Micro-depletion and decay heat calculations with Serpent-DYN3D

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Outline

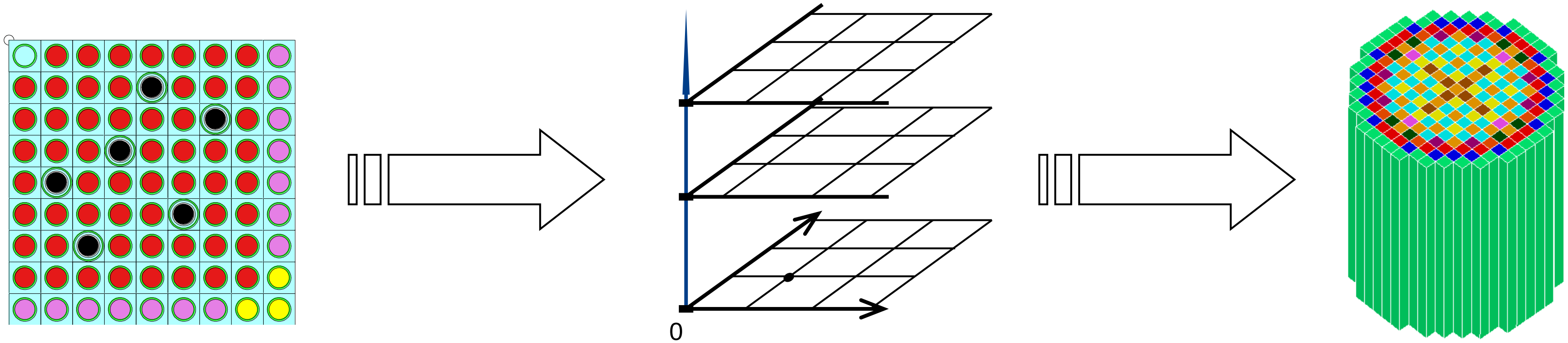
- Few-group cross sections and spectral history effects
- Micro-depletion model in DYN3D
- Test problem

Two-step reactor analyses procedure

Lattice transport
cont. or multi-group
fine-mesh

Parametrized
few-group
XS Library

Full core diffusion
few-group
coarse mesh



- XS parametrized as a function of state parameters:
  - **B** - Burnup
  - $\boldsymbol{\rho_m}$ - Moderator density
  - $\boldsymbol{T_f}$ - Fuel temperature
  - $\boldsymbol{C_B}$ - Boron concentration
  - …

Lattice calculations

1. Depletion:
   - Nuclides content vs. burnup
   - Fixed ρ_m T_f C_B

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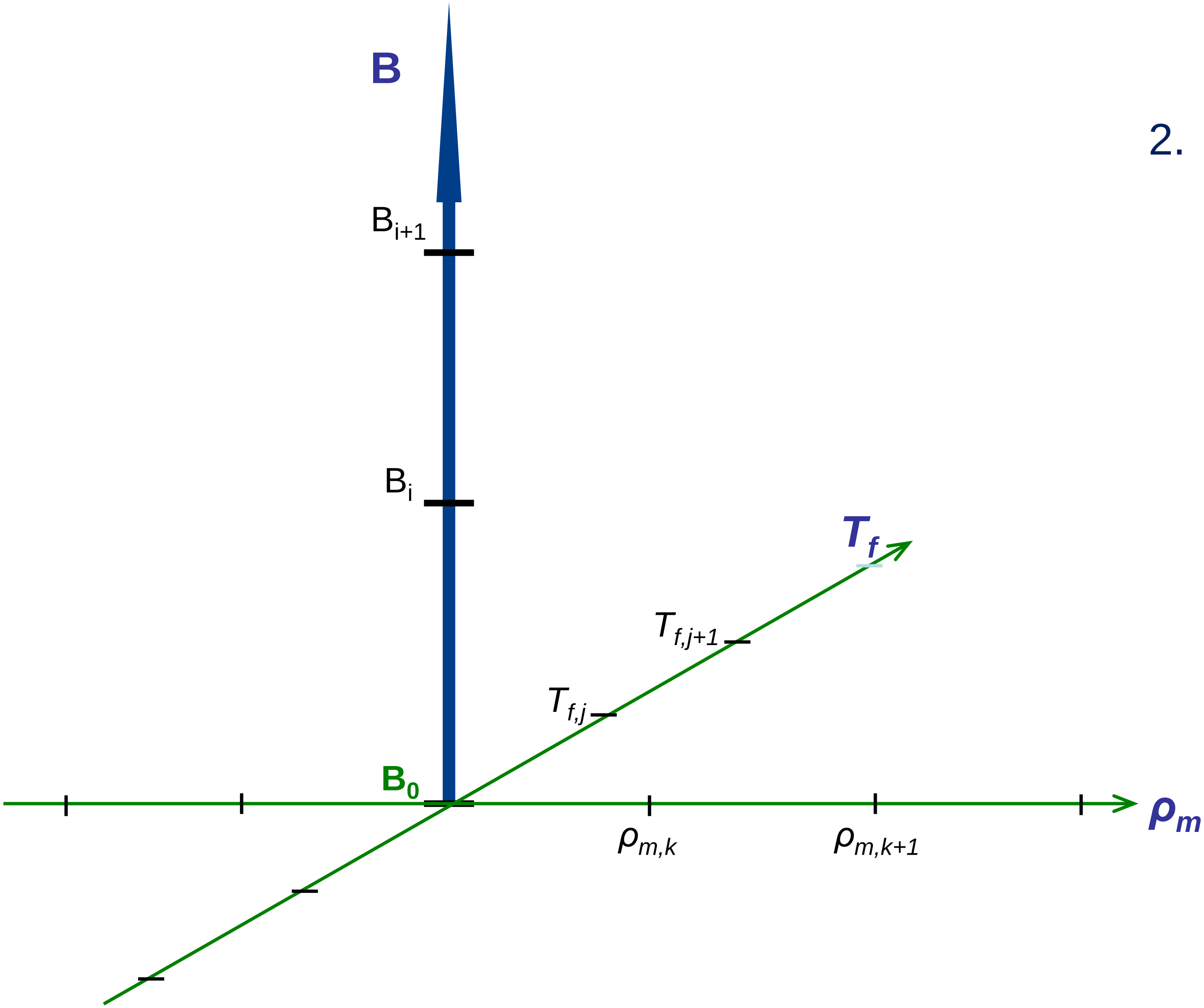
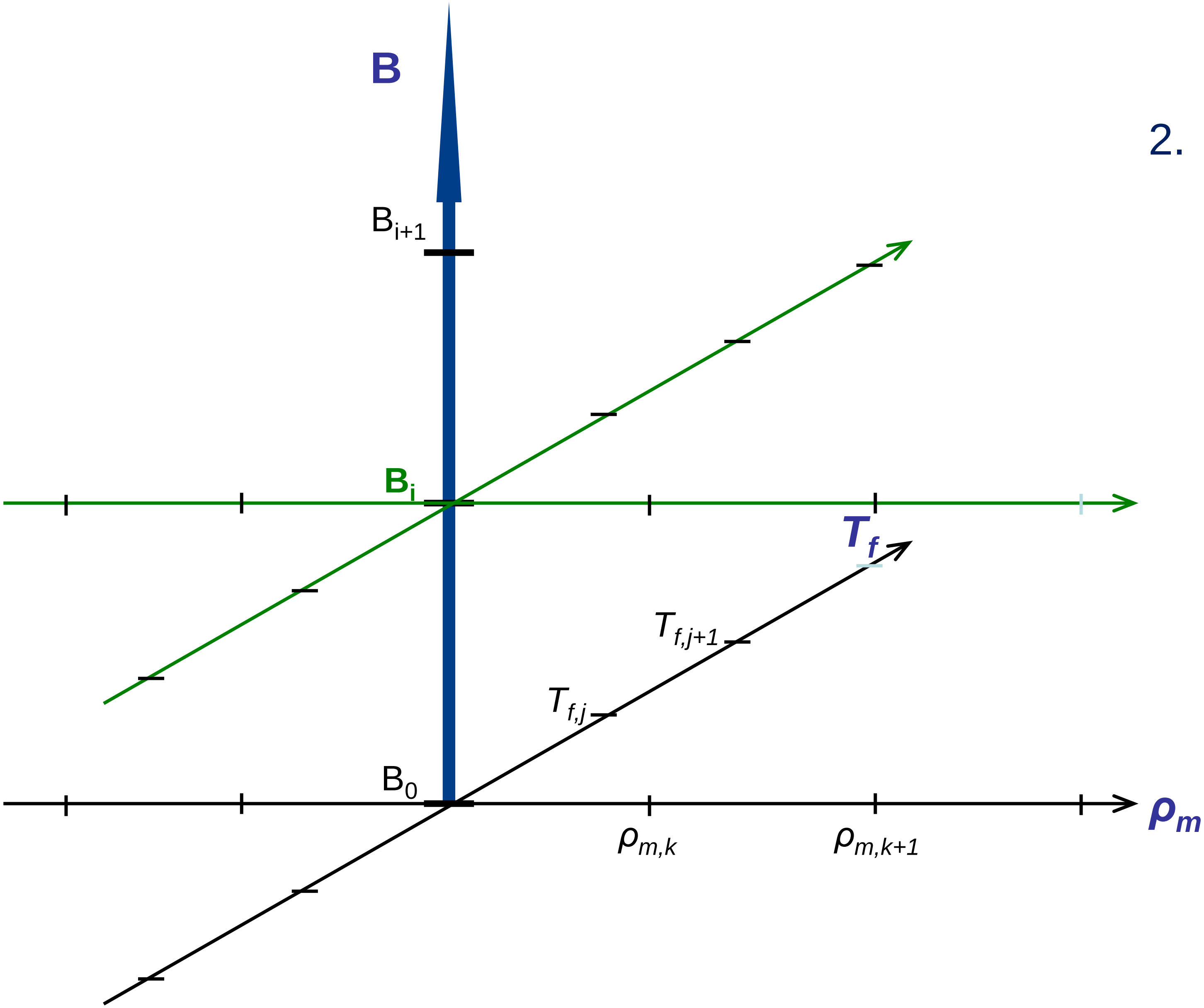
Lattice calculations



1. Depletion:
   - Nuclides content vs. burnup
   - Fixed ρ_m T_f C_B
2. Branching:
   - Homogenized XS $\Sigma(B_0 \quad \rho_{m,k}\ T_{f,j}\ C_{B,n})$

Lattice calculations



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   - Fixed $\rho_m\ T_f\ C_B$
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$\Sigma(\boldsymbol{B_0}\quad \rho_{m,k}\ T_{f,j}\ C_{B,n})$

$\Sigma(\boldsymbol{B_i}\quad \rho_{m,k}\ T_{f,j}\ C_{B,n})$

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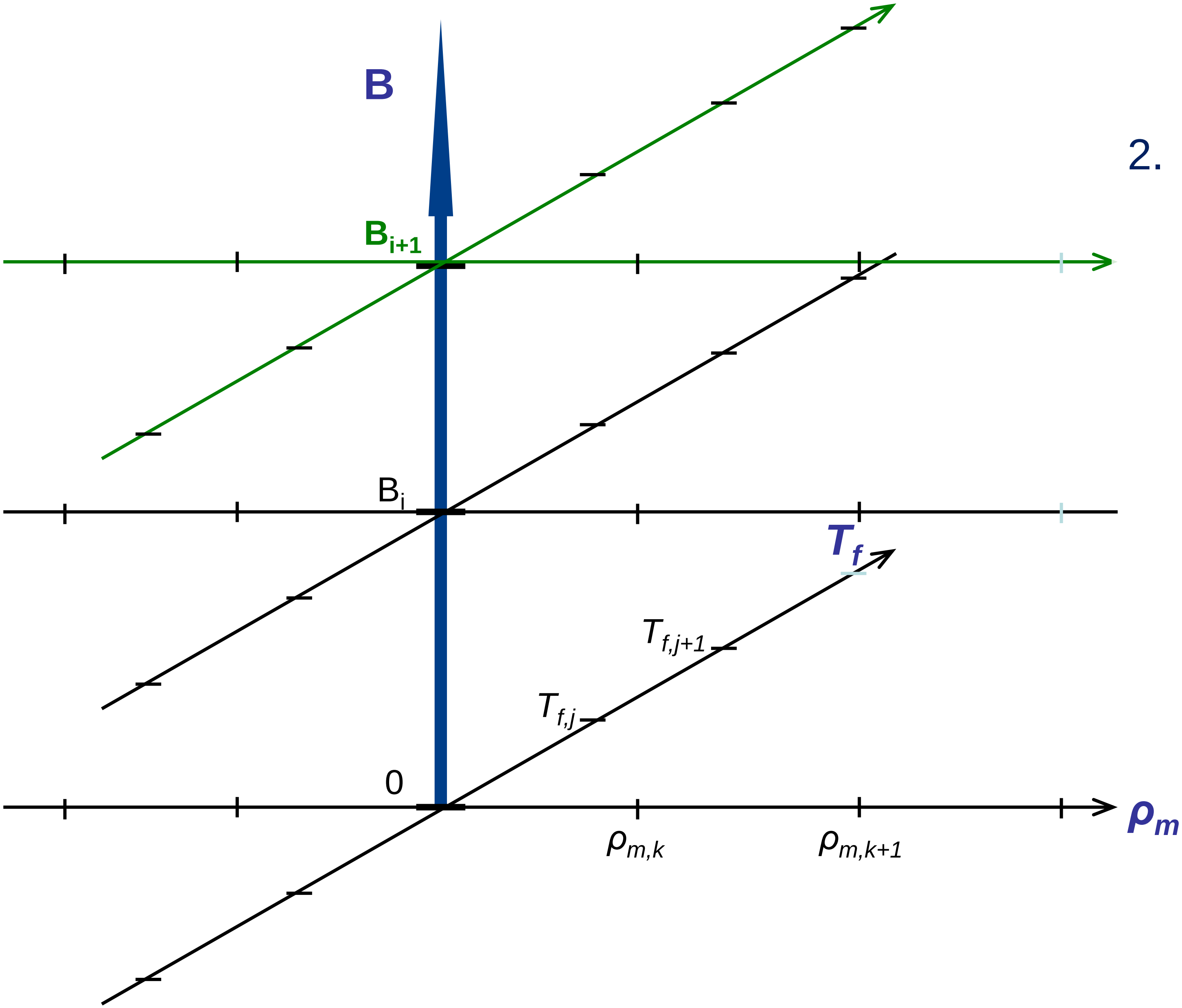
Lattice calculations



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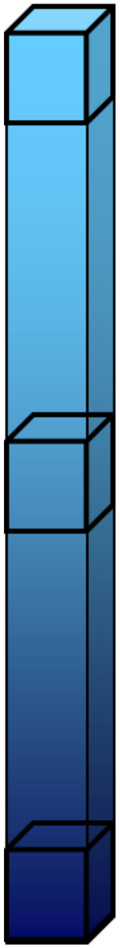
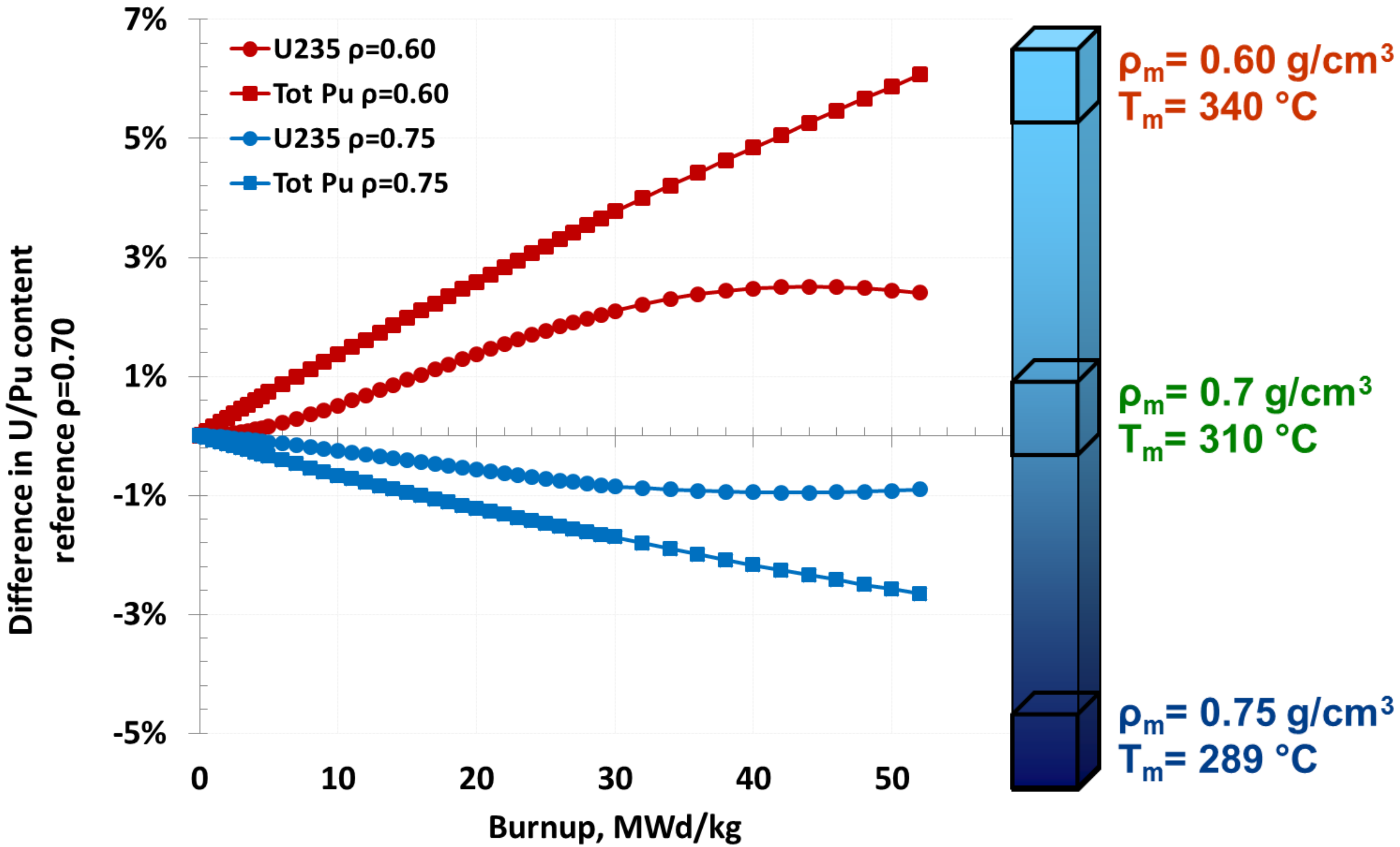
$\Sigma(B_0\ \ \rho_{m,k}\ T_{f,j}\ C_{B,n})$
$\Sigma(B_i\ \ \rho_{m,k}\ T_{f,j}\ C_{B,n})$
$\Sigma(B_{i+1}\ \rho_{m,k}\ T_{f,j}\ C_{B,n})$

Spectral history effect

- Core-averaged conditions are used for depletion
- Local conditions may differ → error accumulation:
  - nuclide content
  - few-group XS

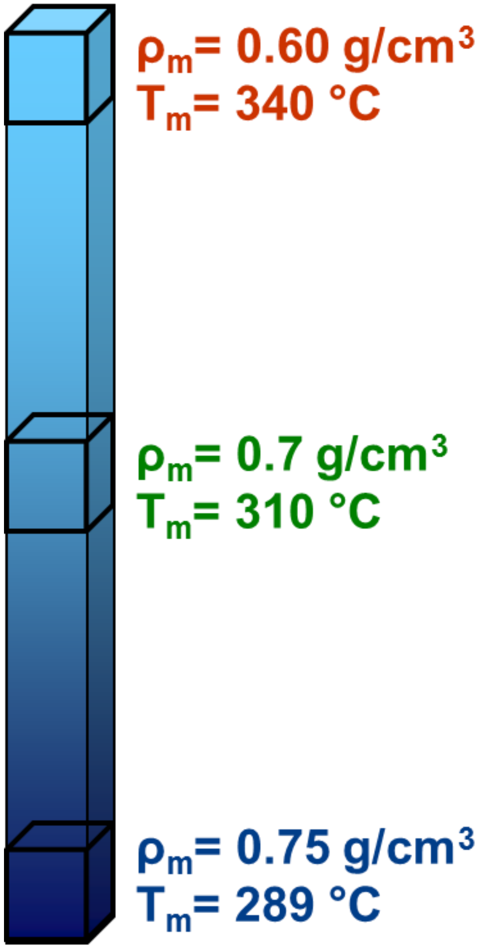


Spectral history effect



Spectral history effect

- Core-averaged conditions are used for depletion
- Local conditions may differ → error accumulation:
  - nuclide content
  - few-group XS
- Referred to as spectral history effects
- Few-group XS should be corrected



Accounting for spectral history effects

- Various methods are in use
  - Exposure-weighting
  - Spectral indices
  - Micro-depletion

- In DYN3D: a new hybrid micro-depletion method

- Accounts for different spectral history effects:
  - moderator density, fuel temperature, boron concentration

- Works for different fuel types:
  - UOX, MOX, Gd as a burnable absorber

- … and different scenarios:
  - outage and power variation

DYN3D: hybrid micro-depletion method

- Hybrid micro-depletion:

$$\Sigma^{corrected} = \Sigma^{Ref} + \sum_{i}^{L} (\sigma_i^{corrected} N_i^{actual} - \sigma_i^{Ref} N_i^{Ref})$$

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- Correction of micro-XS and macro-XS (scattering, diffusion):

$$\sigma^{corrected} = \sigma^{Ref} \cdot \left[1 + k \left(\frac{N_{fissile}^{actual}}{N_{fissile}^{SA}} - 1 \right) \right]$$

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Calculated by DYN3D

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Pre-calculated → XS-library by Serpent

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- Derivation of correction factors → based on variation in fissile content

$$k = \delta\Sigma \Big/ \delta N_{fiss} = \frac{\Sigma^{off} - \Sigma^{nom}}{\Sigma^{nom}} \cdot \frac{\sqrt{N_{fiss}^{nom}}}{\sqrt{N_{fiss}^{off}} - \sqrt{N_{fiss}^{nom}}}$$

DYN3D: hybrid micro-depletion method

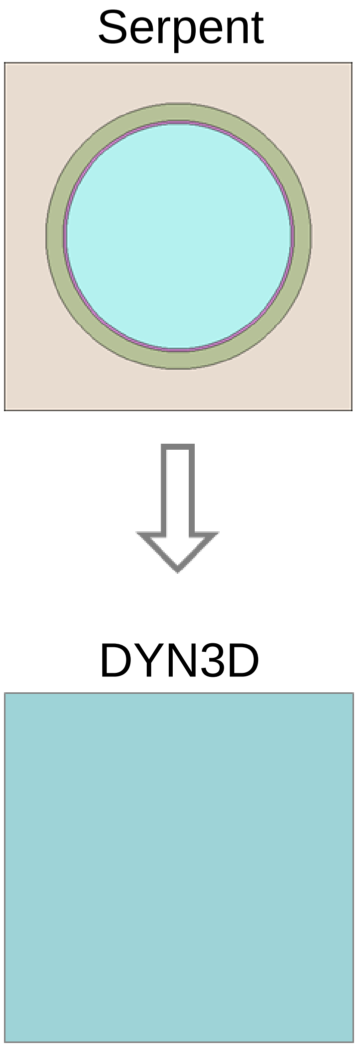
- New depletion solver in DYN3D
  - Transmutation matrix solved by CRAM
  - about 1100 nuclides (300 with cross section data)

- **Decay heat** can be calculated **as a bonus**!
  - Distribution in the core
  - Can be important for safety analyses (e.g. LOCA)

Test problem

Test problem

- 2D BWR fuel pin with 3% UO_2 fuel
- Two depletion histories for XS generation
  - ρ_m = 0.4 g/cm^3 and 0.7 g/cm^3
  - T_f = 900 K
- Serpent 2 with JEFF-3.1 is used for
  - reference solution
  - macro- and micro-XS for DYN3D
  - fission yields, branching ratios for n,γ reactions
- DYN3D solution for ρ_m = 0.5 g/cm^3
  - k-inf
  - decay heat



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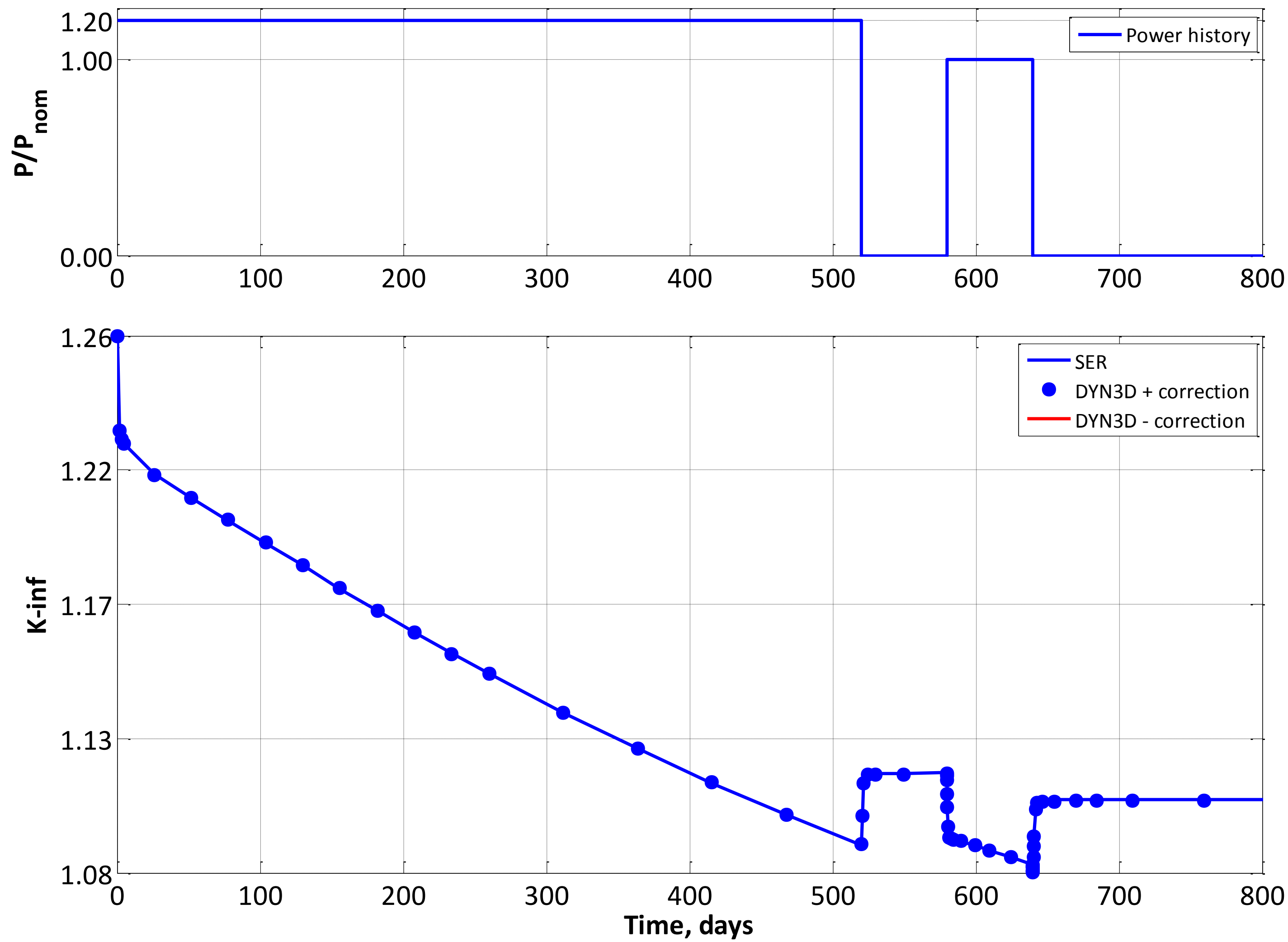
Results: k-inf



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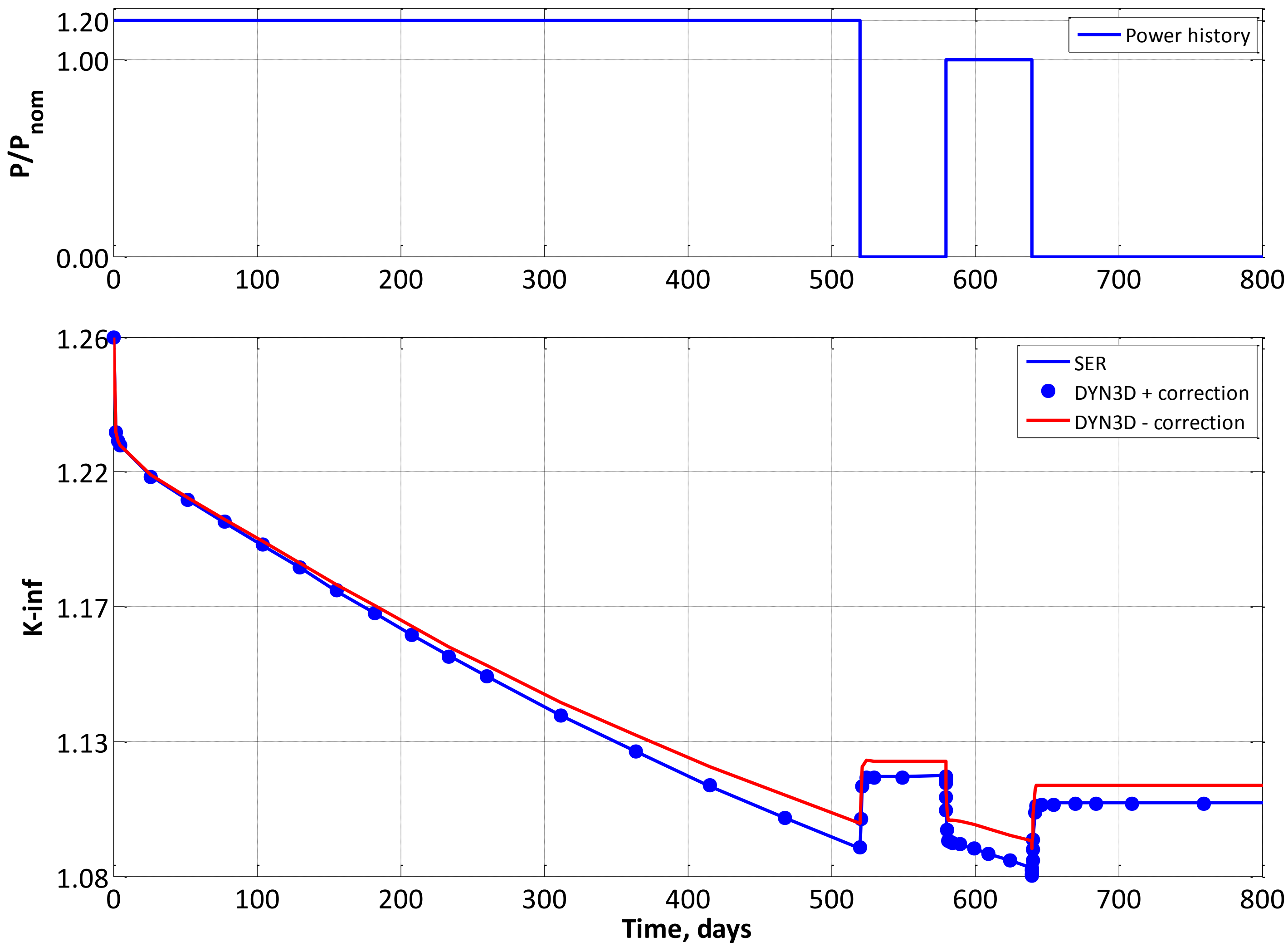

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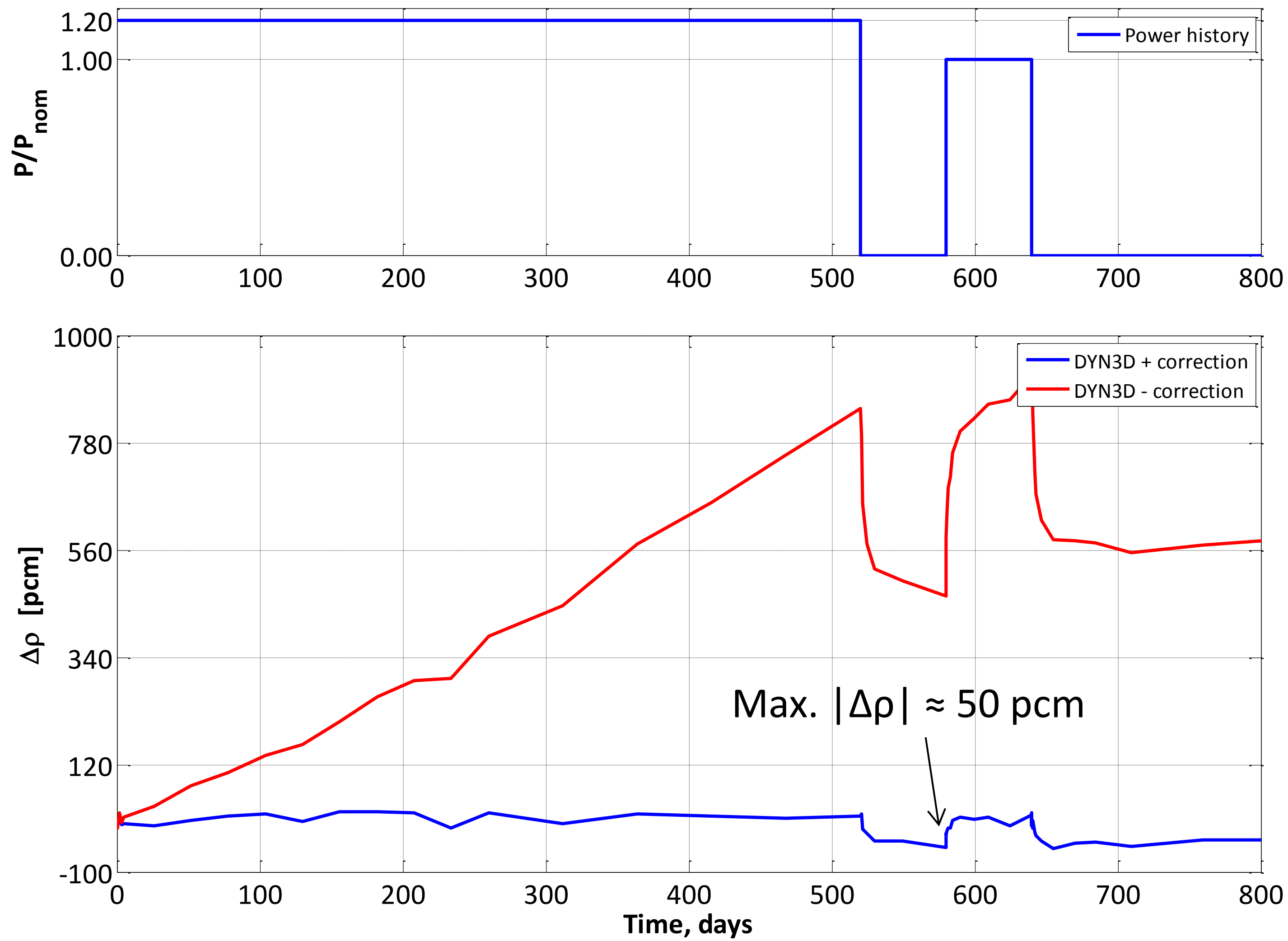
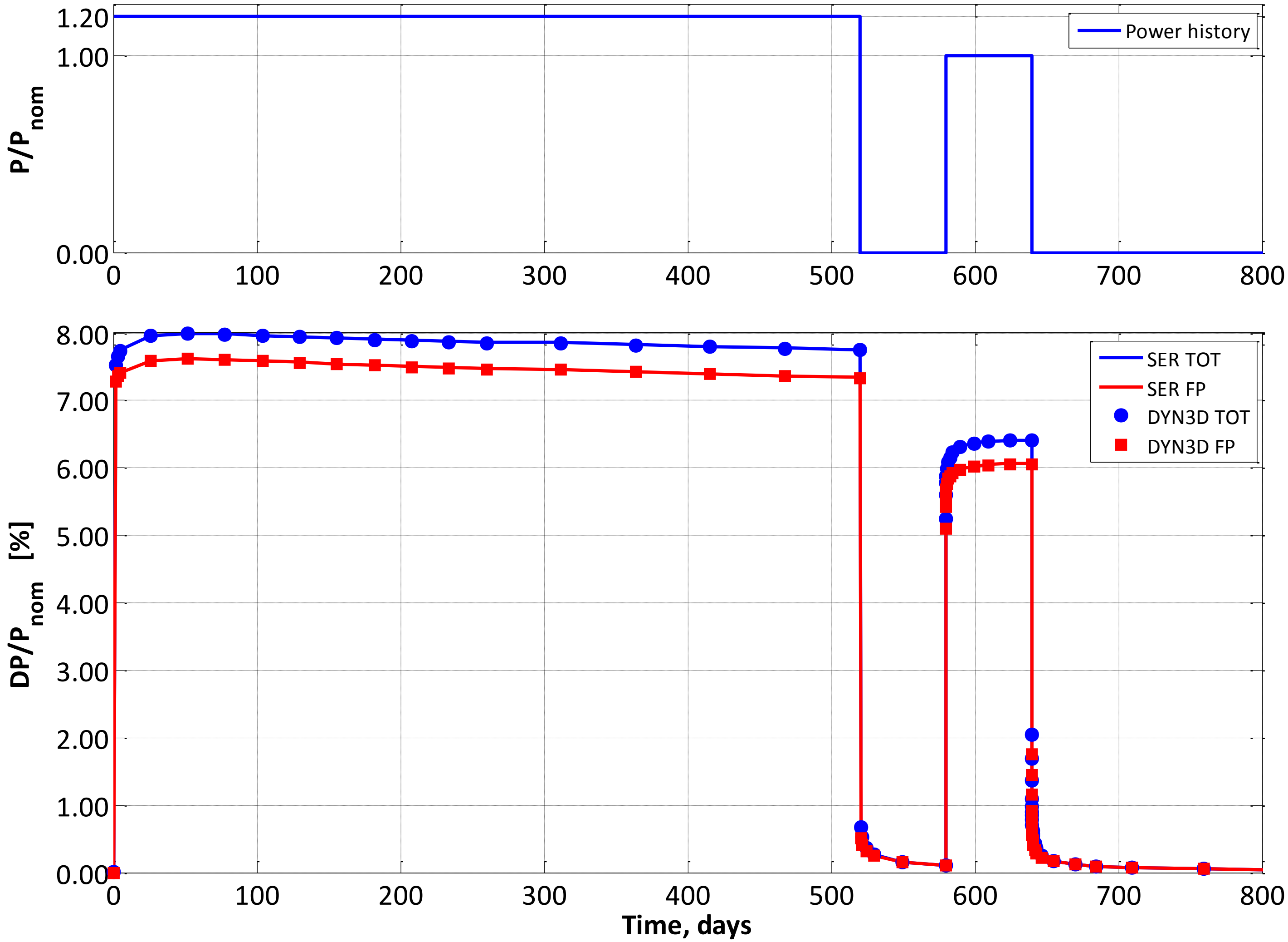


Results: decay power during operation



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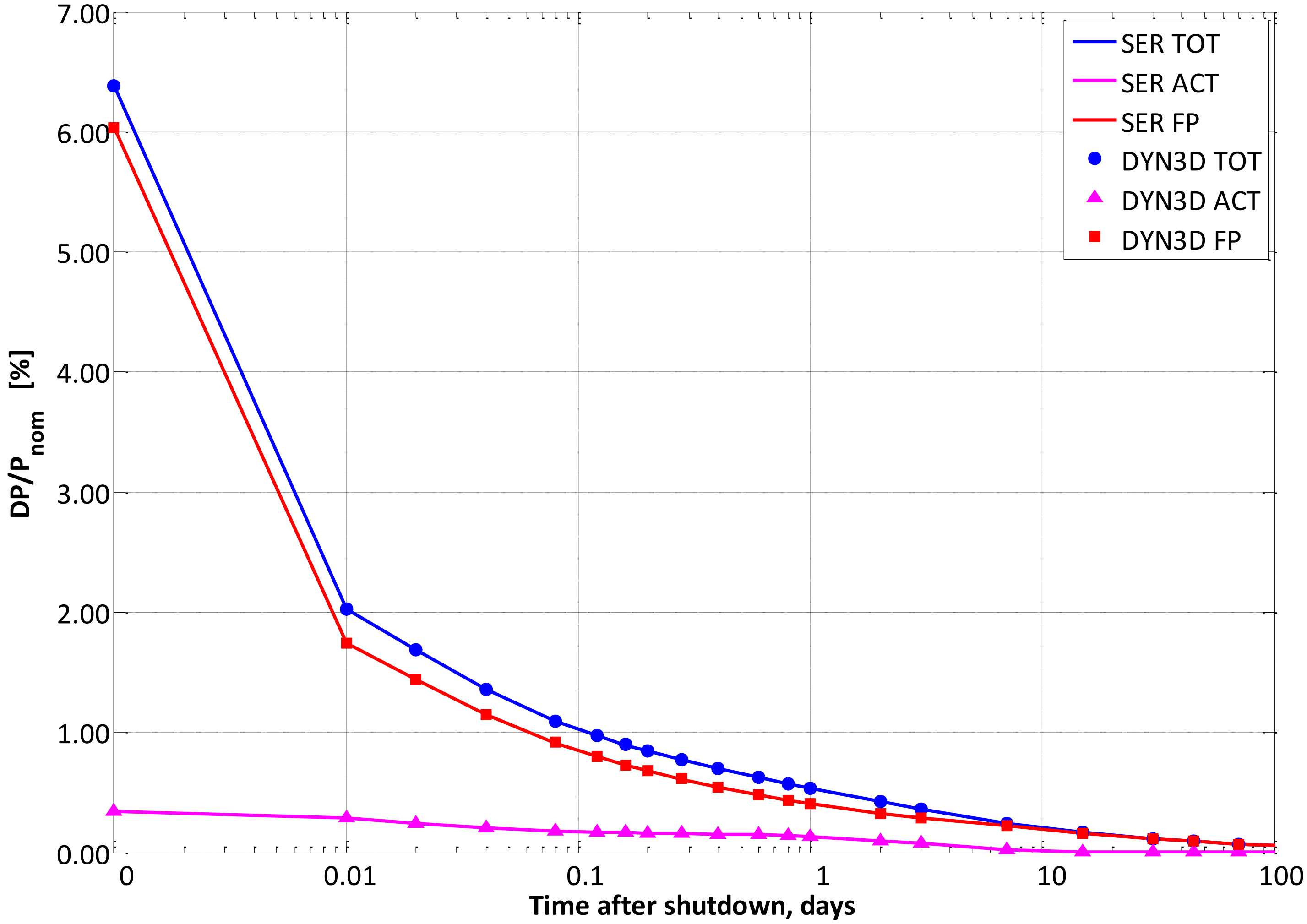
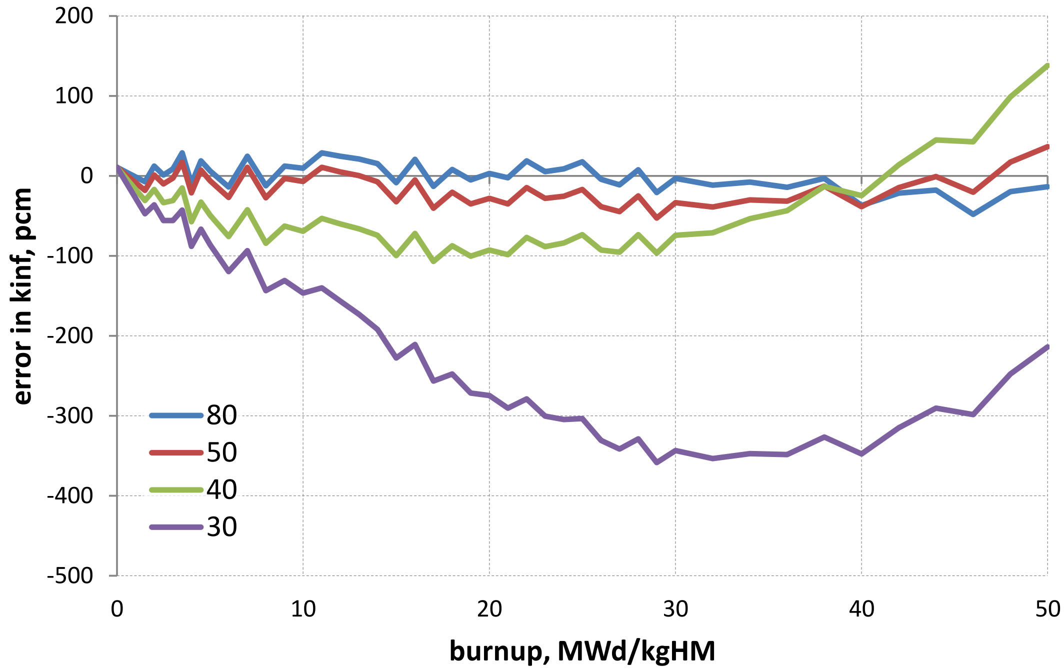
Results: decay power after shutdown



Micro-depletion: how many nuclides are needed?



- **Full nuclide** content for decay heat calculations
- But only **50-80 nuclides** for neutronics only
- Transmutation matrix can be optimized

Summary

- Test case demonstrate very good results
  - coolant density history
  - outage
  - decay heat

- Future work
  - transmutation matrix optimization
  - test on realistic case
  - parallelization of micro-depletion calculations

- Issues with micro-XS from Serpent:
  - detectors are slow
  - homogenization of micro-XS
  - output of fission yields

Thank you!