Reactor Power Distribution Calculation in Research Reactors Using MCNP

Zeyun Wu[1,2] and Robert Williams[1]

[1] NIST Center for Neutron Research, 100 Bureau Drive, Gaithersburg, MD
[2] Department of Materials Science and Engineering, University of Maryland, College Park, MD

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Introduction

- In reactor calculations, a detailed 3-D power distribution is a requisite for core optimization studies and safety analyses.
- There is no a single tally option in MCNP that is capable of directly calculating power information (F7 tally only takes account for prompt fission energy release in a fission event).
- The reactor power is directly related to the reactor specific Q-value, which is essentially the summation of the kinetic energy (K.E.) from all radiation components released from a single fission event.
- It is a complex task to accurately estimate a Q-value for a given reactor.
- This talk presents two alternative methods to predict the 3-D power distribution in a reactor with the assumption that all the recoverable fission energy (effective fission energy release) is deposited at the point of fission and the power density is proportional to the fission rate density.

Energy Release in U-235 Thermal Fission

Energy Source	Emitted Energy (MeV)	Energy (%)	Distance	Recoverable	Time Delayed
K.E. of fission fragments	168	81.2%	<0.01 cm	yes	Instantaneous
K.E. of prompt γ-rays				yes	Instantaneous
K.E. of fission neutrons	5	2.4%	10-100 cm		
- prompt neutron				yes	Instantaneous
- delayed neutron				yes	delayed
Fission-product decay					
- β-rays	8	3.9%	short	yes	delayed
- γ-rays	7	3.4%	100 cm	yes	delayed
- neutrinos	12	5.8%	n/a	no	delayed
Non-fission (n, γ) reaction	3-12[a]				
- radiative capture γ-rays (PGNA)			100 cm	yes	instantaneous
- (n, γ) product decay β-rays			Short	yes	delayed
- (n, γ) product decay γ-rays (DGNA)			100 cm	yes	delayed
- neutrinos			n/a	no	delayed
Total	207				

[a]This value will depend upon the nature of the materials present in the reactor core, this is the reason that the total amount of heat produced by fission will vary, to some extent, from one type of reactor to another.

Method for Power Distribution Calculation in MCNP – I: FMESH Method

- This method applies flux tally (F4 card) or mesh flux tally (FMESH card) and the tally multiplication option (FM card) in MCNP to produce the cell-wise fission rate.
- The superimposed mesh tally capability provides significant convenience for the power distribution calculation.
- The data post-processing is trivial as the hierarchy of output can be controlled and managed by MCNP.
- But this method requires additional mesh definition and computational cost for flux tallies.

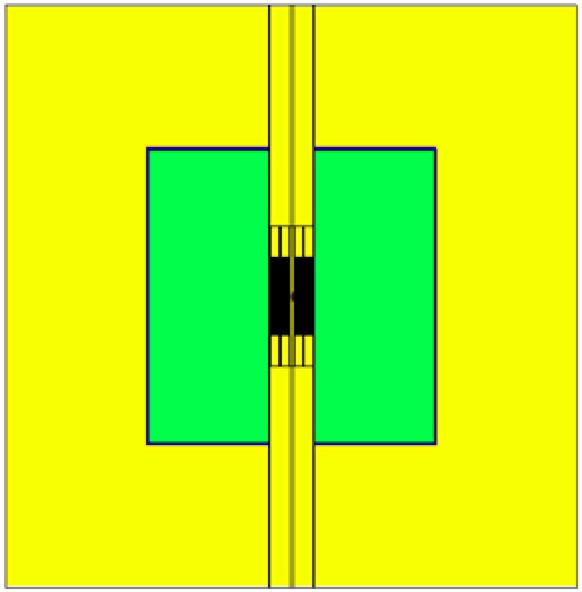
```
c
c Superimposed mesh tally (xyz geometry): FMESH tally for flux
c
fmesh4:n  geom=xyz origin= -18 -17.6 -30  $ Origin at bottom, left, behind of a rectangular
          imesh=-17.387 -10.213 -9.187 -2.013 2.013 9.187 10.213 17.387 18
          iints=1 17 1 17 1 17 1 17 1
          jmesh=-16.567 -10.433 -8.567 -2.433 2.433 8.567 10.433 16.567 17.6
          jints=1 3 1 3 1 3 1 3 1
          kmesh=30 kints=30
          out=ij
c Tally multiplier for fission density (fission/cm3)
fm4       1 0  -6  $ mat = 0 for MCNP to identify the specific material contained in the mesh
c
```

Method for Power Distribution Calculation in MCNP – II: Table128 Method

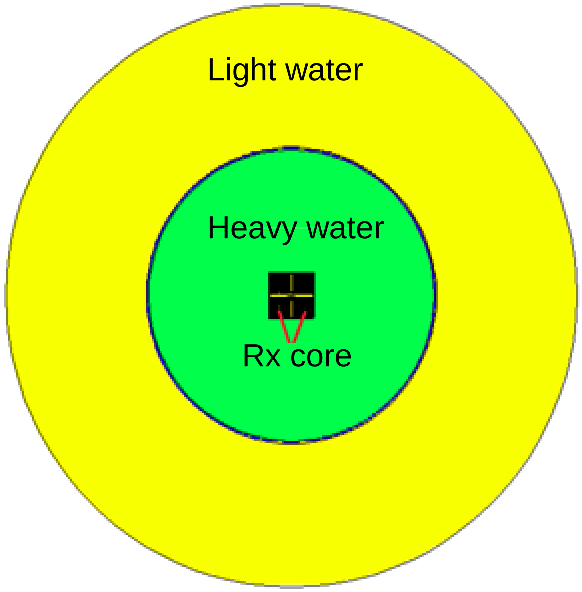
- This method uses the converged fission source number printed in the universe map table (Table 128) in the standard output of MCNP.
- The fission source number of a cell is naturally proportional to the fission density in that cell.
- Thus Table 128 actually provides a straightforward way to obtain power density information in MCNP with no additional computational cost.
- But cells containing fissionable materials need to be divided into multiple sub-cells if detailed power density distribution is desired.

```
1neutron  activity in each repeated structure / lattice element                                   print table 128

     source    entering  collisions        path

          0           0           0      1
          0    36656529  1458224098      2
          0    69178830    25972108      3
          0   108956724 -1273264116      4
          0   148019273    17381057      5
          0      176067       44588   12[-2 -2 0] <     6
          0     7468931     3481360   12[-1 -2 0] <     6
          0     7837726     3596864   12[ 0 -2 0] <     6
          0           0           0   12[ 1 -2 0] <     6
          0     7316904     3410798   12[-2 -1 0] <     6
          0     4596903     2578131   2231[-9 0 0] < 2201 < 12[-1 -1 0] <     6
      2791*       31115        3041   6101[0 -1 0] < 5101 < 2231[-8 0 0] < 2201 < 12[-1 -1 0] <     6
      2480*       30307        2860   6101[0  0 0] < 5101 < 2231[-8 0 0] < 2201 < 12[-1 -1 0] <     6
      2733*       31477        3024   6101[0  1 0] < 5101 < 2231[-8 0 0] < 2201 < 12[-1 -1 0] <     6
      2741*       32521        2937   6102[0 -1 0] < 5102 < 2231[-8 0 0] < 2201 < 12[-1 -1 0] <     6
      2350*       32109        2706   6102[0  0 0] < 5102 < 2231[-8 0 0] < 2201 < 12[-1 -1 0] <     6
      2611*       32753        2969   6102[0  1 0] < 5102 < 2231[-8 0 0] < 2201 < 12[-1 -1 0] <     6
```

Example: A Compact Core Research Reactor



(a) Side view

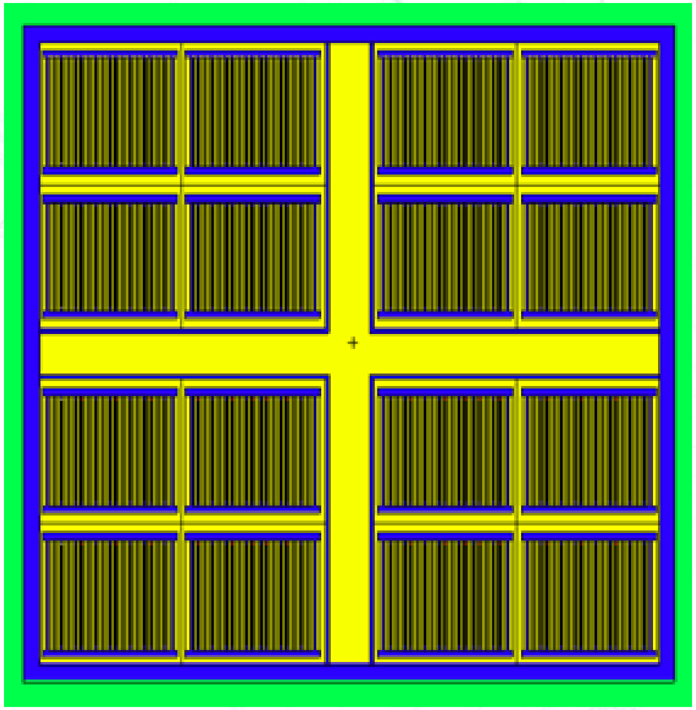


(b) Top view

A Schematic view of cutaway side-plane (left) and mid-plane (right) of the reactor.

Reactor Size (m)	Value
Heavy water tank diameter	2.5
Heavy water tank height	2.5
Light water pool diameter	5.0
Light water pool height	5.0

Fuel Element Layout for the Core



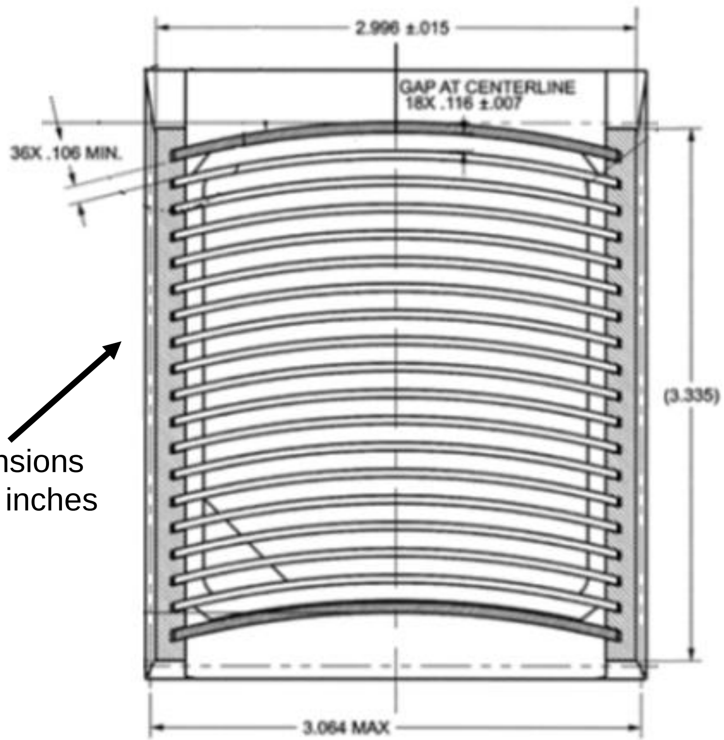
The core is similar to the OPAL (Open Pool Australian Light-water Reactor) core located in a suburb of Sydney, Australia.

Basics of the Core and Fuel Element

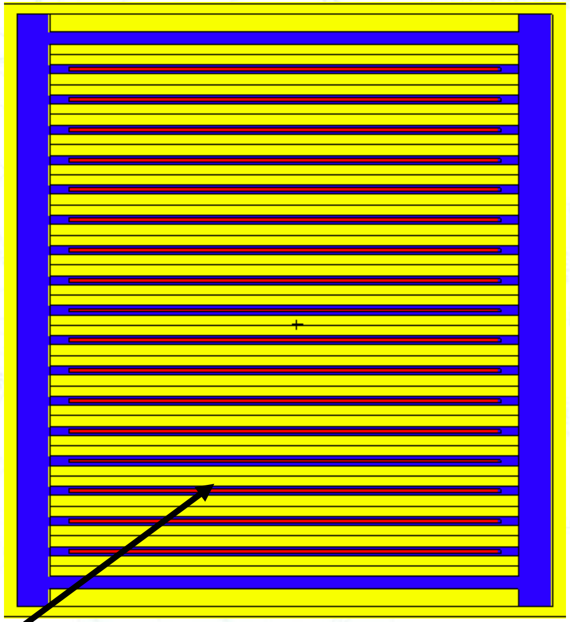
Parameter	Data
Thermal power rate (MW)	20
Fuel cycle length (days)	30
Active fuel height (cm)	60.0
Fuel material	U_3Si_2/Al
U-235 enrichment in the fuel (wt. %)	19.75
Fuel mixture density (g/cc)	6.52
Uranium density (g/cc)	4.8
Number of fuel elements in the core	16

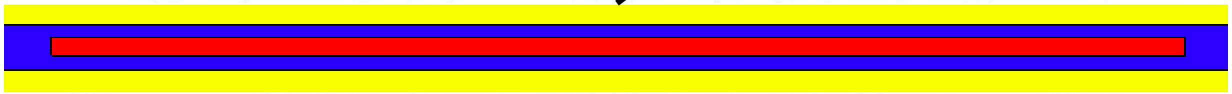
The MTR-type Fuel Plate and Fuel Element

Cross sectional view of the fuel element: 17 fuel plates, 2 end plates and 2 side plates.



Dimensions are in inches

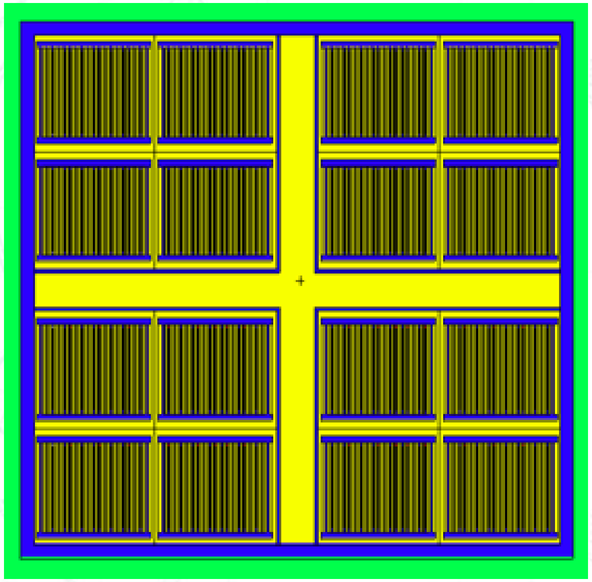




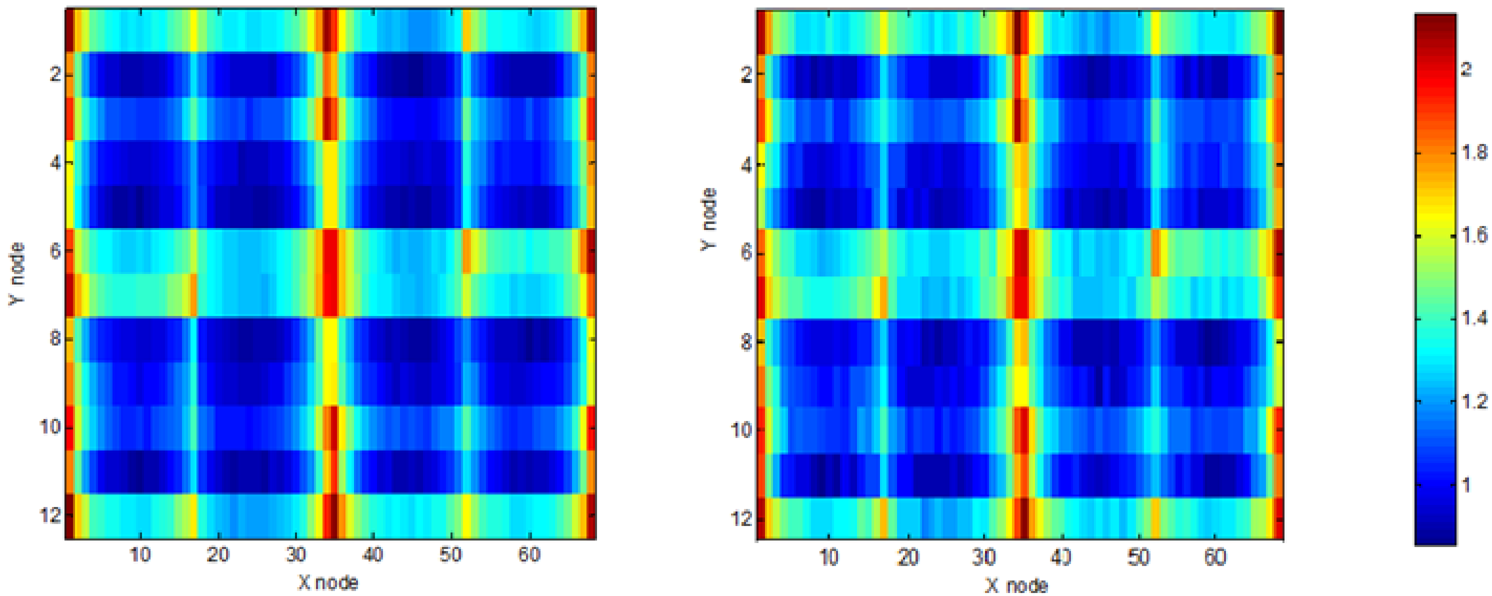
The fuel plate: For the U_3Si_2/Al fuel meat, it is 0.066 cm (26 mil) thick and 6.134 cm wide.

Detailed 3-D Discretization for the Output

- Each fuel meat is divided into 30 intervals in length (axial), 3 intervals in width, and 1 interval in thickness, thus the total number of computational cells in one plate is 30 x 3 x 1 = 90 with the volume ~0.27 cm^3 for each cell.
- Considering the number of plates in one FE (17) and the number of FEs in the core (16), the total number of fissionable cells in the example is 90 x 17 x 16 = 24480.
- As the fuel has 30 segments in axial direction, the output results will be presented as 30 axial levels with 12 x 68 cells in each level.
- The above discretized scheme is applied to both methods discussed previously.



2D Image View of the Power Factors in the Mid-plane of the Core



(a) FMESH method (b) Table128 method

The results yielded from both methods are nearly identical, and all the relative high power spots occur in the plates either in the side or the corner of the core.

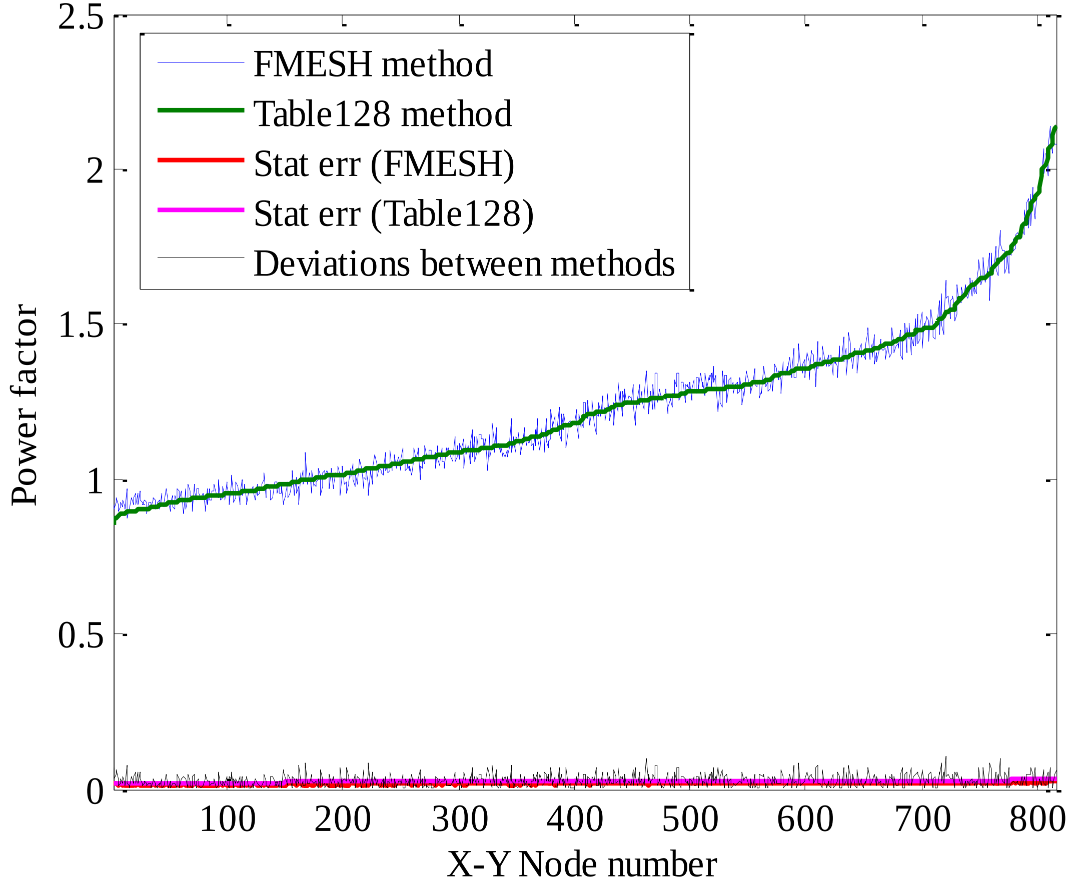
Quantitative comparison of the power factor of some hot spots in the mid-plane of the core

X	Y	FMESH	Table128	z-factor
1	1	2.100±0.025	2.072±0.032	0.689
34	1	2.077±0.023	2.139±0.032	1.573
35	1	1.917±0.021	1.916±0.031	0.027
68	1	2.081±0.025	2.132±0.032	1.256
1	6	1.889±0.021	1.841±0.030	1.311
34	6	1.988±0.019	2.005±0.031	0.468
35	6	1.979±0.019	2.034±0.032	1.478
68	6	2.073±0.023	2.066±0.032	0.178
1	7	2.047±0.023	2.012±0.031	0.907
34	7	1.975±0.019	1.996±0.031	0.578
35	7	1.991±0.019	1.979±0.031	0.330
68	7	1.856±0.021	1.908±0.031	1.389
1	12	2.136±0.026	2.069±0.032	1.625
34	12	1.937±0.021	1.892±0.030	1.229
35	12	2.101±0.023	2.106±0.032	0.126
68	12	2.062±0.024	1.999±0.031	1.607

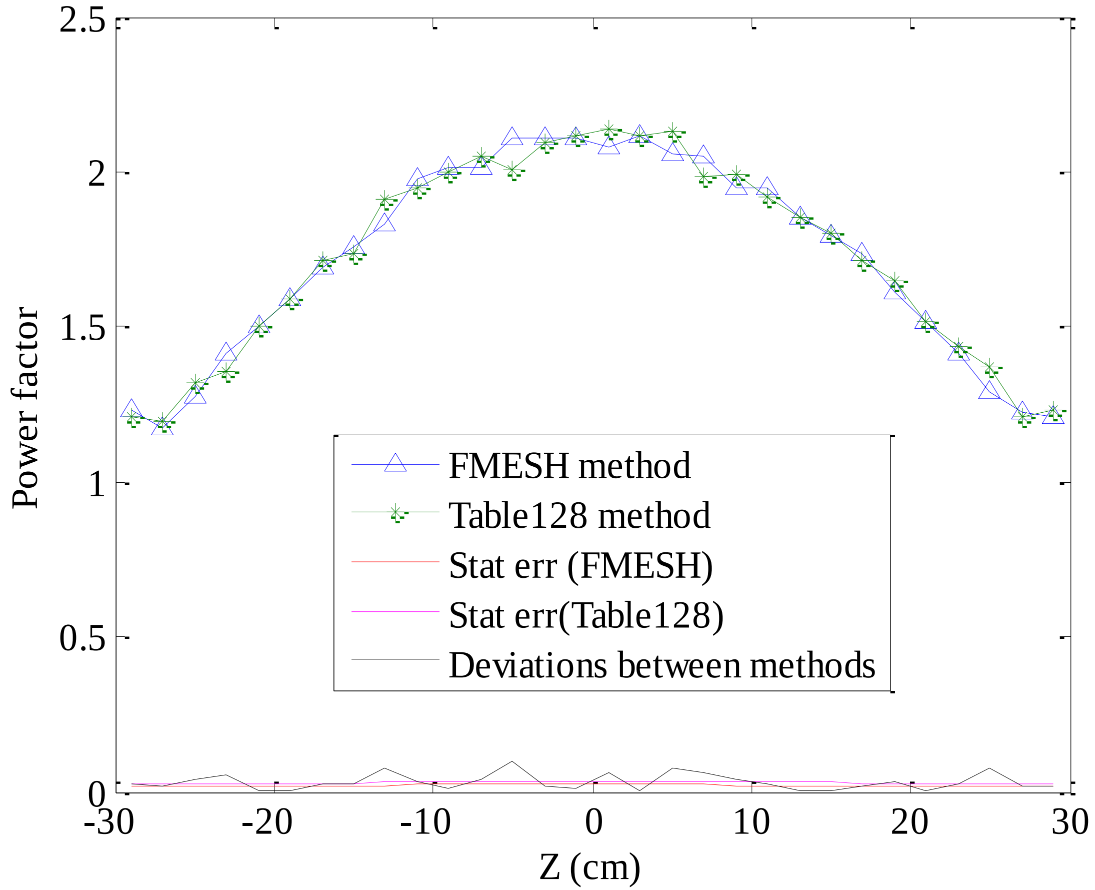
- The statistical error for FMESH method is directly provided by MCNP output.
- The errors for Table128 Method is calculated based on the assumption that the standard deviation of radiation measurement is proportional to the square root of the detected number (e.g., $1/\sqrt{N}$ principle).
- The *z-factor* is used as a measure of accuracy between two statistical quantities, it is defined as:

$$z-factor = \frac{|x_1 - x_2|}{\sqrt{\sigma_1^2 - \sigma_2^2}} \quad .$$

1D Curve Presentation of the 2D Power Factors



Comparison of Axial Power Distribution (hot channel) from Both Methods



Summary

- Two alternative methods for power distribution calculation on research reactors using MCNP are presented.
- FMESH method uses features provided by the superimposed mesh tally, which is advantageous and flexible when applied to different problems. However, computational time would increase in the case of large number of meshes.
- Table128 method uses information provided in the universe map table (Table 128), which requires no additional computational efforts. However, the cells containing fissionable materials would need to be divided into sub-cells if more refined power distribution is desired and data post-processing would be required.
- Our experience on an example problem shows these methods essentially produce statistically identical results.

Thank you!