STARS

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

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Duration of the Project	1.1.2010 – 31.12.2012

ABSTRACT

During 2013, STARS renewed its collaboration agreements with ENSI on scientific support and research related to multi-physics multi-scale modelling of Light-Water-Reactors (LWR) and with emphasis on safety analyses of the Swiss reactors. On this basis, one scientific support activity was to undertake steps towards providing verifications of new core designs and in that framework, the latest KKL core loading submitted for licensing was evaluated with the STARS independent modelling and analysis capabilities. In the area of thermal-hydraulics, research was oriented towards a consolidated approach for the validation of TRACE models of the Swiss reactors. Also, a transition from FLICA to COBRA-TF for more versatile sub-channel capabilities was launched. For the usage of CFD methods in multi-physics simulations, the validation of STAR-CCM+ for coolant mixing phenomena was strengthened and OpenFOAM was in this context also introduced as complementary solver. And in the perspective of a CFD multi-scale approach to enhance heat transfer models of thermal-hydraulic codes, a first assessment of the STAR-CCM+ capabilities for boiling phenomena was carried out. Finally, a PhD thesis was launched to address uncertainties in physical models of thermal-hydraulic codes. Regarding neutronics and core analysis, the transition to CASMO-5 for the Swiss core models was continued and a first assessment of the new SIMULATE-5 advanced core simulator was conducted in the framework of a newly launched PhD thesis. For reactor dynamics, the BWR stability methodology was further developed, both on the basis of an on-going international benchmark as well as for the analysis and interpretation of complex stability tests carried out at the KKL plant. Also, the development of a new spatial coupling scheme for TRACE/S3K plant system/core simulations was initiated and first studies towards a more reliable modelling of reactivity feedback effects in case of strongly heterogeneous coolant conditions were conducted. The complementary TRACE/PARCS code system was also evaluated, revealing several limitations in many parts and steps of the computational flow. Finally, the methodology for nuclear data uncertainty quantification was strengthened by establishing methods to propagate depletion as well as decay related uncertainties and a first assessment was carried out to estimate decay heat uncertainties in the context of spent fuel pool safety analyses. Concerning fuel modelling, the analyses with FALCON of LOCA experiments at the Halden reactor and involving high burnup fuel samples from the Swiss BWRs, was continued. This included a validation against a recent test with cladding burst and the design of a new test aimed at significant ballooning but with no cladding rupture. And in relation to this, a PhD thesis was also launched to develop enhanced models for fuel fragmentation, relocation and dispersal during LOCAs. For PCI/PCMI related fuel failures, selected BWR and PWR power ramp experiments from the SCIP-II program were studied with FALCON in order to understand

the effects from fuel and ramp characteristics. The performance of the FALCON criteria for PCI failures was in this context also evaluated. Finally, an important step was to start the development of uncertainty quantification methodology for FALCON analyses. At this stage, emphasis was given to estimate the effects from modelling uncertainties on the predicted fuel temperature during base irradiation and to establish global sensitivity analysis methods to determine the main contributors to the uncertainty.

Project goals

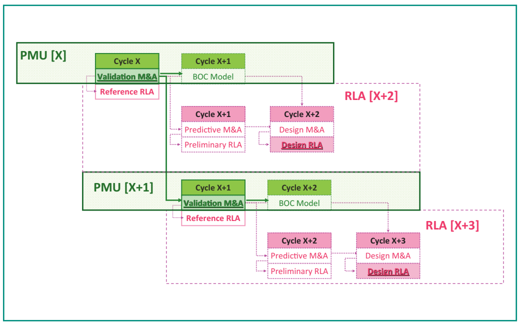
STARS aims at research related to multi-physics multi-scale modelling and simulations of Light-Water-Reactors (LWR) with emphasis on applications to safety analyses of the Swiss reactors. The main components of STARS are the two cooperation agreements with ENSI on scientific support and research respectively. During 2013, these two agreements were renewed for the period 2013–2015 with continued emphasis on: 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. Within this framework, the objectives for 2013 were as shown in Table 1.
This report provides an overview of the status and progress achieved for selected activities conducted in relation to these objectives.

Table 1: *Objectives 2013*

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 Behaviour 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

Scientific support

During 2013, a main scientific support activity was to conduct a pilot study aimed at independent verification analyses of the new KKL core design. For this, the CMSYS platform had first to be updated. So far, this platform was mainly used as framework to conduct for each of the Swiss reactors, periodic model updates (PMU) consisting in the development and validation of reference core models up to the latest completed cycle. To integrate the new types of calculations required for core licensing and referred to as reload licensing analyses (RLA), the platform was updated both in terms of architecture and computational modules such as to accommodate for the modelling and analysis (M&A) flow shown in Fig. 1. Although this update has so far only been made for KKL, it allowed to carry out the RLA calculations for the new KKL Cycle 30 core design. A quantitative evaluation of some main operation and safety relevant parameters was performed including core reactivity, three-dimensional (3-D) power/burnup distributions, core pressure drop, thermal limits, shutdown margins, reactivity coefficients, kinetic parameters, control rod worth for rod drop accidents as well as core characteristics for stability analyses. To qualitatively assess the performance of the new core design, all the parameters were com-



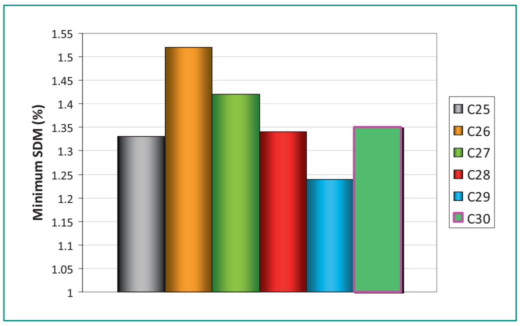


Figure 1 (left):
New CMSYS Modeling and Analysis Flow.

Figure 2 (right):
Comparison of Minimum SDM between Cycle 30 and Previous Cycles.

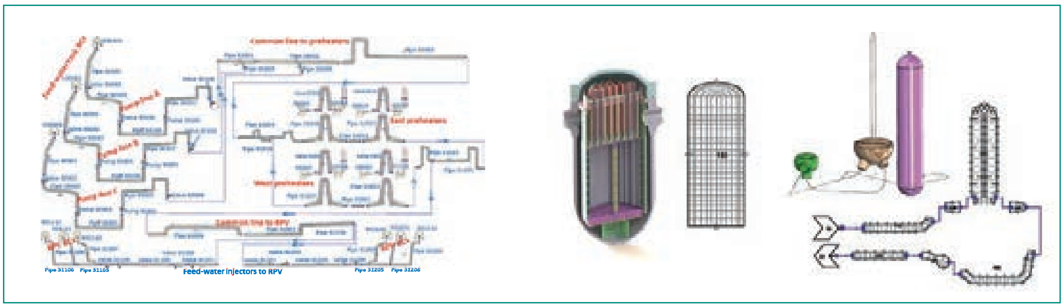


Figure 3:
Left – Revised KKL Feedwater model.
Right – Solid Model of the KKG Reactor used to Derive Updated TRACE Component Models.

pared to those obtained for the five preceding operated cycles. This indicated that the Cycle 30 core design would be well in line with recently operated cycles ([2], [3]) and no particular deviation could be identified. This is illustrated in Fig. 2 where a comparison of the cycle minimum cold shutdown margin, reflecting the margin to cold criticality when the highest worth control rod remains fully withdrawn, is presented.

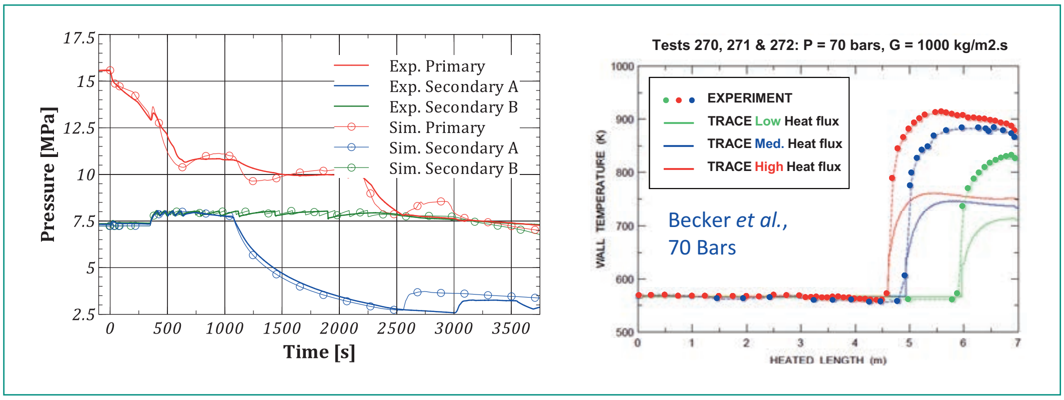
Developement of TRACE plant system models for the Swiss reactors

During 2013, a thorough review towards updates of the TRACE models for the Swiss nuclear power plants (NPPs) was initiated for the KKG, KKL and KKM plants and following the latest TRACE best practice guidelines. On the BWR side, the update of the TRACE KKL model was started with focus on reviewing the steam line and associated control system models. These were identified as the main reason for a rather poor performance of the TRACE plant system response when analysing a turbine trip test conducted at the plant and representing a typical type of transient included in reload licensing submittals. Progress was however limited due to lack of sufficient information and discussions were therefore undertaken with the plant to address this issue. Also, an in-depth review of the feedwater system was carried out, resulting in the updated model illustrated in Fig. 3. As for the TRACE KKM model, a revision of the model for steam line break analysis was completed, ensuring thereby capabilities to analyze such transients. For PWRs, an update of the KKG TRACE model was launched. The new model will use a three-dimensional VESSEL component for the reactor pressure vessel. To do so, a detailed three-dimensional solid model of the KKG reactor core as well as other regions of the primary system has been developed (see Fig. 3). This solid model was already useful for two aspects of the revision work; the exact flow paths of leakage and bypass flows in the reactor core were not well understood and the model helped to clarify a number of issues. The solid model also provides an exact geometric representation of the KKG reactor from which updated TRACE models of the various components can be derived. Some effort has already been made in this direction and preliminary automated approaches for obtaining VESSEL and PIPE components directly from solid model geometry have been tested.

Assessment of TRACE code using ITF and STF experiments

Parallel to the updates of the KKG TRACE model, work towards an On-Call on Steam-Generator-Tube-Rupture (SGTR) accidents was launched. To that aim, an SGTR experiment referred to as Test-4 of the OECD/NEA ROSA-2 project and carried out at the JAEA/LSTF Integral-Test-Facility (ITF) was modeled and analyzed with TRACE. As can be seen in Fig. 4, the results obtained so far show that the primary and secondary pressure of both loops is captured adequately, providing confidence in the TRACE capabilities to simulate such scenarios for the Swiss plants. Sensitivity analyses are now being

Figure 4:
Left- TRACE Analysis of ROSA SGTR
Right - Validation of TRACE against Becker Single Rod CHF Test .

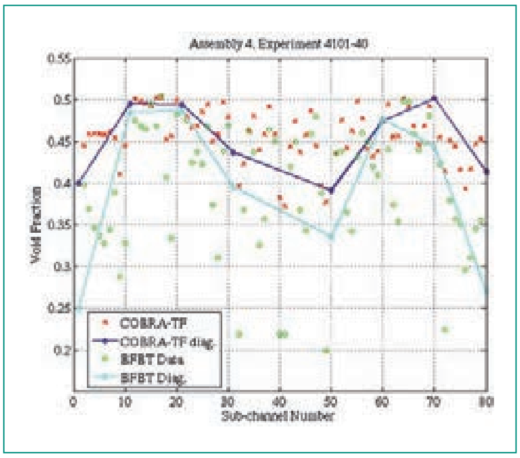


conducted to determine the optimal nodalization to be employed for the KKG simulations. Regarding experiments aimed at separate-effects, main focus in 2013 was on validating the critical-heat-flux (CHF) and post-CHF models of TRACE. A series of CHF tests selected such as to confine the phenomenon to the simplest possible conditions, were analyzed with focus on two aspects: the capability to predict the correct CHF location and the capability to predict the right post-CHF wall axial temperature distribution. The results for one of the Becker test are presented in Fig 4. Although not shown there, for most flow conditions, a CHF multiplier had to be applied in order to capture the correct CHF location. The next difficulty was that despite a correct CHF location, TRACE would tend to underpredict the wall temperature, sometimes by up to a few hundred Kelvin. This gives thus a first quantitative indication of the uncertainty level in TRACE post-CHF predictions.

Sub-channel modelling

After earlier efforts towards using the PWR FLICA4 code for subchannel CHF calculations coupled to 3-D core simulations, a migration to the more versatile code COBRA-TF was for several reasons considered as necessary. In particular, this code should allow for BWR applications since it models the liquid water as two separate fields (a liquid film attached to the wall and an entrained droplet field), something that could theoretically allow for BWR dryout predictions based on mechanistic principles. Therefore, during 2013, a first evaluation of COBRA-TF for BWR simulations was conducted by modelling and analysing the OECD/NEA BWR Full-size Fine-mesh Bundle Test (BFBT) benchmark. For illustration purpose, the results obtained for case 4101-40 of assembly Type 4 (pressure = 7.144 MPa, flow rate = 20.03 t/h, inlet sub-cooling = 52 kJ/kg and exit quality = 0.70) is presented in Fig. 5. There, experimental versus predicted void distributions are shown as a function of the sub-channel number. When considering the sub-channels starting from the corners linked diagonally together, it can be observed that the predicted sub-channel void agrees well with the measurements even in the presence of strong diversion and turbulent mixing. Complementary sensitivity analyses confirmed that COBRA-TF would provide satisfactory local void predictions except for corner and inner subchannels. And a tendency for increased spread between measured and calculations was observed at low flow conditions, pointing to the need for further assessment focused on the code models for turbulent mixing as well as void drift coefficients.

Figure 5:
Modelling and Assessment of COBRA-TF for BFBT Bundle Experiments.



CFD methods for safety applications and multi-scale modelling

To gradually implement CFD models to estimate complex three-dimensional flow structures required in multi-physics simulations, a validation of the STAR-CCM+ code was performed on the basis of mixing experiments from the ROCOM facility and included as part of the OECD/NEA PKL-2 project. Particular focus was given to two modelling approaches for the turbulent heat flux. In the first approach, a standard constant value for the turbulent Prandtl number (CTP) was employed while in the second approach, a variable turbulent Prandtl number (VTP) accounting for turbulent local flow conditions was implemented. The results (see Fig. 6) showed that both models would be capable to describe the time-evolution of the temperature field at the core inlet but the VTP approach would be clearly superior in terms of capturing the separation line between two temperature zones that build up in the downcomer due to thermal stratification and mixing. Regarding usage of CFD for multi-scale modelling, an assessment of STAR-CCM+ for pool boiling was carried out in the perspective of enhancing the TRACE heat transfer models for simulations at very low pressure stagnant flow conditions. This research was conducted with support from a guest scientist invited to the project through an IAEA fellowship program. The study consisted in assessing the latest STAR-CCM+ boiling models for a) full scale representations of passive-containment-cooling-system (PCCS) experiments carried out at the PSI PANDA facility and b) small scale experiments from a simplified heating up device. For the former, no conclusive results could be obtained due to several convergence issues and to the complexity of the PCCS modelling requirements. On the other hand, for the simpler experiment, the obtained results showed a satisfactory performance when comparing against measured temperatures (exp) and previous CFX solutions. As well, qualitatively correct velocity distributions and void fractions, including void fraction rising trends and void appearance time, were obtained.

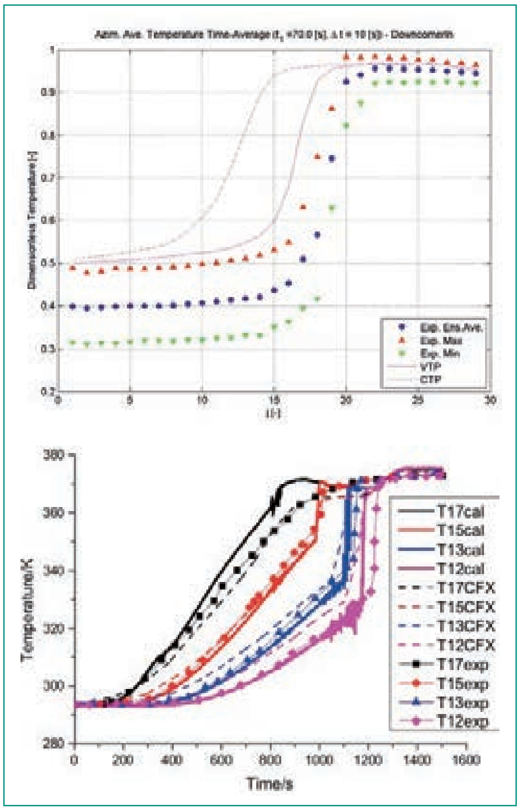


Figure 6:
Top – STAR-CCM+ Assessment against PKL2 ROCOM Test – Bottom – STAR-CCM+ Modelling of Pool Boiling for Simplified Heating Device Experiment.

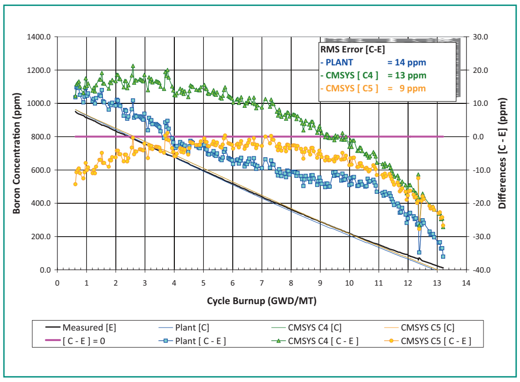


Figure 7:
Assessment of CASMO-5 for CMSYS Predictions of Critical Boron concentration of KKG Cycle 30.

Reactor physics and core analysis methods

During 2013, the transition from the 2-D lattice code CASMO-4 (C4) to its successor CASMO-5 (C5) for the Swiss core models was continued. As part of this, specific emphasis was given to develop a new burnup condensation scheme for the KKG reactor and to refine the modelling of the aeroball detector system. To illustrate the results, the calculated boron concentration for the latest KKG cycle modelled in CMSYS is shown in Fig. 7. Compared to both C4 as well as to the plant reference calculations, the C5 based results provide two significant enhancements. First, the within cycle variation of the reactivity bias does not show as distinct burnup-dependant trends. Secondly, the cycle average bias (RMS) is now below 10 ppm which can be considered as remarquable. Also on the PWR side, an assessment of the advanced SIMULATE-5 core simulator was started with focus on evaluating the effects of the new radial sub-meshing method on pin power calculations. This showed among other things, a non-negligible impact especially for fresh Gd rods as well as MOX assemblies. For BWRs, significant validation activities were carried out to improve the KKL models and the KKM models were also brought in-line with the latest operated cycle. In addition, a study of the void reactivity coefficient (VRC) was carried out for an ATRIUM assembly model. Special emphasis was given to the evolution of an unphysical behaviour seen in the C4 VRC around 70-90% void when using the ENDF/B-IV based L-Library. When quantifying the VRC evolution as function of gradual changes in CASMO code and associated neutron data libraries, it was confirmed that only the older L-library presents this unphysical behaviour but mainly at beginning-of-life (BOL). Reactivity decompositions confirmed that this behaviour is guided by ^{238}U capture and point to library adjustments made for this reaction in order to render the VRC less negative, something that could have been aimed at e.g. improving the cycle length predictions in relation to the BWR stretch-out spectral shift operation phase.

Reactor dynamics and stability

During 2013, the development of the SIMULATE-3K (S3K) methodology for BWR stability analysis was continued in several areas. First, significant progress was achieved for the OECD/NEA benchmark on the Oskarshamn 2 instability event with a very close agreement against plant data of the preliminary S3K solution. Regarding the Swiss BWRs, investigations were continued on the out-of-phase regional instability mode that occurred during the KKL Cycle 7 test. An in-depth analysis of the predicted LPRM responses was performed and allowed to better relate the oscillating versus rotational behaviour of the core symmetry line to the excitation modes of the neutron fluxes. Bifurcation studies were also carried out, indicating that the observed Cycle 7 regional mode corresponds to a super-critical Hopf bifurcation. Steps have now been undertaken to couple S3K with modal analysis methods in order to understand more precisely the excitations of the various neutronic modes and the conditions under which these are triggered. Concerning the global stability mode, in-depth sensitivity analyses were conducted in relation to the S3K validation for KKL Cycle 10 and special focus was given to Record 10 during which the core was practically unstable. This has not been captured by the PSI validation so far and available solutions from other organisations have indicated the same systematic underprediction of the decay ratio (DR). Several hypotheses have been investigated but none could lead to conclusive statements regarding this behaviour. Therefore, attention was given to one aspect not investigated in details before, namely the reactivity feedback under conditions characterized by heterogeneous active versus bypass flow conditions. To that aim, the test was re-analysed by applying the standard code feedback model for such conditions. While this allowed to better capture the resonance frequency, it severely aggravated the DR underprediction (see Fig. 8). Therefore, a new approach was instead developed and tested based on the concept to ensure at the initial steady-state conditions, the same reactivity feedback as the static core simulator SIMULATE-3. Without any intervention on any other models or assumptions, this new approach, referred to the static heterogeneous feedback reactivity model (SHFRM), allowed to capture the core dynamics with remarquable precision. Although this approach needs further research, it is believed to have revealed one eventual reason for the long-standing unresolved issue of poor code performance for this specific test.

Modelling and analysis of fuel rod behaviour during LOCA

A post-test analysis of the Halden LOCA Test IFA-650.13 using a High-Burnup (HBU) fuel sample from the KKL plant was carried out as a part of the project's continuous modeling support to the OECD Halden Reactor Project (HRP). These new experiments were adopted in order to study new phenomena taking place at high burn-up, particularly Fission Gas Release (FGR). Similarly as before, the

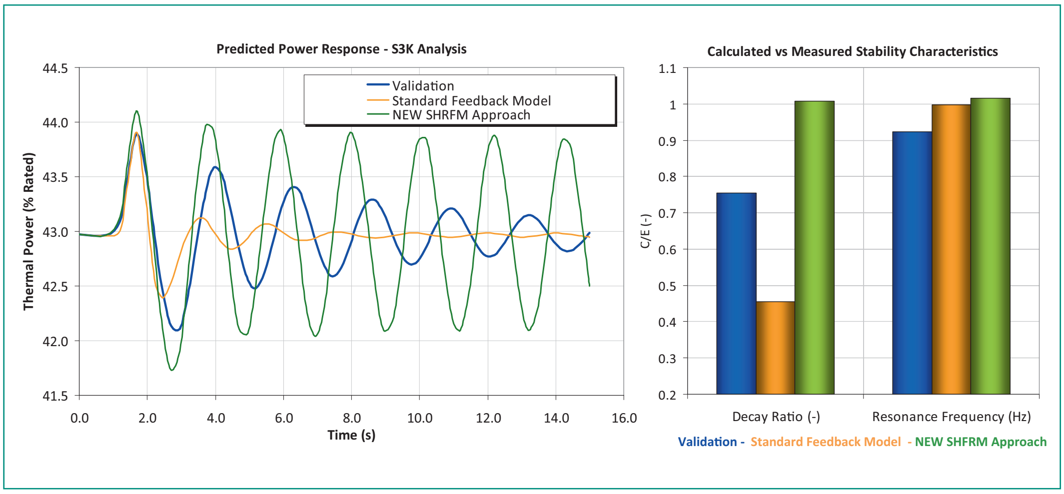


Figure 8: *Modelling and Analysis with S3K Analysis of KKL Cycle 10 Record 10 Test.*

above mentioned test IFA-650.13 was designed by PSI using the FALCON code coupled with the GRSW-A model and with the objective to produce a significant ballooning as well as a burst of the cladding. This was effectively achieved when the test was conducted towards end of 2012. Experimental data obtained after the test were used during 2013 for a post-analysis aimed at general code validation but also aimed at verifying findings obtained from the preceding test (IFA-650.12). This shed light on: (1) Axial distribution of LHGR in the fuel rod, in consideration of its reduced length; (2) the fuel-sample specific cladding surface emissivity to precisely calculate the cladding temperature during the experiment; (3) the narrowed down range of calculated damage index for predicting cladding failure during the LOCA; (4) the method for estimating the quantity of fission gases to be released by the HBU fuel during the early phase of the LOCA transient. Regarding Point 3, the mutual evolution of the calculated Cumulative-Damage-Index (CDI) of the cladding and the rod pressure during the pressure reduction phase caused by the ballooning, is shown in Fig. 9. It can be seen that the measured pressure reduction falls precisely in the range corresponding to CDI values between 0.5 and 0.7. This is well in line with the PSI design recommendations. Moreover, a comparison of the measured and calculated pressure reduction confirms that the cladding failure took place rather at a lower limit of critical CDI range, ca. of 0.5. This is also consistent with a value of 0.56 estimated as the CDI at burst for IFA-650.12. An important outcome of the above post-test validation of FALCON/GRSW-A is that it has underlined the key role of an adequate prediction of the rod pressure reduction – relating to the reached maximum – due to the ballooning until cladding burst.

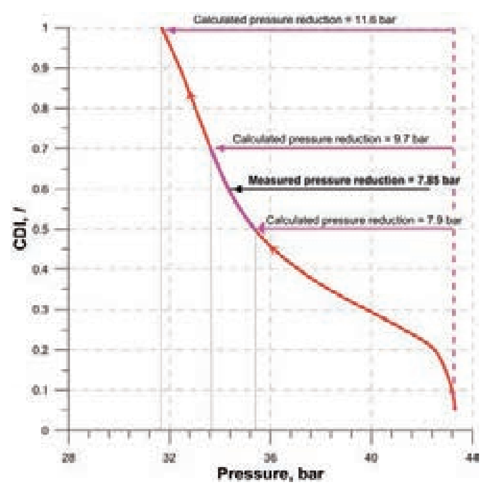


Figure 9:
FALCON/GRSW-A Analysis of HALDEN LOCA IFA-650.13 Burst Test.

Assessment of fuel performance codes for PCI/PCMI failures

During 2013, STARS continued its participation to the OECD/NEA SCIP-II program to assess the modeling capabilities of FALCON coupled to GRSW-A for predictions of PCI/PCMI fuel failures. In this context, analyses of selected ramp tests involving both PWR and BWR samples were carried out. One objective was to investigate and discriminate effects from ramp and fuel characteristics on occurrence of fuel failure. The main results are summarized on the left hand side of Fig. 10. First, it is observed that a stepwise power ramp mode appears to be more crucial in terms of avoiding fuel failure than a large peak LHGR because such ramp mode allows for significant stress and strain relaxations during each ramp hold time. Secondly, it appears that the ramp rate has a stronger role on fuel failure compared to burnup. Third, the ramp mode and rate affect the strain rate which is found to be determinant regarding the time to failure: a high strain rate leads to short time to failure and vice-versa. A second objective was to evaluate the performance of the two fuel failure criteria used by

	Segments			
	PR-1	PR-2	PR-3	BR
Test Outcome	**Failure**	**Failure**	**NO Failure**	**Failure**
Characteristics				
Burnup	**High**	**Low**	**Low**	**High**
Ramp Mode	**Single Step**	**Single Step**	**Stepwise**	**Stepwise**
Ramp Rate	**High**	**High**	**Low**	**High**
Peak LHGR	**Low**	**Low**	**High**	**Low**
Strain Rate	**High**	**High**	**Low**	**Low**
Time-to-Failure	**Short**	**Short**	**-**	**Long**
Lasted time with high Stress	**Short**	**Short**	**Very Short**	**Very Long**
Failure Criteria				
Cumulative Damage Index (-)	1.49	1.02	---	---
Peak Hoop Stress (Mpa)	535.5	522.7	506.9	629.7

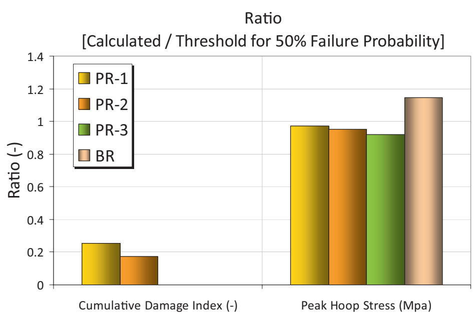
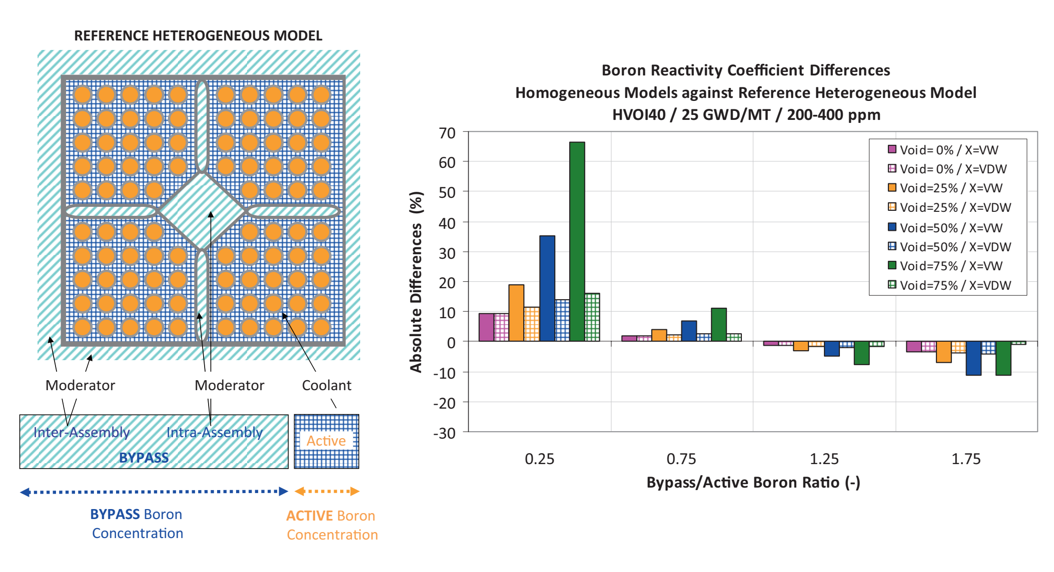


Figure 10:
FALCON/GRSW-A Modelling and Analysis of SCIP-II PWR (PR) and BWR (BR) Ramp Tests.

Figure 11:
CASMO Studies of Heterogeneous Boron Distributions across Active and Bypass Zone of BWR Assemblies.



FALCON in relation to PCI failures: the Cumulative Damage Index (CDI) method and the Peak Hoop Stress (PHS) approach.

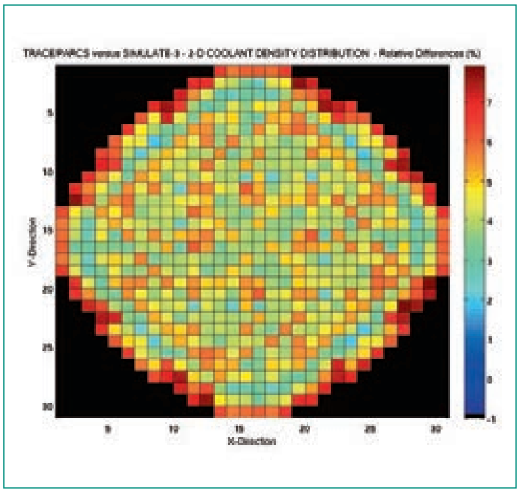
Although the CDI was found to predict a non-negligible failure probability, it remained significantly lower than the currently applied threshold for 50% probability failure (see Fig. 10, right). One reason might be that it does not account for the strain level and relies only on hoop stresses. A second difficulty is that it could not be applied to step-wise ramping modes. The PHS approach was on the other hand found to predict more consistent trends with experimental results: lowest failure probability for the rodlet that did effectively not fail (PR-3) and around or above 50% probability for the ones that failed. However, the studies have indicated that not only a low peak hoop stress but also a short time duration over a certain threshold is critical for survival. Conversely, a high peak stress combined with a long duration above a certain threshold will increase the probability for failure. As the time duration is not accounted for in the PHS approach, this could partially explain that more convincing failure probabilities were not obtained also with this approach.

Multi-physics and coupling methodologies

During 2013, an activity was launched to enhance coupled neutronic/thermal-hydraulic code systems for situations where the thermal-hydraulic (T-H) feedback is characterised by very heterogeneous distributions between the active and bypass zones of fuel assemblies. For BWRs, studies with the CASMO lattice transport solver were first initiated to evaluate the feedback effects from heterogeneous ^{10}B concentrations. Several methods to capture this feedback in downstream coupled codes confirmed that a volumetric-density weighting (VDW) rather than only volume weighting (VW) would be adequate to reduce the over or under prediction of the (negative) boron reactivity coefficient (BCOEFF). However, for transient time periods during which more boron would accumulate in the active zone, the BCOEFF magnitude will be overpredicted specially for highly voided core regions. On the contrary, if the boron is transported to the bypass before entering the active fuel, the BCOEFF will tend to be less negative.

Another main activity was to launch the development of an external coupling scheme for the TRACE/S3K code system. Compared to the existing internal coupling mode, the external mode intends to use S3K for all core physics, i.e. both neutronics and T-H, while TRACE will now handle the T-H only for the system and out-of-core com-

Figure 12:
Comparison of 2-D assembly Average Coolant Density Distributions between TRACE/PARCS and SIMULATE-3.



ponents. Preliminary testing of this new coupling mode has been launched for open public OECD/NEA benchmarks and the work remains under progress. Concerning TRACE/PARCS, a master thesis (MSc) project followed by a 3 month internship was conducted. During the MSc project, focus was given to establish and verify on the basis of the KKL plant, methodologies for 1) PARCS core model initialization; 2) TRACE standalone plant system model initialisation; 3) coupled TRACE/PARCS model initialization. The follow-up practicum had as objective to resolve or mitigate many of the TRACE/PARCS limitations encountered during the MSc project. First, a simplification of the CASMO cross-section structure was found necessary for usage by the PARCS XS model. Secondly, a plenum-to-plenum 648 full core TRACE model was developed in order to reduce the large errors in core void fractions caused by lumped channel models. This allowed to enhance the steady-state TRACE/PARCS solution when compared to SIMULATE-3 and TIP measurements. However, the unclear treatment by PARCS of reflector T/H variables and the limitations in handling coupled reactivity effects, did not allow to fully resolve the k-eff discrepancies and the power distribution errors at the core radial/axial peripheries. On the TRACE side, the lack of water rod models in the current KKL model was found to produce a too low bypass flow fraction, producing thereby a major part of the coolant density differences against SIMULATE-3 (see Fig. 12) and leading through this to relatively large power distribution differences.

Uncertainly analysis

During 2013, the STARS activities on methodologies for uncertainty quantification (UQ) and sensitivity analysis (S/A) were strengthened and spread across all three main technical areas of the project. For thermal-hydraulics, a PhD thesis was started with the aim to derive uncertainties in physical models (e.g. correlations, empirical models) employed by thermal-hydraulic codes. On the neutronics side, the development of methods to propagate half-lives (HL), energy-per-decay (EPD), and fission yield (FY) uncertainties through the CASMO-5 code was launched. Although several simplifications and assumptions remain and will need to be gradually tackled, a first application was to evaluate decay heat uncertainties due to nuclear data in the context of spent fuel pool safety analyses. The relative contributions from the various classes of nuclear data to the decay heat uncertainty were in this context also estimated indicating, as shown in Fig. 13, that the largest uncertainty contributions come from FYs, independently of burnup or cooling time. Decay data (HLs, EPDs) uncertainties play a non-negligible role during the cooling phase but their contributions decrease as function of cooling time. And cross-section (XS) uncertainties start to play a role only for high assembly discharge burnups.

Another important milestone is that UQ methodology development for fuel analyses with FALCON was started through participation to the OECD LWR UAM Phase-2 benchmark (UAM-II). At this stage, focus has been given to the steady-state benchmark cases aimed at assessing the impact from model uncertainties (e.g. design, geometry, materials) on fuel temperature predictions. To that aim, an interface was developed between FALCON and the URANIE platform selected and installed in STARS during the year to serve as general UQ and S/A platform. On that basis, sampling of uncertain model parameters was performed using a Latin

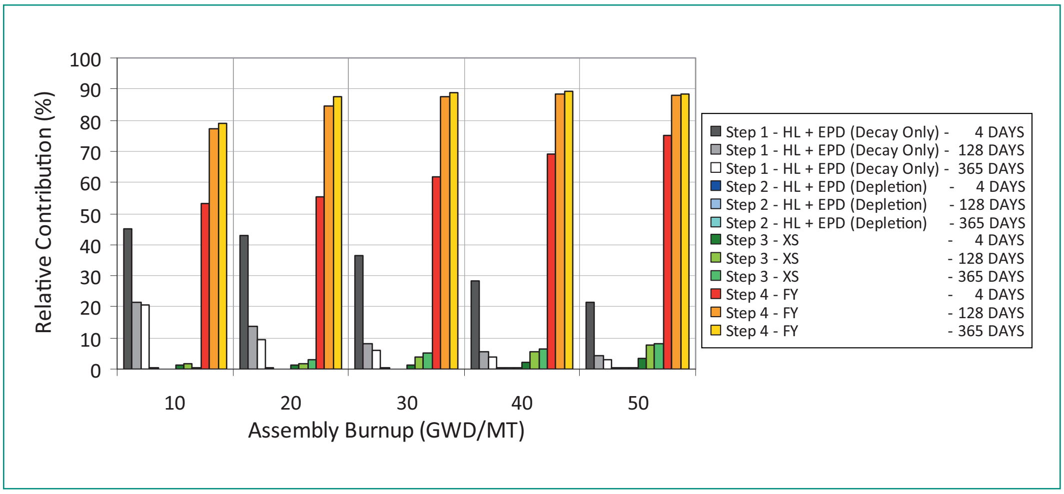


Figure 13:
Nuclear Data Uncertainty Contributions to CASMO-5 Decay Heat Uncertainty.

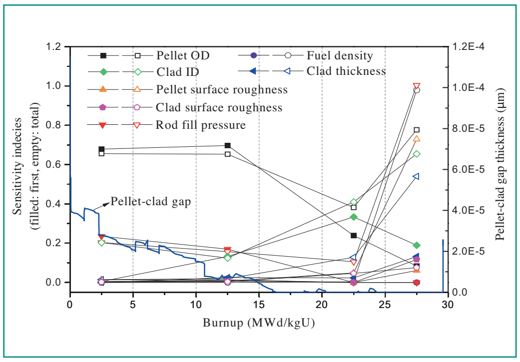


Figure 14:
FALCON/URANIE Estimations of First and Total Sensitivity Indices of Fuel Temperature upon Modelling Uncertainties.

Hypercube Sampling (LHS) method and the uncertainty in predicted parameters was quantified, showing for instance, an increased fuel temperature uncertainty as function of burnup. To assess the contributors to the output uncertainties, a global sensitivity analysis approach based on the Sobol method was also tested. Preliminary results, illustrated in Fig. 14, indicate that at low burnups, the gap thickness plays a key role on the fuel temperature uncertainty while after gap closure, interactions between parameters start to jointly contribute to the uncertainty, noting that rod fill pressure and fuel density are indicated as being mostly involved in the interactions. Further research is now on-going to verify these results, in particular regarding differences between first order and total sensitivity indices as some of the differences seen in Fig. 13 at e.g. low burnup could be due to numerics and/or indicate the need for a larger number of samples.

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 mainly, «Special Topics in Reactor Physics» and 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. Also, the project collaborates with the IAEA for the training of scientists from emerging nuclear countries in the area of LWR safety analyses. The project also collaborates with other research organisations, on the one hand through e.g. EU 7th FP NURESAFE project and on the other hand, through bilateral cooperation e.g. GRS, CEA, IRSN, Urbana-Champaign, Chalmers university. An active cooperation with the Finnish regulatory body STUK for safety evaluations related to the GIII/GIII+ EPR is also being continued and with AREVA, discussions towards research cooperation on advanced core simulation methods has been started. 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 2013 and Perspectives for 2014

During 2013, progress was achieved with regards to most of the goals. In particular, a pilot study on core licensing was completed in order to provide independant verifications of the new KKL core loading. Concerning research, a significant milestone was reached by enlarging the development of uncertainty analysis methodologies to all technical areas of the project, including fuel modelling. On the education side, three new PhD projects were launched including uncertainty analysis related to physical models of thermal-hydraulic codes, development and assessment of advanced methodologies for full core 3-D pin-by-pin transport methods and modelling of fuel fragmentation, relocation and dispersal during LOCA. On the other hand, some few but important goals could not be achieved. Due to the departure of the responsible scientists for KKL plant system analysis and sub-channel modelling, the YUMOD On-call could not be completed as desired and the establishment of a capability for DNB predictions in core simulations was not launched. On the fuel modelling side, the high priority given to LOCA fuel safety work as well as PCI/PCMI and uncertainty analysis, did not allow to complete the studies on clad-lift-off. Also, the development of a fuel management system (FMSYS) for reference models of Swiss fuel rod designs was not started. For 2014, the main objectives are as follows.

Plant System and Thermal-Hydraulics	Completion of On-Calls on KKL Fast Pump Run-up Transients and KKG SGTR Transient
	Consolidate Validation of System Code Thermal Hydraulic Models for CHF Predictions
	Assessment of Sub-Channel Models for Single and Multi-Assembly Analyses
	Development of CFD Vessel Model with Coupling Approach to TRACE for a Swiss PWR
	Validation and Application of CFD Code for Boron Dilution and Coolant Mixing Transients
Core Behaviour and Reactor Physics	Core Analyses for Support to Reload Licensing of the Swiss Reactors
	Start Transition to SIMULATE-5 for a Swiss Reactor
	Establishment of nTRACER for Full Core Transport Analyses
	Modelling and Assessment of 3-D Kinetics Solver for RIA Experiments
	Development of Methodology for Nuclear Data Uncertainty Propagation to Few-Group Cross-Sections
Fuel Modelling and Thermo-Mechanics	Review and Verification of KKM Fuel Performance Code
	Modelling Support to Fuel Safety Experimental programs and Assessment of DIFFOX for Clad Oxygen Diffusion
	Establishment and Assessment of Falcon V1 Reference Steady-State Methodology for Swiss Fuel Rod Designs
	Participation to RIA Code Benchmark with Uncertainty Analysis
Multi-Physics	Development of COBALT-X towards PWR Full Core LOCA Analyses
	Development and Assessment of External Coupling Scheme for TRACE/S3K Analyses
	Continue and Enlarge Participation to OECD/NEA UAM Phase 2 for Bundle Thermal-Hydraulics

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