The Impact of Spatial Self-Shielding on Pin-wise Reaction Rates

William Boyd, Benoit Forget, Kord Smith

Massachusetts Institute of Technology, Department of Nuclear Science and Engineering, 77 Massachusetts Avenue, Building 24, Cambridge, MA 02139, United States

Abstract

Keywords: Multi-group cross-sections, spatial homogenization, spatia self-shielding

1. Introduction

The nuclear reactor physics community has long strived for deterministic neutron transport-based tools for whole-core reactor analysis. A key challenge for whole-core multi-group transport methods is accurate reactor agnostic multi-group cross section (MGXS) generation. The MGXS generation process applies a series of approximations to produce spatially homogenized and energy condensed MGXS in each spatial zone and energy group. Many approximations related to multi-group theory, including the selection of discretized energy group structures and the truncation of the Legendre expansion of the multi-group scattering kernel, are widely studied in the literature. However, the practical impact of the flux separability approximation, which permits the use of the scalar rather than the angular neutron flux to weight the continuous energy cross sections, is less understood. This paper investigates the flux separability approximation and quantifies its significance for heterogeneous PWR problems.

This work employs Monte Carlo (MC) neutron transport simulations to generate MGXS. Monte Carlo methods have increasingly been used to generate few group constants for coarse mesh diffusion, most notably by the Serpent MC code (Leppänen, 2013), and to a much lesser extent, for high-fidelity neutron transport methods (Redmond, 1997; Nelson, 2014; Cai, 2014; Boyd, 2016). The advantage of a MC-based approach is that all of the relevant physics are directly embedded into MGXS by weighting the continuous energy cross sections with a statistical proxy to the “true” neutron flux. However, this paper shows that even when the “true” scalar flux is used to generate MGXS, the flux separability approximation results in a non-negligible eigenvalue bias between continuous energy and multi-group calculations due to under-prediction of U-238 capture in resonance energy groups.

The content in this paper is organized as follows.

2. Methodology

2.1. Continous Energy Calculations with OpenMC

-OpenMC **?** -Python API **?** -NNDC data **?** -distribcell tallies **?** -`openmc.mgxs`

2.2. Multi-Group Calculations with OpenMOC

-OpenMOC **?** -multi-core parallelism **?**

2.3. Spatial Homogenization Schemes

2.3.1. Null Homogenization

2.3.2. Degenerate Homogenization

-OpenCG region differentiation **?**

3. Test Cases

-Benchmarks for Evaluation and Validation of Reactor Simulations (BEAVRS) **?**

-need to describe both pinwise fissoin rates and U-238 capture rates -latter papers can cite this one as reason to only analyze capture rates?

3.1. Benchmark Configurations

3.2. Single Fuel Assembly

3.3. Reflected Assembly Colorset

3.4. Verification Metrics

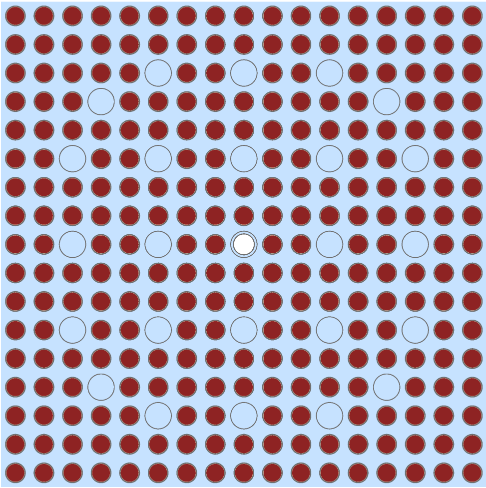
-reference results generated with OpenMC simulations (Sec. 7.3) -eigenvalues -pin-wise fission rates -pin-wise U-238 capture rates

4. Conclusions

Acknowledgments

This work was supported by the Idaho National Laboratory and the National Science Foundation Graduate Research Fellowship Grant No. 1122374. This research made use of the resources of the High Performance Computing Center at Idaho National Laboratory, which is supported by the Office of Nuclear Energy of the U.S. Department of Energy and the Nuclear Science User Facilities under Contract No. DE-AC07-05ID14517.

Email addresses: `wboyd@mit.edu` (William Boyd), `bforget@mit.edu` (Benoit Forget), `kord@mit.edu` (Kord Smith)



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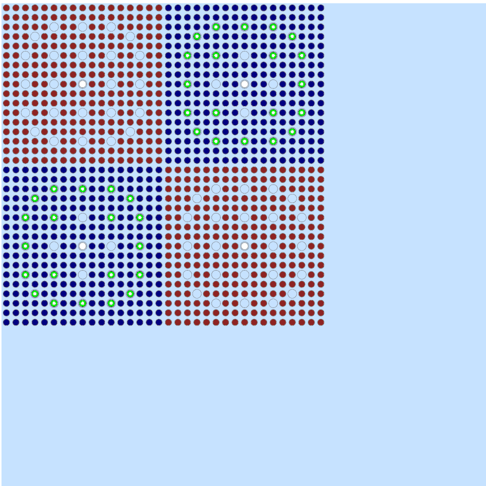

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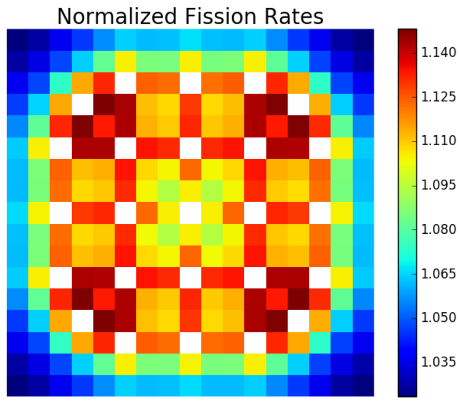
Figure 1. A (a) fuel assembly and (b) 2×2 assembly colorset with reflector.



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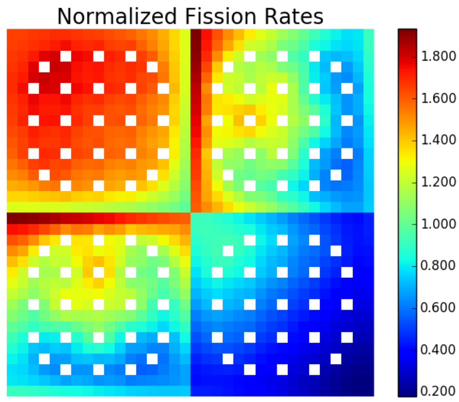

(b)

Figure 2. Fission rates for the (a) assembly and (b) 2×2 colorset.

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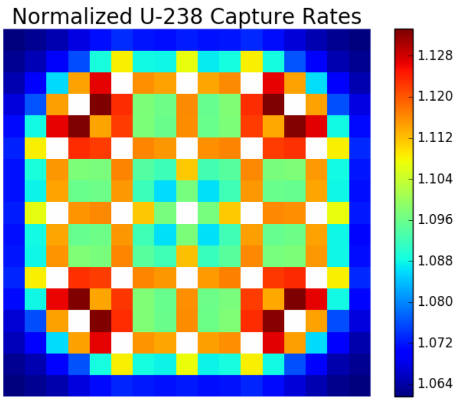

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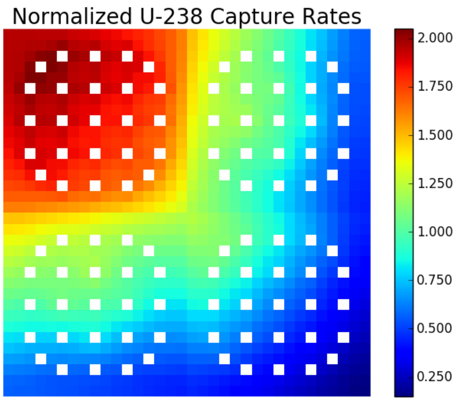

(b)

Figure 3. Capture rates for the (a) assembly and (b) 2×2 colorset.