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CFD/Monte-Carlo neutron transport coupling scheme, application to TRIGA reactor

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ABSTRACT

A new computational model of the JSI TRIGA Mark II, coupling Monte Carlo neutron transport code TRIPOLI and fluid dynamics code CFX was built and verified with a set of new experimental data. A set of subroutines was developed to allow the communication between the Monte-Carlo transport code and CFD code. First, test of the coupling scheme is presented: for a given thermal power of the reactor, the coupled model numerically reproduced fuel temperature monitored during reactor operation and axial water temperature profile measured in the coolant channels. Then axial temperature profiles in the coolant channels were measured with a newly developed sensor during steady-state operation. Predictions of the coupled model are in expected agreement with experimental data recorded during reactor operations. Influence of the coupling has been investigated.

1. Introduction

Nuclear engineering is a complex crossover of several disciplines such as material science, chemistry, thermodynamics and particle physics. Each discipline describes the evolutions of physical quantities through set of mathematical equations. At the early days of nuclear engineering, slower computers required simplified models. Modelling assumptions were numerous (for example: one dimensional problem for thermal-hydraulics, point kinetic for neutronics...). Great improvement in computational power allows development of more accurate models able to describe interactions amongst nuclear, fluid, thermal chemical and structural behaviour of a nuclear reactor.

Multi-physics and multi-scale modelling is a particularly challenging task for the future (Ivanov and Avramova, 2007). More precisely, the coupled phenomena occurring in the core between neutronics and thermal-hydraulics are known as reactivity or thermal feedback. In water cooled reactor, thermal feedback and temperature coefficients of reactivity are important: a temperature variation impacts the neutron spectrum and the neutron flux profile. Those effects can have a significant impact on core's performance and inherent safety of the reactor.

Those reasons were naturally leading in the development of coupling methods in order to correctly predict the behaviour of the core in normal operation and accidental situation (Ivanov et al., 2013; Mylonakis et al., 2014; Zerkak et al., 2015). There are two ways to perform the coupling, the first one is called internal coupling in which all equations are solved simultaneously. Mahadevan (Mahadevan et al., 2012) have shown that this way of coupling provides better convergence, nevertheless, the number of equation to solve could be excessively large therefore several numerical challenges remain to be resolved. The coupling can also be external: a specialized code is used for each area of physics and the coupling is achieved with communication of the data between the codes (Vazquez et al., 2012). From an engineering point of view, this method allows the use of "state of the art" code thoughtfully verified and validated for each discipline. Downsides are the development of communication interfaces that are efficient on massive-parallel computer systems, accurate and stable (Kotlyar and Shwageraus, 2014). This last approach has been retained in the present paper.

The choice have been taken to couple a Monte-Carlo (MC) neutron transport code TRIPOLI (TRIPOLI-4 Project Team, 2013), with a computational fluid dynamic (CFD) code, Ansys CFX (ANSYS, 2011).

MC codes are gaining in popularity over deterministic codes, indeed, they enable the use of continuous nuclear data libraries and allow detailed modelling of complex geometry with minimum approximations (Wu and Kozlowski, 2015). Recently, MC codes were coupled with sub-channels code, MCNP5 and SUBCHANFLOW were used to predict the pin-power distribution of a PWR fuel assembly (Ivanov et al., 2013). Additionally, MCNP was coupled to various sub channel codes (Richard et al., 2015). SERPENT and DYNSUB were used to model a reactivity insertion accident

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(Knebel et al., 2016). Another example is the coupling between Serpent 2 and SUBCHANFLOW to simulate a full PWR under hot full power conditions (Daeubler et al., 2015).

Recently numerous coupling studies have been done with CFD and deterministic neutron transport codes to take advantage of their capabilities to reproduce 3 dimensional effects (Scheuerer et al., 2005). The coupling DeCART/STAR-CD was carried out to perform high fidelity simulations of boiling and pressurized water reactor (Weber et al., 2007). Ansys CFX and DYN3D were coupled to improve the steady-state description of PWR as well as few transient scenarios such as control rod insertion (Grahn et al., 2015).

Nevertheless, coupled studies based on MC and CFD are limited: both methods demand a lot of computational power. MCNP5/STAR-CD coupling was done to simulate a 3-D 3 by 3 array of PWR fuel pins (Seker et al., 2007). Another example are the coupled MCNP/Fluent simulations performed at University of Illinois to provide a high fidelity multi-physics simulation tools for analysis of the steady-state Pressurized Water Reactor core (Hu and Rizwan, 2008). More recently, steady-state of Pebble Bed-Advanced High Temperature reactor has been investigated with a coupled system RMC/CFX (Li et al., 2012). Super critical water reactor was also investigated with an external coupled system MCNP/CFX (Xi et al., 2013).

Fluid flow in the pool and in the core of the TRIGA reactor exhibits strong 3D features. For example, some of our simulations point to the possibility of internal recirculation inside the core: fluid is rising in the centre of the core and is flowing downward in some of the empty channels at the core periphery. This kind of phenomena justifies the implementation of the CFD approach. Another argument for using a CFD rather than subchannel codes for thermal-hydraulic analyses, is relatively simple geometry of the TRIGA reactor core. Roughly 40 cylindrical fuel elements and control rods can be relatively easily modelled with CFD. The same task is much more difficult in commercial reactors with several ten thousand fuel rods.

The purpose of this paper is to define a reference benchmark case for coupled thermal-hydraulic and neutronic calculations in a relatively simple geometry of the TRIGA reactor. The calculations are compared against the new experiments specially designed for the validation of the coupled model. After an overview of the TRIGA reactor, the thermal-hydraulic as well as the neutronic model are presented. The third part describes the coupling scheme. In order to validate the model a measurement campaign was performed to collect water temperature profile; presentation of the experimental protocol is given in the fourth part. Finally, the calculations are compared to the experiments to serve as a validation of the model.

2. TRIGA reactor

TRIGA® (Training Research Isotope General Atomics) is a pool-type nuclear research reactor manufactured by General Atomics. The reactor fuel is uranium zirconium hydride (U-ZrH) fuel. Fuel elements are cooled by demineralised water that flows through the reactor core by natural convection.

The TRIGA MARK II research reactor at the Jožef Stefan institute (JSI) is a typical 250 kW TRIGA reactor, which is used for various applications: such as neutron activation analysis, neutron radiography and tomography, education and training, radiation hardness studies and benchmark experiments for verification and validation of computer codes (Radulovic et al., 2014; Ravnik and Jeraj, 2003; Snoj et al., 2011; Žerovnik et al., 2015). During the long time reactor operation, an external cooling system is operating, which cools the upper section of the pool through a forced convection; however, natural convection remains the driving force for the flow through the core.

The core of TRIGA reactor is placed at the bottom of an open tank (atmospheric pressure) with 5 m of water column above it (Fig. 1). The core has a cylindrical configuration with 91 designed locations to accommodate fuel elements or other components such as control rods, a neutron source and irradiation channels. Elements are arranged in six concentric rings: A, B, C, D, E and F, each having 1, 6, 12, 18, 24 and 30 locations, respectively.

A graphite reflector enclosed in aluminium casing surrounds the core. An annular groove in the upper part of the reflector body is provided to contain a special irradiation facility. Detailed description (dimension, material composition...) of the reflector and every other component can be found in the paper of Ravnik and Jeraj (Ravnik and Jeraj, 2003).

The TRIGA reactor fuel is a homogeneous mixture of enriched uranium and zirconium hydride (U-ZrH) it is clad by stainless steel (SS304). The enrichment and content of uranium in U-ZrH depends on the fuel type (ICSBEP, 2013; Snoj and Ravnik, 2008) The main feature of such composition is that the fuel is homogenously mixed with moderator in the form of U-ZrH causing a large fraction of neutrons to be moderated in the fuel itself. Therefore one should pay special attention to the thermal scattering cross-sections, especially for H and Zr bound in ZrH as well as elastic scattering cross-sections for Zr (see discussion in Section 4.1, Step 4 of the algorithm) (Snoj et al., 2012).

3. Computational models

3.1. Neutron physics

The neutron transport simulations were performed with the Monte Carlo neutron transport code TRIPOLI. The geometry of the computational model is presented in Fig. 2. Fuel elements (red) and control rods (grey) are taken into account with full available precision, meaning that Zr rod, stainless steel cladding, air gaps and Mo supporting disc are explicitly resolved. The supporting grid, graphite reflector (in black) with rotary groove (cyan) and central irradiation channel (white) in the core are also explicitly resolved. Some elements cannot be seen in the picture but are modelled:

- irradiation channels in the core
- graphite of the thermalizing and thermal column
- carrousel with explicitly modelled irradiation tubes
- neutron source element
- triangular irradiation channel
- radial and tangential beam ports

The ENDF/B-VII.0 nuclear data set (Chadwick et al., 2006) was used for neutron transport calculations as well as for reaction rate calculations.

A more accurate description of the modelling assumption of the TRIPOLI model of TRIGA and its validation can be found in (Henry et al., 2015). The core configuration studied is very similar to the one used for reaction rates calculations of TRIGA Mark II reactor. The main difference being in control rod position and number of fuel rods and that the zero power condition is not imposed: temperatures of the fuel and water as well as the density of the water used in TRIPOLI are the one obtained from CFX.

3.2. Thermal-hydraulics

The CFD simulations were performed with the commercial code ANSYS® CFX. Two different models were used.