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Neutronic calculation of an axially heterogeneous ASTRID fuel assembly with APOLLO3®: Analysis of biases and foreseen improvements

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ABSTRACT

Design of modern sodium fast reactors such as the French SFR project ASTRID are axially heterogeneous in order to maximize neutron leakage and therefore improve their natural behavior during transients. This feature is however challenging for neutronic calculations that usually rely on a first step based upon the generation of averaged cross sections over planar geometries. It is in particular the case of the APOLLO3® reference calculation route that is currently considered for SFR applications. This paper investigates the corresponding biases in the case of a 3D fissile assembly calculation and highlights the consequences of the fundamental mode assumption. It is shown that even though an acceptable global accuracy can be achieved, the flux distribution presents a slight axial drift. The homogenization and energy collapsing of anisotropic cross sections over radial geometries is pointed responsible for it. Several proposals are made to correct this inconsistency, leading to new calculation schemes and enhanced performances.

1. Introduction

The French sodium fast reactor (SFR) prototype ASTRID reached in 2015 the end of its conceptual design phase (Beck et al., 2017). The CFV (“coeur a faible vidange”) concept has been confirmed for the core because of an improved natural behavior during unprotected transients. One of its main particular features is the axially heterogeneous fuel assembly design that maximizes neutron leakage and therefore ensures a low sodium void worth. It is based upon the insertion of a UO_2 fertile layer into the (U,Pu) O_2 fissile zone and an upper sodium plenum. A fertile blanket is also located at the bottom of the active core.

This highly heterogeneous design is a breakthrough compared to previous reactors and is likely to have an impact on the way the neutron distribution is numerically calculated over the assembly. In fact, most neutronic calculation tools assume that this distribution can be approximated by the fundamental mode of the Boltzmann neutron transport equation, that is the asymptotic solution for an axially infinite and radially periodic lattice. In such an approximation, the actual core geometry is only considered afterward in order to account for the coupling between assemblies.

Due to the combination of its axially heterogeneous design and of the large mean free path of neutrons in sodium fast reactors, this assumption may be inaccurate in an ASTRID fuel assembly.

The development of the APOLLO3® code for reactor physics analysis at Commissariat l’Energie Atomique et aux Energies Alternatives (CEA) in France is therefore an opportunity to investigate these hypotheses. Relying on the previous generation of deterministic codes such as ECCO/ERANOS (Ruggieri et al., 2006) or APOLLO2 (Sanchez et al., 2010), it takes advantage of modern computer architectures and enhanced numerical methods and opens possibilities toward the definition of innovative calculation schemes (Schneider et al., 2016).

The current paper aims at presenting the limitations of the actual APOLLO3® reference route used for the calculation of a CFV assembly and to show how they can be circumvent. The fundamental mode approximation is at the center of our inquiries.

In Section 2, the neutron transport equation is presented together with the main principles of its resolution, leading to the concept of calculation scheme. The current APOLLO3® calculation route for sodium fast reactor analysis, that relies on a two-step approach, is presented in Section 3. Results are given for an ASTRID fissile assembly together with a detailed analysis of model biases. As a consequence of this analysis, improved calculation schemes are proposed in Section 4, focusing mainly on an advanced modeling of the axial neutron streaming.

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2. Neutronic calculation schemes

Starting from the presentation of the neutron transport equation in Section 2.1, the concept of calculation scheme is defined in Section 2.2. The key points of the traditional two-step approach are recalled in Section 2.3 and some comments concerning SFRs are made in Section 2.4.

2.1. The transport equation

The statistical behavior of a stationary neutron population in a reactor core can be described by the well-known multigroup Boltzmann transport equation:

$$\left(\vec{\Omega}\cdot\vec{\nabla}+\Sigma^{g}(\vec{r})\right)\psi^{g}(\vec{r},\vec{\Omega})=\sum_{g'=1}^{N_g}\sum_{l=0}^{+\infty}\frac{2l+1}{4\pi}\Sigma_{s,l}^{g'\to g}(\vec{r})\sum_{m=-l}^{+l}\phi_{lm}^{g'}(\vec{r})R_{lm}(\vec{\Omega})+\frac{1}{4\pi}\sum_{g'=1}^{N_g}\chi^{g'\to g}(\vec{r})\nu\Sigma_{f}^{g'}(\vec{r})\phi^{g'}(\vec{r})+S^{g}(\vec{r},\vec{\Omega}) \tag{1}$$

where $\Sigma^{g}(\vec{r}),\Sigma_{s,l}^{g'\to g}(\vec{r})$ and $\nu\Sigma_{f}^{g'}(\vec{r})$ are respectively the macroscopic total, l-th order scattering and production cross-sections and $\chi^{g'\to g}(\vec{r})$ is the fission spectrum, which is often considered to be independent of the departure group g'. The angular flux $\psi^{g}(\vec{r},\vec{\Omega})$ is the desired solution of the problem whose boundary conditions must be specified. The angular flux moments are given by:

$$\phi_{lm}^{g}(\vec{r})=\int_{4\pi}d^{2}\Omega\psi^{g}(\vec{r},\vec{\Omega})R_{lm}(\vec{\Omega}) \tag{2}$$

where R_{lm} are real spherical harmonics and the scalar flux is $\phi^{g'}(\vec{r})=\phi_{00}^{g'}(\vec{r})$. The external source term $S^{g}(\vec{r},\vec{\Omega})$ drives the neutron flux given that the number of neutrons produced by fission does not exceed the number of disappearance by absorption or leakage.

Yet, when the reactor is in operating conditions, the fission term domines the external source and Eq. (1) is transformed into a K-eigenvalue problem:

$$\left(\vec{\Omega}\cdot\vec{\nabla}+\Sigma^{g}(\vec{r})\right)\psi^{g}(\vec{r},\vec{\Omega})=\sum_{g'=1}^{N_g}\sum_{l=0}^{+\infty}\frac{2l+1}{4\pi}\Sigma_{s,l}^{g'\to g}(\vec{r})\sum_{m=-l}^{+l}\phi_{lm}^{g'}(\vec{r})R_{lm}(\vec{\Omega})+\frac{1}{4\pi K}\sum_{g'=1}^{N_g}\chi^{g'\to g}(\vec{r})\nu\Sigma_{f}^{g'}(\vec{r})\phi^{g'}(\vec{r}) \tag{3}$$

The maximal eigenvalue K is called the multiplication factor and the corresponding eigenvector ψ is supposed to be an estimation of the real flux in the reactor core.

In the above equations, continuous energy dependencies have been replaced by a multigroup formalism (with N_g groups) as it is the way most neutron transport codes deal with the energy problem.

2.2. Calculation schemes

No analytical solution of Eqs. (1) or (3) can be found in a reactor core. Therefore, numerical methods are used. Those methods are confronted to the great size and highly heterogeneous structure of a reactor core that makes millions of spatial meshes and hundreds of angular directions necessary to properly represent neutron trajectories. Besides, the resonant behavior of cross sections requires a discretization of the energy domain in thousands of energy groups. If super-computers are now able to treat such complex computational problems all at once, it is still out of range of desktop computers. Multi-step approaches are then designed in order to divide the problem into several simplified ones with limited biases.

2.2.1. All at once resolution of the transport problem

Monte Carlo methods (Metropolis and Ulam, 1949) are an example of a one-step approach. They do not solve directly the transport equation but rely on a history-based probabilistic description of the particles. They are able to compute the neutron distribution in the reactor core with no approximation. They are widely used in the research and development field as powerful tools to understand the neutron transport physics or as reference calculations. Recent work have been focused in providing a way to also use them in multi-step approaches (Cai, 2014).

The all at once resolution of (1) or (3) is also a long-term goal of the deterministic neutronic platform APOLLO3®. The possibility to use the accurate method of characteristics (MOC) on three-dimensional geometries associated with the development of innovative algorithms to reduce its computational cost (Sciannandrone et al., 2015; Graziano et al., 2017) is a significant step forward in that direction.

It is worth mentioning that, for deterministic calculations, nuclear data libraries with a few hundreds (or thousands) of energy groups must in all cases be prepared in order to reduce the dimension of the phase-space in energy. Additional informations about resonances (probability tables or equivalence parameters) are stored in those libraries and used by the code to produce geometry-dependent self-shielded cross sections.

2.2.2. Multi-step approaches

Multi-step approaches are still the most widely used, especially in the industry field where super-computers and time-consuming calculations cannot be afforded. They usually rely on two main steps known respectively as lattice and core calculations.

At the lattice level, it is assumed that the core is in first approximation an infinite lattice of identical assemblies. Fine-group nuclear data libraries are used to produced self-shielded cross sections and the flux distribution is computed with Eq. (1) or (3) on a 2D sub-assembly with adequate boundary conditions (translation or reflection). A more or less conservative theory is then applied to produce nuclear data from this fine distribution on a coarse mesh in energy and space. Usually, a homogenization technique is used to produce one set of cross sections by type of assembly and an energy-condensation is performed to reduce the number of groups to a few tens maximum.

At the core level, the reactor is modeled by homogeneous blocks whose nuclear properties are the outputs of the lattice step.

A synthetic overview of such a two-step calculation scheme is given in Fig. 1.

2.3. Treatment of physical phenomena in a two-step approach

This section is dedicated to the mathematical and physical background involved in a traditional two-step calculation scheme. Special attention will be paid to the methods that have been used for this work and that are well adapted for fast neutron reactor analysis.

2.3.1. Resonance self-shielding

In the fine-group data libraries, resonance information is stored in addition to averaged values of cross sections. The geometry-dependent self-shielded cross sections are then either constructed from an appropriate equivalence method (Hébert and Marleau, 1991; Hébert, 2004; Sanchez et al., 2010) or derived from a trans-