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# Analysis of a multigroup stylized CANDU half-core benchmark

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ABSTRACT

An 8-group cross section library is provided to augment a previously published 2-group 3D stylized half-core Canadian deuterium uranium (CANDU) reactor benchmark problem. Reference eigenvalues and selected pin and bundle fission rates are also included. This benchmark is intended to provide computational reactor physicists and methods developers with a stylized model problem in more than two energy groups that is realistic with respect to the underlying physics. In addition to transport theory code verification, the 8-group energy structure provides reactor physicist with an ideal problem for examining cross section homogenization and collapsing effects in a full-core environment. To this end, additional 2-, 4- and 47-group full-core Monte Carlo benchmark solutions are compared to analyze homogenization-free transport approximations incurred as a result of energy group condensation.



## 1. Introduction

A 3D stylized half-core Canadian deuterium-uranium (CANDU) benchmark problem and solution was recently developed and published (Pounders et al., 2010). This benchmark was specifically designed to provide transport theory methods developers with a realistic large-scale reactor problem. The benchmark core preserves the characteristic physics (heavy water moderated natural uranium fuel), geometry (horizontally extended cylindrical fuel bundles), and heterogeneity (alternating channel burnup distributions and interstitial adjuster rods) that is typical in CANDU reactors. To promote the versatility of the benchmark problem, the description of the core was simplified by reducing the number of discrete bundle burnup points to eight and by limiting the number and type of adjuster rods. A 2-group macroscopic cross section library was also furnished with the core description to facilitate code-to-code comparisons that are not biased by library differences, and reference Monte Carlo solutions were presented based on that library.

The present work provides an 8-group macroscopic cross section library and Monte Carlo reference solution to augment the previous 2-group benchmark problem. Two-, four-, and forty-seven-group solutions are also compared to support analyses of multigroup collapsing and the transport approximations that result from energy condensation. These analyses isolate the effect of group condensation by avoiding any spatial homogenization of the cross sections. The two energy groups of the previous library represented fast (>0.625 eV) and thermal (<0.625 eV) neutron interactions; the 8-group structure presented in this work was obtained by dividing the above 0.625 eV group into six subgroups and the thermal group into two subgroups. Two different 4-group structures are investigated that are intermediate between the two and eight group schemes. All of these multigroup solutions are compared to a 47-group solution under cooled, checkerboard voided, and fully voided configurations. The benchmark core geometry and configuration is identical to that given previously.

This addition to the 2-group benchmark problem is valuable because it gives transport theory methods developers the ability to test new methods on a realistic full-core problem with more than two energy groups. Additionally, reactor physics methods developers may use this stylized benchmark to investigate the impact of cross section collapsing on few-group transport solutions—effects which are highlighted in Section 3. Although this benchmark represents an idealization of operating CANDU reactors, the underlying core physics has been preserved. The net result, therefore, is a realistic problem unencumbered from operational details that do not affect neutronics—the ideal situation for methods development benchmarks.

## 2. Benchmark description

A detailed description of the benchmark geometry and configuration has been provided in a previous publication (Pounders et al., 2010) and for the sake of brevity will not be repeated here. This section will, however, present a summary of the benchmark core highlighting the significant components and features.

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The stylized benchmark problem consists of a half-core symmetric rectangular lattice of natural uranium fuel bundles with a lattice pitch (LP) of 28.575 cm. The fuel lattice is 22 LPs in width at the core mid-plane. Each fuel lattice cell consists of a single fuel bundle that has been uniformly burned (under a zero-current boundary approximation) to one of the eight following points: 32.69, 78.38, 342.37, 818.87, 1638.73, 3608.15, 6381.44 and 8721.49 MWd/tU. The distribution of these burnups in the core maintains a realistic degree of heterogeneity because it is consistent with the fuel loading pattern of CANDU reactors: online refueling from both ends of the core in an alternating checkerboard pattern. The fuel lattice is surrounded in the radial direction by a heavy water reflector that extends at least two lattice pitches beyond the outer most fuel cells.

For reactivity control and power shaping, the benchmark contains one type of adjuster rod at 21 interstitial locations within the core. The adjuster rods, which are inserted perpendicularly to the fuel channels, consist of stainless steel shims in zirconium guide tubes. The detailed dimensions of these rods are given in Pounders et al. (2010). A cross sectional view of the benchmark core that includes adjuster rods is shown in Fig. 1.

The library for this benchmark problem includes cross sections for fuel, clad, coolant, pressure tube, calandria tube, and moderator materials at each of the eight burnup points, as well as cross sections for reflector and adjuster rod materials. The fuel region was divided into four material regions corresponding to the central fuel pin plus the three rings of fuel pins surrounding it. The collision probability code HELIOS, version 1.8 (Villarino et al., 1992) was used to calculate burnup-dependent cross sections for each of the nine unique fuel bundle materials. The cross section generation process began with using HELIOS to deplete a fresh natural uranium fuel bundle (with specular reflection boundary conditions) to each of the eight burnup points of interest. At each of these points, the intra-bundle flux distribution was used to collapse region-wise cross sections. The 2-, 4-, and 47-group structures are discussed in the following section.

Reflector cross sections were calculated by extending a fuel lattice cell (depleted to 4000 MWd/tU) by 68 cm in one direction, and homogenizing an area corresponding to one lattice pitch immediately adjacent to the fuel lattice. For a detailed description of this procedure, the reader is referred to Pounders et al. (2010).

## 3. Analysis of group collapsing

### 3.1. Computational results

To demonstrate the effect of using few-group cross sections, the full-core problem was solved using the original 47-group structure, an 8-group structure, two different 4-group structures, and a 2-group structure. The multigroup energy boundaries are shown in Table 1. No spatial homogenization was performed prior to the full-core calculations to isolate the effect of group condensation. Also uranium and plutonium resonances are captured implicitly in the 47-group macroscopic cross sections; no resonance treatments were made beyond the 47-group single-bundle calculations so that attention could be limited on computational transport effects.

For each group structure, full-core Monte Carlo solutions (for cooled, voided and checkerboard voiding configurations) were computed using the code MCNP (2003). The fission source was converged by sampling 500 million particle histories, starting with an initial source distribution of one fission site per rod per bundle throughout the core. An additional 500 million particle histories were sampled for tallying fission densities and estimating the eigenvalue ($k_{eff}$). Table 2 provides the eigenvalue ($k_{eff}$) results for the three states. For ease of comparison, the 8-, 4-, and 2-group eigenvalue differences relative to the 47-group result are shown in Table 3. The coolant void reactivities (CVRs), provided in Table 4, are calculated as follows:

$$\text{CVR } [\text{pcm}] = \left(\frac{1}{k_{\text{cooled}}} - \frac{1}{k_{\text{voided}}}\right) \times 10^5.$$

From these comparisons it is seen that, as expected, the 8-group eigenvalue is closer to the 47-group result than the two and four group results. The 8-group difference is not uniform across the void configurations, however, so the CVR discrepancy, in this case, is not best predicted with this structure.

The distribution of the few-group bundle-averaged-fission-rates was also compared to the 47-group solution. The average and maximum relative differences are shown in Table 5. Pin fission rates were calculated for all channels in planes 13 and 21 of the problem. (See Appendix A for the numbering scheme.) Table 6 shows the relative differences of the pin fission rates with respect to the 47-group solution. The maximum bundle and pin differences all occurred adjacent to either an axial or radial core boundary, where the neutron spectrum is farthest from the infinite medium

**Table 1**
Upper energy bounds for collapsed group structures (MeV).

| 8-Group | 4-Group (A) | 4-Group (B) | 2-Group |
|---|---|---|---|
| 2.0000E+01 | 2.0000E+01 | 2.0000E+01 | 2.0000E+01 |
| 2.2313E+00 | | | |
| 8.2085E−01 | | | |
| 1.8316E−01 | 1.8316E−01 | 1.8316E−01 | |
| 9.1188E−03 | | | |
| 1.3007E−04 | | 1.3007E−04 | |
| 6.2506E−07 | 6.2506E−07 | 6.2506E−07 | 6.2500E−07 |
| 1.1157E−07 | 1.1157E−07 | | |

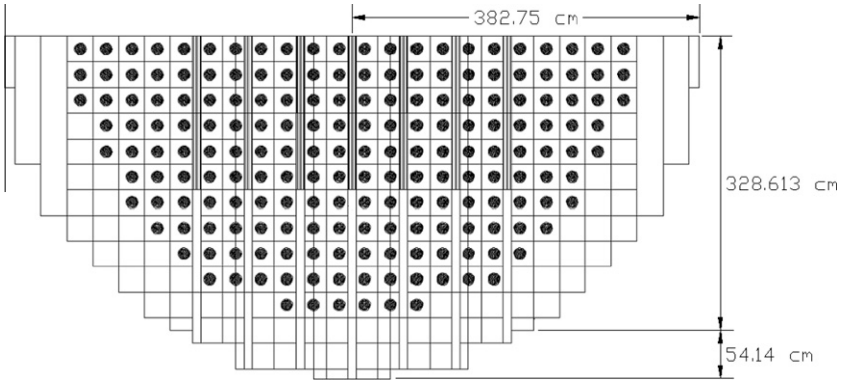


**Fig. 1.** Cross sectional view of stylized core model.