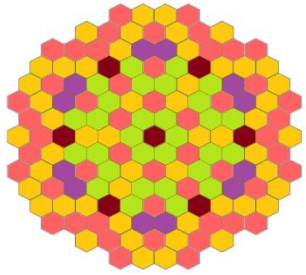
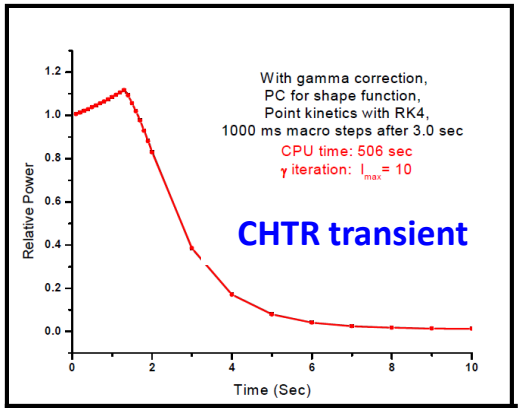
Reactor Space-Time Analysis Code 'ARCH' (AG)

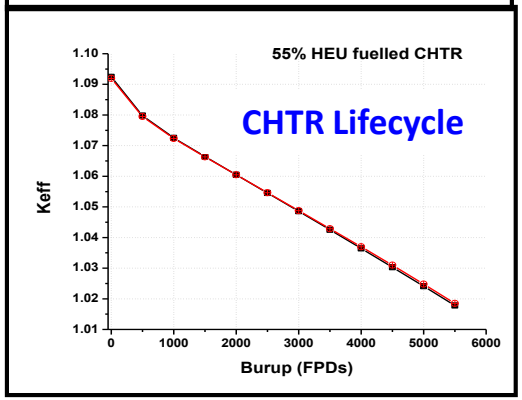
Analysis of **R**eactor Transients in **C**artesian and **H**exagonal Geometries

- **Being used in CHTR, HTRs, AHWR, IPWR, IMSR**
- Rectangular & Hexagonal lattices, K-eff, Source
- Reactor Transients with delayed neutrons
- Xenon spatial Oscillations & Transients
- Fuel Burnup, Core Followup, Eigenmodes
- Refuelling & Fuel Reshuffleing
- Krylov based solution methods, F90/95
- Thermal Hydraulics feedbacks

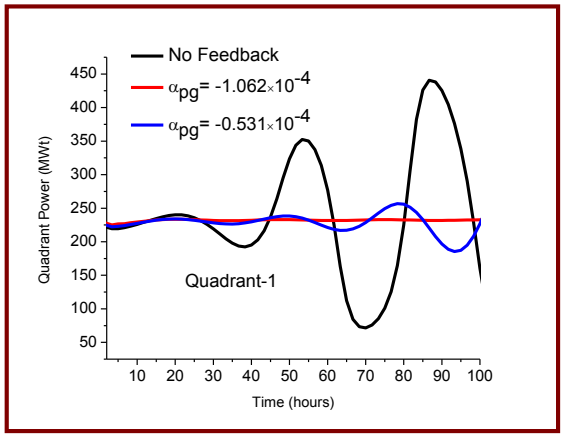


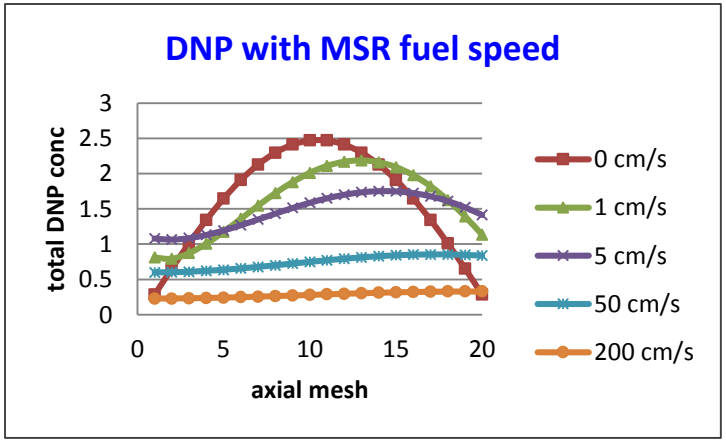
Reload 2 of IPWR

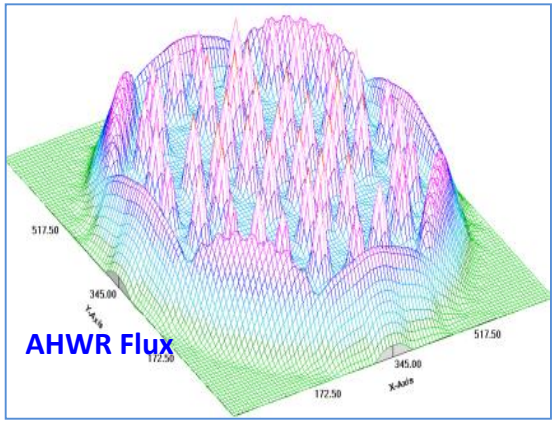




Xenon Oscillationwith feedbacks



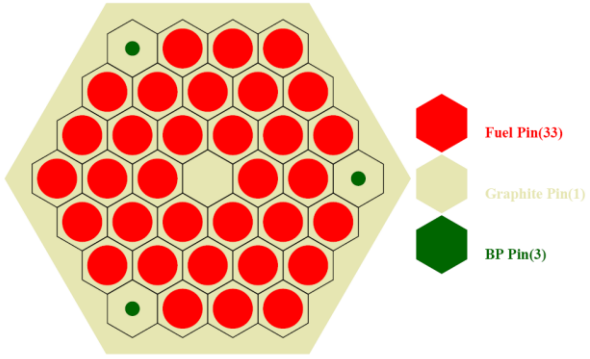




Development of VISWAM and TRANPIN code (SAK)

VISWAM Code

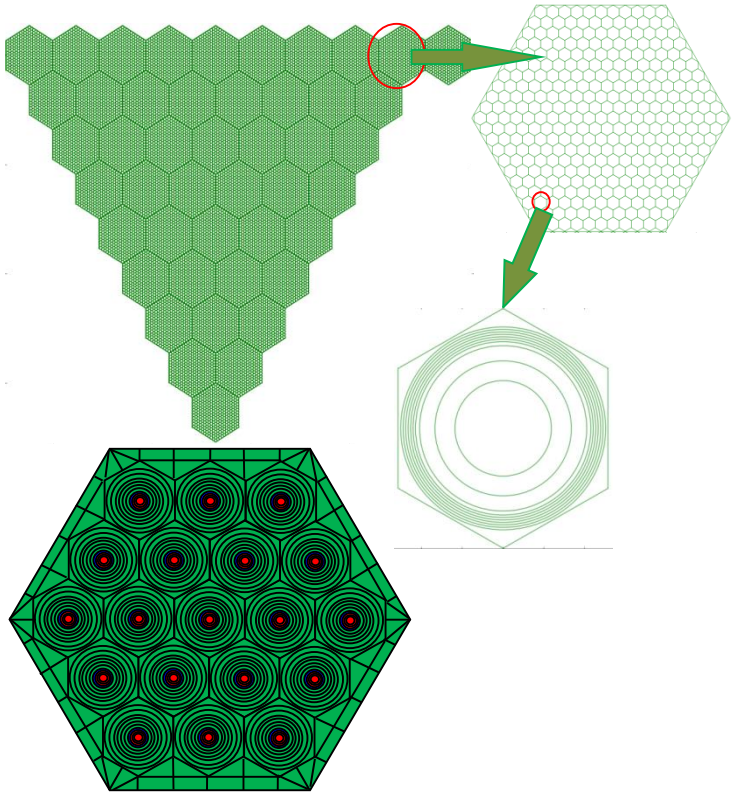
- The new method is the interface current method based on 2D collision probability (CP) developed for heterogeneous lattices, where strong flux gradients are encountered.
- 2D CP method at individual lattice cells level and different lattice cells are linked using interface currents with double P2 (DP2) expansion of angular flux at cell boundaries
- Benchmarks analysed : HTTR benchmark problem, OECD/NEA C5G7 MOX assembly benchmark problem, VVER-1000 OECD LEU and MOX Computational Benchmark



HTTR benchmark problem

TRANPIN Code

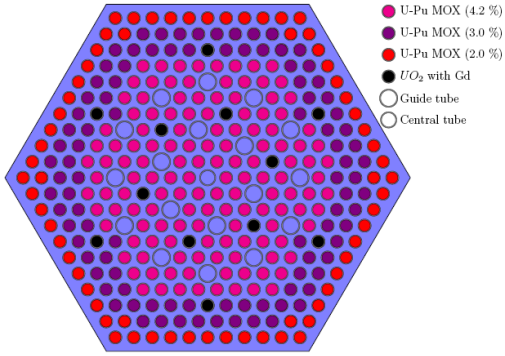
- To perform the whole core pin-by-pin calculation in 2D hexagonal geometry advanced from the conventional 2-step process
- A more exact simulation
- Solves the transport equation for the full core using the interface current method based on 2D collision probability (CP) method
- the heterogeneous lattice structure of fuel rod and absorber rod cells are sub divided into finer regions
- The zone to zone coupling in the lattice cell is achieved using region to region CPs.
- The coupling between the cells in the same FA and cells of different FAs is achieved by expanding the angular flux leaving or entering a lattice cell into a finite set of linearly independent functions
- Successfully tested for VVER



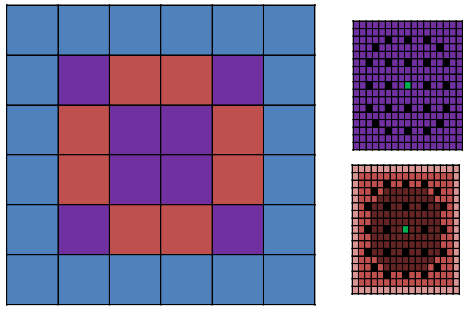
TRANPIN Pin–by–Pin Core Simulation Approach

PATMOC :: A multi-group Monte Carlo code for neutron transport

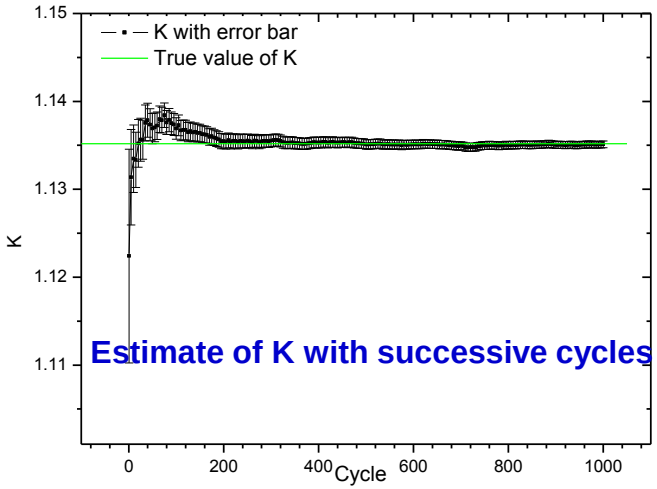
- **Multi-group Monte Carlo code for neutron transport simulation**
- **Current capabilities and features:**
  - Eigen mode simulation to compute K-inf/K-eff and tallies like flux and reaction rates.
  - Fixed source mode simulation Can handle both non-fissile and fissile system. Supports point and regular shaped volume sources like box, sphere and cylinder.
  - Geometry modeling based on Constructive Solid Geometry techniques. Can model geometries (1D, 2D, 3D) made up of 1st and 2nd degree surfaces.
  - Supports major boundary conditions, namely reflective, white, periodic and vacuum.
  - Supports both analog or non-analog (implicit absorption) method of simulation.
  - Coupled with WIMS format nuclear data library for lattice cell simulation.
  - WIMS Library processing based on 'Equivalence Principle'. Dancoff factors required for the processing is computed internally using stochastic methods. Supports both single and double rational approximation for the resonance treatment.
  - Requires user supplied cross section data for other geometries.

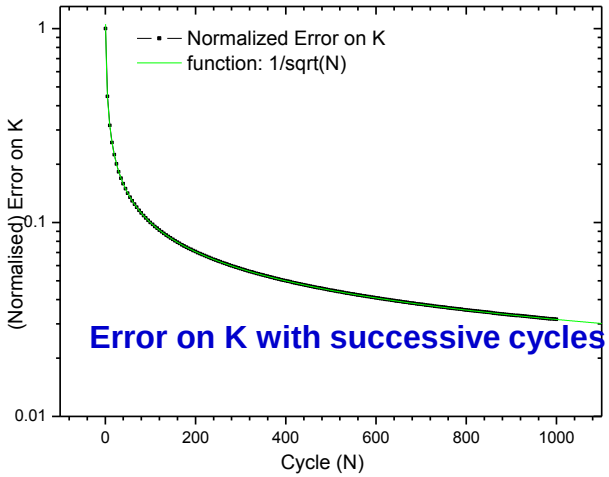


OECD VVER Lattice - MOXGD variant



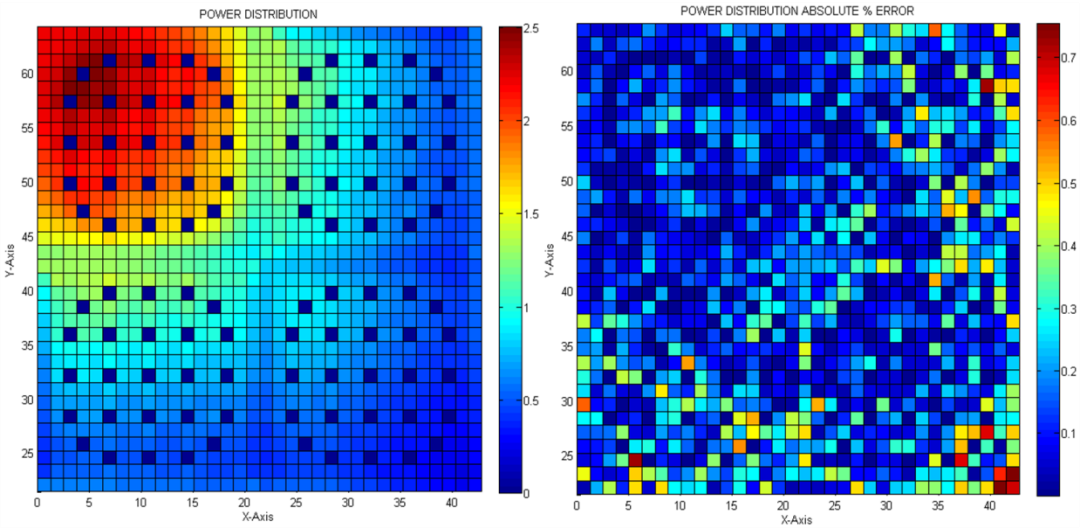
Core and Assemblies of C5G7 Benchmark



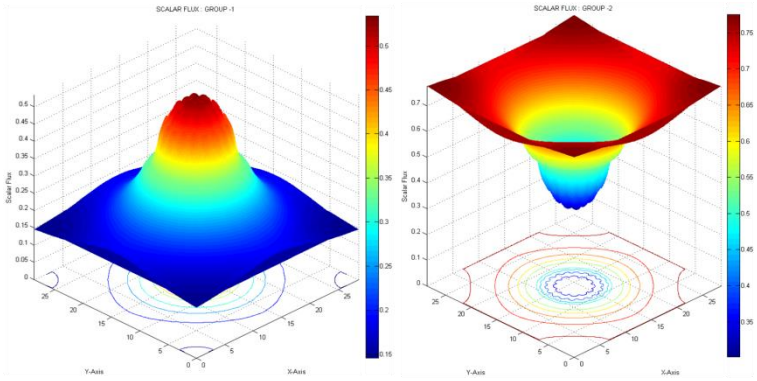


DIAMOND : High fidelity neutron transport code based on MOC

DIAMOND, an advanced code for neutron transport in two-dimensional rectangular assembly lattices to obtain detailed spatial and energy distribution of neutron flux. The concept of Constructive Solid Geometry (CSG) is employed for construction of the complex geometry and intricate structures inside rectangular fuel assemblies. The multi-group neutron transport equation is solved using a flux solver based on the extremely powerful Method of Characteristics (MOC).



Pin power distribution and power distribution error for C5G7 problem using DIAMOND



CANDU-6 annular cell group wise flux distribution