ICAPP 07 Proceedings- May 13-18 2007 NICE (France)
Paper 7205

HEMERA: a 3D coupled core-plant system for accidental reactor transient simulation

GB. Bruna, F. Fouquet, F. Dubois
Institut de Radioprotection et de Sûreté Nucléaire
BP 17 – 92262 Fontenay-aux-Roses Cedex - France
Tel: (33)1 58 35 72 03 , Fax: (33)158359602, Email: frederic.fouquet@irsn.fr

JC. Le Pallec, E. Richebois, E. Hourcade, C. Poinot-Salanon, E. Royer
CEA/DEN/DM2S
Centre de Saclay, 91 191 Gif sur Yvette Cedex, France
Te: (33)169082626 , Fax:(33)169089935 , Email: jc.lepallec@cea.fr

Abstract

A Nuclear Power Supply System (NPSS) is driven and regulated by several diversified and complex phenomena which are distributed in space and coupled in time in different and somewhat varying ways. In a first approximation, they depend on the reactor type, the core design and lay-out, the fuel features, the coolant, the loading strategy and cycle, the operating mode and, more generally, on the whole status of the system.

The main coupling agent in a reactor system is the temperature field inside the fuel, the core and its immediate surroundings. The temperature affects the neutron behaviour, both in normal operation and during transients, through the cross-sections, which account for the probability of neutrons to interact with matter in every zone and at any time. Thus, temperature is always relevant to normal reactor operation and control, but it may become extremely important and sometime decisive in the transients, mainly the reactivity driven ones, which are characterized by very short response-time and severe power variations.

In the framework of their collaboration to develop a system to study reactor transients in "safety-representative conditions", IRSN and CEA have launched the development of a fully coupled 3D computational chain, called HEMERA (Highly Evolutionary Methods for Extensive Reactor Analyses), based on the French SAPHYR code system, composed by APOLLO2, CRONOS2 and FLICA4 codes, and the system code CATHARE. It includes cross sections generation, steady-state, depletion and transient computation capabilities in a consistent approach. Multi-level and multi-dimensional models are developed to account for neutronics, core thermal-hydraulics, fuel thermal analysis and system thermal-hydraulics.

Currently Control Rod Ejection (RIA) and Main Steam Line Break (MSLB) accidents are investigated. The HEMERA system is presently applied to French PWR.

The present paper outlines the main physical phenomena to be accounted for in such a coupled computational chain with significant time and space effects.

A selection of results is presented along with a comparison of the available levels of simulation, ranging from 0D to 3D and from assembly-wise to pin-wise in the core.

I. INTRODUCTION

Safety accident analyses must demonstrate the respect of the safety criteria. The demonstration is performed on the most penalizing initiator.. To do this, one has to set up neutronics, thermal and thermal-hydraulics modelling to simulate normal and accidental transients. In principle, one should make the analysis for the three fields at the same time because:

- The cross-sections are dependent on the fuel temperature and the moderator density,
- The fuel temperature depends on the neutronics power and the thermal exchange with the moderator fluid,
- The thermal-hydraulic depends on the source term corresponding to the power released by convection and by γ radiation.

Up to now, in the methods used in safety reports, the three fields have been more or less decoupled. The major disadvantage of this approach is the impossibility to compute the fine power distribution of the core. Thus, power peaking factors are used. Whereas they are

evaluated in steady-state conditions, they are used for transient adding some corrections to ensure conservatism.
Incorporating full three-dimensional (3D) models of the reactor core into system transient codes enables a “best-estimate” calculation of the interactions between the core behaviour and the plant dynamics. Recent progress in computer technology has made the development of coupled thermal-hydraulic (T-H) and neutron kinetics code systems feasible.
The objectives of the HEMERA system are to perform best-estimate calculations and to develop calculation schemes for safety analysis, in association with uncertainty and sensitivity studies and penalization techniques.
The first part of this paper is dedicated to the description of the new HEMERA (Highly Evolutionary Methods for Extensive Reactor Analyses) chain, based on the French SAPHYR code system, including APOLLO2, CRONOS2 and FLICA4 codes, as well as the system code CATHARE.
The second part of the paper presents two PWR applications of the new system, so the Reactivity Insertion Accident (RIA) and the Main Steam Line Break (MSLB).
Finally, a conclusion presents the main perspectives of this work.

II. DESCRIPTION OF THE HEMERA SYSTEM

The Fig. 1 gathers a core calculation setting up the codes of the SAPHYR system, developed mainly for the PWRs[6] and BWRs[7].
For the coupling, an explicit technique which consists in solving the neutronic and thermal-hydraulic equations separately has been adopted in the system; the coupling is managed by data sharing and an iterative algorithm for convergence. This methodology has been quite easy to implement because it makes use of existing codes, nevertheless it needs external iterations and a specific tool to drive codes and manage data exchanges. The ISAS software, based on PVM, is used for this purpose[3].

It is generally agreed that for PWR multi-group calculations, the cross-section self-shielding is dependent, in a first approximation, on local conditions only, so that it can be evaluated in an assembly-wise scheme. At the opposite, a full core geometry description is necessary to enable a consistent evaluation of the reactor power and the fuel depletion. Accordingly, the calculation scheme is split into two main chained steps: firstly the evaluation of homogenized cross-sections, secondly the coupling of thermalhydraulics and neutronics. For plant transients such as MSLB, there is also a coupling between core and system thermalhydraulics.

II.A Cross-sections

The first step consists in a 2 dimension infinite medium assembly calculation in which the heterogeneities of the assembly are described as precisely as possible. The Boltzmann equation for the neutrons transport and the equations for the depletion of fuel are solved in the APOLLO2 code[4]. In this step, no coupling with thermalhydraulics is made.
The self-shielded cross-sections and isotopic densities of all the media in the fuel rods of the assembly are stored vs. burn-up in tables called “libraries”. Those tables are completed by restart calculations in which core parameters (moderator density and temperature, fuel temperature, boron concentration….) are modified separately to obtain a “multi-parameter” library for every assembly, which allows accounting for the feed-back effects through interpolation.
Each cross-section set (i.e, with a well identified set of parameters) is obtained by homogenisation on the whole assembly (the "homogeneous library") or by homogenisation pin by pin (the "heterogeneous library").
This calculation step is validated by comparison with reference calculations against the CEA Monte-Carlo code TRIPOLI4[5].

II.B Core

The core calculation is performed in 3 dimensions with the CRONOS2[1] and FLICA4[2] codes, coupled by the ISAS software.
The CRONOS2 code is used with the neutrons diffusion approximation, on homogenized assembly-type geometry, a limited number of energy groups is chosen. Typically, 4 meshes per assembly are defined and the cross sections come from the multi-parameters library.
The FLICA4 code solves the fuel thermal equation on one-dimensional geometry and the two-phase flow in 3 dimensions. The two-phase mixture is modelled by a set of four balance equations~: mass, momentum and energy of mixture, and mass of steam. The velocity disequilibrium is taken into account by a drift flux correlation. The user can choose the closure laws for wall friction, drift flux and heat transfer and the correlations for critical heat flux, depending on the fluid, the geometry and operating conditions (e.g. Pressure). The numerical method is finite volume, based on an extension of Roe's approximate Riemann solver to define convective fluxes and on the VF9 scheme to estimate the diffusive fluxes. To go forward in time, a linearized conservative implicit integrating step is used, together with a Newton iterative method.
The coupling between FLICA4 and CRONOS2 consists in: i) the power distribution calculated by CRONOS2 is transferred to FLICA4 to be used as a source term in the energy balance equation of fluid and fuel; ii) the thermal-hydraulic parameters for the evaluation of cross-sections are provided to CRONOS2 (for interpolation in the cross-

sections libraries). After around 10 iterations, this process allows obtaining a steady state for given operation conditions. The main results are the power distribution in the core, the mass flow repartition among the fuel assemblies, the fuel temperatures and the core reactivity.

II.C Plant

The primary and secondary circuits of the plant are modelled by CATHARE[8]. CATHARE is a best-estimate system code developed by CEA, EDF, FRAMATOME-ANP and IRSN for PWR safety analysis, accident management, definition of plant operating procedure and for research and development. Two-phase flows are modelled using a two-fluid six-equation model. There are several modules for 0D, 1D or 3D. In the current PWR model for MSLB, one uses 1D modules for the pipes and 0D modules for the mixing volumes. The core vessel has a channel per loop. The core is simulated by boundary conditions, since it is computed by FLICA4. CATHARE provides mass flow and temperature at core inlet and pressure at core outlet for FLICA4, while FLICA4 sends back the pressure at the core inlet and the mass flow and temperature at the core outlet. The flow mixing between loops in lower plenums and upper plenums is modelled by user-defined mixing coefficients.

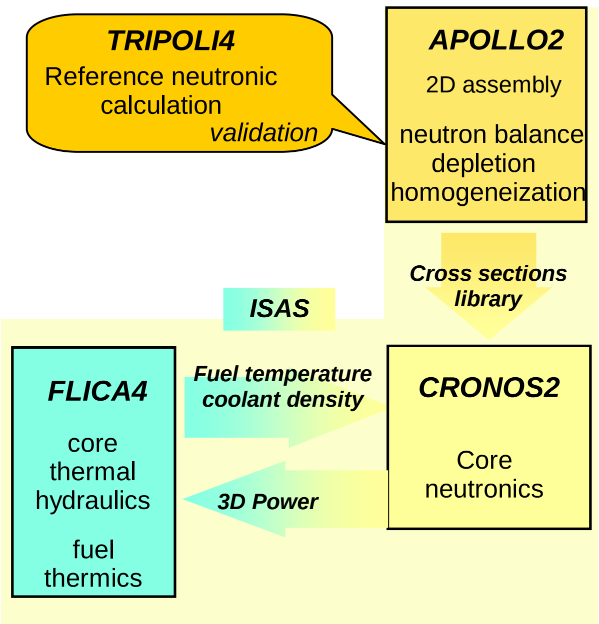


Fig. 1: Description of the neutronic/thermalhydraulic coupling in SAPHYR system

The HEMERA system permits today to simulate two accidental transients: the **Reactivity Insertion Accident (RIA)** and the **Main Steam Line Break (MSLB).** The main parameters of interest for these two accidental situations are local parameters: the power peak for RIA, and the DNBR for MSLB. In order to take into account the local effects within the fuel assembly where the control rod is not inserted, and to predict the safety parameters at the fuel rod level, a two-level calculation scheme is used for CRONOS2 and FLICA4.

For FLICA4, there are a core description at fuel assembly level (or quarter of assembly) and a hot fuel assembly description at the sub-channel level. The two levels are coupled together through hydraulic boundary conditions: mass flow, enthalpy and pressure (cf. Fig. 2).

For CRONOS2, a hybrid description of the core is used. Homogeneous cross-sections are used everywhere except in the refined assembly where heterogeneous cross-sections can be applied (cf. Fig. 3).

Feedback and neutronic power are exchanged between FLICA4 and CRONOS2, with a consistent level of discretization: coarse mesh on the core but fine mesh on the hot fuel assembly. This type of calculation scheme is very well adapted to capture the hot spot during the transient with a reasonable CPU time (possibility to distribute the system thermalhydraulics (CATHARE), core neutronics (CRONOS2), core thermalhydraulics (FLICA4) and hot assembly thermalhydraulics (FLICA4) on separate processors, and optimized discretization of the core).

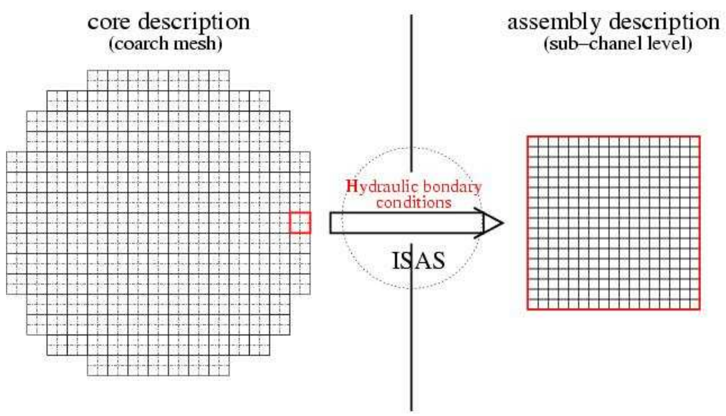


Fig. 2 – Two levels description in FLICA4 code

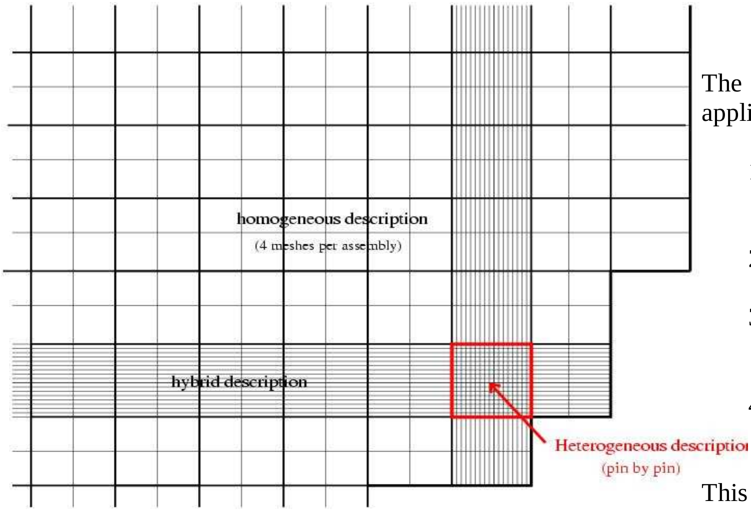


Fig. 3 – Hybrid description in CRONOS2 code

III PRESENTATION OF TWO APPLICATIONS : RIA AND MSLB TRANSIENTS

III.A RIA TRANSIENT TYPE

III.A.1 General concern

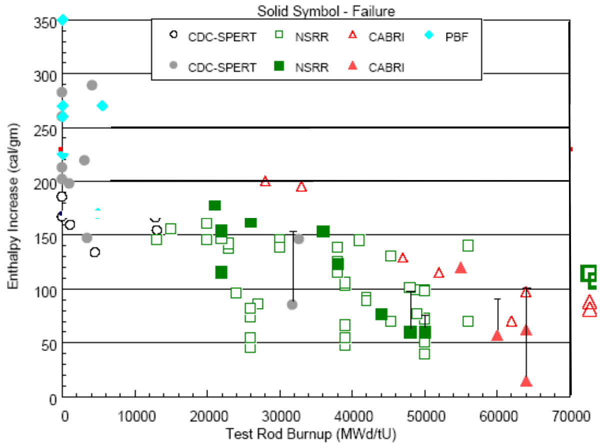
The RIA accident is generated by the ejection of a control rod, which introduces so a large amount of reactivity in the core as to render it prompt-critical and triggers a sudden and important energy release in a localised area of the core (the area surrounding the location of the ejected rod).

For high burn-up fuel managements, the methods used to calculate a rod ejection accident on a PWR rely on 3D kinetics. The former conservative methodology wouldn't permit to demonstrate fuel integrity.

Some experiments prove that in a high burn-up core, during a RIA, high burn-up fuel can fail before a less irradiated one (see Fig.4).

Some experiments prove that in a high burn-up core, during a RIA, high burn-up fuel can fail before a less irradiated one (see Fig.4).

Fig. 4: Estimated Fuel scattering threshold in a RIA as a function of fuel burnup



III.A.2 Nodalisation and boundary conditions

The following steps describe the HEMERA 2D/3D applications for a RIA safety analysis:

1. The calculation of the initial state of the core (3D static calculation in which simplified thermal and thermal-hydraulic models are adopted)
2. The research of the highest-worth control rod with a penalizing Xenon situation
3. The 3D kinetic calculation coupled with 3D thermal/thermal-hydraulic models to determine the behaviour of the core power peak versus time
4. 2D Mesh refinement for hot pin analysis (see above).

This scheme has been used at IRSN to study RIA in a 3-loop PWR loaded core. Results for a 1,26$ reactivity insertion are given bellow (Fig.5 and Fig.6).Power reaches 8 times nominal power and assembly 3D form factor reaches 7 .

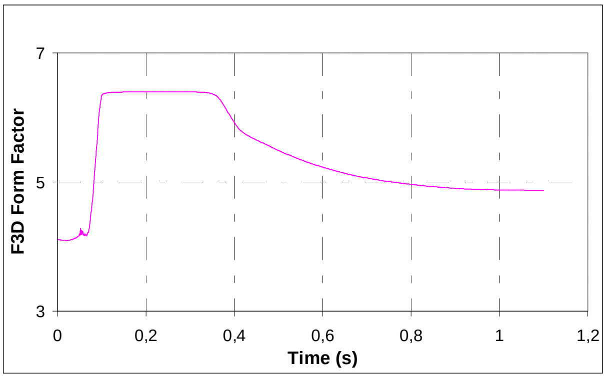


Fig. 5: form factor and reactivity as a function of time

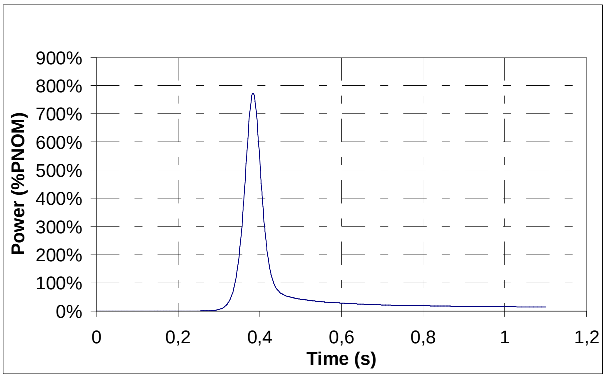


Fig. 6: Core power as a function of time

Hot pin power reconstruction was operated using the mesh refinement function of HEMERA, the pin power peaking within hot assembly was found to be 1.28. This reconstruction technique was previously benchmarked in pin by pin calculation and results were found in good agreement.

III.B SLB TRANSIENT TYPE

The HEMERA system has also been used for Main Steam Line Break (MSLB) studies and, more specifically for a four-loop French PWR transient.

III.B.1 General concern

The Main Steam Line Break is a DBA (Design Basis Accident) in PWRs, which involves coupled physical phenomena such as the thermalhydraulics of the secondary circuit, the thermal exchange between primary and secondary circuits (through the steam generator), the thermalhydraulics of the primary circuit and both the neutronic and thermalhydraulic of the core.
The steam release as a consequence of the rupture of a main steam line results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls down. The energy removal from the RCS generates a reduction of coolant temperature and pressure. The moderator coefficient being generally highly negative in such systems, the cool down leads to an insertion of positive reactivity. The core may then become critical and return to power leading eventually to the boiling crisis. This power increase is more significant when the most penalizing rod cluster control assembly is assumed stuck in its fully withdrawn position after the trip.

The MSLB is a dissymmetric accident because the loop corresponding to the break behaves differently from the others loops. The cooling of the core isn't uniform, which generates disequilibrium in the power distribution. The power peak is worsened by the stuck of a control rod.

III.B.2 Nodalisation

The nodalisation of the primary circuit (except for the core) with its 4 distinct loops and the secondary has been performed using 0D-1D elements of the CATHARE code as shown in the Fig. 7 (only two loops out of four are shown). The vessel is subdivided in four "channels", related to each loop. The core is simulated in 3 dimensions with CRONOS2 and FLICA4 codes, with 4 nodes per assembly for neutronics and 1 mesh per assembly for thermal-hydraulic calculations and 32 meshes on z-axis. A matrix derived from LACYDON-experiment results simulates the mixing between the four loop flow rates and temperatures.

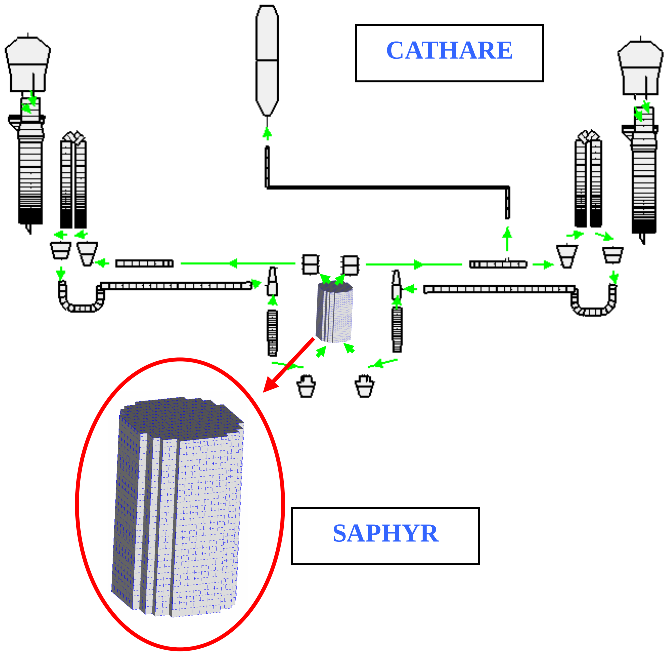


Fig. 7: Nodalisation for the MSLB simulation

III.B.3 Initial state and boundaries conditions

This analysis therefore assumes a non-isolable Main Steam Line Break at hot zero power. A small initial nuclear power showing-up penalizing, with respect to the insertion of a positive reactivity, a conservative value of 10^{-9} of nominal power is assumed. The fuel loading is UOX at the end of equilibrium cycle with no Xenon concentration. The most penalizing single failures, with regard to the DNBR (Departure from Nucleate Boiling Ratio), is a rod cluster control assembly RCCA (located in assembly-position F14 – Fig. 8) having he highest reactivity-worth, stuck in its fully withdrawn position after the reactor trip.

The initial RCS temperature and pressure are those of the hot zero power conditions (297.2°C and 155 bars). According to the end of cycle assumption, the primary boron concentration is put to zero, in order to maximize the reactivity insertion during the cool down. The initial sub-criticality considered in this analysis is of -1800 pcm (1 pcm = $1.0\ 10^{-5}\,\frac{\delta k}{k}$).

To maximize the cool down, the SIS (Safety Injection System) flow rate and SG (Steam Generator) feed water flow rate are maximized with a minimal temperature. The SI lines water is assumed at 0 ppm (1 ppm = $1.0\ 10^{-6}$) and the RWST (Refueling Water Storage Tank) concentration is assumed at 2000 ppm.

The minimum mixing within the RPV (Reactor Pressure Vessel) between loop flows relies on typical data of current 4-loop (from LACYDON tests). The minimum loop flow mixing within the RPV penalizes the core power transient. As for mixing at core inlet, it is assumed that a maximum of 65% of the flow entering through inlet nozzle remains in the associated core quadrant at core inlet.

The Reactor Coolant Pumps (RCP) are assumed not stopped.

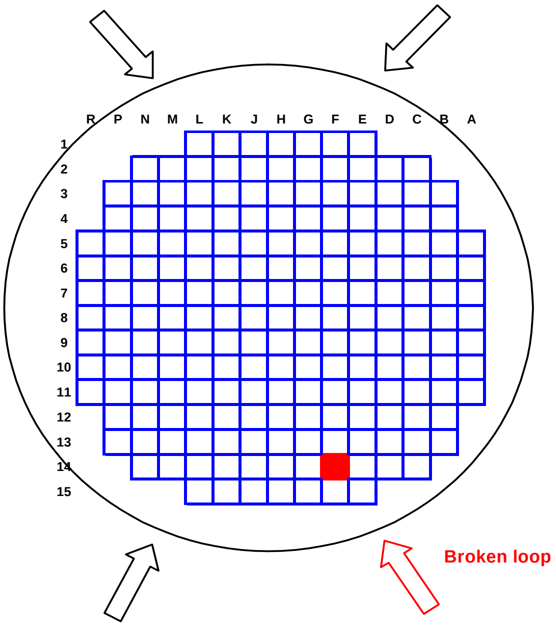


Fig.8: stuck rod and broken loop

III.B.4 Typical sequence of events

Immediately after the break initiation, the secondary system starts depressurizing. The SG pressure drop or pressure low signals actuate the Reactor Trip (RT) (if SLB – Steam Line Break - at power), drive the closure of all Main Steam Isolation Valves (MSIV), and isolation of the Main Feed Water (MFW) of the affected SG. After this isolation, only the affected steam generator, which experiences a non-isolable SLB (break inside containment, or break outside containment with failure to close of the MSIV), continues to depressurize. This SG is supplied by the Emergency Feed Water.

The energy removed from the RCS causes a reduction of coolant temperature and pressure, with actuation of Safety Injection (SI).

Due to the negative moderator coefficient, the RCS cool down results in an insertion of a positive reactivity. The reactor goes critical with a power excursion. Eventually, the Doppler Effect and the boron insertion either limit or stop the power increase.

When the affected steam generator becomes empty, the power is quickly reduced down to a level, which corresponds to the steaming of EFWS (Emergency Feed Water System) flow rate.

A stable state is reached with:

- The core just critical (i.e. reactivity equal zero),
- The core power removed via the leak and EFWS in the affected steam generator,
- A stable coolant inventory.

III.B.5 Results

Figs 9 to 11 show the behavior of the main physical parameters of the reactor system during the transient. The sequence of events is presented on the table hereafter.

Event	**Time (s)**
Main Steam Line Break	0
Lower advanced SG pressure signal	3
MSIV isolation	10
MFW isolation	10
Reactor becomes critical	16
Safety injection	20
Injection of boron in the core	75
Maximum core power is reached (5,3 NP)	145
End of simulation	300

The double-ended guillotine break of the main steam line (figure 9) leads to a quick depressurization of the secondary side and the primary side (figure 10).

The lower advanced SG pressure signal is reached at 3 seconds which drives the steam lines and MFW isolation 10 seconds later.

After MSIV closure, only the affected steam generator continues to depressurize.
The energy removed from the RCS causes a reduction of coolant temperature (figure 11). Reactor becomes critical and hence thermal power is increasing at 16 seconds (figure 12).
The Doppler Effect limits the thermal power excursion but does not stop it. The thermal power increase is limited

when the boron arrives in the core at 75 seconds (figures 13 and 14). The boron propagation in the primary via safety injection lines is in the form of a front at the beginning and leads to power oscillation in the core. The time step corresponds to the time necessary for the boron front to cover all the primary circuit. Due to diffusion in the CATHARE code (mixing), this behavior quickly disappears.

After a quick stabilization of the thermalhydraulic parameters, a stable state (core just critical with the core power removed via the leak and EFWS in the affected steam generator) is then reached. The maximum core power is 5.3 % NP reached at 145 seconds.

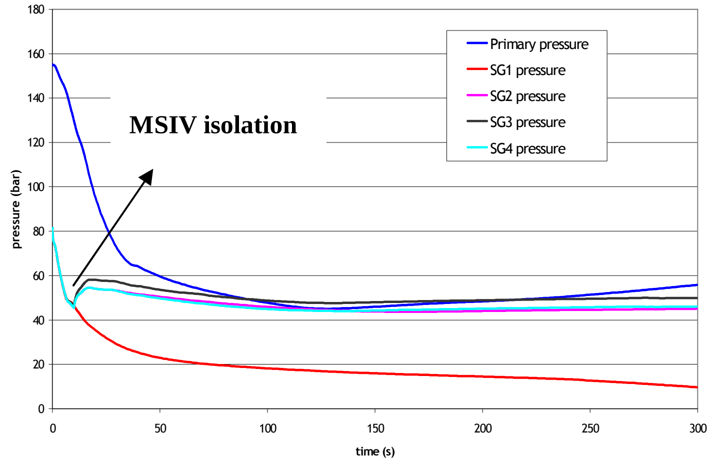


Fig.10: Primary and secondary pressure

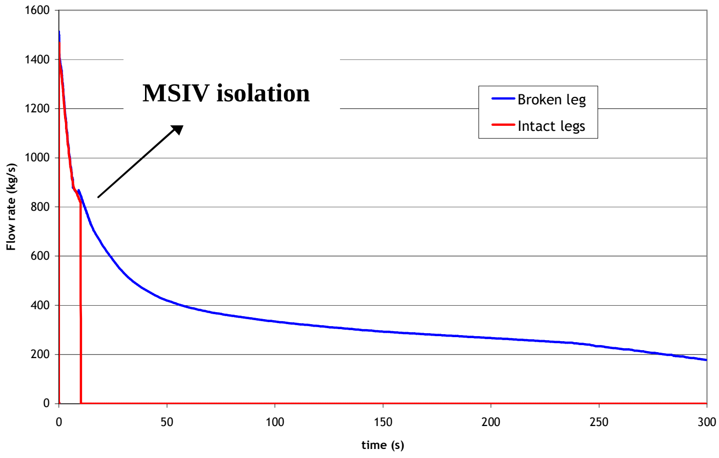


Fig.9: Break flow rate

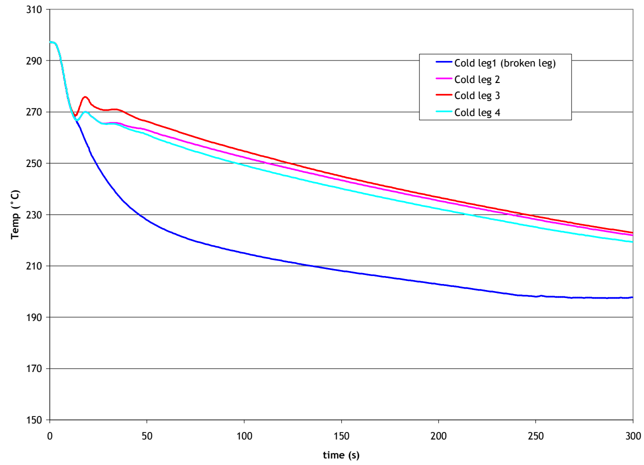


Fig.11: Primary cold leg temperature

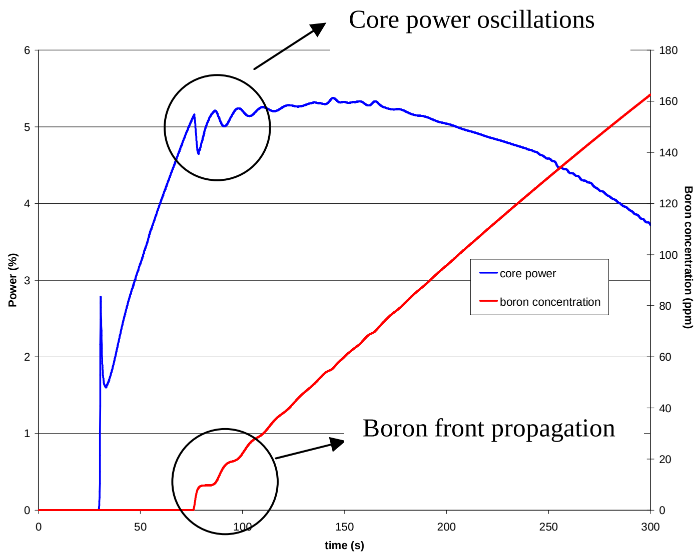


Fig. 12: Core power (blue) and boron concentration (red)

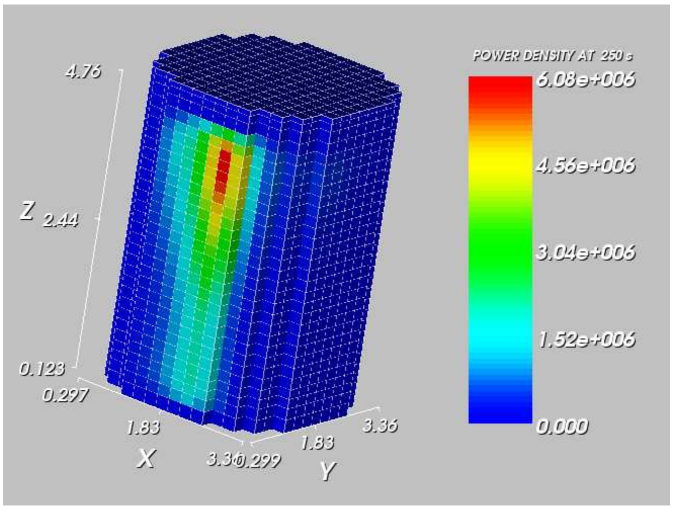


Fig. 13: Power density at 250 s

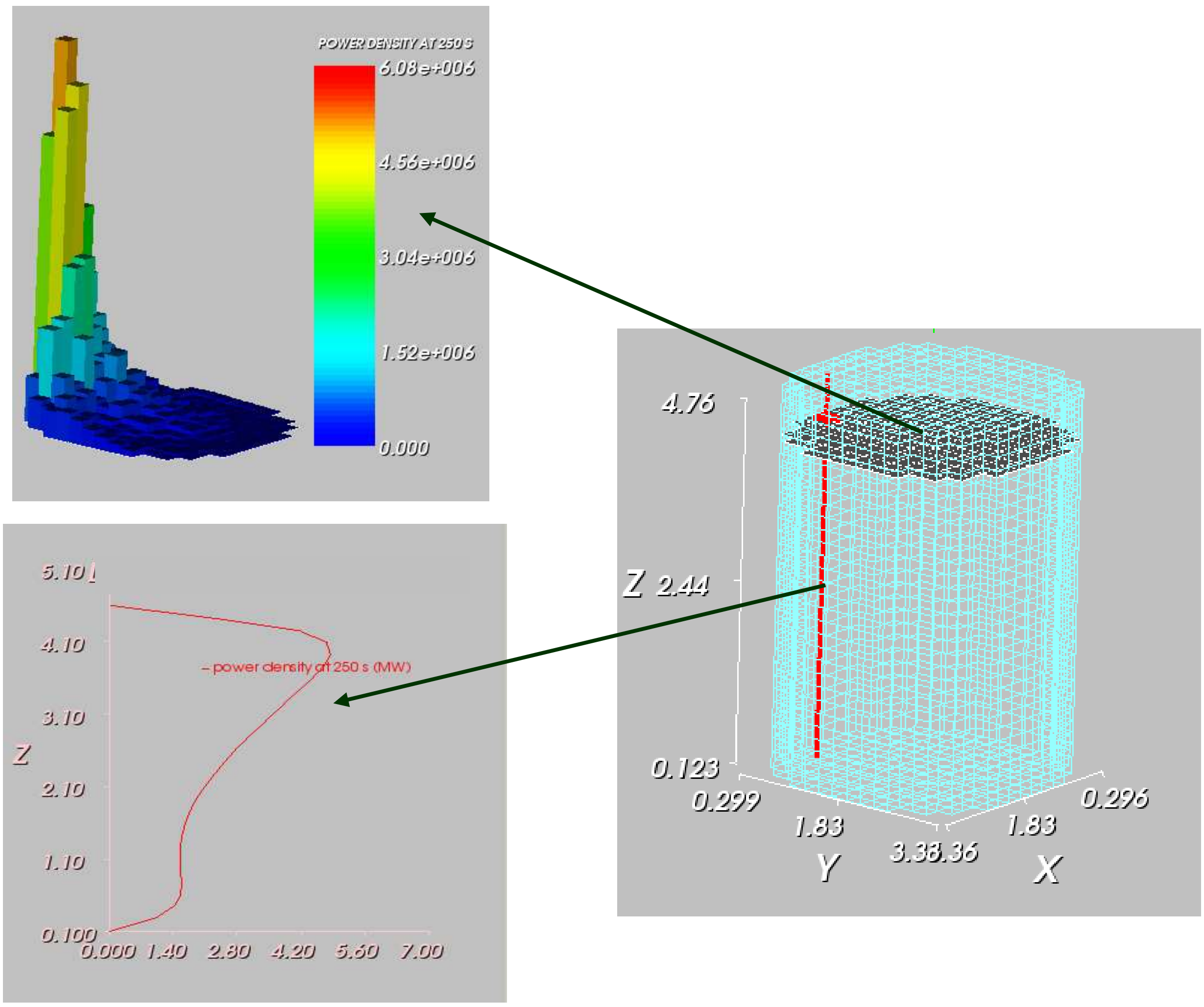


Figure 14: Power density distribution at 250 s (in a x-y plan and in the assembly F14)

Figs 13 and 14 gather that the maximum power density is located at the top of core (lower burnup with quite high density) and in the assembly F15, assembly from first cycle and near the stuck rod.

Another simulation was performed without boron injection (Fig. 15). In this case, after steam generator draining, the thermal power decreases, the power reaches 12.5 % NP at 250 seconds. The comparison of these two calculations shows the importance of boron effect on the thermal transient.

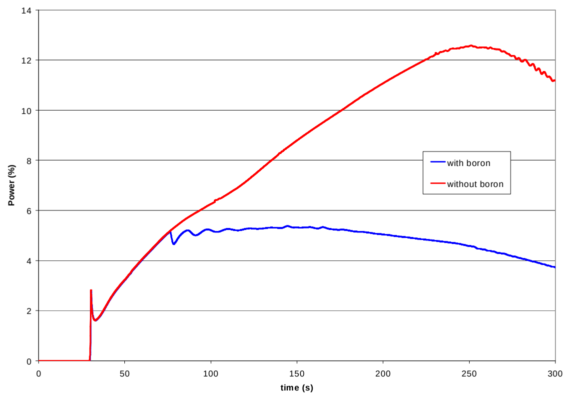


Fig. 15: Core power (blue line: with boron; red line: without boron)

IV CONCLUSION AND PERSPECTIVES

The current scope of neutronics and thermal-hydraulics coupling enables perform to best-estimate calculations for PWR safety analysis, in association with uncertainty and sensitivity studies. Moreover, development of suitable penalization techniques is underway.
For this purpose CEA and IRSN are developing the HEMERA coupled neutronics and thermal-hydraulics computational chain, based on CATHARE, CRONOS2, FLICA4 and APOLLO2. HEMERA is now used by IRSN for PWR safety assessment with application to two accidental transients: MainSteam Line Break, involving the coupling between core and system, and Reactivity Insertion Accident.

Taking advantage from the current experience, several main axis of improvement have already been identified and stressed, such as:

- Necessity to use the best available models in the different physics inside the coupled system (neutronics, thermal-hydraulics…),
- Accounting for the impact of the thermal-mechanics of the fuel on the thermal feed-back,
- Continuous validation of the coupled system with international benchmarks, if possible with actual plant data (e.g. Peach Bottom, Kozloduy…),
- Introduction of methods for uncertainties evaluation (Design of experiments, response surfaces, method of penalization…),
- New coupling techniques, including interpolation and unified data structures, definition and share of common data between coupled models, supervision of calculations,
- Necessity to easy perform sensitivity analysis.

Those improvements either are underway or will be addressed in a near future.
Among the new features already planned for HEMERA, we can mention improvements coming from coupling with a code for fuel integrity analysis; SALOME will replace ISAS for easier supervision of calculations. For mid term, time-step management for complex coupled transient will be introduced, sensitivity matrix could be built and used on analysis, and, in order to extend the scope of the code system, we could add refinements some physical models whose accuracy is too weak. In the long term, we want the multi-scale capabilities will be enhanced and benefits from new solvers developed within the future DESCARTES and NEPTUNE platforms, respectively for neutronics and thermal-hydraulics, will be available.

REFERENCES

1. J.J. LAUTARD, S. LOUBIERE, C. MAGNAUD « CRONOS, A Modular Computational System for Neutronic Core Calculations », *IAEA Specialists Meeting, France,* September 1990.

2. I.TOUMI and al, “FLICA4: a three dimensional two-phase flow computer code with advanced numerical methods for nuclear applications » *Nuclear Engineering and Design, 200* (2000*) 139-155*

3. I.TOUMI and al, “Specifications of the General Software Architecture for code Integration in ISAS” *Euratom Fusion Technology, ITER Task S81TT-01, 1995*

4. R. SANCHEZ, A. HEBERT, Z. STANKOVSKI, M. COSTE, S. LOUBIERE, C VAN DER GUCHT and I.ZMIJAREVIC, « APOLLO2 Twelve Years Later », *Mathematics and Computation, Reactor Physics and Environmental Analysis in Nuclear Applications - Madrid* - (September 99)

5. J.P. BOTH, Y. PENELIAU, “The Monte-Carlo code TRIPOLI4 and its first benchmark interpretations”, *International Conference PHYSOR 96, Mito, Ibaraki, Japan*, September 1996

6. E.Royer, E.Raimond, D.Caruge, "CEA-IRSN participation in the MSLB benchmark", *Nuclear technology, Vol.142*, p.154-165, May 2003

7. G. Mignot, E. Royer, B. Rameau, N. Todorova, "Computation of a BWR turbine Trip with CATHARE-CRONOS2-FLICA4 coupled codes, *Nuclear Science and Engineering, Vol 148*, p;235-246, October 2004

8. G. Geffraye, I. Dor, G. Lavialle, « Recent improvements of physical models in the CATHARE code and their validation", *NURETH 10 conference*, October 5-9, 2003, Seoul, Korea