# STARS

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


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ABSTRACT

During 2009, the STARS project has continued its consolidation around the TRACE code for plant system analysis with on the one hand, base models of all the Swiss plants established and on the other hand, modelling and assessment activities for both integral test experiments and for the EPR reactor. An increased usage of CFD methods along with the establishment of advanced modelling methodologies for system thermal-hydraulic analysis constitutes another significant achievement. As a major step towards advanced multiphysics capabilities, a coupling scheme between TRACE and SIMULATE-3K for 3-D core/plant system transient analyses was developed and its assessment has been started for a BWR turbine trip transient. And related to core physics, enhanced core modelling methodologies, required for the periodic modelling of the Swiss cores, have been established for KKB/KKM and are currently being developed for KKL/KKG. Also, the assessment of cross-section modelling effects on 3-D core dynamics has been continued, confirming among others the need for developing methods for the propagation of neutronic uncertainties. In that context, a forward sensitivity method based on direct perturbations has been implemented for lattice transport and is being assessed the context of an on-going international benchmark. For fuel behaviour, recent JAEA Reactivity-Initiated-Accidents (RIA) tests performed at both room- and high temperature conditions have been analysed as part of the validation of the PSI model for gas swelling and fission gas release coupled to FALCON. These studies have, among others, put forward the effect from the initial fill gas pressure on cladding strain, high-temperature creep and ballooning during RIAs. For fuel behaviour of High-Burnup fuel during LOCAs, an enhanced model for axial gas flow has been developed and shown to yield improved accuracy on the basis of HALDEN experimental tests.

Along the above, several On-Calls requested by ENSI during the year were also completed including: a validation of TRACE for a recent transient in KKL, a study on BWR thermal-limit violations in core analysis calculations for KKM, a verification of fluence licensing calculations for KKB reactor life-time extensions as well as a review, in the same context, of the vendor methodology for analysis of Pressurized Thermal Shocks and a fuel behaviour parametric study to assess causes and eventually outline necessary measures related to recurrent fuel rod failures observed in recent KKG cycles.

Finally, the project carried out a substantial revision and update of its quality management system PMS which was successfully certified according to ISO 9001:2008 in July 2009, fulfilling thereby an important milestone for the year.

## Project goals

The mission of STARS is to maintain and further develop a comprehensive state-of-the-art best-estimate analysis methodology of Light-Water-Reactors (LWRs) for conditions ranging from normal operation to beyond design basis accidents (excluding core melt) and including criticality safety. Through the acquired knowledge and capabilities, the primary objective is to act as a leading national competence centre for scientific support to the Swiss nuclear safety inspectorate (ENSI) for plant/core/fuel related issues relevant for the current as well as for the future fleet of LWRs. Over time, the project has evolved towards a multi-partner program but with ENSI remaining as main external partner. In that context, the present document therefore only includes the status and progress related to the activities carried out with/for ENSI whereas those carried out with other partners are not reported here. On that basis, the specific objectives of the STARS project collaboration with ENSI for 2009 were as follows.

### Plant Behaviour

- Complete the migration of existing plant models for legacy codes (RELAP5/TRAC-BF1)
- Define scenarios for SBLOCAs, LBLOCAs and MSLBs and perform LOCA analyses for the EPR and CFD simulations for mixing in the RPV during boron dilution and MBLOCA scenarios
- In the framework of the EU NURISP project, develop advance coupling methodologies for the sub-channel code FLICA and for CFD/system-code interface within the SALOME platform
- Validation of TRACE for BWR applications
- Validation of the coupled CFD/TRACE tool against experimental results
- Implement scheduler for dynamic event tree simulations and run test cases

### Core Physics

- Update Swiss core models (CMSYS) and assessment of CASMO-5
- Development of SIMULATE-3K/TRACE coupling scheme
- Continue with APOLLO-2 for applications towards higher-order 3-D core analysis within NUR-SIP
- Establishment of a method for statistical sampling of cross-section data in 2-D lattice transport calculations

### Fuel Behaviour

- LS-2 NSRR RIA-test with KKL high-burnup fuel recently performed in the HT-capsule (ALPS project)
- Investigate origin of the high FGR, as observed in a number of ALPS tests, emphasising the relative contribution of the HBS in the pellet rim (subject to availability of suitable data)
- Further code improvement and verification should address the issue of apparent anisotropy of macroscopic fuel stack growth caused by microscopic fuel swelling
- Continue validation of GRSW-A model and extend validation matrix of FALCON-PSI, using available data from international projects, e.g. HRP, ALPS and SCIP or from Swiss industry

This short report only presents an overview of the progress achieved for some of the above activities.

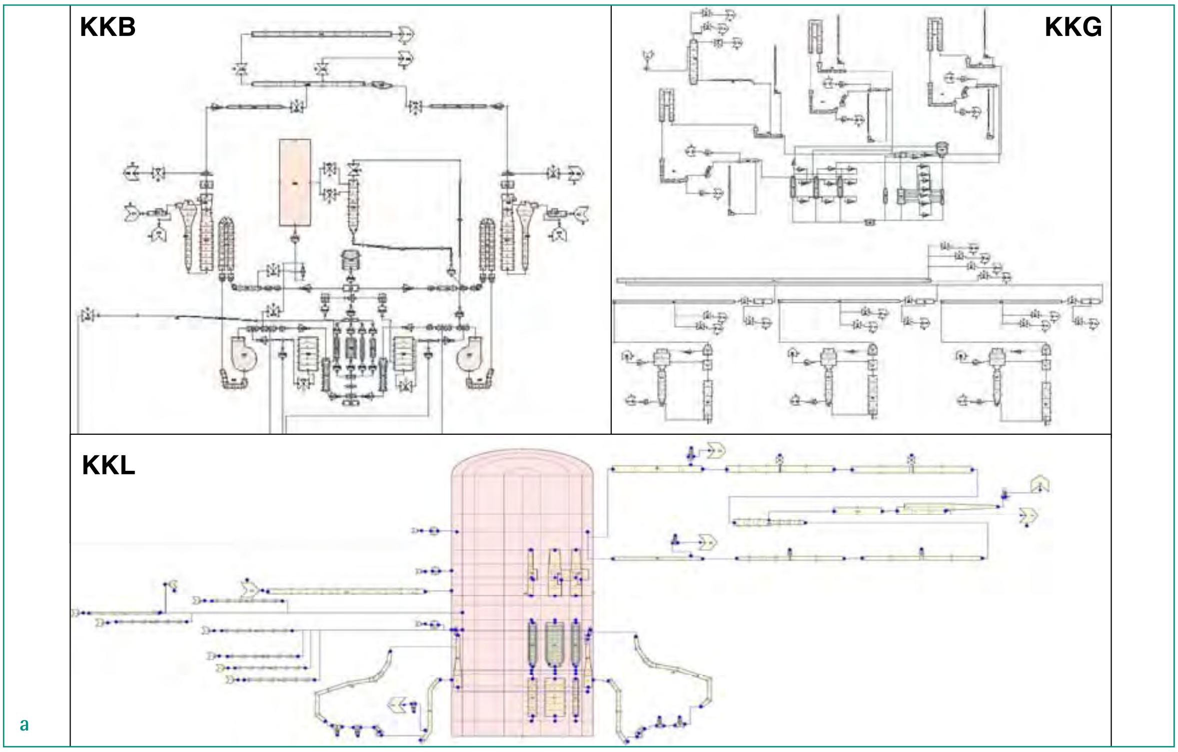
### Plant Modeling with trace for the Swiss Nuclear Power Plants

Considerable efforts have been dedicated to the migration of all models for the Swiss nuclear power plants to the TRACE code, the latest best-estimate thermal-hydraulic code issued by the U.S. NRC for the simulations of LWRs (Light Water Reactors) transients. The models have been validated against available plant data (ADS event for KKL, pump trip for KKG, loss-of-feed-water for KKM) or alternative code simulations (LBLOCA with RELAP5 for KKB). Schemes of the nodalizations are presented in Fig. 1 (a). Below in Fig. 1 (b), results of the validation cases for the KKM and for the KKB model are reported. The KKM simulation corresponds to a total loss of feed-water test performed at KKM in 1993 to demonstrate that after tripping both feedwater pumps, the reactor water level can be maintained above the -227 cm level, with the use of only one, of the two normally available, reactor core-isolation cooling (RCICs) systems. Following the total loss of feed-water flow, the water level decreases gradually due to the negative mass balance. Upon the injection of the RCIC flow, this trend is reversed and the level starts increasing slowly. The level difference between calculated and measured data during the recovery (i.e., after ~150 sec) remained at about 20 cm, which represents a reasonable result (the error in the measured downcomer water level being about +/- 15 cm). The KKB simulation corresponds to a hypothetical large-break loss-of-coolant accident (LBOCA). The maximum cladding temperature and the core quenching time are in good agreement with previous RELAP5 simulations.

Concerning KKL, the development of a base model was completed [1]. This model was used as basis for an On-Call requested by ENSI [2] and aimed at comparing TRACE simulations of the KKL ADS event of March 2007 against both plant data and calculations carried out with TRAC-BF1 and TRACG (the latter performed by General Electric). It was found that the main plant parameters, namely RPV pressure and Wide Range vessel level, are better predicted by TRACE 5.0 code than by TRAC-BF1 and TRACG.

Following an urgent request by ENSI, an intensive 5-days training course on TRACE was also organized and carried out by the project. The course included an introduction on thermal-hydraulic systems codes with special focus on the TRACE code capabilities, models, special components and control system features. The lectures were accompanied by practical cases.

Finally, based on the expertise developed in this area, another On-Call was requested by ENSI [3], this time to review the vendor methodology applied for licensing calculations related to the risks of the KKB1 reactor pressure vessel (RPV) to undergo Pressurize-Thermal-Shocks (PTS). The review was focused on the selection of transients scenarios analyzed by AREVA, and on the computational methods and models employed for the estimation of the time-dependent temperature distribution in the RPV.



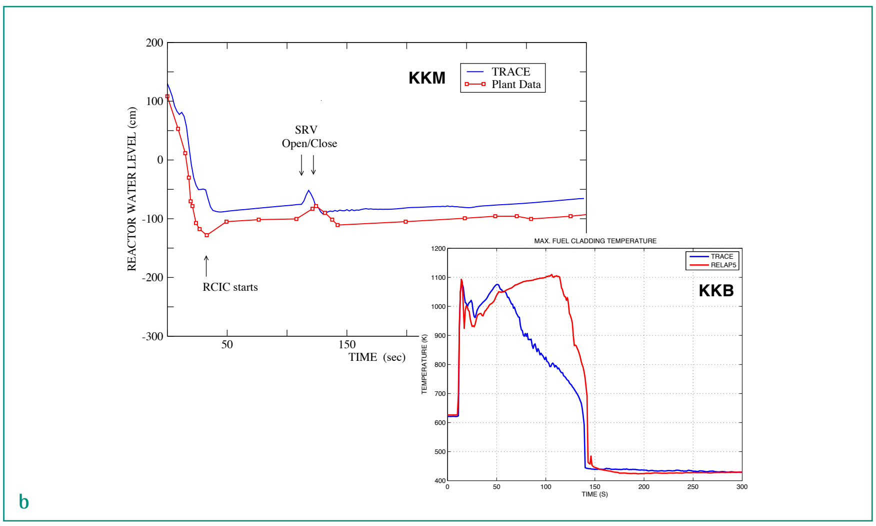




**Figure 1:**
*Scheme of TRACE nodalization for the Swiss plants (a) and results of validation cases (b).*

## Assessment of trace and coupling to 3-D core kinetics

The work on the assessment of the thermal-hydraulic (TH) code TRACE continues to be pursued, based on plant data as well as experimental results gathered with large-scale integral tests and with separate effect tests facility (e.g. [4], [5]). For illustration, the results obtained with TRACE for the large-scale experimental facility PKL (tests F1.1 and F1.2) both related to boron dilution scenarios, have been compared with experimental data and with RELAP5 simulations. TRACE provided results in very good agreement with the experiments (Fig. 2) laying out a high maturity of the code with regards to PWR transients. Similarly, an assessment of TRACE to adequately predict the impact of LOCA conditions on the fuel rod cladding temperature was made on the basis of the HALDEN IFA-650 series of integral experiments. Cladding temperatures predictions (Fig. 3) for the IFA-650.6 test were found to reproduce well, both qualitatively and quantitatively, the measurements providing thereby further confidence in the modelling capabilities developed within the project to simulate LOCA transients using the TRACE code.

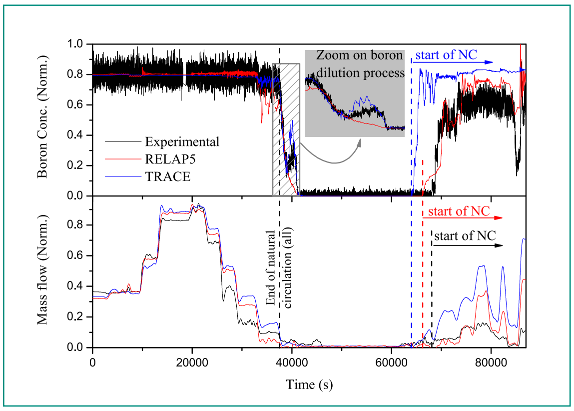



**Figure 2:** *Boron concentration in the loop seal mass flow for PKL test F1.2.*

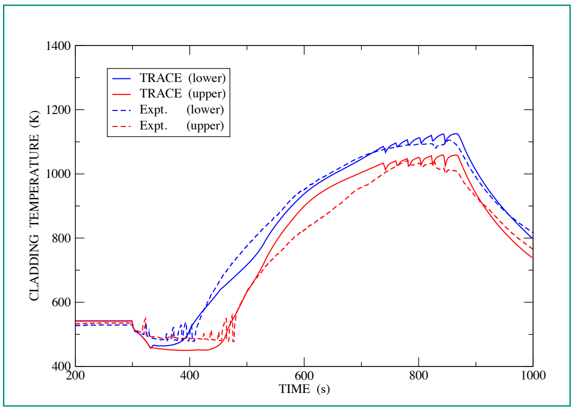



**Figure 3:** *Peak cladding temperature and loop for HALDEN F1.2 IFA-650.6 TEST.*

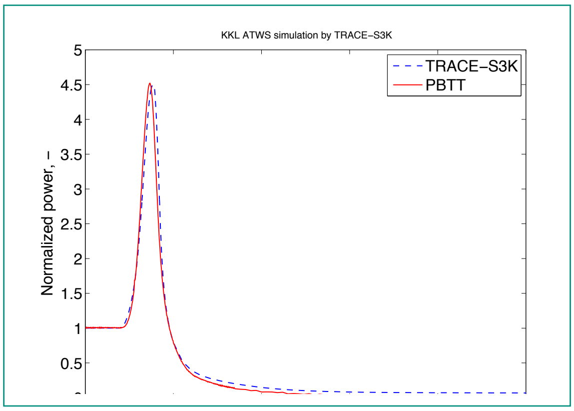



**Figure 4:** *PB2 TTRIP Benchmark Relative Core Power.*

Parallel to this, efforts were undertaken to develop with the core physics sub-project, a coupling scheme between TRACE and the SIMULATE-3K (S3K) 3-D kinetic solver in view of consolidating around these two codes, the project capabilities for advanced integrated multi-physics 3-D core/plant system analysis of the Swiss Nuclear Power Plants (NPPs). A base version of the coupling scheme was developed in collaboration with the S3K code developers and a first validation of the coupled TRACE/S3K code was carried out for the OECD/NEA Peach Bottom 2 Turbine Trip Test 2 (PB2 TT2) benchmark. The comparison with experimental data shows that the coupled code reproduces well the main transient phenomena as shown in Fig. 4. Further code verifications will include the simulation of a PWR MSLB international benchmark and comparison with available Swiss reactors plant data.

## Advanced methodologies for plant system modeling and analyses

As mentioned previously, one of the long-term goals of the project is to establish advanced integrated (multi-physics) analysis methods for the steady-state and transient simulation of the existing and future Swiss nuclear power plants. Physical and numerical consistency of the coupling interfaces between the analysis codes at different scales and for different physics has to be ensured. With regards to the geometrical aspect of the consistency problem, a considered solution is to derive the different codes input information from a single data base (CAD model) describing the simulated system. Based on such data base, the concept would include series of code specific input data pre-processing interfaces that could generate readily and consistently the different input decks required by the simulation codes. As a path-

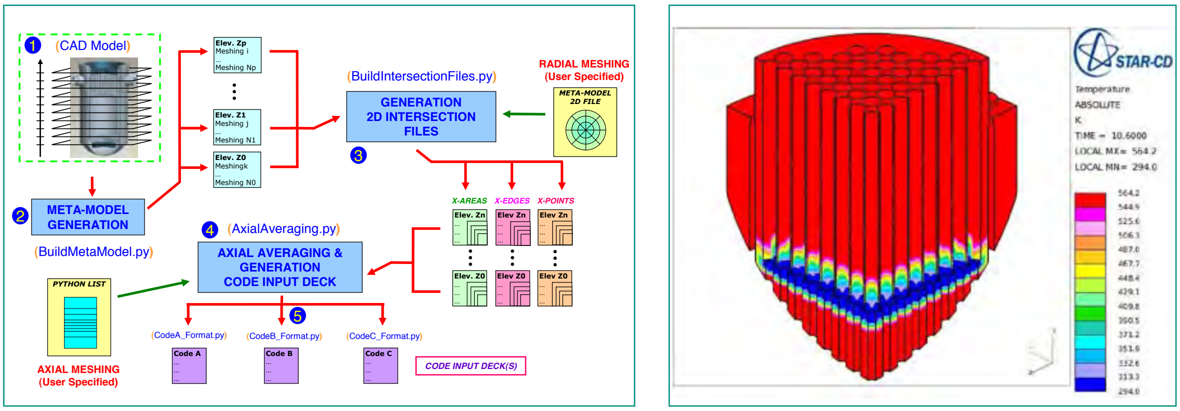



Figure 5: *Architecture of a 3-D coarse mesh pre-processing tool for plant system codes (Left) and temperature distribution in KKL lower plenum computed by means of CFD (Right).*

finder exercise of this envisioned approach, a prototype procedure was developed [6] to compute and generate the input deck for the TRACE thermal-hydraulic system code of a reactor pressure vessel component. The procedure starts from a set of 2-D CAD-based cross-sections describing the system geometry at different elevations and a user-specified 3-D Cylindrical meshing scheme. As a result, input data required by the code for the specified meshing are derived. A first version of this prototype pre-processor is illustrated on the left-hand side of Fig. 5 and was successfully used to generate a 3-D vessel model as part of the project's TRACE model for the EPR Reactor Pressure Vessel [7]. Related to advanced safety analyses, efforts towards probabilistic dynamics, i.e. coupled deterministic and probabilistic safety analyses, continued during the year. Among others, TRACE analyses of a MBLOCA assuming a dominant event-tree sequence and minimum specifications for success were performed in an attempt to quantify the change due to a plant modification, e.g. power uprate, on the exceedance frequency for the given event [8].

Beside best-estimate TH codes, Computational Fluid Dynamics (CFD) is finding more and more application in the nuclear field, as a satisfactory level of maturity has been achieved, at least for single-phase problems. The use of CFD can give particularly insights, for processes where 3D effects are important and for which TH-codes cannot provide correct information, with the sufficient level of detail. A CFD expertise has been built within the STARS team ([9], [10]) and first applications for nuclear power plants have been carried out for the EPR reactor [11]. Also, CFD simulations have been performed to investigate the occurrence of thermal stratification in the lower plenum of a BWR/6 reactor (KKL) during natural circulation conditions. The formation of a stratified layer of cold water, as shown on the right hand side of Fig. 5, is caused by the cooling water of the control rod guide mechanism in the lower plenum and might take place during accidental conditions, when the main circulation pumps are tripped and natural circulation does not contribute significantly to fluid mixing. The correct prediction of thermal stratification is important for the evaluation of thermal shocks on the RPV internals.

## Core modelling and steady-state analysis of the swiss reactors

The PSI Core Management System CMSYS has been established within STARS to serve as code/model integrated platform for the development and validation of reference core models of all the Swiss reactors using the state-of-the-art CASMO-4/SIMULATE-3 codes.

During 2009, CMSYS updates were performed for the PWR KKB1/KKB2 Beznau units ([12], [13]) and for the BWR Mühleberg plant (KKM). For the later, the updates were made following an On-Call requested by ENSI [14] to investigate the reasons for linear heat generation rate (LHGR) related Thermal-Limit (TLM) violations) observed at several TIP measurement points for the CMSYS Cycle 29 and 30 models. Noting that thermal-limit calculations had at this stage not been part of the CMSYS core analysis methodology, a review of the TLM models and correlations was performed after that vendor data had for that purpose been made available. This allowed to eliminate the TLM violations for Cycle 29 but was not sufficient to resolve those for Cycle 30. Further studies identified as most important cause for the violations, an overestimation of the negative void reactivity feedback particularly in the upper part of the core. An overestimated negative void feedback could be caused by many different modelling aspects. However, one reason found

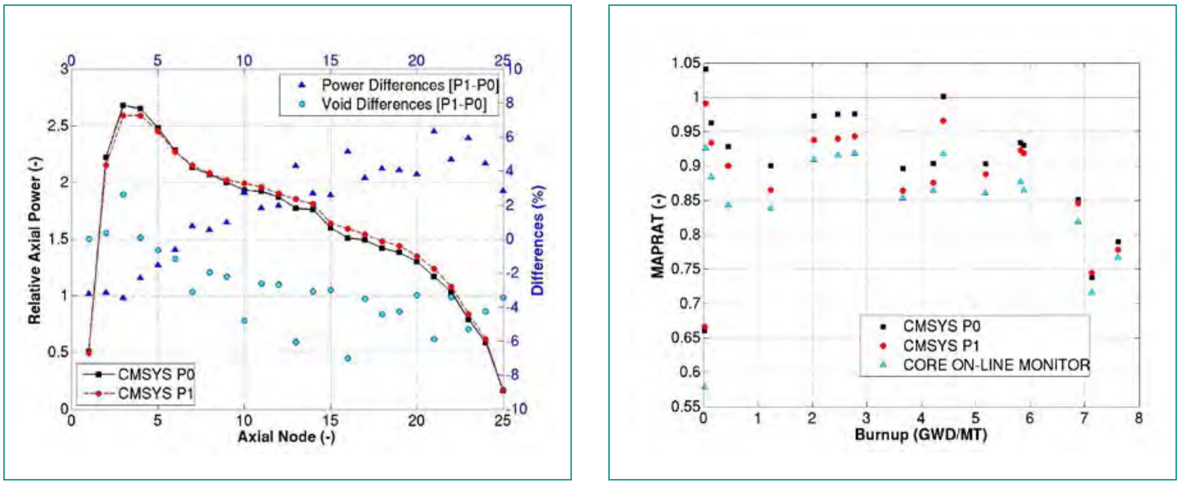



**Figure 6:** *Effect of Spacer void Model on Axial Power/Void Distributions (Left) and Thermal Limits (Right) for Cycle 30 TIP Measurements.*

here is that the special spacervoid model, which was designed to increase the void at spacers locations in order to compensate for a too small negative void coefficient in older neutron libraries, is no longer appropriate with the newer JEF-2.2 library employed for BWRs and characterised by more negative void coefficients. As can be seen on the left-hand side of Fig. 6 showing CMSYS results based on the new JEF-2.2 library, a slightly less top-peaked power distribution is obtained without the spacer-void model (P1) compared to when this model is applied (P0). This is because it reduces the upper core zone void fraction by around 4–5 %. Nevertheless, this small power peaking factor reduction is sufficient to eliminate the thermal-violations of Cycle 30 as can be seen in the right hand side of Fig. 6 where the CMSYS calculated ratio between average planar LHGR and operating limit, referred to as MAPRAT, is shown with (P0) and without (P1) spacer-void model and for completeness, compared to the on-line core monitoring system.

Consequently, all CMSYS models of Cycles 1-21 for the Leibstadt KKL plant were also updated in a similar manner as above, noting that moreover, the development of models for Cycles 22-24 was also initiated. Finally, the assessment of CASMO-5 as new lattice solver was started for the KKM core models and as well as to assess, using its full 2-D core modelling capabilities, the 3-D core simulator pin power reconstruction accuracy for high burnup cores.

## Neutron cross-section modeling and 3-D core dynamics

A large OECD/NEA international benchmark (UAM) has been launched to attempt developing and benchmarking methodologies for the quantification of uncertainties in coupled multi-physics transient analysis. Since highly relevant to STARS, the project intends to participate in this benchmark. The first major challenge relates to the development of methods for the propagation of neutronic uncertainties (covariance data) in reactor physics calculations. Within STARS, the objective is to attempt for a statistical sampling of the XS covariance data but one current main limitation is that this approach is not applicable to CASMO-4, since this code uses adjusted XS libraries in binary format only accessible to the code developers. To partly overcome this, a forward direct perturbation (DP) approach has instead been established that permits, through perturbation of the XS within the code itself, to estimate first-order sensitivity coefficients and to couple these with the specified benchmark covariance data to estimate uncertainties.

Now beside uncertainties directly related to the XS data, the effects on 3-D core dynamics related to the assumptions made to model and parameterize these XS, were subject of further studies during 2009. First, this was done as part of the assessment of SIMULATE-3K as main kinetic solver for both RIAs [15] and for Main Steam Line Break Analyses [16]. Secondly, the studies [17] related to verify the code on the basis of a BWR nuclear heating transient at cold-zero-power (CZP) were continued, this time with focus on XS modelling effects on the Isothermal Temperature Coefficient (ITC) and its consequent effect on 3-D power/void distributions. For instance, as can be seen in Fig. 7, by increasing slightly the ITC through assuming no fission products (XS-B), a much better qualitative and quantitative prediction of the reactor behaviour could be obtained

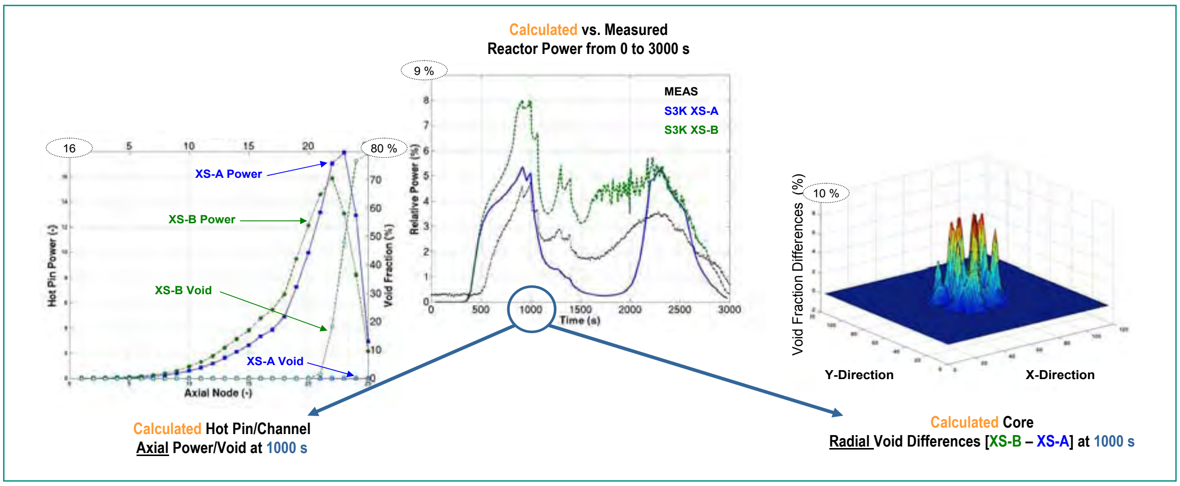



Figure 7: *Sensitivity of 3-D Power from ITC Uncertainties related to XS Modelling.*

compared to the previous model (XS-A) with fission products estimated according to the shutdown period. The reason is that the larger ITC with the XS-B model yields an initial power rise sufficient enough to create void which strongly affects the control rod efficiency (inserted at 1000 s) due to spectral effects, affecting thereby the remaining part of the transient simulation. At the 3-D level, it can be seen that void is produced in the very top part of the core but only with the XS-B model (Bottom-left) and this occurs (at 1000 s) mainly the central core zone (bottom-right). This analysis illustrates the strong multi-physics nature of the processes taking place in a reactor even at very cold low temperature conditions where because of very strong feedback between neutronic/thermal-hydraulics, an accurate simulation can only be done with proper accounting of the major uncertainties, here clearly shown to be related to XS data.

## Fast neutron fluence analysis

Assessment of radiation exposure for the reactor pressure vessel (RPV) material in terms of the fast neutron fluence (FNF) accumulated during the reactor operation is an important part of a PWR reactor safety analysis. Within STARS, an accurate methodology for FNF estimations based on the CASMO-4/SIMULATE-3/MCNPX-2.4.0 system of codes has in previous years been developed in collaboration with swissnuclear. During 2009, the further qualification of the methodology was pursued [18] and an extension towards BWR fluence analyses initiated. And on the basis of the expertise developed so far in this area, an On-Call was requested by ENSI to perform FNF calculations for the KKB-1 RPV in the perspective of life-time extension [19]. Specific aspect of the On-Call request was to provide a comparison of the FNF predictions obtained with the PSI methodology versus the results of the vendor licensing analyses using

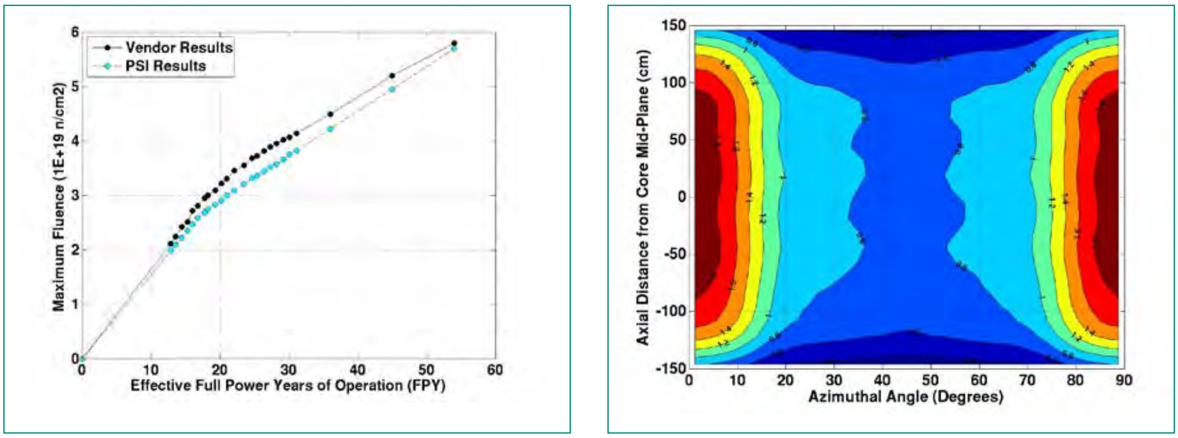



Figure 8: *Comparison of Maximum Fluence (Left) and PSI Estimations of RPV Surface Distribution after 54 FPY (Right).*

its own independent calculation methodology and associated dataset on KKB-1 reactor design and operation history. First as such, the neutron source cycle-bycycle distributions used for the MCNPX neutron transport modelling were evaluated based on the existing CMSYS CASMO-4/SIMULATE-3 models for cycles 16-35. Because the core models for the first 15 cycles are not presently available at PSI, the FNF accumulated during this first period of operation had to be estimated using extrapolation assumptions. Similarly, the maximum FNF beyond cycle 35 and up to 54 years of full power operation (FPY) was extrapolated assuming minor changes in fuel management schemes compared to the later cycles 31-35. The results obtained in that manner were compared with the vendor calculations (estimated from available reports) and a reasonably good agreement was observed as shown on the left-hand side of Fig. 8. Important to note in that context is that up to 54 FPY, the PSI analyses do not yield more penalizing estimations versus the references vendor values. Secondly, an additional objective with the PSI results was to provide FNF axial/azimuthal distribution across the RPV surface as such were required by ENSI in the context of a parallel study on Pressurized Thermal Shocks. An example of the RPV surface normalized FNF distribution, accumulated after 54 FPY according to the PSI estimations, is shown on the right-hand side of Fig. 8, indicating clearly the radial and axial regions where the highest FNF values are predicted.

## Fuel Behaviour model development and assessment

The research activities within the area of fuel behaviour aim principally at enhancements related to the modelling with FALCON of fuel thermo-mechanical behaviour during base irradiation as well as for high-burnup LOCA and RIA transients. This includes on the one hand, model/method development activities and on the other hand, continuous assessment through participation in a wide range of experimental programs. During 2009, progress was achieved in many areas. Among others, FALCON analyses of HALDEN cladding lift-off [20] and CABRI experiments [21], as currently available to the project, as well as an assessment of the code for predictions of advanced PWR/BWR cladding behaviour during power ramp tests ([22], [23]) were completed. Similarly, participation in the HALDEN LOCA experimental program continued. Among others, the FRELAX sub-code, developed specifically to treat complex thermal effects related to axial fuel relocation during LOCAs, was extended with an enhanced model for axial gas flow during ballooning and post-burst phase. The FRELAX model was thereafter coupled to FALCON for analyses of the IFA-650.3/5 tests, confirming an improved accuracy. A major activity during 2009 was also to continue with the development and validation of the FALCON code coupled with the GRSW-A model [24] in the context of RIA transients. More specifically, the analysis of recent RIA tests performed at room (LS-1) and high-temperature

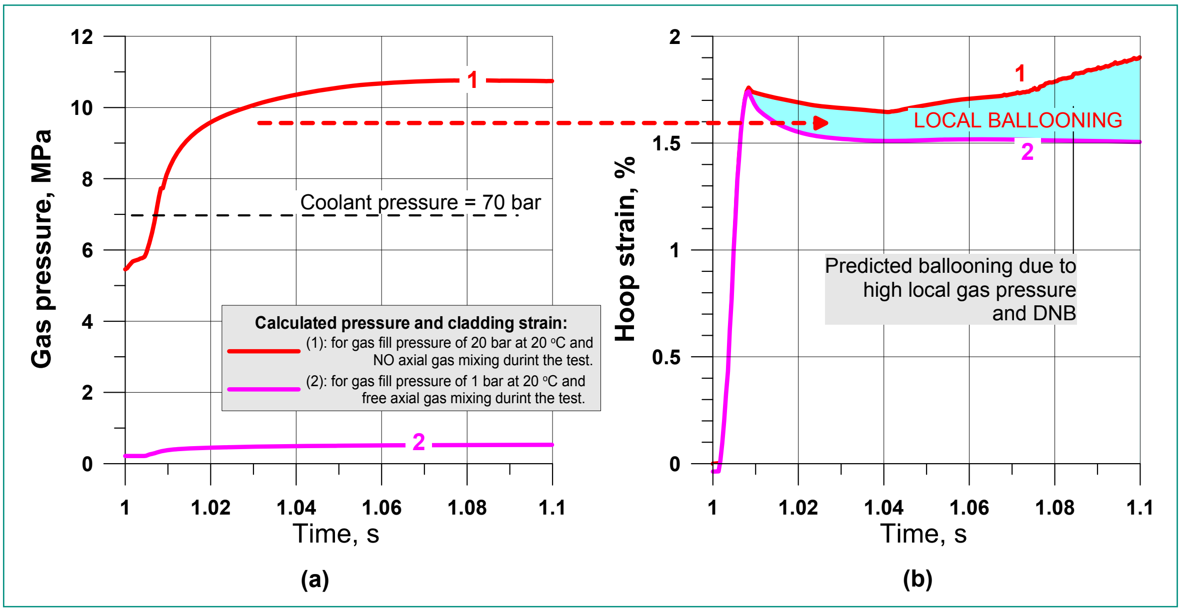



Figure 9: *Calculated dynamics of internal gas pressure against cladding strain during RIA test for the different assumptions on fill-gas pressure (FALCON-PSI).*

conditions (LS-2) in the research reactor NSRR (JAEA-Japan) were continued [25] to study in particular, the effects of coolant conditions (temperature and pressure), initial filling-gas pressure, power impulse width and active fuel length in LWR fuel rod during RIA transients [26]. In that context, the use of low fill-gas pressure (e.g., atmospheric pressure at RT) was studied with FALCON and shown to be a characteristic feature of the NSRR RIA tests that may defer them from the real situations feasible in a LWR. As shown on the left-hand side of Fig. 9, the calculated inner gas pressure in low-pressure test-fuel-rods remains throughout the transient tests well below the coolant pressure – both in the RT- and high-temperature -capsules. Based on a wide range of FALCON calculations using very conservative assumptions, a ballooning type deformation of the cladding is not expected under such conditions. However, further calculations show that if a higher fill-gas pressure (namely: 20 bar at RT, which is typical for a BWR fuel rod at a peak-pellet burn-up of ~70 MWd/kgU) is assumed, a drastic increase of the cladding strain is predicted, causing high-temperature creep (i.e., the local ballooning of the cladding) after the energy insertion, as shown on the right-hand side of Fig. 9. Moreover, as part of the FALCON analyses related to the LS-2 test, a hypothesis was put forward regarding a fast-transient specific mechanism of FGR and fuel swelling in order to address the impact on these from High-Burnup-Structures (HBS). This mechanism, assessed and checked against LS-2 test data, indicates that the normal structure (pellet centre and bulk) and not the HBS (pellet rim) contributes mostly to the transient FGR during a RIA. However, these are only very preliminary results and therefore, further verification using a wider range of experimental data remains highly necessary.

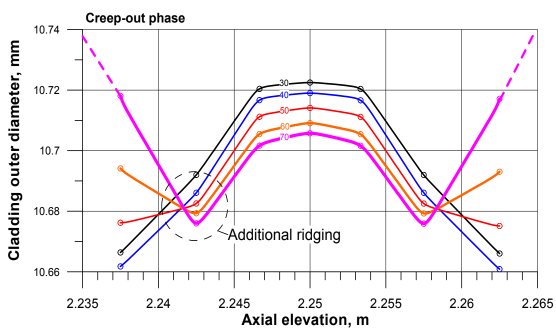



Figure 10: *Calculated evolution of cladding outer diameter profile around a hypothetical missing pellet region for the stabilized creep-out phase.*

## Fuel modelling studies of cladding failures in KKG Cycle 29

Fuel damage (e.g. primary cracks and cladding fretting/wear, along with secondary fissures) has been observed in KKG fuel during Cycle 29. Although the reasons for this are not yet fully understood, a significant local cladding shrinkage was also observed in some of fuel rods with moderate burnups (i.e. 2 cycles of operation). On the basis of the project fuel behaviour expertise developed as part of the R&D activities mentioned previously, an On-Call was requested by ENSI [27] with the objective to perform numerical studies to investigate if eventual deviations in local parameter characteristics could cause such a significant cladding shrinkage as observed at KKG and if such deviations could in turn result in cladding failures.

To this end, a number of hypothetical assumptions, regarding local deviations of the pellet parameters and cladding thermal conditions were considered. To start, an eventual physical absence of a pellet or an incidental presence of a low-enrichment (or totally inactive) pellet were assumed but the FALCON analyses out-ruled these as plausible causes. The results, as illustrated in Fig. 10, showed indeed that before reaching very high burnups, local prominences (regions with a larger diameter) would have been observed along with the regions of cladding shrinkage.

And this was not the case according to the vendor documents made available at the time of the present study.

Among other assumptions considered, an excessive densification of singular fuel pellets and/or local overheating of the cladding, accompanied by an increase in the cladding creep rate, were studied with FALCON. And in both these cases, it was found that a significant cladding shrinkage could indeed be obtained at burnups similar to the affected rods. On that basis, an additional study was carried out to investigate if these local features identified as plausible causes for the shrink-age, could have resulted in cladding failures. In that context, it was found that cladding overheating and, as a result an enhanced creep-down rate, could result in a considerable aggravation of the critical characteristics of fuel rod behaviour during the power ramp. A recommendation on the reduction of the limit-power-ramp-level was finally proposed as a possible countermeasure. The applied methodology to quantify such reduction was also presented, noting that it could later serve as basis to modify an existing operation criterion relating to the limiting power ramp level.

## National Cooperation

To carry out its research and scientific support activities, the STARS project collaborates with ENSI as well as with swissnuclear and the Swiss individual nuclear power plants. Along this, the project also collaborates with other laboratories and departments 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 involved in an increased collaboration with the Swiss federal polytechnic institutes ETHZ/EPFL for the elaboration, supervision and realisation of relevant challenging MSc and PhD theses.

## 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 7$^{th}$ FP NURISP project and on the other hand, through bilateral cooperation e.g. GRS, CEA, Purdue 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-4/SIMULATE-3/SIMULATE-3K) and EPRI/ANATECH (FALCON).

## Assessment 2009 and Perspectives for 2010

Most of the objectives set for 2009 were achieved as planned without affecting the project ability to also perform the On-Calls requested during the year by ENSI, including the realisation on short-notice of a training course. Along with this, the project team was also involved during 2009 in a substantial revision and update of its quality management system PMS, which was successfully certified according to ISO 9001:2008 in July 2009 with no major recommendations for enhancements by SQS, the certifying body. The only significant deviation from the objectives concerns the NURISP project where no substantial progress could be achieved. However, this is mainly due to the non-availability of the NURISP codes which in most cases, are planned to be distributed by the consortium organisations only during 2010.

Concerning the perspectives for 2010, the starting point is to renew the two contracts with ENSI consisting of a Scientific Support and an R&D contract. On that basis, the technical objectives of the project are as follows.

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| Performance of On-Calls according to 2009 STARS/ENSI bilateral plan |
| Safety analyses with TRACE of EPR, incl. SBLOCAs, LBLOCAS and MSLB, and application of CFD for boron dilution |
| Assessment of FLICA and CFD towards development of Sub-Channel Methodology |
| Application of Dynamical Coupled CFD/TRACE model for selected PWR Transient |
| Development and Optimisation of CRONOS/FLICA numerical coupling schemes within NURISP SP3 |
| Further development of TRACE/S3K coupling scheme and associated modelling methodologies for the Swiss Plant/Core models |
| Core Model Updates for CMSYS KKG Cycles 22-30 and Assessment of CASMO-5 also for full 2-D core analyses |
| Development of BWR Stability Analysis Methodology using S3K and on the basis of KKL |
| Establishment and assessment of deterministic neutronic uncertainty propagation methods |
| Development of APOLLO-2 lattice computational routes for nodal and cell cross-section libraries within NURISP SP1 |
| Development of non-axis symmmetric modelling capability in FALCON |
| Assessment of MOX Models in FALCON and/or application to CABRI WL tests |
| Pre-analysis and post-analyses of next HALDEN LOCA test with high-burnup KKL fuel |

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