Using RAPID for solving VT^{3}G-SNFP benchmark

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Criticality Calculation

- **Standard approach - Full Monte Carlo calculations face difficulties in this area**
  - Convergence is difficult due to low coupling between regions (due to absorbers)
    - Convergence can also be difficult to detect
  - Computation times are very long, especially to get detailed information
  - Changing pool configuration requires complete recalculation

- **Developed a Multi-stage Response-function Transport approach using the Fission Matrix (FM) technique**

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Derivation of Fission Matrix (FM) Formulation

- Eigenvalue formulation in operator form is expressed by

$$H\psi = \frac{1}{k} F\psi$$

Where,

$$H = \hat{\Omega} \cdot \nabla + \sigma_t(\bar{r}, E) - \int_0^\infty dE' \int_{4\pi} d\Omega' \, \sigma_s(\bar{r}, E' \rightarrow \mathrm{E}, \mu_0)$$

$$F = \frac{\chi(E)}{4\pi} \int_0^\infty dE' \int_{4\pi} d\Omega' \, \nu\sigma_f(\bar{r}, E')$$

FM Derivation (cont)

- We may write above equation as

$$H\psi = \frac{1}{k}\chi\tilde{F}\psi$$

Where, $$\tilde{F} = \frac{1}{4\pi}\int_0^\infty dE'\int_{4\pi} d\Omega' \nu\sigma_f(\vec{r},E')$$

- Then, solve for $$\psi = \frac{1}{k}H^{-1}\chi\tilde{F}\psi$$

- Then, we obtain fission density by operating $\tilde{F}$ onto the above equation to obtain

$$S = \frac{1}{k}AS$$

Where,

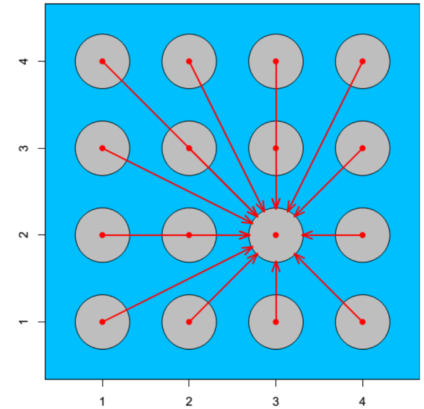
$$S = \tilde{F}\psi$$

$$A = \tilde{F}H^{-1}\chi$$

FM Formulations

- **Eigenvalue** formulation

$$AS(\overline{P}) = \int_{P'} d\overline{P}' a(\overline{P}' \to \overline{P}) S(\overline{P}')$$



  - *k* is eigenvalue
  - is fission source, is fixed source in cell j
  - is the number of fission neutrons produced in cell due to a fission neutron born in cell .

- **Subcritical multiplication** formulation

  - is the number of fission neutrons produced in cell due to a source neutron born in cell .

Developed a Multi-stage methodology for determination of FM coefficients

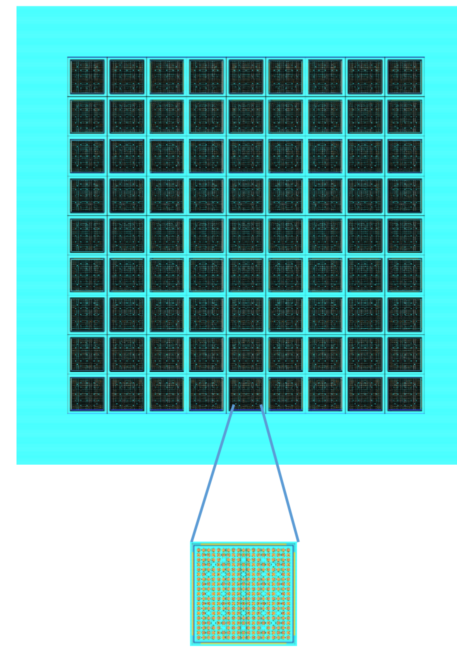
- As the computational cell size, use a single pin
  - $N = 9 \times 9 \times 336 = 27{,}216$ total fuel pins/ fission matrix cells
  - Considering 24 axial segments per rod, then
  - $N = 653{,}184$
  - Allows for good accuracy and pin-resolved fission rates

- Standard FM would require $N = 653{,}184$ separate fixed-source calculations to determine the coefficient matrix
  - One calculation for each pin
  - A matrix of size N x N = 4.26649E+11 total coefficients (6 GB of memory is needed)

- The standard approach is clearly NOT feasible

- We have developed a multi-stage approach to obtain detailed FM coefficients (*in the process of filing for a patent*)

9x9 array of assemblies in a pool



Assembly with 19x19 lattice; 25 positions are reserved for control rods

Remarks on the multi-stage methodology

- **Coefficients are calculated at different stages including:**
  - Pin-wise (**axially dependent**) for one assembly for different burnups, cooling times, lattice structures, and enrichments
  - For assemblies in the pool (**pin-wise or regional)**

- **Reduction in computation time and memory**
  - **Computation time**
    - Geometric similarity
    - Geometric symmetry
    - Degree of coupling
    - Sensitivity of the coefficients to different parameters
  - **Memory by indexing**

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Solution method - FM approach for eigenvalue problems

- After determination of *FM* coefficients, then we solve a system of equations based on the power iteration to obtain *k* and fission density (fundamental eigenfuction)

$$F_i^{(0)} = \frac{1}{N} \quad \& \quad k^{(0)} = 1$$

$$F_i^{(m+1)} = \frac{1}{k^{(m)}} \sum_{j=1}^{N} a_{i,j} F_j^{(m)}$$

Where, $k^{(m)} = \sum_{i=1}^{N} F_i^{(m)}$

RAPID tool

- **Developed the RAPID (Real-time Analysis for spent fuel Pool *In situ* Detection) tool for determination of**
  - Eigenvalue
  - Subcritical multiplication
  - Pin-wise, axial fission density

- **With application to**
  - Criticality safety
  - Safeguards
  - Nonproliferation and materials accountability

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RAPID code system - Structure

Pre-Calculation (one time):

1. Burnup Calculation – to obtain material composition
2. Fission Matrix Coefficient Generation

Real-time Analysis:

1. Run Fission Matrix Code
2. Process Results

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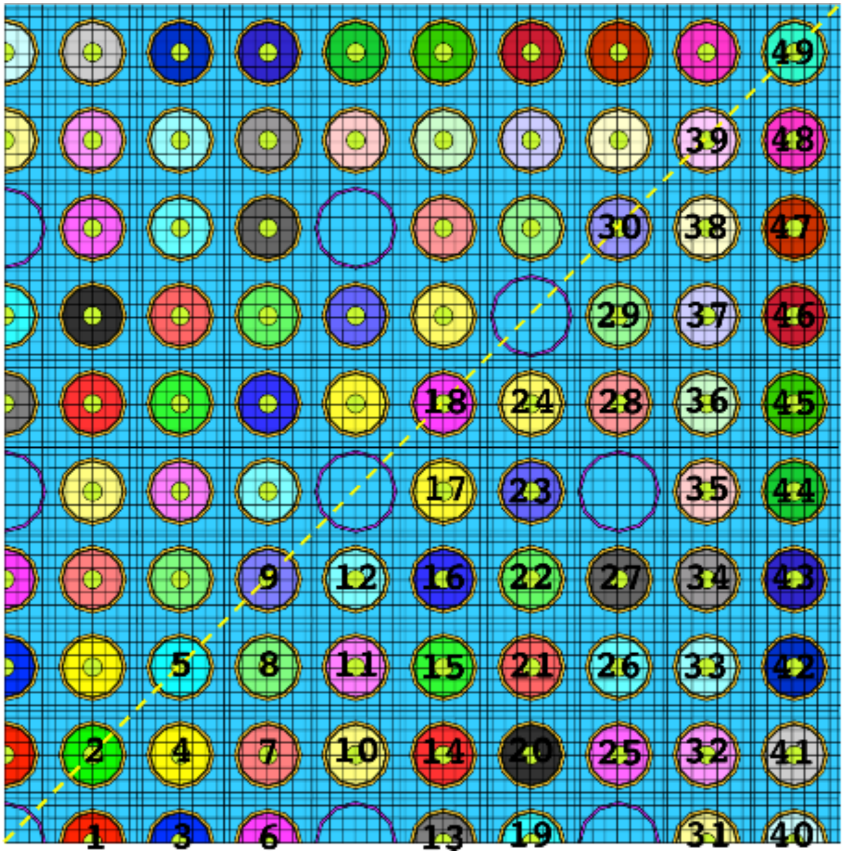
Pre-calculation

1. SCALE* burnup calculation
2. SCALE output processing
3. MCNP input generation & calculation
4. MCNP tally processing

*SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations, ORNL/TM-2005/39, Version 5.1.

Pre-calculation – Step 1

- At each desired burnup, run a quarter assembly SCALE (`t-depl` module) model
  - Reflected on –x and –y
  - Octal symmetry
  - 49 fuel materials (each pin within octant is unique)



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Pre-calculation – Step 2

- SCALE outputs:
  - neutron/gamma spectra
  - actinide/fission product concentrations
  - Fission Spectrum (χ)
    - Fit to Watt fission spectrum (nonlinear regression)

- Process SCALE outputs
  - **`getdat.sh`** - prepares material and source information for MCNP input file
  - **`fitChi.R`** - prepare continuous energy fission spectrum (Watt's spectrum format) from multigroup SCALE generated spectrum for MCNP input file

Pre-calculation –Step 3

- Automatic input file generation for MCNP
  - **`calcMat.f90`** – generates necessary input block segments as a function of burnup and cooling time (source definition and material composition)
  - **`makeMCNP.sh`** – concocts input block segments to generate a full MCNP input file
  - **`mkzmcnp.sh`** – generate 55 unique input files for each *a* & *b* calculation
- Run MCNP for each coefficient as a function of burnup and cooling times

Pre-calculation – Step 4

- Processing MCNP output files to generate database
  - **`getFMco.sh`** - extract fission density tally from each MCNP output file
  - **`rdmc.f90`** – generates FM coefficient database file

Pre-processing– Estimated Time Requirements

- For a single coefficient calculation and processing.

Step		Time (serial)
1.1) SCALE Run		~158 min
1.2) SCALE Output Processing		n/a
3) MCNP Fixed-source Calculation:	Input Generation	n/a
	Calculation	~28 min
4) Tally Processing/Consolidation		~1 min
	Total	**~187 min**

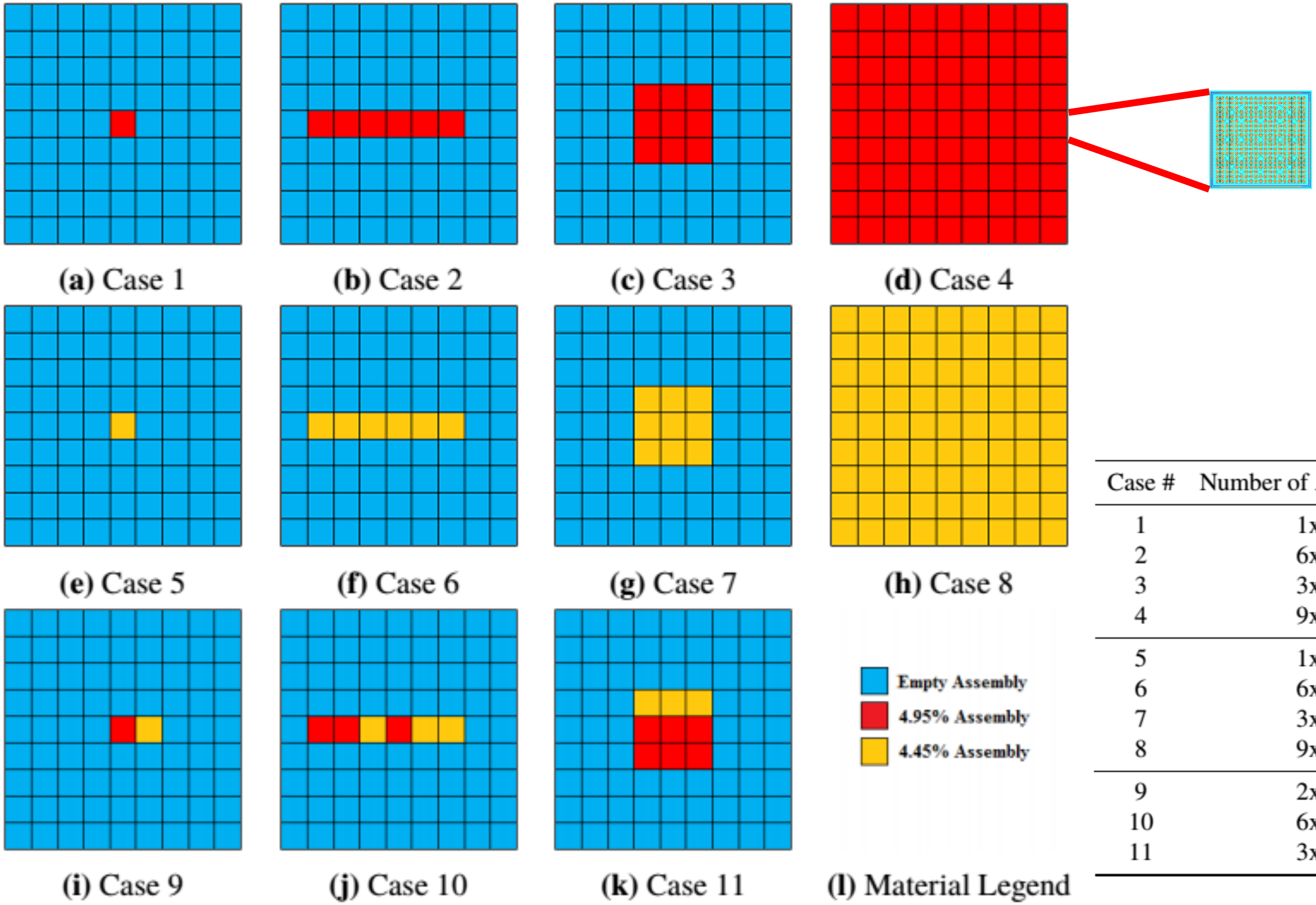
Real-time Analysis

1. Inputs for pool setup
2. Examples

Input Files

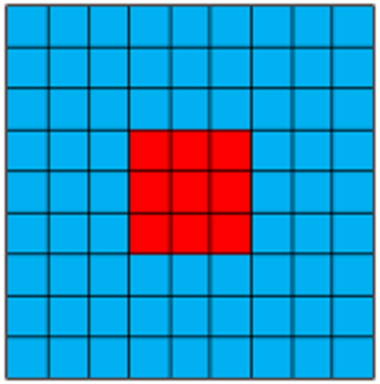
- `pool.inp` – defines the pool structure and range of burnups and cooling times (driver file)
- `runName.burn` – defines assembly-wise axial burnup distribution
- `runName.cool` – defines assembly-wise cooling times

Test Problems (9x9 assemblies)

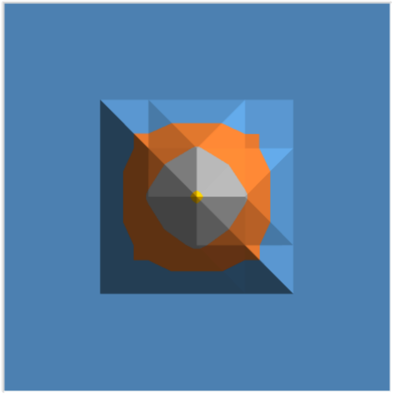


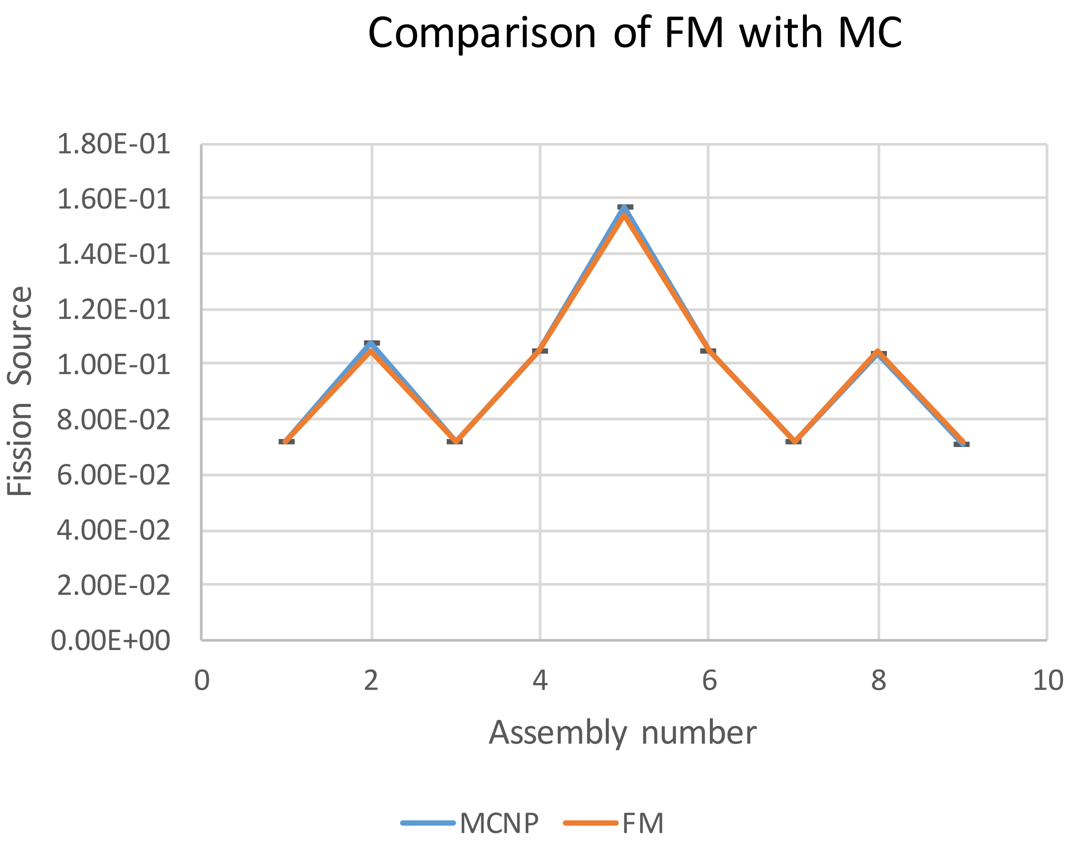
Case #	Number of Assemblies	Fuel Type
1	1x1	4.95%
2	6x1	4.95%
3	3x3	4.95%
4	9x9	4.95%
5	1x1	4.45%
6	6x1	4.45%
7	3x3	4.45%
8	9x9	4.45%
9	2x1	Mixed
10	6x1	Mixed
11	3x3	Mixed

Case 3 Eigenfunction

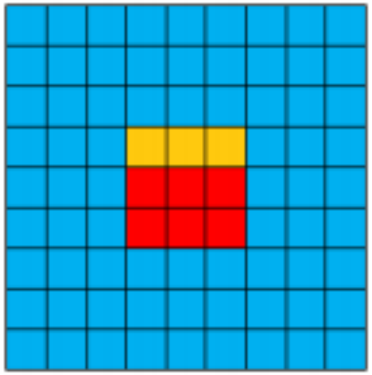


Reference Solution

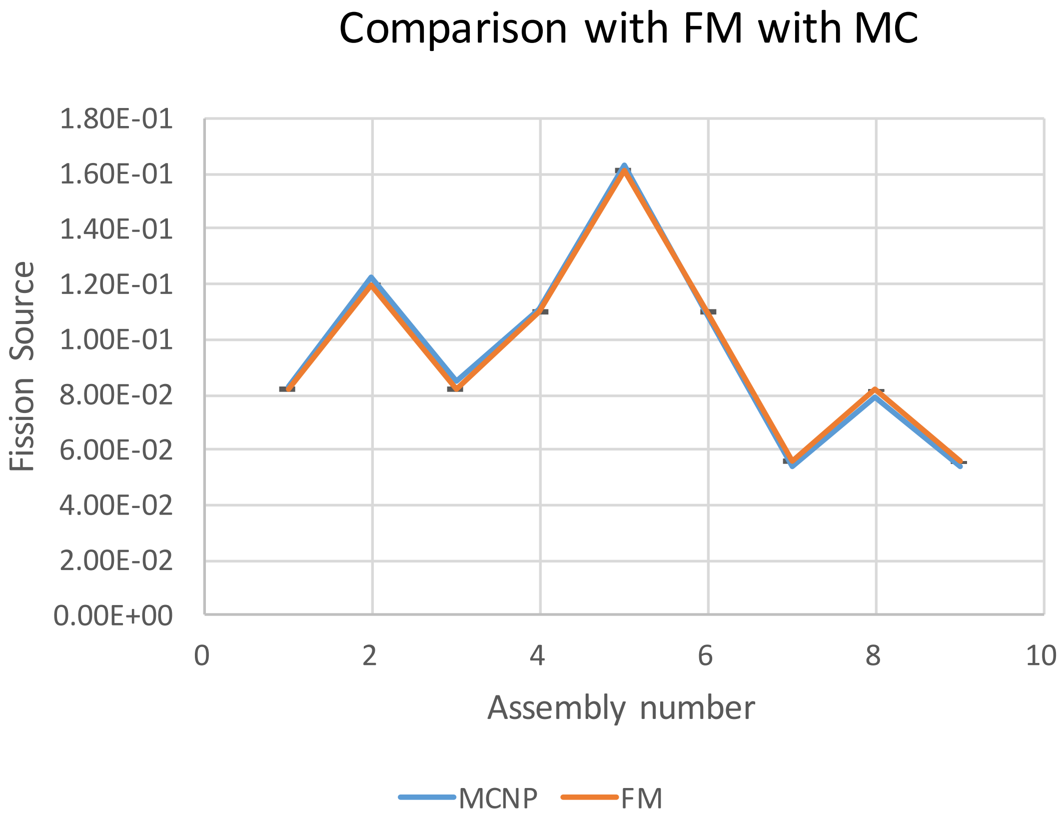




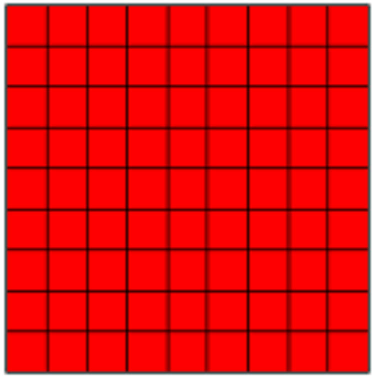
Case 11 Eigenfunction



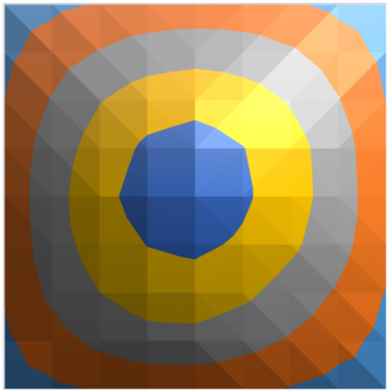
Reference Solution



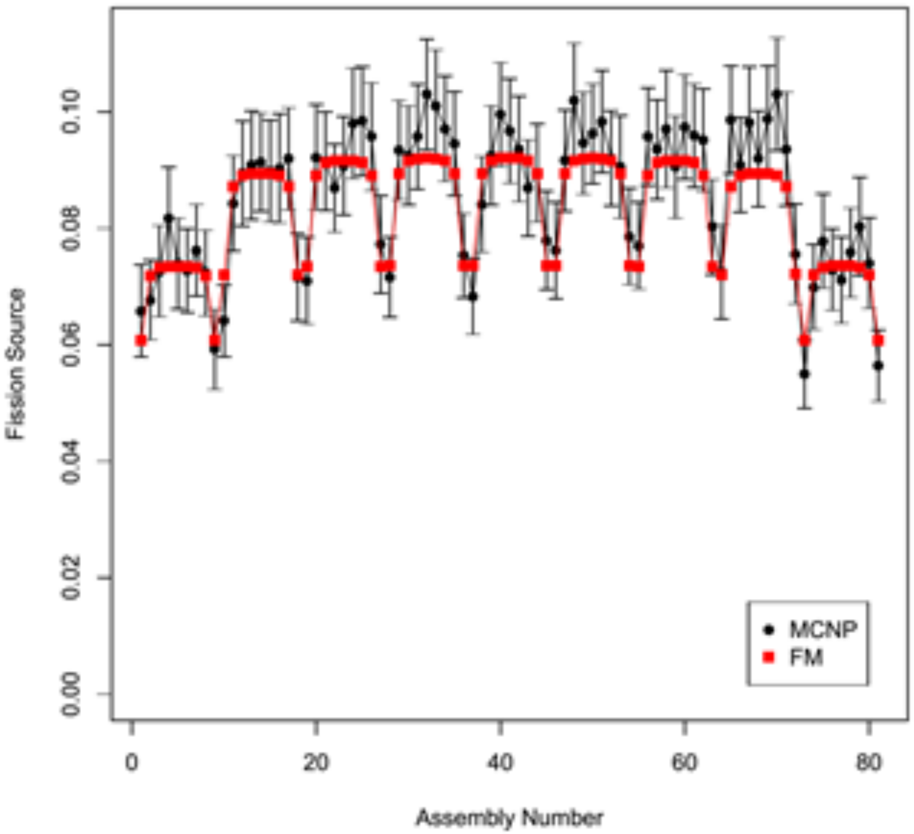
Case 4 Eigenfunction distribution



Reference Solution



Comparison with FM with MC



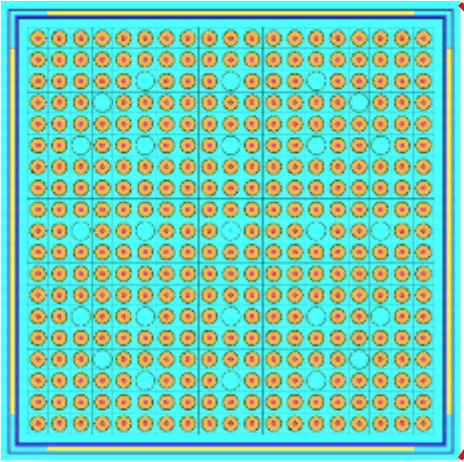
Comparison of calculated M - FM vs. MCNP

Case	FM		MCNP			Error in M (FM vs MCNP)	Speedup (FM vs MCNP)
	M	Time (min)	M	Time (min)	1-σ Uncertainty		
1x1	3.343353	0.092	3.33155	925	0.0010	0.35%	10062
6x1	4.328244	0.213	4.31336	1198	0.0010	0.35%	5613
3x3	5.428051	0.965	5.40992	1502	0.0011	0.35%	1558
9x9	6.697940	8.17	6.67674	1928	0.012	0.32%	236

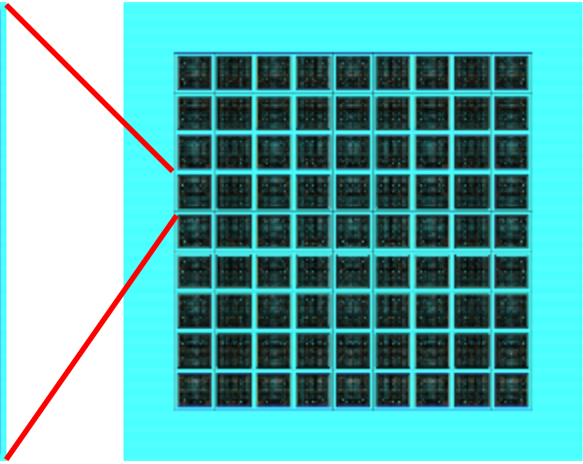
*Note that the FM technique also provide pin-wise, axial-dependent fission source or power.

Reference Spent Fuel Pool

- Being developed for I2S-LWR reactor design
- 19x19 U_3Si_2 fuel assemblies
- 4.45 and 4.95 w/o U-235
- Metamic® absorbers (Al-B4C) used between assemblies
- Burnup up to 55 GWd/MTU



One 19x19 Assembly



9x9 segment of spent fuel pool

Whole pool made up of 8 9x9 segments

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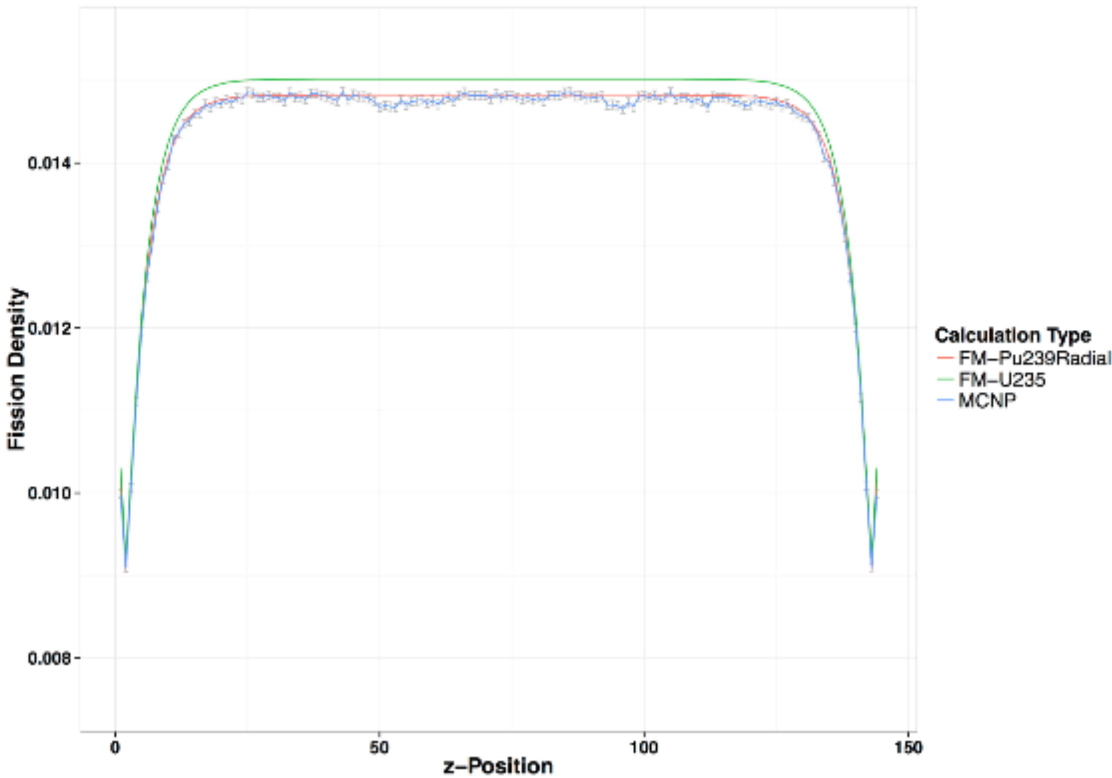
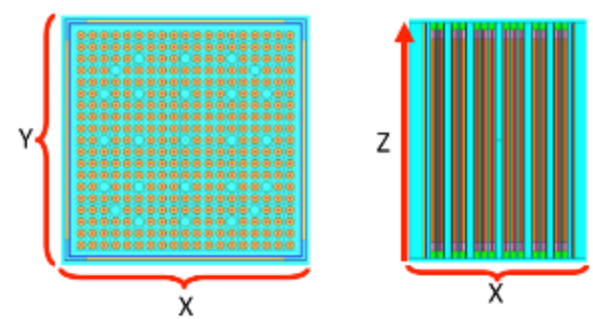
Fission density plots

- Axial fission density for entire assembly (x-y integrated)





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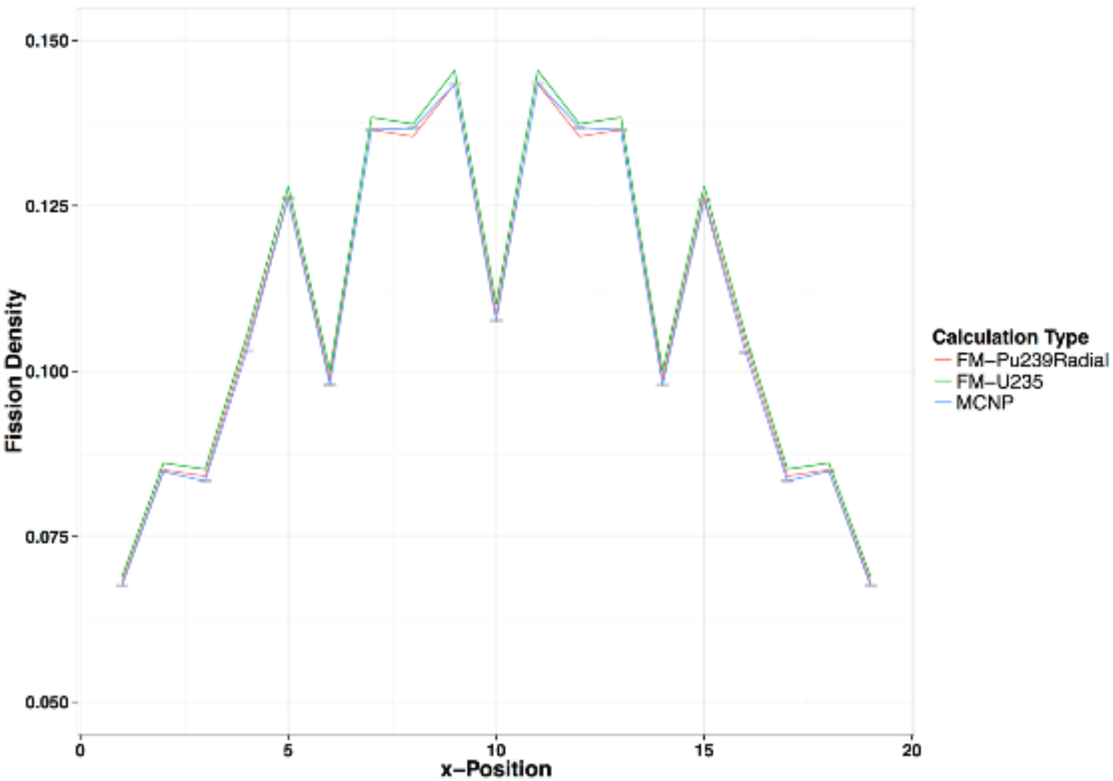
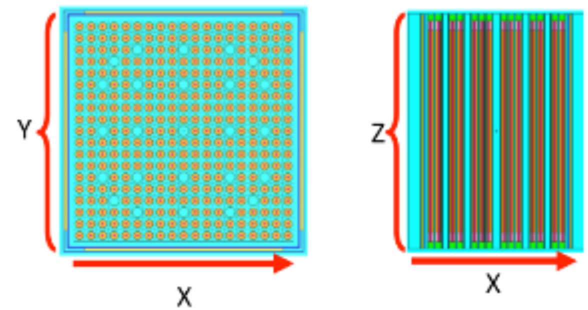
Fission density plots

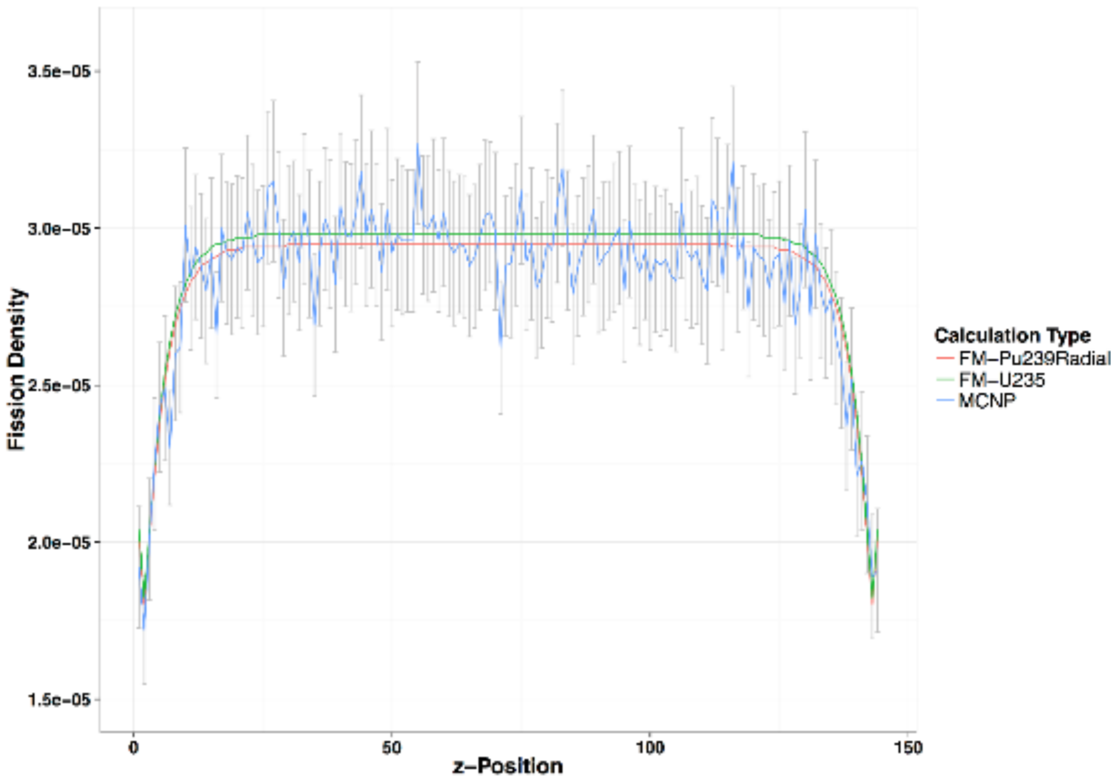
- *x*-fission density for entire assembly (*y*-*z* integrated)

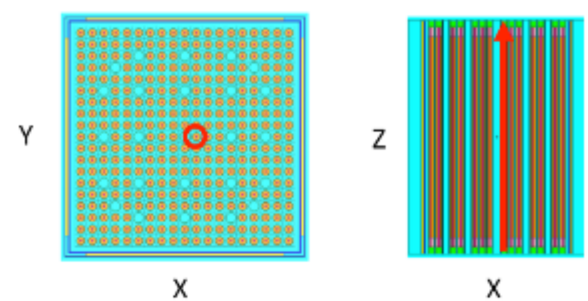




Fission density plots

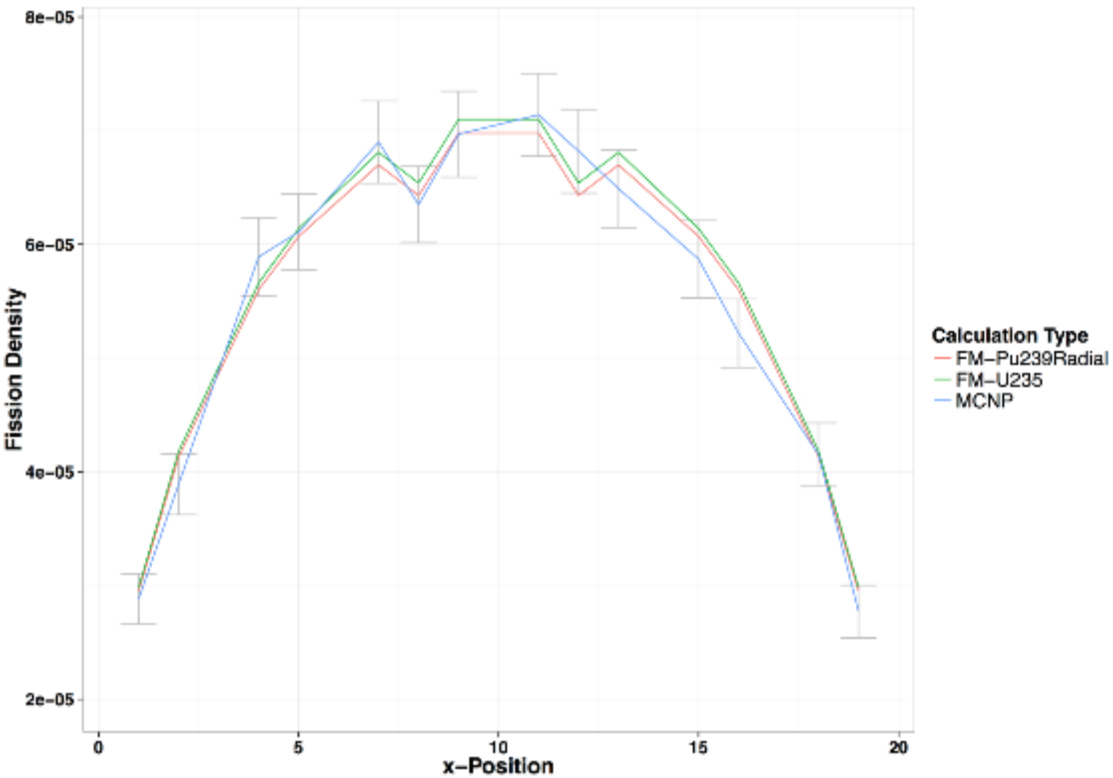
- Axial fission density for a single pin (x=10, y=11)

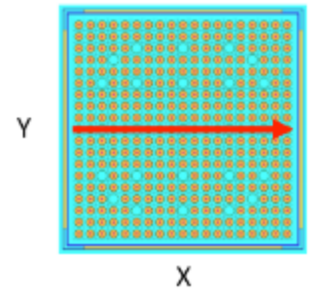


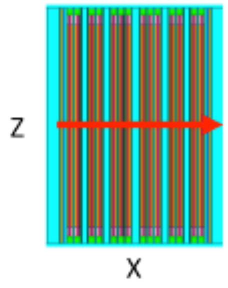


Fission density plots

- *x*-fission density for a single z-level (*y*=10, *z*=72)







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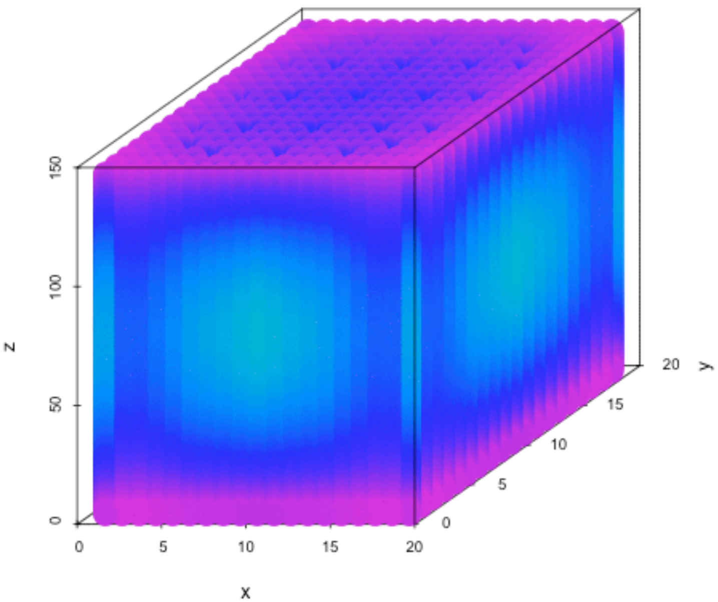
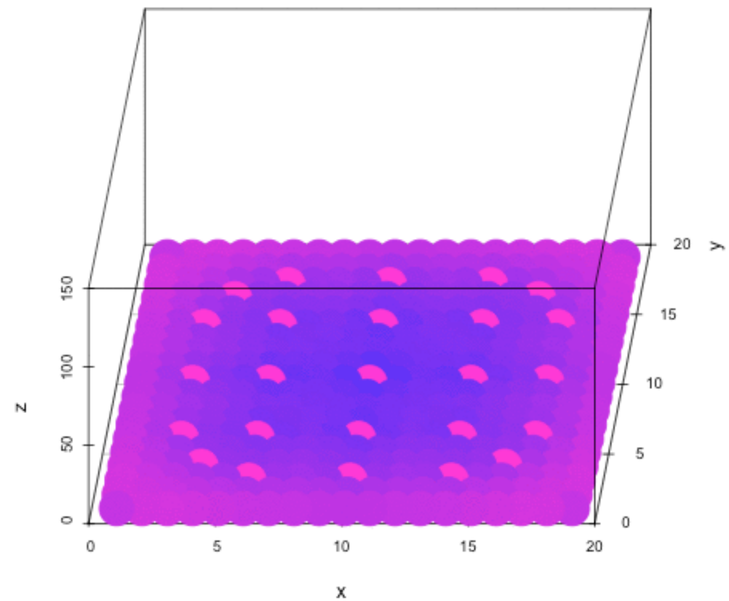
Post Processing: 1x1 Pool Layout

- 3-D Fission Density

Y-LEVEL ANIMATION



Z-LEVEL ANIMATION



RAPID tool

- The current version of the RAPID can quickly and accurately calculate the eigenvalue and eigenfunction for a spent fuel pool
  - By pre-calculating a database of FM coefficients for different conditions, pool simulation can be performed in real-time allowing for changes in configurations (assembly shuffling, removal and addition) for various burnup, cooling times, lattice structure, and enrichments