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Stability and accuracy of 3D neutron transport simulations using the 2D/1D method in MPACT ☆

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ABSTRACT

A consistent "2D/1D" neutron transport method is derived from the 3D Boltzmann transport equation, to calculate fuel-pin-resolved neutron fluxes for realistic full-core Pressurized Water Reactor (PWR) problems. The 2D/1D method employs the Method of Characteristics to discretize the radial variables and a lower order transport solution to discretize the axial variable. This paper describes the theory of the 2D/1D method and its implementation in the MPACT code, which has become the whole-core deterministic neutron transport solver for the Consortium for Advanced Simulations of Light Water Reactors (CASL) core simulator VERA-CS. Several applications have been performed on both leadership-class and industry-class computing clusters. Results are presented for whole-core solutions of the Watts Bar Nuclear Power Station Unit 1 and compared to both continuous-energy Monte Carlo results and plant data.

1. Introduction

Significant advances in multi-core, multi-node computing clusters over the past several years have enabled the development of practical numerical methods for solving the Boltzmann Transport Equation (BTE) to calculate fuel-pin-resolved neutron fluxes in full 3D nuclear reactor cores. Historically, the nuclear reactor industry has relied upon a two-step procedure to solve for full-core power distributions – by using pre-generated few-group cross sections, homogenized over a fuel assembly, and low-order 3D nodal diffusion approximations to obtain neutron flux distribution throughout the reactor. Then, the detailed fuel pin powers (required for core design and safety analysis) are determined by post-processing the whole core

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neutron diffusion solution and utilizing pre-computed shape functions, to reconstruct the detailed intra-assembly flux and power distribution. For several decades, these methods – which provide reasonable accuracy and require limited computing resources – have been the workhorse of the nuclear industry. The use of more accurate full-core pin-resolved neutron transport methods to model the reactor requires significant additional computational resources [1–3], which are made available through the use of leadership-class computing facilities such as the U.S. Department of Energy INSITE program, which are capable of supporting applications using more than 100,000 compute cores and occasionally hardware accelerators such as GPUs. However, in order to deploy whole core transport methods to the nuclear industry, a methodology is needed that can run on industry-class computing clusters, which are typically between 500 and 5000 compute cores.

During the last few years, the Method of Characteristics (MOC) has become a nuclear industry standard for solving the transport equation for fuel assembly-sized problems [4,5], to generate the few-group homogenized cross sections for whole-core nodal diffusion methods [6]. Because of the computational appeal of MOC and the familiarity of the industry with this method, several researchers investigated the extension of MOC to larger 3D reactor problems [7,8,2]. However, it became evident that even with leadership-class computing platforms, the MOC method is too costly for these problems. A group of Korean researchers then investigated "2D/1D" methods, which utilize MOC in the 2D radial (x and y) directions and a lower-order transport solution in the 1D axial (z) direction [9,10]. This approach was motivated by the fact that most of the material heterogeneity in Light Water Reactors (LWRs) occurs in the radial directions, whereas the axial material heterogeneity is comparatively minimal.

The first 2D/1D method was introduced as the "2D/1D Fusion" method in the CRX code [11], which utilized a 2D MOC solution radially with a discrete-ordinates solution axially. Specifically, the radial 2D MOC method was discretized on a "fine" radial grid (in which each pin cell is divided into 50–100 "fine" spatial cells or "flat source regions"), while the axial solution was discretized on a "coarse" radial cell (consisting typically of one pin cell). The second major implementation of the 2D/1D method was in the DeCART code, developed at the Korean Atomic Energy Research Institute (KAERI) [9]. This method differed from the 2D/1D fusion method in that the axial solver was based on the diffusion approximation (and later, on SP_N) [10]. A more recent implementation of the 2D/1D method in DeCART has been the nTRACER code [12]. In the KAERI codes, the 2D MOC methods are also discretized on a fine radial grid, and the axial methods on a coarse radial grid. During the past few years, the KAERI 2D/1D methods have achieved success for practical reactor applications [13].

However, significant limitations in the numerical stability and accuracy were observed in DeCART, particularly when refining the axial mesh. None-the-less, the general concept of a 2D/1D method for whole-core reactor methods research provided a useful starting point when the U.S. Department of Energy (DOE) initiated the Nuclear Reactor Simulation Hub, CASL, in 2010. The first step in this development was to derive the 2D/1D equations directly from the 3D transport equation and to formalize the sequence of approximations used in the derivation. The result of this work was a numerically robust 2D/1D method that provided the foundation for the MPACT computer code – which has become the whole-core deterministic neutron transport solver for the CASL core simulator VERA-CS. The purpose of this paper is to describe the principal features of the 2D/1D method in MPACT.

MPACT is a 3D whole-core transport code based on the 2D/1D method. It provides pin-resolved flux and power distributions, which are important for the "challenge problems" in CASL. This fine spatial resolution is achieved by obtaining transport solutions for heterogeneous reactor problems in which the detailed geometrical configuration of fuel components, such as the pellet and cladding, is explicitly retained. The cross-section data needed for the calculation is obtained directly from a multigroup microscopic cross section library, similar to libraries used in lattice physics codes that generate the few-group cross sections for the full-core nodal simulators. Because MPACT involves neither a priori homogenization nor group condensation for the core spatial solution, it represents a significant advance in the fidelity and accuracy of the full-core flux solution, without compromising the stability and robustness required for industry applications.

To provide a sense of the magnitude of the 3D problems solved by MPACT and other 2D/1D codes, a typical reactor core contains:

- $N_{fa} \approx 200$ fuel assemblies,
- $N_{pc/fa} \approx 17 \times 17 = 289$ pin cells per fuel assembly.

Furthermore, a typical reactor core is discretized using:

- $N_{fc/pc} \approx 50$ fine 2D spatial cells ("flat source regions") per pin cell,
- $N_{fc} = N_{fc/pc} \cdot N_{pc/fa} \cdot N_{fa} \approx 2.4 \times 10^6$ fine 2D spatial cells,
- $N_{cc} = N_{pc/fa} \cdot N_{fa} \approx 6 \times 10^4$ coarse 2D spatial cells,
- $N_{ax} \approx 400$ axial cells,
- $N_{eg} \approx 50$ energy groups,
- $N_{aq} \approx 120$ directions in the angular quadrature set,
- $N_{tot} = N_{fc} \cdot N_{ax} \cdot N_{eg} \cdot N_{aq} \approx 10^{13} \approx$ total number of primary unknowns.

However, due to the details of the Method of Characteristics, which effectively imposes a finer radial spatial grid on the problem than the one indicated above, the actual total number of numerical unknowns is often approximately two orders of magnitude greater. For time-dependent simulations (reactor safety or depletion), the time-variable t must also be included,