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Neutron Kinetics Modelling for Simulations of Loss of Coolant Accidents in The Nuclear Power Plants

Bartlomiej Klis, Simon Primeau, Patrick Blaise, Jean-Christophe Lecoy, Marie-Christine Grouhel

Summary **— The code used by Framatome to predict the progression of a LOCA is the system scale thermal-hydraulic code CATHARE (Code Avancé de THermoHydraulique pour les Accidents deRéacteurs à Eau). It calculates the full primary and secondary circuits including the core and the fuel elements. CATHARE currently utilizes a 0D neutronic model that solves the PointKinetics Equation (PKE) to determine the evolution of instantaneous fission power. This approach is suitable for transients where the core's moderator density changes rapidly in a quasi-uniform manner, such as in large break LOCA scenarios (from the initial situation with a core full of liquid to a sudden and complete voiding leading to a full vapour environment). However, during Intermediate Break (IB) scenarios, with slower dynamics, the assumption of a uniform core's moderator density is no longer valid. This assumption results in a significant underestimation of void antireactivity in the upper part of the core and a slight overestimation at the bottom. Thus, using PKE for IB-LOCA leads to an overestimation of the fission power, and the more heterogeneous the core, the higher the conservatism of this hypothesis is expected. In fact, the specific application of IB-LOCA involves a precise neutronic calculation in a strongly diphasic fluid environment which is first due to the uncovering of the core and then to its reflooding by the safety injections. The presented work relies on the development of a fine 3D coupling between neutronics and thermal-hydraulics at the assembly scale plunged into a fullreactor simulation. Such a development goes beyond the known limitations of current neutronics/thermal-hydraulics couplings (dealing with low void fraction situations) which are notsuitable for LOCA safety studies.**

Keywords **— nuclear safety, thermo-hydraulics, neutronics, code coupling**

I. Introduction

In this work, authors present the neutron kinetics model commonly used for the loss of coolant accidents (LOCA) safety studies by system scale thermal-hydraulics (TH) codes. We hereby present the results of calculations done by CATHARE 3 with full 3D core modelling at the assembly scale of a 3-loop pressurized water reactor (PWR) for Intermediate Break LOCA (IB LOCA). The main observation we would like to emphasize is that many other studies involving system scale TH codes use the point kinetics equation (PKE) for prediction of fission power evolution during the LOCA transients. Keeping in mind this approach is well penalized and validated to be conservative in variety of safety studies, we will challenge 0D neutronic modelling and justify the need for new developments of multi-physics methods for LOCA safety studies.

More advanced neutron kinetics models exist than PKE, which are either the result of deterministic neutron transport codes solving Boltzmann equation or Monte Carlo codes using large statistics of tracing down each individual particle interactions. Some authors in their studies, often focused on operational conditions and non-LOCA accidents, use tools which involve thermal-hydraulics/neutronics (TH-N) loose couplings (also called black box couplings) to incorporate better neutron kinetics modelling with possibility to model 3D power distribution changes in the core. An example is a study done with RELAP5/PARCS by Peakman et al. [1]. Some other researchers studied direct TH-N coupling methods using for example the Newton-Krylov approach like Zhang et al. [2]. Unfortunately, they are not yet implemented in the form of mature codes and methods for the nuclear industry. However, based on bibliography, we spotted a gap in studies related to loss of coolant accidents (LOCA), which we think are worth investigating using high-fidelity methods involving the TH-N couplings. This was also identified in Wang et al. work from 2019 where the author presented a thorough review of multiple TH-N couplings [3]. It seems nontrivial that any recent code coupling described so far be specifically designed to address the physics of LOCAs for PWRs. An example can be the recent work of Peakman et al. [4]. A recent review on TH and neutronics fuel screening methods made by Gorton et al. from Oak Ridge in 2023 [5] mentioned the limitations of PKE in the system scale code RELAP5 (Reactor Excursion and Leak Analysis Program) and a multi-physics coupling was pointed to solve these issues. Similar conclusions were pointed by P. Ruyer from IRSN (nowadays ASNR) [6]. Lastly, the need for developments of multi-physics tools at system scale has also recently been discussed during the OECD/NEA/CSNI Specialists Meeting on Transient Thermal-hydraulics [7].

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CATHARE (Code for Analysis of Thermal-Hydraulics during Accident of Reactor and safety Evaluation) is the system scale TH code used in Framatome to predict the transient simulation of LOCA [8]. Currently, CATHARE utilizes a 0D neutronic model that solves PKE to determine the evolution of instantaneous fission power.

II. Point Kinetics Equation

The neutron point kinetics is described by a system of ordinary differential equations composed of one equation for neutron population and N_d-equations for delayed neutron precursor groups. Many authors use six precursor groups as initiated by the study of G. R. Keepin et al. [9].

Neutron population is described by the following equation:

$$\frac{d}{dt}n(t) = \frac{\rho(t) - \beta}{\Lambda}n(t) + \sum_{i=1}^{Nd} \lambda_i\, C_i(t) \quad (1)$$

Whereas the N_d equations for delayed neutron precursors are shown below:

$$\frac{d}{dt}C_i(t) = \frac{\beta_i}{\Lambda}n(t) - \lambda_i C_i(t) \quad (2)$$

Where: $n(t)$ – number of neutrons, $\rho(t)$ – reactivity, β – effective delayed neutron fraction, β_i - delayed neutron fraction of i-th precursor group, Λ – average neutron generation time, λ_i - decay constant for i-th delayed neutron precursor group, $C_i(t)$ - concentration of i-th delayed neutron precursor group.

By solving this system of equations, one can predict fission power evolution for a given temporal reactivity input. The global reactivity calculation is based on TH data from transient and includes different reactivity feedbacks.

III. Methods

The following study was done using the CATHARE 3 code. For sake of the demonstration, we implemented the complete numerical input of 3-loop PWR nuclear power plant (NPP) with the 3D cartesian core modelling. This level of fidelity was enabled in CATHARE 3 because of recent developments made by Prea et al. from CEA [10]. Thus, our input deck simulates the whole primary and secondary circuits, as well as key events for the IB LOCA transient. The new 3D core model can simulate local effects such as gas and liquid crossflows, departure from nucleate boiling due critical heat flux, as well as thermomechanical effects such as cladding ballooning and worsening of heat transfer in the fuel due to burnup fatigue. Figure 1 presents the 3D core model used for this study. The core bypass is modelled via one more axial element connecting the lower and upper plenum objects. Figure 2 reproduces the penalized core map. Three concentric rings, each using a different average linear heat generation rate (LHGR) for the rods were used. This allows to generate a bounding TH environment for the *hot rod*, which is the fuel rod with the highest LHGR located in centre of core model. Thus, the *hot rod* reaches the highest possible cladding temperatures and is used to evaluate safety criterions. Total power of the core was slightly higher than maximal nominal value. The central ring (red) concentrates the assemblies with rods having high LHGR and contains the *hot rod*, next ring (yellow) represents assemblies of mean core, and finally assemblies in outer ring (green) have the decreased LHGR to balance the total core power. In this cartesian core modelling, some corner regions are inactive (grey), no TH connection is allowed with these nodes and no calculations are done inside them.

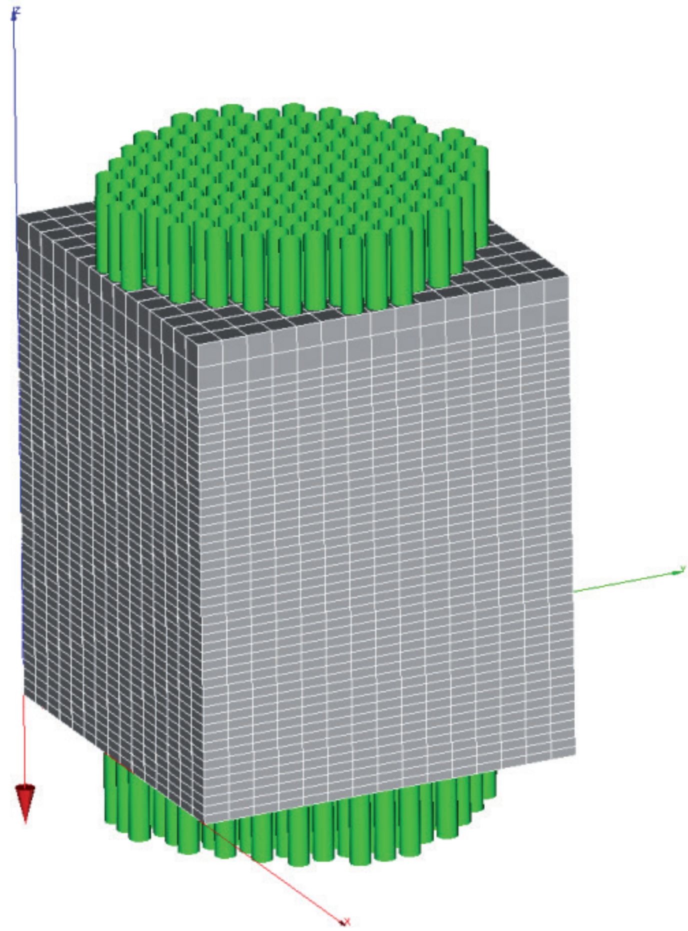

Fig. 1. CATHARE 3D core model

Additionally, the 0D neutronics model using the PKE was utilized to predict the fission power evolution. This allows the validation of CATHARE 3 for simulation of the core pre-accident phase and the transient after the break opening. The corresponding input data for PKE model is a curve of antireactivity along moderator mean core density computed for defined core geometry, fuel loading, burn-up and initial conditions. It is computed separately using assembly lattice deterministic neutronic codes like APOLLO. These calculations result in moderator antireactivity expressed by non-linear formula as a function of average density, which is referred as the *voiding curve*. Additionally, among other neutronic effects, the Doppler feedback law was included.

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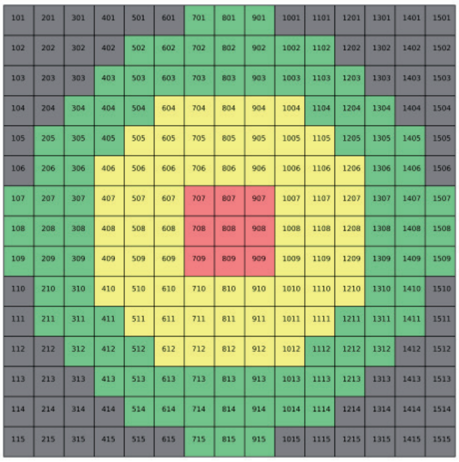

Fig. 2. Core fuel map

IV. Results

A. Description of the IB LOCA Transient

The IB LOCA transient starts when a breach opens in one of the primary coolant loops at full core power. This results in depressurisation of the primary circuit and start of fluid loss from the system. This period is called blowdown phase. Meanwhile, critical heat flux for some of the fuel elements is reached and leads to departure from nucleate boiling regime in their proximity. This causes occurrence of first peak cladding temperature (PCT1) for *hot rod* within the first few seconds of the transient. As the core fission power is going down mostly due moderator antireactivity feedback and SCRAM, the cladding temperature decreases. However, because of decreased pressure and lack of liquid in the core, some claddings are not able to achieve the wet contact with the fluid again. Inside the core, the steam produced does not allow to maintain cooling via heat exchange with fuel rod claddings. The several dozen seconds later, a second peak cladding temperature (PCT2) for *hot rod* occurs when the core is almost empty, this is core uncovery phase, and claddings are heating up adiabatically due to the decay heat. For PCT2 value, the fission energy released in the fuel from the beginning of the transient impacts the adiabatic heating, defining the initial temperature from which this second rise starts. The ultimate and last phase of the transient is referred as core reflooding, when water from the accumulators rushes into the core, rapidly cooling down the claddings and restoring the wet contact with liquid, which improves the heat transfer. From this moment, the active safety systems sustain the cooling of the core, preventing the more severe consequences.

B. Selected Data from the IB LOCA Transient

Some of the IB LOCA transient data are presented below. Due sensitivity of some information presented here, the data was normalized, but main transient features necessary to explain the subject are valid. The normalized PCT evolution during IB LOCA for *hot rod* is presented on the Figure 3. Similarly, the evolution of average fluid density in the core is presented on Figure 4.

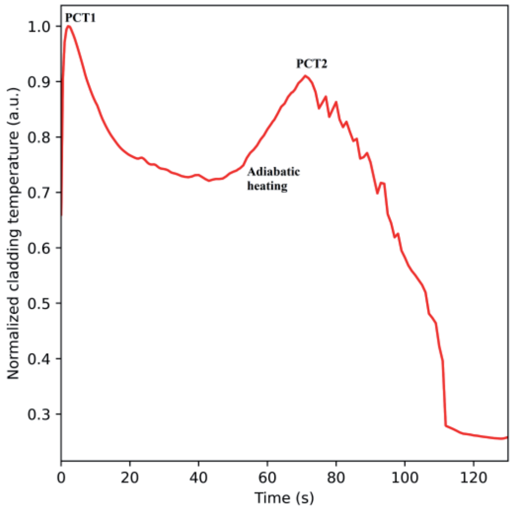

Fig. 3. Normalized maximal cladding temperature evolution with two peaks: PCT1 and PCT2

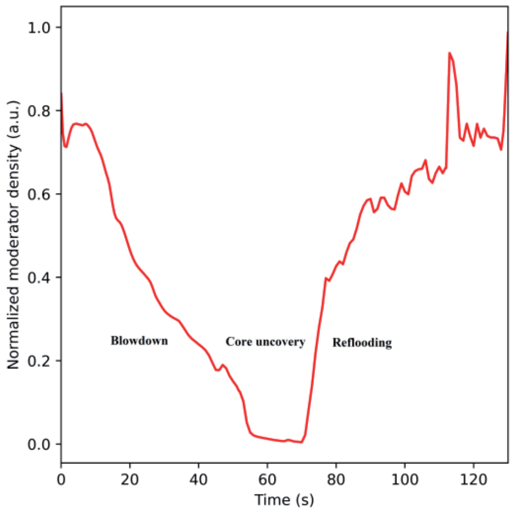

Fig. 4. Normalized average moderator density evolution in the core

We want to focus on the fact that fission power is driven by neutronic feedbacks responding to voiding of the core and fuel temperature evolutions. This power is a result of 0D neutron kinetics model of CATHARE 3. The global fission power evolution is presented on Figure 5. Moreover, the residual and decay power are included in the simulation as additional laws, not presented hereby. In CATHARE, the average fluid density in the core is used to get the moderator antireactivity, based on highly penalized voiding curve law. This feedback is presented on Figure 6. Additionally, the Doppler feedback is modelled via average effective temperature, as presented on Figure 7. However, the Doppler in LOCA is in fact strongly spatially dependent and brings positive reactivity feedback when the fuel cools

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down, which may result in neutronic instabilities at the beginning of transient. The average density profile in the core after 1 second of breach opening is presented on Figure 8. Please note that voiding is concentrated locally near the hottest assemblies.

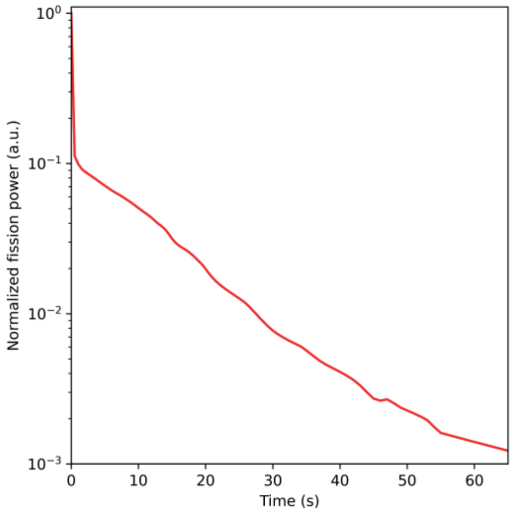

Fig. 5. Normalized global fission power evolution

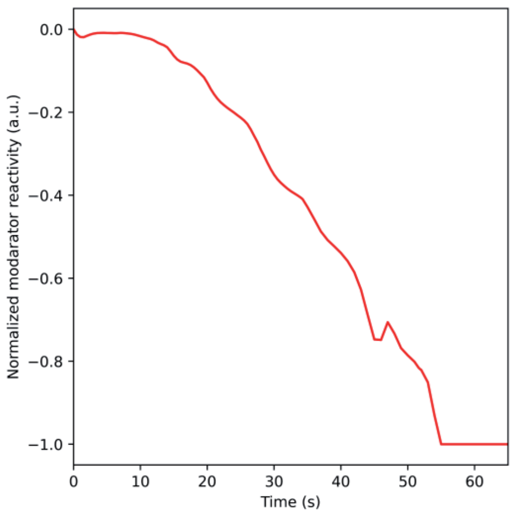

Fig. 6. Normalized moderator reactivity feedback

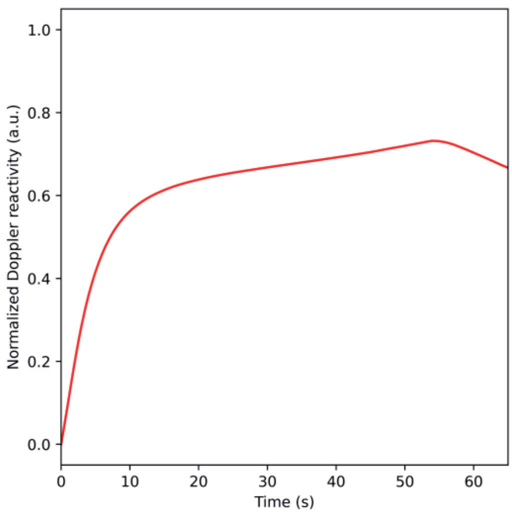

Fig. 7. Normalized Doppler reactivity feedback

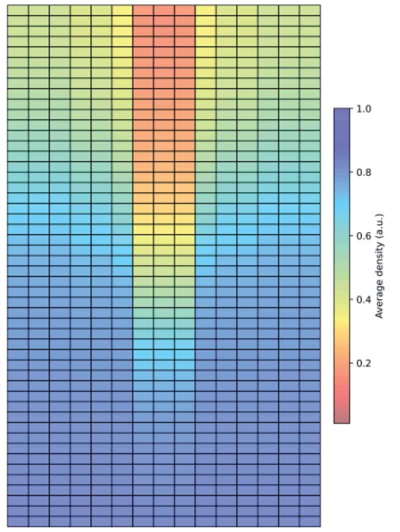

Fig. 8. Normalized average moderator density profile in the core after 1 s of transient

C. Challenging the PKE neutron kinetics

Using new 3D core capacity, we see that density is not changing uniformly in the core for IB LOCA transients, but it is strongly correlated with places where boiling occurs as presented on Figure 8. The other places in the core, mostly near its bottom have similar TH conditions as at the pre-accident phase. Thus, in a local sense, the environment for neutrons has not yet changed there, which may be under-conservative to assume that antireactivity feedback influences these regions. Keeping in mind, the PKE uses penalized law for moderator antireactivity and power is lower at these axial positions. Contrary, the top part of the core and the hottest assemblies present strong void fraction, and fuel is hotter than in any other places in the core. If their power change with the global kinetics, it is clearly an over-conservative behaviour. Moreover, time taken to void the core is much longer than in Large Break LOCA (LB LOCA). These facts are justifying the research on other high fidelity neutron kinetics models for system scale TH codes. We like to mention that in the LB LOCA simulations we observe strong voiding effects, leading to quasi-uniform rapid decrease in moderator fluid density. Such scenarios justify the use of the PKE model, since core voiding brings fast and large antireactivity feedback, stopping fission reaction within a few seconds.

V. Conclusions

The neutron PKE is used in many of the current system scale TH codes to predict fission power for the LOCA transients. It is well justified in the case of rapid uniform voiding of the core, since resulting antireactivity can be well modelled via global approach. However, the slower core voiding with strong fluid heterogeneities common for the IB LOCA transients are challenging these neutronic assumptions. Thus, the investigation of multi- physics phenomena in the IB LOCA transients are of high importance for the development of new high-fidelity tools for the LOCA safety studies. In this work the authors presented preliminary results of IB LOCA transient on a stylized full 3D core modelling with CATHARE 3 and PKE 0D neutronics to justify a roadmap for the new developments. We plan to continue our research by developing and validating a more advanced multi-dimensional neutron kinetics model for CATHARE 3, that could furthermore be coupled with a 3D Neutronics core calculation. The main objective of our upcoming research is then to perform TH-N 3D/3D coupling for IB LOCA, thus improving current approaches such as *voiding curve* methodologies, and more accurate flow dissymmetry during the transient.

VI. Acknowledgments

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