Title: Development of a multi-physics, multi-scale coupled simulation system for LWR safety analysis

Author: Yann Périn

Organisation: GRS

Introduction

In a nuclear reactor core, different fields of physics interact. A prime example is the interaction between Neutron Physics (NP) and Thermal-Hydraulics (TH). In the early years of the nuclear industry, the available computational power only allowed the simulation of one field of physics at once and only in a limited fashion. In the nineties, thanks to the constant improvements in computational capabilities, thermal-hydraulic system codes were coupled with 3D neutronics models. Since then, the use of coupled three-dimensional Thermal-Hydraulics / Neutron Physics (TH/NP) multi-physics codes has become state-of-the-art for performing reactivity initiated accident and several operational transient analyses.

Usually those couplings are made at the assembly level. However, most of the acceptance criteria for rigorous safety analyses are based on local values within the reactor core: thermal-hydraulic, neutron-kinetic or thermal-mechanic parameters at the fuel pin level. In order to simulate the behavior of a fuel pin during a transient, one often needs to simulate the behavior of the whole nuclear power plant.

The aim of this work is to develop a multi-scale simulation tool that can model the whole plant behavior as well as part of the core at the pin level in order to evaluate local safety parameters.

In order to model the power distribution at the pin level, several approaches are available. Pin-by-pin reconstruction methods within a neutron diffusion core usually provide a satisfactory accuracy for safety analysis while being fast. In order to model the plant level as well as the pin level, in addition to TH/NP multi-physics couplings, the system code and the sub-channel code also need to be coupled.

In the framework of the NURESAFE project [1], a multi-physics multi-scale simulation tool was developed. The TH system code ATHLET, the subchannel code CTF and the neutronics code DYN3D were integrated and coupled on the Salomé platform. The resulting tri-code coupled system has been tested on representative PWR and BWR transients [5, 6]. The results from a MSLB transient simulation are presented in this paper.

Codes description

In this section a brief introduction of the ATHLET, CTF, DYN3D codes is given. The NURESIM platform is also introduced.

The DYN3D reactor simulator

DYN3D [2] is a reactor core simulator developed at the Helmholz Zentrum Dresden Rossendorf (HZDR), Germany. It is applied for performing of steady state and transient analysis in LWR for hexagonal or square fuel assemblies' geometries. The diffusion equation is solved using a nodal expansion method. Pin-by-pin simulations are possible using the pin power reconstruction method implemented in DYN3D.

The CTF TH subchannel code

Coolant-Boiling in Rod Arrays | Two Fluids (COBRA-TF) [3] is a 3D Thermal/Hydraulic simulation code designed for LWR subchannel analysis. It has been improved and updated at the North Carolina State University (NCSU), USA by the Reactor Dynamics and Fuel Management Group (RDFMG) and subsequently re-branded as CTF.

The ATHLET TH system code

The thermal-hydraulics computer code ATHLET (Analysis of Thermal-hydraulics of LEaks and Transients) [4] is developed by the GRS for the analysis of anticipated and abnormal plant transients, small and intermediate leaks as well as large breaks in light water reactors.

The NURESIM platform

The NURESIM platform is based on the open-source software Salomé (http://salome-platform.org) which is an open-source software co-developed by EDF, CEA and OpenCascade.

Originally created for CAD applications, it has since evolved into a platform for code coupling in the framework of a series of three consecutive European Commission founded projects: NURESIM (2006-2008), NURISP (2009-2011) and NURESAFE (2013-2015).

Code integration and coupling on the NURESIM Platform

The standard coupling approach for TH/NK codes at the GRS uses the "serial integration" whereby the coupled codes are compiled together. One code is the master program while the other(s) code(s) is a subroutine of the master program. With the parallel processing approach however, the data exchange is made by a third party program such as the NURESIM platform. The simulations are not necessarily run in parallel but can actually be consecutive. The coupling interfaces for each code must be compatible with the coupling supervisor without depending directly on the other code. When a well-defined standard is adopted, a coupling with one code in such an environment allows a coupling to any other code compliant with the standard.

During the NURISP project, a standard API for code integration and coupling, the ICoCo API (Interface for COde Coupling), was developed for NURESIM. It is a pure abstract API defining a standard way for two physical codes to exchange information between them.

Codes are integrated on the NURESIM platform as "components". In the case of a full integration, single code's functions can be called and run from the platform (e.g. initialize code, read input, perform steady-state, etc.).

Data exchange on the platform is performed directly through memory using a dedicated data structure: MEDCoupling. The MEDCoupling format was developed by EDF and CEA. The goal was to design a standardized approach that could be used to exchange data between codes. The MEDCoupling data model consists of two components:

- Mesh: The mesh contains the geometry of a domain which is represented by a set of cells and nodes.
- Fields: The fields are the physical quantities that the codes actually exchange.

Generally meshes generated by different codes do not match. For this purpose, interpolations tools are available on the platform, which automatically interpolate the exchanged fields from one mesh to the other.

Coupling of CTF (and ATHLET) with DYN3D

The thermal-hydraulic code (either CTF or ATHLET) describes the fluid dynamics and the heat exchanges in the reactor core while the neutron physics code simulates neutronics. TH feedback core distributions (moderator density and/or temperature, fuel temperature, boron concentration) are transferred to the neutronics code which returns core power distribution.

Coupling of CTF with ATHLET

The coupling of CTF and ATHLET was achieved using the parallel coupling method. With the parallel coupling approach, both thermal-hydraulics codes model the core region usually with different level of precision. This means that the ATHLET model retains its core nodalization. ATHLET provides the core inlet (mass flowrates, fluid enthalpy, and boron concentration) and core exit boundary conditions (pressures) to CTF.

This is a one-way coupling, since no information from the sub-channel code is transferred to the system code.

When the CTF model is more detailed than the ATHLET one, which is usually the case, the inlet mass flow in a CTF channel is proportional to the ratio between the CTF and the ATHLET channel flow area. For the pressure and the fluid enthalpy (/temperature) a uniform distribution is passed from the ATHLET channel to all connected CTF channels.

This approach was expanded to include mixing matrices for the transfer of inlet temperature. The inlet temperature in each channel in CTF is calculated as a weighted sum of the temperature in the four cold loops. The mixing matrix can be provided by measurements or by higher resolution simulations as provided by CFD codes.

Time coupling scheme

For transient simulations, explicit time step synchronization is used. The time step size is the smallest one proposed by all three codes. This solution increases the computation time but improves the stability and the precision. No stability problems and no divergence of the parallel running TH models have been encountered in any of the performed coupled simulations.

Tri-code coupling between ATHLET-CTF-DYN3D

The tri-code coupled system ATHLET, CTF and DYN3D covers the whole range, from plant level to the pin level. It has been tested on several PWR and BWR transient analyses [5, 6]. Results of for a MSLB transient are presented in the next section.

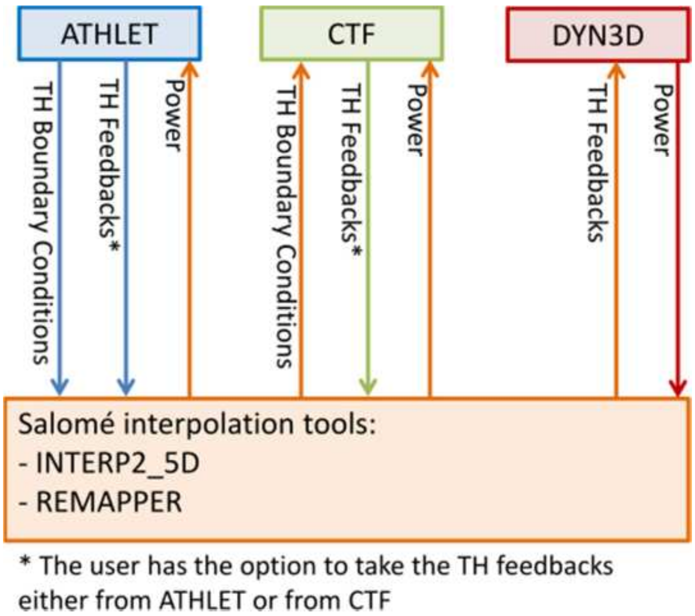

Fig. 1 - Flow chart of the ATHLET/CTF/DYN3D coupling on the Salome platform

Application of the three-code coupled system to a MSLB transient

One objective of the NURESAFE project was the "development and execution of a set of simulation schemes towards higher-fidelity simulation of a PWR MSLB transient for improved predictions the key safety parameters". The results obtained with the tri-code coupling are presented in this section.

For this analysis, the Zion PWR nuclear power plant (NPP) is selected. The core loading considered is described in the specification of the OECD/NEA UOX/MOX Core Transient Benchmark [7].

Physical description of the transient

The initiating event is a double-ended main steam line break in one of the secondary loop. The break causes a pressure drop in the secondary side of the Steam Generator (SG). The temperature in the affected cold leg depends directly on the decrease of the saturation temperature (i.e. the pressure) on the secondary side. A (high) heat transfer between primary and secondary side exists as long as the SG contains enough liquid water.

The coolant temperature distribution in the reactor core, which determines the criticality conditions, depends mostly on the mixing phenomena between the coolant flows from the affected and intact primary loop in the downcomer and in the lower plenum.

For conservativeness, the transient is simulated at End of Cycle (EOC) from Hot Zero Power (HZP) state (undercriticality fixed to 1$). In addition, a stuck out control rod is considered in the core zone where the minimum temperature.

Models description

The system code ATHLET models the primary and secondary sides using boundary conditions for the steam line pressure and the feed water mass flow/temperature. In ATHLET, the mixing in the downcomer and lower plenum is modeled with a coarse model 8 branches model.

In order to provide an assembly-wise coolant distribution at the core inlet, a mixing matrix, calculated by a CFD code on the base of the ROCOM experiment, is used. The CFD simulations were conducted at UJV Rez [5]. The resulting temperature distribution, when the minimum average inlet temperature is reached, is shown in Fig. 2

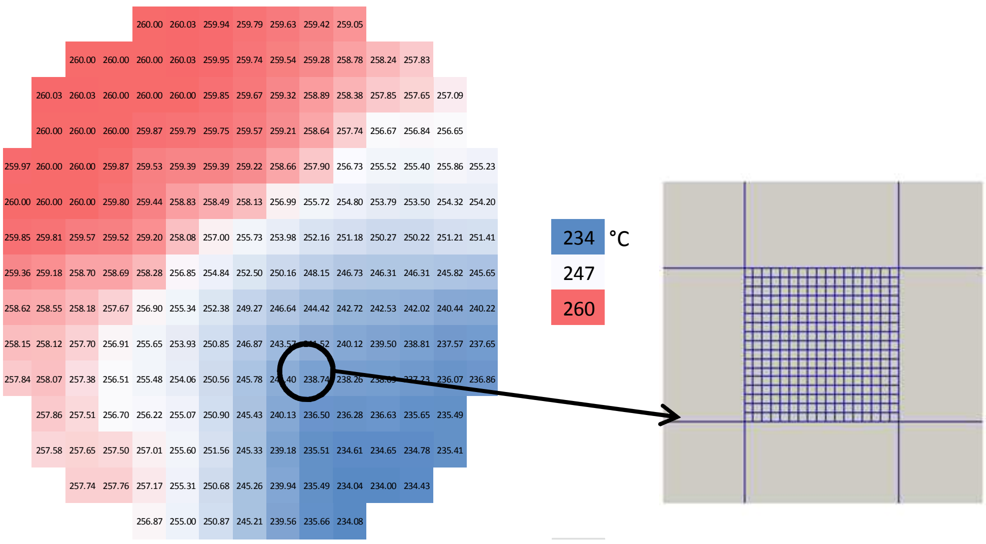

Fig. 2. CFD Simulated Mixing Matrix at T_{inlet} = min – Refined mesh at stuck control rod position

In the CTF model each assembly is modeled by a dedicated channel. The assembly with the stuck out control rod is modeled at pin level with a rod-centered CTF model. This makes for a total of 192 + 264 subchannels. The model features cross-connections between the subchannels. The CTF model is coupled either directly to the ATHLET lower plenum model or through the mixing matrix described above.

The DYN3D input deck for PWR ZION was developed at the HZDR. The core loading pattern from the UOX/MOX Core Transient Benchmark is at BOC. The generation of the EOC cross-sections was performed jointly by thee IRSN and the UPM. The pin-by-pin flux reconstruction capability is applied in the assembly with a control rod stuck out. The resulting refined mesh (in CTF and in DYN3D) around the control rod stuck out is shown in Fig.2.

Results

The power response in the ATHLET-CTF-DYN3D simulation with and without the mixing matrix is shown in Fig. 4. With the mixing matrix no prompt critical power peak is observed whereas it can be seen in the other version. Similarly the power peak is lower with the mixing matrix (352MW) than without (420 MW).

Investigations have been conducted to explain those differences. During the first phase of the transient, the overcooling is very similar with and without the mixing matrix. However, as shown in the mixing matrix (Fig. 2), the overcooling is not homogeneous in the affected quarter and the stronger overcooling does not occur near the stuck control rod but closer to the reflector.

The second phase of the transient shows two very different behaviors which are due to a threshold effect. Indeed, in the case without mixing matrix, the primary pressure reaches the SIS actuation value which is not the case of the other simulation. The SIS actuation causes the feedwater pump trip. With the feedwater pump trip the steam generator falls dry and at the end of the transient, the power level decreases down to 6MW. When no feedwater pump trip occurs, the power stabilizes at a higher level (225MW).

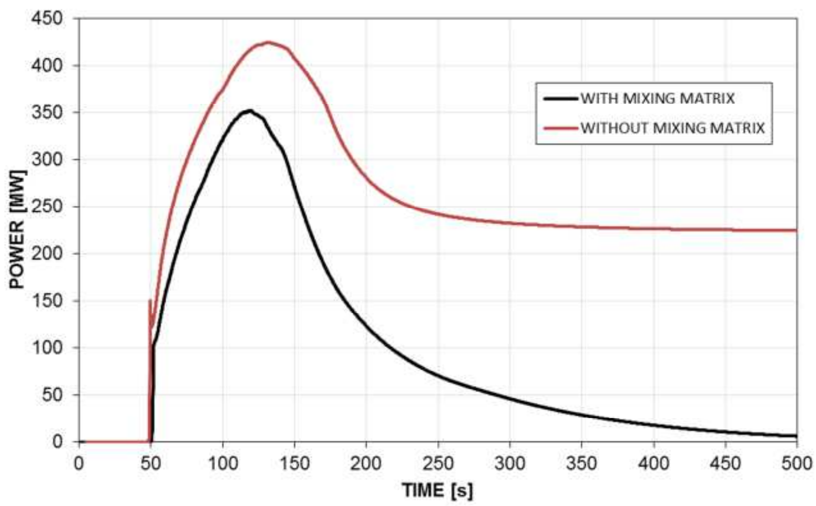

Fig. 4. Fission power history during MSLB

Fig. 5 shows the minimum DNB history during the MSLB transient. The results are extracted from the hottest pin from the assembly using a refined pin-by-pin mesh. The results confirm the behavior observed in the power history.

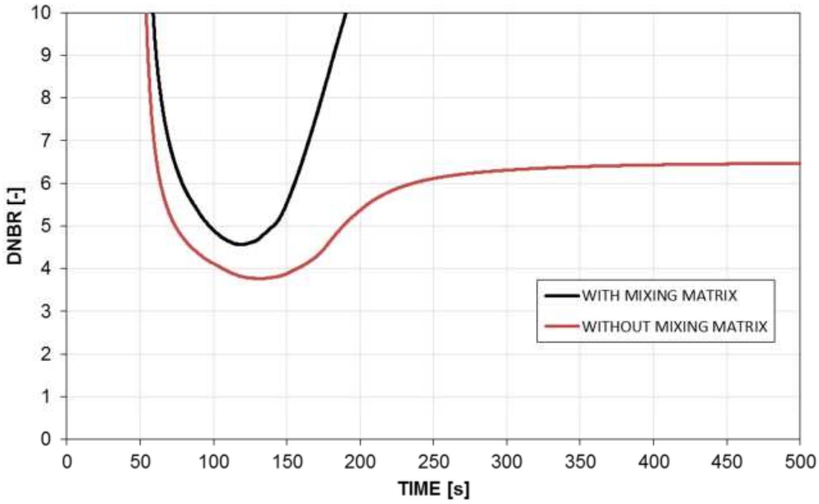

Fig. 5. Minimum DNB Ratio history during MSLB

Conclusions

The objective of this work was the development of a multi-physics, multi-scale simulation tool for LWR transient analysis. This tool should be able to simulate the whole power plant behavior while providing access to parameter at the pin level.

The PWR Main Steam Line Break Transient was successfully simulated on the Salomé platform with a three-code coupled system. The ATHLET-CTF-DYN3D simulation was performed applying additionally a CFD pre-calculated temperature mixing matrix at core inlet.

The results confirm that the mixing phenomena between the affected/unaffected loops are primordial for the simulation of the overcooling transient, as it helped to identify a threshold effect. The coarser mixing modeling in ATHLET is more conservative than the version using a mixing matrix. Although far from any safety criteria in the present case, both models give access to local (pin) value of relevant safety parameters.

Acknowledgements

This work was partly founded by the German Federal Ministry of Economics and Technology (BMWi) and by the European Commission under the 7th EURATOM Framework Program within the NURESAFE Project contract No. 323263.

References

1. B. E. A. CHANARON, “Advanced multi-physics simulation for reactor safety in the framework of the NURESAFE project,” Annals of Nuclear Energy, 84 (2015).
2. U. ROHDE, et al. “The reactor dynamics code DYN3D - models, validation and applications”, Progress in Nuclear Energy, 89, 2016
3. M. AVRAMOVA, “COBRA-TF Development, Qualification, and Application to Light Water Reactor Analysis”, Master's thesis, The Pennsylvania State University, 2003
4. G. LERCHL, et al., „ATHLET mod 3.1 cycle a - models and methods”, 2016
5. KLIEM, S. ET AL., “Testing the NURESIM platform on a PWR main steam line break benchmark,” Nuclear Engineering and Design (2017).
6. Y. PERIN and J. JIMENEZ, “Application of the best-estimate plus uncertainty approach on a BWR ATWS transient using the NURESIM European code platform,” Nuclear Engineering and Design (2017)
7. T. KOZLOWSKI and T. DOWNAR “OECD/NEA and U.S. NRC PWR MOX/UO2 Core Transient Benchmark, Final Specifications”, 2003