PROJECT FINAL REPORT

Name of the scientific representative of the project's co-ordinator: Bruno Chanaron (Commissariat à l'Energie Atomique et aux Energies Alternatives) bruno.chanaron@cea.fr

Other authors:

- **Carol Ahnert (Universita politecnica de Madrid) carolina.ahnert@upm.es**
- **Dominique bestion (Commissariat à l'Energie Atomique et aux Energies Alternatives) dominique.bestion@cea.fr**
- **Martin Zimmermann (Paul Scherrer Institut) martin.zimmermann@psi.ch**
- **Dan Cacuci (Karlsruhe Intitute Für technologie) dan.cacuci@kit.edu**
- **Nicolas Crouzet (Commissariat à l'Energie Atomique et aux Energies Alternatives) nicolas.crouzet@cea.fr**

Project website address: http://nurisp.eu

1. Executive summary

The target of the NURISP Collaborative Project is to make new and significant steps towards the NURESIM European Reference Simulation Platform. NURESIM is a set of state of the art software devoted to the simulation of normal operation and design basis accidents of light water reactors: BWR, PWR, VVER. This platform currently includes 11 codes (figure 1) covering different physics: neutronics, thermal-hydraulics, fuel thermo-mechanics and relevant scales: local (sub-channel or pin), fuel assembly, core and reactor system. Given their complementary features, the selected codes offer solutions suitable for various situations.
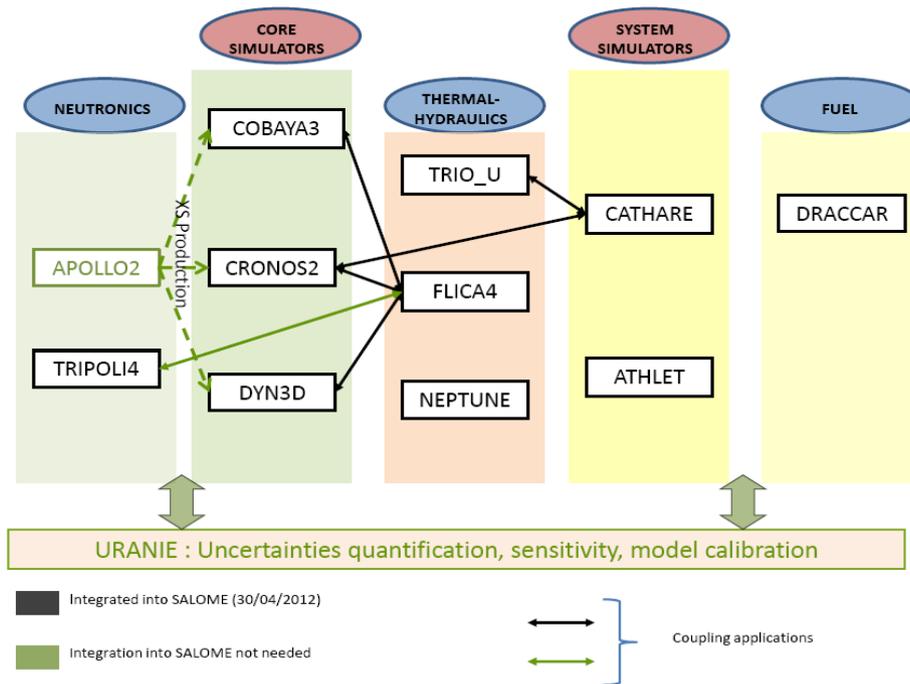

Figure 1
The NURESIM platform at the end of NURISP project

The codes have been extensively benchmarked and validated against experiments during the course of NURISP European collaborative project (2009-2012). A further extension of NURESIM is planned in order to extend its fuel thermo-mechanics capacity, currently limited to LOCA, to other transients and normal operation.

Coupling applications can run simultaneously two codes while exchanging data between them for a more accurate simulation of transients. Up to now, 5 coupling applications have been developed, in particular those involving a sub-channel thermal-hydraulics code and a core simulators. Development of more coupling applications is planned during the three next years.

A **SALOME** platform: The codes use generic features provided by the SALOME open-source software for pre-processing, post-processing of codes and use of the coupling applications, except those of codes or coupling interfaces which main purpose is not full core or reactor system simulation.

NURESIM includes a comprehensive capability for **uncertainty** quantification, sensitivity analysis and model calibration provided by the URANIE software. URANIE is designed so it can analyze data provided by all NURESIM codes.

2. Context and objectives

The target of the NURISP Collaborative Project is to make new and significant steps towards the NURESIM European Reference Simulation Platform for applications relevant to present and future PWR and BWR, especially to reactor safety. The roadmap of this Simulation Platform is a part of the Strategic Research Agenda of the Sustainable Nuclear Energy Technology Platform (SNE-TP). The first step towards this ambitious target has been made during the FP6 NURESIM Integrated Project. The NURISP project started from this basis and develop further the already common and well-proven NURESIM informatics platform based on the SALOME open source tool (www.salome-platform.org). It also strengthened and enlarged the united team of top level international experts already federated during the NURESIM project.

The platform must provide a more accurate representation of the physical phenomena by developing and incorporating into "best estimate" codes, better validated and qualified, faster models in core physics, two-phase thermal-hydraulics and fuel codes. The project also intends to develop significant capacities for multiscale and multiphysics calculations, and for deterministic and statistical sensitivity and uncertainty analysis, facilitating their use in a generic environment. It should permit more precise computations, with detailed meshing.

The platform must also provide generic pre-processing and post-processing and supervision functions which will make the codes more user-friendly.

A European platform

The NURESIM-Platform must integrate the products and the knowledge of the European organizations and ease collaborative work between them thanks to standards and a common informatics environment for testing and comparing different codes. For this purpose, it is essential to permit connection of the codes in a standardized way. The standards are being progressively built concurrently with the process of developing the platform.

On this basis, European collaborative work becomes easier and more efficient. It will enable European organizations to better share knowledge, development and validation and to decrease the costs, while improving code usability, quality and best practices.

The roadmap of the NURESIM-platform

The development of the NURESIM European Reference Simulation Platform for Nuclear Reactors comprises three steps:

1/ *The FP6 NURESIM project* provided the initial step towards the European Simulation Platform and demonstrated the potential of the proposal_a first integration of codes into the NURESIM platform and experienced feedback for the definition of integration standards; a first demonstration of a generic method for multiphysics coupling showing the potential of the method, the ease of use and its flexibility and functionality

2/ The FP7 NURISP project (2009 – 2012) aims at consolidating and extending the results of FP6 NURESIM in continuity with it. New codes have to be connected to the platform and new steps are made for integration, model development (including fuel), coupling, sensitivity and uncertainty analysis, and validation, with broader applications. Validation of the codes versus previous experiments or versus experiments funded elsewhere (for instance those of the NEA or IAEA programs) are an important part of the program.

3/ the FP7 NURESAFE project (2013-2015) By developing, validating and delivering to end-users fully integrated practical applications relevant to reactor safety analysis, operation and design, NURESAFE should achieve the initial objectives of the NURESIM roadmap.

Subprojects

The objectives of NURISP were realised through five scientific Sub-Projects:

- RTD Sub-Project 1 (SP1): Core Physics (Coordinator: UPM);
- RTD Sub-Project 2 (SP2): Thermal-Hydraulics (Coordinator: CEA);
- RTD Sub-Project 3 (SP3): Multi-Physics (Coordinator: PSI);
- RTD Sub-Project 4 (SP4): Model Validation & Calibration, Sensitivity and Uncertainty Quantification (Coordinator: UniKa);
- RTD Sub-Project 5 (SP5): Integration (Coordinator: CEA).

Objectives of SP1 (Core physics)

The overall objective of the Core Physics subproject is the development, verification and demonstration of advanced Core Physics safety-relevant numerical simulation codes and tools for LWR. To achieve this goal, it is necessary to build and validate the NURESIM-platform with capacity to simulate neutron kinetics at the needed resolution in the neutron energy, spatial and time scales and to provide consistent coupling with thermal-hydraulics and fuel thermo-mechanics.

The specific objectives of the Core Physics subproject are:

- To achieve the final stages of development of a mature system of neutron kinetics codes for safety-relevant numerical simulation. This goal has to be reached, not only by adding the needed additional functionalities, efficient and accurate performance to the NURESIM Core Physics platform, but also by increasing its usability and consistency, by well-established calculational routes that are consistent in terms of data (nuclear cross-sections and other data, geometry and materials definitions …), modelling options (mesh, solution order, …) and code integration. One of the relevant beyond the state-of-the art objectives is to develop pin-by-pin neutronics analysis needed for improved understanding of safety margins.
- To achieve the verification/demonstration process of the platform by furnishing the tools in order to compute realistic power plant configurations and/or experimental facilities, in coordination with the subprojects on Thermal Hydraulics, Multi-Physics coupling, Validation and Sensitivity and Uncertainty analysis, and Integration, SP2 to SP5, respectively.

Objectives of SP2 (Thermal-Hydraulics)

The overall objective of the thermalhydraulics subproject (SP2) is the development and qualification of advanced thermalhydraulics numerical simulation tools for LWR reactor safety in order to contribute to a European pole of excellence in reactor safety computations. To achieve this goal, it is necessary to build and validate a common standardized European software platform able to simulate flow phenomena at four spatial scales and to provide easy coupling with neutronics and fuel thermo-mechanics.

Whereas NURESIM addressed mainly the meso-scale (CFD in open medium) with some applications of the micro-scale (Direct Numerical Simulation), one objective is to include a new generation of standardized modules for component scale and system scale to build the next generation of experimentally validated "best estimate" tools for modelling present and future LWRs. Coupling between the four spatial scales will be further developed in order to perform more precise computations, with a local zoom when and where necessary.

Multi-scale analysis and multi-scale coupling of thermalhydraulics tools and coupling of thermalhydraulics with core physics and fuel thermo-mechanics will be used to investigate safety issues such as PTS (Pressurized Thermal Shock), CHF (Critical Heat Flux), LOCAs (Loss of Coolant Accidents) and steam condensation. PTS will be investigated through coupling of system and CFD codes and validated with the new data of the TOPFLOW and ROSA (OECD-NEA test program) test facilities. LOCA transients including not only LBLOCA but also any kind of SBLOCA will be investigated with advanced system scale modelling and coupling with advanced fuel thermo-mechanics to better simulate transients with ballooned fuel and pellet relocation. Based on the results of FP6 actions, two-phase CFD tools will be further developed, improved, and validated on new available experimental data including data from NEA data base, in a new step towards their

industrial application. Thanks to the involvement of several partners in OECD-NEA Committees and Working Groups, strong interactions with these international activities will be effective.

Objectives of SP3 (Multi-Physics)

A first objective of the Multi Physics subproject SP3 of the NURISP CP project is to develop and demonstrate state-of-the-art simulation capabilities for improved understanding of safety margins, which requires that multi-physics simulations be extended to the pin-by-pin scale of the fuel pins and subchannels, as stated by the scientific reviewers of the NURESIM FP6 Integrated Project.

The second objective of the Multi-physics element of the NURISP project is to fully integrate a range of thermalhydraulics, core physics and fuel thermo-mechanics codes and solvers within the SALOME platform to form an integrated European platform with the aim of providing a state-of-the-art code system to support safety analysis of current and evolving LWRs. The integration will include common post-processing and common data structure based on the "MED" data exchange model, which will permit the efficient exchange of information between all integrated codes and the development of generic multi-physics functionalities.

Starting from the coupling schemes developed during NURESIM, new features are systematically investigated and new coupling schemes will be developed: Emphasis is on integrating cross-flow for detailed (hot-channel) PWR core-analysis, a systematic evaluation of the coupling schemes in the time-domain emphasizing rapid transients and finally the coupling between a systems thermal-hydraulic code and a CFD code. An important additional result of this effort will be the availability of solutions of different platform solvers/codes for the selected benchmark problems (situation targets), further demonstrating the added value the NURESIM platform will bring to the analyst by offering alternatives.

The integration of a fuel behavior code and its coupling to thermal-hydraulic codes completes the necessary code tool-box for LWR safety analysis. It is expected that it will likewise cover a wide range of the needs for the safety analysis of advanced reactor systems, even though no type-specific testing is foreseen in the current project due to the limited resources available.

Furthermore, in order to reduce the required computational effort, first steps towards adaptive modeling, switching from higher-order to lower-order models based on the evolution of the transient, will be pursued.

Objectives of SP4 (Model Validation & Calibration, Sensitivity and Uncertainty Quantification)

The Sub-Project SP4 aims at providing essential tools for developing predictive experimentally validated "best-estimate" tools, within the NURESIM simulation platform. This would be particularly important for designing new technologies and facilities based on novel processes, while striving to avoid, as much as possible, the costly and lengthy procedures of building representative mock-up experiments which might confirm—but would not necessarily explain— the predictions of simulation tools.

The tools developed within SP4 will help to identify uncertainties and shortcomings for the neutronics and thermal-hydraulics codes which are part of the NURESIM platform. Such uncertainties arise from imprecisely known parameters, modeling errors, boundary and/or initial conditions.

Loosely speaking, "*code verification*" means "are you solving the mathematical model correctly?"; "*code validation*" means "does the model represent reality?"; and "*code qualification*" means certifying that a proposed simulation/design methodology/system satisfies all performance and safety specifications. Validation and qualification can be done only by selected benchmarking,

taking into account systematically (i.e., using sensitivities) all the uncertainties (computational, experimental, etc). Verification and predictive validation must be based on a well-established set of scientific approaches that will allow the a priori announcement, with strong confidence, of computational uncertainties, both for cases which are well known (for example, where strong experimental evidence exists in the form of confirmatory mock-ups) and for cases which are less well known (for example, for novel design where only elementary experimental evidence is available). The framework for this predictive verification and validation should enable a graded, science-based approach, with applications ranging from scoping-type design activities to generating information for the regulatory bodies, where the ability to predict the reactor's behaviour accurately and convincingly is particularly important.

Both selected legacy codes and new codes will require verification and validation relying on sensitivity analysis and uncertainty quantification. Sensitivity and uncertainty analyses can be considered as formal methods for evaluating data and models because they are associated with the computation of specific quantitative measures that allow, in particular, assessment of variability in output variables and importance of input variables.

The specific objectives of SP4 include:
- development of (adjoint-functions-based) deterministic and statistical model/modules, including software modules based on stochastic finite element methods ("polynomial chaos expansion"), within the novel software-platform URANIE,
- development and implementation into URANIE of adjoint-functions-based model/modules for global sensitivity and uncertainty analysis;
- development and implementation into URANIE of software modules based on "RaFu"-method for combining stochastic and epistemic uncertainties;
- development of new hybrid methods and software modules by combining adjoint and statistical sensitivity and uncertainty analysis methods;
- development of methods and software modules for data assimilation and best-estimate adjustment of simulation modules/codes;

Objectives of SP5 (Integration)
The overall objective of the integration sub-project (SP) is to build with the other sub-projects a qualitatively improved European platform (NURESIM), which will integrate in a simulation framework (SALOME) thermal-hydraulics, neutronics and fuel thermo-mechanics codes and solvers. Integrated codes and SALOME simulation tools will work concurrently within the applications specified and developed by the other sub-projects.

The integration team will assist and advise the partners in integrating their modules and codes. It will provide specific training to enhance autonomy, efficiency and advanced usage of SALOME platform features. It will also participate to specification and development of integrated applications, and adapt the SALOME platform to meet the requirements of these applications.

A last objective is to ensure the production and the maintenance of the NURESIM platform, its quality and non-regression, and also to promote collaborative development by providing partners with tools like CVS repositories and bug-tracker.

3. Main S&T results/foregrounds

3.1 Core physics

Advanced Monte Carlo Core Physics development

Sampling of the fission source distribution using power distribution obtained from an (approximate) deterministic calculation to obtain fast convergence of the fission source distribution in the Monte Carlo calculation.

CEA has implemented in TRIPOLI-4 an option allowing to depict the fission source in a meshing independent of the geometry volume description in TRIPOLI-4 code. The corresponding work done is summarized thereafter.

For Monte Carlo criticality calculations, the user needs to specify some initial source distribution, in space, energy and angle. This distribution is actually only used in the first cycle of the calculation, that is in the first step of the power iteration, but its choice can influence the convergence of the power iteration to the steady-state solution, that is the number of cycles that need to be discarded before starting the collection of the tallies.

The approach has been to use seamlessly an APOLLO2 calculation which gives multi-group homogenized cross sections to CRONOS2, in order to produce a deterministic solution to feed to TRIPOLI-4 as the used defined initial guess. This (almost) automatic chain has been applied to the Hoogenboom-Martin benchmark, to test the new developments in TRIPOLI-4 and CRONOS2 and to quantify the possible gain in a real-size problem.

It has been shown that this development allows the saving of almost the entire convergence phase of the usual power iteration, which corresponds to about one hundred cycles for a large scale PWR reactor. This saving is most valuable for parallel calculations, where the convergence needs to be repeated on each independent simulator. The use of a single tabulated source over the entire grid allows keeping the source initialization time to a negligible level, even for a heterogeneous source with millions of values.

Implementation and testing of an interface between TRIPOLI-4 Monte-Carlo code and a thermalhydraulics code to transfer the local power to the thermal-hydraulics model and to feedback the calculated temperature distribution for the fuel rod and coolant into the neutronics calculation.

The Delft University of Technology (TUD) and the Karlsruhe Institute of Technology (KIT) developed coupling schemes between a Monte Carlo (MC) calculation of the neutronics of a system with a thermal-hydraulics (TH) calculation of that system.

This development and its first applications led the partners to propose four types of future improvements for the TRIPOLI-4 Monte Carlo code concerning the temperature cross section treatment (calculation of the cross section at a desired temperature), the convergence process (save fission source distribution for next iteration, use of additional convergence criteria, ...) the physical modelling (heat deposition, ...) and the time calculation (parallel processing, ...) respectively.

The demonstration of the feedback coupling between a thermal-hydraulics code and a Monte Carlo transport code has been done by TUD and KIT. They have each developed a coupling system allowing to carry out cross comparison studies which involves both deterministic and stochastic solvers. The codes considered are FLICA and SUBCHANFLOW for the thermal-hydraulics (TH), TRIPOLI-4, MCNP-5 or KENO for the neutron transport (NTR) using Monte Carlo method. Among

the main physical quantities of interest to calculate there are the fuel temperature, the coolant density, coolant temperature and the fuel power distributions.

Because this type of coupling calculations have been performed currently with deterministic neutron transport codes, it seemed useful to define a numerical benchmark to compare the constraints and results related to these two coupling approaches. APOLLO-2 lattice code and FLICA-4 have been chosen to carry out such calculations.

A BWR UO$_2$ fuel pin standard configuration has been specified by CEA as a numerical benchmark definition (3.8 m for active zone and 0.2 m for bottom and top reflector) introducing axial dependence of enrichment, moderator density and temperature axial profiles. CEA provided also results of fuel temperature, coolant density and fuel power distributions obtained from APOLLO-2 - FLICA-4 coupling in order to compare with 3D TH/NTR Monte Carlo couplings.

KIT and TUD provided solutions for the benchmark problem for five main coupling combinations: FLICA-4 / MNCP-5, FLICA-4 /TRIPOLI-4, SUBCHANFLOW / MCNP-5, SUBCHANFLOW / TRIPOLI-4, FLICA with MNCP-5, FLICA / TRIPOLI-4, SUBCHANFLOW / KENO, SUBCHANFLOW / MCNP.

It is clear that the comparison between the deterministic TH / deterministic NTR coupling results and the deterministic TH / Monte Carlo NTR coupling results has a limitation due to deterministic neutron transport hypotheses and approximations which are ignored by the 3D approach.

Despite some differences, the results obtained by the different calculations coupling schemes show a very similar shape for fuel temperature, coolant density, fuel power axial distributions.

One can conclude that the schemes developed for coupling a thermal-hydraulics code with a Monte Carlo code for the neutronics are appropriate. The set of these results give confidence to deal with more complex calculations. They allow to highlight several key aspects which get attention:

- the performing treatment of cross-sections (accuracy and processing speed up) at the right temperature, especially in the thermalization range (TUD and KIT have investigated this point),
- the iteration process using the neutron source distribution from the previous iteration (this functionality is implemented in a new development version of TRIPOLI-4 code) in order to reach faster the source stationarity,
- the acceleration of the convergence process: it has been investigated by KIT using the SUBCHANFLOW-KENO coupling to accelerate the SUBCHANFLOW-MCNP coupling.
- the need for good convergence of the source distribution before the new thermal-hydraulics calculation,
- the need of convergence criteria,
- the need to give an estimation of the standard deviation associated to each physical quantities. This point raises the uncertainty propagation problem in the frame of the coupling problems.

This kind of study contributes to open the way towards a future reference 3D TH/NTR coupling calculation scheme using neutron Monte Carlo transport. This aim is in agreement with the need to improve the predictive capability of the nuclear reactor physics tools.

Development of the basic Monte Carlo techniques for long-time kinetic and dynamic calculations.

The aim of this task was to investigate the feasibility of kinetic and dynamic Monte Carlo simulations. It is recalled that the difficulty of doing a dynamic Monte Carlo simulation is the

different orders of magnitude in time scale that the various processes in a nuclear reactor have. Prompt neutrons operate at micro to milliseconds time scales, but precursors operate at time scales from seconds to minutes.

The feasibility of the long-time kinetic Monte Carlo has been demonstrated and advanced variance reduction methods developed can increase *Figure of Merit* with a factor of 10.

The dynamic scheme developed has been implemented into the general purpose Monte Carlo code TRIPOLI-4. A new option and functionalities have been added to the standard version of TRIPOLI-4 to add the simulation of precursors, to add the right source distribution of both precursors and neutrons and to add the dynamic scheme for simulating particles. The preliminary results of the Dynamic TRIPOLI-4 code show that it is also possible to do dynamic calculations with a general purpose code like TRIPOLI-4. A demonstration of TRIPOLI-4 is made in a steady state system, in a subcritical system, in a system which has a short period of super-criticality. The results agree nicely with point-kinetics in these simple systems. In addition, with this code, the power produced in a fuel assembly due to the movement of control rods has been simulated successfully using exact geometry and still within reasonable time.

With the development of the basic Monte Carlo techniques for dynamic calculations, the goal has been achieved. The Dynamic TRIPOLI-4 code can be used to do benchmark calculations, which are also solvable with deterministic methods to validate the method thoroughly. The next step is to couple the dynamic code to a thermal-hydraulics code for the simulation of accidental transients for instance.

Advanced deterministic Core Physics development
New developments in the NURESIM core physics codes
As APOLLO2 is the reference lattice code to be used to provide the homogenized multigroup cross-sections for the NURISP 3D core codes (CRONOS2, COBAYA3 and DYN3D), it was first necessary to develop the interfaces between APOLLO2 and these core codes. It comprises the specification of the cross-section libraries generated by APOLLO2 and of the interface software. In particular, this specification includes a detailed definition of all the possible correction factors for performing transport-corrected diffusion calculations; it describes the required configurations and parametric space for transient calculations; and finally, it describes the required auxiliary tool to read generate the libraries for any core code from APOLLO2 output database. It was also necessary to describe the multigroup cross section libraries used in core codes and the parametric space covering the operational conditions for PWR and VVER and the required configurations.

A new tagged version APOLLO2.8-3.E was released in early 2011 as it offers some interesting additional functionalities for NURISP partners. This version includes a new module for collecting isotopic compositions in an APOLLO2 calculation to be directly used by the Monte Carlo code TRIPOLI-4. That offers a consistent verification between these two codes. In addition, some developments were carried out on the Assembly Discontinuity Factors (ADF) providing the capability to extract flux and currents on internal surfaces of a MOC (Method of Characteristic) geometry or a multicell geometry.

Concerning core codes, a nodal method for solving the within-group SP3 equations on trigonal geometry was developed and implemented in the DYN3D code. This method is applicable to reactors with hexagonal fuel assembly geometry. This geometry is of particular importance for VVER-type PWRs as well as for most innovative reactor concepts like high-temperature reactors or sodium fast reactors. A particular advantage in developing a nodal expansion method for hexagonal geometry on a trigonal basis is the possibility of further mesh refinement.

The trigonal SP3 method was derived. Within this approach, a pair of coupled diffusion-type equations has to be solved iteratively for each node and each energy group.

The methodological capability and the consistency of the implementation of the SP3 method was shown via a steady-state test calculation for a simplified VVER-type core consisting of 127 fuel assemblies. The DYN3D SP3 solutions, carried out with six trigonal nodes per hexagonal fuel assembly, were compared to Monte-Carlo calculations obtained by the SERPENT code. The results can be considered a good approximation.

Thereby the new DYN3D-module provides possibilities for advanced visualization of the pin-wise results and extended code coupling inside SALOME. Moreover, the procedure can be extended to visualize not only results from pin-wise power reconstruction, but also from pin-wise calculation using diffusion or SP3 approach. Some results of calculations are presented. SALOME tools were used to visualize the results on trigonal lattice

In the COBAYA3 code, the capability of using a new library format to apply functionalized cross sections and discontinuity factors has been programmed and tested. Additional developments and debugging have been made related to the solution of hexagonal pin-by-pin and triangular nodal geometries. The capability of representing the water gap cells for VVER reactors formed by irregular hexagon has been implemented. These developments have led finally to perfect agreement with the APOLLO2 solutions when using interface discontinuity factors.

Advanced few-group XS libraries generation
Innovative calculation schemes for cross-section production

An industrial calculation scheme using two-level MOC and a reference scheme for XS library generation with APOLLO2/JEFF3.1.1 at the nodal level have been implemented and tested for PWR and VVER.

These schemes are based on a generic two-step methodology and reactor type-specific options and procedures (figure 2). The first step uses a Pij multi-cell model in 281 energy groups and an advanced space-dependent self-shielding model above 23eV. The second step is a MOC calculation using 281energy groups and fine segmentation for the reference scheme or a collapsed energy mesh of around 40g and a simplified spatial mesh for the two-level industrial scheme.

The accuracy of the two-level mode is close to that of the reference one while the computation is around 10 times faster. The corresponding simulations last minutes for each run.

At the pin level, reference and standard calculation schemes for pin-by-pin XS library generation with APOLLO2 have been developed and tested for PWR and VVER. The schemes use pin-cluster branching calculations to take into account the cell neighbourhood. In the current version, the two-step methodology is used in 281g without condensation.

Used with suitable scripts for multi-processor ('parallel') calculations the resulting industrial calculation schemes are able to generate accurate parameterized cross-section libraries in an automatic way. Results are presented in 3 NURISP reports and 4 publications on PWR and VVER benchmarking and XS calculation schemes.
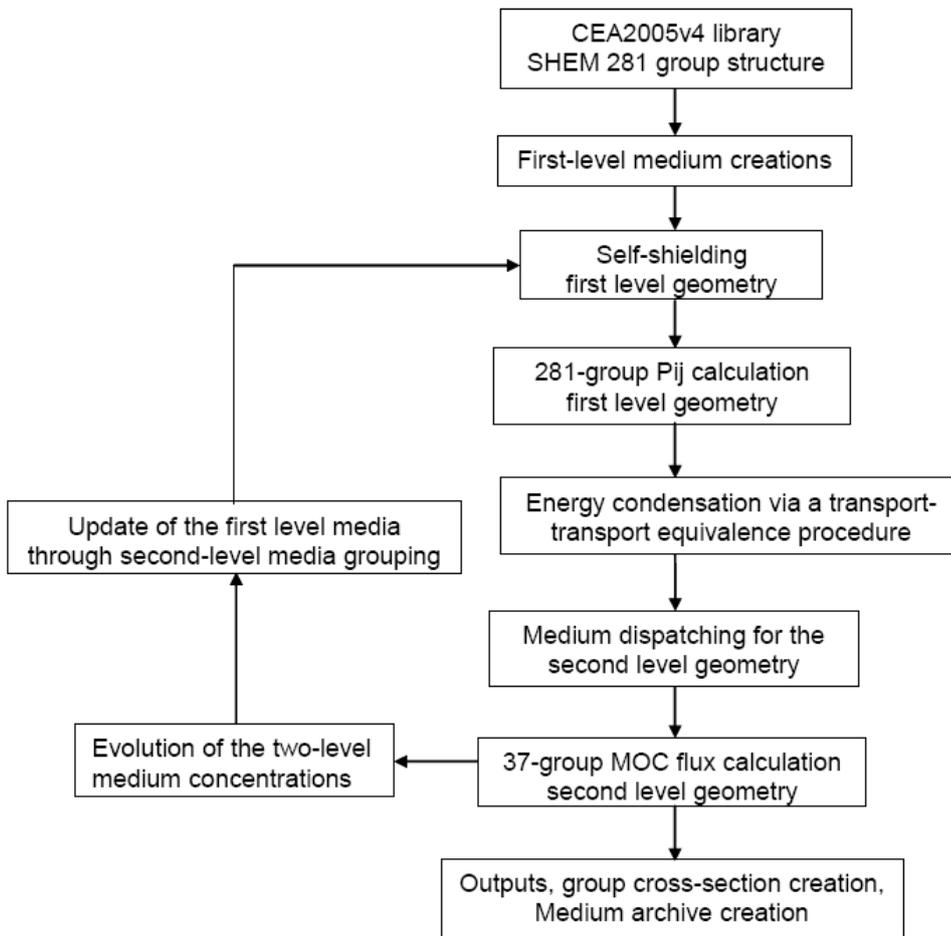
Figure 2 : flow chart of the two-level scheme for depletion calculations

A mathematical benchmark for VVER-440 UOX-Gd assembly calculation was solved with APOLLO2.8.3e MOC/JEFF3.1.1/SHEM 281g library. Depletion evolution to 60 MWd/kgHM was considered. The results for k-infinity, power distributions and U, Pu, Xe and Gd isotopic concentrations vs. burnup were investigated. The results were compared with some other deterministic codes (CASMO4E, HELIOS, KARATE, TVS-M) and good agreement was displayed.

A functional fitting to deal with spectral history effects has been developed by HZDR. A spectral correction index based on the actual Pu-239 concentration was proposed and tested in DYN3D code. The method aims to take into account the deviations in nuclides concentrations occurring during irradiation under different spectral conditions. A clear correlation between a history-related change of homogenized XS and the history-related change of the nodal ^{239}Pu concentration in PWR fuel depleted under nominal and deviated spectral conditions has been found.

This means that having determined the deviation of the ^{239}Pu concentrations from the value in a nominal depletion (and knowing the proportionality factor) it is possible to derive the respective difference in the XS. ^{239}Pu thus serving as an indicator for the actual nuclide content ("burnup-history indicator").

Results of nodal cross section calculation with and without application of historical correction are compared with HELIOS 1.8 for three types of 18x18 PWR fuel assemblies (FA). The application of historical correction significantly improves the cross section estimation accuracy both for UOX and MOX fuel. Thus, the effectiveness of the method − to evaluate spectral-history-related changes of macroscopic cross sections by changes in ^{9}Pu concentrations – has been proved. The methodology is independent from the lattice code.

A new method for the automatic and efficient selection of multivariate polynomials terms as well as a new sampling scheme for lattice calculations that allows analyze local and history effects was proposed by Chalmers university. The new methods were numerically tested for BWR examples.

A step-by-step approach to build functional fittings for XS and IDF with local state variables was developed by UPM. Least squares fitting were applied to get polynomial expressions (statistical program R).

   a. A methodology to optimize the obtained polynomials was proposed, based on sensitivity analysis.

   b. Comparison between the performance of tabulated and parameterized libraries was performed for PWR and VVER cores (in collaboration with INRNE).

A step-by-step approach for functional-fitting with respect to neighborhood has been developed by UPM and INRNE:

   c. For PWR cores, suitable expressions were found to parameterize IDF at the pin level for Cartesian geometry. The extension to nodal IDF and cross sections is still under development and validation.

   d. For VVER cores, to extend the previous methodology further research was required, and some advances were made: a formulation in hexagonal pins where interface fluxes are related to interface currents, averaged fluxes and transverse leakage currents was developed in order to parameterize IDF. The extension to nodal IDF and cross sections is still under development and validation.

Validation
Methodology
During the NURISP project, advanced LWR calculation schemes, based on APOLLO2 for LWR XS library generation and COBAYA3 and DYN3D 3D for pin-by-pin core calculations, have been developed and tested in the NURESIM platform. The aim was to verify and validate the developed codes and calculation routes in lattice and core calculations against Monte-Carlo solutions and clean experiments, within the objective of pin-by-pin neutronics fidelity for safety margin analysis.

The *numerical* validation of codes and calculation schemes vs. TRIPOLI4 reference solutions is performed step-by-step and comprises:

- APOLLO2.8E code functionalities
- reference and industrial calculation schemes in APOLLO2 for nodal and pin-by-pin XS library generation for PWR and VVER
- pin-by-pin calculation schemes in COBAYA3 and DYN3D for LWR
- improved nodal schemes with node sub-division and pin-power reconstruction for VVER

The *experimental* validation vs. VVER mock-up data includes:

- APOLLO2 MOC calculation of three experimental benchmarks using ZR-6 VVER mock-up measurements at KFKI.

- DYN3D nodal calculation of the V1000-LR0-STAT VVER mock-up measurements at NRI.

Conclusions
The following conclusions can be drawn from the validation process :
- During the NURISP project, advanced LWR calculation schemes, based on APOLLO2 for LWR XS library generation and COBAYA3 and DYN3D 3D for pin-by-pin core calculations, have been developed and tested in the NURESIM platform
- Assembly calculation schemes for PWR, VVER or BWR with APOLLO2 are similar, based on a two-step approach with a self-shielding strategy using the multicel-Pij solver followed

by a flux calculation with the MOC solver applied on an unstructured mesh. The geometry capabilities of the MOC solver allow an exact description of the different assemblies or subsets, the meshes and the tracking parameters depending on the treated case

- For numerical validation, a set of LWR lattice benchmarks has been defined and solved. The solutions have been compared to deterministic and Monte Carlo reference solutions and optimized options have been selected to define industrial calculation schemes
- For experimental validation, critical configurations of VVER reactor mock-up ZR-6 have been analyzed with APOLLO2.

The benchmarking results of APOLLO2 show that:

- APOLLO2 is an accurate and reliable lattice physics code and is a good choice for LWR applications,
- the APOLLO2 reference solutions are very close to the TRIPOLI4 Monte Carlo results for PWR and VVER. Differences are more significant for BWR but can certainly be reduced by refining the spatial mesh.
- the industrial calculation schemes for XS generation with APOLLO2 provide solutions which are close to the reference while the CPU time is reduced by a factor of 10.
- ZR-6 experiment analysis with APOLLO2 and the JEFF3.1.1 cross section library gives satisfactory results (figure 3). The k-effective and the power map of several configurations with different absorbers, including Gadolinium, are well reproduced.

As a result of this work, the code has been improved and advanced calculation routes for each type of LWR have been developed. Multi-parameter XS libraries at the nodal and pin level have been generated, including Interface Discontinuity Factors for COBAYA3 and SPH factors for DYN3D.
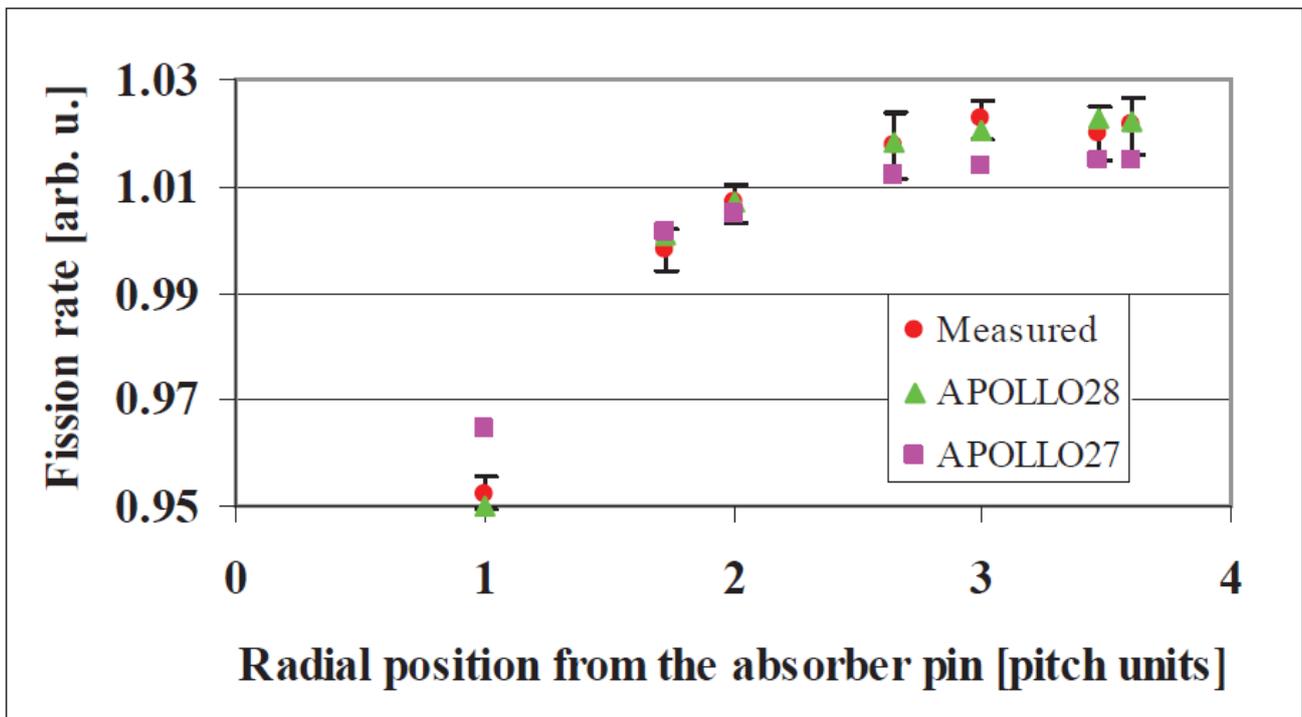

Figure 3 : APOLLO2 simulation of a ZR-6 critical experiment – fission distribution analysis

The testing of pin-by-pin (pbp) core simulators shows that:
- the COBAYA3 pbp transport-corrected few-group diffusion solutions vs. APOLLO2 results show very close agreement for the different LWR. The results with side-dependent DF taking into account the cell neighborhood allow to achieve the target accuracy.
- The DYN3D transport simplified (SP3) solutions with SPH factors provide also excellent results for the tested application (PWR).

From the neutronics point of view, the NURISP objective of pin level accuracy has been achieved in 2D for subsets of the core. For whole 3D core calculation, the treatment of the reflector, which is not simple, has to be done.

After that, the coupling with thermohydraulics being functional at the pin level, the extension to full 3D multi-physics core calculations should be easy.

3.2 Thermal hydraulics

Pressurized Thermal Shoc (PTS)
Main results for PTS analysis
Numerical investigations of PTS were performed using different CFD Codes: NEPTUNE_CFD, TransAT, ANSYS FLUENT and ANSYS CFX. The work focused on validation, benchmarking and improvement of the CFD models for PTS situations using so-called combined-effect experiments (e.g. COSI, TOPFLOW-PTS). Improved model approaches are proposed. The analysis shows the capabilities of CFD codes for the simulation of single-phase and two-phase PTS. During the course of the project, we also described the state of development of one-dimensional two-fluid model for modeling of condensation-induced water hammer that is implemented in the WAHA code. At the end, conclusions and recommendations are made for future modeling work.

The main achievements of the activity on PTS and DCC made in the NURISP project are the following:
- Condensation-Induced Water Hammer: The capability of a 1-D two-fluid model to predict the condensation induced water hammer and the associated pressure peaks has been demonstrated (at least when the initial water level is not low) using a new experimental data base which has a more simple and repeatable procedure of tests than previous data used in NURESIM. The closure laws have to be specific to this situation and should correspond to a two-fluid model obtained by a simple space averaging in order to capture the time evolution of the liquid slug. Standard 1-D two-fluid models implemented in system codes have not this capability since closure laws are adapted to a steady state or slow transient situations and correspond to a two-fluid model obtained by a double space and time averaging.
- Two CFD modeling approaches of free surface flow with condensation (LEIS and URANS-LIM) have been implemented in the two CFD tools of the NURESIM common platform (TransAT and NEPTUNE_CFD). They were compared on a COSI test and applied to several tests of Lim et al. and both methods show good capabilities to predict the most important phenomena (interfacial transfers and turbulent diffusion) governing the PTS issue. The URANS-LIM method has now a wide validation base including air-water tests, steam water tests (Lim et al) in rectangular channel, jet impingement data (Bonetto and Lahey, Iguchi experiments), COSI tests and TOPFLOW-PTS tests. The data base includes both smooth interface and wavy interface and specific models are used to predict the effects of waves on interfacial transfers. In addition, the URANS-LIM method was compared with results obtained by the AIAD model within CFX and by LES used in a FLUENT simulation. For the TOPFLOW-PTS air-water case the AIAD model showed a good agreement with the experimental data.

- A coupled system-CFD PTS simulation method was applied with success and a validation method for this coupled calculation has been applied using ROSA test data in a single phase scenario.

Conclusions and recommendations

The main conclusions of the PTS activity are:

- The benefits of a multiscale analysis of thermalhydraulic issues have been demonstrated. Condensation-induced water hammer was investigated at both CFD (NURESIM) and system (NURISP) scales and the two approaches benefitted from each other. The system-CFD coupled simulation of ROSA test demonstrated the applicability of the method to PTS simulation.
- The CFD modeling of two-phase PTS is now closer to the industrial application since both NEPTUNE_CFD with URANS-LIM and TransAT with LEIS could simulate COSI test and several tests of Lim et al. including smooth interface and wavy interface. These tests include the most important phenomena governing the PTS issue (interfacial transfers and turbulent diffusion). Further validation on steam-water TOPFLOW tests or two-phase ROSA tests is still necessary.

Recommendations for further PTS investigations are:

- The extensive simulation campaign revealed very large differences between the codes results. For this reason, further studies and validation are required to explain and to minimize this inconsistency between the codes and to identify the best models. It requires that each parameter in the closure laws should be validated to avoid model compensating errors. In this context, experimental data for velocity and turbulence are necessary. Therefore, from the CFD validation point of view, there is still a large demand of further development of the measurement technique for the two-phase flows.
- Turbulence modeling of interfacial turbulent flows still remains a difficulty although progress has been made. The modeling of the interfacial shear and interface turbulence parameters (e.g. kinetic energy) was already developed within NURESIM from air-water tests and probably still needs to be continued and validated in the most complex situations including a wavy interface and condensation.
- In case of the two-fluid (U)RANS approach, a specific modeling of the interfacial friction should be further improved. In particular, the efforts to take into account friction coming from the waves that are smaller than the grid size must be continued. The influence of the heat and mass transfer on the friction and turbulence should also be continued but to go further on this point, it would be necessary to get dedicated experimental data or DNS.
- The DCC approaches available in TransAT (LEIS) and NEPTUNE_CFD (URANS with LIM) present the current state-of-the art. From the simulation campaign we can only conclude that both methods seem to be applicable to the PTS simulation. However, it is recommended to further validate them against the TOPFLOW-PTS steam-water experiments.
- It is expected that new simulations of the TOPFLOW-PTS steam-water experiments which are now available would allow further validation of the models and a better identification of the best existing models. Benchmarking of different codes and models also provides valuable information on the strengths and weaknesses of the single approaches.
- Finally, before reactor application for PTS simulation, it is recommended to validate a frozen version of a modeling approach at least on the following validation base, as it is done with NEPTUNE_CFD (Coste et al., 2011): air-water Fabre et al data, Lim et al. (1984), jet impingement data (Bonetto and Lahey, Iguchi experiments), COSI tests. TOPFLOW-PTS and ROSA experiments should be added in the future.

Departure from Nucleate Boiling (DNB)

The multiscale strategy for boiling flow simulation and CHF prediction

A general long term multi-scale strategy for boiling flow simulation and CHF prediction was elaborated at the beginning of the FP6-NURESIM project and was followed during both NURESIM and NURISP projects. It is illustrated in figure 4. DNS tools are developed and applied for micro-scale simulations of local phenomena such as activation of nucleation sites, bubble growing and detachment, influence of wall cavities, and may later identify the DNB mechanism itself which remains unknown. Today, such DNS were applied only to pool boiling and should be extended to convective boiling. Then CFD 2-fluid RANS models are developed and validated using both adiabatic flow and boiling flow. In case of Dry-out investigations Euler-Lagrange simulations are also used. Wall transfers, interfacial transfers and turbulent transfers are later validated on more complex geometries including rod bundle tests. At the end, the resulting CFD simulation tool may be used as a tool for design optimization of the fuels and/or for CHF prediction.
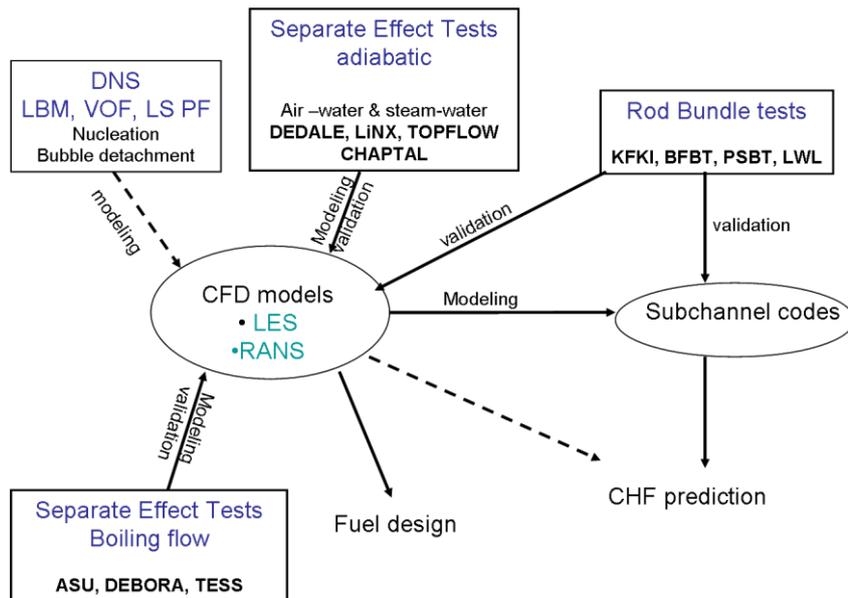

Figure 4: The multiscale strategy for modeling and validation of boiling flow and CHF prediction

Conclusions on capabilities of CFD codes for the simulation of boiling flows and for DNB prediction

Although no big step forward in the modeling of boiling flow and DNB prediction was made within NURISP, significant progress were made thanks to new data that brought precious and unique information obtained in difficult experimental conditions.

Adding data also adds to the difficulty to obtain good results in a wide domain since momentum transfers, wall transfers and turbulence have very complex interactions. Various options still exist in the CFD tools for each closure law and the work of selecting the best combination of models for a best prediction of the whole data base requires more and more understanding of the complex physics of boiling bubbly flow. The main advances in the understanding of flow processes and in the modeling obtained within NURISP came from the following works:

- Wall function laws for momentum specific for boiling bubbly flow were validated on the TAMU boundary layer data

- Mechanical laws including interfacial forces and turbulence were consolidated by a new validation on adiabatic bubbly flow in CHAPTAL

- Polydispersion effects were first investigated in presence of boiling and condensation

- CFD was validated on the local data obtained in real rod bundle geometry (KFKI & PSBT).

- Many attempts to develop a physically based local DNB criterion were made. They were not successful and a very simple criterion on the local void fraction or the Podowski & Podowski criterion are still the best available predictors.

The state of the art in CHF prediction with two-phase CFD is the following:

- CHF in heated tube with steam-water is predicted with a 10% accuracy in the domain: 15< P <20MPa, -0.5 < X < 0.1, 1800 < G < 5000Kg/m2/s, 4 < D < 16mm

  (P: Pressure, X: thermodynamic quality, G: mass flow rate, D: diameter)

- Worse results are obtained at lower pressure and lower mass flux

- CHF in rod bundle was predicted with a 20% accuracy and the effect of the spacer grids was qualitatively correct

- Larger errors were found for CHF in heated tube with Freon

New data required and expectations from DNS

New experimental programs and DNS applications are recommended :

1. Experiment providing information on the wall boiling heat transfer and on the DNB mechanism by using innovative measurement or visualization techniques are expected

2. Experiment providing several parameters together such as velocities, void fraction, bubble size, liquid turbulence and wall and fluid temperatures. Existing test facilities in simple geometries should make efforts to add new measurement devices.

3. Experiments to provide data with evaluation of measurement uncertainty and with the minimum possible uncertainty

4. Micro-scale simulation techniques should be further developed and applied to wall boiling investigations first in pool boiling and then in convective boiling.

Recommendations for further model development

The limited accuracy of present CFD simulation requires further model developments and improvements:

- Wall transfers: wall boiling heat transfer and the DNB local criterion are still weak points of the CFD modeling and require further efforts.

- Turbulence is modeled either by a two-equation model or by Rij-ε model and several options are available for the source terms. In particular it should be made clear whether the Sato's turbulent viscosity model or the addition of two-phase source terms in transport equations is the best approach. Rij-ε model was found necessary in complex flow conditions like downstream a spacer grid. However in two-phase conditions the experience is still limited and further comparison with more simple models is welcome.

- Interfacial momentum transfers were extensively validated. The drag force is rather well predicted. The mean added mass force is not playing an important role in most cases. Both lift and turbulent dispersion forces play important roles on the void distribution. If the Tomiyama's lift model seems to be generally accepted and validated in simple geometry (pipe), several formulations of dispersion force still exist without a clear consensus on the best one. Efforts should then be directed on the modeling of dispersion effects.

- Condensation modeling requires a good heat transfer coefficient together with a good modeling of the polydispersion effect. Separating the two is still difficult. Further measurements in subcooled boiling such as TESS tests with measurement of bubble size distribution should be able to separate these effects if the accuracy of measurement is improved.

- Polydispersion modeling is a difficult issue in bubbly flow. Since the remaining effort may be too long to finalize the closure laws and since it would require a lot of non available data, attempts to simplify the modeling of polydispersion effects are required.

BWR thermohydraulics

Dry-Out investigations with model developments and validation.

a) <u>Development and implementation of the two-fluid model of annular two-phase flow with dryout capability</u>

A three-dimensional model for the description of drops in continuous form was put together. The model requires an implementation of the turbulent stress equation for the drop phase. *A priori* implementation of the three-dimensional model, a channel-averaged model for the pipe flow is checked for the applicability to describe the deposition of drops in annular two-phase flow. The comparison to the experimental data of different conditions and fluids as well as comparison to other calculation methods such as empirical correlation and Lagrangian particle tracking was performed.

It has been concluded that several closure relationships have to be formulated to represent the specifics of the annular two-phase flow. In particular, in such flows the velocity and turbulence intensity distributions close to the wall exhibit strong variations. To avoid prohibitively small grids in that region, law-of-the-wall modeling is applied to relate the wall shear stress to the velocity field away from the wall. This approach is widely used in single-phase calculations, for which the law-of-the-wall is well established. This is not the case for annular two-phase flows, for which the influence of the second phase on the velocity and turbulence distributions in not well understood. For bubbly two-phase flows it was observed that near the wall, velocity distribution preserved the linear log-law shape, however, with different coefficients. More recent experiments indicate that log-law is preserved only for low void fractions, when the buoyancy effects can be neglected. With increasing void fraction, at low and moderate liquid phase Reynolds numbers (up to ~10000), a break-down of the log-law is observed. Measurements performed for annular two-phase flows indicate that the single-phase log-law is preserved in case of well developed thin liquid film. However, for developing thicker liquid film, a modified log-law should be used with a smaller value of the intercept constant in the law-of-the-wall equation.

b) <u>Validation of CEA's Pilot Code of CATHARE-3</u>. The Pilot Code is basically a system code, a new version of the CATHARE code, It has a three-field model to describe the annular regime of two-phase flow with continuous water phase as a liquid film on the heated surfaces, continuous vapor phase in the core of the flow channel, and dispersed liquid phase (water droplets) among the vapor. All three fields have their own equations of conservation of mass, momentum and energy. Transfer of water between the continuous film and dispersed droplets is controlled by, among other things, correlations for entrainment of liquid as droplets into the vapor core and deposition of droplets back onto the film.

This code was used to simulate measurements of liquid film flow in a vertical heated tube. Based on the results of the performed calculations it can be stated that the simulation results, when compared with observations, are usually acceptable but not very good. Looking at the worst cases, there is no one systematic way of under- or over-predicting film flow rates, but dry-out may be predicted when film flow was observed, and vice versa.

Generally, it would be very important to predict correctly the velocities of the liquid film and the gas core. Only with these data correct may we even expect correct entrainment rates from the appropriate correlation. The velocities are affected by friction factors (between liquid and vapor, and between liquid and wall). If we are still in a flow regime with a thick film, there may be a velocity gradient across the thickness of the film (higher velocities near the fast-moving vapor and lower near the

fixed wall). This consideration, and considerations of velocity and droplet density gradients in the vapor core, raise the question whether it is necessary to consider the transverse dimension in film flow modeling, instead of using a 1D model, even when it is as mechanistic as possible.

Core transient problems with advanced modeling of core TH.

The work on validation of the NEPTUNE_CFD code was focused on steady state conditions at low and moderate void fractions in subchannel experiments. For this point, the experimental data were provided by the PSBT experiments. Two different cases were selected within the single subchannel void distribution experiments of the PSBT test series. The simulations were performed with three different meshes and the results indicate a significant dependency of the void fraction close to the heaters on the cell spacing in normal direction.

The OECD/NRC benchmark NUPEC BWR Full-Size Fine-Mesh Bundle Test (BFBT), provides a suitable database for validation and improvement of subchannel and CFD two phase flow models. The BFBT benchmark is composed of two different parts, the void distribution benchmark (Phase I) and the critical power benchmark (Phase II). An exercise from Phase I was selected for the validation of NEPTUNE_CFD. The transient tests performed in the frame of this benchmark represent the thermal hydraulic conditions that may be encountered during a postulated BWR turbine trip transient without bypass and recirculation pump trip. From those postulated turbine and pump trip transient important thermal hydraulic parameters were derived for the test such as the evolution of the pressure, total bundle power, mass flow, radial and axial power profile which serves as initial and boundary conditions for the CFD simulations. Many parameters were measured during the tests which will be used for comparisons with the code predictions.

During the power peak of the turbine trip, the superheat of steam is several hundred K but for this case the heat exchange coefficient between the wall and the liquid was not modeled. Therefore, the coupling NEPTUNE_CFD with SYRTHES has been used in this work and its capabilities have been tested. This coupling allows the calculation of the heat transfer coefficient and furthermore the wall thermal inertia effect, leading to a better steam temperature calculation. Different heat transfer models available in NEPTUNE_CFD have been applied and compared. Taking into account thermal inertia of the rod during the transient or not has a real effect on the simulation.

The code is reproducing reasonably well the experimental data. The closure laws implemented in the simulation are valid for bubbly flow. But the description of different flow regimes such a annular flow or the transition from bubbly to annular flow is still an open issue for the CFD codes, because it is complex to define different continuous and disperse phases in the same domain. The improvement area is located mainly at the evaporation modeling where the code is underestimating the void fraction.

Steam injection in pressure suppression pool simulated at CFD scale.

The simulations of the POOLEX STB-31 and STB-28 experiments were performed (figure 5). STB-31 are steady steam/water-interface experiments and STB-28 consisted of one long-running steam blowdown (duration 3195 s). The purpose of this experiment was to study the formation and condensation of steam bubbles at the blowdown pipe outlet as a function of pool water temperature.
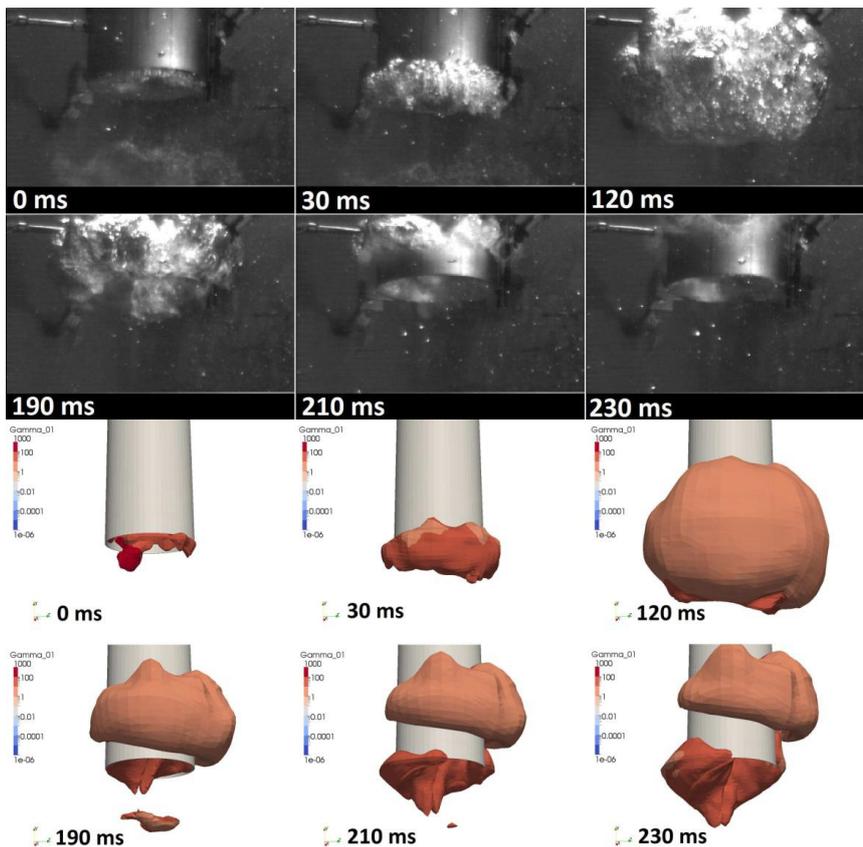
Figure 5 : Inflation and collapse of a bubble during the STB-28-4 blowdown POOLEX experiment and NEPTUNE_CFD simulation

The NEPTUNE_CFD simulations tested the condensation models of Hughes-Duffey, Lakehal et al. 2008b, Coste 2004 and Coste-Laviéville. They confirmed earlier results; the Hughes-Duffey condensation model overpredicted the condensation rate by one order of magnitude whereas the Lakehal et al. condensation model predicted the condensation rate very accurately. The Coste-Laviéville model predicted condensation rates close to the rates of the Lakehal et al. condensation model.

The TransAT simulations utilized the DNS method and tested the direct phase change solution, the surface renewal model of Lakehal et al 2008a, the model of Lakehal 2008b and the model of Banerjee 1968. They led to the underestimation of condensation rates by 1-2 orders of magnitude.

In the TransAT 2D-axisymmetric simulations, grid resolutions of 10-188 μm were tested. With the smallest resolution: 10 - 37.5 μm grids, the condensation rates corresponded with the measured values but with coarser grids the condensation rates were underestimated by the both methods.

The simulations with RANS based condensation models of Lakehal et al. 2008b and Banerjee 1968 predicted condensation rates similar to the values of Lakehal 2008b in NEPTUNE_CFD.
A new model has been developed using ASCOMP's CMFD code TransAT for turbulent multifluid flows. For the purpose of this work package, the modeling strategy based on Interface Tracking Methods (ITM) should be assessed by addressing the following issues: (1) Implement interfacial models and conduct a systematic study comparing these with direct phase change modeling, and (2) Compare the V-LES and LES simulation strategies for turbulent interfacial flows.

To address these issues within this task, the POOLEX experiment has been selected. The main objective of the numerical work consists in studying the influence of the jet velocity and variations of jet falling lengths on the jet penetration depth and the volume fraction of entrained air. Two different approaches were used, namely the mixture model approach with the standard two- equation turbulence model, and interface-tracking approach using the Level-set technique. The results were

compared to the experimental data obtained at FZD Rossendorf, Germany. The results show that the two modeling approaches predict well the jet penetration depth, but visible differences are observed as to the jet instability and free surface deformation, and lateral spreading.

LOCA

TH modeling for macroscopic scale: BFTB and PSBT analysis.

An extensive validation activity of CATHARE 2 and CATHARE 3 codes against experimental data on void distribution provided by NUPEC (Japan) was carried out. BFBT and PSBT benchmarks are devoted to the study of boiling flow in rod bundles in BWR and PWR conditions. They are useful to validate the choices for closure laws, especially for friction and mixing terms in rod bundle geometry. The used experimental database includes:

- 126 steady-state test with test sections representing a single subchannels of a PWR,
- 252 steady-state tests with test section representing an entire, full-scale bundle, according to different possible configurations of the assembly,
- 12 transient tests with bundle tests section (same as for steady-state tests); starting from steady-state conditions, transient boundary conditions are then applied, leading to void production increase.

All single sub-channel tests were simulated with both codes, using a 1D nodalization. The calculations yielded reasonably good agreement with measured data (with some tendency to over prediction) for low void fraction cases (subcooled boiling), as well as for relatively high void fractions (> 0.4).

The steady-state rod bundle tests were also simulated with both codes; both a 1D and a 3D nodalization were used for CATHARE 2 calculations. CATHARE 3 simulations were performed with a 3D module at subchannel scale. Comparisons between data and CATHARE 3 results using either the 6-equation or the 3-field model show that the simulations give roughly satisfactory results for void fraction prediction. Such fair axial profiles validate the main friction closure laws.

The results were finally encouraging and BFBT and PSBT were very useful to improve the validation and assessment of the CATHARE 3 system code, giving the opportunity to implement new mixing terms for bundle geometry in single phase flow and also for mixing of vapour in a continuous boiling liquid.

TH modeling for macroscopic scale: modeling of the critical flow.

In the context of nuclear reactor safety, a pipe breach in the primary circuit is the initiator of a Loss of Coolant Accident (LOCA). The calculation of leak rates involving the discharge of water and steam mixtures plays an important role in the modeling of LOCA's for all types of LWR. Indeed, the flow through the breach determines the depressurisation rate of the system and the time from when the core is no longer under water which in turn are of major concern for when and how different mitigation auxiliary systems will be initiated and be efficient [1]. The way in which the flow evolves as a function of time can be different for the case of a small broken pipe from that corresponding to a small hole in a large pipe even if the initial break flows are the same in both cases. In many licensing applications, the knowledge of the actual flow rate through a break of a given size is not required because what is of interest is the behaviour of the plant for a range of break sizes. Exceptions are the determination of the maximum flow for particular types of breaks (for instance from an instrument penetration in the pressure vessel), and the likely flow from a broken steam generator tube.

The modeling of critical flow in several of the thermal-hydraulics codes is based on semi-empirical models which in general require user defined adjustment factors to obtain a satisfactory agreement

with data in individual situations. In this regard, more universal models should be developed taking into account a wide range of operating and geometry conditions.

In this context, the work deals with the 1-D Delayed Equilibrium Model (DEM) for choked or critical flow rate in steady state or quasi-steady state conditions and the selection of the relevant experimental data for assessing such models. In particular, the focus is made on thermodynamic non-equilibrium conditions, which prevail in the flashing process near the critical section. Three different sets of experimental data have been chosen as reference test for assessing the DEM Model: Super Moby-Dick tests, Marvken tests and UCL test performed in the frame of the EC Environment Programme. This new benchmarking of the DEM model against these set of experiments are successfully demonstrated.

Another part of the work was the implementation of the DEM model in the CATHARE system code. A methodology to implement relaxation models like the DEM has been developed and can be used in every code based on a two-fluid model. The implementation of the DEM in the CATHARE code is still under progress. Several stages of development are already done to verify the Homogeneous Equilibrium Model based on the mechanical as well as the thermodynamic equilibrium between the phases during the flashing process. Both the CATHARE and the UCL codes give the same results. The next stage will be the verification of the DEM model, which is already implemented in the CATHARE code.

Modeling for CFD scale.

The first part of the work aims at building a general-purpose CFD model of heat transfer for the dispersed flow regime in sub-channel geometry. The model encompasses an Eulerian description of the continuous vapour field coupled with Lagrangian tracking of the droplet population. In the first phase of the work, the models governing the dynamics of the droplet motion have been assembled. These are needed for solving the Newton equation of motion for the individual droplets according to the external forces acting upon them.

The second part of the work is the implementation of the heat transfer models for the droplet phase. The principal heat transfer mechanisms are (i) wall-to-vapour (this is a standard feature in all general-purpose CFD codes, and no extra programming is required); (ii) vapour-to-droplet (this will result in droplet evaporation, which will feed back into mechanical equations of motion); (iii) wall-to-droplet by radiation (this also causes droplet evaporation, and wall cooling); and (iv) wall-to-droplet heat transfer as a result of direct droplet impact.

All models have been tested by comparing code predictions against analytical solutions in simple, though relevant, situations. With all models assembled and tested, a computational tool to examine post-dryout heat transfer within a CFD framework has been assembled. In any project following NURISP, the physical bases of the models coded will need to be examined by direct comparison of numerical results against measured data. Most experimental data relate to pipe flow, but the models in the code are quite general and are ready to be applied to sub-channel geometries under LOCA conditions.

In addition, CFD modeling of phenomena governing vapour droplets flow in ballooned core regions during the LOCA has been investigated. This task deals with the development of a CFD code to simulate such droplet dispersed flows with the final aim of carrying out sensitivity studies of blockage ratio and length on wall cooling.

From the analysis of a dedicated experiment performed at IRSN, a model has been proposed in the frame of the NURISP project. The prediction of the correct surface area of the droplets is a key issue to correctly model the inter-phase transfers. The use of a transport equation for the volumetric

interfacial area in the context of dispersed droplets flow has been proposed and the possible models for the closure laws for coalescence and break-up have been reviewed.

From the evaluation of the contributions from the radiative heat in dispersed drop flows, it was concluded that it is required to compute this heat transfer. A model has been released for this purpose and its validation against a large set of test cases has been given.

Finally, a set of test cases of dispersed droplets flow in a geometry representative of three sub-channels has been performed. The flow deviation induced by partially deformed rods has been analyzed as well as the corresponding droplet dynamics. Moreover an analysis of wall to fluid heat transfer has shown some consequences of the latter flow features on the cooling capability of such configuration. It remains to perform more realistic studies in particular including the radiative heat transfer.

3.3 Multiphysics

Status and limits of current methods for plant analysis

An extensive literature survey has been conducted. The current state-of-the-art for codes that are applied in the framework of safety evaluations for LWRs can be characterized as follows:

1. Neutronics analysis is performed using diffusion theory, mostly in 2 energy groups. A wide variety of codes are available. Audit calculations at the core-level can be performed using deterministic or stochastic neutron transport methods, but these are very computer intensive and have not yet entered the production environment.

2. Core dynamics analyses are almost exclusively performed using a variant of nodal neutron diffusion theory, typically in 2 energy groups.

3. Core thermal-hydraulics is performed using an extension of the HEM-model (or a constrained 2-fluid model), e.g. 4 equation model for both PWR and BWR applications, using sub-channel codes. There is a tendency to move to two-fluid sub-channel codes, even including 3-field models, explicitly representing a droplet field. *(the "field" means the geometrical configuration of a phase; for instance: droplets or continuous water)*

4. Current system thermal-hydraulics codes that are used for accident simulation are based on the 2-fluid model (6 equation model). Also in this area, there are trends to move to 2-fluid 3 field modeling, at least within the core region.

5. Fuel performance and fuel behavior codes feature a two-dimensional representation of individual fuel-pins and solve the thermal and mechanical equations. Many additional models are incorporated into these codes: Thermal and mechanical properties, special models to describe the restructuration of the fuel materials and detailed models describing the behavior of fission products, mostly the fission gases. Progress is frequently achieved with improved modeling approaches for the mentioned additional models. There is a trend towards 3D representation of the thermo-mechanics. For LOCA application, the most advanced code represents multiple rods.

Implementation and improvements of coupling schemes for PWR and VVER applications
Extend existing two-level N-TH coupling to include cross-flow
All the developments needed to allow coupling between neutronics and thermal-hydraulics with non conforming meshes have been achieved and tested. An explicit boron transport scheme has been developed in FLICA-4 in order to improve the accuracy of boron dilution.

The software package implementing this new functionality (FLICA-4 + CRONOS-2 + SAPHTOOL + INTERP_2_5D (in collaboration with SP5) has been delivered to end-users within the project.

Perform PWR benchmark of the coupling schemes for boron dilution

The PWR boron dilution benchmark was conducted. Results are provided for the mini-core scenarios and for the PWR whole core transients. This benchmark was calculated using the coupled code systems DYN3D/FLICA COBAYA3/FLICA (figure 6) and the thermal hydraulic module of DYN3D (FLOCAL). Such a demanding case was useful to develop and improve several capabilities of the codes, as the subcritical transient model for COBAYA3, or the boron transport modeling options in FLICA-4, and helps to consolidate the integration of the NURESIM platform codes. A comparison between solutions was provided. The following conclusions can be drawn:

• The coupling of the codes within the NURESIM platform is working,

• The newly implemented into the FLICA code boron transport scheme provides an acceptable evolution of the boron distribution when using the CFL option. In all other cases the numerical diffusion is too high to calculate such boron dilution transients.

• The particle-in-cell method for the boron transport in the thermo-hydraulic module FLOCAL of DYN3D is practically numerical-diffusion-free and can serve as a reference for other models.

• The obtained power maximum after the recriticality depends very much on the correct transport of the deboration front, the numerical diffusion smoothes the perturbation and leads to lower maximum power values.

• The introduced power leads to local boiling in the core. Maximum nodal values of void fraction of more than 80 % are reached.

• The introduction of neutron kinetic sub-steps has a greater influence on the calculated power behaviour. This could be an option for further development of the possibilities of the NURESIM platform.
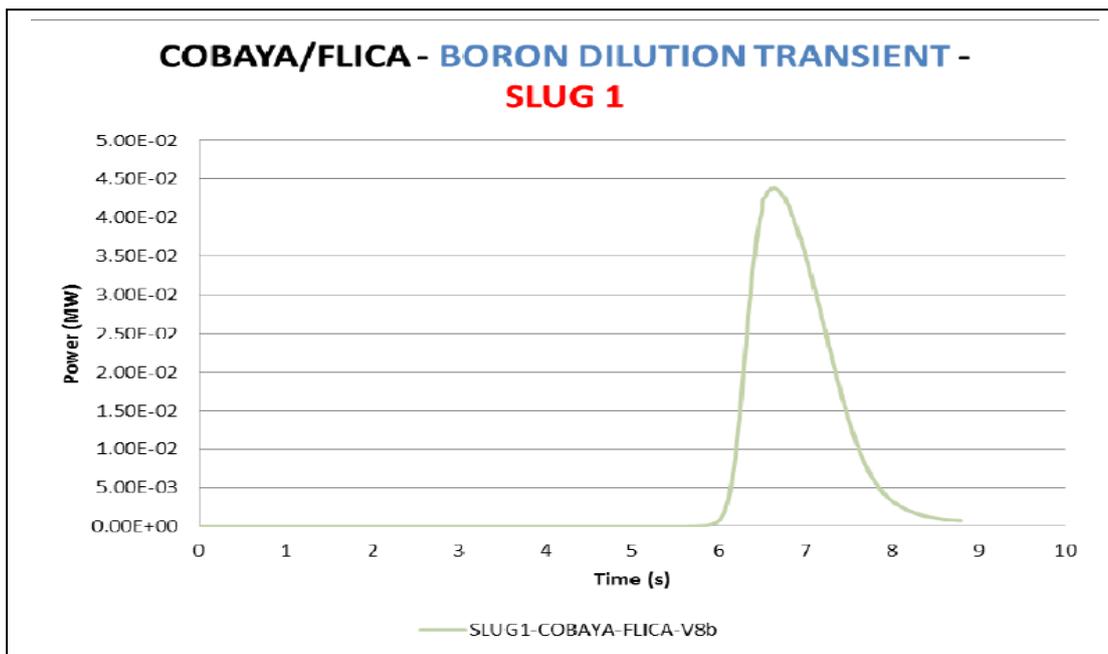
Figure 6 : Simulation of a boron dilution transient using the FLICA-COBAYA3 coupling interface: boron dilution causes a decrease in the boron concentration and thus an increase of the power

As a conclusion, the boron dilution comparison between the codes was highly appropriate for testing the neutronics-thermohydraulics coupling within SALOME platform, as the results are quite good. The neutronics codes performed adequately the transients, and several improvements have been done to simulate precisely the boron dilution. It would be necessary to improve also FLICA4 and SALOME to perform correctly the cases for the future.

Perform VVER benchmark of the coupling schemes for MSLB (hexagonal assembly geometry, nodal and pin level)

The benchmark consists in a nodal test problem which was defined as a core boundary condition problem and was derived from the OECD VVER-1000 MSLB benchmark. It also includes 2D/3D pin-by-pin test problems for subsets of the core.

At the nodal level, the coupling schemes are operational and the VVER MSLB benchmark was conducted during the project. Solutions with standalone and coupled codes have been tested step-by-step. Coupled nodal CRONOS2/FLICA4 and COBAYA3/FLICA4 steady state and transient solutions were compared to each other and to results of independent couplings of COBAYA3/COBRA3 and DYN3D-FLOCAL. The variety of couplings allowed separation of the effects of mixing models, node subdivision in neutronics and the coupling schemes. The results show good overall agreement in steady state and transient.

The calculation schemes including coupled codes & nodal XS libraries were validated vs. steady-state Kozloduy-6 plant data at hot power. The biases are well within the uncertainty band of the Core Monitoring System.

On the basis of this analysis, the coupling schemes in CRONOS2/FLICA4 and COBAYA3/FLICA4 at nodal level were improved with respect to geometrical and parametric coupling. The improved schemes show very good agreement of coupled codes results. At the pin level, the coupling is not yet operational and the testing is ongoing.

Development and testing of automatic adaptive coupling and switching algorithms between different level of detail of the neutron kinetic models (0D, 1D-, and 3D-kinetics) for the coupled code system CRONOS-FLICA

The adaptive coupling & switching algorithms have been initially developed and demonstrated for the RELAP5/PANBOX/COBRA coupled thermal-hydraulics and neutron kinetics code system to switch between three-dimensional (3D), one-dimensional (1D), and point neutron kinetics (PK) models during a transient calculation.

Then, an adaptive 3D–point kinetic method has been implemented in the multi-group version of the DYN3D code coupled with the thermohydraulics model FLOCAL.

The method has been tested on two examples: Boron Dilution Transient and Rod Movement. Validation calculations show that the results produced by the automatic dimensionally adaptive switching algorithm retain the accuracy of the 3D reference calculations. They are faster, typically requiring only 30 to 70% of the CPU time needed by the 3D reference calculations while maintaining 3D-accuracy.

Implementation and improvements of coupling schemes for BWR applications
Investigate coupling modes in the time-domain

The goal of this task was to identify and suggest areas for improvements to the temporal coupling schemes that will be ultimately used to perform multi-physics transient analysis in the NURISP platform. Following the findings from the state-of-the-art review, a set of functional specifications for semi-implicit code coupling was developed. These specifications should allow for the implementation in SALOME of semi-implicit and implicit code coupling schemes that should increase the numerical accuracy and stability of the coupled multi-physics simulations.

Three different temporal coupling schemes with increasing level of complexity were presented. The first two schemes are based on Fixed-Point-Iteration (FPI) methods. These methods require

minimum modifications to the codes, and the corresponding Application Programming Interface (API) is very similar in nature to the one employed in the current version of the SALOME/NURESIM platform, which is based on an Operator Splitting coupling scheme. As a result of this proximity, some relatively detailed specifications could be provided. The third coupling scheme is based on an Approximate Block-Newton method (ABN). This scheme replaces the straightforward FPI loop by a Newton method, and would require a more different API and more intrusive integration of the individual codes. It is therefore not expected that such API could be developed within the time frame of the NURISP project. As result, the corresponding specifications were provided in a less detailed way.

This task is completed and the actual implementation of these different schemes is now up to the partners that own the different codes, since the implementation would require modifications to the integration procedure of the codes and to their corresponding API (access to the source of the codes is needed).

Development of new multi-physics coupling schemes
New coupling schemes between system codes and CFD codes
This task aims at developing new coupling schemes between System and CFD codes. First, a comprehensive study on the state of the art on available coupling schemes has been carried out.
An Application Protocol Interface, named ICOCO, has been proposed, implemented in the system code CATHARE and in the CFD Trio_U and integrated in SALOME. This API has been used for several System-CFD coupling exercises involving CATHARE (in sequential mode) and Trio_U (in parallel mode) in order to analyze various coupling schemes including overlapping and non-overlapping schemes, and synchronous or asynchronous time stepping coupling schemes.

In a first phase, only single phase flows where dealt with. It should be noticed that the proposed API and the single phase coupling scheme has been widely and successfully used within the THINS European project. During the last months, an extension of the coupling scheme to two phase flows has been proposed and encouraging results has been obtained.

The coupling interface between CFD code Trio_U and the system thermal hydraulic code CATHARE was tested on a numerical benchmark based on an actual main-steam-line-break (MSLB) accident scenario. A simplified model of primary circuit of VVER-1000 nuclear reactor (including 4 loops and primary sides of steam generators) was created in CATHARE. Trio_U computational domain covering the reactor downcomer and lower plenum was overlapped by the CATHARE model of primary circuit so as to properly simulate the mixing in the reactor inlet part. This work demonstrated that the coupled system of Trio_U and CATHARE codes can be used in a satisfactory way to simulate the coolant mixing in the downcomer and the flow reversal in the afflicted loop during the MSLB event in a VVER-1000 nuclear power plant.

Develop coupling schemes between component or system and fuel-behavior codes
  Coupling a system thermal-hydraulic code with the DRACCAR code devoted to analysis of rods deformation during thermal transients was foreseen for at three reasons:
- first, the thermal-hydraulic scale is the good one to be able to model ballooning effects on the sub-channel flow regime,
- second, the CATHARE code developers have proposed new API based on more general framework called ICOCO,
- finally, some common background concerning code coupling has made this coupling between the DRACCAR and CATHARE codes obvious within the NURISP project.

First, DRACCAR was integrated into the SALOME platform and an Application Programming Interface (API) was developed in order to process the exchanges between codes. This API is based

on two levels; the first one is called by the pilot module (the main task in fact, like the time or iteration manager,…) and the second one can be called directly by both modules (some services like the geometry update,…). This API is the simplest one, already tested and validated within the stand-alone version of the DRACCAR code but does not verify total programming orthogonality. To overcome this drawback, a second API has been developed after the first release of the DRACCAR code coupled to the simplified 1-phase thermal-hydraulic module of DRACCAR named GAS3D within the NURISP project.

Nevertheless because of some time and internal developing constraints for both codes, final consensus on the definitive API was not achieved early enough within the project to allow a validated integration of the DRACCAR code. This is why the DRACCAR code is finally released within the NURISP framework with the first API version.

A test-case which deals with an interesting reflood problem on a deformed bundle has been conducted. For this, the DRACCAR code has been coupled with the CESAR code following the first API. The CESAR code is the 2-phase thermal-hydraulic module of the ASTEC code, ASTEC being the reference code of the EU SARNET project. This test-case demonstrates the great interest to couple thermal-mechanics and thermal-hydraulics during thermal transients when the structure deformation is significant enough. Indeed an important physical coupling occurs between restriction of the fluid section (due to the structure deformation), thermal exchanges (convection, radiation) and flow regime evolution (annular, inverted annular,…).

At the same time, the second API has been implemented in the DRACCAR code and tested for both the GAS3D and CESAR codes. The development of the coupling interface CATHARE-DRACCAR is now the next step of the work-program.

3.4 Integration

Assistance and integration

The main objective of this work package is to provide fast and efficient support to the users of NURESIM platform, and facilitate integration of codes. To fulfil this objective, the work-package delivers a two level support to the users of the NURESIM platform. The first level of assistance consists in answering questions and qualifying bugs. The second level of assistance consists in supporting and advising code developers in the integration of their codes.

During the first half of the project, the focus was put on code integration, with the introduction of three new codes (CATHARE, TRIO-U, DRACCAR). During the second period, the support on code integration continued, notably on the use of MED data format, but the focus was more put on multi-physics applications.

With the set-up of transient cases like the boron dilution scenario, which requires the extension of the coupling schemes at the pin level (and large computational resources), problems arose linked with memory leaks. These problems are common in the context of distributed computation, but were too important to allow the completion of demanding transients. Thus it was necessary to parse the integrated codes, and track all the significant memory leaks. Non regression memory tests were also developed in order to avoid the introduction of new leakages in the coming developments. After this campaign against memory leaks, it was possible to run all transient cases until the end.

The integration tool hxx2salome was initially written in bash (Linux script shell), and was only dealing with sequential C++ components. TRIO-U being a parallel code, its integration has been done manually. A parallel C++ component was firstly developed, and then the SALOME/CORBA component above was also developed (the underlying component being parallel, it was not possible to automatically generate the CORBA layer). Also the DRACCAR SALOME/CORBA component

was not generated, because the DRACCAR development team made the choice of python language. Taking advantage of the feedback of these two components, the integration tool was extended to manage parallel C++ components, and also python components.

All the complicated aspects of MPI management are now handled by the generator. This will in the future simplify the development of parallel SALOME components. In order to manage python components, hxx2salome tool was merged with an existing SALOME tool (yacsgen). The new tool, called yacsgen, is developed in python language (more portable than bash) and it is able to build PYTHON and C++ sequential or parallel components. The parallel aspects were tested with components developed with LIBMESH (an open-source C++ parallel finite elements library). It is planned now to apply the new generator to TRIO-U, and check that identical results are retrieved with a generated component.

In collaboration with the uncertainties sub-project, the URANIE platform was adapted to HPC architectures. This required two actions: the adaptation to a queuing system (taking into account the type of queuing system and the MPI implementation), and a mechanism for launching the codes, regardless of their own level of parallelism.

Adaptation of the SALOME platform

The SALOME platform has been adapted continuously to satisfy the needs of the other sub-projects. The first version of the software was delivered to the project in December 2009. The development of new use cases revealed some bugs, notably in the MED and YACS library, and in the integration tools. Also some improvement was required in relation with the run of large transient cases.

This led to the delivery of a patched version of SALOME containing all the necessary corrections, available on CEA ftp site. These patches were at the end merged in a SALOME new version, delivered to NURISP in November 2011.

Production of NURESIM platform

To ensure the building of the platform and its maintenance, a "production environment" is provided: CVS repositories and bug-tracker tool, hosted and maintained by CEA, and a testing procedure, which was developed in the first half of the project and delivered through NURISP web site. This robust and user-friendly procedure enables to perform automatic test campaigns on various Linux platforms and on Windows. At the end of the period, there was twenty-five functional test cases, plus six memory tests.

Every night, SALOME and the NURESIM components are compiled, and tested against the non regression base. Doing this allows an early detection of bugs, and facilitates their correction. During the course of the project, four versions of the NURESIM platform were delivered.

3.5 Uncertainty quantification

Status and limits of current methods for plant analysis

The objective of this work-package was to identify weaknesses, current remedies and future work concerning *model verification, model validation and predictive estimation.*
Model verification:

- SQA procedures are well developed in general, but in the future, improvements will be needed specifically for software operating on massively parallel computer systems.

- For the purpose of using the method of manufactured solutions, existing and future computer codes need to be able to accommodate the addition of source terms (i.e., the right-hand side) of the physics-model PDEs programmed in the code.

- The discretization errors must be quantified in order to separate them, in principle, from other error and uncertainty sources, such as physics modeling errors and variability in physical

properties. Two major shortcomings affect current verification methods, namely: (i) estimating discretization errors using solutions on multiple mesh resolutions is a computationally expensive process, and (ii) current methods for complex physics simulations are not robust. While developing methods for alleviating these shortcomings, it is presently recommended to account for such numerical and epistemic modeling errors by using biases as upper bounds for their overall effects.

Model validation:

- Improved quantitative methods are needed for statistical comparison of simulation results and experimental results for situations where very few experimental realizations are available but a large number of spatially or temporally distributed measurements are available at locations, or times, over the system of interest.

- Improved methods are needed for quantifying and properly interpreting, for informed decision-making, differences in probability distributions from computations and experiments for various system response quantities (e.g., how should differences in mean response quantities be weighted relative to tails in each distribution?).

- Development and formal compilation of accurate benchmarks are needed in each of the elements of NURISP reactor core simulation (including neutronics, fluid flow and heat transfer, and multi-physics modeling); such benchmarks include a variety of benchmarks, from unit problems to system-scale problems.

Predictive estimation:

- *The development of new sampling methods*. Predictive estimation and sensitivity analysis require ensembles of multiple related runs of the code. In multi-physics and multi-scale physics models with nonlinear coupling and large numbers of input parameters, each code run is computationally expensive. New and efficient sampling techniques are needed that employ a combination of statistical features and applied mathematics features of the PDEs (e.g., the elliptic nature of the PDEs).

- *Uncertainty propagation from modules to multi-physics systems*. Simulations of multi-physics comprise codes with different physical understanding and different levels of validation. This requires extension of current capabilities in calibration of models, particularly for physical models, to estimate credible prediction uncertainty.

- *Extrapolation to higher levels in the validation hierarchy*. Within NURISP, there will be more data for component and subsystem testing, and less for mock-ups and full system testing. It is an open question how to integrate this data to make credible predictions with defensible uncertainty estimations at the full system level.

Development of New Methods and Software Modules for Local and Global Adjoint Sensitivity and Uncertainty Analysis for the VSUQ-Platform URANIE

Sensitivity and uncertainty analysis procedures can be either local or global in scope. The objective of local analysis is to analyze the behaviour of the system response locally around a chosen point or trajectory in the combined phase space of parameters and state variables. On the other hand, the objective of global analysis is to determine all of the system's critical points in the combined phase space formed by the parameters and dependent (state) variables, and subsequently analyse these critical points by local sensitivity and uncertainty analysis. The methods for sensitivity and uncertainty analysis are based on either deterministic or statistical procedures. In principle, both types of procedures can be used for either local or for global sensitivity and uncertainty analysis, although, in practice, deterministic methods are used mostly for local analysis while statistical methods are used for both local and global analysis.

Illustrative examples for *local* and *global* sensitivity and uncertainty analysis have been reviewed

Development of New Methods and Software Modules for Statistical Sensitivity and Uncertainty Analysis for the VSUQ-Platform URANIE

Development and implementation into URANIE of software modules based on stochastic finite element method (SFEM)

Polynomial Chaos (PC) expansions have been used for uncertainty quantification in a large variety of domains (e.g. in solid mechanics, fluid flows, thermal sciences, etc.). PC expansions is a probabilistic method consisting in the projection of the random model output on a basis of orthogonal stochastic polynomial with respect to the random inputs. The stochastic projection provides a compact and convenient representation of the model output variability with respect to the inputs. PC are suitable for the sensitivity analysis. In fact, Sobol's sensitivity indices and even more ANOVA functional decomposition, can be immediately deduced from the PC expansion of the model output. PC functionalities have been implemented in the toolbox NISP (Non Intrusive Spectral Projection). For NURISP Project, NISP has been integrated in the URANIE platform to use polynomial chaos for uncertainty modeling. This will provide access to this method for the codes and coupled schemes which are built in the other sub-projects.

Development and implementation into URANIE of software modules based on "RaFu"-method

In order to motivate the use of the RaFu method in uncertainty analysis, the general uncertainty analysis practice is first recalled with a special attention to the classical probabilistic modeling used by many analysts in nuclear safety. Since it becomes clear that this kind of approach is not fully satisfactory to handle both aleatory and epistemic uncertainties that usually arise in safety studies, the RaFu method is introduced and fully described. It is based on a more flexible framework than the probabilistic one allowing to respect the real state of knowledge on uncertainties and leading to a reliable estimation of uncertainty margins.

The integration of the RaFu method in the NURISP platform has been made through the URANIE software. It is coupled with an optimal numerical treatment (based on an extension of Monte-Carlo simulations to the theory of evidence framework and on the introduction of a decision step before the propagation). The new RaFu-URANIE tool has been applied to the uncertainty analysis of the Couplex model related to deep storage of nuclear waste. It came out that this approach provides robust uncertainty margins related to the assumptions about pdfs that are required within the classically probabilistic modelling.

Development of New Hybrid Methods and Software Modules for Combining Adjoint and Statistical Sensitivity and Uncertainty Analysis Methods for the VSUQ-Platform URANIE

An illustrative example (radioactive decay chain), suited for integrated statistical / deterministic methods, using adjoint functions, for performing sensitivity and uncertainty analysis has been proposed. It has been implemented in the dedicated platform URANIE. Once the adjoint functions have been calculated, they are used to obtain the sensitivities to all system parameters, by simple quadratures, without needing to solve repeatedly differential and/or integral equations. Therefore, for this example, the sensitivities have been calculated numerically (making use of the adjoint functions) and their analytical forms have been used as verification.

Development of Methods and Software Modules for Data Adjustment and Assimilation for the URANIE Platform

Development of methodologies and modules for time-independent data assimilation and best-estimate adjustment

A rigorous methodology for computing best-estimate predictive results using experimental and computational information in conjunction with models of time-independent systems has been presented. This methodology uses Bayes' theorem in conjunction with information theory to assimilate consistently all available experimental and computational uncertainty-afflicted information (including discretization-modeling errors) for obtaining best-estimate calibrated model parameters and responses, together with correspondingly reduced uncertainties. This new

methodology also provides quantitative indicators for assessing the consistency among parameters and responses, for consequent acceptance or rejection of information within the overall assimilation procedure. The main features of this new methodology have been illustrated analytically through a paradigm application to a stationary neutron diffusion problem.

Development of methodologies and modules for time-dependent data assimilation and best-estimate adjustment

A representative application of a general mathematical framework for simultaneously calibrating model parameters and responses through the assimilation of experimental data, leading to "best-estimate" values with reduced uncertainties for both parameters and responses in a generic time-dependent system is presented. This mathematical framework provides an indication of the agreement between the computed and experimentally measured responses while performing: simultaneous calibration of all parameters and responses, treatment of systems involving correlated parameters and responses, simultaneous calibration over all time intervals.

In premiere, a large-scale application of assimilating transient experimental data from the OECD/NRC BWR Full-Size Fine-Mesh Bundle Tests (BFBT) benchmarks using computations performed with the core thermal-hydraulics code FLICA4 has been performed (figure 7).
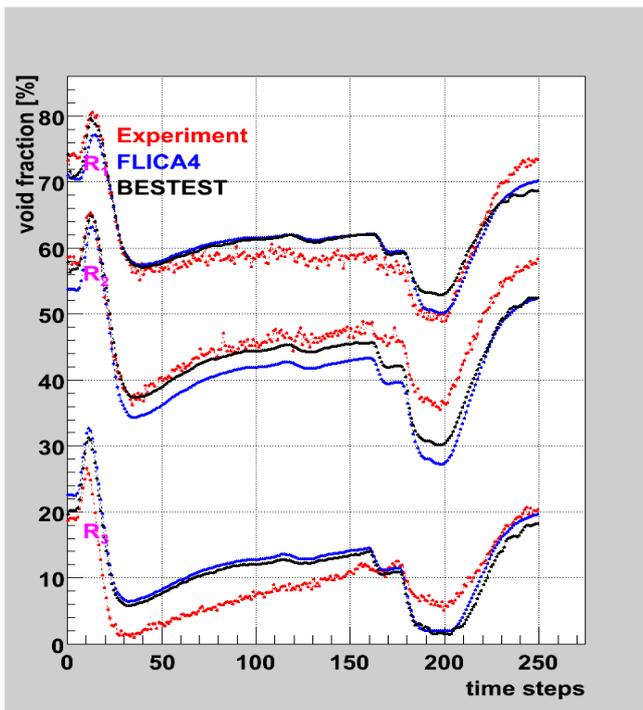
Figure7: Model Calibration using void fraction measurements in the BFBT Facility

Comparison of
1) experimentally measured,
2) FLICA4 computations
3) best-estimated
responses for the void-fractions at three elevations

Adaptation of the VSUQ-Platform URANIE to High Performance Computing (HPC) Architectures

Uncertainty analysis can take two shapes: local sensitivity and statistical methods. When adjoint models are available, local sensitivity can be computed around a nominal point with very little extra computational cost. However, such adjoint models are sometimes difficult or impossible to determine. Defining local sensitivity by numerical approximation or using statistical methods requires much more computational power, but they can be used in a very generic manner.

Defining a plan of numerical experiments can be used:
- for uncertainty propagation in order to compute distributions of the output parameters and/or global sensitivity indices,
- to compute local sensitivity matrix using Forward Sensitivity,

- to provide a base for the training of a meta-model, be it a polynomial model, an artificial neural network, or a NISP spectral decomposition.

Depending on the number of parameters involved, the model non linearity, and the quantities of interest, the number of model evaluations typically range from tens to thousands of evaluations.

The work presented consists in the adaptation of URANIE platform to HPC environments and to the presentation of a sensitivity analysis performed on a RIA scenario computed by the CRONOS code. The adaptation of URANIE to supercomputers offers very promising prospects for the realization of uncertainty analysis.

Development of a Formal Procedure to Account for Scaling Effects on Sensitivity and Uncertainty Analysis of Thermal-Hydraulics System Codes within the VSUQ-Platform URANIE

Scaling is a reference 'key-word' in engineering and in physics. The relevance of scaling in the water cooled nuclear reactor technology constitutes the motivation for the present activity.

A critical survey (or 'scaling state-of-art') was first carried out. Then, we classified the information about achievements in the area of thermal-hydraulics which are relevant to scaling: the concepts of 'scalingpyramid' and related 'scaling bridges' were discussed establishing a logical path across the scaling achievements (represented as a 'scaling puzzle'), thus proposing the 'roadmap for scaling': the objective is addressing the scaling issue when demonstrating the applicability of system codes in the licensing process of nuclear power plants. The database from the operation of properly-scaled ITF and the availability of qualified system codes are identified as main achievements in NRSTH connected with scaling. The 'roadmap to scaling' constitutes a unified approach to scaling that solves the 'scaling puzzle' created by researches performed during a half-a-century period.

Some steps of the proposed methodology were applied to a benchmark case. The results from a Loss-Of-Coolant Accident (LOCA) test in the PSB facility, simulating a VVER 1000 NPP at a scale 1/300, constitute the experimental reference for performing the scaling analysis. The so-called "PSB/VVER transient" simulating the LOCA on a virtual VVER with re-scaled PSB results has been compared with CATHARE2 calculation of the LOCA of a real VVER reactor (figure 8).
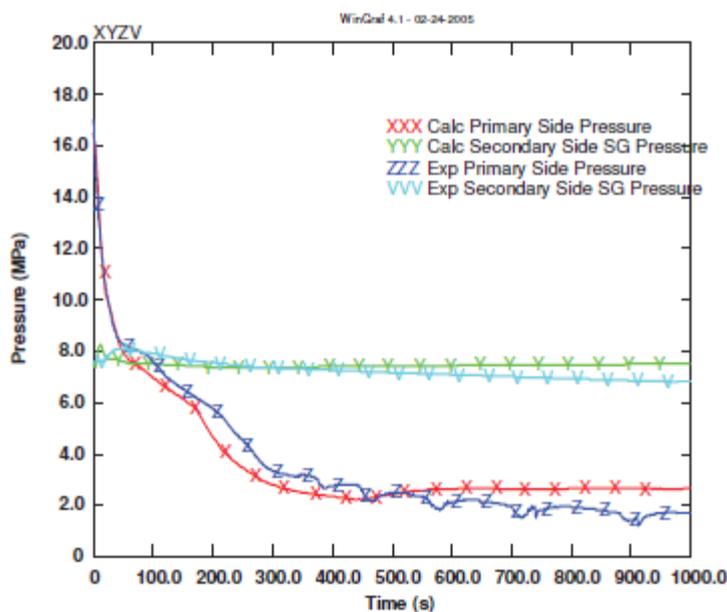
Figure 8: Comparison between the scaled (calc) transient and the reactor (exp) transient

3.6 Conclusion

The initial objectives of the project were achieved for core physics, thermal-hydraulics, multi-physics and uncertainty quantification, all these topics having been supported successfully by the integration effort.

Core physics: New Monte-Carlo methods were successfully implemented especially the coupling between Monte-Carlo codes and thermal-hydraulics codes for thermal feedback. This coupling was implemented on a small scale and the full core implementation should be done in the framework of another new project. Concerning deterministic methods, advanced LWR neutronics calculation schemes for pin-by-pin core calculations have been developed and successfully tested against experiments. They are based on a two-step approach with a self-shielding PIJ approach followed by a MOC flux calculation applied on an unstructured mesh. These schemes are usable in the framework of the NURESAFE follow-up project.

Thermal-hydraulics: LOCA modeling was improved: an extensive validation of the NURESIM system codes against experimental data on void distribution was conducted and the Delayed Equilibrium Model was implemented for critical flow modeling. In addition, significant progress were made towards the DNB prediction with CFD codes thanks to new experimental data. However, this effort must be pursued in the framework of the NURESAFE project.

Multi-physics: Neutronics thermal-hydraulics coupling applications were developed in order to model PWR boron dilution accident or steam line break. These developements are pursued within NURESAFE in order to progress towards higher fidelity pin-by-pin modeling and to validate the models.

Uncertainty quantification: many innovative methods for uncertainty quantification were developed and succesfully tested. These methods should be used in the framework of the follow-up project (NURESAFE) in order to assess the uncertainties of some accidental transient simulations and multiscale thermal-hydraulics simulations.

4. Dissemination activities and exploitation of the results

As NURISP objective was to provide more accurate representation of core physics, thermal-hydraulics and fuel thermo-mechanics in a standard environment for easy multi-physics and multi-scale simulations and uncertainty analysis, the project outcome consists of physical models, numerical methods, software modules, code integration and supervision tools, and know-how in core physics, thermal-hydraulics, fuel modeling, multi-physics coupling and uncertainty methods. This knowledge can be used through the codes incorporated in the NURESIM platform which are the property of NURISP partners.

The NURISP partners promote the dissemination of the codes and most of NURESIM codes have long been widely used by the nuclear industry (utilities, vendors), TSOs, R&D labs and universities. The users of codes who belong to these customer organizations already benefit from the improvements made during the NURISP project that have been incorporated in the recent versions of codes. Thus, the position of the NURESIM codes is consolidated by the end of the project. This effort will be strengthened through the FP7-NURESAFE project (2013-2015) which intends to deliver state-of-the-art simulation tools relevant for safety analysis of LWR reactors.

In addition, the use of the codes has been extended during the course of the FP6-NURESIM and NURISP projects among the NURESIM community (NURISP partners, members of the NURISP Users' Group) which will continue to use the codes after the end of NURISP, especially in the framework of the FP7-NURESAFE project.

During the NURISP period (three and half years), the impact of the project has been enhanced by carrying out specific dissemination actions : conferences, publications, training sessions and courses, interaction with the users' group and the open web site. These activities contributed to spread the scientific advances of the project, to receive comments from non-NURISP partners, to improve the developments and to promote the NURESIM platform as a state-of the-art set of codes. The target of the communication actions are the potential users of simulation tools for nuclear reactors, inside and outside Europe:
- Nuclear professionals: industry, TSOs, and research organizations,
- Scientific nuclear community.

The NURISP open website (http://www.nuresim.com/) is public and offers an overview of both projects NURESIM and NURISP including many reports, synthesis of NURISP actions, a list of NURISP publications available in the open literature and link to NURISP related information (for instance, information about SALOME). One can also find in this site the periodic and final activity reports of NURESIM and NURISP projects and presentations at the general meetings.

Many training sessions were organized within NURISP for dissemination of the know-how of new solvers and code functionalities. The training sessions are firstly directed towards the NURISP partners and the members of the Users' Group, especially students at pre- and postdoctoral levels in the universities involved in the project. They included core physics codes, thermal-hydraulics codes, coupling tools and uncertainty quantification methods and tools.

Moreover, two-phase CFD training sessions were presented in the framework of international workshops: 3D-SUNCOP training sessions, THICKET Seminars (OCDE), IAEA Regional Workshop on CFD and some others.

The NURISP Users' Group comprises seven organizations from the industrial nuclear sector or universities. The Users' organizations had an access to the NURISP codes and methods and, in turn, provided the consortium with their own feedback in a well balanced agreement.

During the project period (three and half years) more than 70 communications were published. They consist in:

- papers presented in international nuclear conferences,
- articles in nuclear and energy press and in scientific journals.

The main conferences for NURISP presentations were: NURETH, ICONE, PHYSOR, M&C, ICAPP, ENC and ANS. They were best suited to spread the NURISP knowledge and results within the international nuclear community.

The NURISP website adress is: http://www.nuresim.com

Publications

No.	DOI	Title	Main author	Title of the periodical or the series	Number, date or frequency	Publisher	Place of publication	Date of publication	Relevant pages	Permanent identifiers (if applicable)	Is open access provided to this publication?
1		extension of the analytical nodal diffusion solver ANDES to triangular-Z geometry and coupling with COBRA-IIIc for hexagonal core analysis	JA LOZANO	annals of nuclear energy	vol 37; 380-388, 2010			06/09/2010	380-388		Yes
2	10.1016/j.anucene.2011.03.006	building the nodal nuclear data dependences in a many-dimensionnal state variable space	J Dufek	anals of nuclear energy							Yes
3		A Monte Carlo method for calculation of the dynamic behaviour of nuclear reactors	BL Sjenitzer	progress in nuclear science and technology	to be completed	atomic energy society of Japan		05/09/2011	to be completed		Yes
4		variance reduction for fixed source Monte Carlo calculations in multiplying systems by improving chain-length statistics	BL Sjenitzer	anals of nuclear enregy vol 38:2195-2203, 2011	38			06/06/2011	2195-2203		Yes
5		The European Project NURISP for Nuclear reactor simulation	B Chanaron	transactions of the American Nuclear Society	vol 103, pp 671-672 (2010)	american nuclear society		08/11/2010	vol 103, 671-672 (2010)		Yes
6	10.1002/fld.269 5	a conservative local	SATO,								Yes
		interface sharpening scheme for the constrained interpolation profile method	NICENO								
7	10.1016/j.mucengdes.2011.08.06 6	validation of the large interface method of NEPTUNE_CFD 1.0.8 for pressurized thermal shock applications	COSTE	Nuclear engineering and design							Yes
8		A new modelling strategy for phase-change heat transfer in turbulent interfacial two-phase flow	Lakehal	International journal of Multiphase flow	37,6 pp 627-639	Elsevier		02/05/2011	627-639		Yes
9		Qualification of the APOLLO2 lattice physics code of the NURISP platform for VVER hexagonal lattices?	HEGYI	KERNTECHNIK Journal	issue 2012/04	HANSER		03/09/2012	218-225		Yes
10		Development of multi-physics code systems based on the reactor dynamics code DYN3D?	KLIEM	KERNTECHNIK Journal	issue 2011/3	HANSER		06/06/2011	160-165		Yes

LIST OF SCIENTIFIC (PEER REVIEWED) PUBLICATIONS, STARTING WITH THE MOST IMPORTANT ONES

No.	Type of activities	Main Leader	Title	Date	Place	Type of audience	Size of audience	Countries addressed
				LIST OF DISSEMINATION ACTIVITIES				
1	Conference	INSTITUTE OF NUCLEAR RESEARCH AND NUCLEAR ENERGY - BULGARIAN ACADEMY OF SCIENCES	TRIPOLI4 solutions to VVER-1000 assembly and core benchmarks	04/05/2009	international conference on mathematics, computational methods MC2009 (saratoga springs)	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
2	Conference	UNIVERSIDAD POLITECNICA DE MADRID	performance of whole pin-by-pin calculations by domain decomposition through alternate dissections	04/05/2009	international conference on mathematics, computational methods MC2009 (saratoga springs)	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
3	Conference	UNIVERSIDAD POLITECNICA DE MADRID	development and performance of the ANDES/COBRAIII coupled system in hexagonal-z geometry	04/05/2009	international conference on mathematics, computational methods MC2009 (saratoga springs)	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
4	Conference	TECHNISCHE UNIVERSITEIT DELFT	zero variance scheme to accelerate the fission source convergence in a Monte Carlo calculation	04/05/2009	international conference on mathematics, computational methods MC2009 (saratoga springs)	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
5	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	validation of DYN3D pin power calculation against experimental VVER full core benchmark	04/05/2009	international conference on mathematics, computational methods MC2009 (saratoga springs)	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
6	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	application of a stepwise verification and validation procedure to the neutron kinetics code DYN3D w	13/07/2009	ICONE17 international conference on nuclear engineering, Brussels	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
7	Conference	Karlsruher Institut fuer Technologie	extension of the CAD interface program McCad for TRIPOLI4	19/04/2010	RPSD2010 international conference, LAS VEGAS	Scientific community (higher education, Research)		countries developing nuclear energy
8	Conference	TECHNISCHE UNIVERSITEIT DELFT	A monte-carlo method for calculation on the dynamix behaviour of nuclear reactors	18/10/2010	international conference "Supercomputing in Nuclear Applications -Monte-Carlo" 2010, TOKYO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
9	Conference	INSTITUTE OF NUCLEAR RESEARCH AND NUCLEAR ENERGY - BULGARIAN ACADEMY OF SCIENCES	two level MOC calculation schems in APOLLO2 for cross-section libraries generation for LWR hexagonal	09/05/2011	international conference M&C 2011 - RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
10	Conference	UNIVERSIDAD POLITECNICA DE MADRID	a proposed parametrization of IDF depending on neighborhood for pin-by-pin computations for LWR	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
11	Conference	UNIVERSIDAD POLITECNICA DE MADRID	interface discontinuity factors in the modal eigenspace of the multigroup diffusion matrix	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
12	Conference	TECHNISCHE UNIVERSITEIT DELFT	implementation of the dynamic Monte Carlo method for transient analysis in TRIPOLI4	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
13	Conference	TECHNISCHE UNIVERSITEIT DELFT	demonstration of a zero variance based scheme for variance reduction to a monte carlo calculation	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy

14	Conference	TECHNISCHE UNIVERSITEIT DELFT	a flexible coupling scheme for Monte Carlo and thermohydraulics codes	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
15	Conference	Karlsruher Institut fuer Technologie	development of a coupling scheme between MCNP5 and subchanflow for simulation of LWR	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
16	Conference	CHALMERS TEKNISKA HOEGSKOLA AB	complex models of nodal nuclear data	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
17	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	a trigonal nodal solution approach to the simplified P3 equations in the reactor code DYN3D	09/05/2011	M&C 2011 international conference, RIO DE JANEIRO	Scientific community (higher education, Research)		countries developing nuclear energy
18	Conference	INSTITUTE OF NUCLEAR RESEARCH AND NUCLEAR ENERGY - BULGARIAN ACADEMY OF SCIENCES	advanced calculation schemes and cross-section library generation in hexagonal geometry with APOLLO2	19/09/2011	21st symposium of AER on VVER reactor physics and safety - DRESDEN	Scientific community (higher education, Research) - Industry		countries operating VVER reactors
19	Conference	MAGYAR TUDOMANYOS AKADEMIA KFKI ATOMENERGIA KUTATOINTEZET	qualification of the APOLLO2 lattice physics code of the NURESIM platform for VVER hexagonal lattice	19/09/2011	21st symposium of AER on VVER reactor physics and safety - DRESDEN	Scientific community (higher education, Research) - Industry		countries operating VVER reactors
20	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	influence of spectral history on PWR full core calculation results	19/09/2011	21st symposium of AER on VVER reactor physics and safety - DRESDEN	Scientific community (higher education, Research) - Industry		countries operating VVER reactors
21	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	the simplified P3 approach on a trigonal geometry in the nodal reactor code DYN3D	19/09/2011	21st symposium of AER on VVER reactor physics and safety - DRESDEN	Scientific community (higher education, Research) - Industry		countries operating VVER reactors
22	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	the trigonal nodal SP3 method of the reactor code DYN3D	16/05/2011	ICONE19 international conference on nuclear engineering - CHIBA (Japan)	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
23	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	SP3 solutions versus diffusion solutions in nodal codes	12/09/2011	international conference on transport theory - PORTLAND (USA)	Scientific community (higher education, Research)		countries developing nuclear energy
24	Conference	Karlsruher Institut fuer Technologie	optimisation of a coupling scheme between MCNP5 and SUBCHANFLOW for high fidelity modeling of LWR	16/04/2012	international conference PHYSOR 2012 - KNOXVILLE (USA)	Scientific community (higher education, Research)		countries developing nuclear energy
25	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	trigonal nodal SP3 method with mesh refinment capabilities - development and verification	16/04/2012	international conference PHYSOR 2012 - KNOXVILLE (USA)	Scientific community (higher education, Research)		countries developing nuclear energy
26	Conference	TECHNISCHE UNIVERSITEIT DELFT	general purpose dynamic Monte Carlo with continuous energy for transient analysis	16/04/2012	international conference PHYSOR 2012 - KNOXVILLE (USA)	Scientific community (higher education, Research)		countries developing nuclear energy
27	Conference	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	two-phase advances in the NURESIM and NURISP projects	17/05/2010	international conference on nuclear engineering ICONE18 - XIAN (China)	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
28	Conference	INSTITUT JOZEF STEFAN	2 fluid model for simulation of water-hammer induced by condensation of hot vapor on stratified flow	31/05/2010	ICMF2010 international conference on multiphase flow - TAMPA (USA)	Scientific community (higher education, Research)		countries developing nuclear energy
29	Conference	ASCOMP GmbH	A level-set based	31/05/2010	ICMF2010	Scientific		countries

						Scientific community (higher education, Research) - Industry	countries developing nuclear energy
			interfacial heat and mass transfer model built in TransAT		international conference on multiphase flow - TAMPA (USA)	community (higher education, Research)	developing nuclear energy
30	Conference	UNIVERSITE CATHOLIQUE DE LOUVAIN	revisiting modeling techniques and validation experiments for 2 phase chocked flows relevant to LOCA	11/10/2010	NUTHOS-8 international conference on nuclear thermohydraulics, operation and safety SHANGHAI	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
31	Conference	KUNGLIGA TEKNISKA HOEGSKOLAN	Eulerian description model for deposition of drops in annular flow regime	11/10/2010	NUTHOS-8 international conference on nuclear thermohydraulics, operation and safety SHANGHAI	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
32	Conference	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	validation of the large interface method of NEPTUNE_CFD 1.0.8 for PTS applications	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
33	Conference	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	best practice guide lines for two phase CFD calculations	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
34	Conference	GESELLSCHAFT FUER ANLAGEN- UND REAKTORSICHERHEIT (GRS) MBH	transient CFD analysis of emergency core cooling injection at natural circulation conditions	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
35	Conference	ELECTRICITE DE FRANCE S.A.	NEPTUNE-CFD validation	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
36	Conference	INSTITUT DE RADIOPROTECTION ET DE SURETE NUCLEAIRE	numerical simulation of condensation in boiling flow (validation of NEPTUNE-CFD on Debora-Tess data)	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
37	Conference	ASCOMP GmbH	PTS prediction using the CFMD code TransAT: the COSI test case	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
38	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	pre-test simulation of TOPFLOW PTS experiments with ANSYS CFX 12.0	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
39	Conference	USTAV JADERNEHO VYZKUMU REZ A.S.	CFD simulation of critical heat flux in a tube	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
40	Conference	USTAV JADERNEHO VYZKUMU REZ A.S.	CFD simulation of critical heat flux in a rod bundle	14/09/2010	CFD4NRS-3 Worshop on application of CFD codes to nuclear reactor safety issues WASHINGTON	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
41	Conference	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	2 phase CFD advances in the NURESIM and NURISP projects	17/05/2010	international conference on nuclear engineering ICONE18 - XIAN (China)	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
42	Conference	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	multiscale thermohydraulics analyses - performed in NURESIM and NURISP projects	30/07/2012	ICONE20 international conference on nuclear engineering ANAHEIM (USA)	Scientific community (higher education, Research) - Industry	countries developing nuclear energy
43	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	numerical simulation of the TOPFLOW-PTS	26/09/2011	NURETH14 thermohydraulics international	Scientific community (higher education,	countries developing nuclear energy

			steam water experiment		conference - TORONTO	Research) - Industry		
44	Conference	ASCOMP GmbH	new perspective in the simulation of PTS from RANS to VLES&LES to interface tracking methods	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
45	Conference	INSTITUT JOZEF STEFAN	slug modeling with 1D two fluid model	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
46	Conference	INSTITUT JOZEF STEFAN	modelling and validation of turbulent boiling flow in a rectangular channel	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
47	Conference	PAUL SCHERRER INSTITUT	RANS and LES simulations of the air-water TOPFLOW-PTS experiment	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
48	Conference	Karlsruher Institut fuer Technologie	validation of NEPTUNE-CFD two-phase flow models using the OECD/NRC BFBT benchmark database	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
49	Conference	GESELLSCHAFT FUER ANLAGEN- UND REAKTORSICHERHEIT (GRS) MBH	development of a coupled 1D-3D thermohydraulics code for PWR simulation, application to PTS scenario	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
50	Conference	UNIVERSITE CATHOLIQUE DE LOUVAIN	delayed equilibrium model and validation experiments for two-phase chocked flows relevant to LOCA	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
51	Conference	PAUL SCHERRER INSTITUT	revisiting temporal accuracy in Neutronics/Thermohydraulics code coupling	26/09/2011	NURETH14 thermohydraulics international conference - TORONTO	Scientific community (higher education, Research) - Industry		countries developing nuclear energy
52	Conference	Karlsruher Institut fuer Technologie	validation of NEPTUNE-CFD two-phase flow models using PSBT experiments	03/05/2010	Kern Technik Gesellschaft (KTG) meeting - BERLIN	Scientific community (higher education, Research)		Germany
53	Conference	PAUL SCHERRER INSTITUT	quantitative phase field modeling of boiling phenomena	12/09/2011	20th international conference on nuclear energy for new europe - BOVEC (Slovenia)	Scientific community (higher education, Research)		european countries
54	Conference	PAUL SCHERRER INSTITUT	a new conservative phase change model for nucleate boiling	12/09/2011	20th international conference on nuclear energy for new europe - BOVEC (Slovenia)	Scientific community (higher education, Research)		european countries
55	Conference	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	integration of DYN3D inside the NURESIM platform	25/10/2010	17th pacific basin nuclear conference - CANCUN (Mexico)	Scientific community (higher education, Research) - Industry		countries in Asia - Pacific ocean
56	Conference	INSTITUTE OF NUCLEAR RESEARCH AND NUCLEAR ENERGY - BULGARIAN ACADEMY OF SCIENCES	DYN3D/FLOCAL vs COBAYA3/FLICA4 solutions of the VVER MSLB benchmark	19/09/2011	21st symposium of AER on VVER reactor physics and safety - DRESDEN	Scientific community (higher education, Research) - Industry		countries operating VVER reactors
57	Conference	UNIVERSIDAD POLITECNICA DE MADRID	presentation of the multi-physics subproject of the NURISP European project	28/09/2011	37th annual meeting of the spanish nuclear society - BURGOS (spain)	Scientific community (higher education, Research)		Spain
58	Workshops	Karlsruher Institut fuer Technologie	first seminar on model validation&calibration, sensitivity and uncertainty	30/11/2010	University of PISA (Italy)	Scientific community (higher education, Research)	20	European countries

			quantification					
59	Workshops	Karlsruher Institut fuer Technologie	first seminar on model validation&calibration, sensitivity and uncertainty quantification	04/04/2012	Karlsruhe University - KARLSRUHE (Germany)	Scientific community (higher education, Research)	20	European countries
60	Presentations	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	first NURISP open seminar	30/11/2009	DRESDEN (Germany)	Scientific community (higher education, Research) - Industry	70	European countries
61	Presentations	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	second NURISP open seminar	02/04/2012	KARLSRUHE (Germany)	Scientific community (higher education, Research) - Industry	70	European countries
62	Conference	USTAV JADERNEHO VYZKUMU REZ A.S.	VALIDATION OF DYN3D PIN-POWER CALCULATION AGAINST EXPERIMENTAL RESULTS FROM THE LR-0 REACTOR	01/10/2012	22nd Symposium of AER on VVER Reactor Physics and Reactor Safety, Pr#honice (Czech Republic)	Scientific community (higher education, Research) - Industry		countries operating VVER reactors
63	Thesis	TECHNISCHE UNIVERSITEIT DELFT	A zero-variance based scheme for Monte Carlo criticality simulations	06/09/2010	DELFT University	Scientific community (higher education, Research)		European countries
64	Thesis	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	Spectral History Modelling in the Reactor Dynamics Code DYN3D", HZDR,	11/10/2010	Dresden	Scientific community (higher education, Research)		European countries
65	Thesis	FORSCHUNGSZENTRUM DRESDEN-ROSSENDORF EV	Development of a Nodal Simplified P3 Neutron Transport Method on a Trigonal Geometry	06/09/2010	Dresden	Scientific community (higher education, Research)		European countries
66	Thesis	UNIVERSIDAD POLITECNICA DE MADRID	Development of an analytic nodal code for multigroup neutronic diffusion for reactors in 3D rectangu	04/10/2010	Madrid	Scientific community (higher education, Research)		European countries
67	Thesis	UNIVERSIDAD POLITECNICA DE MADRID	Multiscale Core-Cell Advanced Methods in Three-Dimensional geometries and Multi-Groups for Light Wat	02/07/2012	Madrid	Scientific community (higher education, Research)		European countries
68	Thesis	TECHNISCHE UNIVERSITEIT DELFT	Transient Analysis of a Nuclear Reactor Using the Monte Carlo Method	03/06/2013	Delft	Scientific community (higher education, Research)		European countries
69	Thesis	INSTITUTE OF NUCLEAR RESEARCH AND NUCLEAR ENERGY - BULGARIAN ACADEMY OF SCIENCES	Implementation of higher-resolution schemes for VVER simulation	10/09/2012	SOFIA	Scientific community (higher education, Research)		European countries
70	Thesis	INSTITUTE OF NUCLEAR RESEARCH AND NUCLEAR ENERGY - BULGARIAN ACADEMY OF SCIENCES	Reference calculation schemes for VVER	10/09/2012	SOFIA	Scientific community (higher education, Research)		European countries
71	Thesis	UNIVERSIDAD POLITECNICA DE MADRID	Development and implementation of the subdomain decomposition in neutronic-thermalhydraulic coupling	14/06/2010	Madrid	Scientific community (higher education, Research)		European countries
72	Thesis	UNIVERSITA DI PISA	Development and validation of CFD methods for two-phase flow analysis	07/09/2009	PISA	Scientific community (higher education, Research)		European countries

73	Thesis	KUNGLIGA TEKNISKA HOEGSKOLAN	modeling of droplet deposition for dryout prediction in nuclear fuel assemblies	15/06/2009	Stockholm	Scientific community (higher education, Research)		European countries
74	Thesis	LAPPEENRANNAN TEKNILLINEN YLIOPISTO	cfd modeling of direct contact condensation by applying condensation models of separated flow	11/06/2012	Helsinki	Scientific community (higher education, Research)		European countries
75	Thesis	INSTITUT JOZEF STEFAN	cfd simulation of condensation induced water hammer	14/09/2009	Ljubljana	Scientific community (higher education, Research)		European countries
76	Thesis	MAGYAR TUDOMANYOS AKADEMIA KFKI ATOMENERGIA KUTATOINTEZET	use of lattice Boltzman method to simulate boiling flows	20/09/2010	Budapest	Scientific community (higher education, Research)		European countries
77	Thesis	USTAV JADERNEHO VYZKUMU REZ A.S.	CFD simulation for DNB prediction	05/09/2011	Prague	Scientific community (higher education, Research)		European countries
78	Thesis	Karlsruher Institut fuer Technologie	validation of two phase flow modeling	25/06/2012	Karlsruhe	Scientific community (higher education, Research)		European countries
79	Thesis	COMMISSARIAT A L ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES	Modeling of boiling flows with poly-dispersed bubbles	17/10/2011	Grenoble (France)	Scientific community (higher education, Research)		European countries

Appendix: Glossary of acronyms

- **AIAD** *(Algebraic Interfacial Area Density)*

- **ANOVA** *(ANalysis Of VAriance)* collection of statistical models, and their associated procedures, in which the observed variance in a particular variable is partitioned into components attributable to different sources of variation

- **API** (*Application Programming Interface)* Interface of a computer program that allows its interaction with other software, within the NURESIM platform, with SALOME,

- **BFBT** *(BWR Full-size Fine-mesh Bundle Tests)*

- **BLOCA** *(Break Loss-of-Coolant Accident)*

- **BWR** *(Boiling Water Reactor)*

- **CAD** *(Computer Aided Design)*

- **CFD** *(Computational Fluid Dynamics)*: a CFD code solves 3D equations of fluid dynamics

- **CHF** *(Critical heat flux)*

- **CMFD** *(Computational Multi-Fluid Dynamics)*

- **CORBA** *(Common Object Request Broker Architecture)* Standard that enables software components written in multiple computer languages and running on multiple computers to work together. The SALOME platform is based upon CORBA.

- **COSI** *(Condensation at Safety Injections)*: Experiment to investigate Condensation at Safety Injections

- **CVS** *(Concurrent Versions System)* Client-server free software revision control system in the field of software development. It keeps track of all work and all changes in a set of files, and allows several developers (potentially widely separated in space and time) to collaborate.

- **DCC** *(Direct Contact Condensation)*

- **DEM** *(Delayed Equilibrium Model)* : Model of depressurization of the Primary circuit in the initial phase of a LOCA

- **DNB** *(Departure from Nucleate Boiling)*: Critical Heat Flux that may occur in boiling bubbly flow conditions

- **DNS** *(Direct Numeric Simulation)*

- **FA** *(Fuel assembly)*

- **HEM** *(Homogeneous Equilibrium Model)*

- **HPC** *(High Performance Computing)*

- **IDF** *(Interface Discontinuity Factor):* Numerical factor often used in innovative core physics methods

- **LES** *(Large-Eddy Simulation)*

- **LGPL** *(Lesser General Public Licence)*

- **LIM** *(Large Interface Model)*

- **LOCA** *(Lost of coolant accident)*

- **LWR** *(Light Water Reactor)*

- **MED** *(Modèle d'Echange de Données) or Data Exchange Model.* It is the SALOME standard to exchange numerical fields and meshes

- **MOC** *(Method Of Characteristics):* A flux calculation method in core physics

- **MPI** *(Message Passing Interface)* It is a language-independent communications protocol used to program parallel computers.

- **MSLB** *(Main Steam Line Break)*

- **NEA** *(Nuclear Energy Agency)*: Agency of the OECD

- **NISP** *(Non Intrusive Spectral Method)*

- **NURESIM** *(Nuclear Reactor SIMulation)*: Name of the reference European simulation platform and of an FP6 project

- **PDE** *(Probability Density Estimator)*

- **PIJ** *(Probability of collision)*: A method that can be used to calculate the neutron fluxes distribution inside the fuel assemblies

- **PSBT** *(PWR Sub-channel and Bundle Test)*

- **PPOOLEX:** Name of a test facility designed and constructed at Lappeenranta University of Technology (Finland)

- **PTS** *(Pressurized Thermal Shock)*

- **PWR** *(Pressurized Water Reactor)*

- **RANS** *(Reynolds Average Navier Stockes)*: Fluid dynamic equations resulting from a time or ensemble averaging in a steady flow

- **REGARDS:** Experiments investigating stratified flow with droplet entrainment and deposition

- **ROSA:** *(Rig of Safety Assessment)* : name of an Integral Effect Test Facility

- **RPV** *(Reactor Pressure Vessel)*

- **SBLOCA** *(Small Break Loss-of-Coolant Accident)*

- **SQA** *(Software Quality Assurance)*

- **SYRTHES** Name of a heat conduction code coupled to NETUNE-CFD

- **THINS** *(Thermal Hydraulics for Innovative Nuclear Systems)*: FP7 project

- **TOPFLOW:** Name of an experimental facility operated by HZDR (Germany) dedicated to the simulation of flow during PTS

- **URANS** *(Unsteady Reynolds Average Navier Stokes)*

- **VLEIS** *(Very large-Eddy and Interface Simulation)*

- **VSUQ** *(Validation, Sensitivity and Uncertainty Quantification)*

- **VVER** *(Water-Water Energetic Reactor)*: Pressurized water reactor with triangular lattice

- **XS** *(Cross Section)*

- **YACS** It is the SALOME supervision module, used to couple codes together