Implementation of the SP_3 equations in a MOOSE-based application

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INTRODUCTION

Multi-physics simulations are necessary for assessing the safety characteristics of nuclear reactors [1]. Solving the neutronics equations provides information on the power distribution, which plays a crucial role in the thermal-fluids behavior of the reactor. Therefore, the right modeling of the neutron flux is necessary. This work presents the implementation and verification of the Simplified P_3 (SP_3) equations [2] in a Multiphysics Object-Oriented Simulation Environment (MOOSE)-based application as an alternative for solving the neutronics equations in a nuclear reactor.

MOOSE [3] is a computational framework that supports engineering analysis applications. In a nuclear reactor, several partial differential equations describe its physical behavior. These equations are typically nonlinear, and they are often coupled to each other. MOOSE is an open-source Finite Element Method (FEM) framework that supports the development of applications for solving such systems. All the software built on the MOOSE framework shares the same Application Programming Interface (API), facilitating relatively easy coupling between different phenomena. While the SP_3 equations solve the neutronics equations in a nuclear reactor, other applications may solve the thermal-fluids, and given they share the same API; their integration is straightforward.

The P_N method [4] discretizes the transport equation by expanding the angular dependence of the neutron flux in spherical harmonics, considering up to order N polynomials. If $N \to \infty$, the solution of the P_N equations tends to the exact transport solution. In three-dimensional geometries, the number of P_N equations is proportional to $(N+1)^2$, whereas, in one-dimensional planar geometries, the number of P_N equations is $(N+1)$. Gelbard [2] proposed the SP_N approximation by replacing the second derivatives in the one-dimensional planar P_N equations with three-dimensional Laplacian operators. This approximation considerably reduces the number of equations conserving a reasonable accuracy [5].

The SP_N approximation has the disadvantage that the solution does not usually converge to the true transport solution as $N \to \infty$. Additionally, the theoretical basis of Gelbard's formulation of SP_N approximation was weak. For these reasons, the method did not gain widespread use until the 2000s, when thanks to Pomraning [6], Brantley, and Larsen's [7] contribution, the method gained a stronger theoretical basis.

In practice, the SP_N equations are most accurate for diffusive problems or for problems in which the solution behaves nearly one-dimensionally and has weak tangential derivatives at material interfaces. For problems with strong, multidimensional transport effects, such as voids, streaming regions, or geometrically complex regions, the SP_N solutions are less accurate [8]. However, several results show that the SP_N approximation yields more accurate solutions than the diffusion approximation [9] [10] [11] [12] with considerably less computational expense than the discrete ordinates (S_N) method [7]. For example, the SP_3 approximation is preferable over the diffusion approximation for modeling reactors using MOX/UO_2 fuel assemblies. MOX fuel assemblies have higher thermal absorption and fission cross-sections than UO_2 fuel assemblies, and consequently, their thermal flux is lower while their power production higher. Modeling these characteristics using the diffusion approximation may be challenging [7] [5].

The SP_3 approximation gained usage throughout the last couple of decades and currently, different software uses it to solve the neutron transport equation. Some of those software are PARCS [8], DYN3D [9], and COCAGNE [10].

METHODOLOGY

This section describes the methodology followed for solving the equations. Davidson [4] defined the one-dimensional multi-group P_N equations. For $N = 3$ and steady-state, the equations become

$$\frac{d}{dx}\phi_{1,g} + \Sigma_{t,g}\phi_{0,g} = \sum_{g'=1}^{G}\Sigma_{s0,g'\to g}\phi_{0,g'} + \frac{\chi_g}{k_{eff}}\sum_{g'=1}^{G}\nu\Sigma_{f,g'}\phi_{0,g'} + Q_{0,g} \quad (1)$$

$$\frac{1}{3}\frac{d}{dx}\phi_{0,g} + \frac{2}{3}\frac{d}{dx}\phi_{2,g} + \Sigma_{t,g}\phi_{1,g} = \sum_{g'=1}^{G}\Sigma_{s1,g'\to g}\phi_{1,g'} + Q_{1,g} \quad (2)$$

$$\frac{2}{5}\frac{d}{dx}\phi_{1,g} + \frac{3}{5}\frac{d}{dx}\phi_{3,g} + \Sigma_{t,g}\phi_{2,g} = \sum_{g'=1}^{G}\Sigma_{s2,g'\to g}\phi_{2,g'} + Q_{2,g} \quad (3)$$

$$\frac{3}{7}\frac{d}{dx}\phi_{2,g} + \Sigma_{t,g}\phi_{3,g} = \sum_{g'=1}^{G}\Sigma_{s3,g'\to g}\phi_{3,g'} + Q_{3,g} \quad (4)$$

where

$\phi_{n,g}$ = n^{th} moment of group g neutron flux
$\Sigma_{t,g}$ = group g macroscopic total cross-section
$\Sigma_{sn,g'\to g}$ = n^{th} moment of the group g' to group g macroscopic scattering cross-section
$\nu\Sigma_{f,g}$ = group g macroscopic production cross-section
χ_g = group g fission spectrum
k_{eff} = eigenvalue
$Q_{n,g}$ = n^{th} moment of group g external neutron source
G = number of energy groups.

Assuming an isotropic external source and negligible anisotropic group-to-group scattering [7]

$$Q_{n,g} = 0, \quad n > 0$$
$$\Sigma_{sn,g'\to g} = 0, \quad g' \neq g, \quad n > 0$$

simplifies equations 2 and 4, allowing to express the odd moments of the flux $\phi_{1,g}$ and $\phi_{3,g}$ as functions of the even moments $\phi_{0,g}$ and $\phi_{2,g}$. Introducing $\phi_{1,g}$ and $\phi_{3,g}$ into equations 1 and 3 reduces the system from four to two equations. Introducing the variables $\Phi_{0,g}$ and $\Phi_{2,g}$, reorganizing the equations, and replacing the second derivatives by Laplacian operators [2] yields the SP_3 equations [9]

$$-D_{0,g}\Delta\Phi_{0,g} + \Sigma_{0,g}\Phi_{0,g} - 2\Sigma_{0,g}\Phi_{2,g} = S_{0,g} \tag{5}$$
$$-D_{2,g}\Delta\Phi_{2,g} + \left(\Sigma_{2,g} + \frac{4}{5}\Sigma_{0,g}\right)\Phi_{2,g} - \frac{2}{5}\Sigma_{0,g}\Phi_{0,g} = -\frac{2}{5}S_{0,g} \tag{6}$$

where

$$\begin{aligned}
\Sigma_{n,g} &= \Sigma_{t,g} - \Sigma_{sn,g'\to g} \\
\Phi_{0,g} &= \phi_{0,g} + 2\phi_{2,g} \\
\Phi_{2,g} &= \phi_{2,g} \\
D_{0,g} &= \frac{1}{3\Sigma_{1,g}} \\
D_{2,g} &= \frac{9}{35\Sigma_{3,g}} \\
S_{0,g} &= \sum_{g'\neq g}^{G} \Sigma_{s0,g'\to g}\left(\Phi_{0,g'} - 2\Phi_{2,g'}\right) \\
&\quad + \frac{\chi_g}{k_{eff}} \sum_{g'=1}^{G} \nu\Sigma_{f,g'}\left(\Phi_{0,g'} - 2\Phi_{2,g'}\right) + Q_{0,g}.
\end{aligned} \tag{7}$$

The Marshak-like vacuum boundary conditions (BCs) complete the system of equations [9]

$$\frac{1}{4}\Phi_{0,g} \pm \frac{1}{2}\hat{n}\cdot J_{0,g} - \frac{3}{16}\Phi_{2,g} = 0 \tag{8}$$
$$-\frac{3}{80}\Phi_{0,g} \pm \frac{1}{2}\hat{n}\cdot J_{2,g} + \frac{21}{80}\Phi_{2,g} = 0 \tag{9}$$

where

$$J_{n,g} = -D_{n,g}\nabla\Phi_{n,g}.$$

Finally, multiplying equations 5 and 6 by a test function and integrating over the domain yields the weak form of the equations modularized into kernels in the MOOSE-based application. For brevity, we will not display the derivation of the kernels here. Such procedure is standard in weighted residual methods and can be found in [11] and any finite elements book [13].

RESULTS

This section presents several numerical results that validate the calculation scheme. The following sections discuss the results of a one group exercise presented in [7] and describe the results of the C5 MOX Benchmark [14].

One group two-dimensional problem

This section describes a one group, isotropic-scattering eigenvalue problem introduced by Brantley and Larsen [7]. This section also presents the eigenvalue obtained with the SP_3 solver and compares it against the reference value. Figure 1 shows the problem's geometry, and Table I specifies its cross-sections. Due to the problem's symmetry, the model included only a quarter of the core. We created the mesh using the *Gmsh* [15] meshing tool. The mesh had 6×10^3 elements. The simulation convergence criterion was 10^{-8} for the neutron flux.

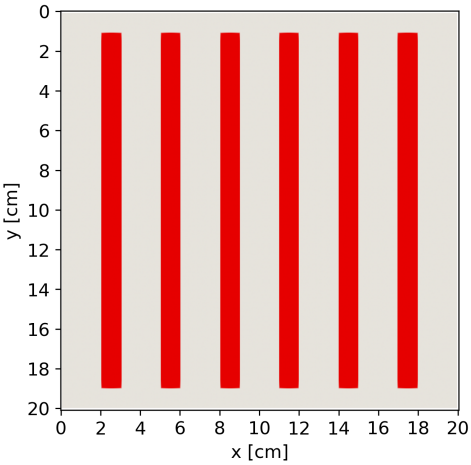

Fig. 1: Geometry of the one group eigenvalue problem. Fuel in red. Moderator in gray.

Material	Σ_t	Σ_{s0}	$\nu\Sigma_f$
Moderator	1.00	0.93	0.00
Fuel	1.50	1.35	0.24

TABLE I: Cross-sections of the one group eigenvalue problem [7]. Values expressed in cm^{-1}.

Table II compares the eigenvalue obtained with the SP_3 solver and the reference value [7] using the equation

$$\Delta_\rho = \left|\frac{k_{SP_3} - k_{Ref}}{k_{SP_3}k_{Ref}}\right| \tag{10}$$

where

Δ_ρ = reactivity difference $[pcm]$
k_{SP_3} = eigenvalue obtained with SP_3 solver
k_{Ref} = reference eigenvalue.

k_{Ref}	k_{SP_3}	Δ_ρ
0.79862	0.79854	12

TABLE II: Comparison between the result obtained with the SP_3 solver and the reference result for the one group eigenvalue problem.

C5 MOX Benchmark

This section introduces the C5 MOX Benchmark [14] and presents the SP_3 solver results. The Organisation for Economic Co-operation and Development (OECD)/Nuclear Energy Agency (NEA) developed this benchmark to validate methods and identify their strengths, limitations, and accuracy, and suggest needs for method development. Two types of fuel assembly (MOX and UO_2) and a reflector comprise the core, shown in Figure 2. Each fuel assembly consists of a 17×17 array of squared pin cells, as displayed in Figures 3 and 4. The dimensions of each pin cell are 1.26×1.26 cm, being 21.42 $\times$ 21.42 cm the dimensions of each assembly, and 128.52 $\times$ 128.52 cm of the whole core. The benchmark [14] specifies the cross-sections, which have a two-energy group structure.

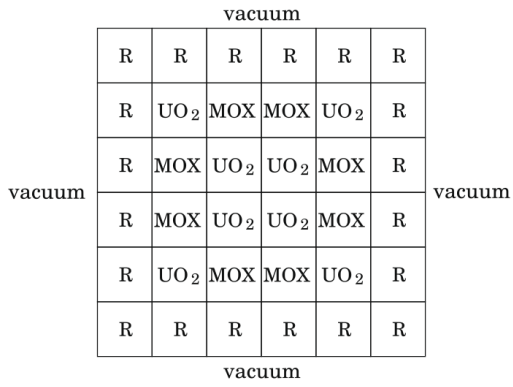

Fig. 2: C5 MOX benchmark configuration. *R* correponds to the reflector region. Image reproduced from [5].

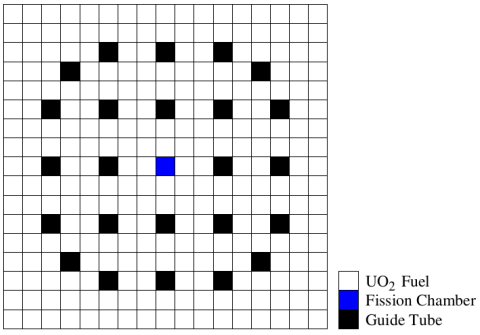

Fig. 3: Structure of the UO_2 assembly. Image reproduced from [5].

When no anisotropic component of the scattering cross-section is available, the benchmark recommends applying the diagonal transport correction

$$D_{0,g} = \frac{1}{3\Sigma_{tr,g}} \qquad (11)$$
$$\Sigma_{tr} = \Sigma_{t,g} - \bar{\mu}_g \Sigma_{s0,g}$$

where

$\Sigma_{tr,g}$ = group g transport cross-section
$\bar{\mu}_g$ = group g average cosine deviation angle.

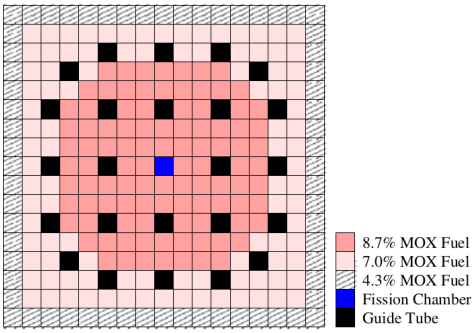

Fig. 4: Structure of the MOX assembly. Image reproduced from [5].

For the sake of comparison, we conducted the exercise with and without the transport correction for calculating $D_{0,g}$ with equations 11 and 7, respectively. Due to the problem's symmetry, the model included only a quarter of the core. The mesh had 2.4×10^4 elements. The simulation convergence criterion was 10^{-8} for the neutron flux.

Table III compares the eigenvalues obtained with the SP_3 solver and the references. For the method without correction, we used an eigenvalue reference value from Capilla et al. [5] as they conducted the exercise without correction. For the method with transport correction, we used the reference value from the benchmark [14].

	k_{Ref}	k_{SP_3}	Δ_ρ
No correction	0.96969	0.97106	145
Transport correction	0.93755	0.93792	43

TABLE III: Comparison between the results obtained with the SP_3 solver using no correction (equation 7), the transport correction (equation 11), and the reference results for the C5 MOX Benchmark.

The eigenvalues obtained with the SP_3 solver are within 145 pcm of the reference values. However, the difference between the reference values of the different schemes is 3535 pcm, suggesting that the use of the transport correction is necessary.

Our calculations included the pin power distribution, whose reference values the benchmark specifies [14]. For conciseness, we only calculated the power distribution using the transport correction. To simplify displaying the results,

Figure 5 presents the assembly power distribution with the largest pin power relative difference in each assembly. The MOX assembly hosts the largest pin power relative difference, its value being 1.8%.

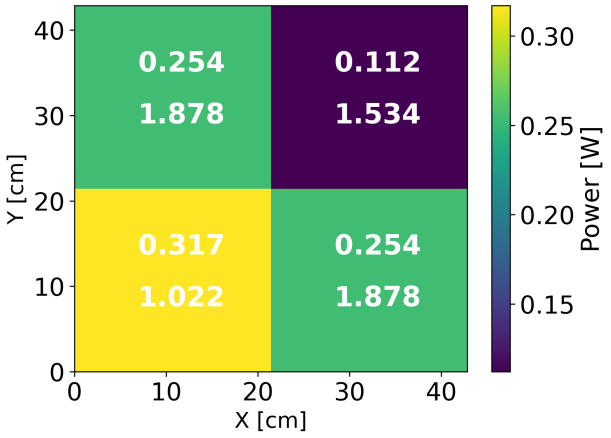

Fig. 5: Assembly power distribution in the C5 MOX Benchmark. Top: Assembly power distribution. Bottom: pin power relative difference expressed in %.

CONCLUSIONS

MOOSE is a computational framework that solves systems of nonlinear differential equations. As part of this work, we implemented the kernels to solve the steady-state SP_3 equations in a MOOSE-based application. Additionally, we carried out two exercises whose reference results were known. The first exercise solved a one-group eigenvalue problem with a simple geometry, with a result within 12 pcm. The second exercise studied the C5 MOX Benchmark, solving it using different approaches and obtaining results within 145 pcm. The calculated pin power values were within 2% error from the reference.

While the SP_3 equations solve the neutronics equations in a nuclear reactor, future work may develop other applications to solve the thermal-fluids or integrate this application to existing applications. MOOSE-based applications share the same API making their integration straightforward.

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