MONTE-CARLO COUPLED DEPLETION CODES EFFICIENCY FOR RESEARCH REACTOR DESIGN

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Introduction

- **TechnicAtome**: specializes in the design, construction, operation and maintenance of compact nuclear reactors

- **Early stages of core design** and industrial studies require a **quick and efficient calculation** of key neutronic parameters at any time
  - Mainly achieved by **deterministic calculation schemes**
  - COCONEUT (COre COnception NEUtronic Tool)

- Nevertheless
  - Deterministic codes: **problem dependent / V&V process** for various kinds of cores
  - Improvement of CPU power: Monte-Carlo burnup calculations for industrial studies

- **Aims of this paper:**
  - Monte-Carlo burnup codes for industrial studies *(TRIPOLI4®, MCNP, Serpent)*
  - Describe a case study part of the V&V process undergone by COCONEUT

- **Case study:**
  - A multipurpose dummy core designed by TechnicAtome
Contents

1. Depletion calculation methods

2. Codes used in this study

3. Case study: Dummy core

4. Results and Analysis

5. Conclusion and Outlooks
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Depletion calculation methods (1/2)

General diagram for depletion calculation

- Step 1: Deterministic
  - New compositions calculation
  - 1st or 2nd order methods

- Step 2: Material balance 2

- Step 3: Material balance 3

Deterministic / stochastic calculation?
Depletion calculation methods (2/2)

**Deterministic approach**
- Fast method for flux calculation → industrial studies
- Cross sections collapsing
- Self shielding
- Spatial mesh
- Time related mesh
- Geometry dependent

Approximations / biases to quantify

**Stochastic approach**
- Exact 3D geometry
- Punctual XS for flux calculation
- **Slower** than deterministic calculation
- Statistical uncertainties
- Spatial mesh for depletion
- Time related mesh

Results depending on statistical convergence
- Uncertainties propagation
- Time consuming
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Codes used in this study: **MC codes**

**TRIPOLI4®**
- Code developed by CEA (French Alternative Energies and Atomic Energy Commission)
- **Safety studies reference** at TechnicAtome
- Polyvalent code
- Large V&V process
- **Root based interfaces** (pre / post processing)
  - Geometry modification during depletion
  - Refueling module
  - Possibility to develop a tool for uncertainties propagation

**MCNP**
- **International reference**
- Code largely benchmarked
- Many applications at TechnicAtome
- Assessment of JHR neutronic performances

**Serpent**
- **Fast**
- New methods (perturbation, coupled physics…)
- Automatic mesh
- Undergoing V&V process
1) XS generation
   - Standard FA
   - Supercritical pattern for Absorber FA

2) Core calculation
   - 2D model (exact)
   - 3D model

Validation: fuel pattern and core
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Case study: Dummy core (1/2)

**AIMS**
- Comparing methodologies / non-regression tests
- Education and Training object
- Validation and qualification of both calculation and computational techniques
- V&V purposes

**DESIGN**
- Only describe two assembly types
- Simple to model
- Fuel lattice pattern
- Add components
  - Reflector vessel (heavy water...)
  - Experimental devices
- Add ex-core environment

**Standard Fuel Assembly (SFA)**

**Absorber Fuel Assembly (AFA)**

Depending on the case study

<table>
<thead>
<tr>
<th>Material</th>
<th>SFA</th>
<th>AFA</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel (MTR type)</td>
<td>8 cm x 0.8 cm</td>
<td>8 cm x 0.3 cm</td>
</tr>
<tr>
<td>Cladding</td>
<td>0.2 cm</td>
<td>0.2 cm</td>
</tr>
<tr>
<td>Aluminum</td>
<td>0.2 cm</td>
<td>0.2 cm</td>
</tr>
<tr>
<td>Boron</td>
<td>0.356 cm</td>
<td>0.3055 cm</td>
</tr>
<tr>
<td>Water</td>
<td>8.288 cm</td>
<td>7.6 cm</td>
</tr>
<tr>
<td>Hafnium</td>
<td>9 cm</td>
<td>9 cm</td>
</tr>
<tr>
<td>[1/2]</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Fuel (MTR type) Cladding Aluminum Boron Water Hafnium
Case study: Dummy core (2/2)

Fuel assembly lattice configuration (2D)

- 3 configurations
  - Temperature: 300°C
  - Total: 62 depleted medium

**Full core configuration (2D)**

- 32 assemblies core
  - 16 SFA / 16 AFA
  - Reflector / coolant: light water
  - Fuel lattice: 10 cm
  - Total: 672 depleted medium

For more information

Dummy Core for V&V and Education & Training Purposes at TechnicAtome: In and Ex-Core Calculations

S. Nicolas, A. Noguès, L. Manifacier, L. Chabert
Results and analysis: **Standard FA (1/3)**

- **Benchmark considerations**
  
  - The same consistent parameters are taken into account for each code and simulation

  - Reflecting surface are defined as boundary conditions
  - **50 burnup steps** with a maximum value of **100 GWd/tU**
  - Assembly power of: **1.5625 MW<sub>th</sub>**
  - **Depletion** in fuel and boron plates
  - Temperature: **300 K**
  - **JEFF-3.1.1** nuclear data library
Results and analysis: **Standard FA (2/3)**

- **Standard FA: multiplication factor comparison**

  - **$k_{inf}$ comparison**

  ![Graph showing $k_{inf}$ comparison](image)

  - **Reactivity comparison (mean value of the three MC codes as reference)**

  ![Graph showing reactivity comparison](image)

- **Initial $k_{inf}$**
  - Monte Carlo codes are all the same within the $1\sigma$ range / **+150 pcm** bias for COCONEUT

- **Depletion:**
  - Maximum reactivity peak at ~64 GWd/tU (less than **10%** of $[^{10}B]$ remains)
  - **COCONEUT**: Maximum reactivity discrepancy is found when $^{10}B$ is half consumed (**+280 pcm**)
  - Bias **relatively constant** between **MCNP and Serpent** even after **60 GWd/tU**
  - Discrepancies become visible after 60 GWd/tU between TRIPOLI4® and other MC codes
Results and analysis: **Standard FA (3/3)**

- **Whole concentrations comparison: MCNP as reference**

  - $[^{135}\text{Xe}]:$ Less than 2.5 % (all codes)
  - $[^{10}\text{B}]:$ at 60GWD/tU (10% $^{10}\text{B}$ remains)
    - MC codes: until 5.0 %
    - COCONUE: until 5.5 %
  - $[^{235}\text{U}]:$ at the end of depletion
    - Serpent: 0.5 %
    - TRIPOLI4®: 1.0 %
    - COCONUE: 1.5 %

  Good agreement between the codes
**Results and analysis: 2D full core (1/3)**

- **Benchmark considerations**
  - The same consistent parameters are taken into account for each code and simulation
    - Reflecting surface on Z axis
    - 50 burnup steps with a maximum value of 80 GWd/tU
    - Core power of: 50 MW<sub>th</sub>
    - Temperature: 300 K
    - JEFF-3.1.1 nuclear data library
Results and analysis: 2D full core (2/3)

- **Core calculation:** multiplication factor comparison

  ![Comparison of multiplication factor (K_{inf})](image)

- **MC codes**
  - Discrepancy between -110 pcm and +78 pcm
  - Simulation time: Serpent faster than TRIPOLI4®

- **COCONEUT vs mean of MC codes**
  - Fresh fuel: -235 pcm
  - Constant bias during the depletion: (between -395 pcm and -235 pcm)
  - 6 factor formula has to be calculated during the depletion to determine compensations

- **Reactivity comparison (mean value of MC codes as reference)**

  ![Comparison of reactivity (Δρ)](image)

  Slight discrepancy between the codes.

  Next step:
  3D core calculation and critical configurations research during the depletion.
Results and analysis: 2D full core (3/3)

Whole concentrations comparison

- $^{135}$Xe discrepancies
  - MC codes: less 0.6 %
  - COCONUT vs MC: Maximum of 1.6 %

- $^{149}$Sm
  - MC codes: less 0.5 %
  - COCONUT vs MC: 4.0 % during the first steps

- $^{10}$B
  - MC codes: Maximum of 2.5 %
  - COCONUT vs MC: Close to 4.5 %

Good agreement between the codes
Contents

① Depletion calculation methods

② Codes used in this study

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Conclusions

- **Good agreements** between MC codes for 2D assembly and 2D full core
  - Serpent faster than other MC codes
  - Quantify the differences between normalization methods

- **Small discrepancy** between COCONEUT and MC codes
  - Constant bias around -300 pcm during the entire depletion for 2D full core

- Dummy core is well suitable for core calculation studies and gives a better understanding of design purpose
Outlooks

- **Optimize MC coupled depletion codes**
  - Adapt time mesh with the flux gradient
  - Test of refueling algorithm proposed by Serpent and TRIPOLI4®
  - Changing depletion mesh step by step during the depletion (3D calculation)
  - Perform uncertainties propagation (compositions / flux)
  - Comparison with experimental core data

- **COCONEUT**
  - Estimate compensations with 6 factors formula
  - 281 groups calculation
  - Perform self shielding during the depletion

- **Dummy core**: future works on new methods for neutron propagation from core to ex-core system
Depletion calculation codes and benchmark considerations (1/2)

- **Neutronic analysis:**
  - Rise to equilibrium / Material balance
  - Flux / power distribution, Absorbers worth

- **Mainly used for export fuel assembly burnup compositions to MC codes**

- **Principal model consideration**
  1) **XS calculation**
     - XS collapsing: MOC calculation 281 → 26 groups
     - Self-shielding performed at the initial step
     - AFA is treated as a supercritical pattern representative of neutronic spectrum in the AFA.
  2) **Core calculation**
     - Transport theory (26-group) - 2D exact → APOLLO2
     - Diffusion theory (4-group) - 3D model → CRONOS2

- **Currently undergoing a large V&V process**
  - Part of this process: estimate the impact of main assumptions on depleted composition with MC codes
Results and analysis: Standard FA (3/4)

- COCONEUT: discrepancy analysis

**Relative concentration comparison (%) between COCONEUT and mean value of MC codes**

<table>
<thead>
<tr>
<th>Burnup (Gd/tU)</th>
<th>20</th>
<th>40</th>
<th>60</th>
<th>80</th>
<th>100</th>
</tr>
</thead>
<tbody>
<tr>
<td>Δ[^235]U</td>
<td>-0.12</td>
<td>-0.27</td>
<td>-0.46</td>
<td>-0.69</td>
<td>-1.03</td>
</tr>
<tr>
<td>Δ[^239]Pu</td>
<td>2.42</td>
<td>1.89</td>
<td>1.33</td>
<td>0.82</td>
<td>0.21</td>
</tr>
<tr>
<td>Δ[^148]Nd</td>
<td>1.95</td>
<td>2.07</td>
<td>2.12</td>
<td>2.14</td>
<td>2.15</td>
</tr>
<tr>
<td>Δ[^149]Sm</td>
<td>1.36</td>
<td>1.13</td>
<td>1.59</td>
<td>2.03</td>
<td>2.48</td>
</tr>
</tbody>
</table>

- Main concentrations
  - Less than 3% discrepancy
  - Constant bias on ^148Nd (burnup indicator)

- [^10B]
  - Burned faster with TRIPOLI4® and COCONEUT depletion chain / power normalization?

- COCONEUT Outlook
  - Depletion with 281 group
  - Self shielding during depletion (several steps)