LA-UR- 02-161 c.1

Approved for public release; distribution is unlimited.

Title: PARTISN RESULTS FOR THE C5G7 MOX BENCHMARK PROBLEMS

Author(s). J.A. Dahl
R.E. Alcouffe

Submitted to PHYSOR 2002, International Conference on the New Frontiers of Nuclear Technology: Reactor Physics, Safety and High-Performance Computing, October 7-10,2002, Seoul, Korea

Los Alamos
NATIONAL LABORATORY

Form 836 (8/00)

PARTISN Results for the C5G7 MOX Benchmark Problems
(invited)

Jon A. Dahl and Raymond E. Alcouffe
Transport Methods Group
Los Alamos National Laboratory
Los Alamos, NM 87545

Introduction

In early 2001 the Nuclear Energy Agency solicited participants for a proposed new benchmark [1]. The benchmark, known as C5G7 MOX, is intended to be a basis to measure current transport code abilities in the treatment of reactor core problems without spatial homogenization. We have participated with the code transport code PARTISN [2]. PARTISN (PARallel TIme Dependent SN), PARTISN solves the linear Boltzmann transport equation in static and time dependent forms on one, two and three dimensional orthogonal grids using the deterministic (S_N) method. A variety of spatial discritization methods are incorporated into PARTISN, however all calculations performed here used the diamond difference approach, coupled with a volume fraction method for non-Cartesian problem geometries. Acceleration of the source iterations is accomplished with diffusion synthetic acceleration (DSA).

Description of Work

For the two dimensional C5G7 MOX problem we used an X-Y Cartesian grid. Rather than creating a "stair stepped" grid to represent each fuel pin, we began with an unstructured quadrilateral grid created using ICEM CFD Engineering grid generation tools. This unstructured grid more closely represents the actual geometry of each water channel-fuel pin combination. We then overlay a Cartesian mesh onto the unstructured grid and calculate to volume fractions of each material where a Cartesian mesh cells is intersected by an unstructured mesh material boundary. This methodology allows the preservation of mass without changing the density in cells which are not intersected by actual material interfaces. We generated three grids for the two dimensional case, a coarse, medium, and fine, corresponding to a 5x5, 10x10 and 15x15 grid in each water channel-fuel pin cell, and 30 mesh cells in each direction of the water reflector.

The grid for the three dimensional problem was generated in the same way, beginning with a three dimensional unstructured mesh and calculating the corresponding volume fractions for Cartesian X-Y-Z mesh cells intersecting material interfaces. We calculated on two three dimensional grids, one coarse and one medium, which, as in the two dimensional case, contained 5x5 and 10x10 mesh cells respectively in each water channel-fuel pin cell, and 30 mesh cells in each direction of the water reflector. The coarse mesh fuel region contained **20** cells in the Z direction, for a total of 200x200x50 or **6** million cells. The fuel region of the medium mesh was comprised of 50 cells in the Z direction for a total of 370x370x80, or 10,952,000 cells. A fine three dimensional mesh we also generated, consisting of 17,496,000 cells, but two dimensional convergence studies indicated that a mesh this fine is not required.

We ran each problem using the square Tchebechev-Legendre quadrature set [3] and diamond difference spatial differencing. The pointwise fluxes were converged to an error of 1.0E-05. The diffusion equations for the DSA were solved using a parallel multigrid technique described by Alcouffe [4]. Varying S_N orders for each mesh were run to determine that spatial and angular convergence was achieved.

Both the two and three dimensional problems were run on the Bluemountain computer, located at the Los Alamos National Laboratory. Bluemountain consists of 48 SGI Origin 2000 boxes with 128 processors each. The two dimension problems were run with 16 processors, while the three dimensional problems were run with 126 to 768 processors, depending on mesh size and S_N order.

Computational Results

Table 1 presents the two dimensional k eigenvalue results for varying S_N orders for the coarse, medium and fine meshes. Table 2 gives the three dimensional k eigenvalues for varying S_N orders for the coarse and medium meshes. These eigenvalues exhibit that the calculation is generally converged to a precision of 1.0E-4 with an S_N order of 26 for this quadrature set. We also notice that the differences between the coarse and finer meshes are small, indicating that the angular dependence in stronger that the spatial.

S_N Order	Coarse Mesh	Medium Mesh	Fine Mesh
4	1.18540	1.18496	1.18459
6	1.18554	1.18478	1.18468
8	1.18572	1.18489	1.18477
12	1.18621	1.18578	1.18556
16	1.18648	1.18596	1.18599
20	1.18665	1.18626	1.18607
26	1.18672	1.18637	1.18628
32	1.18677	1.18637	1.18630
40	1.18680	1.18641	1.18632

S_N Order	Coarse Mesh	Medium Mesh
4	1.18220	1.18117
6	1.18284	1.18153
8	1.18303	1.18184
12	1.18352	1.18264
16	1.18379	1.18289
20	1.18396	1.18344
26	1.18401	1,18362

In Table 3, the normalized minimum and maximum pins powers for the two dimensional coarse and medium mesh and the three dimensional coarse mesh are shown. These results were generated with a quadrature order of S_{26}. Pin power results for the three dimensional medium mesh will be presented when available.

Mesh	Maximum	Minimum
2-D Coarse Mesh	2.5231	0.2318
2-D Medium Mesh	2.5025	0.2319
3-D Coarse Mesh	2.5223	0.2317

Acknowledgements

We wish to thank Todd Wareing for generating the unstructured quadrilateral grid with ICEM CFD Engineering software.

This work was performed under the auspices of the Department of Energy.

References

1. E.E. Lewis, et. Al., "Benchmark Specification for Deterministic 2-D/3-D MOX Fuel Assembly Transport Calculations Without Spatial Homogenization," Nuclear Energy Agency, Nuclear Science Committee, Expert Group on 3-D Radiation Transport Benchmarks Report, NEA/NSC/DOC(2001)17, (2001).

2. R. E. Alcouffe, R. S. Baker, J. A. Dahl,, and S. A. Turner, "PARTISN Code Abstract," Physor 2000 International Topical Meeting, Advances in Reactor Physics and Mathematics and Computation into the Next Millennium, Pittsburgh, (2000).

3. R. D. O'Dell, R. E. Alcouffe, "Transport Calculations for Nuclear Analysis: Theory and Guidelines for Effective Use of Transport Codes, LA-10983-MS, Los Alamos National Laboratory, September 1987.

4. R.E. Alcouffe, "A Parallel Multigrid Method for Inversion of the Diffusion Operator in Neutronics Applications," Proceedings ANS International Meeting on Mathematical Methods for Nuclear Applications, Salt Lake City, UT, September (2001).