MSRE CROSS SECTION GENERATION

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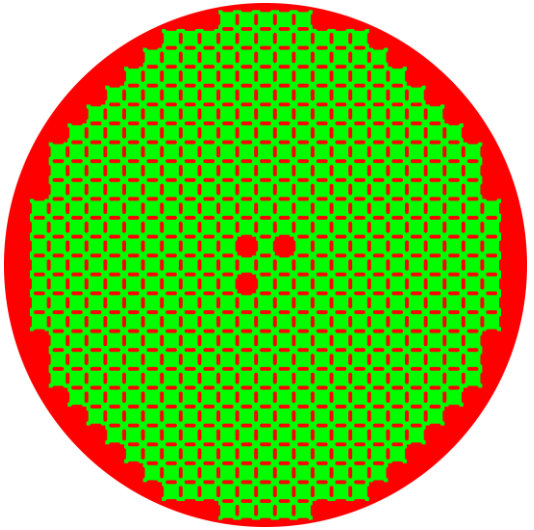
OPENMC MGXS INTRODUCTION

- Using OpenMC MGXS module to automatically create multi-group tallies.
  - Manual: https://docs.openmc.org/en/stable/pythonapi/mgxs.html
- Information required
  - Energy structure
  - Cells/Universes/Materials of interest
  - Reaction type: total, transport, nu-transport, absorption, reduced absorption, capture, fission, nu-fission, kappa-fission, scatter, nu-scatter, scatter matrix, nu-scatter matrix, multiplicity matrix, nu-fission matrix, scatter probability matrix, consistent scatter matrix, consistent nu-scatter matrix, chi, chi-prompt, inverse-velocity, prompt-nu-fission, prompt-nu-fission matrix, current, diffusion-coefficient, nu-diffusion-coefficient
  - macro or micro

MSRE CROSS SECTION GENERATION (I)



- Step 1: create mgxs.Library
  - domains are the cells in the fuel lattice and control rod lattice
  - geometry is the openmc.Geometry object
  - scatt_order = 1
  - domain_type is cell
  - by_nuclide is True

```
lib = mgxs.Library(geometry)
lib.energy_groups = mgxs.EnergyGroups(groups)
lib.mgxs_types = ["total", "absorption", "nu-fission", "fission", "chi",
                  "consistent nu-scatter matrix", "consistent scatter matrix",
                  "kappa-fission"]
lib.correction = None
lib.legendre_order = scatt_order
lib.domain_type = domain_type
lib.domains = domains
lib.by_nuclide = by_nuclide

lib.build_library()
```

MSRE CROSS SECTION GENERATION (II)

- Step 2: add the tallies to the openmc tallies object

```
mgxs_lib = PrepareLibrary(talcl,'cell',1,energy,geom,True)
tals = omc.Tallies()
mgxs_lib.add_to_tallies_file(tals,merge=True)
tals.export_to_xml()
```

- Step 3: run openmc
- Step 4: load openmc statepoint file back into the magx.Library and use external scripts to convert the cross section into isoxml format
  - convert to ISOTXS, then use Griffin utility program to convert ISOTXS to isoxml.
  - direct conversion to isoxml using Griffin utility program.