Hybrid Micro-Depletion model in DYN3D

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Content

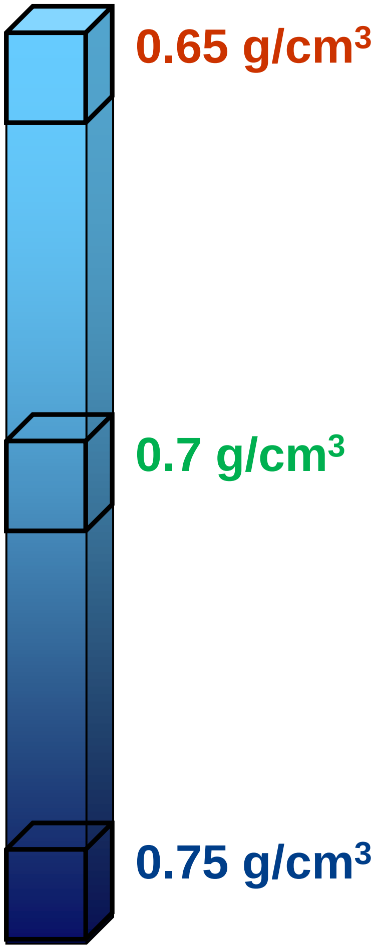
- Reactor dynamic code DYN3D
- Spectral history effects
- Pu-correction method
- Micro-depletion method
- Hybrid micro-depletion
- Verification
- Conclusions

Reactor dynamic code DYN3D

- 3D multi-group diffusion or SP_3, NEM
- 1D two-phase flow in parallel channels
- Steady-state and burnup analyses
- Transient analyses
  - startup, SCRAM, RIA, Xe dynamics, burnup
- Coupling with:
  - system codes
  - sub-channel TH
  - CFD
  - fuel performance
- LWR + HTGR and SFR

Spectral history effect

- Homogenized cross sections (XS) are generated in single assembly (SA) by lattice codes
- Core-averaged conditions are used for SA depletion
- Depletion conditions for each node differ from SA
- That leads to differences in
  - nuclide content
  - XS
- Nodal XS should be corrected



Pu-history correction

$$\Sigma = \Sigma^{SA} \cdot \left[1 + k \left(\sqrt{\frac{N_{Pu}^{actual}}{N_{Pu}^{SA}}} - 1 \right) \right]$$

Σ^{SA} – diffusion parameter (XS), calculated by lattice code;

N_{Pu}^{SA} – concentration of Pu239 in the nominal depletion;

k – proportionality coefficient, defined for each XS type;

N_{Pu}^{actual} – actual local concentration of Pu239.

$$k = {}^{\delta\Sigma}\!/\!_{\delta Pu} = \frac{\Sigma^{off} - \Sigma^{nom}}{\Sigma^{nom}} \cdot \frac{\sqrt{N_{Pu}^{nom}}}{\sqrt{N_{Pu}^{off}} - \sqrt{N_{Pu}^{nom}}}$$

* Bilodid, I., Mittag, S., 2010. Use of the local Pu-239 concentration as an indicator of burnup spectral history in DYN3D. Ann. Nucl. Energy 37, 1208–1213.

Pu-history correction

Calculated by DYN3D

$$\Sigma = \Sigma^{SA} \cdot \left[1 + k \left(\sqrt{\frac{N_{Pu}^{actual}}{N_{Pu}^{SA}}} - 1 \right) \right]$$

Pre-calculated by Serpent
↓
XS-library

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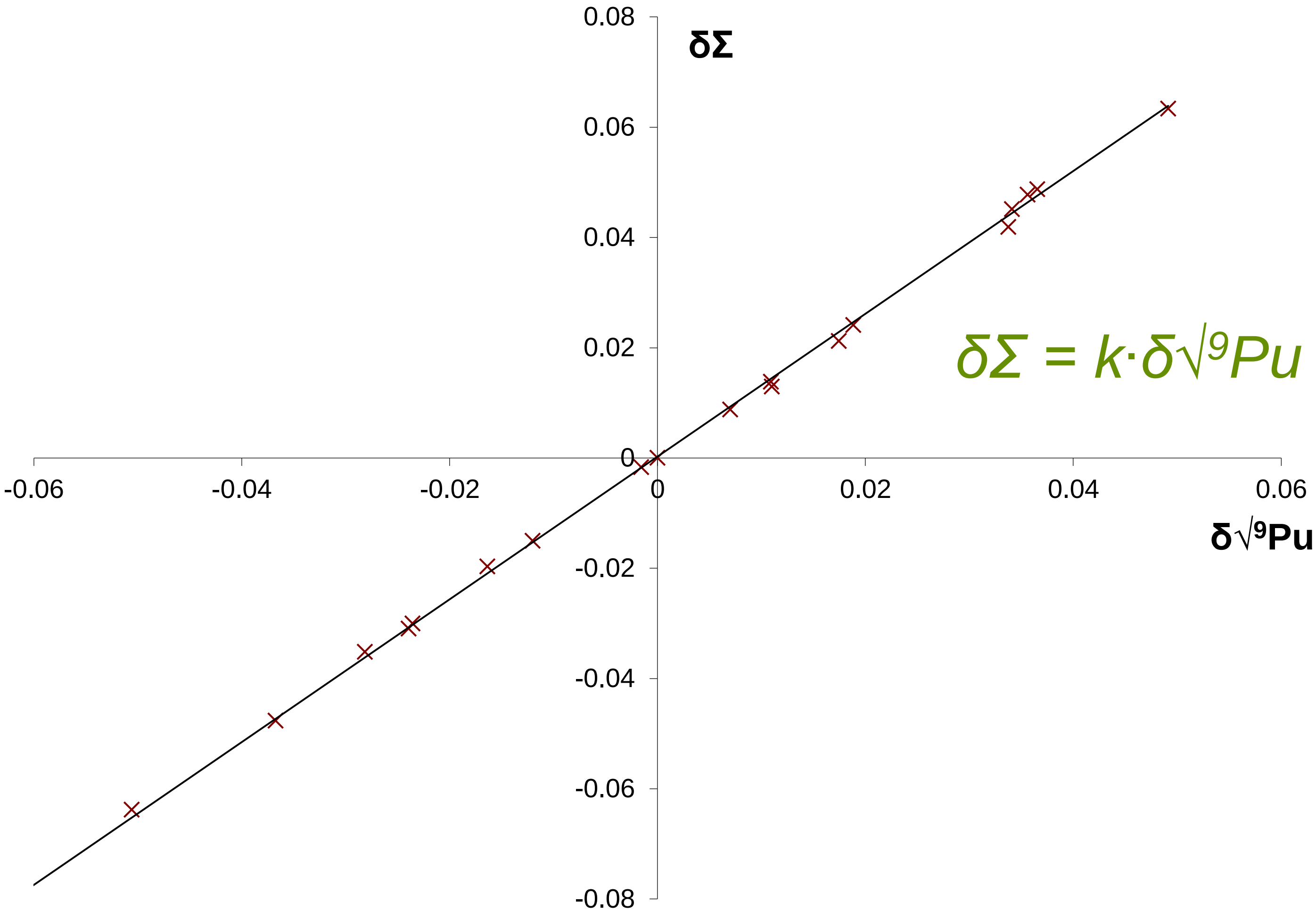
Method idea



- The linear proportionality between deviation of XS and deviation of $\sqrt{{}^9Pu}$ concentration can be assumed

$\delta\sqrt{{}^9Pu} = (\sqrt{{}^9Pu} - \sqrt{{}^9Pu}_{nom}) / \sqrt{{}^9Pu}_{nom}$

Pu-history correction

$$\Sigma = \Sigma^{SA} \cdot \left[1 + k \left(\sqrt{\frac{N_{Pu}^{actual}}{N_{Pu}^{SA}}} - 1 \right) \right]$$

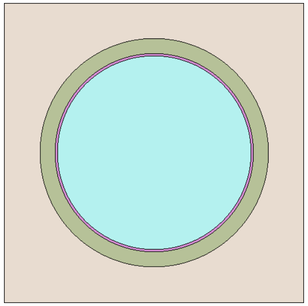
- Different history effects:
  - moderator density
  - fuel temperature
  - boron acid concentration

- Different fuel types:
  - WWER, PWR and BWR
  - UOX and MOX
  - Gd – burnable absorber

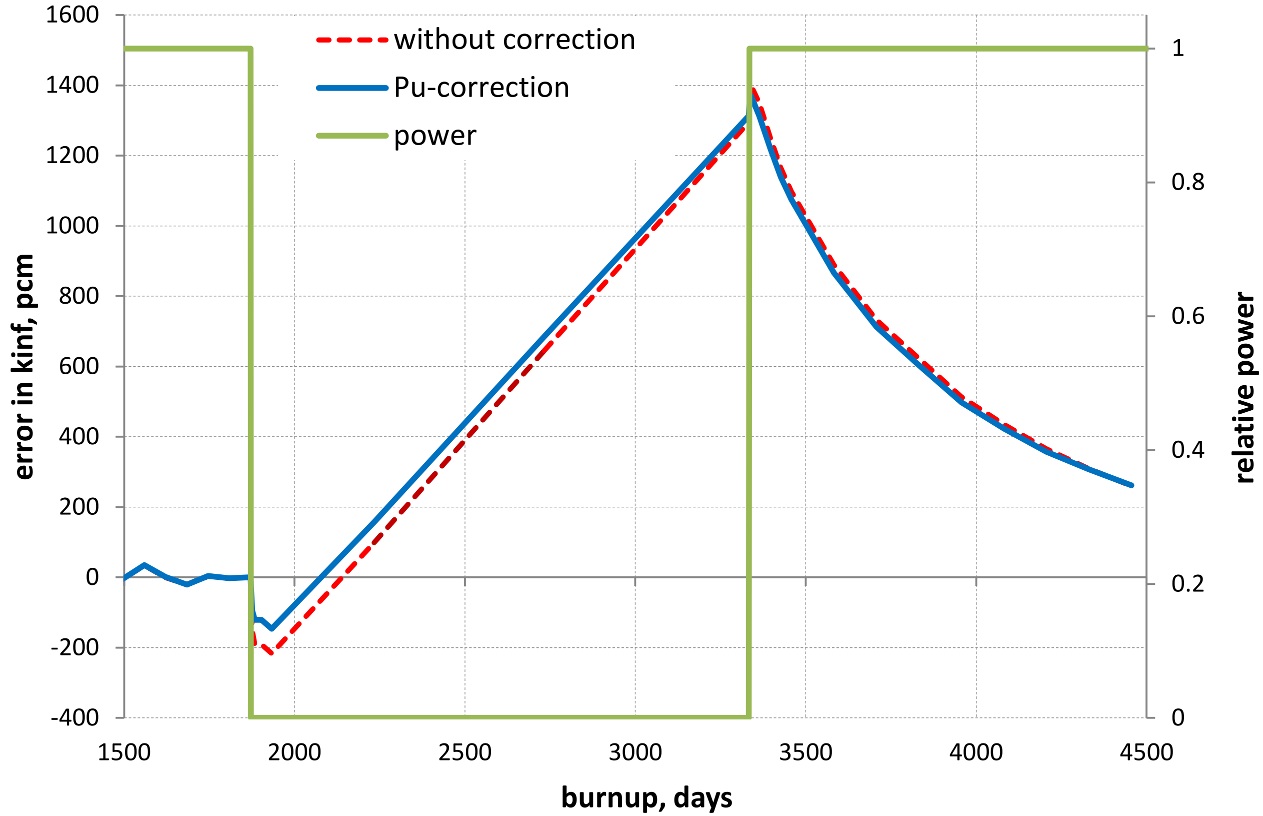
- Fails to reproduce outage and power variation

Test model

- BWR fuel pin:
  - 2D with reflective boundaries
- Nominal conditions for SA depletion:
  - water density 400 kg/m^3
  - fuel temperature 900 K
- Serpent 2 with JEFF-3.1 is used for
  - reference solution
  - macro- and micro-XS for DYN3D



4-years outage simulation



- burnup until 30 MWd/kgHM; 4 years outage; burnup up to 50 MWd/kgHM
- Shutdown cooling reactivity (except Xe and Sm)
  - Pu241 → Am241
  - Eu155 → Gd155

Micro-depletion method

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

$\boldsymbol{\Sigma^{SA}}$ – diffusion parameter (XS) from lattice code

$\boldsymbol{N_i^{SA}}$ – concentration of nuclide $\boldsymbol{i}$ in the SA depletion

$\boldsymbol{N_i^{actual}}$ – actual local concentration of nuclide $\boldsymbol{i}$

$\boldsymbol{\sigma_i}$ – micro-XS (absorption or fission)

$\boldsymbol{i}$ – nuclide index

$\boldsymbol{L}$ – total number of nuclides

Micro-depletion method

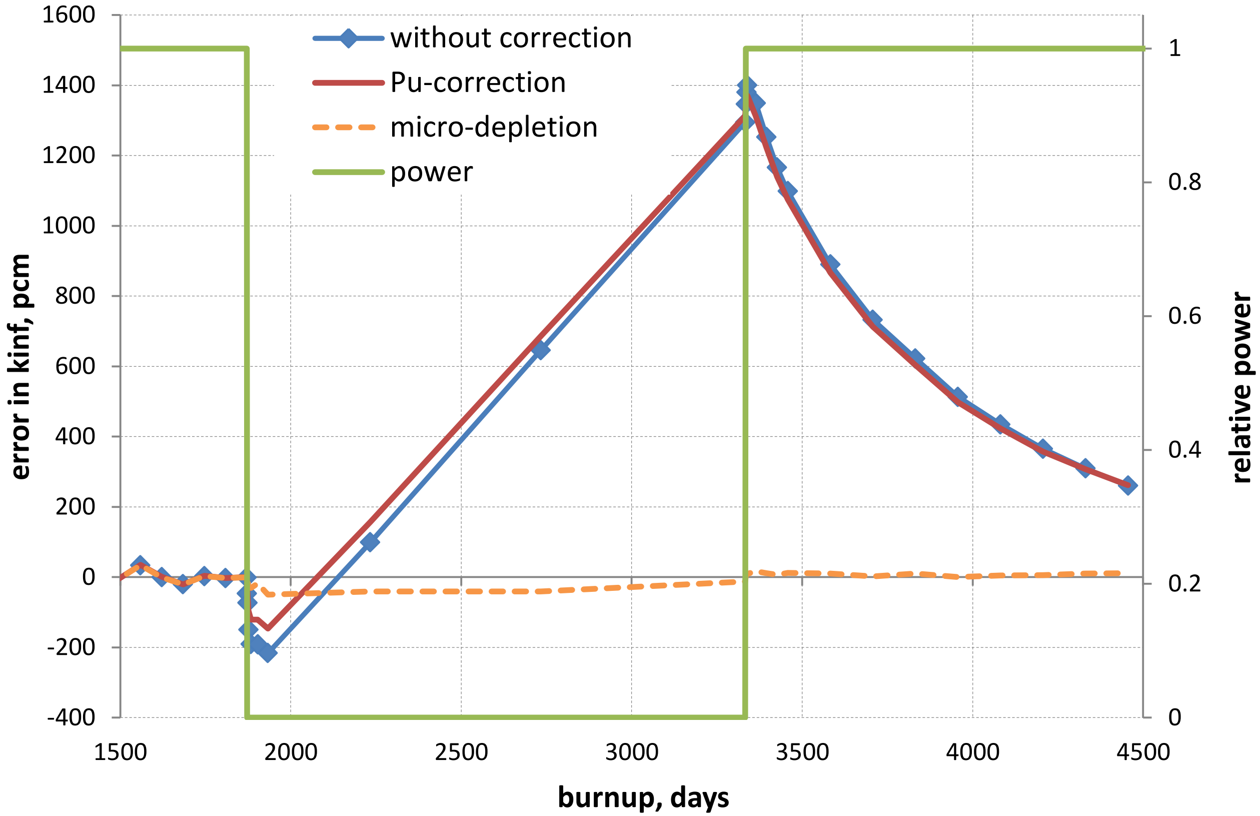
$$\Sigma = \Sigma^{SA} + \sum_{i}^{L} \sigma_i (N_i^{actual} - N_i^{SA})$$

- In DYN3D – only for Xe and Sm
- In some other codes (SIMULATE, ANC, etc.) up to 50 nuclides
  - simplified and linearized chains
- New depletion solver in DYN3D
  - transmutation matrix solved by CRAM
  - full nuclide content (~1100 nuclides)

$$\mathbf{N}(t) = \exp\left[\mathbf{M}(\phi, T)(t - t_0)\right] \times \mathbf{N}(t_0)$$

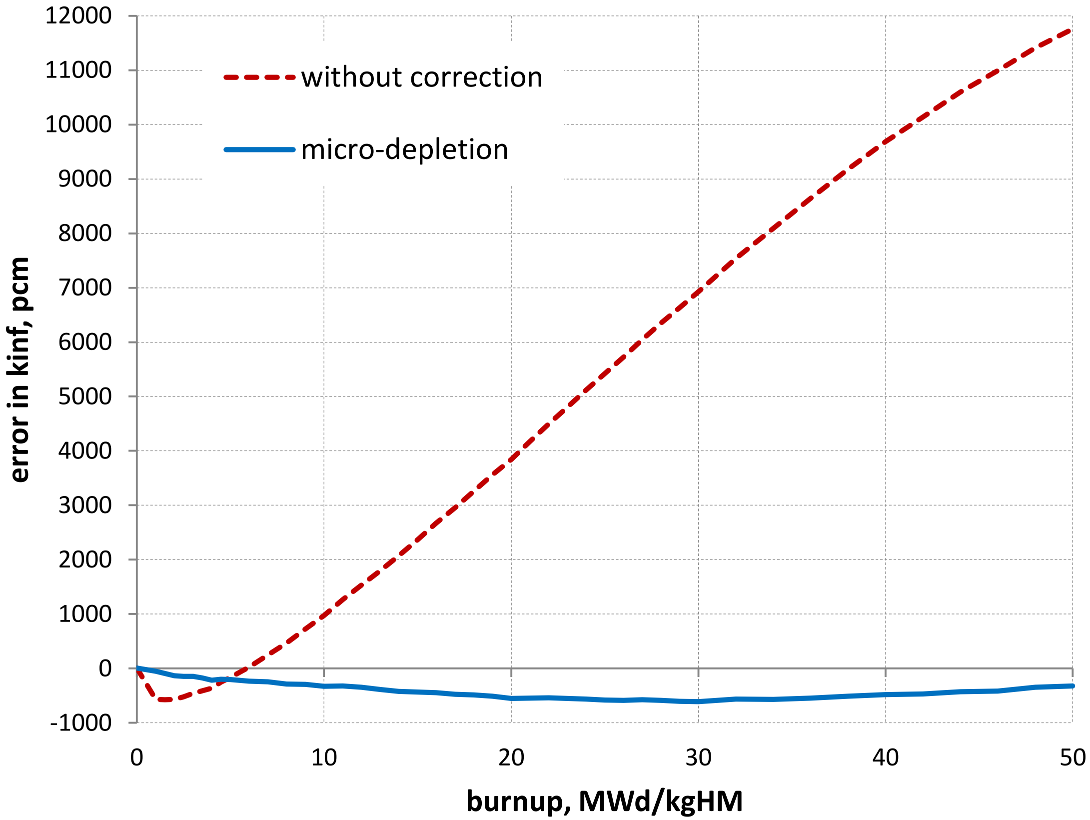
- Micro-depletion correction in DYN3D
  - about 300 nuclides have cross sections

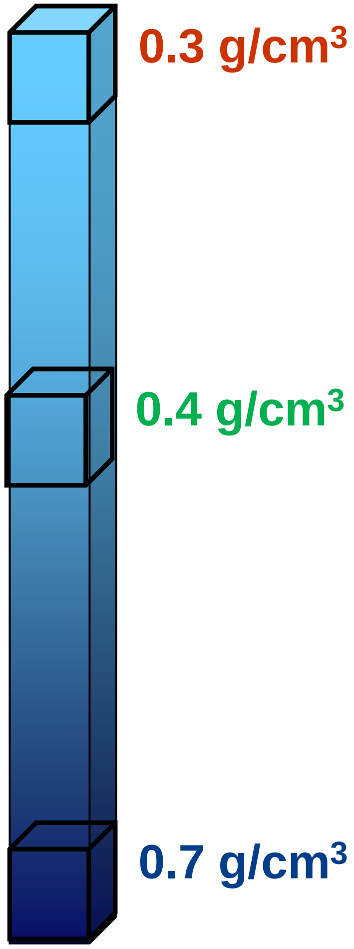
4-years outage simulation



- Micro-depletion reduces error to <50 pcm

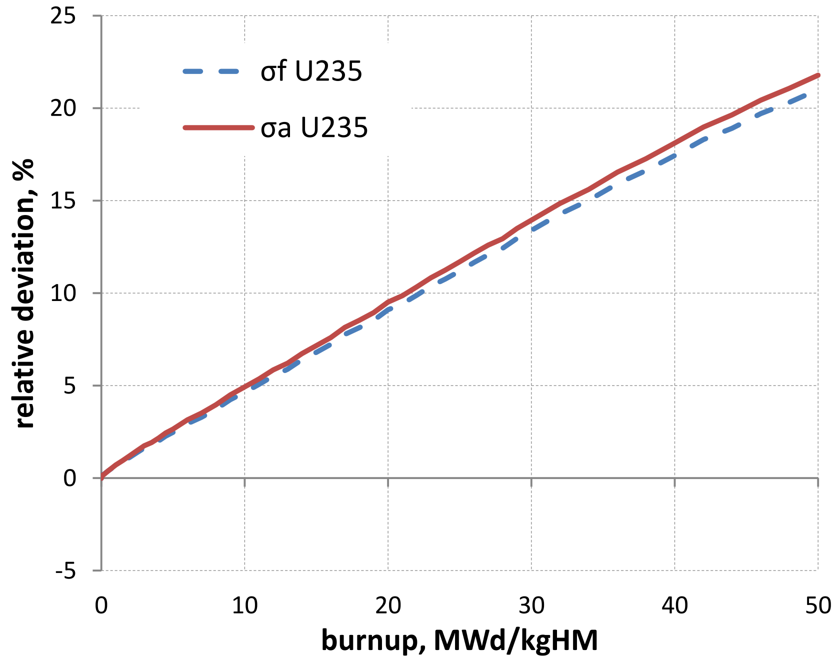
Moderator density effect

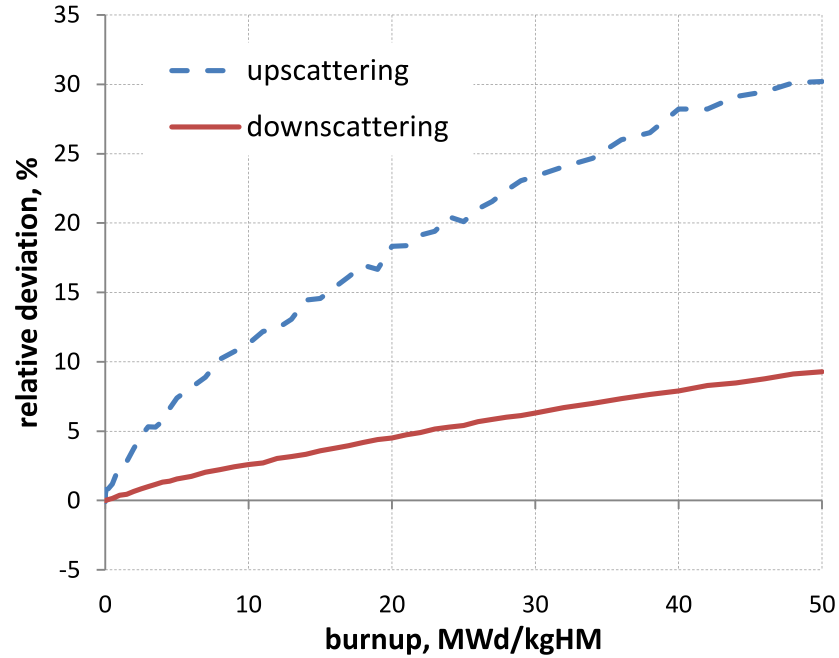




- Without correction error is >11000 pcm
- Micro-depletion reduces error to ~600 pcm

Scattering matrix and microscopic cross sections





- Only absorption and fission XS are corrected by micro-depletion
- Scattering matrix and micro-XS should be corrected too

Hybrid micro-depletion method

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

- Correction of micro-XS
- Correction of macro-XS (scattering, diffusion, etc.)
- Sum of fissile nuclides
    - U233 + U235 + Pu239 + Pu241

- Hybrid micro-depletion:

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

Hybrid micro-depletion method

Calculated by DYN3D

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

Pre-calculated by Serpent
↓
XS-library

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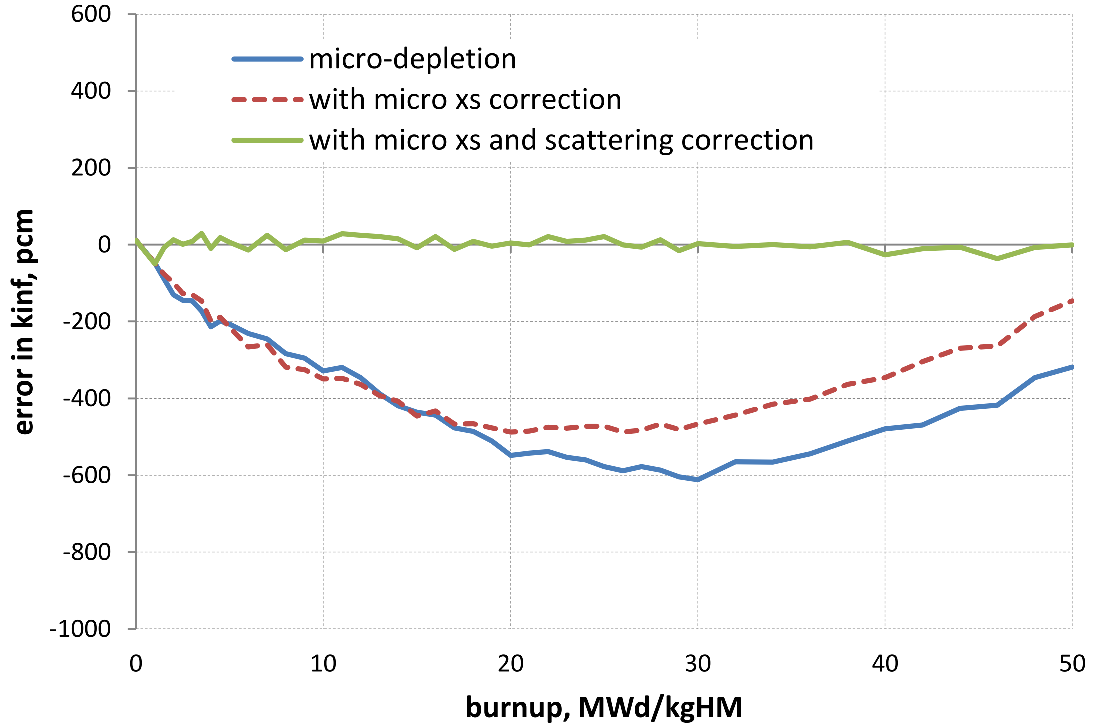
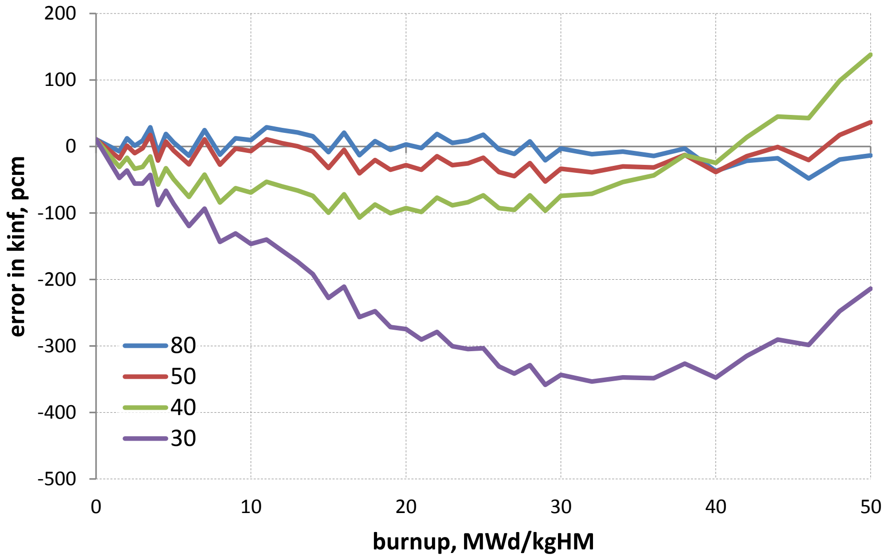
Hybrid micro-depletion method



- Correction of both micro-XS and scattering matrix
- Hybrid micro-depletion reduces error <50 pcm

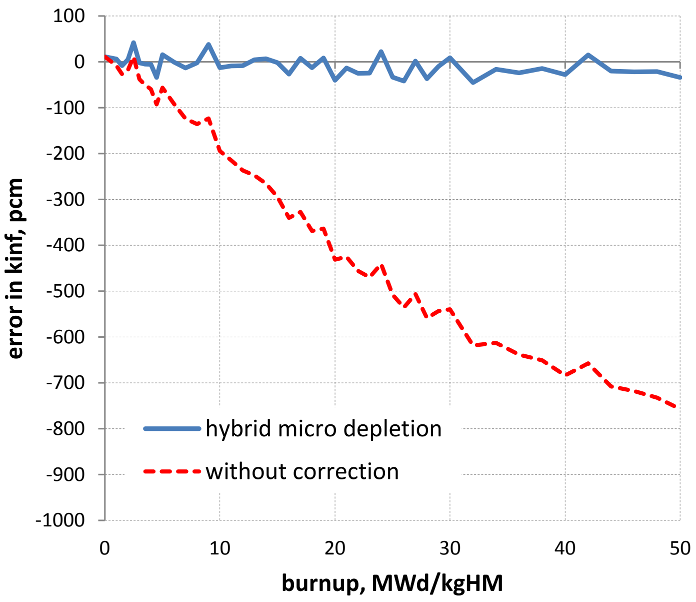
How many nuclides?

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

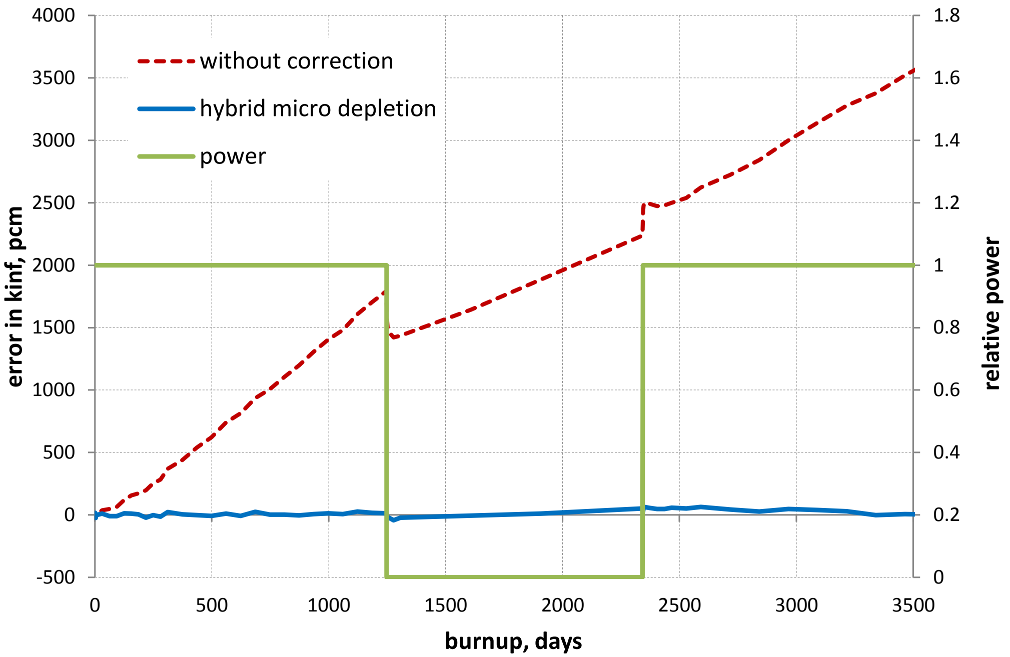


- Nuclides are sorted by their contribution
- *L* is reduced consequently
- Only 50-80 nuclides actually influence results
- Transmutation matrix could be optimized (~300 nuclides)

Hybrid micro-depletion method – BWR UOX

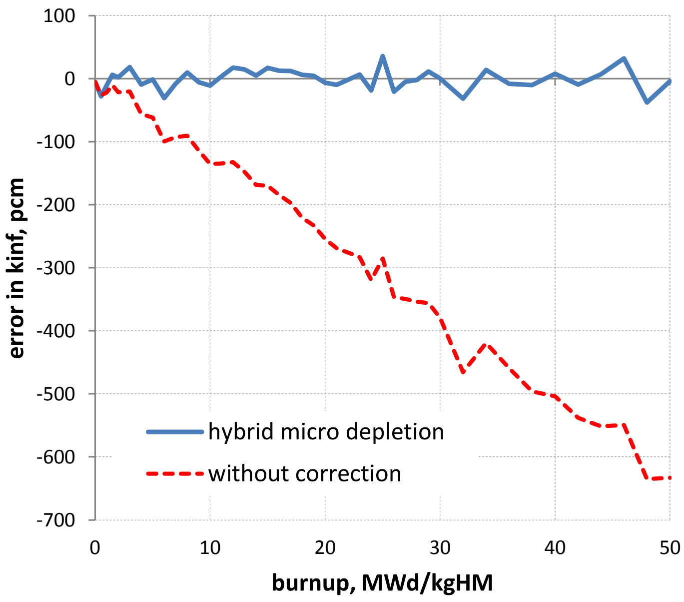


fuel temperature 1200 K

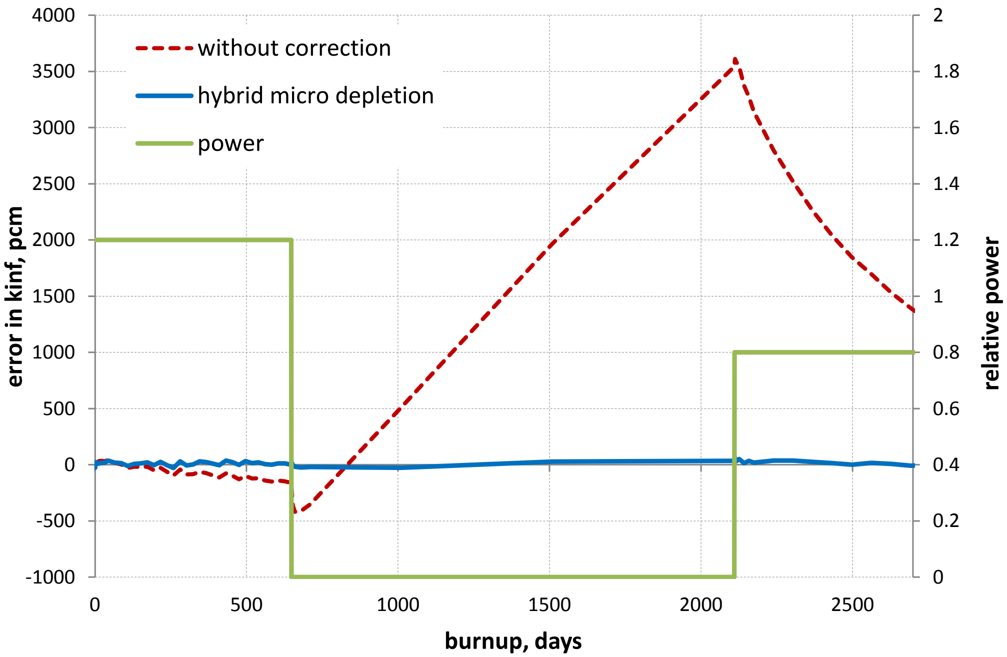


water density 500 kg/m^3 with 3-year outage

Hybrid micro-depletion verification – PWR MOX



boron concentraion 500 ppm



power variation with 4-year outage

Summary

- Main advantages of the new hybrid micro-depletion:
  - fast and flexible depletion solver
  - correction of micro-XS and other macro-XS
- Test cases demonstrate very good results
  - various history effects
  - various fuels
- Issues with micro-XS from Serpent:
  - detectors are slow
  - homogenization of micro-XS
  - output of fission yields
- Future work:
  - transmutation matrix optimization
  - test on realistic cases
  - decay heat calculation

Thank You!