Translation from Unstructured Meshes to Combinatorial Geometry for Nuclear Energy Applications

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Introduction

Monte Carlo methods are invaluable in generating the nuclear data needed for simulating neutron behavior.

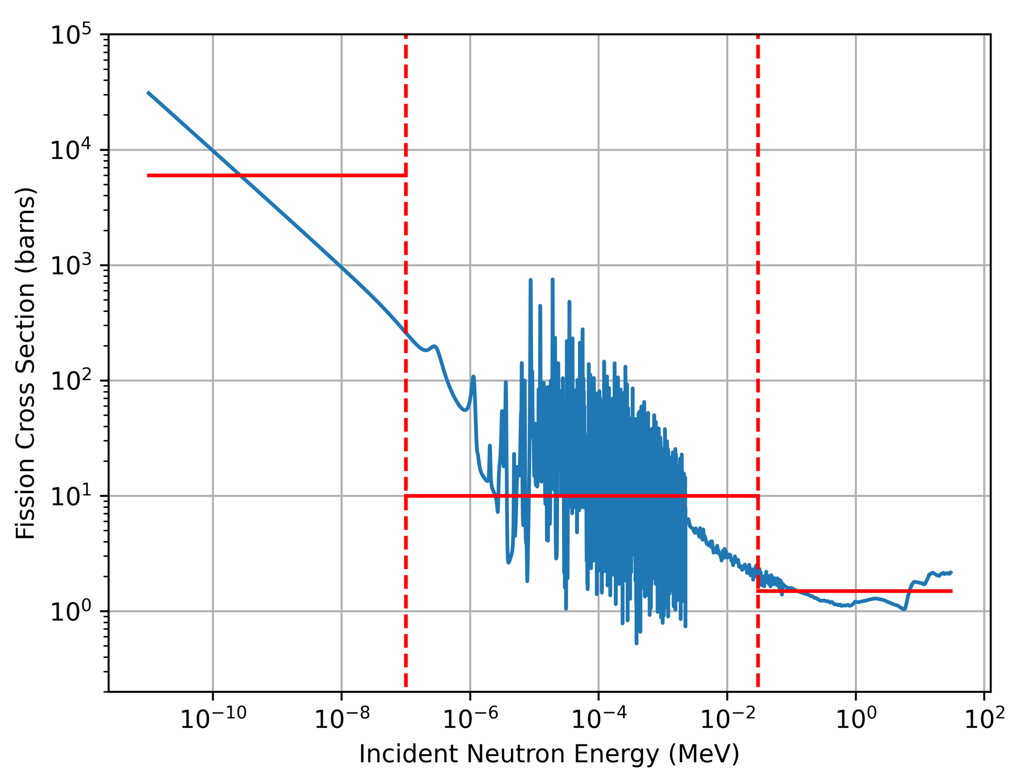
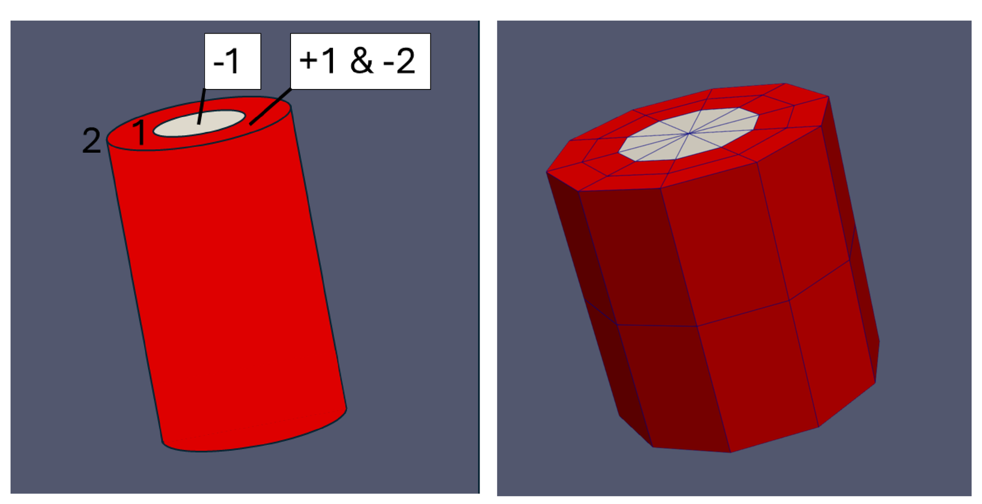


Figure 1: Continuous and multigroup ^{235}U fission cross sections

Deterministic methods run faster and allow for design optimization and time-dependent studies. Monte Carlo and deterministic methods use unique geometry representations:



(a) Monte Carlo, combinatorial solid geometry (CSG) model **(b)** Deterministic, unstructured mesh model

Figure 2: CSG and unstructured mesh models of concentric cylinders

Goal

We are developing translational models to recreate unstructured meshes created in MOOSE as CSG representations in OpenMC. These models are being integrated into the MOOSE meshing library, specifically the `reactor` module.

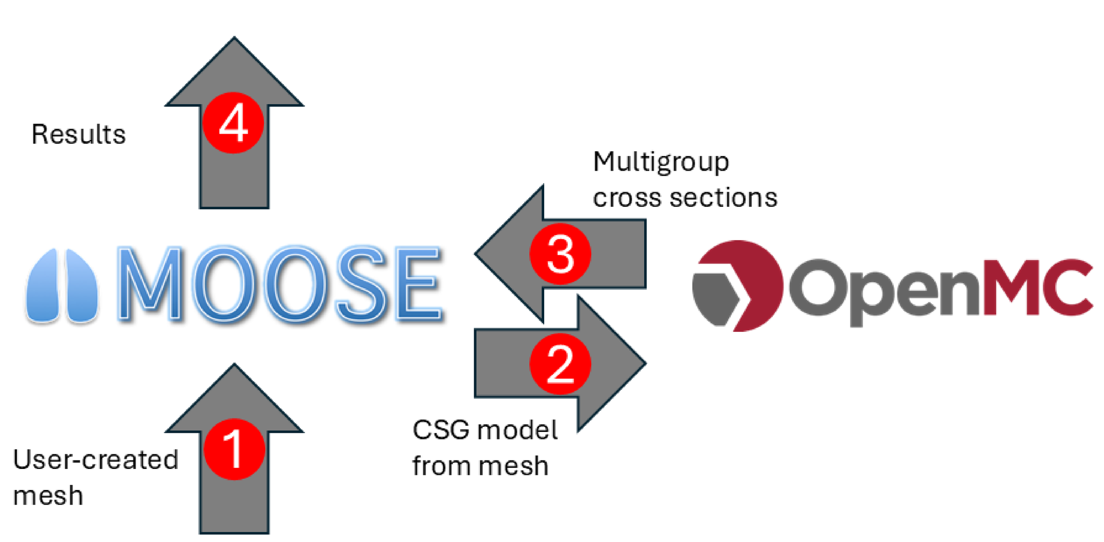


Figure 3: Desired simulation workflow

Methods and Results

Implementation of Translational Models

Mesh generation in MOOSE operates on a mesh in steps using `MeshGenerator` objects. Each object provides information about the type of geometry being created and the input parameters.

OpenMC input files are in the form of XML files. OpenMC contains functions to construct the model geometry from `pugixml` objects. We implement the translational models by constructing the corresponding `pugixml` tree in each `MeshGenerator` object.

```
[Mesh]
  [ccyl]
    type = AdvancedConcentricCircleGenerator
    ring_intervals = '1 2'
    ring_radii = '1 2'
    num_sectors = 10
  []
  [extrude]
    type = AdvancedExtruderGenerator
    direction = '0 0 1'
    heights = '4'
    input = ccyl
    num_layers = 2
  []
[]
```

Figure 4: MOOSE input file with consecutive mesh generators

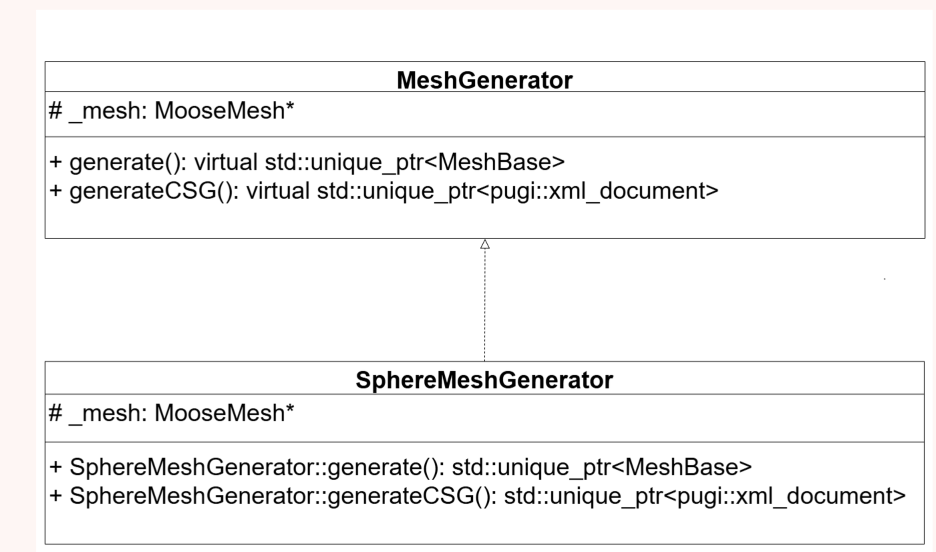


Figure 5: UML diagram of the implementation of the `MeshGenerator` class

OpenMC geometry consists of surfaces, cells bounded by surfaces, and universes filled with cells. The corresponding `pugixml` tree consists of a root node with additional nodes for surfaces and cells:

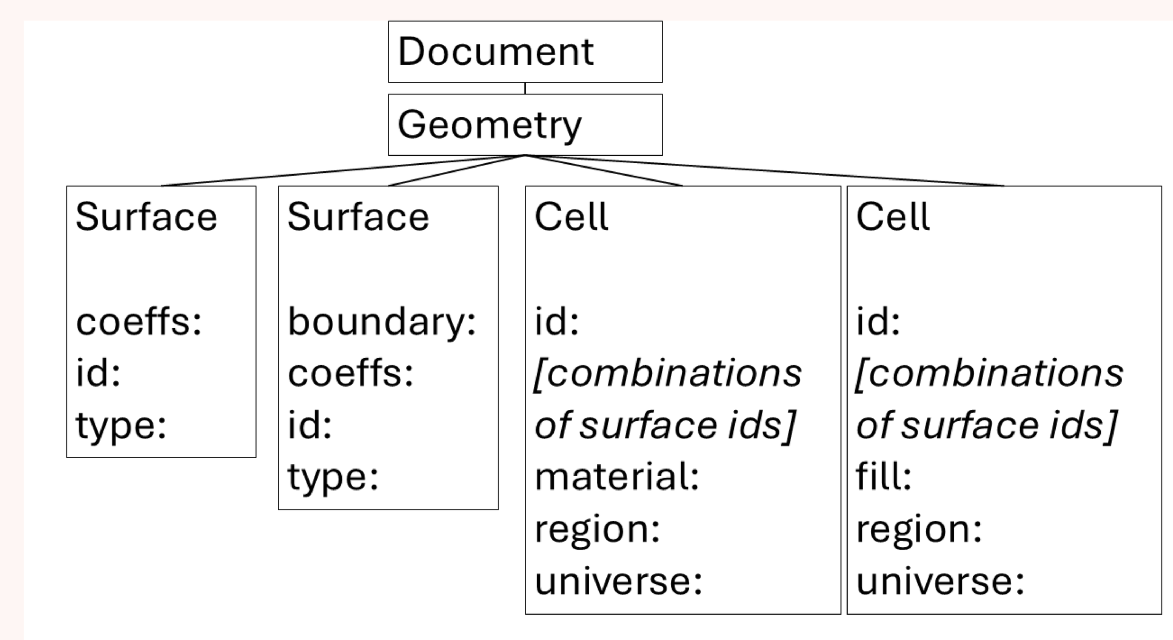
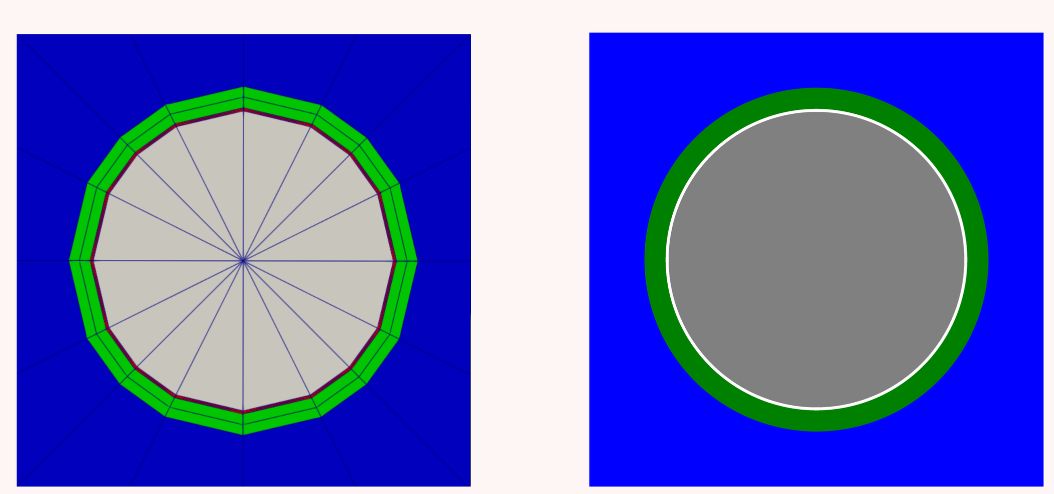


Figure 6: Structure of XML tree to represent CSG geometry

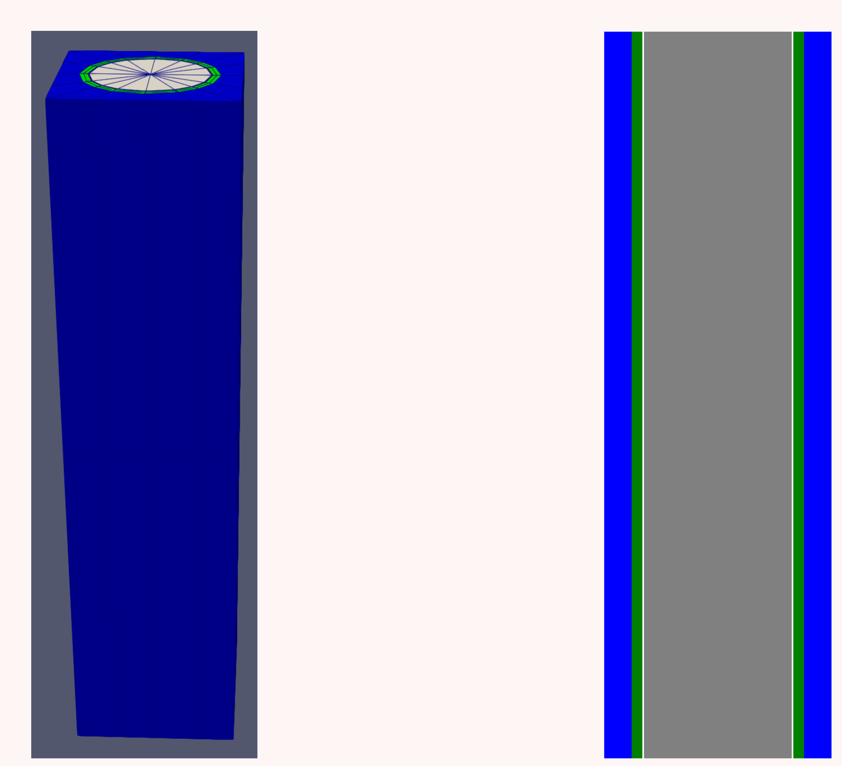
Pin Cell Demonstration Model

A single nuclear fuel pin is a simple geometry common in nuclear reactors that can be used as a proof-of-concept. The fuel pin consists of concentric cylinders surrounded by a square and is created with two mesh generators:



(a) MOOSE pin cell, first created in 2D **(b)** OpenMC pin cell, first created as infinite in height

Figure 7: First step in creating the mesh and CSG models of the pin cell

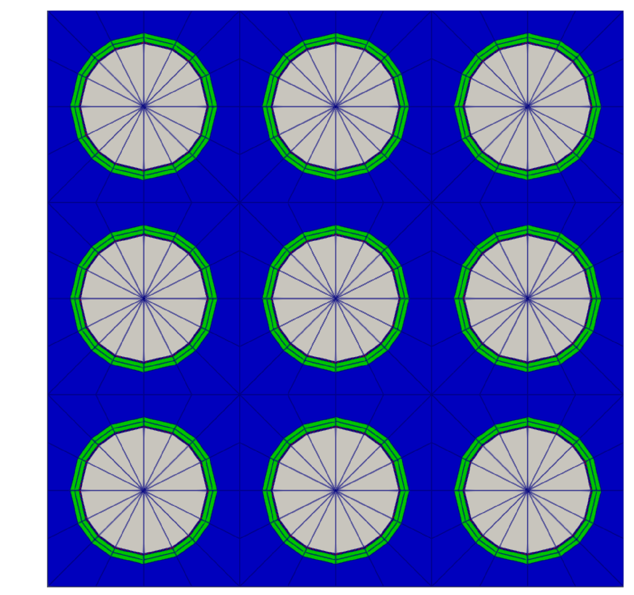


(a) MOOSE pin cell, extruded to 3D using second mesh generator **(b)** OpenMC pin cell, bounded in 3D by top and bottom planes

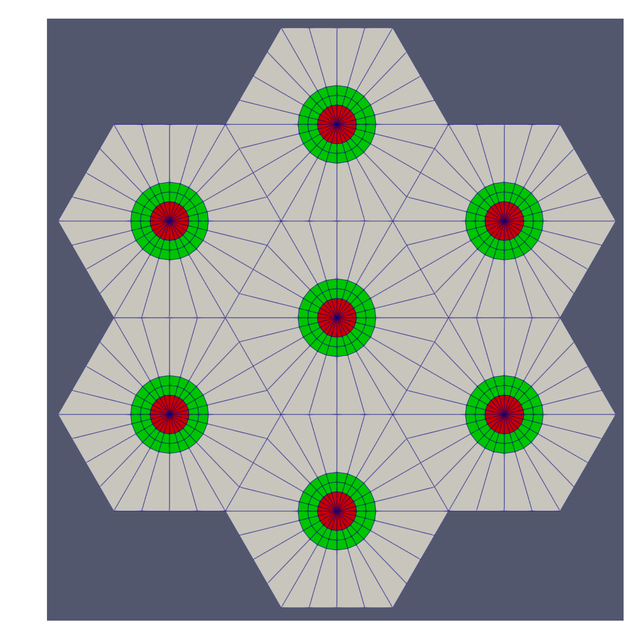
Figure 8: Second step in creating the mesh and CSG models of the pin cell

Future Directions

- Pattern unit cells into lattices for full core modeling:



- Add support for hexagonal cells (common geometry in high temperature gas cooled reactors):



- Find workaround for issues with working with surfaces with boundary conditions and mesh generators that modify an existing mesh
- Find way to leverage MOOSE id system to automatically assign materials to cells in CSG models

References and Acknowledgments

References:

[1] Brookhaven National Laboratory. Evaluated Nuclear Data File (ENDF). `https://www.nndc.bnl.gov/endf/`, 2024. Accessed: 2025-04-17.

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[4] Paul K. Romano et al. OpenMC: A State-of-the-Art Monte Carlo Code for Research and Development. *Annals of Nuclear Energy*, 82:90–97, 2015.

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