Parameter interpolation for MSR core physics modules

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Physics & Technology Modeling

Kyle Anderson
Steve Skutnik, Alex Wheeler, Ondrej Chvala
Outline

• Overall Research Project and Motivation
• Reactor Core Physics Module
• Reactor Data Libraries
• Interpolation and Spacing of Libraries
Project Goal and Motivation

• Project Goal is:
  • Develop a modular software framework to allow the testing of MC&A methods on a variety of MSR designs

• The Motivation is:
  • No reactors to test methods on
  • Large variety in the design space
Block Diagram of a Simple Model
Reactor Core Physics Module - Requirements

• Requirements:
  • Calculate depletion of fuel salt and PK data
  • Fast computational solution time
  • Ability to cover large design space
Reactor Core Physics Module - Methodology

• Requirements:
  • Calculate depletion of fuel salt and PK data
  • Fast computational solution time
  • Ability to cover large design space

• Methodology:
  • Utilize pre-generated collapse data libraries to run point depletion calculations
Reactor Core Physics Module - Methodology

• Requirements:
  • Calculate depletion of fuel salt and PK data
  • Fast computational solution time
  • Ability to cover large design space

• Module Methodology:
  • Break up module by large scale design parameter (Fuel cycle, Spectrum type, etc.)
  • Interpolate over smaller design parameters
Reactor Core Physics Module Diagram

On Run Startup

- Select Library Group
- Input Parameters (Core Size, Enrichment, etc.)
- Generate ORIGEN library with OBIWAN interpolation

During Run

- Isotopic & Mass Flow Data
- Module IN
- Call ORIGEN for Point Depletion Calculation
- Calculate Point Kinetics Data
- Module OUT
- Updated Isotopic, Mass Flow, & Point Kinetics Data
Interpolating between Data Libraries

• Reactor data libraries are interpolated over transition matrices

• The transition matrices can be studied to understand:
  • Importance of parameter
  • Predicted error in interpolation
### Thermal FLIBE Group Library

#### Quarter Core Geometry

- **Fuel Salt**
- **Steel Vessel**
- **Graphite Structure**
- **Air**

#### Parameter Table

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Range</th>
<th>Spacing</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Uranium Enrichment [wt. % U-235]</strong></td>
<td>2 – 10</td>
<td>1</td>
</tr>
<tr>
<td><strong>Lattice Pitch [cm]</strong></td>
<td>10 – 50</td>
<td>10</td>
</tr>
<tr>
<td><strong>Salt-Moderator Ratio</strong></td>
<td>0.1 – 0.5</td>
<td>0.1</td>
</tr>
<tr>
<td><strong>Number of Fuel Channels</strong></td>
<td>7 – 169</td>
<td>Varies</td>
</tr>
</tbody>
</table>
Averaged Effect of Lattice Pitch

Effect of Lattice Pitch on Transition Matrix Values Across Burnup (0 - 23.7 GWd/MTIHM)

- **Average Value**
- **Range of Values**

Average Relative Difference of Fission and Removal Transition Matrices

Lattice Pitch

Increasing Burnup
Pu-239 Mass Difference

Effect of Transition Matrix Differences on Plutonium-239 Production

Relative Difference of Pu-239 Mass

Relative Difference in Transition Matrix for Actinide Reactions
Library Spacing Algorithm

1. Start Spacing Algorithm
2. For Each Library in List
   - Determine Best Nearest Libraries
   - Interpolate to Selected Library
   - Calculate Error of Interpolation
3. Determine Interpolation with Lowest Error
4. Is Error < Error Tolerance?
   - Yes: Drop Library with Lowest Error from List
   - No: End Spacing Algorithm
Library Spacing Algorithm

Determining Next Libraries to Interpolate

• Potential to use difference in transition matrix value to determine next library
Acknowledgments

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Summary

• A modular framework for testing MC&A methods on MSRIs is being developed
• Core physics in the framework is being done using reactor data libraries
• The spacing and interpolation of the libraries is being studied through transition matrices
Questions or Clarifications?

Additional Clarification Slides:

- Molten Salt Reactor Design Space
- Definition of Transition Matrix
- Averaged Effects of Enrichment and Salt-Moderator Ratio
MSR Design Space

Neutron Spectrum
- Thermal
- Fast
- Spatially or Temporally Varying

Fluid Number
- Single
- One-and-half
- Dual

Fuel Cycle
- Th/U
- U/Pu

Additional Parameters
- Salt Choice
- Fissile Content (Enrichment)
- Salt-Moderator Ratio
- Pitch
Transition Matrix

Nuclide Generation, Depletion, and Decay

\[
\frac{dN_i}{dt} = \sum_{j\neq i} (l_{ij} \lambda_j + f_{ij} \sigma_j \phi) N_j(t) - (\lambda_i + \sigma_i \phi) N_i(t) + S_i(t)
\]

Term Definitions

\(N_i\) = Amount of nuclide i
\(l_{ij}\) = Fractional yield of nuclide i from decay of nuclide j
\(\lambda_j\) = Decay constant of nuclide j
\(f_{ij}\) = Fraction yield of nuclide I from neutron-induced transmutation of nuclide j
\(\phi\) = Neutron flux
\(S_i\) = Source term
Transition Matrix

Nuclide Generation, Depletion, and Decay

\[
\frac{dN_i}{dt} = \sum_{j \neq i} \left( l_{ij} \lambda_j + f_{ij} \sigma_j \phi \right) N_j(t) - (\lambda_i + \sigma_i \phi) N_i(t) + S_i(t)
\]

Decay in  Transmute in  Decay out  Transmute out  Source

Simply with \( T \) as “Transition Matrix”

\[
\frac{dN}{dt} = TN(t) + S(t)
\]
Averaged Effect of Enrichment

Effect of Enrichment on Transition Matrix Values Across Burnup (0 - 23.7 GWD/MTIHM)

- Average Value
- Range of Values

Average Relative Difference of Fission and Removal Transition Matrices

Increasing Burnup

Enrichment (U-235 wt%)
Averaged Effect of Salt-Moderator Ratio

Effect of Salt-Moderator Ratio on Transition Matrix Values Across Burnup (0 - 23.7 GWD/MTIHM)