Benchmarking of the RAPID Tool for a Subcritical Facility

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OUTLINE

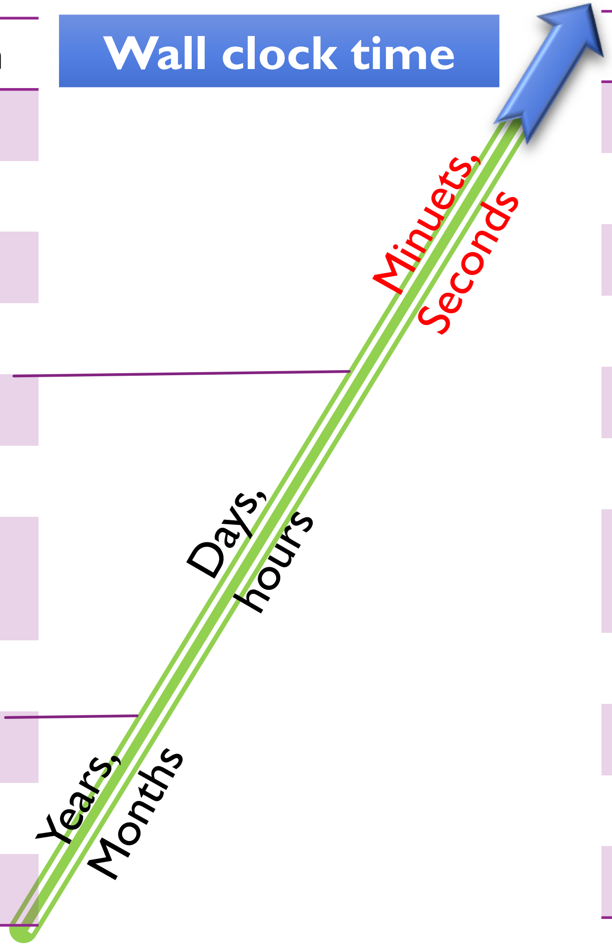
- Purpose
- RAPID tool
  - Theory
  - Capability
- USNA Subcritical (USNA-SC) facility
- Benchmarking of RAPID vs MCNP & USNA-SC facility
- Concluding Remarks

PURPOSE

- The goal of this project is to perform ***experimental*** benchmark studies for further validation of the RAPID (Real-time Analysis Particle-transport In-situ Detection) code system:
  - Experimental facility – the USNA subcritical (USNA-SC)

VT^{3}G – Path to real-time simulation of nuclear systems (power, security & medicine)

Year	Methodology	Computer code system	Former students
2016	MRT*	RAPID**	Dr. Walters
2015	MRT	TITAN-IR	Dr. Royston
2013	MRT	AIMS	Drs. Royston and Walters
2009	MRT	INSPCTs	Dr. Walters
2007	Hybrid MC-det. (AVR)	ADIES (e^-)	Dr. Dionne
2005	Hybrid det. – det.	TITAN (n, ϒ)	Dr. Yi
1997	Hybrid MC-det. (automated VR - AVR)	A^{3}MCNP (n, ϒ)***	Dr. Wagner
1996	Parallel (3-D)	PENTRAN (n, ϒ)	Dr. Sjoden
1992	Vector & parallel (2-D)		Drs. Hunter, Mattis & Petrovic
1989	Parallel processing (1-D)		
1986	Vector processing (1-D)		



*Haghighat, K. Royston, and W. Walters, "MRT methods for real-time simulation of nonproliferation problems," Annals of Nuclear Energy, Vol. 87, Part 1, pp 61-67, April 2016.
**W. Walters, N. Roskoff and A. Haghighat, "Use of the Fission Matrix Method for Solution of the Eigenvalue Problem in a Spent Fuel Pool" Proc. PHYSOR 2014, Kyoto, Japan, Sep. 28-Oct 3, 2014
***Haghighat, A. and J.C. Wagner, "Monte Carlo Variance Reduction with Deterministic Importance Functions," Progress of Nuclear Energy Journal, Vol. 42 (1), Jan. 2003.

RAPID (Real-time Analysis Particle-transport In-situ Detection)
(Wlaters, Haghighat, & Roskoff, 2015)

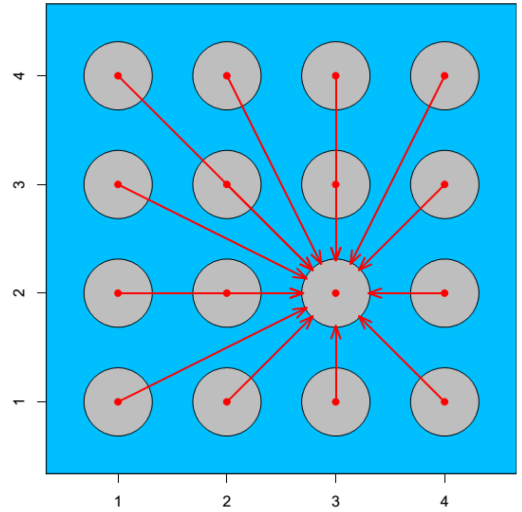
- The RAPID code system is developed based on the MRT (Multi-stage Response-function Transport) methodology; the MRT methodology is described as follows:
  - *Partition a problem into stages*
  - *Represent each stage by a response function or set of response coefficients*
  - *Pre-calculate response functions and/or coefficients (one time)*
  - *Couple stages through a set of linear system of equations*
  - *Solve the linear system of equations iteratively in real-time*

- Currently, RAPID is used for ***criticality safety and safeguards*** of spent nuclear fuel (SNF) pools and casks.

RAPID – Formulation & Algorithm

- RAPID uses the Fission Matrix (FM) approach:
  - **Eigenvalue formulation**

$$\boldsymbol{S_i = \frac{1}{k}\sum_{j=1}^{N} a_{i,j}S_j}$$



    - k is eigenvalue
    - S_j is fission source
    - $a_{i,j}$ is the number of fission neutrons produced in cell i due to a fission neutron born in cell j.
  - **Subcritical multiplication formulation**

$$\boldsymbol{S_i = \sum_{j=1}^{N}(a_{i,j}S_j + b_{i,j}S_j^{intrinsic})}$$

    - $b_{i,j}$ is the number of fission neutrons produced in cell i due to a source neutron born in cell j.

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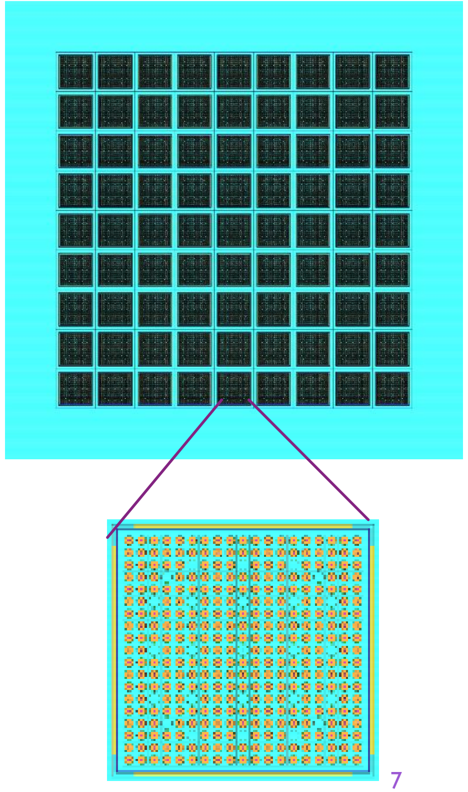
FM Coefficients Determination - a Multi-stage approach

- **Brute force approach:**
  - For a typical spent nuclear fuel pool with a sub-region of 9x9 assemblies:
    - $N = 9 \times 9 \times 264 = 21{,}384$ total fuel pins/ fission matrix cells
    - Considering 24 axial segments per rod, then
      - $N = 513{,}216$
  - Standard FM would require $N = 513{,}216$ separate fixed-source calculations to determine the coefficient matrix
    - A matrix of size *N x N* = 2.63391E+11 total coefficients (> 2 TB of memory is needed)
  - The straightforward approach is clearly **NOT feasible**

- **Multi-stage, regional approach:**

  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



RAPID Code System - Structure

Pre-Calculation (one time):

1. Burnup Calculation – to obtain material composition (ORNL-SCALE6 is used)
2. Fission Matrix Coefficient Generation (LANL-MCNP is used)

Real-time Analysis:

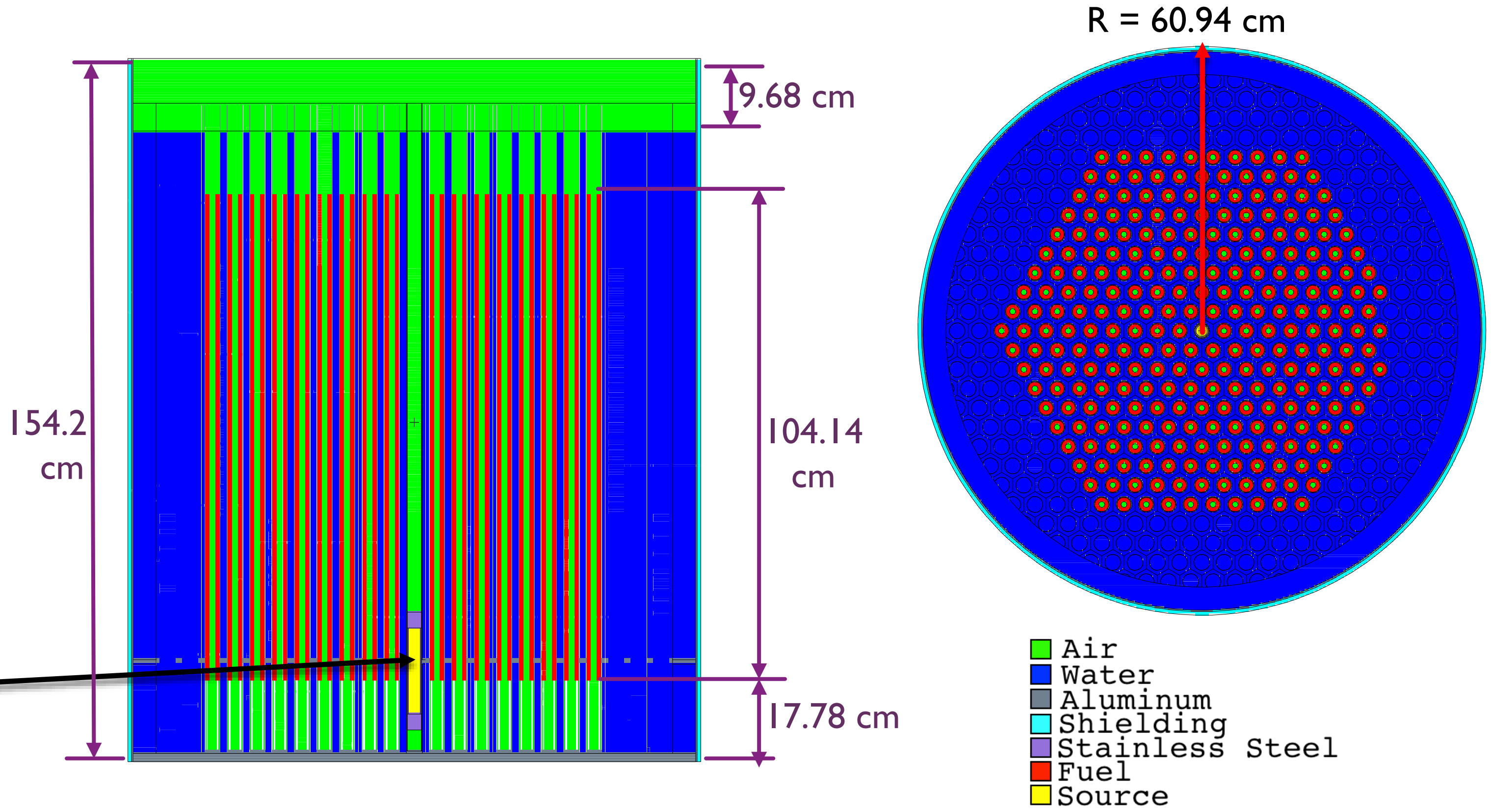
1. Run Fission Matrix Code to determine
   - Eigenvalue, subcritical multiplication, and axial-dependent, pin-wise fission density
2. Process Results and multi-dimensional graphs (in *jpeg* format)

Benchmarking of RAPID

PHASE 1

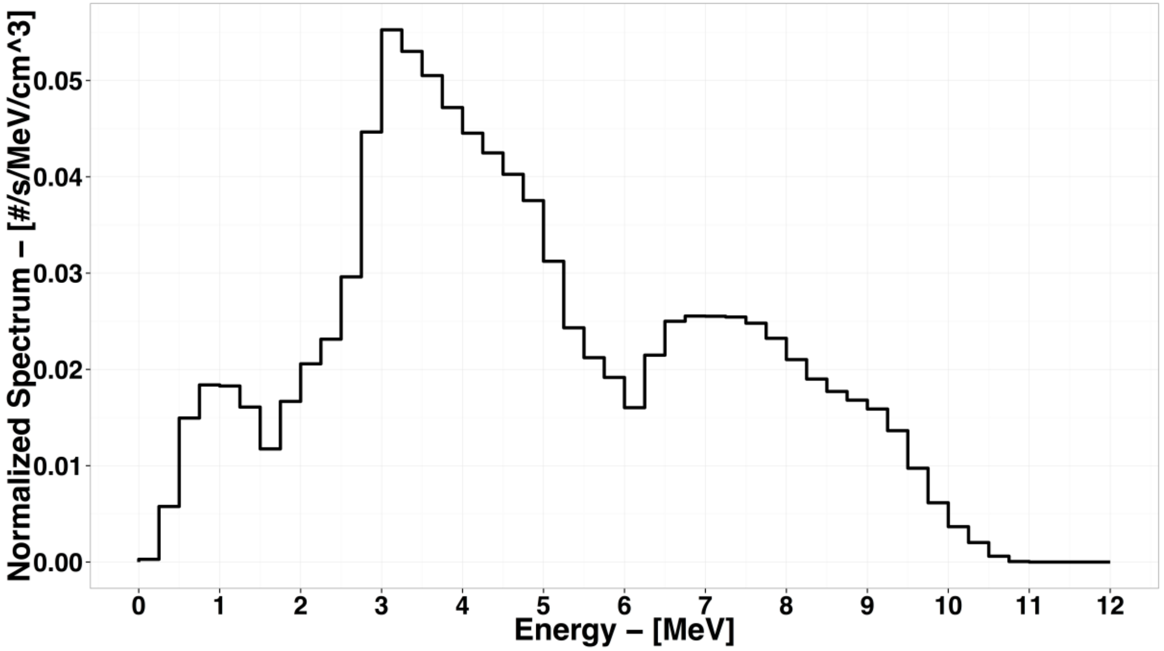
Benchmark facility - US Naval academy Subcritical (USNA-SC)

- A cylindrical pool with natural uranium (fuel) and light water (moderator)
- There are a total of 268 fuel rods, arranged in a hexagonal lattice
- Fuel: hollow aluminum tubes containing 5 annular fuel slugs
- Neutron source: PuBe



Determination of the PuBe Source Spectrum

- SOURCES-4C code (Wilson, 2002) has been used to calculate the PuBe neutron source (from DOE 3031 standard) spectrum.



$$E_{MP} \cong 3.1\ MeV$$

$$\bar{E} = 4.8\ MeV$$

$$S = 2.32x10^6\ \frac{\#}{s - cm^3}$$

Modeling

- Subcritical
  - flux
    - Spatial mesh size (1.22 x1.22x1.58 cm^3)
    - 49-group structure (next slide from SCALE)
- Criticality
  - Eigenvalue
  - Fission density [41-axial node (1" segment) & pin-wise]

49-group structure (SCALE6.1 package) for tallying

Group	Upper Energy (MeV)	Group	Upper Energy (MeV)	Group	Upper Energy (MeV)
1	2.0000E+01	17	1.0000E-04	33	3.7500E-07
2	8.1873E+00	18	3.0000E-05	34	3.5000E-07
3	6.4340E+00	19	1.0000E-05	35	3.2500E-07
4	4.8000E+00	20	8.0000E-06	36	2.7500E-07
5	3.0000E+00	21	6.0000E-06	37	2.5000E-07
6	2.4790E+00	22	4.7500E-06	38	2.2500E-07
7	2.3540E+00	23	3.0000E-06	39	2.0000E-07
8	1.8500E+00	24	1.7700E-06	40	1.5000E-07
9	1.4000E+00	25	1.5000E-06	41	1.0000E-07
10	9.0000E-01	26	1.2500E-06	42	7.0000E-08
11	4.0000E-01	27	1.1500E-06	43	5.0000E-08
12	1.0000E-01	28	1.1000E-06	44	4.0000E-08
13	2.5000E-02	29	1.0500E-06	45	3.0000E-08
14	1.7000E-02	30	1.0000E-06	46	2.5300E-08
15	3.0000E-03	31	6.2500E-07	47	1.0000E-08
16	5.5000E-04	32	4.0000E-07	49	7.0000E-09

Energy Range

Fast
Resonance
Thermal

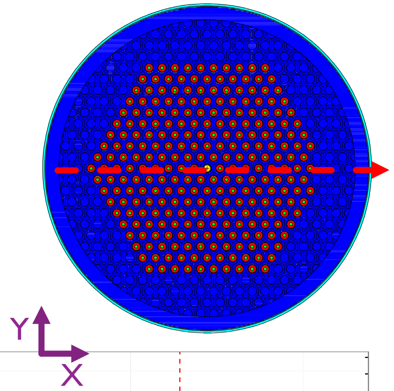
Monte-Carlo MCNP subcritical calculations

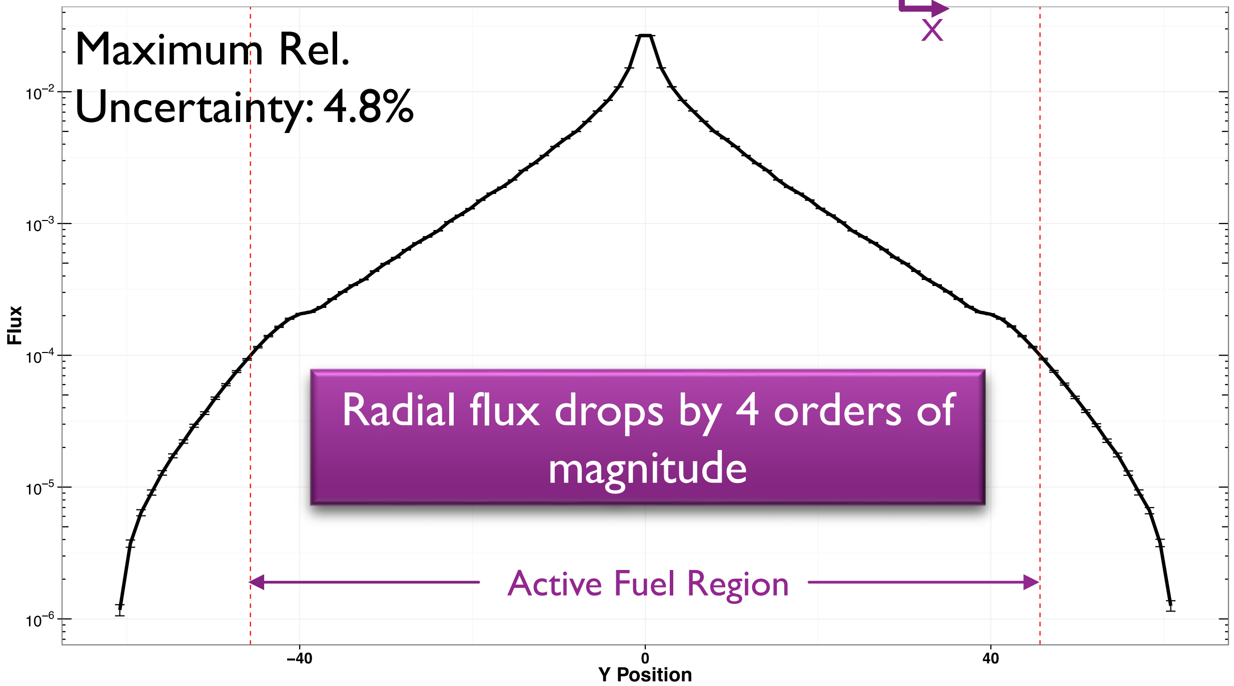
WHOLE CORE - TOTAL NEUTRON FLUX

Cross Core Slice

(y=0cm & z=-102cm)

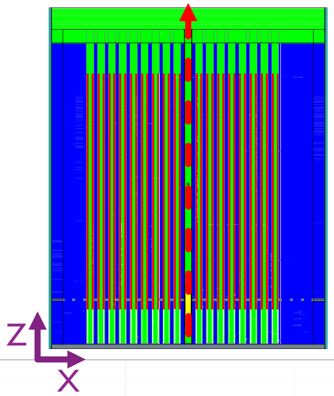
Source Centerline

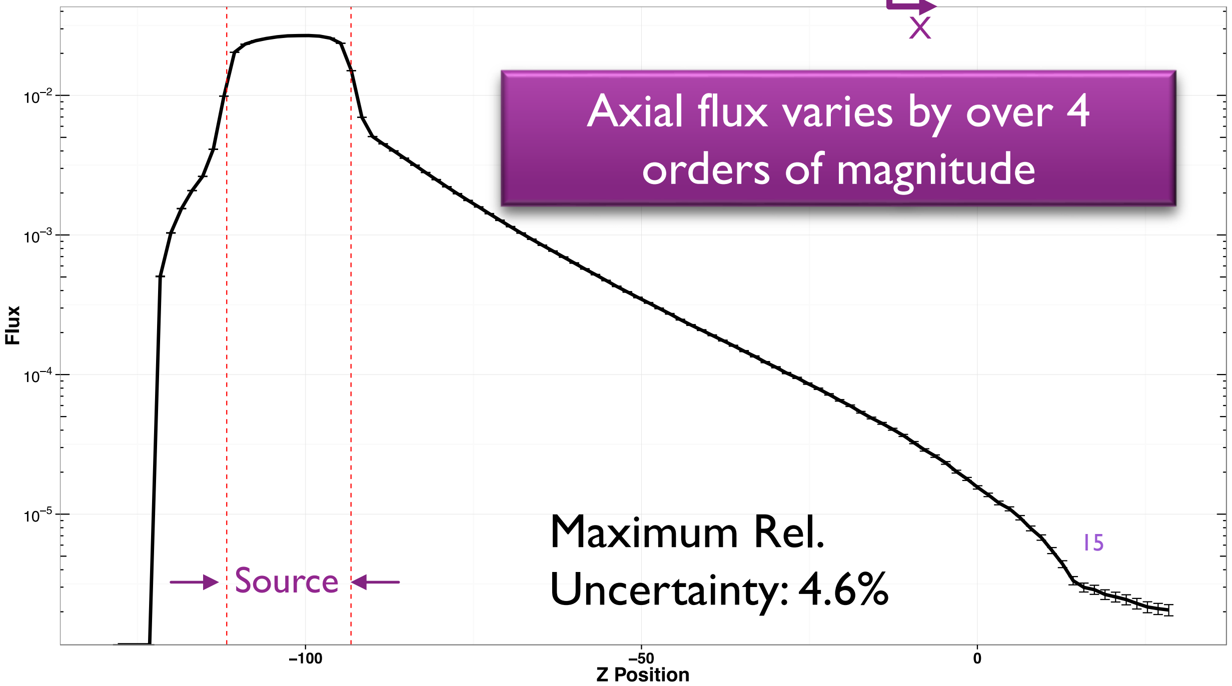




Axial Profile

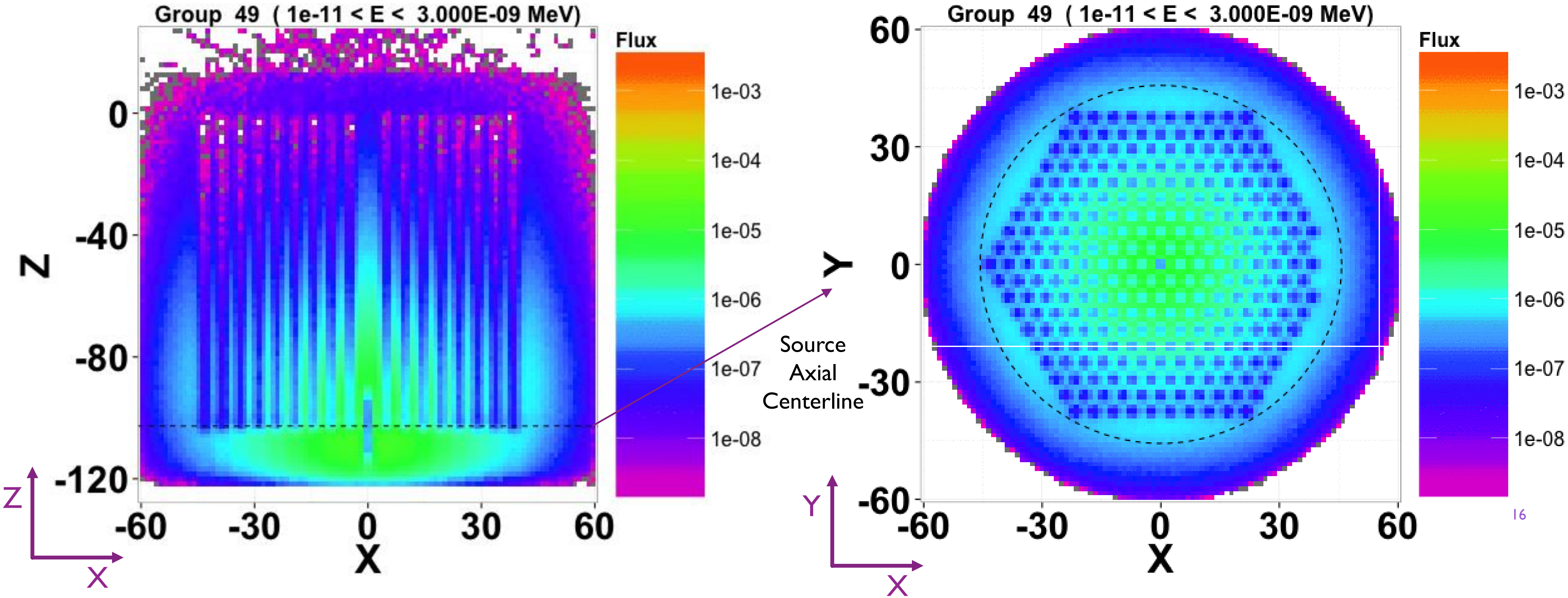
(x=0 & y=0)





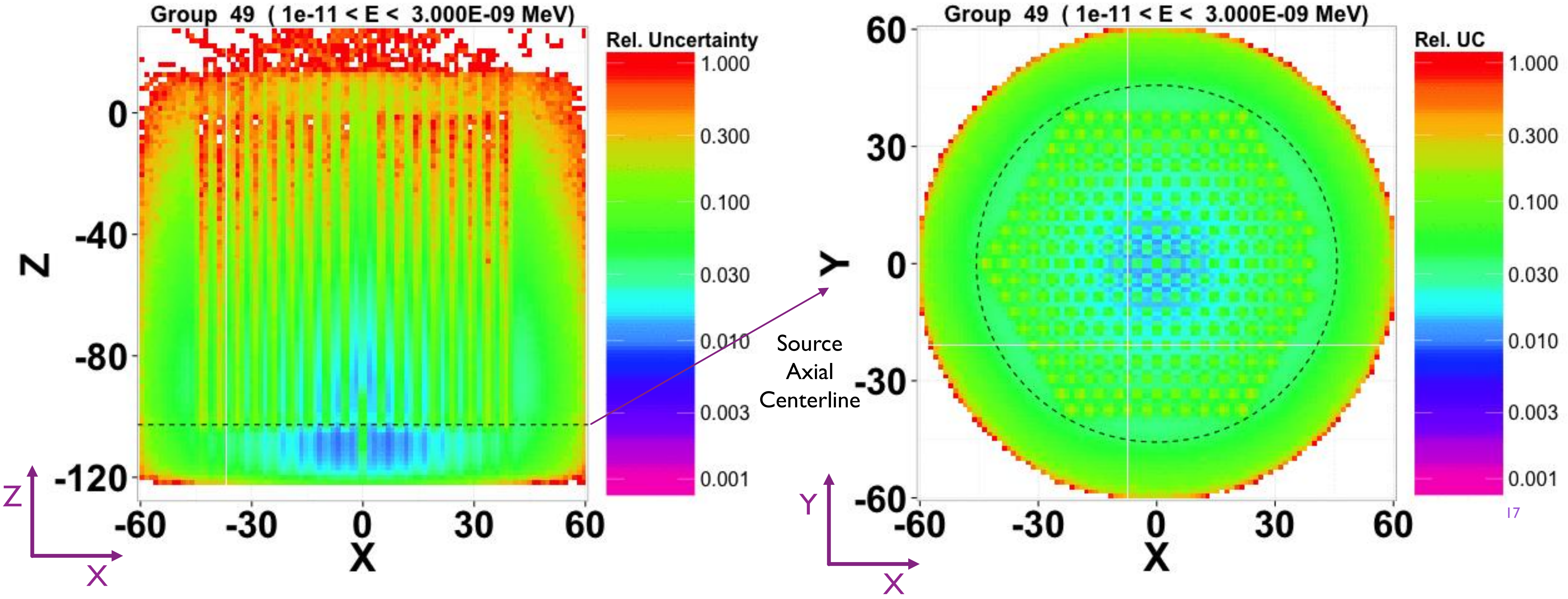
WHOLE CORE : NEUTRON FLUX DISTRIBUTION

(Fast: 1-12; Resonance: 13-22 & Thermal: 23-49)



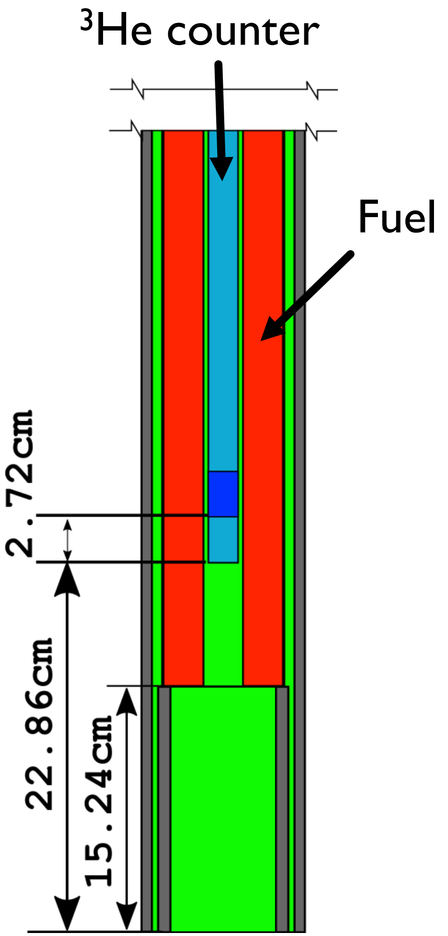
WHOLE CORE : RELATIVE UNCERTAINTIES OF NEUTRON FLUX

(Fast: 1-12; Resonance: 13-22 & Thermal: 23-49)



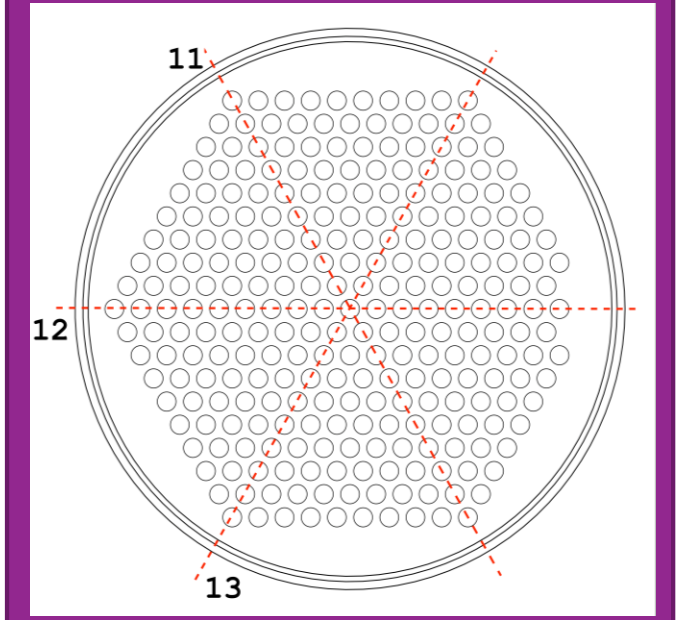
EXPERIMENTS

- Count rate in a ^3He proportional counter was measured by placing the counter within the annulus of each fuel pin
- Neutron counts are determined in fuel pins along three radial profiles (11, 12, & 13) shown in the figure.



Fuel rod

Radial Profiles



Comparison of reaction rates (Counts) of ^{3}He detector [experiment vs calculation]

Estimated Detector Efficiency based on least-squares minimization

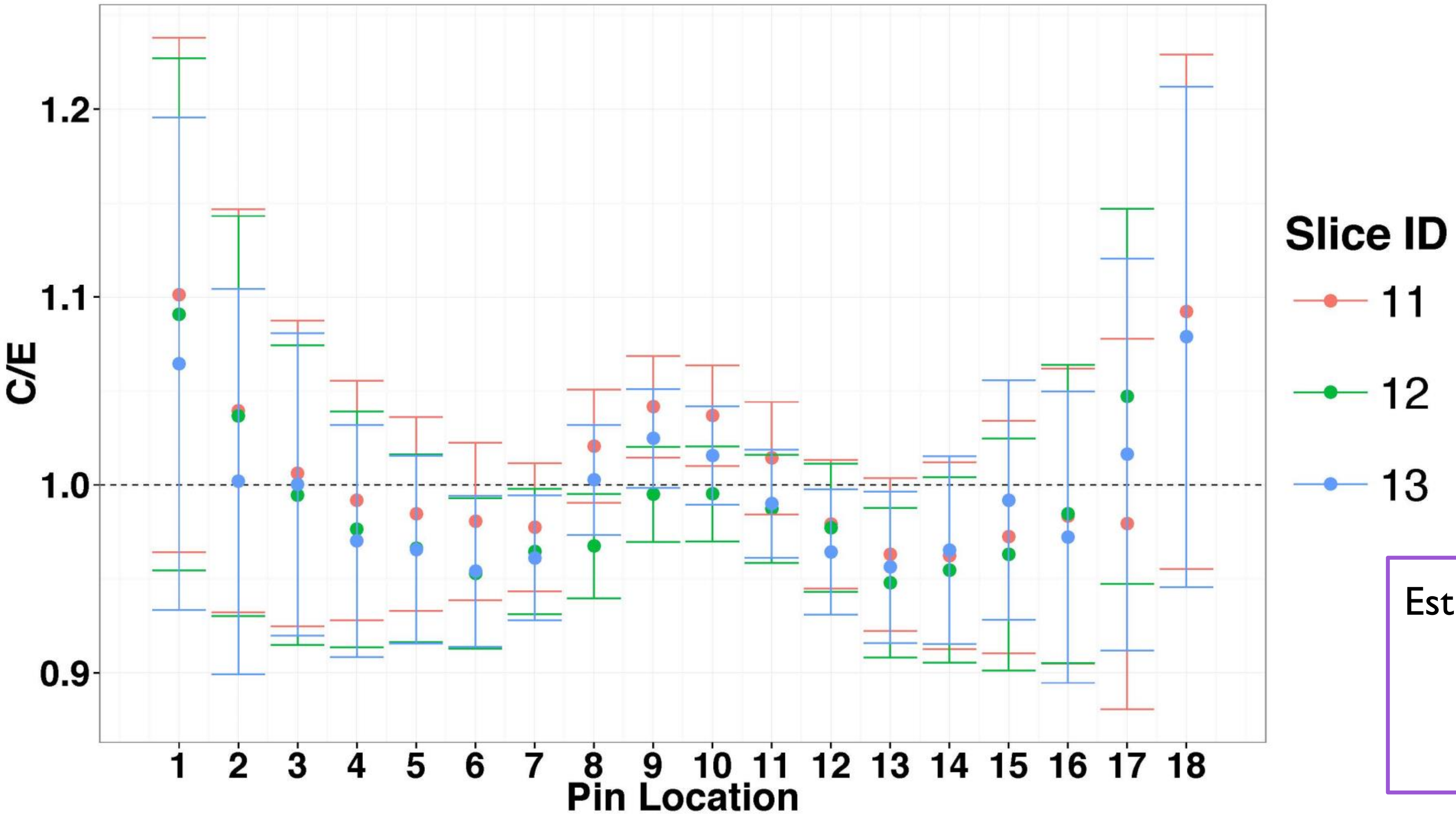
$$Eff = \frac{\sum_i c_i m_i}{\sum_i c_i^2}$$

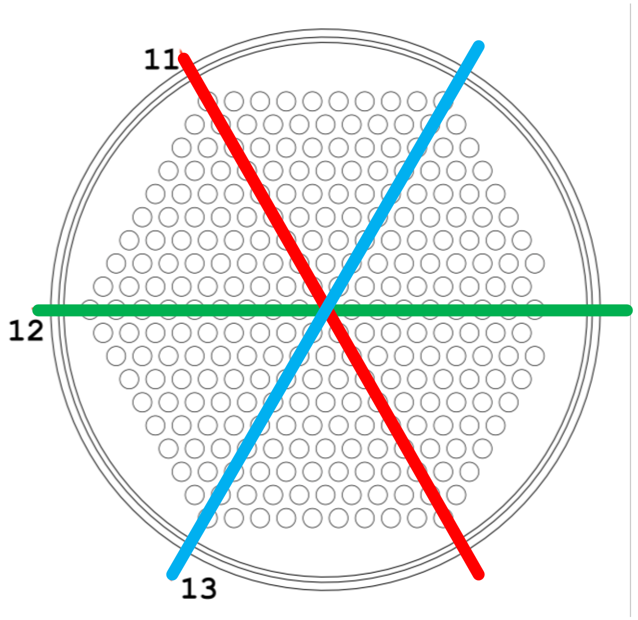
Where,

m_i = Measured response at position *i*

c_i= Calculated response at position *i*

$f = \frac{c}{m}$ (Ratio of calculated to measured responses)





Estimation of uncertainty in $f = \frac{c}{m}$

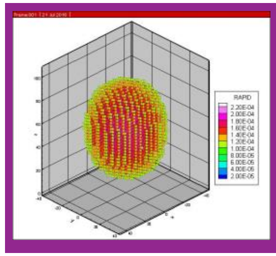
$$= \sqrt{\frac{\sigma_c^2}{m^2} + \frac{c^2 \sigma_m^2}{m^4}}$$

COMPARISON WITH RAPID

EIGENVALUE CALCULATION

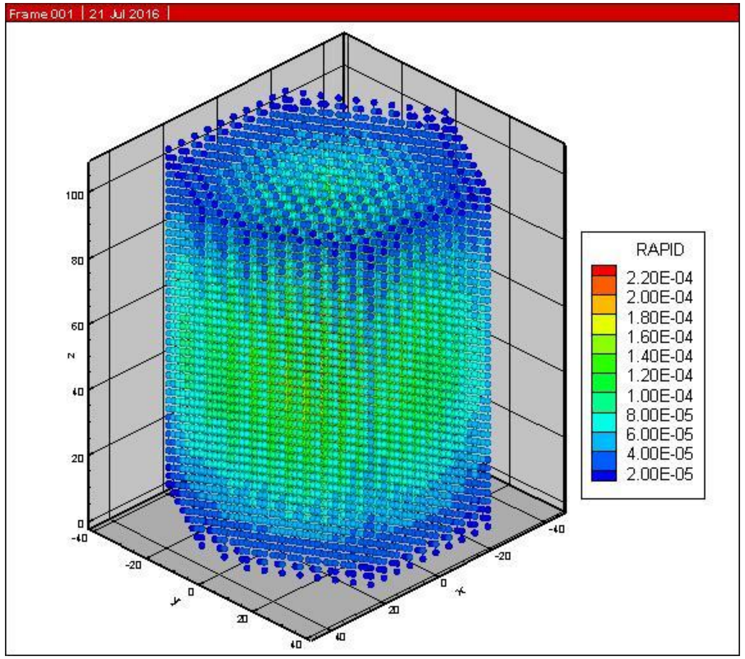
Comparison of Eigenvalues

Code	K-eff	Relative difference (pcm)
MCNP	0. 87278 (±3.5 pcm)	-
RAPID	0.87289	12

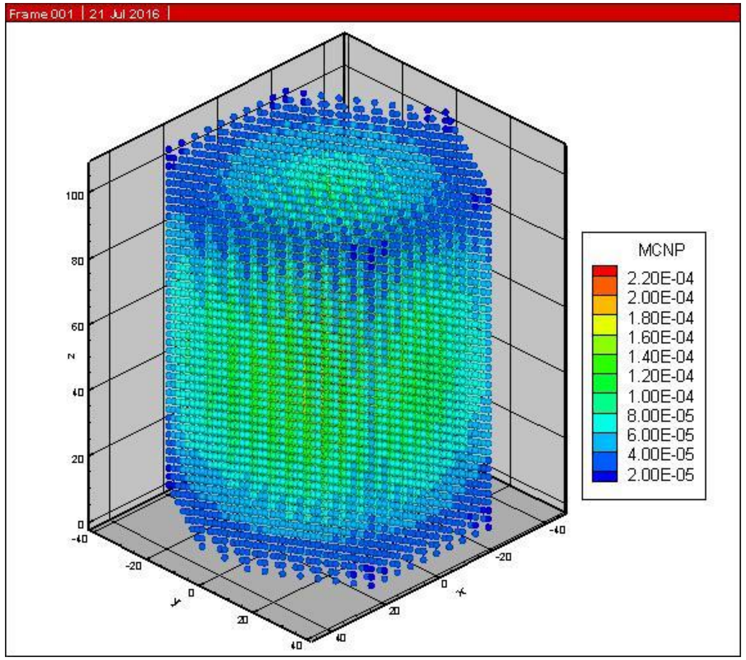


Axial-Dependent (41 nodes), Pin-Wise Fission Density Distribution (10,988 tallies)

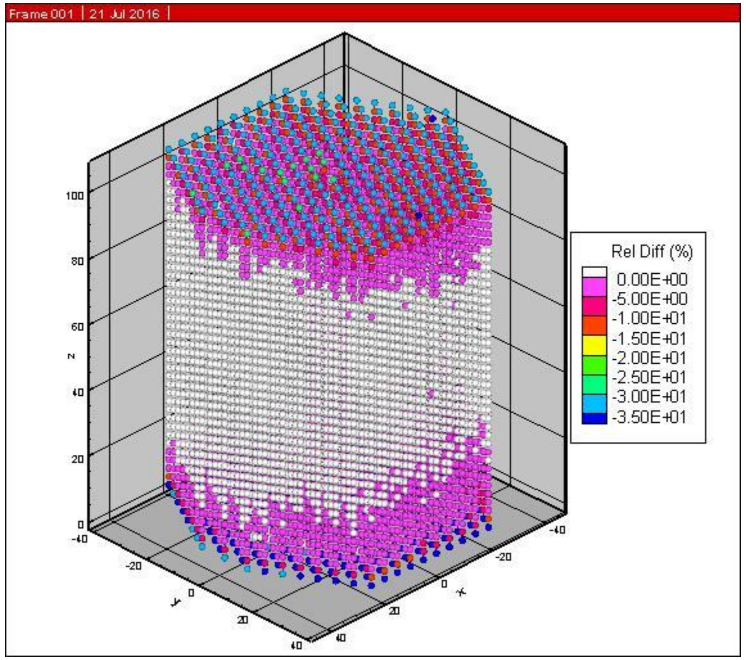
RAPID code system



MCNP code system



Relative difference (%)
RAPID vs. MCNP



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Computation Time

Code	# Processors	Time (min)	Speedup
MCNP	8	819.5 (over 13 hours)	-
RAPID	1	0.22 (about 13 seconds)	3774

Conclusions

- Using the USNA-SC facility, we have demonstrated that the measured and calculated ^{3}He detector responses along three radial profiles are within ±10.
- RAPID results are in excellent agreement with the MCNP predictions:
  - Eigenvalue
  - Eigenfunction - Detailed axial-dependent, pin-wise fission density (10,988 tallies)
- RAPID can solve for eigenvalue and a detail fission density distribution in real time (< 1 min!)

Ongoing & Future Studies

- Performing subcritical multiplication calculation using RAPID
- Determining the FM coefficients with further spatial detail
- Performing further experimentation
  - In-core experiments using a BF_3 detector
  - Ex-core experiments using a larger ^3He detector
  - Axial-dependent measurements using BF_3 & ^3He detectors
- Will prepare benchmark problems for both critical and subcritical reactor systems

QUESTIONS?

THANKS

Appendix – Relative uncertainty in the MCNP fission density