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Neutronic Modelling and Computation of TEPLATOR Core using COMSOL Multiphysics Code

Dipanjan Ray

Research and Innovation Centre for Electrical Engineering
University of West Bohemia
Univerzitní 8
30100, Pilsen, Czech Republic

dipanjan@fel.zcu.cz

Martin Lovecký, Jiří Závorka, Radek Škoda

Research and Innovation Centre for Electrical Engineering
University of West Bohemia
Univerzitní 8
30100, Pilsen, Czech Republic

Czech Technical University in Prague,
Czech Institute of Informatics, Robotics and Cybernetics
Jugoslavskych partyzanu 1580/3, 160 00 Prague 6,
Czech Republic

lovecky@fel.zcu.cz, zavorka@fel.zcu.cz, radek.skoda@cvut.cz

ABSTRACT

TEPLATOR is an innovative heavy water small modular reactor concept that has the potential to meet district and industrial heating demands by harnessing nuclear energy as a heat source. This approach offers a substantial reduction in pollution and environmental impact compared to conventional heating methods reliant on fossil fuels.

The primary objective of this article is to develop a two-dimensional, multi-group neutron diffusion model for the TEPLATOR reactor core using the COMSOL Multiphysics software package. COMSOL code employs a finite element numerical scheme to solve the partial differential equations associated with the neutron diffusion model. Monte Carlo transport code Serpent version 2.2.1 with the latest ENDF/B-VIII.0 nuclear data library is employed to calculate the multi-group constants for different reactor regions. These multi-group constants are then incorporated in the developed COMSOL neutron diffusion model to perform calculations using adaptive mesh refinement technique to enhance the accuracy of the solution. Neutronic behaviour of the TEPLATOR reactor core is obtained by calculating the criticality of the system and analysing the steady-state neutron-flux distribution profile. The calculated results are subsequently compared with those generated by the Serpent, providing a basis for validation and further analysis.

1 INTRODUCTION

In today's technologically advanced era, abundant computational capability enables an in-depth analysis of various physical phenomena involved in nuclear technology. However, when confronted with the challenge of developing a novel reactor concept, an efficient approach is required for the evaluation of various design options and the computing of specific physical parameters for each design. Such situations necessitate the availability of a simplified mathematical framework essential. This perspective remains relevant to reactor core analysis as well. Although detailed neutron transport equations can be solved using Monte Carlo codes, doing so during the designing stage can be time-consuming. Therefore, a simplified multi-group neutron diffusion model poses an efficient alternative as it expedites the process of various design analysis and parameter computation.

The primary objective of this study is to perform a neutronic study related to the heavy water small modular concept reactor TEPLATOR, a project presently being developed at the University of West Bohemia and Czech Technical University Prague. [5-11]. A two-dimensional, two-group neutron diffusion model is developed to perform this research. The COMSOL multi-physics software package is being utilized to create the model. This software provides a compelling option for nuclear reactor neutronic, thermal-hydraulic, and thermomechanical computation due to its ability to combine different physics, facilitating a comprehensive analysis [1-4].

2 MATHEMATICAL MODEL

2.1 Description of the Model

Geometric configuration of the TEPLATOR reactor core is shown in Figure 1 (a). Thermal power of the core is 50 MW. Core consists of 55 hexagonal pressure channels containing fresh fuel assemblies of the VVER-440 reactor, radial reflectors, and 22 control rods. Fuel assemblies are divided into two categories with enrichments of 1.1% and 0.9%, and their spatial arrangement is illustrated in Figure 1(b). Within the reactor, three distinct types of control rods are implemented: 3 regulating rods, 15 compensation rods, and six emergency

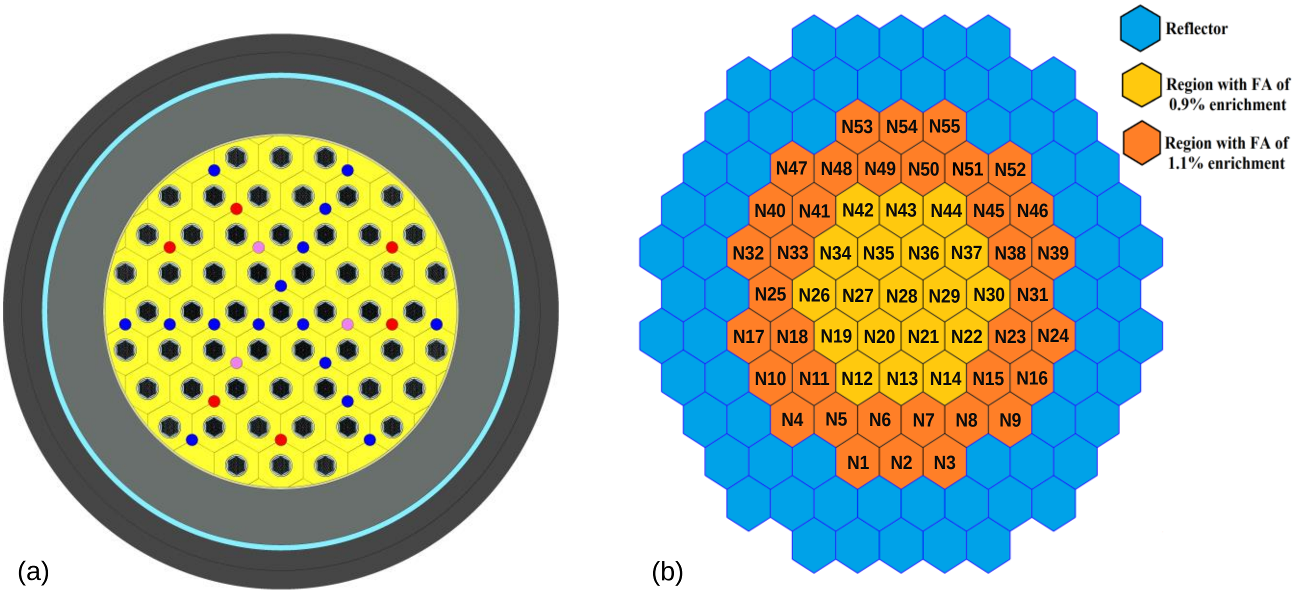

Figure 1: (a) TEPLATOR reactor core with control rods (pink represents regulating rod, blue represents compensation rods and red represents emergency rods, (b) reactor core nodalization with different enrichments.

rods, with their respective positions mentioned in Figure 1(a).

For the mathematical model, the whole reactor core region is divided into hexagonal cells of pitch 40 cm, and each control rod is a part of three adjacent cells (see Figure 1 (a) and (b)).

2.2 System of Equations

Two group steady-state neutron diffusion equations for the core regions are shown below in Eq. (1) and (2).

$$-\nabla D_1 \nabla \emptyset_1 + (\Sigma_{a1} + \Sigma_{12})\emptyset_1 = \frac{1}{k_{eff}}(\upsilon\Sigma_{f1}\emptyset_1 + \upsilon\Sigma_{f2}\emptyset_2) \tag{1}$$

$$-\nabla D_2 \nabla \emptyset_2 + \Sigma_{a2}\emptyset_2 - \Sigma_{12}\emptyset_1 = 0 \tag{2}$$

Where subscripts 1 and 2 denote the fast and thermal neutron energy group, respectively. D represents diffusion coefficient, Σ_a and Σ_f represent absorption and fission cross-section respectively, υ is the number of neutrons emitted per fission, k_{eff} is the effective multiplication factor, and Σ_{12} is the scattering cross section fast to thermal group respectively. The model is simplified by assuming there is no up-scattering from thermal to fast neutron group.

The production term in Eq. (1) (right-hand side of the equation) is not present for the calculation of the non-multiplying region (reflector). Vacuum boundary condition is applied at the external surface of the reflector region. Neutronic parameters for different hexagonal regions of the reactor core are generated using Monte Carlo transport code Serpent version 2.2.1 with the latest ENDF/B-VIII.0 nuclear data [12]. Operating temperature of the Teplator is quite low, and it is expected that the cross-section is weakly dependent on the variation of the operating temperature. Moderator (333 K) and coolant temperatures (465 K) are exact from the thermal-hydraulic analysis, and fuel with 600 K is considered to avoid possible interpolation in the calculation of group constants.

2.3 Numerical Method

COMSOL software employs a finite element discretization technique for the solution of non-linear partial differential equations (PDEs). In the present study, steady-state equations are presented in coefficient form PDE mode in COMSOL solver.

Stationary calculation procedure is divided into two sequential stages. The first step includes the eigenvalue study framework for solving multi-group diffusion equations. The equations for eigenvalue calculation corresponding to two distinct energy groups (referred to as Eq. (1) and (2)) are presented in terms of the coefficient form PDE eigenvalue mode:

$$\lambda^2 e_a u - \lambda d_a u + \nabla.(-c\nabla u - \alpha u + \gamma) + \beta.\nabla u + au = f \tag{3}$$

Table 1 shows the representation of different coefficient terms (d_a, e_a, c, α, β, γ, a) from Eq. (3) in terms of two-group diffusion equations mentioned in Eq. (1) and (2), and the term λ in Eq. (3) represents the eigenvalue.

Table 1: Representation of equations in COMSOL using co-efficient form PDE eigenvalue mode

Equations	**Dependent Variables (u)**	e_a	d_a	c	α, β, γ, a	f
Neutron diffusion (2 equations)	ϕ_g where $g = 1\ to\ 2$	0	1	$\mathrm{D_g}$	0	Rest of the terms of Eq. (1) and (2)

In the second step, the stationary study mode is applied with an initial flux guess calculated from the eigenvalue stage. Stationary analysis of neutron diffusion equations is presented by coefficient form of PDE mode in COMSOL. Equations are expressed as:

$$e_a \frac{\partial^2 u}{\partial t^2} + d_a \frac{\partial u}{\partial t} + \nabla.(-c\nabla u - \alpha u + \gamma) + \beta.\nabla u + au = f \tag{4}$$

For the stationary calculation, value of d_a is equal to zero, and rest of the values of co-efficient in Eq. (4) are same as mentioned in Table 1. Effective multiplication factor and the flux distribution are then calculated using stationary study mode by applying the criticality normalization condition (normalized to 1 fission neutron) expressed below:

$$k_{eff}: \left(\iiint \sum_{g=1}^{2} \nu\Sigma_f^g \phi_g dV\right) - 1 = 0 \tag{5}$$

External surface boundary conditions of the geometry are described by Dirichlet boundary conditions. Adaptive mesh refinement technique in COMSOL is utilized to increase the solution accuracy of the model within reasonable computational time frame. This technique effectively reduces the overall error by constructing finer meshes in the regions exhibiting substantial errors. Figure 2 depicts the two-dimensional mesh structure created using adaptive mesh refinement technique for the neutronic calculation of the Teplator core.

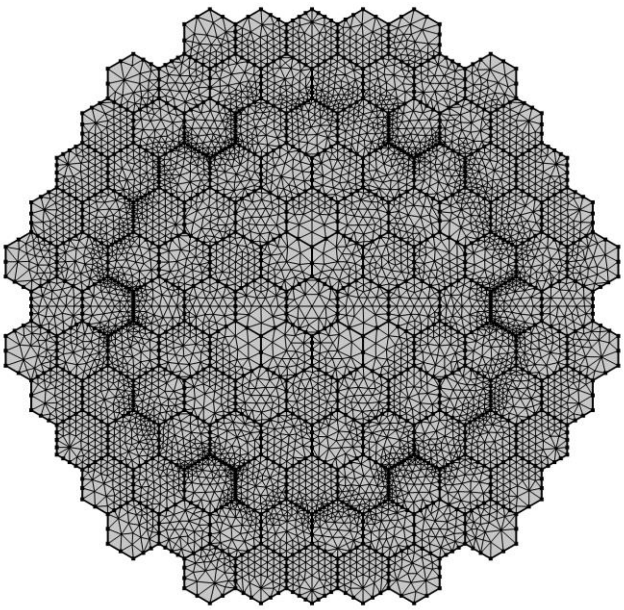

Figure 2: Two-dimensional mesh structure generated by COMSOL for the Teplator core.

3 RESULTS AND DISCUSSIONS

Calculation for the COMSOL two-group neutron diffusion model is performed for two cases in different burn-up cycles (Beginning of Cycle (BOC), Middle of Cycle (MOC), and End of Cycle (EOC)) of the TEPLATOR core, and these cases are mentioned below:

Case 1: All the rods are inside the reactor core.

Case 2: All the rods are outside the reactor core.

Effective multiplication factor for each case is compared with the result from the Serpent code and is mentioned in Table 2. Maximum difference in pcm between the Serpent and COMSOL model is found to be 347.56 (percentage difference of 0.3791). Calculated fast and thermal neutron flux value for case 1 and case 2 in BOC are shown in Figure 3. The values are normalized to their respective maximum flux values.

A comparative analysis of normalized total flux values (normalized to 1) between the COMSOL diffusion model and the Serpent code is presented in Figure 4 for Case 1, considering the BOC. The observed maximum percentage difference between the two models is approximately 8%. Notably, the highest flux value is obtained at node 19 of the core region, as illustrated in Figure 1(b) and Figure 4, which is consistent between the COMSOL and Serpent simulations.

Table 2: Effective multiplication factor (k_{eff}) calculated from COMSOL diffusion model and Serpent code for the case of different rod positions and burn-up cycles

Cases: Different rod positions	**Burn-up cycles**	**k_{eff} from COMSOL diffusion model**	**k_{eff} from Serpent**	**Difference in pcm**	**Percentage Difference (%)**
Case 1	BOC	0.94897	0.95232	334.74	0.3518
	MOC	0.91443	0.91791	347.56	0.3791
	EOC	0.90739	0.91040	301.0	0.3306
Case 2	BOC	1.10884	1.11127	242.81	0.2187
	MOC	1.05918	1.06250	332.23	0.3125
	EOC	1.04766	1.05056	289.88	0.2760

Additionally, the COMSOL model, due to its simplified diffusion approximation, exhibits significantly less computational time (computation time of approximately 8 min) compared to the Serpent neutron transport model. Thus, obtained results demonstrate the suitability of COMSOL diffusion model as a viable alternative tool for the potential design evaluations and preliminary design parameter calculations for the TEPLATOR reactor.

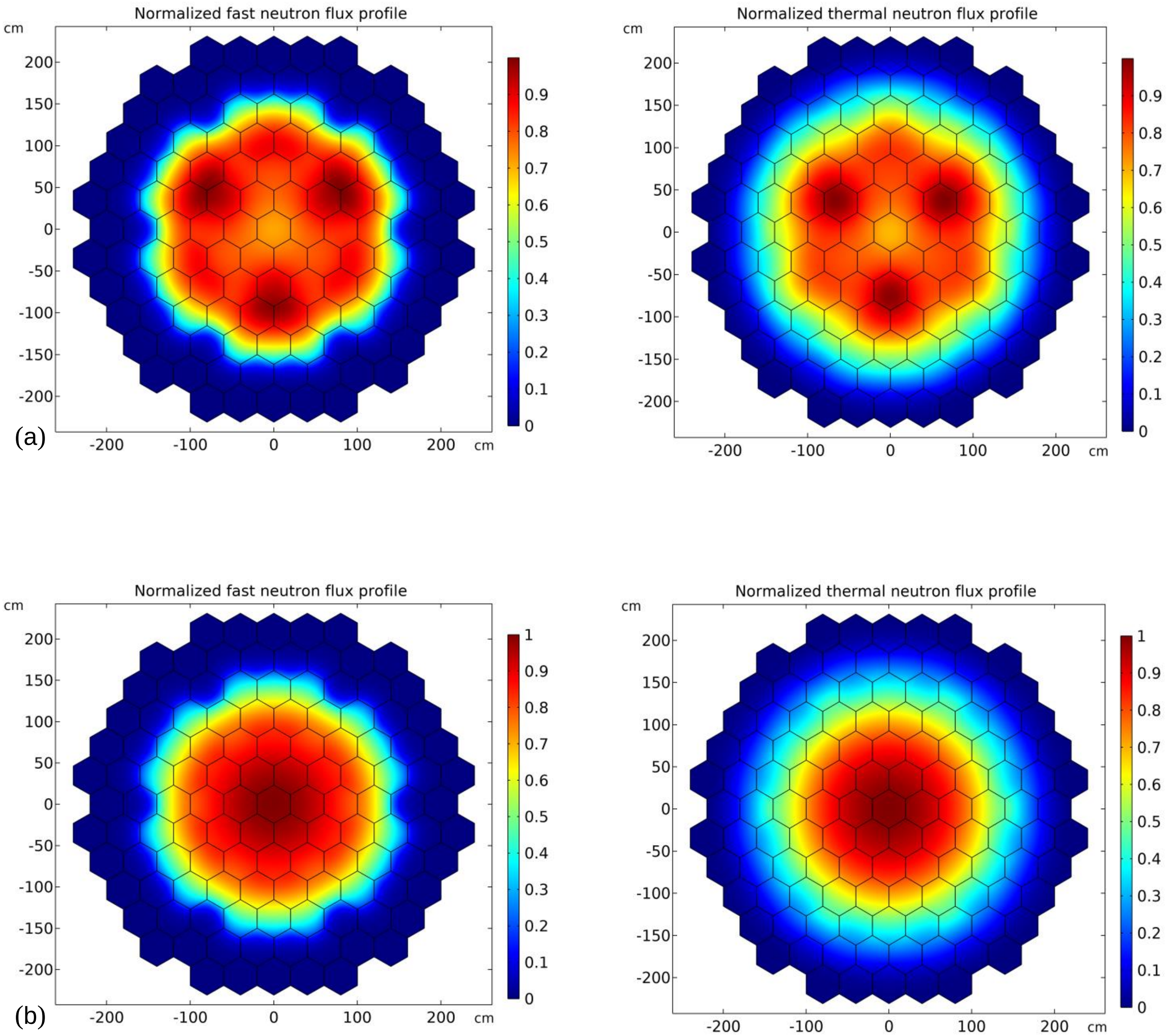

Figure 3: Normalized fast and thermal flux profile for BOC (a) case 1 (All the rods are inside the reactor core) (b) case 2 (All the rods are outside the reactor core).

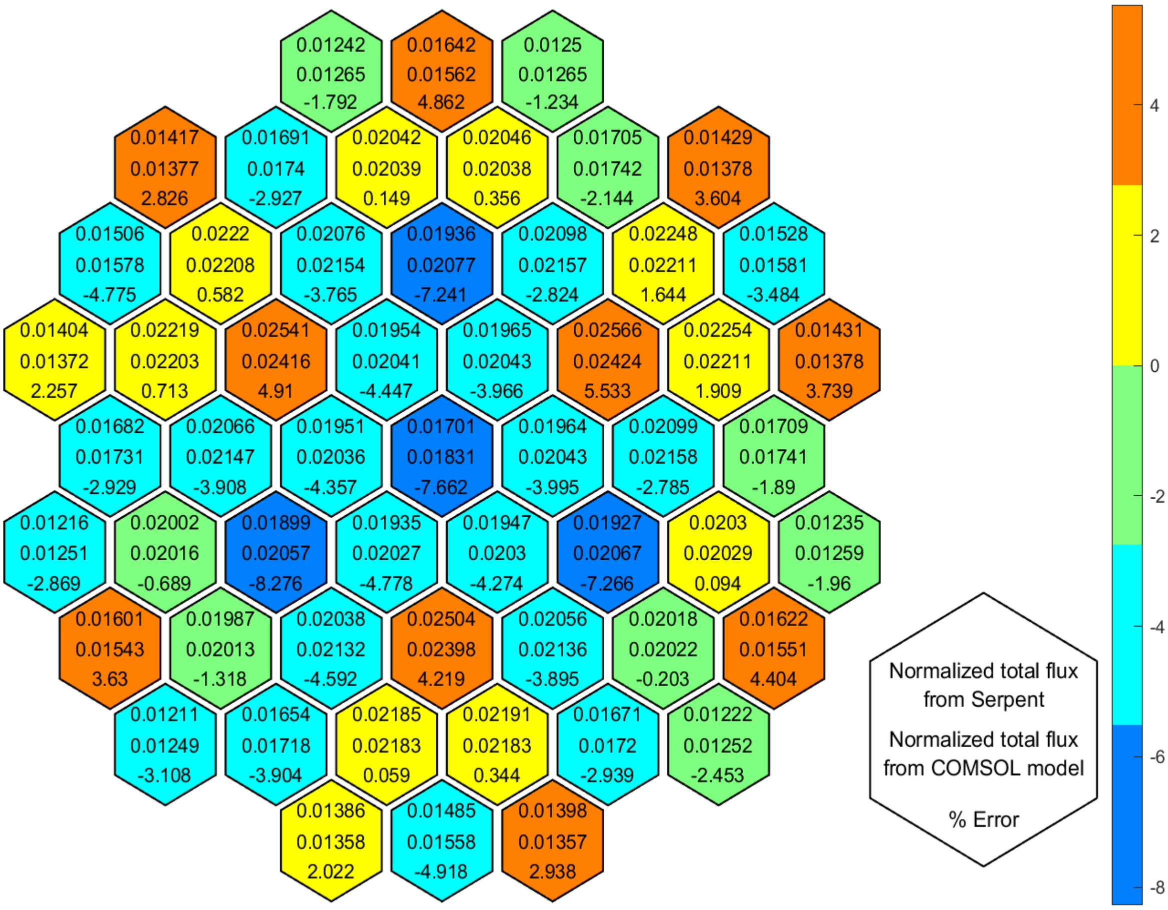

Figure 4: Normalized total flux distribution (normalized to 1) in hexagonal cell calculated from COMSOL diffusion model and Serpent code, for case 1 BOC.

CONCLUSIONS

A two-dimensional, two-group neutron diffusion model is developed using the COMSOL Multiphysics software package to analyse the steady-state neutronic behaviour of the TEPLATOR core. Two simulation conditions are considered, where all control rods are placed both inside and outside the core, and both cases are solved for different burn-up cycles. The calculated results, including the effective multiplication factor and flux distribution, obtained from the COMSOL diffusion model are compared with the results from the Monte Carlo neutron transport code Serpent, demonstrating good agreement between the two models.

In future research work, the emphasis will be placed on extending the existing two-dimensional model of the TEPLATOR core to a three-dimensional model using COMSOL software. The focus will involve performing both steady-state and transient calculations with the goal of calculating various neutronic parameters, such as the reactivity worth of the control rods and reactivity-initiated transients.

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