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Extension of the GeN-Foam neutronic solver to SP_3 analysis and application to the CROCUS experimental reactor

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ABSTRACT

The Laboratory for Reactor Physics and Systems Behaviour at the PSI and at the EPFL has been developing since 2013 a multi-physics platform for coupled reactor analysis named GeN-Foam. The developed tool includes a solver for the eigenvalue and transient solution of multi-group neutron diffusion equations. Although frequently used in reactor analysis, the diffusion theory shows some limitations for core configurations involving strong anisotropies, which is the case for the CROCUS research reactor at the EPFL. The use of an SP_3 approximation to neutron transport can often lead to visible improvements in a code predictive capabilities, especially for one-directional anisotropies, with acceptable added computational cost vs diffusion. Following some modelling issues for the CROCUS reactor, and in order to improve the GeN-Foam modelling capabilities, the GeN-Foam diffusion solver has been extended to allow for SP_3 analyses. The present paper describes such extension and a preliminary verification using a mini-core PWR benchmark. The newly developed solver is then applied to the analysis of the CROCUS experimental reactor and results are compared to Monte Carlo calculations, as well as to the results of the diffusion solver.

1. Introduction

Several efforts have been spent in recent years for the modernization of the numerical toolset available in the nuclear engineering community. This is witnessed by important initiatives like the CASL project (Hursin et al., 2008; Yan et al., 2011; CASL, 2011) in the US and the NURESAFE project (NURESAFE, 2013) in Europe. Some initiatives have also been undertaken aiming at a paradigm shift in the nuclear code development, the main idea being a code development based on modern numerical libraries and methodologies and not on legacy codes. Two main examples in this sense are the MOOSE project in the US (Gaston et al., 2009) and the several initiatives dedicated to the use of the CFD-oriented OpenFOAM® library (Weller et al., 1998; OpenFOAM, 2016) for nuclear reactor applications (Aufiero, 2014; Aufiero et al., 2014; Clifford and Ivanov, 2010; Clifford, 2013; Clifford et al., 2013; Jareteg et al., 2014, 2015). In this framework, the Laboratory for Reactor Physics and Systems Behaviour at the PSI and at the EPFL has started developing new tools for reactor analysis and has decided to privilege the use of OpenFOAM mainly due to its open-source philosophy, which is expected to promote collaborative work and support educational efforts. A main result of this research work has been the development of a multi-physics tool for steady-state and transient analysis of nuclear reactors named GeN-Foam (Fiorina et al., 2015a,b,c, 2016). A main objective of this tool is to provide enough flexibility to analyze non-conventional reactor designs and specific phenomena that legacy codes cannot easily investigate. Such a flexibility derives from: the use of unstructured meshes allowing for fully arbitrary geometries; and the high-level object-oriented programming style of OpenFOAM, which allows for easy and relatively fail-safe modifications of the source code. A second important objective is to promote collaborative efforts on research and education, which derives naturally from the open-source philosophy associated to OpenFOAM, and from the possibility to easily develop and share stand-alone solvers and routines (in the form of C++ classes). It is worth mentioning that OpenFOAM is based on finite-volume discretization techniques and preferentially relies on operator-splitting for the coupling of different equations. This poses some limitations if compared to a finite-element framework with arbitrary coupling like that of MOOSE. On the other hand, finite volumes and operator-splitting are well-known and relatively easy techniques, which can promote the creation of a wider user and developed

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Nomenclature

Latin symbols

C_k	concentration of the kth delayed neutron precursor group [m^{-3}]
D_i	neutron diffusion coefficient for the ith energy group[m]
k_{eff}	effective multiplication factor [–]
S_d	delayed neutron source [m^{-3}·s^{-1}]
$S_{n,i}$	fission neutron source from neutron energy groups others than the ith [m^{-3}·s^{-1}]
$S_{s,i}$	scattering neutron source from neutron energy groups others than the ith [m^{-3}·s^{-1}]
t	time [s]
$\boldsymbol{u}$	fuel velocity [m·s^{-1}]
v_i	average neutron velocity for the ith energy group [m·s^{-1}]

Greek symbols

α	albedo coefficient [–]
β_k	delayed neutron fraction for the kth delayed neutron precursor group [–]
β_t	total delayed neutron fraction [–]
λ_k	decay constant for the kth delayed neutron precursor group [s^{-1}]
υ	average number of neutrons per fission [–]
$\Sigma_{a,i}$	absorption cross section for the ith energy group [m^{-1}]
$\Sigma_{r,i}$	removal (disappearance) cross section for the ith energy group [m^{-1}]
$\Sigma_{f,i}$	fission cross section for the ith energy group [m^{-1}]
$\Sigma_{j\rightarrow i}$	group-transfer cross section from the jth to the ith energy group [m^{-1}]
$\varphi_{0,i}$	0th moment of the neutron flux for the ith energy group [m^{-2}·s^{-1}]
$\widehat{\varphi}_{0,i}$	flux defined in Eq. (1) [m^{-2}·s^{-1}]
$\varphi_{2,i}$	2th moment of the neutron flux for the ith energy group [m^{-2}·s^{-1}]
$\chi_{d,i}$	delayed neutron yield for the ith energy group [–]
$\chi_{p,i}$	prompt neutron yield for the ith energy group [–]

community. In addition, the large OpenFOAM user community continuously develops and shares several new solvers and routines for various applications and problems, which adds to the reliable, high-performing tools for thermal-hydraulics and thermal-mechanics that are already available in the standard OpenFOAM release.

The GeN-Foam solver tightly couples a fine/coarse-mesh sub-solver for single-phase thermal-hydraulics, a displacement-based sub-solver for thermal-mechanics, a simple sub-solver for fuel temperature prediction, and a diffusion-based sub-solver for neutronics. These features provide basic capabilities for the transient analysis of single-phase reactors. However, the diffusion approximation for neutron transport can be unsuited to analyze core configurations featuring strongly anisotropic fluxes. The important limitations associated to a diffusion approximation have been for instance observed in the modelling of the CROCUS research reactor at the EPFL (Rais et al., 2015). Aiming at a better modelling of the CROCUS reactor and at a general improvement of the GeN-Foam predictive capabilities, GeN-Foam has been modified to allow for the possibility to evaluate neutron fluxes based on an SP_3 methodology (Azmy and Sartori, 2010). An SP_3 solver can in fact provide significant improvements in a code predictive capabilities if compared to diffusion, with only a mild impact on computational requirements (IAEA-TECDOC-1700, 2013; Lee et al., 2015; NEA, 2016; Zhong et al., 2003).

Compared to most legacy codes (see e.g. (Rohde et al., 2016; PARCS, 2009)), the SP_3 solver developed for GeN-Foam has the specificity of employing general unstructured meshes. On one hand, this negatively impacts code performances, as it forces to use relatively general solution algorithms. On the other hand, it offers the possibility to study unconventional core geometries, as well as to better investigate specific phenomena, such as the reactivity effect of non-uniform core deformations in fast reactors (Fiorina et al., 2015a).

Main objectives of this paper are to describe in details and preliminary verify the extension of GeN-Foam to SP_3 analyses (Sections 2 and 3), as well as to test its new predictive capabilities via application to the CROCUS research reactor (Section 4). In particular, the results obtained using the GeN-Foam diffusion and SP_3 solvers are compared between each other and with the results obtained using the Serpent Monte Carlo code (Leppänen, 2007). The conclusions of the work are drawn in Section 5.

2. Solver description

The SP_3 model has been implemented in GeN-Foam as an extension of the available diffusion model described in Refs. (Fiorina et al., 2015b, 2016). Structure and main features of the code have been maintained, but the equations and boundary conditions have been modified to allow for an optional transition from diffusion to SP_3 equations.

2.1. Mathematical model

Spherical harmonics (P_N) approximation (Azmy and Sartori, 2010) of the neutron transport equation is obtained via expansion of the angular dependence of the neutron flux in the first N spherical harmonics. In three dimensions, this results in a complex system of differential equations, with a number of equations growing as $(N+1)^2$. The simplified spherical harmonics (SP_N) approximation has been introduced as a trade-off between accuracy of results and complexity of the model. It was first derived heuristically (Gelbard, 1962) by replacing second derivatives with Laplacian operators in a 1-D planar formulation of the P_N approximation. In a 1-D planar geometry, and thus in the SP_N approximation, the system of equations to be solved is limited to $(N+1)/2$ elliptic (diffusion-like) equations, which provide a significant simplification over a standard P_3 approximation. In the specific case of the SP_3 approximation, 4 equations can be obtained for the 4 moments φ_0 to φ_3, and substitution of odd-order moments into even-moment equations then allows to reduce to 2 the number of equations to be solved. It is worth mentioning that the same 2 equations have also been derived rigorously by Brantley and Larsen (2000) on the basis of variational methods using a trial function featuring a one-dimensional variation in an arbitrary direction, which provides a sounder theoretical foundation to the SP_3 methodology. Based on both the heuristic and the more rigorous derivations, it is evident that an SP_3 model can generally be considered as accurate for systems in which flux variations near