STARS

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

Author and Co-Author(s)	Martin A. Zimmermann and collaborators from the project team
Institution	Paul Scherrer Institut
Address	5232 Villigen PSI
Tel., E-mail, Internet address	+41 (0)56 310 27 33, Martin.Zimmermann@psi.ch http://stars.web.psi.ch
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ABSTRACT

During 2008, the STARS project has pursued research in the fields of fuel behaviour modelling for high-burnup fuel, system behaviour modelling for Swiss NPP's as well as for the EPR using TRACE (including the incorporation of CFD) and investigations on the core behaviour.

An explorative investigation of the impact of gaseous swelling for the hypothetical case of a BWR RIA experiment in the NSSR using the high-temperature capsule suggested its relevance especially in view of high-burnup fuel. The FALCON code with the coupled GRSW-A fission gas model fission was also very effective in analysing the swelling behaviour of a BWR fuel rod up to a burnup of 95 GWd/t. Furthermore, good predictions of fission gas release for a set of BWR rods was achieved.

The TRACE code has further matured. The migration of the legacy BWR plant models for TRAC-BF1 was completed and good TRACE results for the KKL SEHR ADS event of 2007 as well as a good comparison to the TRAC-BF1 results for a KKM LBLOCA case can be reported. Good progress has also been achieved for the EPR-project for both the system and the CFD modelling approach.

The participation in BEMUSE documented the good performance of the uncertainty evaluation methodology adopted as PSI, using TRACE and SUSA. Results well in line with the majority of the participants were obtained.

A PhD study developing the coupling between a CFD code (CFX) and TRACE has progressed well: The comparison of the coupled system against data from a dedicated experiment reveals good performance of the coupled codes.

Considerable effort was again spent on updating the core models, and very satisfactory agreements with the plant measurements were achieved.

SIMULATE-3K was employed to supplement an earlier CORETRAN RIA analysis for MOX cores, investigating the realistic pulse width in function of the core composition. The investigation of a BWR start-up transient using SIMULATE-3K emphasized the necessity of a detailed branching scheme a cold zero power for the development of the cross-section library in order to properly represent the positive moderator coefficient found in these core conditions.

In the framework of the NURESIM project, the Peach Bottom turbine trip has been reanalyzed using the SALOME tools, thereby establishing the BWR situation target. A parametric mixing study for a MSLB scenario using RETRAN-3D pointed to the sensitivity of the assumption of the blocked control rod to the mixing pattern. Finally, the criticality safety evaluation of the external wet storage pool for enriched reprocessed uranium confirmed the KKG values submitted to HSK.

Project Goals

Goals for 2008

- Fuel Behaviour
  - – Continue analysis of selected RIA and LOCA experiments from the ALPS program
  - – Continue analysis of SCIP ramp tests
  - – Further develop fission gas models and perform necessary validation
  - – Participate in CSNI/WGFS LOCA benchmark with analysis of IFA-650.4 / IFA 650.5
  - – Establish framework for statistical fuel analysis
- Systems Behaviour
  - – Develop necessary plant models for EPR reactor
  - – Pursue the migration of existing plants models for legacy codes (RELAP5/TRAC-BF1)
  - – Perform selected analyses to assess BWR capabilities of TRACE
  - – Complete uncertainty analyses in the framework of CSNI/BEMUSE program
  - – Couple TRACE with a CFD code: first test runs
  - – Initiate the research program on combination of dynamic event tree and best-estimate thermal-hydraulic codes (Seed-action '06)
- Core Physics
  - – Update Swiss core models (CMSYS)
  - – Initiate migration to CASMO-5 / SIMULATE-4
  - – Couple SIMULATE-3K to TRACE / RETRAN-3D
  - – Within NURESIM, continue exploration of APOLLO-2 for application to core analysis at nodal and pin-level and perform the work necessary to achieve the BWR situation target (turbine trip transient at the core level)
  - – Participate in first exercise of NSC/UAM benchmark (neutronic uncertainty in view of coupled analysis)

Work Carried Out and Results Obtained

Fuel Behaviour

Computational Interpretation of the ALPS Test LS-1 Conducted using KKL Fuel

Further work has been conducted during 2008 in continuation of the analysis of high-burnup fuel behaviour during tests simulating Reactivity Initiated Accident (RIA) [1] in the Nuclear Safety Research Reactor (NSRR, Japan). The activity has been aiming at the exploration of the physical processes and material properties governing fuel rod behaviour (e.g. Pellet Cladding Mechanical Interaction (PCMI), cladding strain and failure, etc.) during RIA. The analysis has been performed using the improved version of the FALCON fuel behaviour code recently developed at PSI by means of coupling of the standard FALCON code [2] with the mechanistic model GRSW-A [3] for fission gas behaviour and structural evolutions in the uranium dioxide fuels.

After only a minor effect of gaseous bubble swelling had been shown in the LS-1 test rod failure by the previous study [4], the recent analysis was directed towards the evaluation of the potential impact of retained fission gases in the irradiated fuel, particularly through gaseous bubble swelling, on the fuel rod thermal-mechanical behaviour during RIA for the LS-1 test of the fuel segment pre-irradiated in the KKL BWR [5]. To this end, a numerical study was performed for a hypothetical test with parameters similar to the LS-1 test, but implemented in the newly-designed high-temperature (HT-) test-capsule with coolant conditions of 559 K and 7.0 MPa. The analysis has shown a significant impact of gaseous bubble swelling in the transient behaviour of the fuel rod, given high enough cladding ductility to survive the power pulse. This comparison of two calculations performed with and without consideration of transient fission gas behaviour is illustrated in Fig. 1.

It was a priori assumed that the coolant conditions prevailing in the HT-capsule should have resulted in a higher resistance to cladding failure than for the comparably low coolant temperatures and pressure of the RT-capsule. Partly, this assumption was based on the increasing solubility limit for hydrogen in Zr-alloys [7]. On the other hand, an essentially different mechanical behaviour has been revealed by the calculations for the two types of boundary conditions, which is shown in Fig. 2, viz., a purely elastic regime up to the experimentally observed moment of failure in case of the RT-capsule against the early transition to plastic straining in case of the HT-capsule conditions. The result of these calculations was interpreted as additional evidence in favor of the necessity to discriminate between these two types of coolant conditions when defining the critical enthalpy for cladding failure caused by RIA. This statement is to be verified by the new series of the RIA-simulating tests in NSRR using the recently designed HT-capsule.

Application of the GRSW-A Model to the Analysis of Steady-State Fuel Swelling under Irradiation to a High Burn-up

Fuel swelling under irradiation is an important issue of fuel performance modeling as far as pellet growth due to swelling (along with cladding creep deformation) is a key factor determining the moment of pellet-cladding gap closure and the level of tensile hoop stress in the cladding thereafter.

This work aimed at the application of the GRSW-A model [3] to the analysis of the processes related to fuel swelling in high-burnup fuel under normal irradiation in LWRs. The analysis was performed for pellet-averaged swelling in the pellet of an idealized fuel rod with the parameters similar to those of the BWR KKL fuel rods [5]. Note that a hypothetical history of the Linear Heat Generation Rate (LHGR) was used in the present calculation, which extended the fuel pellet burn-up at End-of-Life (EOL) to quite a high level of about 100 MWd/kgU. This was achieved by adding four equal low-power cycles to the real 7-cycle power history from the BWR KKL [5].

The results of the corresponding calculation are presented in Fig. 3 where a very good agreement can be seen between the calculation by FALCON-PSI and the fundamental experimental finding [6], viz. (1) reduction of the intragranular pellet swelling rate with burn-up due to the depletion of the matrix fission gas at the pellet periphery subjected to intragranular polygonization, which (2) entails the onset and further growth of grain-boundary bubble swelling due to the formation of pores in the so-called high burnup structure (HBS). The resulting tendency towards an increase of the total swelling rate is predicted by the calculation, as shown in Fig. 3. This finding seems to be consistent with the available data on pellet immersion density. The latter prediction seems to be especially important for the thermal-mechanical analysis of the fuel rod behaviour under normal operation conditions, as far as the pellet swelling rate is the crucial factor for the strain and stress arising in the cladding after gap closure.

Computational Interpretation of the Data for Fission Gas Release in the High-Burnup Fuel Rods from the BWR KKL

A new model for fission gas behaviour and evolution of the fuel microstructure in uranium dioxide fuel under irradiation has been developed at PSI. Moreover, the model is coupled with the FALCON fuel behaviour code, which is extensively used for the analysis of fuel performance during normal operation and slow power transients, as well as for the analysis of the fast thermal transients during the Reactivity Initiated Accident (RIA) and Loss Of Coolant Accident (LOCA). The work addressed

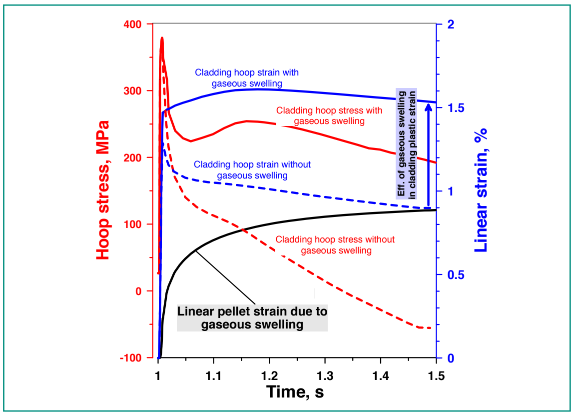

Fig. 1: Characteristics of cladding mechanical Behaviour against pellet gaseous swelling calculated for the LS-1 test assuming real Room Temperature conditions and hypothetical High Temperature High Pressure conditions.

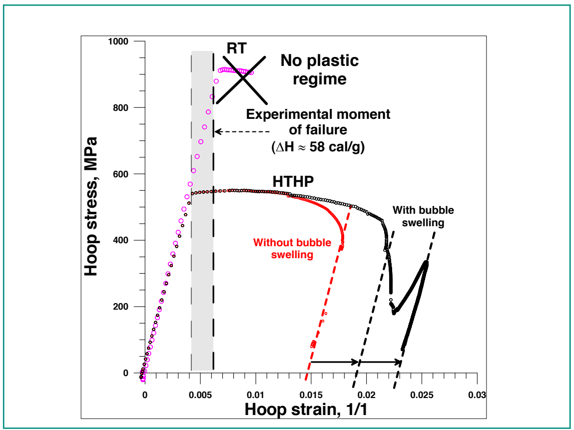

Fig. 2: Strain-stress diagrams calculated for the transient phase of LS-1 test assuming the real RT- and hypothetical HTHP-capsule (High Temperature High Pressure) conditions.

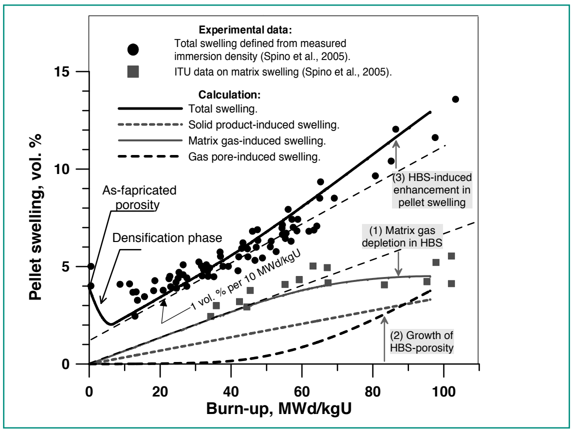

Fig. 3: Comparison of pellet-averaged swelling results in a high-burnup BWR fuel rod against experimental data.

the validation of the PSI version of the FALCON code as applied to normal steady-state operation of the fuel rods in a commercial LWR, namely the BWR KKL. Besides, some possible mechanisms and parameters responsible for the increase of Fission Gas Release (FGR) observed in the KKL fuel rods were offered on the basis of the analysis.

The FALCON code coupled with the GRSW-A model can be applied to a wide range of fuel modeling problems, including analysis of fuel behaviour under normal conditions during the commercial irradiation in the power reactors to an extended level of burn-up. In particular, the investigation has been recently conducted for seventeen fuel rods irradiated in the KKL BWR to a high level of burn-up. The different levels of burn-up ranged from a relatively modest value of about 35 MWd/kgU to about 70 MWd/kgU (pellet). All the fuel rods selected for the present analysis were fitted with the experimental data for the relative FGR at EOL [5], [7] obtained by puncturing in the Hot Laboratory and/or by measurement of the Krypton-85 activity in the plenum conducted at the NPP.

Generally, the nominal values were accepted for the code parameters in the present calculations, specifically parameters of the models for pellet fragment relocation and fuel densification, except for the supplementary parametric study of the effects of these parameters on some of the predictions. It is to be noted that the higher temperature can be obtained from the calculation when setting the parameters of fuel relocation and densification to values corresponding to the larger effective fuel-cladding gap at the beginning of irradiation and vise versa.

The agreement of the calculation with the experimental data, as shown in Fig. 4, is reasonably good in view of the large scatter of the experimental points. Also evident is the significant improvement of the results of FALCON-PSI compared to the results from the ESCORE FGR model of the standard FALCON code [2]. Specifically, it is seen in Fig. 5 that the condition for the onset of FGR has been determined reasonably well by FALCON-PSI compared to the experimental data. This refers also to the predicted rate of FGR increase with burn-up. Furthermore, rod-by-rod verification of Fig. 4 reveals that nearly all the points of the FALCON-PSI prediction fall into the so-called licensing-acceptable sector (shown as the grey region limited by the two broken lines), which corresponds to the error in relative FGR prediction not exceeding 100 %. It is to be noted that virtually all the experimental points obtained by puncturing, which is believed to be the most credible of the experimental methods for FGR measurement, are located in the upper half of the mentioned segment, which suggests a conservative prediction.

It is well seen from the comparison of the calculated peak centerline fuel temperature in one of the fuel rods with the empirical FGR threshold [8], presented in Fig. 3, that the only way for the present modeling to cope with the sharp increase of the measured FGR is by assuming the existence of some a-thermal mechanism(s) of FGR acting in extended burn-up fuel. Evidently, these mechanisms are to be somehow related to the well-known features of the High Burn-up Structure (HBS) in the fuel pellet rim. Indeed, according to the present analysis, the contribution of standard FGR mechanisms resulting from the thermally-induced diffusion of the fission gas

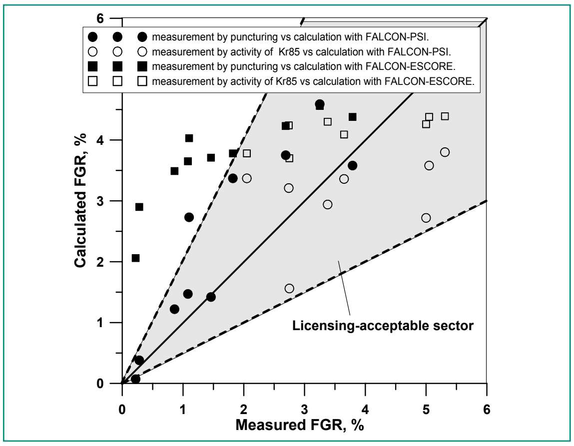

Fig. 4: Results of calculation against measurement for FGR in selected fuel rods irradiated in BWR KKL.

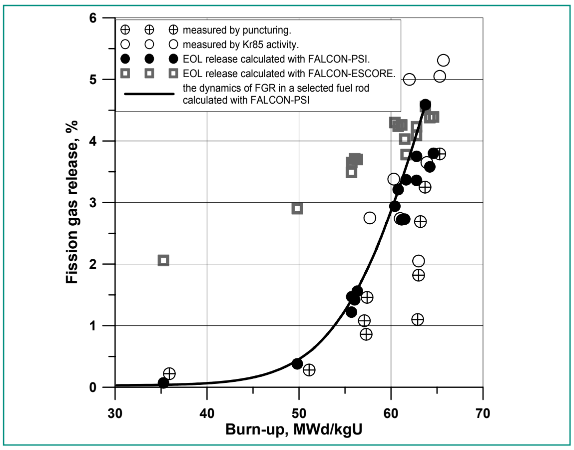

Fig. 5: Calculated and measured FGR in function of fuel rod-averaged burn-up for selected fuel rods irradiated in BWR KKL.

in the pellet center is essentially limited by very low fuel temperatures throughout irradiation. The prediction of low temperatures has been checked using most conservative assumptions for the parameters characterizing fuel pellet relocation and densification (see high-temperature curve of Fig. 6).

Systems Behaviour

KKL ADS event simulation by TRACE 5.0

The event «Fehlerhaftes Aktivieren von SEHR-ADS» that occurred at the KKL power plant on March 6 2007 has been formerly simulated with the TRAC-BF1 code [10] in order to support the interpretation of the plant data [11]. Recent efforts have been dedicated to the employment of the TRACE [12] code for LWRs transients analyses. TRACE is a modernized thermal-hydraulic code designed to consolidate the capabilities of US NRC's legacy safety codes (RELAP, TRAC).

The existing TRAC-BF1 input deck of KKL was converted into a TRACE deck. The control system model, consisting of more than 2000 control blocks, was rewritten to reflect the differences in the two codes logics. To validate the newly developed TRACE input deck of KKL, the KKL SEHR event [13] was simulated. The results were successfully compared with the TRAC-BF1 results. Then the KKL ADS event of March 2007 was simulated. The transient was initiated by the spurious opening of the 8 relief valves of the Special Emergency Heat Removal (SEHR) Automatic Depressurization System (ADS) during an integrated test on the SEHR system [11]. This resulted in a fast depressurization of the primary system. The primary pressure decreased from 7.4 MPa to 0.6 MPa. During the transient, significant water level swelling in the RPV occurred as a consequence of the steam flashing in the system. The possibility of liquid carry-over to the steam line and from there to the Safety Relief Valves (SRVs) needed to be evaluated, since these valves are not designed for operation under two-phase flow conditions.

The results of the TRACE simulation are in good agreement with the plant data. As a matter of fact, the main reactor parameters, namely reactor levels, reactor pressure (see Fig. 7), flow rates, etc., are even better predicted by the TRACE code when comparing with the results previously obtained by TRAC-BF1. The amount of liquid reaching the SRVs was estimated to be sufficiently less compared to the former TRAC-BF1 results, as illustrated in Fig. 7.

The TRACE results will be used to define the boundary conditions for further detailed CFD analyses aimed at the estimation of risks of thermal stresses for the RPV internals.

KKM modeling with TRACE

The purpose of this work was to convert a TRAC-BF1 thermal-hydraulics code input deck model of KKM power plant into a TRACE code model and assess the new model.

After establishing steady-state conditions, through a so-called «null transient» whereby the driving functions remain constant, two scoping transients have been performed to assess the input deck conversion, and also to identify potential impacts of the differences between the thermal-hydraulic models implemented in these two codes.

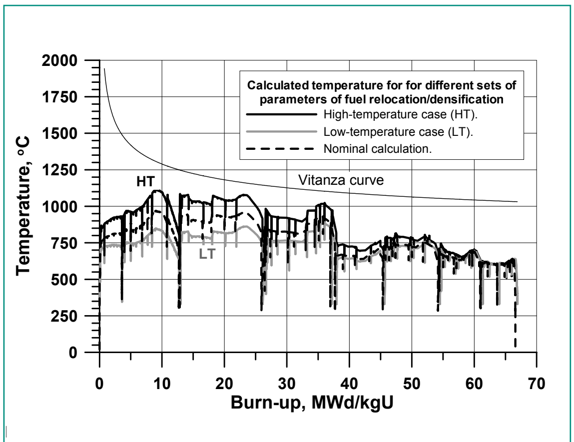

Fig. 6: Calculated centerline fuel temperature in a high-burnup fuel rod during irradiation in KKL BWR.

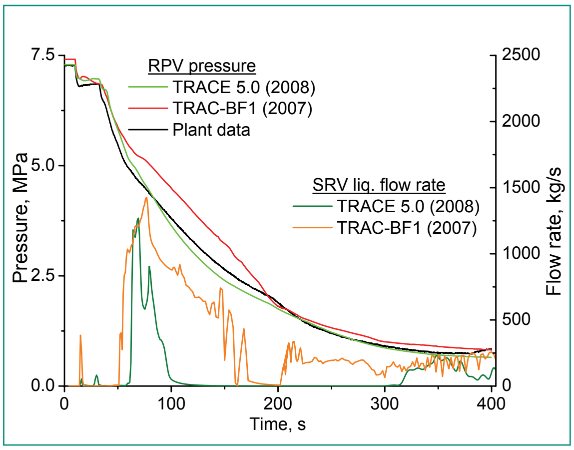

Fig. 7: Reactor pressure and liquid flow through SRVS.

Thus, a large-break loss-of-coolant accident (LBLOCA) and a loss-of-feed-water (LOFW) with stuck-open safety relief valve (SRV) scoping transients have been performed for comparisons with existing TRAC-BF1 code results [14], [15]. These transients were driven as in the TRAC-BF1 simulations for consistency and have not revealed significant difference between the code predictions, as depicted in Fig. 8 for the LOCA.

After establishing steady-state conditions, the blowdown is initiated at 100 seconds. Subsequently to reaching a peak cladding temperature of ~1060 K, all rods are quenched at ~690 seconds. The TRAC-BF1 code predictions are similar with a peak temperature of 1085 K and an earlier quench time (~100 seconds sooner). The first temperature peak is followed by a short-duration return to nucleate boiling (at ~130 seconds) before significant post-CHF conditions develop. One can observe that TRACE results are consistent in predicting delayed quench times after both temperature peaks.

As for the LOFW with stuck-open SRV simulation, the system pressure behavior was of particular interest due to its impact of the SRV behavior (e.g., valve cycling in the case of a total LOFW) and was found to compare reasonably well with the TRAC-BF1 results.

This work will be pursued by performing simulations of a total loss-of-feed-water (LOFW) transient and comparisons of predicted results with KKM plant data.

Safety Evaluations of EPR

In cooperation with the Finnish nuclear authority (STUK), a project has been launched aimed at safety evaluations of the EPR reactor. Main focus is given to large- and small-break LOCAs and to main steam-line breaks. The mixing in the reactor pressure vessel, responsible for the boron and temperatures distributions at the core inlet, has to be analyzed by means of a CFD tool, with the boundary conditions supplied on the basis of the TRACE simulations. The results of the TRACE simulations will be used as input for detailed fuel behaviour evaluation during LOCAs.

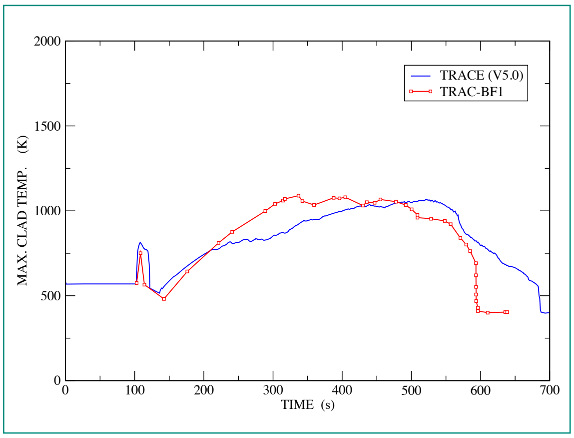

Fig. 8: Comparison of maximum cladding temperatures for KKM LBLOCA.

A full TRACE nodalization of the EPR has been developed. The model represents all four loops and the corresponding steam generators separately. Two nodalization schemes are employed for the vessel, a one-dimensional and a three-dimensional respectively. TRACE three-dimensional components have been employed for the nodalization of the steam generators, in order to take into account the special construction of these components (the feed-water flows into a downcomer wrapper that surrounds only the cold side of the steam generator). In Fig. 9, the nodalization of a single loop including steam generator and PZR is reported (left) together with a scheme illustrating the special features of the EPR steam generators (right). For the development of the TRACE 3D RPV nodalization, a novel tool has been developed within the NURESIM Salome' platform in collaboration between PSI and CEA, in order to automatically generate a TRACE input for the RPV by superimposing a TRACE mesh to the geometry CAD files.

In parallel to the TRACE nodalization, a complete set of 3D CAD models (Fig. 10, left) of the reactor pressure vessel and its internals has been generated on the basis of the supplied drawings. The CAD model has been used as starting point for the construction of the CFD meshing. Particular attention has been dedicated to the generation of a hexahedral mesh, since this type of mesh is subject to less numerical diffusion and provides more accurate results. In Fig. 10 (right) an example of the mesh strategy is shown, as applied to the mixing device present before the core inlet plate.

Scoping simulations are currently being carried out in order to verify the TRACE model. In parallel, stationary STAR-CD simulations are being performed, aimed at optimizing the CFD mesh of the EPR RPV and its internals.

OECD/NEA BEMUSE Programme Phase V: Uncertainty analysis for a LBLOCA in ZION nuclear plant

The BEMUSE (Best Estimate Methods – Uncertainty and Sensitivity Evaluation) Programme, promoted by OECD/NEA, aims at the evaluation of uncertainty methodologies applied to the predictions of best estimate system codes. The first phases of the Programme addressed the application of best-estimate (BE) codes and uncer-

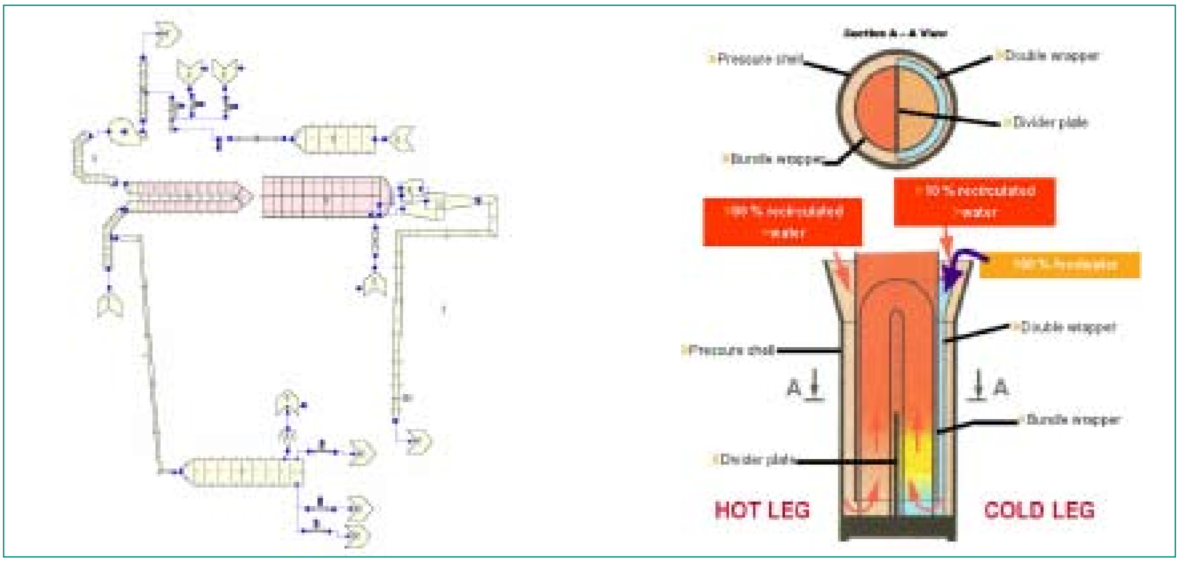

Fig. 9: TRACE LOOP model (left) and steam generator configuration (right).

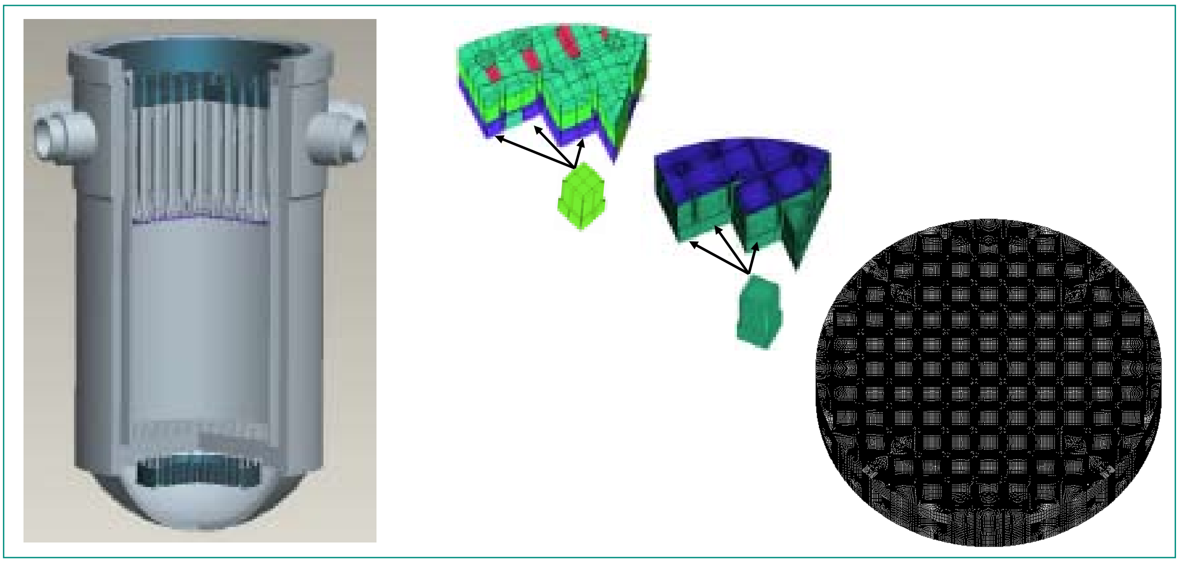

Fig. 10: CAD model of reactor pressure vessel (left) and example of meshing strategy (right).

tainty methodologies to a LOCA in the LOFT integral test facility. Phase IV [16] was focused on the sensitivity analysis for a LBLOCA in the Zion nuclear power plant, a 4-loops PWR. In the following, the results obtained during Phase-V of the BEMUSE programme are reported. This phase is concerned with the uncertainty analysis of a LBLOCA in the Zion NPP.

The Phase V of the BEMUSE Programme was carried out and completed. It consisted of the un-certainty analysis of a LBLOCA in the 4-loop PWR Zion reactor. This analysis is a follow-up of the activities carried out during phase IV of the program, during which the LBLOCA base case was evaluated and the sensitivity of the results to different parameters was investigated. The TRACE

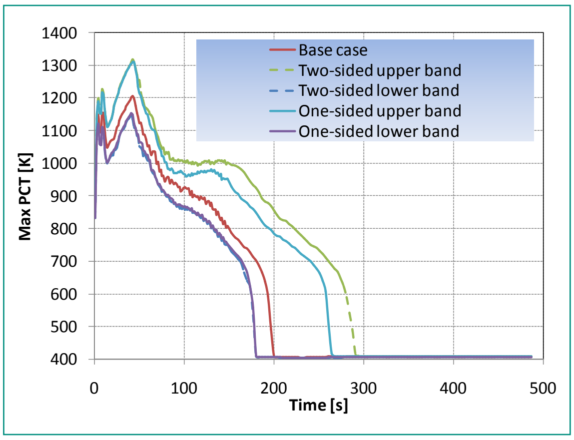

Fig. 11: PSI TRACE results for BEMUSE LBLOCA maximum cladding temperature.

1st PCT	Lower band (K)	Base case (K)	Upper band (K)	2nd PCT	Lower band (K)	Base case (K)	Upper band (K)
AEKI	1160	1216	1343	AEKI	1126	1200	1467
CEA	1168	1252	1326	CEA	1045	1127	1373
EDO	1212	1306	1382	EDO	1216	1326	1450
GRS	1190	1293	1393	GRS	1112	1251	1365
IRSN	1142	1218	1379	IRSN	960	1149	1308
JNES	1075	1185	1234	JNES	998	1076	1132
KAERI	1129	1187	1237	KAERI	1174	1247	1336
KINS	1178	1244	1375	KINS	1213	1291	1435
NRI1	1017	1191	1301	NRI1	1090	1220	1304
NRI2	1080	1189	1374	NRI2	1075	1219	1459
PSI	1131	1178	1237	PSI	1163	1208	1313
UNIPI1	906	1054	1176	UNIPI1	848	1198	1418
UNIPI2	792	1204	1368	UNIPI2	994	1218	1342
UPC	1069	1187	1324	UPC	1114	1189	1342
Mean	1089	1207	1318	**Mean**	1080	1208	1360
Std Dev	117	60	70	**Std Dev**	104	64	88

Table 1: Uncertainty results for 1st and 2nd PCT.

Complete core quench	Lower band (s)	Base case (s)	Upper band (s)
AEKI	112.3	259.0	334.6
CEA	247.3	370.3	583.5
EDO	124.1	136.1	379.3
GRS	179.4	273.1	423.8
IRSN	248.7	430.8	616.5
JNES	230.0	332.0	395.0
KAERI	152.1	209.8	< 1000.0
KINS	145.7	194.7	286.4
NRI1	125.0	162.8	197.8
NRI2	158.9	192.4	214.0
PSI	178.7	199.5	263.1
UNIPI1	172.0	264.0	356.0
UNIPI2	228.0	324.0	420.0
UPC	151.0	205.0	265.0
Mean	175.2	253.8	364.2
Std Dev	46.2	84.6	128.1

Table 2: Uncertainty results core quenching time.

nodalization developed at PSI within BEMUSE includes all four loops, and corresponding steam generators, individually a 3D representation of the reactor pressure vessel. ECCS injections and accumulators are modeled only for the intact loops. In total, the nodalization consists of 908 hydraulic volumes and 5117 heat structures nodes. A guillotine break in a cold leg is assumed at time zero. Following the BEMUSE-V specifications [17], 20 parameters where taken into account in the uncertainty analysis, including material properties, initial and boundary conditions. The distribution and variation range of the selected parameters where assigned in the specifications and based on expert opinion [17].

120 simulations were performed for the uncertainty analysis, and the probabilistic GRS method (SUSA) was used in order to propagate parameters uncertainties to the output variables. In Fig. 11 the maximum cladding temperature is reported as function of time, together with the one- and two-sided uncertainty bands corresponding to the 5 % (lower bands) and 95 % (upper bands) quantiles respectively, with a confidence level of 95 %. The calculated upper limit for the PCT lies below the safety criterion. This was the case also for the other participants of the BEMUSE Programme.

A comparison between PSI and the other participants' estimations is reported in Table 1 for the 1st and 2nd peak cladding temperature and in Table 2 for the core quenching time. The PSI results are within the mean values and corresponding standard deviations obtained by averaging the results from all organizations. The standard deviation for the upper value of the core quenching time is rather high due to the fact that some organizations (CEA, IRSN and KAERI) predicted a rather long quenching period.

A draft report on the results of phase V has been prepared [18]. The finalized version will be issued in early

2009. The last phase of the Programme (phase VI), summarizing methods and guidelines, will be completed in the first half of 2009.

Thermal Hydraulic Analysis on PKL Boron Dilution tests

The PKL-1 test investigated pressurized water reactor (PWR) safety issues, specially, such as: Boron dilution accidents and loss of residual heat removal in mid-loop operation (during shutdown conditions).

During recent years, the best estimate system code TRACE has evolved as the recommended and supported tool by the United States Nuclear Regulatory Commission (USNRC) to simulate Light Water Reactors (LWR). The improvements in version 5 are conspicuous as the code has been able to simulate accurately different scenarios in test facilities and commercial NPP's.

One of the main areas of interest is the simulation of SBLOCA with boron dilution transients. From 2000 to 2007 different tests have been carried out in the PKL test facility with the purpose to provide valuable experimental data on boron dilution transients. Many participants presented simulations of these tests with satisfactory results using different codes. The worked performed at the Polytechnic University of Catalonia (UPC) was of particular interest as all SBLOCA with boron dilution tests were simulated with the same RELAP5 nodalization [19].

A well tested RELAP5 model of the PKL test facility developed by Universitat Politècnica de Catalunya, Barcelona, (UPC) has been converted into a TRACE input deck and the results obtained for tests F1.1 [20] and F1.2 [21] have been compared.

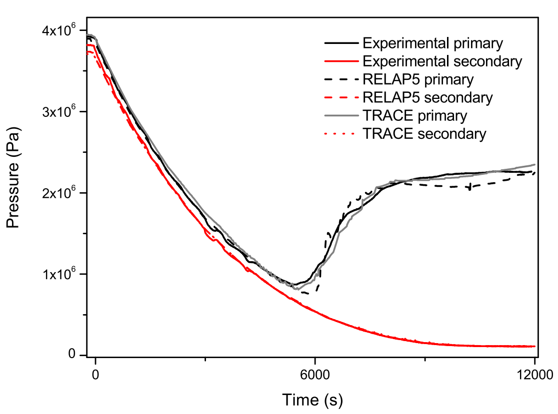

Fig. 12: Primary and secondary pressures for test F1.1: Experimental, RELAP5 and TRACE.

The results obtained for Test F1.1, which is a SBLOCA with boron dilution transient, are satisfactory. A comparison between a RELAP5 simulation and the experimental data show no major discrepancy. The transport of boron is also tested and the results of the two codes are compared, even though a substantial reduction of numerical diffusion is needed for the TRACE code. Therefore, a new numerical scheme for the solute transport equation, aimed at the reduction of numerical diffusion and the introduction of a physical turbulent diffusion coefficient, will be implemented in the TRACE code. The results will be compared to the one obtained with a modified Godunov scheme implemented in RELAP5 [23].

PKL Test F1.2, which is a parametrical study of the inherent boron dilution due to reflux-condenser conditions as a function of the primary coolant inventory, has been simulated as well. Some discrepancies related to the performance of the separators need still further investigations.

After the completion of PKL-1 project, the PKL-2 test program has started in April 2008. The program is focused on safety issues relevant for current PWR plants as well as for new PWR designs, and in particular on complex heat transfer mechanisms in the steam generators and boron precipitation processes under postulated accident situations.

Validation of TRACE reflood model

Reflooding is a phase of a LOCA accident which occurs after the injection of ECCS water. The maximal cladding temperatures during a LOCA transient are usually realized during the Reflooding stage. The proper simulation of reflooding is a crucial issue for the safety assessment. Three sets of reflood experiments, ACHILLES [24], NEPTUN [25] and FLECHT-SEASET [26], were used to validate the TRACE 5.0 code applicability to reflooding simulation.

A TRACE model of the ACHILLES facility has been developed and 74 experiments have been analyzed. The results obtained showed a satisfactory agreement with the experimental data for cladding temperatures, quench front and collapsed water level. The typical behaviour of the cladding temperature is shown in Fig. 13 for RUN A1R029 with a heat generation rate corresponding to 90 % of the ANS decay heat curve. Some differences have been observed when using a VESSEL component (three-dimensional) instead of a PIPE component (one-dimensional) to model the test section. TRACE input decks for two available NEPTUN experimental runs were developed as well and the simulations results were compared with the experimental data. For the FLECHT-SEASET experiment, instead, the input deck developed by Gene Rhee and Jae-Hoon Jeong [27] was used. The simulation results were in agreement with the one reported in the TRACE Assessment Manual data [28].

The reflood model in TRACE performs satisfactorily. The differences between simulations and experimental data are observed mainly for the upper part of test sections. These differences may be caused by a) lack of a spacer grid model in TRACE; b) lack of simulating capability for the top quench behaviour; c) vessel shroud heat transfer and 3D effects, e.g. different mass flow in the central and peripheral areas. It can be concluded, based on the comparison with the experimental data, that the TRACE code is capable of calculating the reflood process with reasonable accuracy. The most important deficiency is the under-prediction of the cladding temperature in the upper part of test section. The usage of the most recent TRACE 5.0 version patch 1 should lead to an improvement of the results [29].

Coupling between the best-estimate thermal-hydraulic code TRACE and the CFD code CFX

Computational fluid-dynamics (CFD) is gaining increasing relevance for nuclear applications, due to its capability of accurately treating multidimensional flows. The ability to reproduce three-dimensional phenomena is of special interests when dealing with asymmetric scenarios (e.g. boron dilution or Main Steam Line Break transients). Since CFD simulations require large computational resources, a coupling between CFD and best-estimate system codes is a worthwhile endeavor, especially for the simulation of transients where three-dimensional flows play an important role (e.g. mixing during boron dilution or Main Steam Line Break transients).

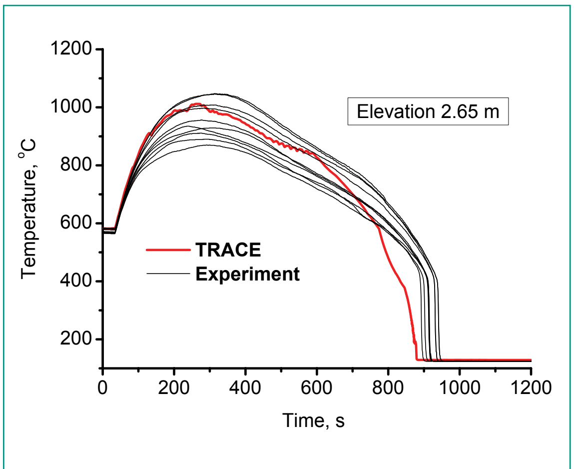

Fig. 13: Cladding temperature during reflooding (RUN A1R029).

The coupling between the commercial CFD code ANSYS-CFX and the best-estimate system code TRACE has been realized at PSI [30], [31]. In order to validate the coupled code and to gain better insight in the relevance of boundary conditions at the interfaces between the two codes, a simple experimental facility has been built at PSI, consisting of two loops connected by a double T-junction (Fig. 14, left). A constant flow-rate of 80l/min is set for each loop. Therefore, a stationary velocity field is established (Fig. 14, right: CFX result). Wire-mesh sensors [32], located in three sections of the experimental set-up, allow the measurement of the two-dimensional distribution of a tracer in the given cross-section. The transient starts with the injection of a certain amount of tracer in the side loop.

Simulations have been performed by means of TRACE, where the entire loop has been modeled with TRACE alone, and by means of the coupled CFX-TRACE code, where the double T-junction has been modeled in CFX and the side loop with TRACE. At the interface TRACE-to-CFX, the cross-section averaged velocity calculated by TRACE is transformed into a developed turbulent velocity profile, before input into CFX. The experimental cross-section averaged tracer concentration is compared with the stand-alone TRACE results and with the coupled CFX-TRACE simulations (Fig. 15). TRACE alone predicts a symmetric splitting of the tracer between the main and the side loop, in accordance with the mass flow ratio between the two loops (1:1 in the experiment under discussion). CFX, instead, correctly predicts a higher amount of tracer flowing back into the side loop. In Fig. 15 (right), the integral of the tracer concentration is shown for the three wire-mesh locations respectively. The integral increases at the same time for experiments, stand-alone TRACE, and coupled CFX/TRACE simulation, demonstrating that the coupling correctly reproduces the transport of the tracer plug in the system. For location WM1, where the assumption of a turbulent velocity profile is justified by the configuration of the facility, excellent agreement is obtained with the coupled tool. This indicates that CFX correctly predicts the amount of tracer mass which is recirculated in the side loop. The fact that the same level of agreement is not obtained for the locations WM2 and WM3 is due to the less accurate prediction of the velocity profiles at these two locations. Unfortunately, the experimental tracer integral concentration cannot be weighted with the velocity profile, as the latter is not measured

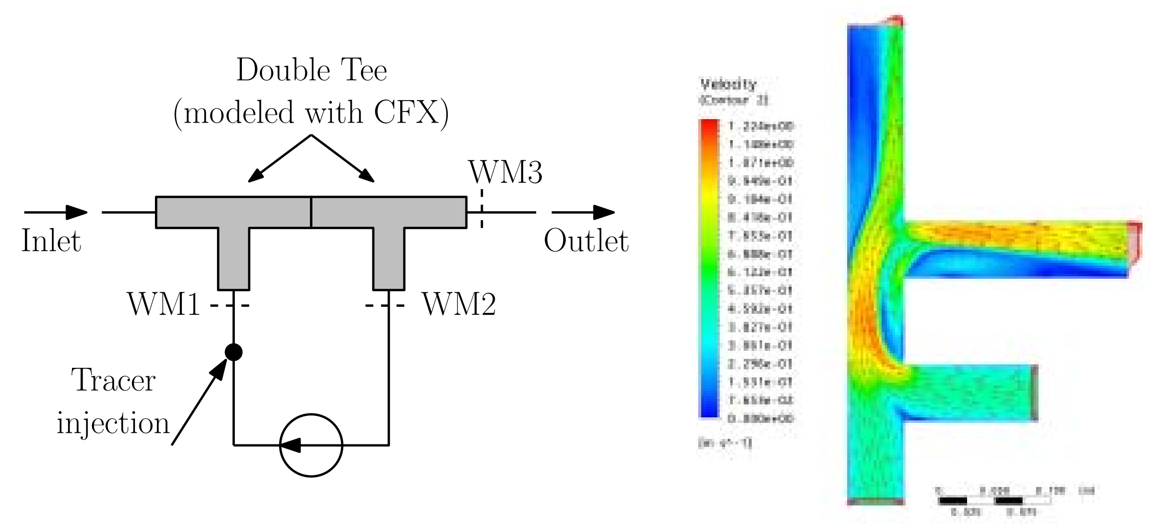

Fig. 14: Scheme of mixing set-up (left) and stationary velocity field (right).

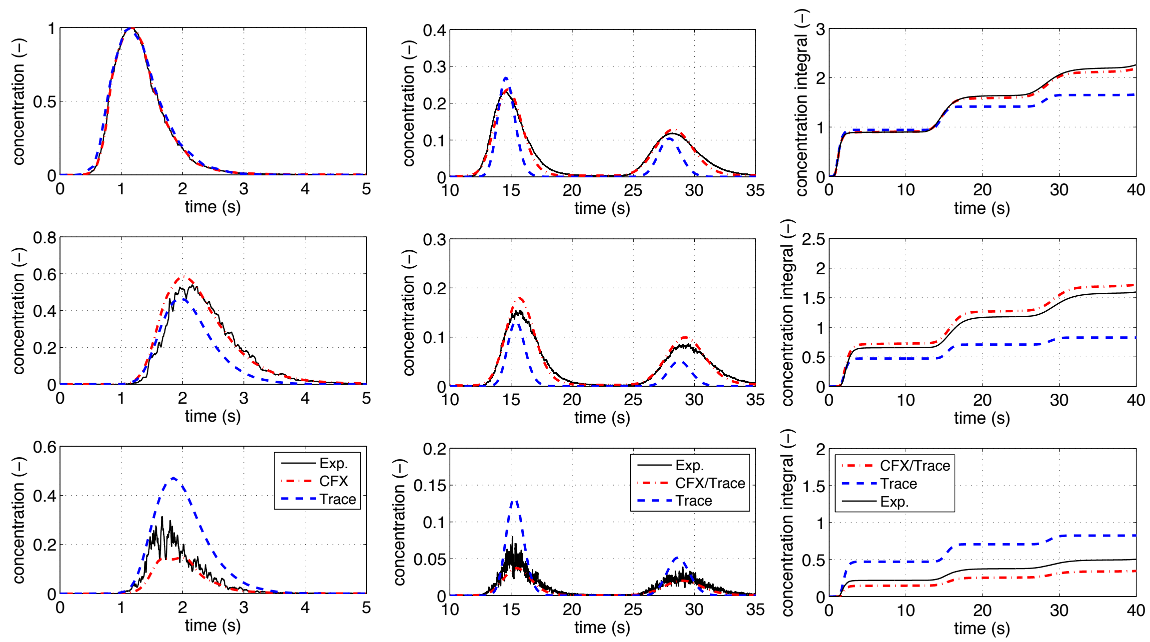

Fig. 15: Comparison between TRACE only and coupled TRACE-CFX simulation.

in the experiment. The fact that a correct prediction of the total mass of tracer recirculated in the side loop is obtained, but an incorrect cross-section average concentration results in locations WM2 and WM3 points out to a disagreement between experimental and calculated velocity profiles. This disagreement is larger for the WM3 location.

A second experimental set-up is currently under construction. The set-up is a scaled representation of a BWR lower plenum and corresponding internals. Experiments with a time-dependent velocity field are planned, in order to challenge the coupling of the momentum balance equation as well. In addition, it has been found that the tracer transport equation present in TRACE exhibits a high level of numerical diffusion. Therefore, efforts are currently being dedicated to the implementation in TRACE of a less diffusive numerical scheme for the tracer transport equation.

Core Behaviour

Steady-State Core Analysis of the Swiss Reactors

The steady-state analysis of the Swiss reactors is carried out within the code/model integrated CMSYS (PSI Core Management SYStem) platform which was presented at the PHYSOR conference [33]. With regards to the codes, CASMO-4 (C4) is employed for the 2-D transport calculations while the 3-D two-group nodal diffusion code SIMULATE-3 (S3) is used to perform the core follow analyses.

Concerning the Swiss reactors, CMSYS models for KKB1 Cycle 35 [34] and KKB2 Cycle 33 [35] were developed and qualified during 2008. In that context, deficiencies with regards to the methods employed for the comparisons between 3-D reaction rates and in-core detector flux traces were identified and resolved, yielding an improvement of the nodal RMS accuracy from 5–7 % to ~ 3 % for both reactors.

On the BWR side, the main activity during 2008 was to develop and qualify core models for KKM cycles 29 to 34. As the most recent CASMO-4E and JEF-2.2 libraries were used for that purpose, all CMSYS models for the previous cycles 19-28 were also updated. An overview of the achieved accuracy in terms of 3-D power distribution is presented in Fig. 16 where the RMS of the differences between calculations and TIP measurements are shown at the assembly, axial and nodal level.

The accuracy for the later cycles 29-34 is seen to be similar if not better than for previous cycles, reflecting thereby an overall adequate performance of SIMULATE-3 also for cores with an increased content of Partial Length

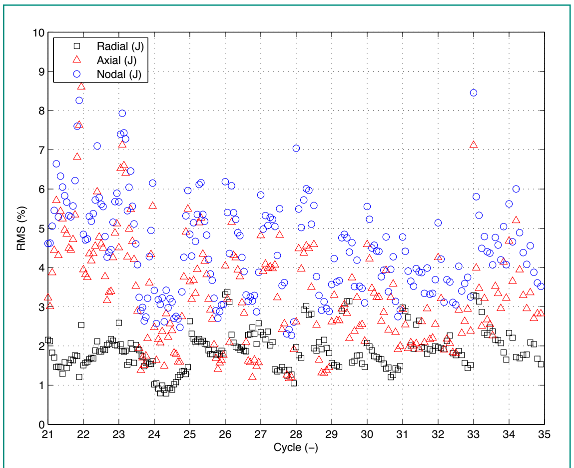

Fig. 16: CMSYS Modeling of KKM Cycles 21 – 34.

Rods (PLR) fuel assemblies. A nodal accuracy with RMS values around 4–5 % can moreover be considered as rather satisfactory for the analysis of heterogeneous BWR cores. Nevertheless, as the largest local deviations are systematically obtained just above the core inlet and just below the core outlet, a review of the axial reflector modeling appears necessary since this will affect the axial neutron leakage. Also, since larger-than-average deviations are usually observed in the lower part of the core, a review of the thermal-hydraulic models related to the axial void distributions, specially in the single-phase and subcooled boiling regions, appears as valuable, particularly when considering moreover that the risk for a violation of the thermal-limits becomes larger in that zone due to the bottom-peaked axial power distribution during a large part of the BWR cycle operation.

PWR Core transient analysis – Reactivity Initiated Accidents

The SIMULATE-3K code is gradually being established within STARS as a replacement of CORETRAN for 3-D kinetics. In that framework, the assessment of SIMULATE-3K (S3K) for PWR reactivity-initiated-transients (RIA) at hot-zero-power (HZP) was completed during 2008 on the basis of a previous study where the full-width-at-half-maximum transient power pulse width was compared between UO_2 and MOX cores using the CORETRAN code and based on real operated KKB1 cycles. These previous CORETRAN analyses were performed, among others, to provide boundary conditions for fuel behavior analyses that were carried out during the development of the revised Swiss RIA acceptance criteria for MOX cores.

From that point of view, an assessment of S3K on the basis of this study was considered as appropriate. The results of the study have recently been submitted for publication that is currently under peer-review [36]. But to summarize, the main findings are that qualitatively, a very similar pulse width behaviour is obtained between S3K and CORETRAN, both for UO_2 and MOX cores as well as when comparing the two types of cores at different times during the cycle operation. Quantitatively, the S3K pulse width is generally slightly larger than CORETRAN but this translates to nodal enthalpy deposition differences of ~5 cal/g and this can be considered as a very good agreement. Since S3K also provides the capability for explicit pin enthalpy calculations, investigations were carried out and showed that only when the pulse width is close or below 10 ms do strong intra-assembly power peaking effects start to occur, yielding thereby increasingly larger pin enthalpies compared to nodal values.

Finally, several sensitivity studies were also carried out using S3K in order to assess the effects related to 3-D kinetic modeling options/assumptions on the pulse width. Based on these studies, pulse width (PW) curves as function of control rod excess reactivity (ER) above prompt criticality were estimated for a representative MOX and UO_2 core respectively. These curves are presented in Fig. 17 and mainly show that for the types of cores similar to the analyzed KKB1 cycles, the MOX pulse width will usually be in the range 15–20 ms and around 5–10 ms smaller compared to UO_2 cores. For ER just below 500 pcm, which corresponds to control rod worth's around 2.0 $, the average MOX pulse width is ~15 ms. Only when assuming unrealistically large rod worth's (ER> 500 pcm) would the pulse width approach a 10 ms lower limit.

To conclude, noting that only CORETRAN had so far been assessed/benchmarked at PSI for RIA applications, the close agreement obtained with S3K for all the investigated KKB1 cores indicates that this code can be considered as equally adequate for the RIA analyses of the Swiss PWR plants. Moreover, the above results have confirmed that sufficiently representative CORETRAN 3-D calculations were provided as boundary conditions for the fuel performance code during the development of the revised MOX acceptance criteria.

BWR core transient analysis –cold zero power startup dynamics

At End-of-Cycle 20 (EOC20), the KKL reactor was shutdown for maintenance and re-started some 20 hours later. During start-up, the coolant heating rate exceeded the maximum allowed limit because of a too rapid heat insertion caused by the combination of a positive moderator temperature coefficient (MTC) and a non-active Residual Heat Removal system (RHR). This resulted in an rapid increase of the thermal power which was attempted to be stabilized through the insertion of control rod banks but eventually continued to increase before being finally reversed by the inherent feedback mechanism (void formation) after around 2300 s.

As part of the overall assessment of S3K for 3-D kinetic LWR applications, this transient was modeled and analyzed during 2008. An overview of the S3K results obtained so far is provided in Fig. 18 where one measured neutron flux signal (APRM) is used as basis to qualitatively assess the S3K performance with regards to the predicted transient core power when using a plenum-to-plenum core model with transient boundary conditions specified at the core inlet/outlet.

As the first attempt showed a rather poor performance (C1), the cross-section methodology with regards to the MTC modeling was studied and on that basis, a more recent CASMO-4 version with a finer cross-section parameterization at cold conditions as main difference was used to update all KKL core follow models from Cycle 1 to 20. The updated core models were found to yield a slightly more positive MTC but this had a sufficiently large impact to allow S3K capturing much better the initial transient behaviour phase up to around 1000 s (case C2 in Fig. 18). During the secondary phase between ~1000 s to ~1500 s when control rods were inserted by the operators, the transient behavior is also better captured compared to the calculation

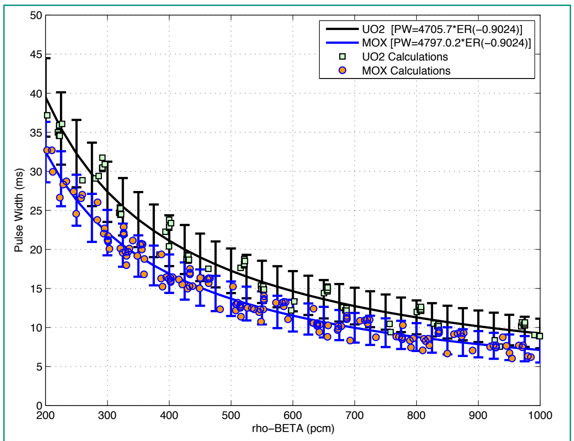

Fig. 17: PWR REA Pulse width Curves for UO_2 and MOX cores – CORETRAN/S3K Analyses.

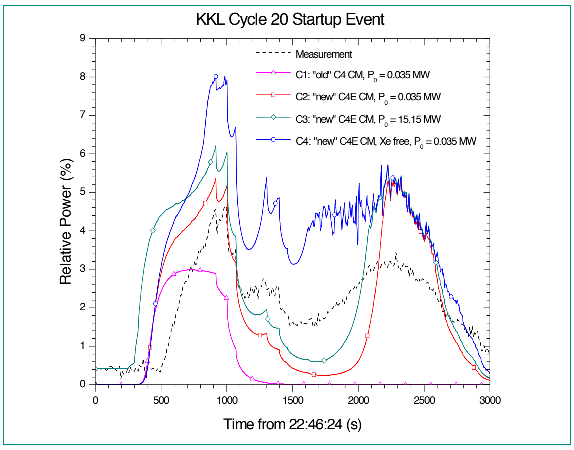

Fig. 18: SIMULATE-3K Analysis of the KKL EOC 20 Start-Up Event – Transient Reactor Power.

using the older XS methodology (C1) for which, a too low (positive) magnitude of the initial MTC combined with the CR insertion was sufficient to render the MTC negative and thereby cause S3K to predict a too early termination of the transient. Nevertheless, despite the improved MTC modeling, the S3K power remains substantially under-predicted during the second transient phase. Further studies have indicated that this is related to the challenges in capturing adequately the strong coupled neutronic/T-H effects which start to occur due to the appearance of void combined with the CR movements. This induces spectral effects on the MTC magnitude and this affects in turn, the remaining part of the transient during which, coolant heating continues. This is illustrated in when assuming a higher initial power (case C3) or when reducing the initial Xenon concentration (case 4). In both cases, a more rapid initial power excursion occurs, leading thereby to a stronger coolant density reduction (and eventually earlier void formation), affecting in turn through the MTC, the subsequent power reduction when CRs are inserted. Similarly, the power increase rate during the later transient phase, during which the coolant heating continues, is also affected. As a main next step, an S3K detector model is now being implemented in order to assess in more details the 3-D local fission power quantitative results based on all available measured APRM/LPRM neutron fluxes, noting that at such conditions, the uncertainty in the measured signals must also be carefully considered.

Coupled transient analysis – PWR Main Steam Line Break events

A study initiated during 2007 to analyze PWR Main Steam Line Break (MSLB) transients was continued during 2008. On the one hand, the effects on 3-D kinetics MSLB analyses at Hot-Zero-Power (HZP) related to the few-group homogenized cross-section (XS) modeling and carried out with CORETRAN/S3K on the basis of the KKB1 Cycle 26 UO_2/MOX, were continued [37] and were moreover extended to investigate the impact on 3-D power distributions both at HZP and at Hot-Full-Power (HFP) conditions [38]. On the other hand, a study of thermal-hydraulic (T/H) mixing related effects was continued, using a RETRAN-3D coupled core/system model of the two-loop KKB1 reactor based on the same 3-D neutronic Cycle 26 model as CORETRAN and with a 1-to-1 mapping between neutronics and T/H core channels. To study these effects, a very detailed nodalization of the core inlet region along with an associated parameterized radial inlet mixing model was implemented in the RETRAN-3D model. The mixing model, illustrated in the left part of Fig. 19, allows adjusting the local mixing efficiency (i.e. inlet temperature) in each individual Fuel Assembly (FA) through the use of fictitious valves assigned to each FA and whose opening areas («valve position coefficient», VPC) are parameterized to the radial distance of the FAs relative to the core mixing separation line (APL).

Based on this model, RETRAN-3D analyses were carried out for a large variety of radial mixing profiles ranging

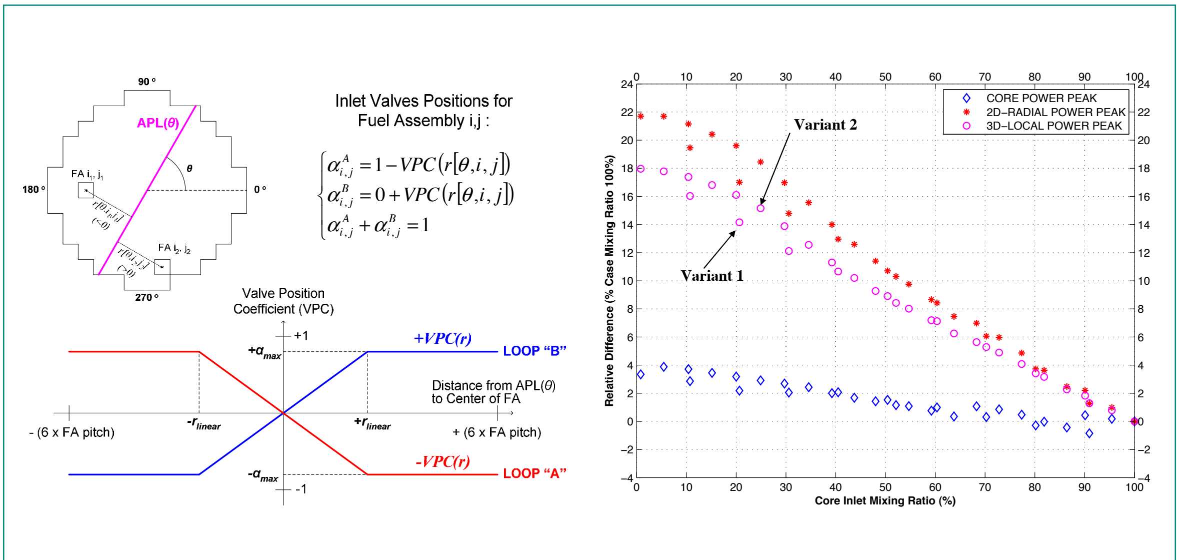

Fig. 19: RETRAN-3D Core Inlet Mixing model (left) and Effects on Transient Power at HFP (right).

from no mixing to full mixing assumptions as well as for different azimuthal positions of the separation line and taking also into account in this context, different radial positions of the assumed stuck control rod. An overview of the results obtained at HFP conditions [39] is presented in the right part of Fig. 19 where the impact on the core integral power maximum and on the 2-D/3-D transient power peaking factors is presented as function of the core inlet mixing ratio (CIMR), which characterizes the overall mixing efficiency between the coolant flows from the two cold legs [39]. In the first variant, a uniform distribution of the local mixing efficiency coefficients is assumed for all FAs belonging to a given core region. In the second variant, an inclined transition region in terms of mixing efficiency is assumed between the two core regions where again a uniform mixing efficiency is assumed. (The width of the transient region was also varied up to the length of 6 times the fuel assembly pitch [39].) It can be seen that the gradual lowering from perfect mixing (100 %) to no mixing (0 %) results in an increase of the local power peaking factors by up to 18 % (expressed relative to the perfect mixing case which was the least conservative case) while showing a much smaller impact on the total core power (4 %). In this context, it is noted that with Variant 1, an almost linear trend is obtained while for Variant 2, a more convex shaped curve is seen. To some extent, this shows that estimating a CIMR or any equivalent type of overall (scalar) mixing indicator between two (or more) loops in 3-D coupled analyses will probably not be sufficient in order to limit the uncertainties in predicted transient power peaking factors. And in turn, this emphasizes that deriving the coefficients for such core inlet mixing model, either from experimental results or from an advanced T-H solver (e.g. CFD), could be valuable for best-estimate MSLB analyses in the near-term while for the longer term, calculations based on a coupling between large plant system models and CFD pressure vessel model should be aimed at. However, at the same time, it must be emphasized that the impact not only on local power peaking but also on DNBR predictions should first also be studied.

Core Physics and Multi-Physics activities within the NURESIM project

The NURESIM integrated project, carried out within the 6th Framework Programme of the European Union, officially ended in December 2008 but was renewed for the next three years as the NURISP project. Similarly as for NURESIM, STARS will continue to participate in the sub-project «Core Physics» (SP1) and will continue to lead as well as participate to the «Multi-Physics» (SP3) sub-project.

Within SP1, STARS is mainly participating in the development and benchmarking of computational schemes for LWRs based on the NURESIM deterministic neutronic solvers APOLLO/CRONOS. During 2008, the computational scheme developed at PSI to perform APOLLO-2 transport calculations for pin-cell models of UO_2 and MOX fuel rods was continued to be assessed against CASMO-4 but this time, with regards to the evolution of isotopic compositions as function of burnup. The development and extension of the PSI APOLLO scheme to full 2-D lattice calculations was however not initiated but is planned as part of the PSI participation to the next project phase.

Concerning SP3, apart from coordinating the sub-project, STARS is mainly participating in the establishment

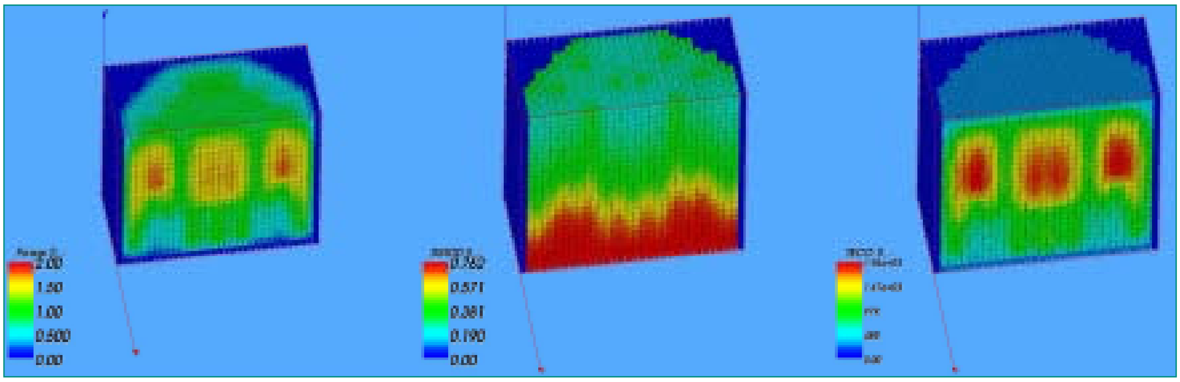

Fig. 20: BWR steady-state results using the coupled cronos2-flica4 solvers in SALOME: Power (left), Moderator Density (center) and Fuel Temperature (right).

within SALOME (software on which the NURESIM Multi-Physics simulation platform is based) of coupling schemes between the 3-D core neutronic- and thermal-hydraulic solvers CRONOS and FLICA for PWR and BWR transient analyses. In that framework, a BWR analysis was performed during 2008 [40] using the CRONOS-FLICA dynamic coupling scheme implemented in SALOME. The main objective was to verify for a BWR situation target, the operability of the different solvers and coupling tools that were developed and integrated in the NURESIM platform during the previous years, and in doing so to complement the set of situation targets that were in parallel developed by other international partners [41]. Thus, the exercise was to simulate the behaviour of a BWR core based on Exercise 2 of the OECD/NEA BWR Turbine Trip Benchmark. The thermal-hydraulic transient boundary conditions of the problem (core inlet pressures, enthalpies, velocities and core outlet pressures) represent a pressurization wave originating from the sudden closure of the turbine stop valve and that propagates throughout the system into the core, affecting thereby the void distribution and thus resulting in an insertion of reactivity due to the moderator density effect.

The steady-state results of the BWR analysis using the SALOME dynamic coupling of CRONOS and FLICA are shown in Fig. 20, which illustrates the strong coupling between power/fuel temperature distributions and the moderator density as water is being heated up. For the transient analysis, the integral reactor power was found to be in reasonable agreement with previous solutions provided by other international organizations in the context of the OECD/NEA benchmark although difficulties in achieving full convergence in the initial null-transient phase was observed.

HSK criticality safety analysis of KKG wet storage pool

The PSI methodology for Criticality Safety Evaluations (CSE) based on MCNPX with modern continuous energy cross section libraries was applied during 2008 for a first practical application [42]. This analysis was requested by the HSK and aimed at serving as an independent verification of the licensing criticality safety evaluations performed by AREVA for the new KKG external wet storage pool and for the new enriched reprocessed uranium (ERU) fuel assemblies (FA) containing a nominal ^{235}U enrichment above 5.0 wt-%. In comparison to standard fuel, ERU FAs contains a significantly higher fraction of ^{236}U, which acts as a neutron absorber early in-life but which after irradiation allows for a longer fuel cycle. The concept of ERU fuel design is therefore to increase the nominal initial ^{235}U enrichment and to add simultaneously, a certain amount of ^{236}U such that the effective initial ^{235}U enrichment remains below 5 wt-%. Within that framework, several configurations were analyzed at PSI: the complete storage pool, a single FA with an infinite water reflector, and two configurations corresponding to accident conditions (the positioning of a FA outside a storage rack and in a corner of the storage pool). The PSI results using MCNPX along with ENDF/B-VII.0 and JEFF-3.1 for the single ERU-FA with an infinite water reflector and without ^{236}U credit are shown in Fig. 21 along with the upper subcriticality limits (USL), derived from the PSI validation suite and depicted as horizontal lines.

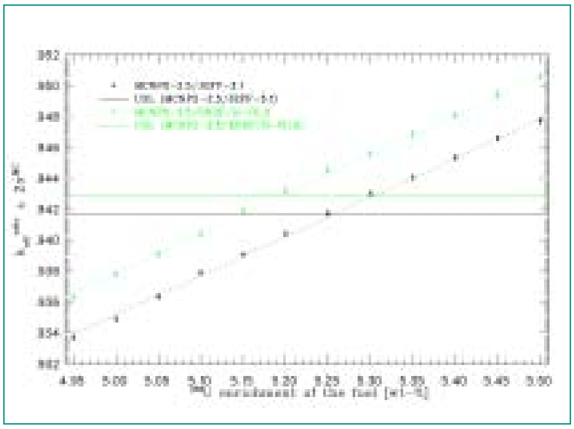

Fig. 21: MCNPX Results for a single ERU-FA in infinite water reflector configuration.

Using ENDF/B-VII.0, a maximum allowed nominal enrichment (e_{nom}^{max}) around 5.17 wt-% is obtained with the PSI approach and this turns out to be identical to the AREVA result obtained with a different methodology including a different code/library system. Similarly, a very close agreement was obtained when taking credit of the ^{236}U content of the fuel as illustrated by Table 3.

Finally, a very close agreement was also obtained when analyzing all other configurations including the nominal entire compact storage pool. And in that context, both methodologies moreover confirmed in a consistent manner that the most stringent configuration is the single FA with an infinite water reflector. Considering the very good agreement for that case mentioned above, the PSI analyses thus confirmed the AREVA licensing results. Nevertheless, it must be noted from Fig. 21 that for the single FA case, the differences obtained by PSI in terms of maximum enrichment when applying two different

Nominal enrichment in ^{235}U [wt-%]	5.17	5.25	5.35	5.45
AREVA result for minimum ^{236}U required [wt-%]	0.0	0.15	0.42	0.76
MCNPX/ENDF/B-VII.0 result for minimum ^{236}U required [wt-%]	0.0	0.14	0.40	0.80

Table 3: Comparison of AREVA and PSI results for minimum ^{236}U content required.

libraries are larger than those that were was expected when considering the USL differences from the validation suite. More precisely, while ENDF/B-VII.0 yields a maximum e_{nom}^{max} of ~5.17 wt-%, JEFF-3.1 shows a larger value of around 5.23 wt-%. This was therefore further studied as part of the continuous development/consolidation of the PSI CSE methodology.

Assessment 2008 and Perspectives for 2009

Fuel Behaviour

Work towards further developing the fission gas behavior model has progressed well during the last reporting period, and a set of KKL fuel rods have been analyzed with good results. As a consequence, a (yet) small database of base irradiations for fuel rods has been established. The description of fuel swelling up to high-burnup has also greatly benefited from the improved model as well as it enabled the successful participation in the SCIP project with modeling of the ramp tests.

The application of the improved fission gas behaviour model to a high-temperature RIA scenario as it can be expected after the introduction of the HT-capsule in NSSR indicated that the behavior of the fission gas captured especially in the bubbles of the High-Burnup Structure (HBS) might be of considerable interest. Meanwhile, a first high-temperature experiment using KKL fuel has been conducted in NSSR, but relevant experimental data are not yet available. Similarly, due to the high temperatures prevailing during a LOCA, the behavior of the fission gas in the bubbles of the HBS might also play an important role.

The participation in the LOCA-benchmark exercise organized by CSNI/WGFS did not materialize due to shortage of resources. However, both experiments have already been analyzed at PSI previously. Work towards establishing a statistical framework for fuel analysis could not be launched again due to shortage of resources. This was caused partly by the necessity to train a young staff, but also due to the fact that more effort was required for the work on further developing the fission gas behaviour model and the participation in the SCIP project with extensive modeling work.

Systems Behaviour

Significant progress has been made with the migration of existing plants models for legacy codes (RELAP/TRAC-BF1/RETRAN) to TRACE and with the assessment of TRACE capabilities for the modeling of BWR transients. The latter point is of special importance, since most of the TRACE validation cases pursued to date have been mainly focused on PWR applications. Scoping simulations of LBLOCA and loss-of-feed with the KKM model, and the successful simulation of the SEHR-ADS event occurred in KKL (March 2007) support the idea that the TRACE code has now reached a considerable level of maturity for LWRs applications.

In view of the future needs for expertise on Gen-III and Gen-III+ reactors, efforts have been dedicated to the development of TRACE and CFD models of the EPR. In 2009 these models will be employed for the simulations of SBLOCAs, LBLOCAs and MSLBs. The results of the TRACE simulations will be used as boundary conditions for the CFD model, in order to evaluate thermal and boron mixing in the EPR reactor pressure vessel, during boron dilution and MSLB transients. In addition, in collaboration with the Laboratory for Thermal-hydraulics (LTH), a study on the performances of the emergency condenser of the SWR1000 has been carried out. RELAP (LRS) and GOTHIC (LTH) simulations have been performed. The results will be compared with experimental data that will be recorded by AREVA at a large-scale facility.

CFD is gaining more and more importance for nuclear safety applications, given its capabilities of capturing three-dimensional phenomena. Therefore, a CFD expertise has been built within the STARS project. Beside the CFD EPR model, a model of the KKL lower plenum has been generated as support to specific investigations of the SEHR-ADS event. The coupling between a CFD code (CFX) and TRACE has been further pursued. Experiments on a double T-junction loop built in LTH have been used as validation cases, showing the superiority of the coupled tool CFX-TRACE in comparison to a stand-alone TRACE model. In addition, the importance of the assumptions on the boundary conditions on the interface between 1D (TRACE) and 3D code (CFX) have been investigated. The obtained results have found a rather good international resonance.

The participation in the international programme CSNI/BEMUSE, aimed at the assessment of un-certainty methodologies for best-estimate thermal-hydraulic codes, has been continued. In PSI, TRACE, combined with a probabilistic approach based on the GRS methodology has been employed for the uncertainty and sensitivity analysis of a LBLOCA in a Westinghouse 4-loop PWR. The results obtained at PSI are well enclosed within the band obtained by averaging among all participants (for obvious reasons, no plant data were available as reference). Beside the BEMUSE programme, the efforts toward the development of a state-of-the-art computational framework have been address by initiating a research program aimed at the combination of best estimate plus uncertainty (BEPU) methods and dynamic event tree (DET), and to the coupling of the plant response, hardware reliability, and failure model, together with a human response model.

Core Physics

On the core modeling side, significant progress was made with regards to the KKM models which were updated up to cycle 34 and as part of this, were improved with regards to the nodal power accuracy. On that basis, several processors were developed and implemented in CMSYS to facilitate the periodic KKM model update for future cycles. Also, procedures for the periodic update of KKB1/KKB2 models were finalized, including improved methods for a comprehensive comparison of the calculation results against plant measurements. The plan is to establish similar procedures for the updates of the KKL and KKG models, the intended benefit being to allow through such CMSYS processors and interface routines, to ensure efficient periodic model updates and to release thereby, more time for detailed investigations necessary to continuously improve the methodologies and the knowledge related to 3-D core analyses. Preliminary steps towards an upgrade to the successor codes CASMO-5/SIMULATE-4 were also undertaken during the year. However, as many of the method improvements in CASMO-5 are tightly linked to SIMULATE-4 and since this later code remains under testing by the code developers, a migration to CASMO-5 was not considered to be of immediate necessity and hence not started.

With regards to core dynamics, the assessment of S3K for 3-D core kinetics was pursued, both for PWR reactivity-initiated accidents (RIA) and main steam line break transients (MSLB) and for BWR nuclear heating transients on the basis of a KKL event. For the PWR analyses, the applicability of S3K was confirmed and this constitutes therefore a confirmation that S3K can now replace CORETRAN as main kinetic solver for these types of transients. For the BWR cold zero-power transient, the performed studies have shown that S3K is capable of capturing the overall transient behaviour but have also illustrated that complex core dynamic phenomenon with strong neutronic/T-H interactions occur during this event. Based on this, areas of improvements were identified although verification against local 3-D detector signals remains to be completed for a final assessment, noting that this will require accounting for the large uncertainties in measured APRM/LPRM detectors that are expected at such low power conditions.

Concerning the coupling of S3K with the TRACE system code, a review of the different coupling strategies was carried out during the year and on that basis, an internal coupling scheme was selected as preferred option in order to allow taking advantage of the more advanced thermal-hydraulic (T/H) modeling capabilities in TRACE (e.g. 6-equation model) as well as to ensure a consistent T/H solution scheme between core and system. On that basis, a collaboration frame-work, where PSI would handle the TRACE side of the coupling scheme while the S3K side would be handled by the code developers, is currently being discussed.

Since it is important that a capability to perform coupled 3-D core/plant system analyses remains available in STARS before an S3K/TRACE coupling scheme can be implemented, work related to the analysis of PWR MSLB transients with a RETRAN-3D fully coupled 3-D core/plant system model was pursued. A detailed core inlet mixing model was implemented and allowed to assess in the mixing effects on the predicted MSLB behaviour. This RETRAN-3D analysis could moreover certainly serve as basis for the future assessment of S3K/TRACE for PWR MSLB transients, noting that a strength of the RETRAN-3D analysis was that it could be carried out without any grouping of the core T/H channels, something that remains common practice in the area of coupled 3-D core/plant system analysis based on large plant models. With regards to coupled analyses, it should also be mentioned that the analysis of the BWR situation target within the NURESIM platform was successfully performed and completed according to the plan. With regards to the core physics activities of NURESIM, an extension of the PSI APOLLO-2 computational scheme for full 2-D lattice applications was not initiated mainly since significant efforts are expected and this had to be balanced relative to other project priorities.

PSI continues to participate as an observer in the first exercise of NSC/UAM benchmark (neutronic uncertainty in view of coupled analysis). However, in the perspective of an active participation, efforts were undertaken towards developing a method based on statistical sampling for the propagation of cross-section uncertainties. Within that framework, discussions were undertaken with both a) GRS in order to initiate collaboration for the development of such sampling method and b) with the CASMO-4 code developers to assess the feasibility to perturb the multi-group cross-section libraries employed by the lattice code. It must be mentioned in this context that the code developers were at this stage not willing to guarantee any support in this area due to the complexity of the task, confirming thereby to some extent that the development of such neutronic uncertainty propagation methodology is a very challenging task and should therefore be approached in a gradual step-wise manner. The planned collaboration with GRS is therefore precisely aimed as a first step towards that objective.

Technical services were also performed and completed both upon request from the HSK and from the utilities. For the HSK, a criticality safety evaluation using the PSI MCNPX-based methodology was performed to analyze enriched reprocessed uranium fuel for the new external wet storage pools at KKG. The PSI calculations were found to be in very close agreement with the AREVA analyses and on that basis, the new wet storage pool as well as the introduction of ERU fuel was finally licensed. Moreover, as follow up to this application, studies were carried out to address the applicability for criticality safety of the data sets for thermal neutron scattering in water employed in different neutron data libraries. On that basis, a scientific publication is currently being prepared. This example illustrates that technical support activities can also be very valuable in order to gain new insights of scientific relevance and/or to develop novel methodologies and thus underlines the strength of the close collaboration of STARS with the national authorities and/or utilities.

Perspectives 09

The projected work for 2009 develops in three main domains:

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

Systems Behaviour

- Complete the migration of existing plant models for legacy codes (RELAP5/TRAC-BF1)
- 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, initiate the development of advanced 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 Behaviour Modeling

- Update Swiss core models (CMSYS)
- 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 NURSIP
- Establishment of a method for statistical sampling of cross-section data in neutronics calculations

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