An Evaluation of Eigenvalue Calculations Using Three-dimensional Deterministic Methods

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INTRODUCTION

The growing interest in next generation reactors is producing advanced fuel and reactor system designs of significant heterogeneity. These designs require more advanced transport methods, especially methods that are capable of flexible and accurate three-dimensional modeling. The current standard for accurate 3-D modeling and analysis is based on Monte Carlo methods. While Monte Carlo methods can provide benchmark quality results, their routine application for production calculations is not generally practical. As an alternative, TransWare Enterprises Inc. is developing advanced transport software based on 3-D deterministic methods.

This paper describes TransWare's new TransLAT [1] 3-D lattice physics software and provides an evaluation of the software for solving light water reactor fuel lattices. Initial testing of the new TransLAT 3-D software has focused on solving simple pin cell problems, although more complex problems are being investigated. In all instances, the results calculated with TransLAT are compared to the MCNP4C [2] Monte Carlo code.

METHODOLOGY

The TransLAT 3-D lattice physics software is a nuclear fuel burnup code that is capable of modeling highly heterogeneous fuel designs in three-dimensional geometry. TransLAT is a full featured software package that includes extensive user input and error checking capabilities, resonance self-shielding treatments, fuel burnup calculations, and editing calculations. The significant features tested in this paper include the 3-D geometry modeling capabilities and the performance of the Dancoff, resonance, transport and eigenvalue calculations.

The geometry modeling technique in TransLAT is based upon the MARS [3] software package. The MARS methodology provides a very general and flexible approach for describing three-dimensional geometry and is easily interfaced to solution techniques using ray-tracing methods. In TransLAT, ray-tracing is used to perform the Dancoff and particle transport calculations.

TransLAT incorporates an integrated Dancoff and resonance treatment capability for determining resonance self-shielding. The resonance calculation is based upon the space-dependent resonance self-shielding method described by Williams, et al [4]. The Dancoff treatment described by Williams is very flexible and easily handles regular and irregular lattices in two-dimensional geometry. For TransLAT, the Dancoff treatment has been extended to also handle three-dimensional geometries. The 3-D Dancoff treatment shows excellent agreement with comparable 2-D treatments and has recently been verified for spherical fuel designs [5].

The resonance self-shielding method described by Williams assumes fuel lumps of right circular cylinder design in two-dimensional planar space. The current formulation, although restrictive for more exotic fuel designs, is acceptable for three-dimensional modeling providing that the fuel lumps are described as right circular cylinders and the axial meshing of the fuel is specified explicitly by user input.

TransLAT supports two deterministic transport modules for calculating neutron and gamma-ray fluxes. The primary transport solver is based on the Method of Characteristics technique and is capable of solving both 2-D and 3-D transport problems. The second transport solver is based on the Method of Collision Probabilities, which performs only 2-D transport calculations. Results for both transport solvers are presented in this paper, as appropriate for the test problems.

The comparisons between TransLAT and MCNP are facilitated by a model translation routine in TransLAT that generates MCNP inputs from the TransLAT model description. To further minimize discrepancies in the comparisons, new cross sections for the ^{235}U and ^{238}U fuel isotopes are generated from ENDF/B-VI release 5 for both codes.

RESULTS

Three test configurations are evaluated in this paper. Table 1 illustrates the problem geometries and lists the results of the TransLAT and MCNP calculations for the test configurations.

The first test configuration is a simple 3-region pin cell which is easily solved with 2-D and 3-D methods. This problem provides an excellent cross comparison of the different transport methods in the TransLAT software. Table 1 shows that both TransLAT 2-D solvers produce excellent agreement with MCNP. The 3-D calculation shows a slight under-prediction of the eigenvalue by 0.0030 delta K.

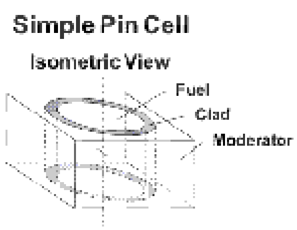
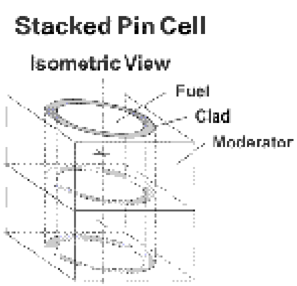
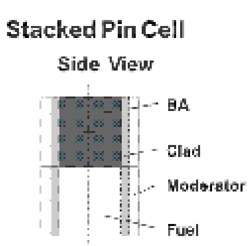
The second test configuration is a column of fuel comprised of two simple pin cells. This test configuration is functionally identical to the first test configuration, except that a column of pin cells are modeled to further test the axial integration capability of the TransLAT 3-D solver. Consistent with the single pin cell problem, the TransLAT 3-D solver under-predicts the eigenvalue by 0.0022 delta K.

The third test configuration is a column of pin cells similar to the second test problem, except the top pin cell is replaced with a burnable absorber pin cell. This configuration poses a more challenging calculation of the TransLAT 3-D solver. The resulting zebra-style fuel configuration is run with both the TransLAT 3-D solver and MCNP. Table 1 shows that TransLAT over-predicts the eigenvalue by 0.0022 delta K.

CONCLUSIONS

The excellent agreement between the TransLAT 2-D results and MCNP provides confidence that the modeling approach and cross sections are properly implemented in TransLAT. The 3-D results provide a strong indication that the 3-D deterministic methods are capable of producing very accurate results as well. Testing will continue with TransLAT to evaluate more complex problems.

Table 1 TransLAT and MCNP4C Results for Three Test Configurations

Geometry Model	Code / Method (1,2)		K-Infinite	Delta K (LAT - MCNP)
	**MCNP4C**		1.34132 ± 0.00039	
	TransLAT	MCPS-2D	1.34205	0.0007
		MOCS-2D	1.34191	0.0006
		MOCS-3D	1.33836	-0.0030
	**TransLAT**	MOCS-3D	1.33908	-0.0022
	**MCNP4C**		0.52231 ± 0.00029	
	TransLAT	MOCS-3D	0.52446	0.0022

1. MCNP cases are run to 2 million histories.
2. MCPS-2D are two-dimensional, Method of Collision Probabilities cases.
   MOCS-2D are two-dimensional, Method of Characteristics cases.
   MOCS-3D are three-dimensional, Method of Characteristics cases.

REFERENCES

1. TransLAT 3-D Lattice Physics Software, TransFX Computer Software Manuals, TWE-TFX-001, June 2001.

2. J. F. Briesmeister, Ed., “MCNP - A General Monte Carlo N-Particle Transport Code, Version 4C,” LA-13709-M (April 2000).

3. J. T. West and M. B. Emmett, “MARS: A Multiple Array System Using Combinatorial Geometry,” Oak Ridge National Laboratory, Radiation Shielding Information Center (Dec 1980).

4. M. L. Williams and R. Raharjo, “Space-Dependent Resonance Self-Shielding,” *Nucl. Sci. Eng.*, **126**, 19 (1997).

5. Charlotte H. Potze, Steven. P. Baker, and Dean B. Jones, “Dancoff Calculations for PBMR Fuel Using TransLAT 3-D Lattice Code,” *Trans. Am. Nucl. Soc.*, submitted for publication (2001).