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# Semi - Analytic Benchmarks for MCNP6

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LANL

09/15/16

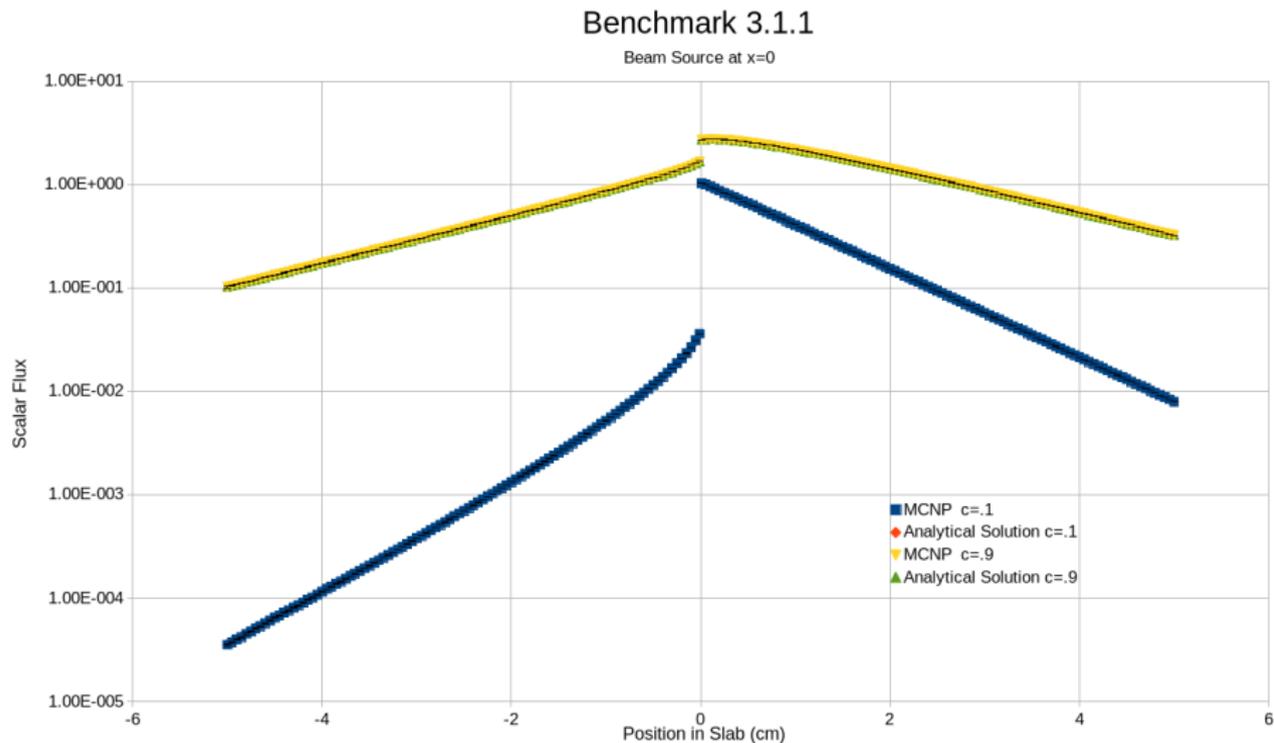
# Introduction

- Created an automated fixed source benchmark suite for code verification in MCNP6 (18 individual problems).
- Benchmarks come from Barry Ganapol's book, *Analytical Benchmarks for Nuclear Engineering Applications*.
- Compared analytical results with MCNP6 using Python.

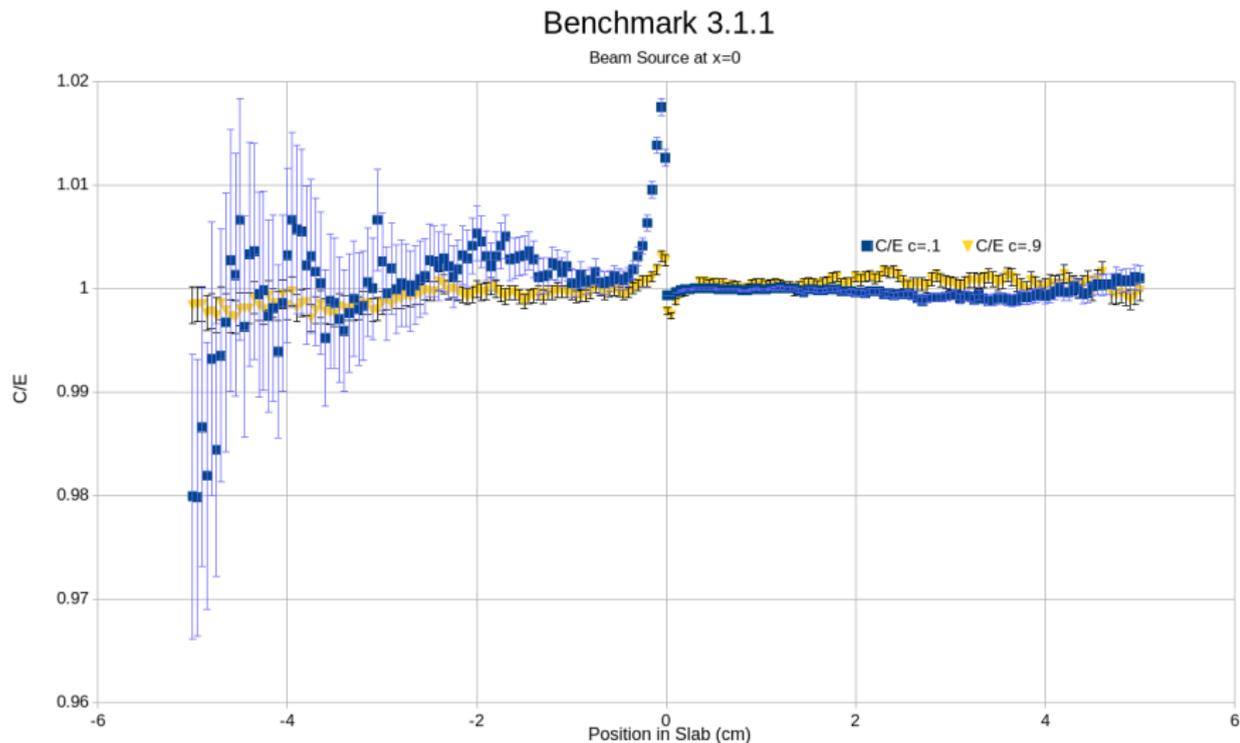
Benchmark	Geometry	Source	Varied
3.1	Infinite Medium	Beam/Isotropic	Source/Beam Angle
3.2	Half Space	Beam/Isotropic	Source
3.3	Finite Slab	Beam	Slab Thickness
3.4*	Infinite Cylinder	Volume Source	Radius

\*Fixed Source & Criticality Problems

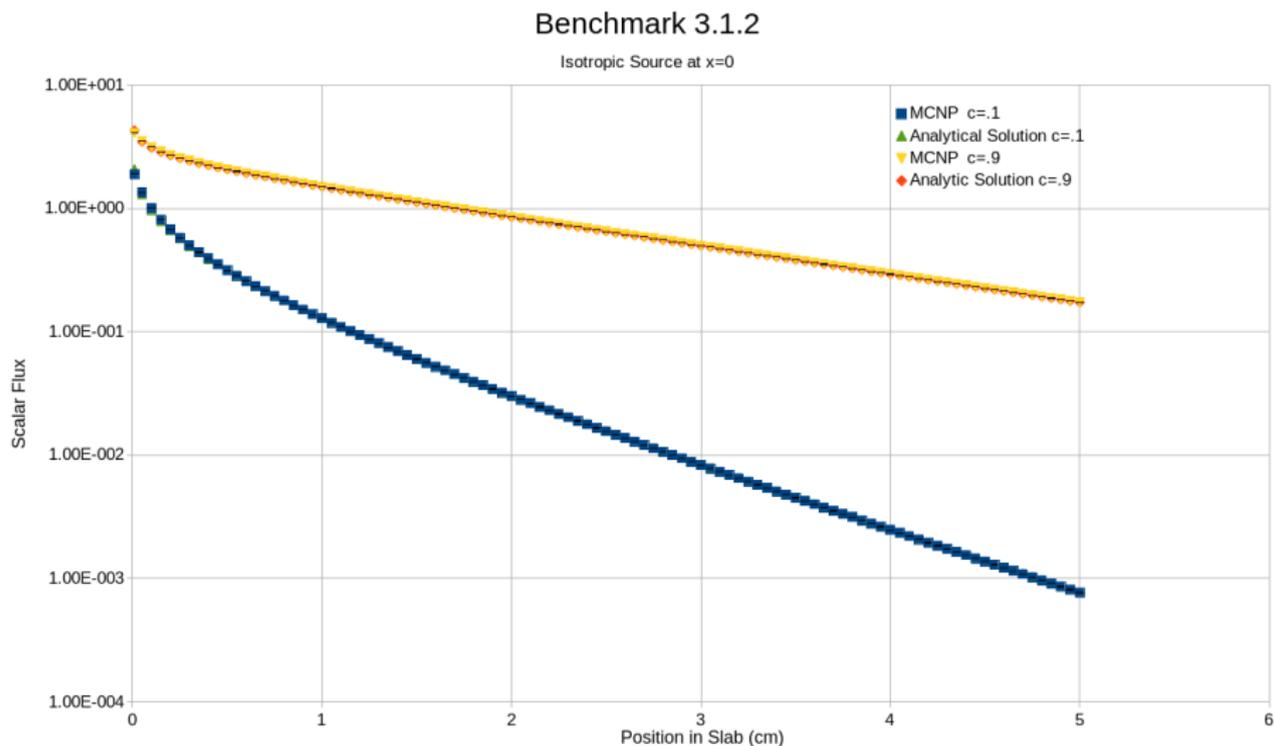
# 3.1 Beam Source in Infinite Medium



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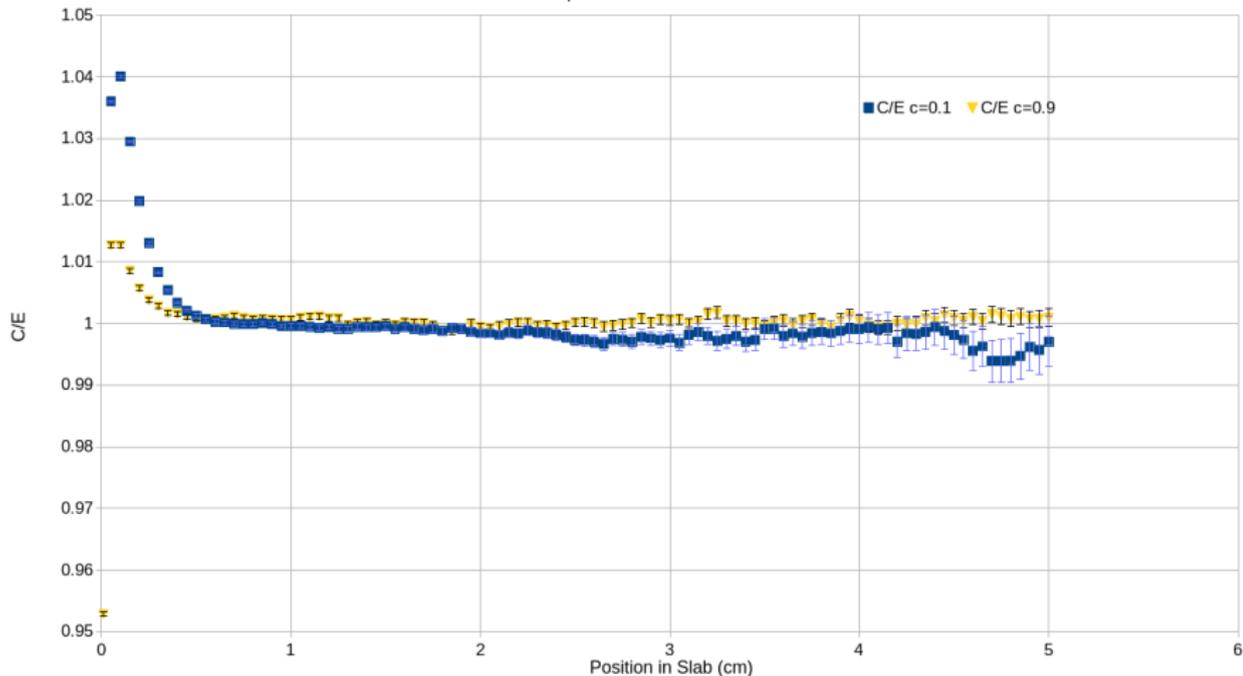


# 3.1 Isotropic Source in Infinite Medium

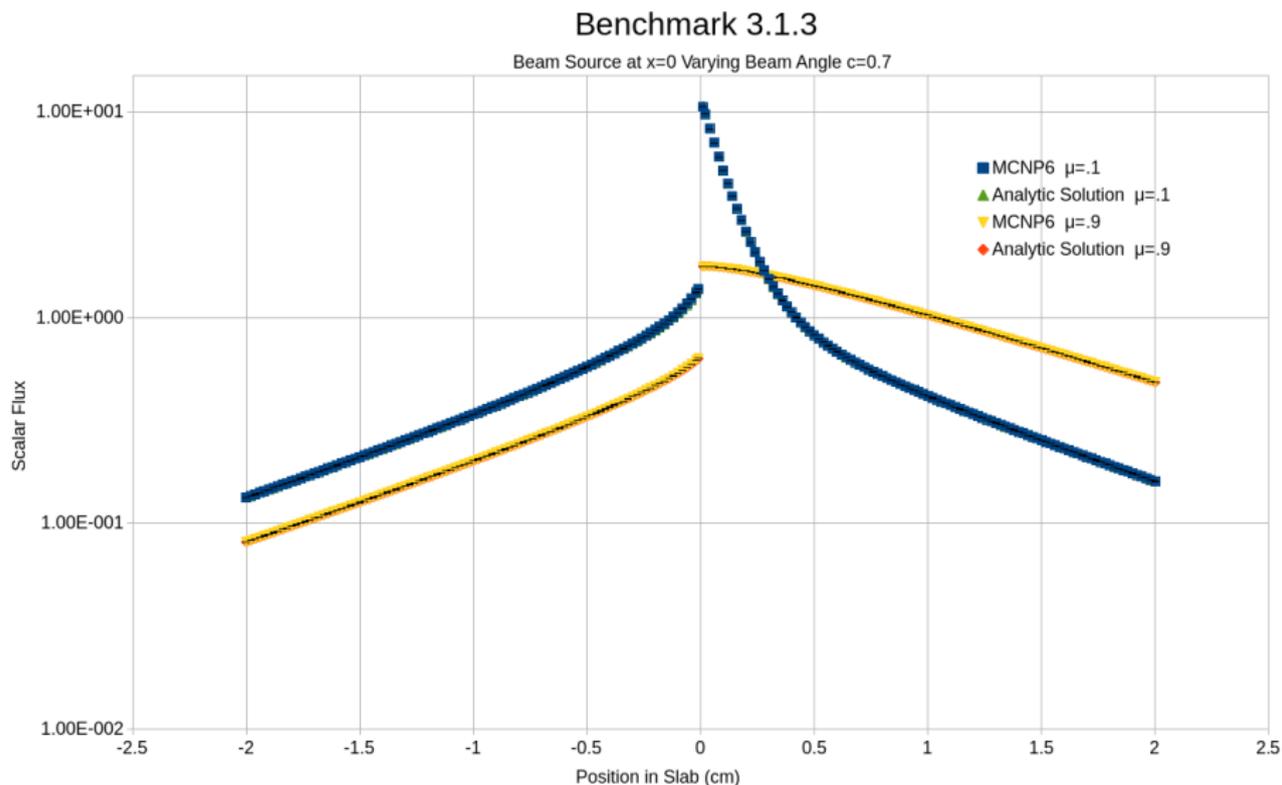


# 3.1 Isotropic Source in Infinite Medium

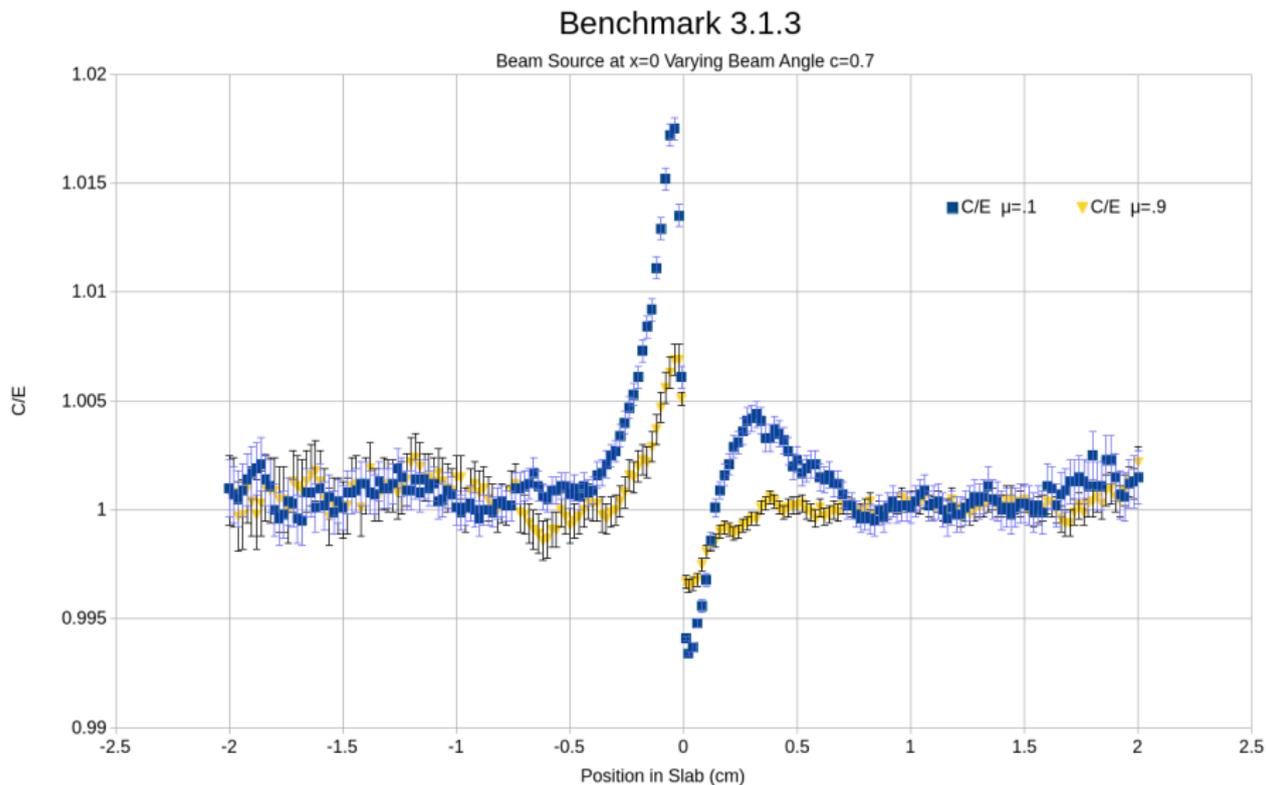
Benchmark 3.1.2

Isotropic Source at  $x=0$ 

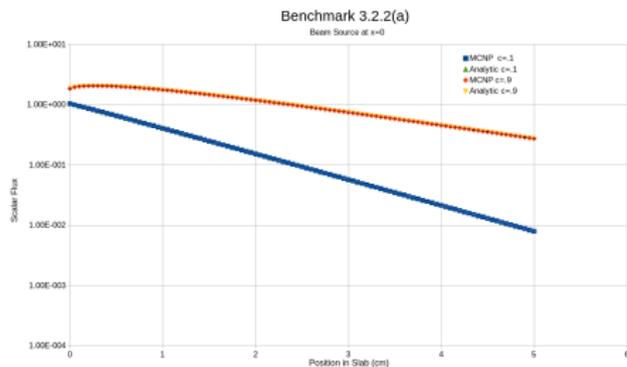
# 3.1 Varying Beam Angle in Infinite Medium



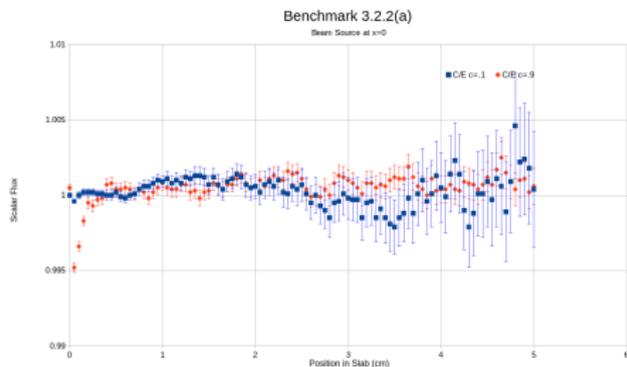
# 3.1 Varying Beam Angle in Infinite Medium



## 3.2 Beam in Half Space

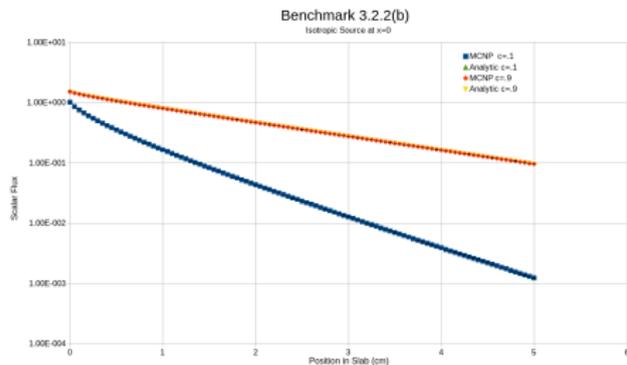


Flux Shapes

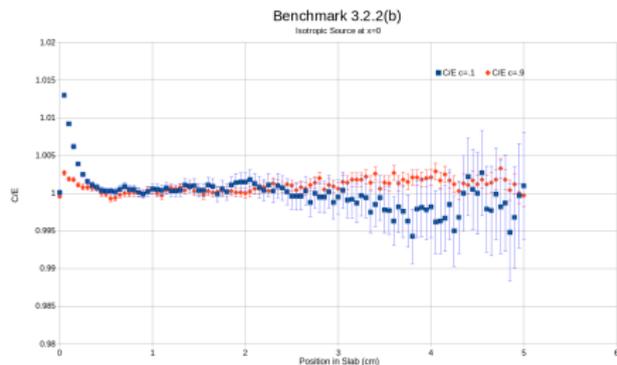


MCNP6 Divided by Analytic Solution

## 3.2 Isotropic Source in Half Space



Flux Shapes



MCNP6 Divided by Analytic Solution

## F2 Surface Flux Tally

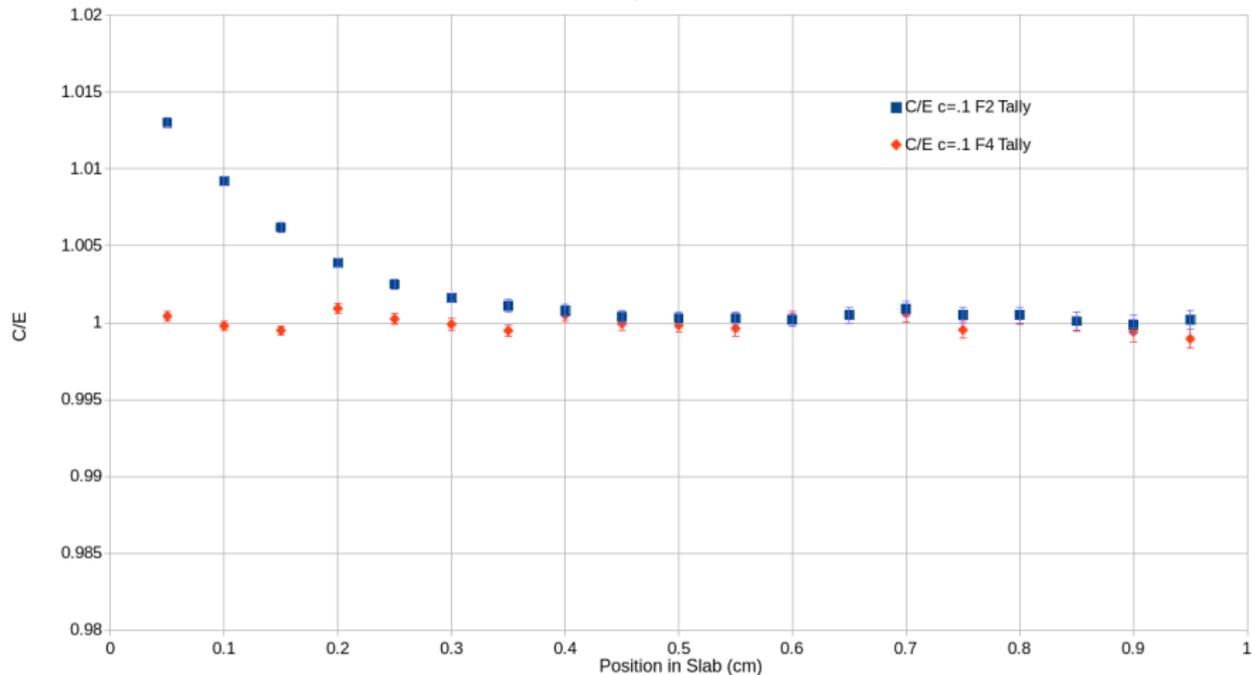
Below  $|\mu| < .1$  MCNP6 makes a constant contribution of  $|\mu| = .05$  to the F2 tally (isotropic flux is assumed), in order to ensure that the variance is finite and to maintain good statistics.

$$\phi = \frac{1}{A * W} \sum \frac{wgt}{|\mu|} \quad (1)$$

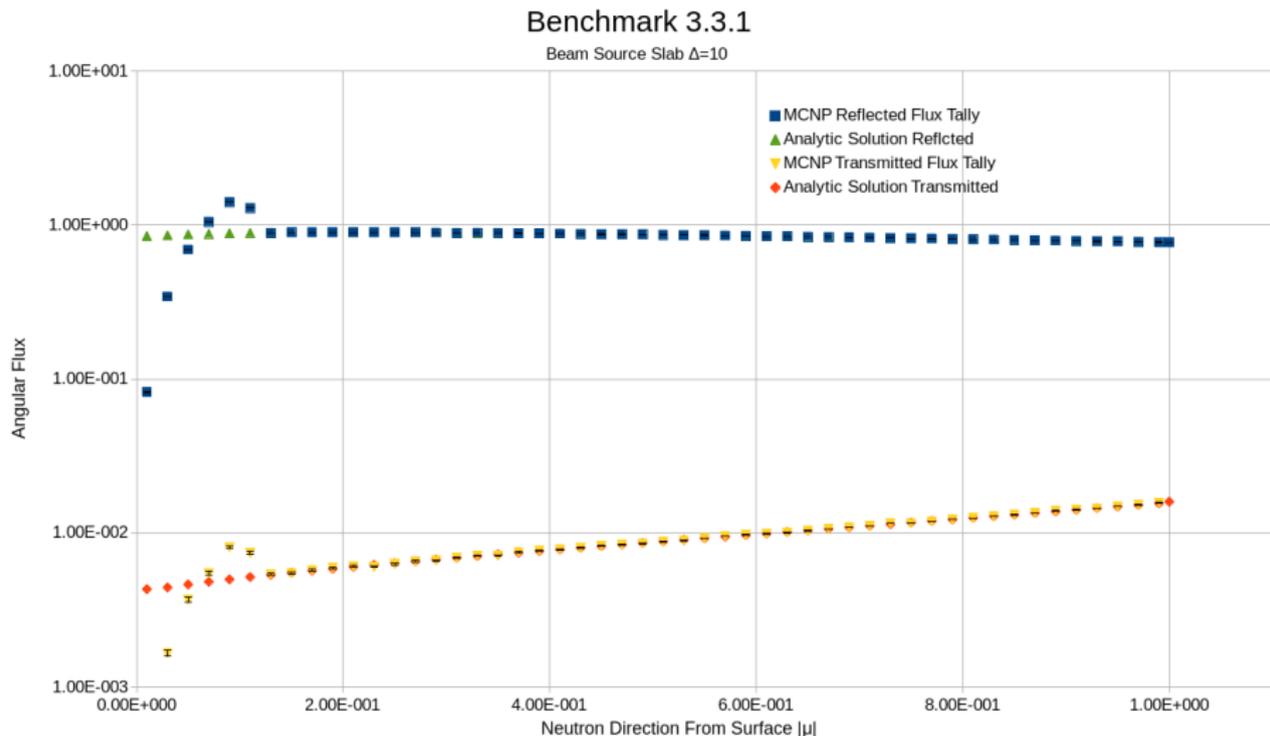
To be sure that the F2 tally is the cause for the difference in values, some problems were reworked with different tallies.

## 3.2 Isotropic Source in Half Space with F4 Tally

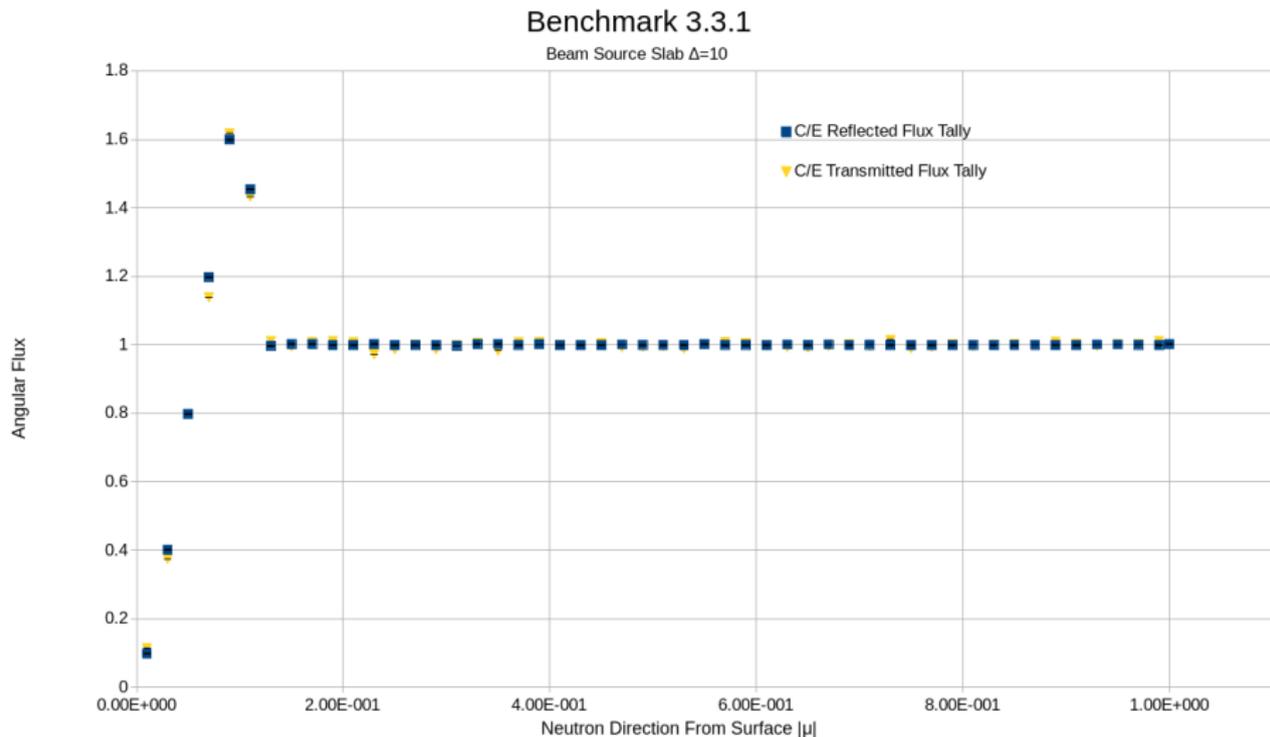
Benchmark 3.2.2(b)

Isotropic Source at  $x=0$ 

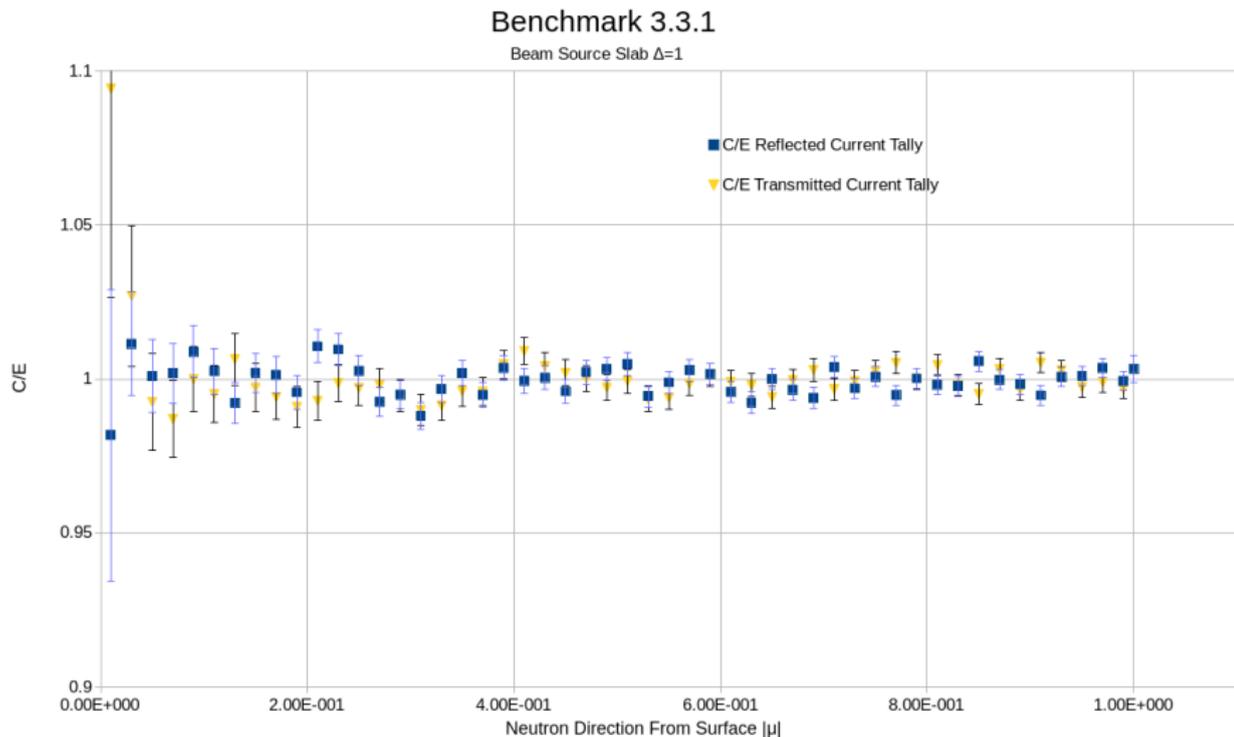
# 3.3 Beam Through Finite Slab



# 3.3 Beam Through Finite Slab

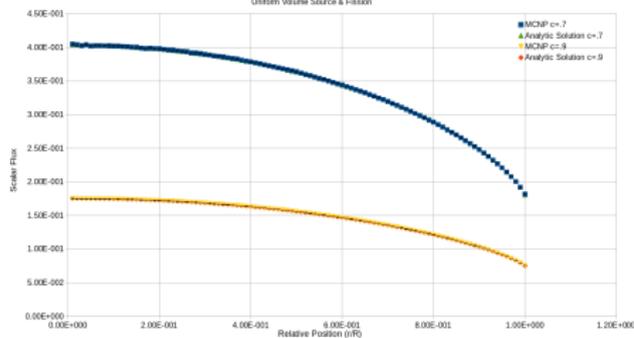


# 3.3 Beam Through Finite Slab with F1 Tally



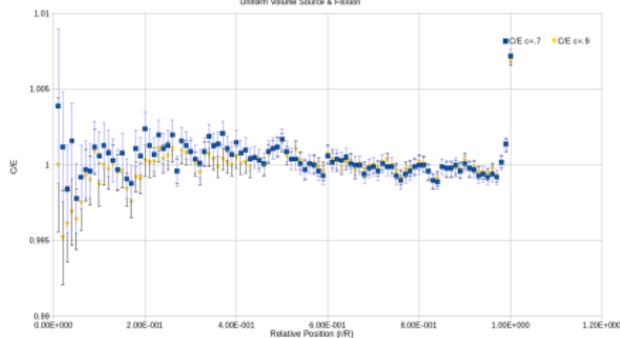
# 3.4 Infinite Cylinder

Benchmark 3.4.1  
Uniform Volume Source & Fission



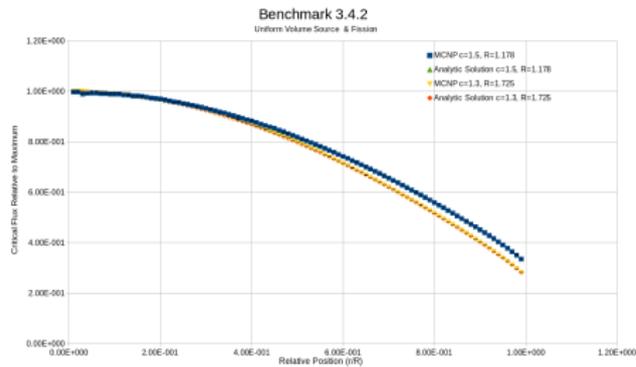
Radial Flux Shape

Benchmark 3.4.1  
Uniform Volume Source & Fission

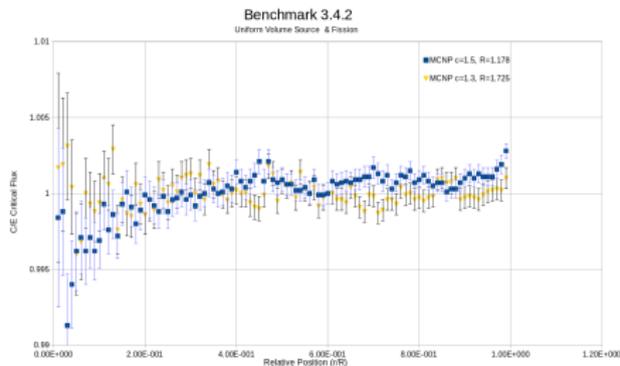


MCNP6 Divided by Analytic Solution

# 3.4 Critical Infinite Cylinder



Critical Flux Profile  $\frac{\phi(r)}{\phi(max)}$

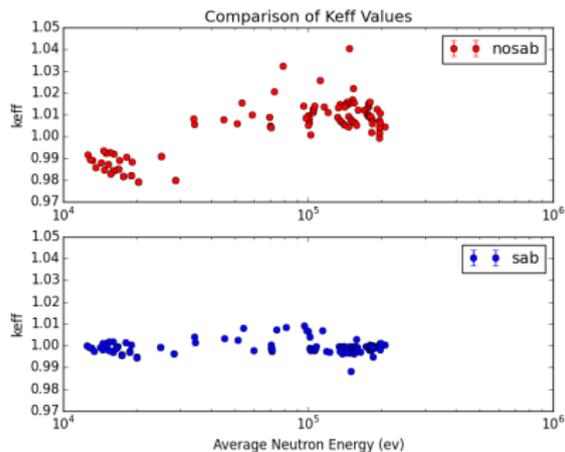


MCNP6 Divided by Analytic Solution

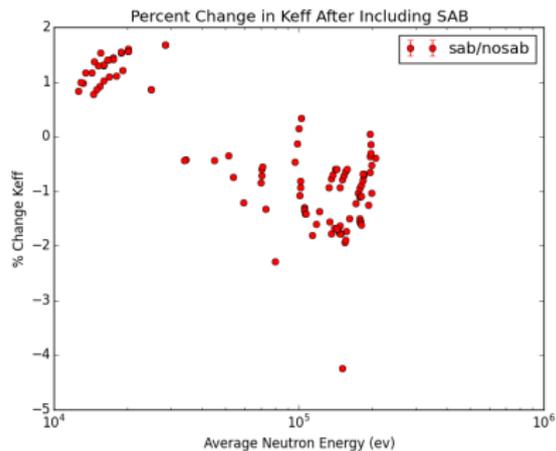
# Discussion

- The F2 tally can cause issues in high scattering systems and near sources.
- To mitigate this effect the F4 tally should be used for scalar flux measurements, and the F1 tally should be used for angular flux measurements.
- For the most part MCNP6 results match the analytical solution within statistics.

# Other Work



Effect of  $S(\alpha, \beta)$  of  $k_{eff}$



Percent Change in  $k_{eff}$

# Future Work

- Add slowing down (chapter 2) or multidimensional (chapter 4) benchmarks from Barry Ganapol's book.
- Rework problems using the F4 tally.
- Compare continuous cross-sections and multigroup cross-sections.
- Add more systems to document effect of  $S(\alpha, \beta)$  on  $k_{eff}$ .

# Acknowledgements

- Thank you to Michael Rising, Forrest Brown, and the XCP-3 group at LANL for the support with this project.
- This work was supported by the DOE Nuclear Criticality Safety Program.

# Questions

