JEFDOC-1222: Development of the PSG Monte Carlo Reactor Physics Code

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Outline

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History of PSG

- * The development of PSG (Probabilistic Scattering Game) started in September 2004.
- * First results in January 2005 (comparison to MCNP).
- * VTT Research project since April 2005.
- * Doctoral thesis finished in June 2007
- * Related conference papers:
  - XII Meeting on Reactor Physics Calculations in the Nordic Countries, Halden, Norway, May 17-18, 2005.
  - M&C 2005, Avignon, France, Sept. 12-15, 2005.
  - PHYSOR-2006, Vancouver, BC, Canada, Sept. 10-14, 2006.
  - M&C 2007, Monterey, CA, USA, April 15-19, 2007.

Background

- PSG is a Monte Carlo neutron transport code, specifically intended for reactor physics calculations:
  - Main motivation for code development is the production of homogenised input parameters for deterministic reactor simulator (nodal diffusion) codes.
  - Research tool to be used in parallel with the production codes (MCNP, CASMO, etc.).
  - Development of PSG "Version 2" with burnup capability currently under way.
- The long-term objective of the project is to develop a fast Monte Carlo neutron transport code to meet the future needs in reactor physics calculations.
- New fields of application may arise during development.

Code description: overview

* Summary of main features:

- Analog Monte Carlo game
- k-eigenvalue criticality source calculation
- Woodcock Delta-tracking method for neutron transport
- Neutron interactions based on classical collision kinematics and ENDF reaction laws (all reaction channels and thermal scattering).
- Interaction data read from ACE-format cross section libraries
- Parallel calculation using MPI
- PSG is capable of calculating all input parameters needed in few-group nodal diffusion calculations: multiplication factors, homogenised few-group cross sections, scattering matrices, diffusion coefficients, assembly discontinuity factors, kinetic parameters, effective delayed neutron fractions and decay constants.

Code description: geometry routine

- PSG has a generalised 3D geometry model. Neutron transport (in "version 1") entirely based on a modified delta-tracking method[a].
- Advantages of delta-tracking:
  - Simplified geometry routines
  - Potentially very efficient in complex geometries
  - Enables some special features (inhomogeneous material compositions etc.)
- Disadvantages:
  - Neutron track lengths not recorded (flux integrals must be calculated using the collision estimator)
  - Efficiency problems in geometries containing localised heavy absorbers

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[a] The delta-tracking method was developed by E. R. Woodcock in the 1960's and is used in M/C codes MONK and MCBEND (the HOLE geometry package), RCP01, RACER, and VMONT.

Code description: interaction physics

- Continuous-energy interaction data is read from ACE format cross section libraries.
- Cross sections reconstructed using a uniform energy grid for all nuclides:
  - Significant speed-up in calculation (grid search has to be performed only once after neutron changes its energy)
  - Dramatic increase in memory usage (large number of redundant data points stored)
- Memory usage is generally not a limiting factor for today’s PCs (>2GB RAM).
- Problems may be encountered, however, if number of nuclides is very large (burnup calculation).

Group Constant Generation Using Monte Carlo

- Interest in the use of Monte Carlo codes for the production of input parameters for deterministic simulator codes has increased over the past few years. Previous studies:
  - R. C. Gast 1981
  - E. L. Redmond II 1997
  - G. Ilas and F. Rahnema 2003
  - M. Tohjoh et al. 2005
  - S. C. van der Marck 2006
  - J. E. Hoogenboom et al. 2007
- Related research topics:
  - Calculation of effective delayed neutron fractions (R. K. Meulekamp and S. C. van der Marck 2006).
  - Development of Monte Carlo burnup calculation codes: Monteburns (LANL), MCB (KTH), ABURN (VTT), etc.

Group constant generation using Monte Carlo

- Calculation of homogenised reaction cross sections:

$$\Sigma_i = \frac{\displaystyle\int_V \int_{E_g}^{E_{g-1}} \Sigma_i(\mathbf{r},E)\phi(\mathbf{r},E)d^3r dE}{\displaystyle\int_V \int_{E_g}^{E_{g-1}} \phi(\mathbf{r},E)d^3r dE}$$

  and assembly discontinuity factors is trivial and can be performed by any Monte Carlo code.

- Calculation of group-transfer cross sections ($\Sigma_{\mathrm{s},g'\rightarrow g}$) is relatively simple.

- Methods for calculating kinetic parameters and β_{eff} exist (Meulekamp & van der Marck 2006).

$\Longrightarrow$ The problem is basically reduced to the calculation of the neutron diffusion coefficient, which is the only parameter without any continuous-energy counterpart in the Monte Carlo calculation.

Example results

- PSG has mainly been validated in LWR lattice calculations by comparing to MCNP4C and CASMO-4E. Typical results:
  - Differences in k_∞ < 100 pcm compared to MCNP, < 500 pcm compared to CASMO.
  - Homogenised reaction cross sections in the fast energy group (E > 0.625 eV) within statistical accuracy from MCNP results. Larger discrepancies (up to $\sim$0.5%) in thermal group.
  - Diffusion coefficients within 5-10% of CASMO results. Differences in group-wise reaction cross section < 5%.
  - PSG runs about 5 to 15 times faster than MCNP4C in LWR lattice calculations.
- Group constants generated by PSG successfully used in nodal diffusion reactor simulator calculations (conceptual EPR initial core at HZP conditions).

Example results

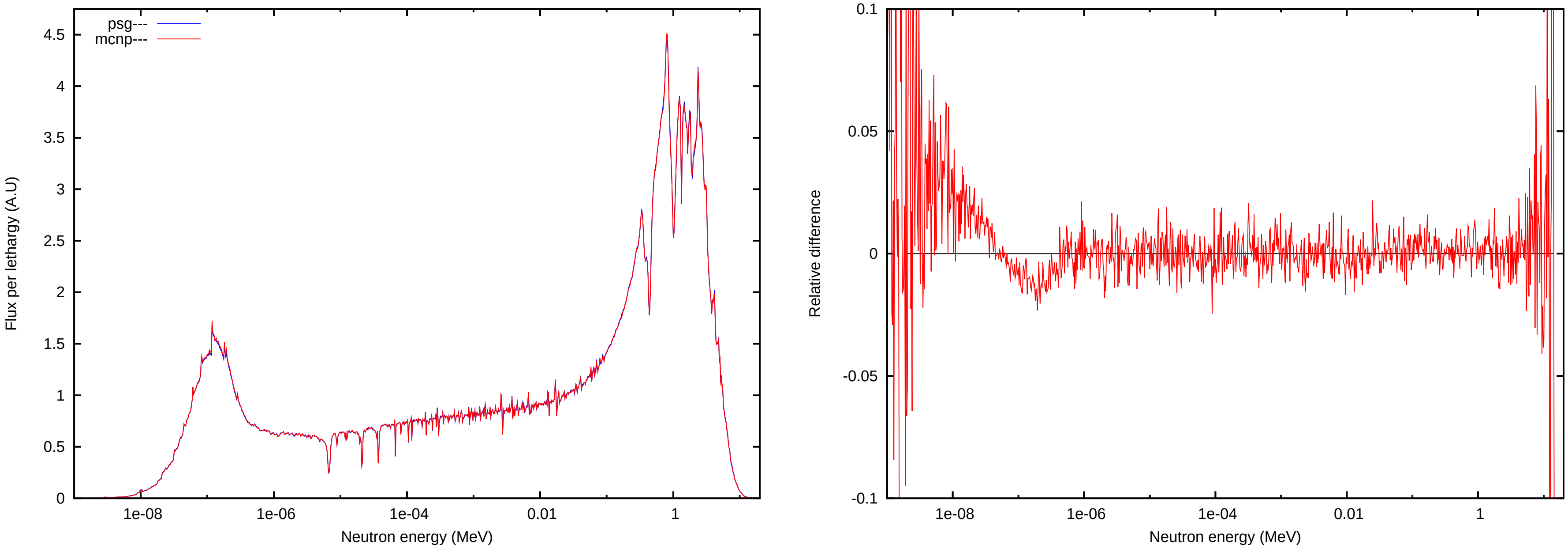

Figure 1: Flux spectra in a VVER-440 fuel assembly (left) and comparison between MCNP and PSG results (right).

Example results

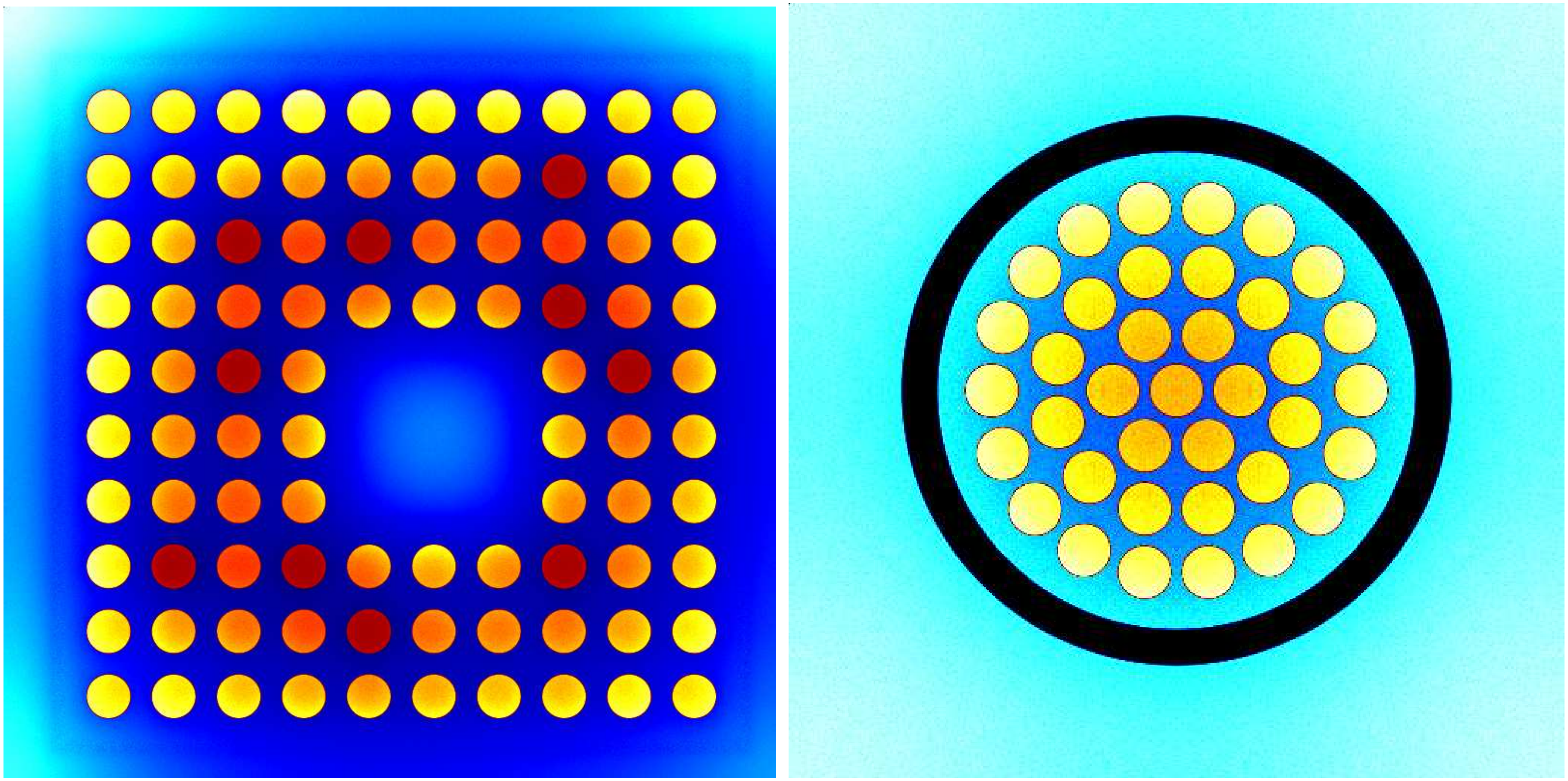

Figure 2: Relative fission rate (“hot” shades) and thermal flux distributions (“cold” shades) in BWR and CANDU lattice calculations.

Example results

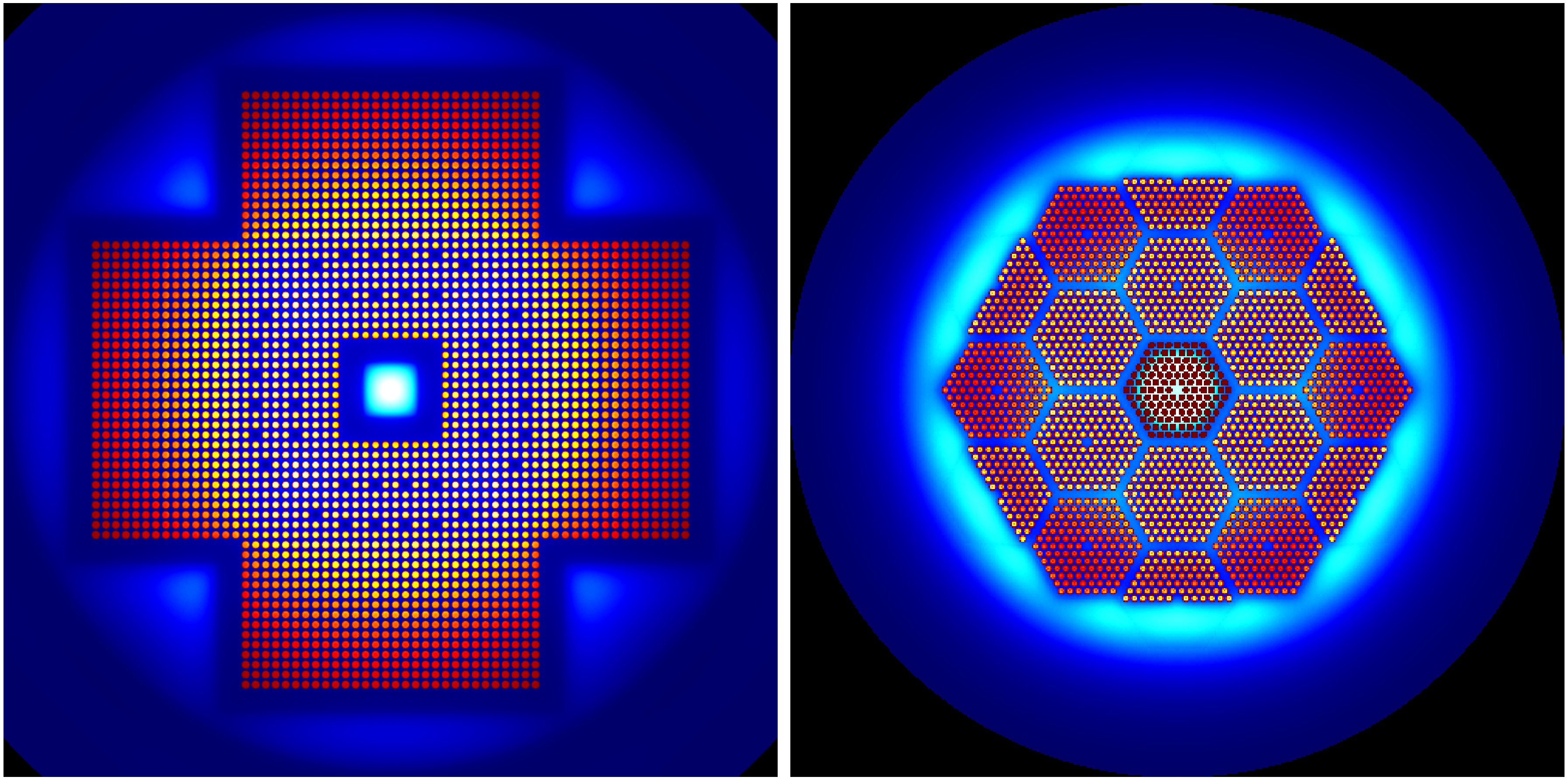

Figure 3: Relative fission rate (“hot” shades) and thermal flux distributions (“cold” shades) in VENUS-2 and LR-0 cores.

References

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Thank you for your attention !