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CENTRE D'ÉTUDE DE L'ÉNERGIE NUCLÉAIRE

BR2 Reactor Department, SCK∗CEN, Mol, Belgium

3-D FUEL MODELING WITH MCNP&ORIGENS

S. Kalcheva and E. Koonen

&

SIMULATION OF IRRADIATION HISTORY IN COMBINED ANNULAR MULTI-TUBE FUEL ELEMENT OF BR2

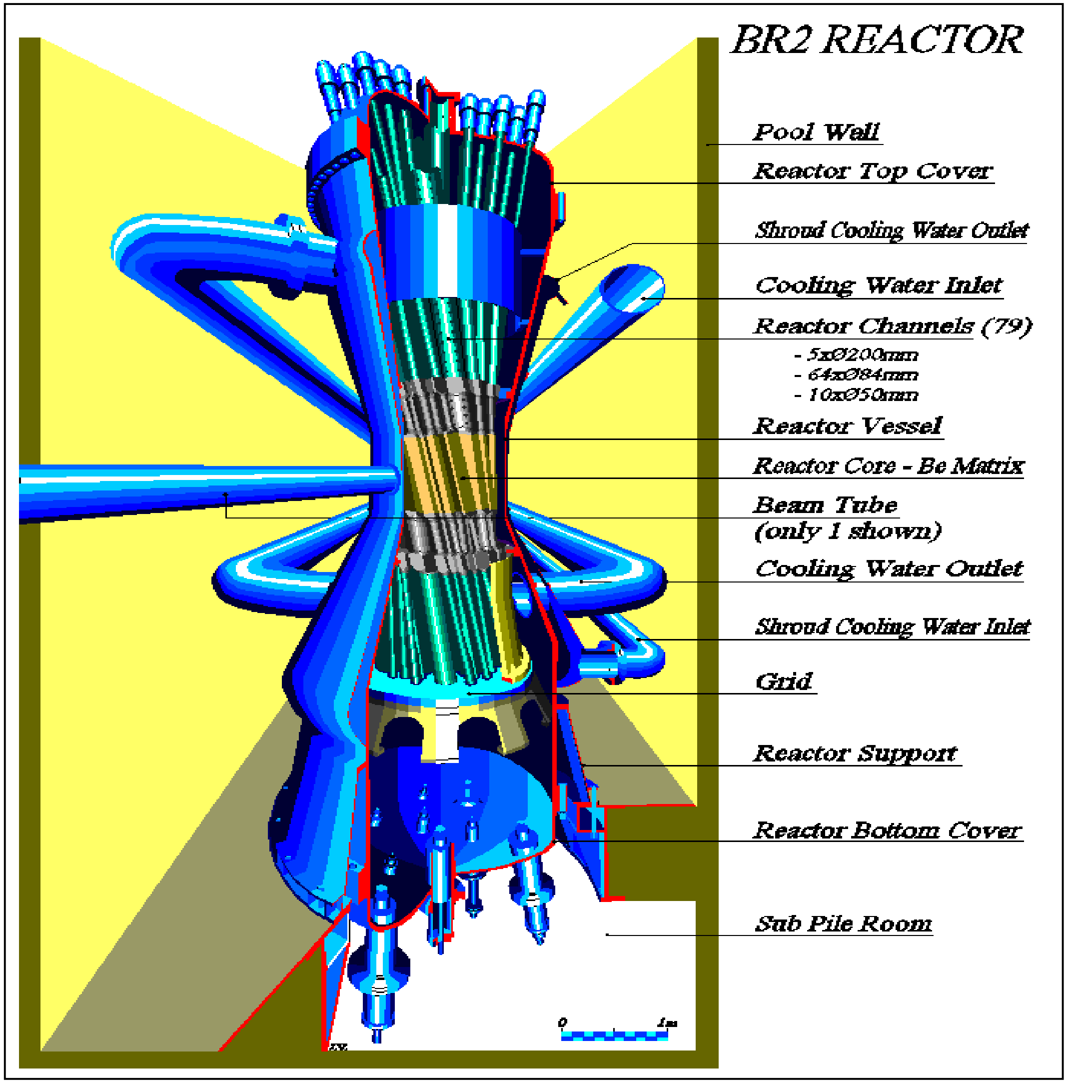
V. Kuzminov and E. Koonen

RRFM 2006, Sofia Bulgaria, 30 April – 3 May 2006

BR2 reactor:
HEU core positioned in beryllium matrix

Be-matrix:

inclined beryllium hexagons arranged in a form of a twisted hyperboloid bundle



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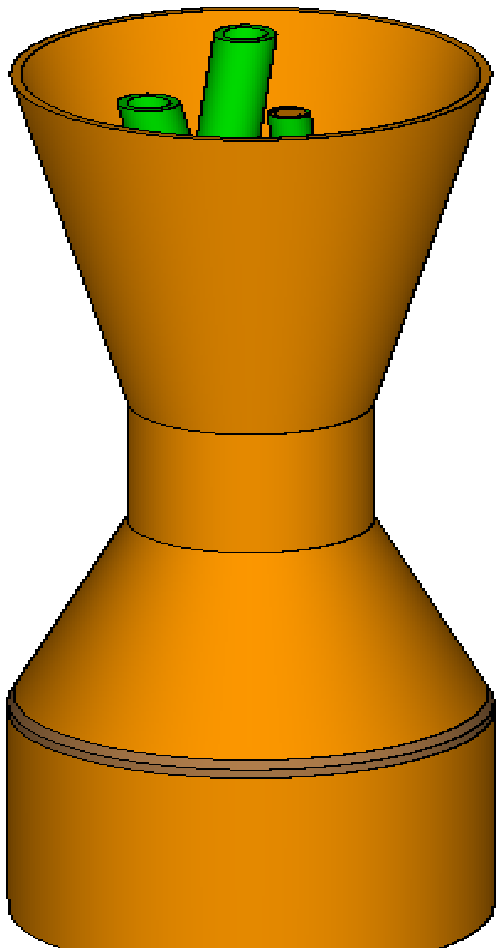
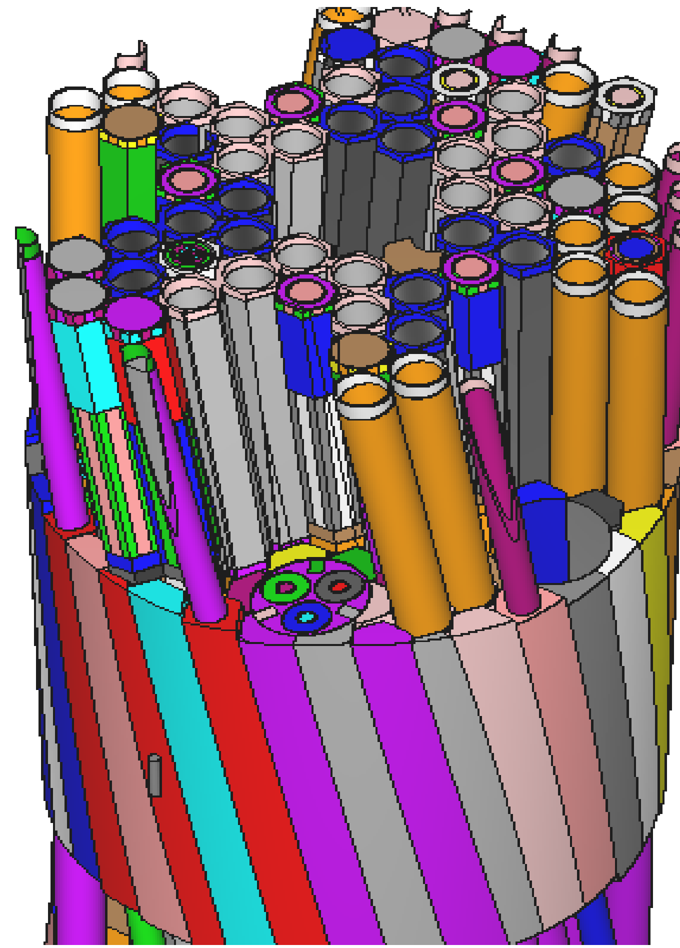
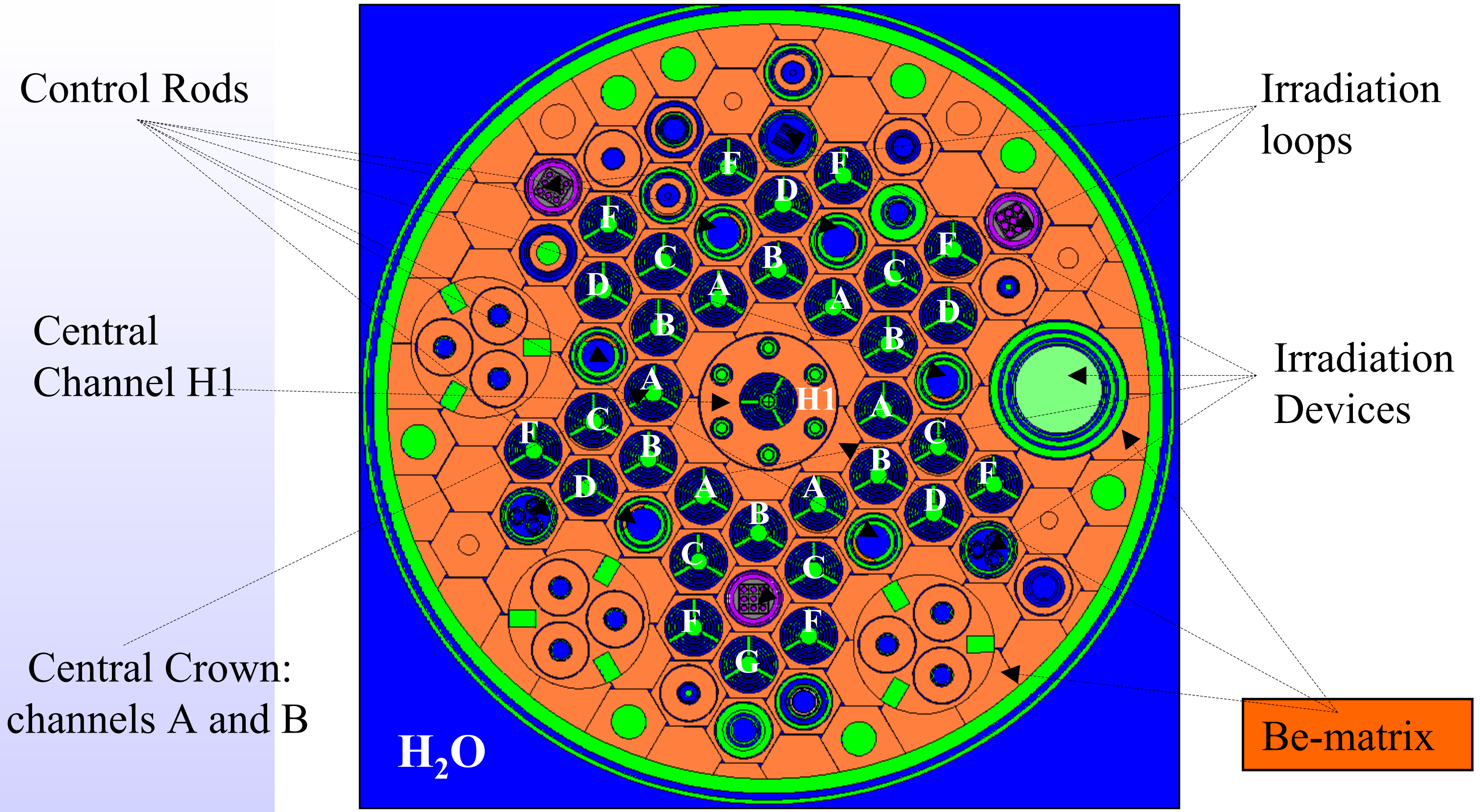
Belgian High Flux Materials Testing Reactor BR2

3-D MCNP simulation model of the BR2 reactor core containing inclined channels





Full-scale 3-D heterogeneous geometry model of BR2, developed with MCNP



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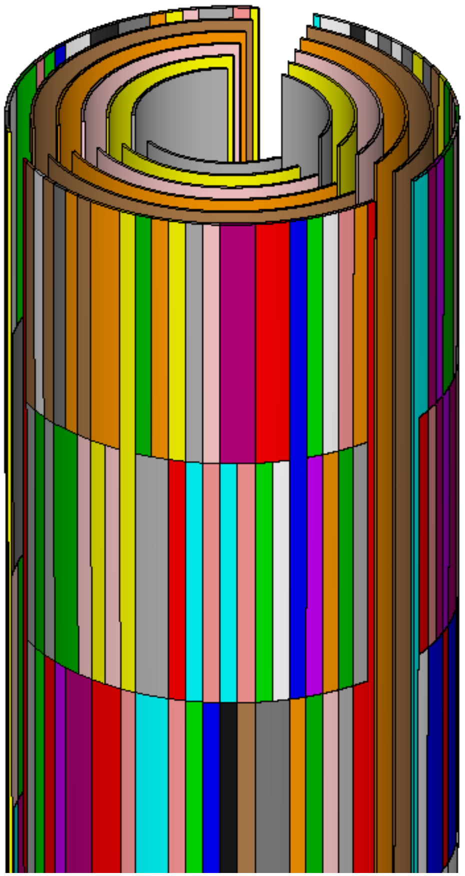
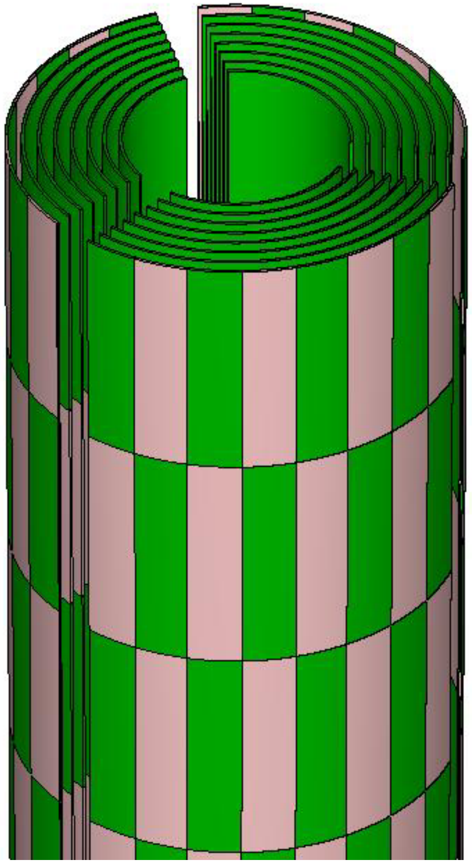
BR2 Multi-tube annular fuel element

MCNP simulation modeling:

- radial
- axial
- azuimuthal



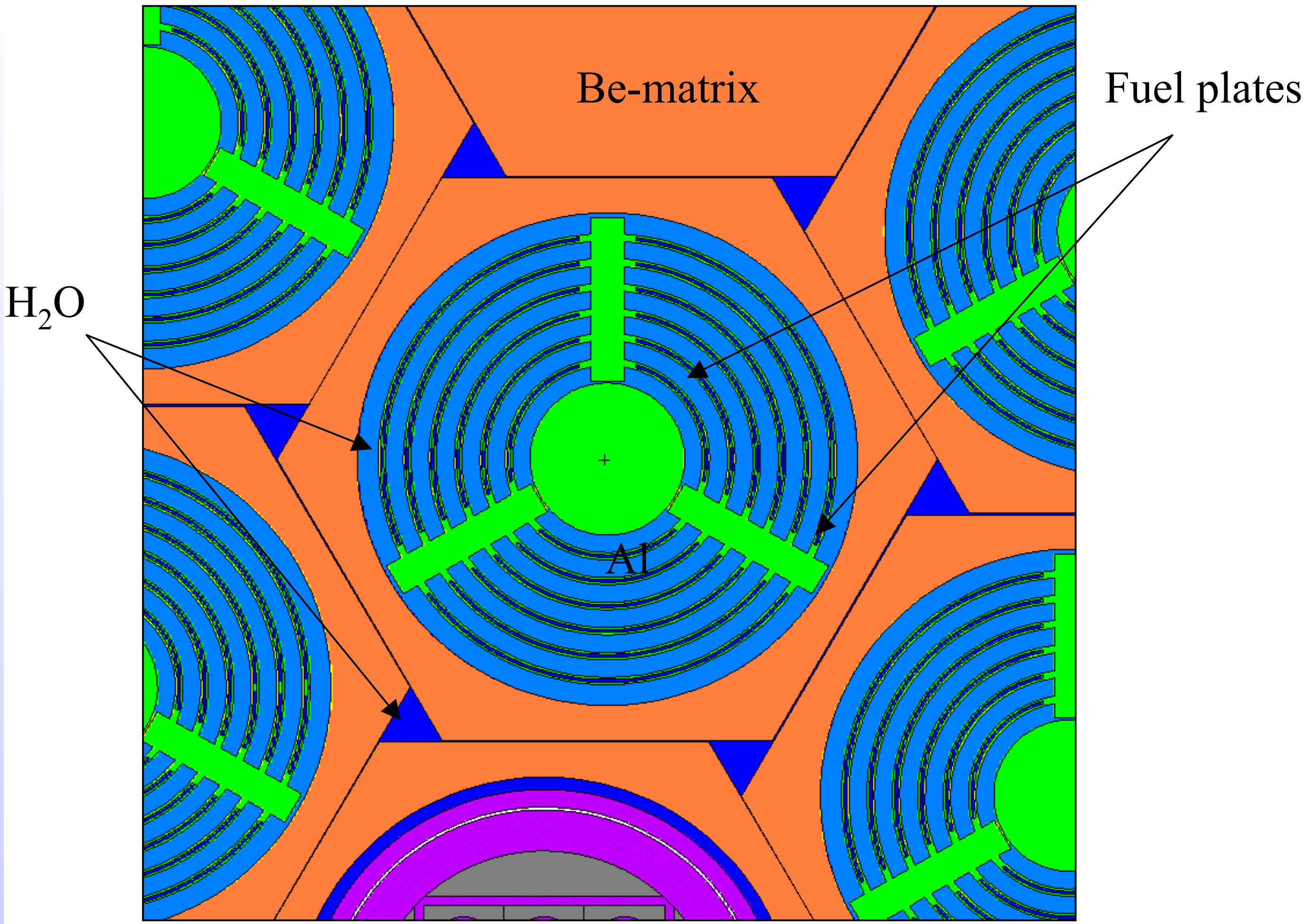
- 6 rings



- 8 rings

BR2 Multi-tube annular fuel element

MCNP simulation model



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Combined MCNP&ORIGENS model: Validation

- Model validation:
  - comparison with reactivity measurements:

    > than 20 operation cycles
  - comparison with thermal balance and γ-spectroscopy methods for determination of the linear power in fuel rods during transient tests
  - comparison with the dosimeter measurements of activation foil reaction rates
- High accuracy of the assessed reactor safety characteristics:
  - **reactivity values** → the computational predictions at various restarts during reactor operation are within: **Δk/k ~ 0.4%**
  - **Linear power, neutron fluxes:**

    typical uncertainties are within **3÷6%.**

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Objectives in this presentation

- Description of our combined MCNP&ORIGENS method used for 3-D isotopic fuel depletion modeling
  - equation of fuel depletion versus burn up
  - correlation between fuel depletion and 3-D relative power distribution
  - creation of a data base with stored isotopic fuel depleted compositions for each fuel type
- Uncertainties in criticality calculations during operation:
  - effect of burn up profile on reactivity values
- Uncertainties in the heat flux values:
  - estimation of the maximum value of the heat flux at the hot spot due to different orientations of FE

Why the method was developed?

- **Practical tool for calculations of reactivity evolution in time of the whole core during operation**
  - prediction of 3-D fuel burn-up distribution in plates during several cycles of the irradiation history
- Requirements:
  - high accuracy of criticality calculations: $\Delta k/k \leq 0.4\%$
  - easy and flexible for use
  - relatively “fast” (Monte Carlo) method for estimation of reactivity variations during an operation cycle
- Alternative to the existing methods

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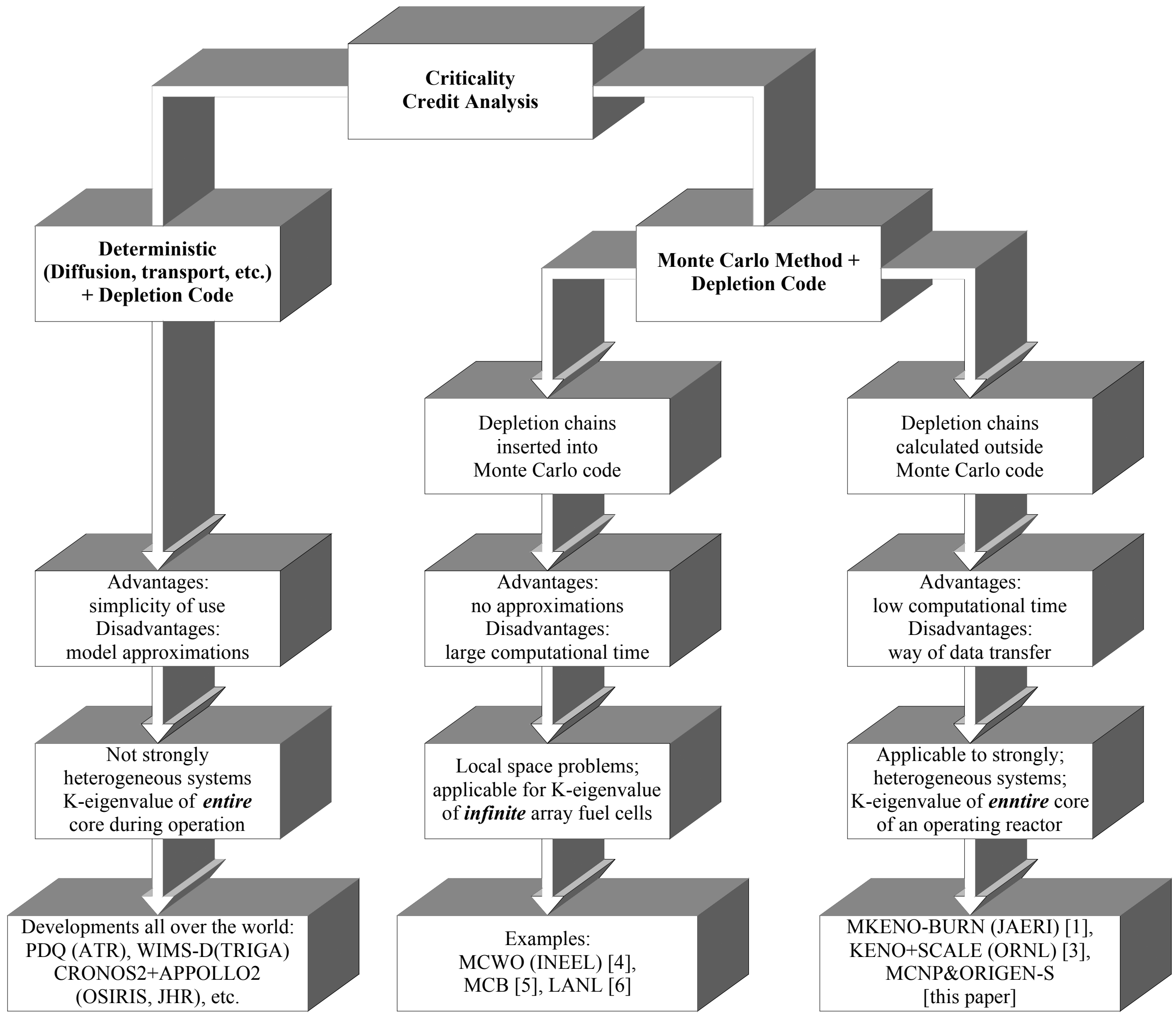
General overview of linking Monte Carlo and deterministic methods with depletion codes



Equation of fuel depletion vs. burn up

- Equation for fuel depletion β^k [%] for total fuel of type "k", including formation and depletion of all TRU - isotopes:

$$\overline{\beta^k}[\%] = \frac{M^k(0) - M^k(T)}{M^k(0)} \approx \text{const}^k \times \frac{T \times \overline{P^k}}{M^k(0)} = \frac{\text{const}^k \times T}{c^k(0)} \times \frac{\overline{P^k}}{V} \qquad (1)$$

$$\text{const}^k = \frac{A^k}{N_A \cdot E^k_{eff}} \cdot \frac{\langle \sigma_f + \sigma_c \rangle^k}{\langle \sigma_f \rangle^k} \qquad (2)$$

$$\alpha^k = \frac{\langle \sigma_f + \sigma_c \rangle^k}{\langle \sigma_f \rangle^k}$$ is constant for fuel type "k" in fuel zone $\{v\}$

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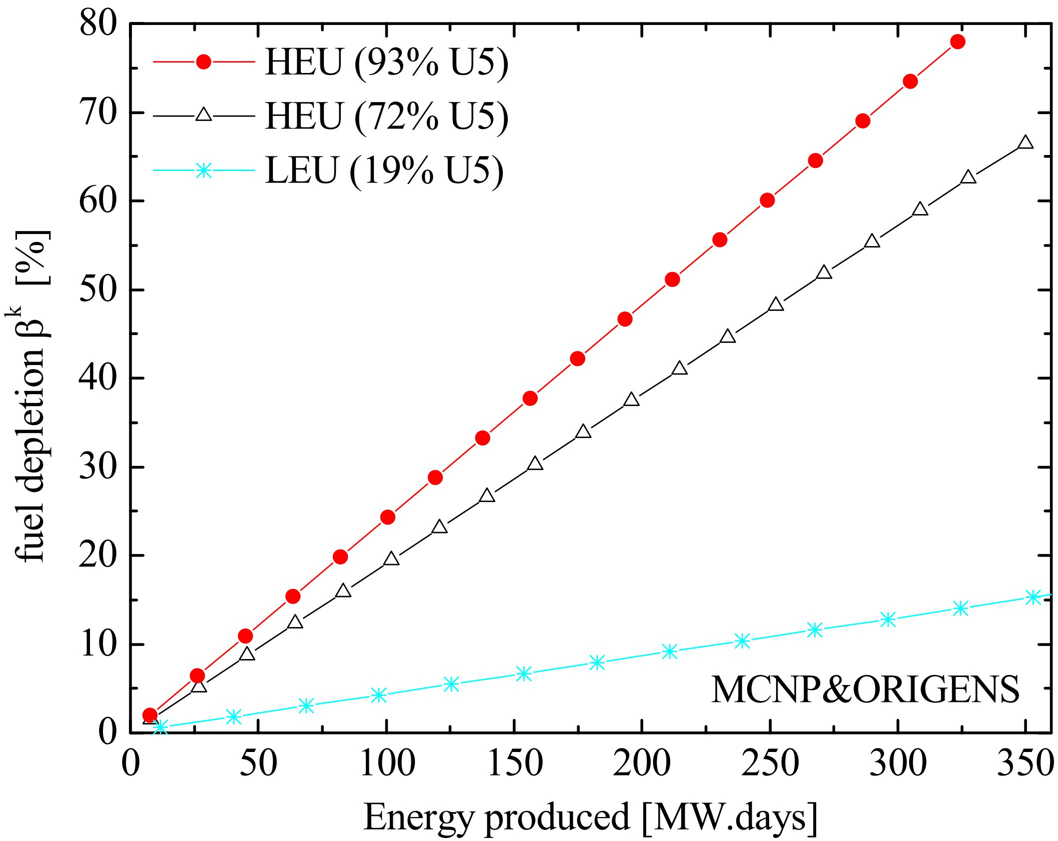
Dependence of total fuel depletion β^k [%] vs. burn up

Linear dependence:

β^k [%] =f(MW.d)

- valid for any fuel type "k" in a same fuel zone
- included formation and depletion of TRU-isotopes
- different slopes of the graphs → influence of ^{238}U content and TRU

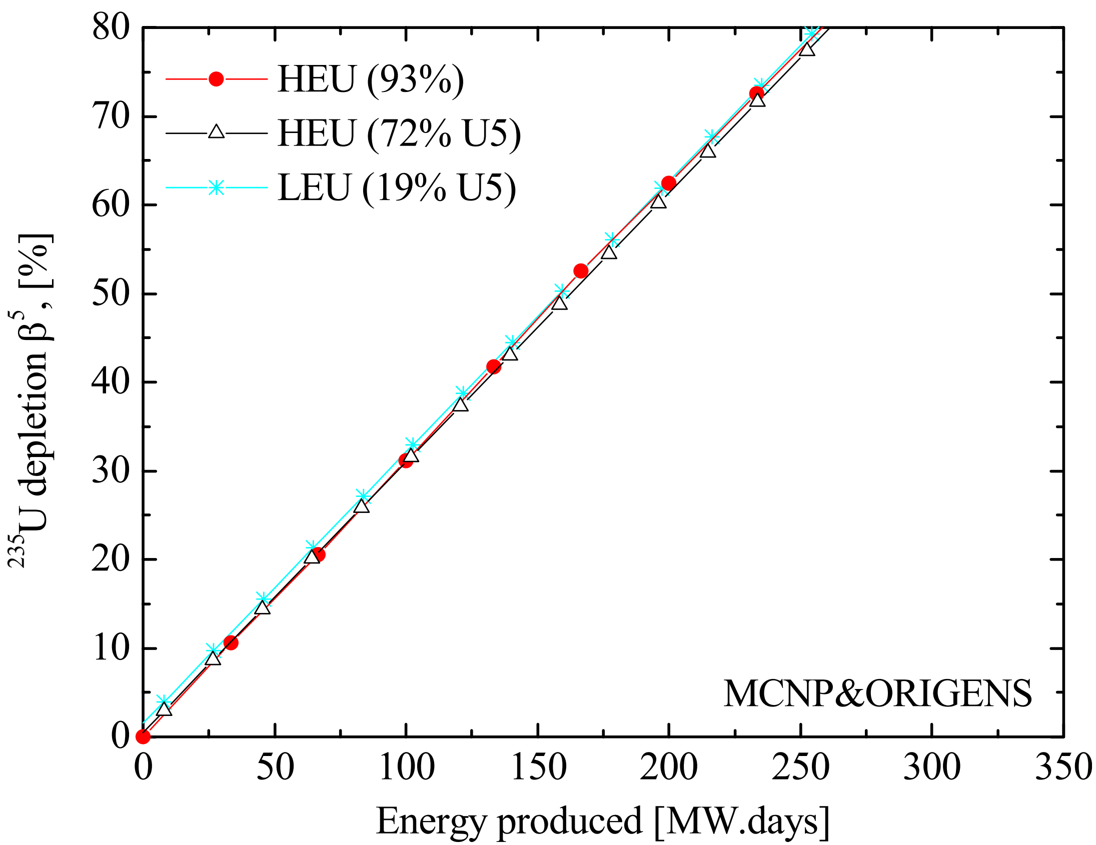


Dependence of ^{235}U depletion β^5 [%] vs. burn up

Linear dependence:

β^5 **(%)** =f(MW.d)

- equivalent for all fuel types for a same fuel zone at the same power



Isotopic evolution vs. burn up

Light elements:

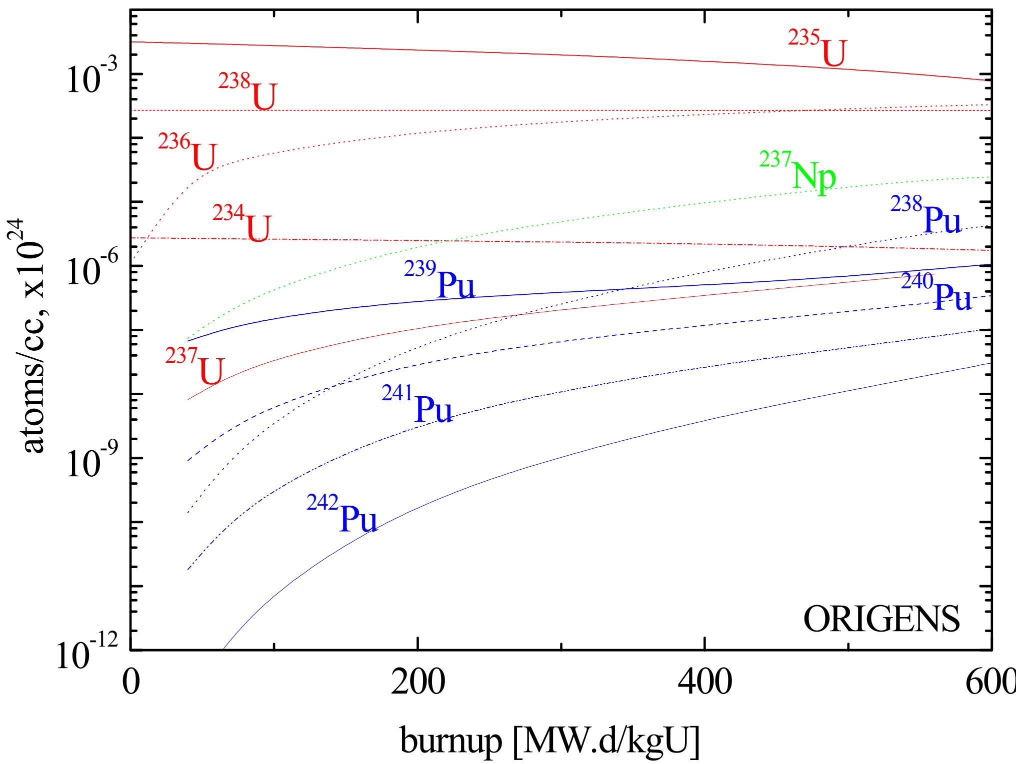
C, O, Al, ^{10}B, ^{11}B

Actinides:

^{234}U, ^{235}U, ^{236}U, ^{237}U, ^{238}U,

^{237}Np, ^{238}Np, ^{239}Np,

^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu



Evolution of fission products inventory vs. burn up and reactivity effect

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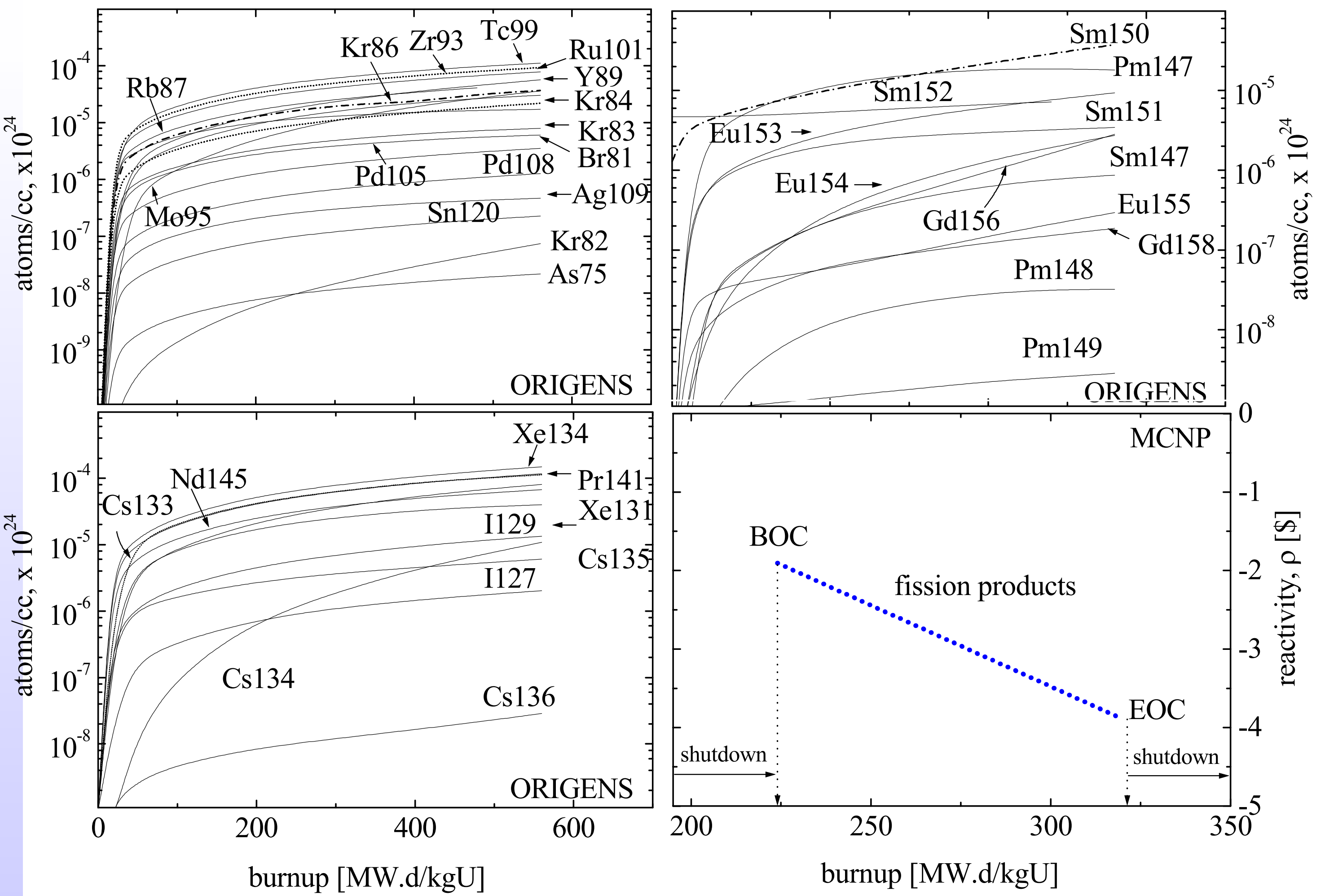
^{75}As, ^{81}Br, ^{82}Kr, ^{83}Kr, ^{84}Kr, ^{85}Kr, ^{86}Kr, ^{85}Rb, ^{87}Rb, ^{89}Y, ^{93}Zr, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Ru, ^{103}Rh, ^{105}Rh, ^{106}Pd, ^{108}Pd, ^{109}Ag, ^{110}Cd, ^{111}Cd, ^{113}Cd, ^{114}Cd, ^{115}Cd, ^{116}Cd, ^{120}Sn, ^{127}I, ^{129}I, ^{135}I, ^{131}Xe, ^{134}Xe, ^{135}Xe, ^{133}Cs, ^{134}Cs, ^{135}Cs, ^{136}Cs, ^{137}Cs, ^{138}Ba, ^{141}Pr, ^{145}Nd, ^{147}Nd, ^{148}Nd, ^{147}Pm, ^{148}Pm, ^{149}Pm, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{153}Eu, ^{154}Eu, ^{155}Eu, ^{156}Gd, ^{158}Gd, etc.



Evolution of fission products inventory vs. burn up and reactivity effect

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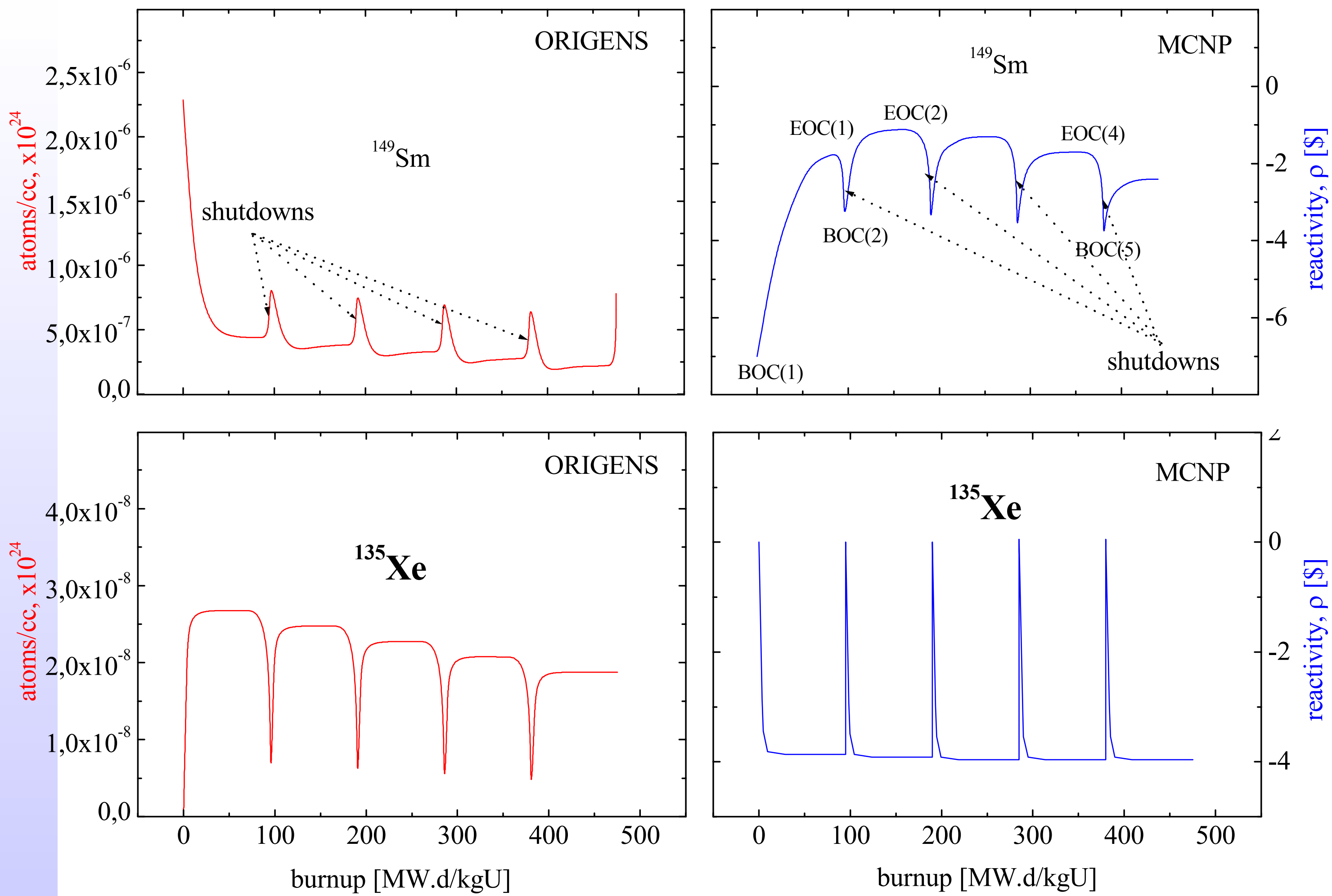
^{149}Sm

^{135}Xe



3-D distribution of isotopic fuel depletion in FE

- Correlation between local fuel depletion and relative power distribution

$$\beta^{k}(T_i, r_m, z_n, \varphi_l) = \overline{\beta^{k}_{FE}}(T_{i-1}) \times K^{k}_{V}(T_{i-1}, r_m, z_n, \varphi) + \frac{const^{k}_{FE} \times (T_i - T_{i-1})}{c^{k}_{FE}(0)} \times \frac{\overline{P^{k}_{FE}}(T_i)}{V_{FE}} \times K^{k}_{V}(T_i, r_m, z_n, \varphi_l) \qquad (3)$$

- Power peaking factor:

$$K^{k}_{V}(T_i, r_m, z_n, \varphi_l) = \frac{\int_{r_m} dr \int_{z_n} dz \int_{\varphi_l} d\varphi\, p\,(T_i, r, z, \varphi) \Big/ \int_{r_m} dr \int_{z_n} dz \int_{\varphi_l} d\varphi}{\overline{P^{k}_{FE}}(T_i) \big/ V_{FE}} \qquad (4)$$

Simulation of fuel burn up irradiation history

- After N irradiation cycles (duration of N cycles is T_N) the local fuel burn-up, $\beta(T_N)$, in each fuel zone {V}

$$\beta(T_N) = \beta(T_{N\text{-}1}) + \left[\overline{\beta}_{FE}(T_N) - \overline{\beta}_{FE}(T_{N-1})\right] \times k_V(T_N)$$

$$= \beta(T_1) + \sum_{i=2}^{N} \left[\overline{\beta}_{FE}(T_i) - \overline{\beta}_{FE}(T_{i-1})\right] \times k_V(T_i) \qquad (5)$$

- The fuel burn-up $\beta(T_N)$ in registration zone {V} after T_N irradiation time can be re-written:

$$\beta(T_N) = \sum_{i=1}^{N-1} \overline{\beta}_{FE}(T_i) \times \left[k_V(T_i) - k_V(T_{i+1})\right] + \overline{\beta}_{FE}(T_N) \times k_V(T_N) \qquad (6)$$

Equation (6)

- functional dependence of local burn-up $\beta(T_N)$ on mean burn-up $\beta_{FE}(T_N)$ after the N^{th} irradiation cycle
- takes into account the changing of the power peaking factors in different cycles
- $\beta_{FE}(T_N)$ is calculated for the mean operation power (determined by MCNP) using the SCALE code.
- the dependence of the nuclide composition versus the released energy (fuel burn-up) in a fuel element (plate) is calculated only one time and kept in the form of a table.
- this table is used each time when it is necessary to obtain the fuel composition for the known local burn-up in the registration mesh.

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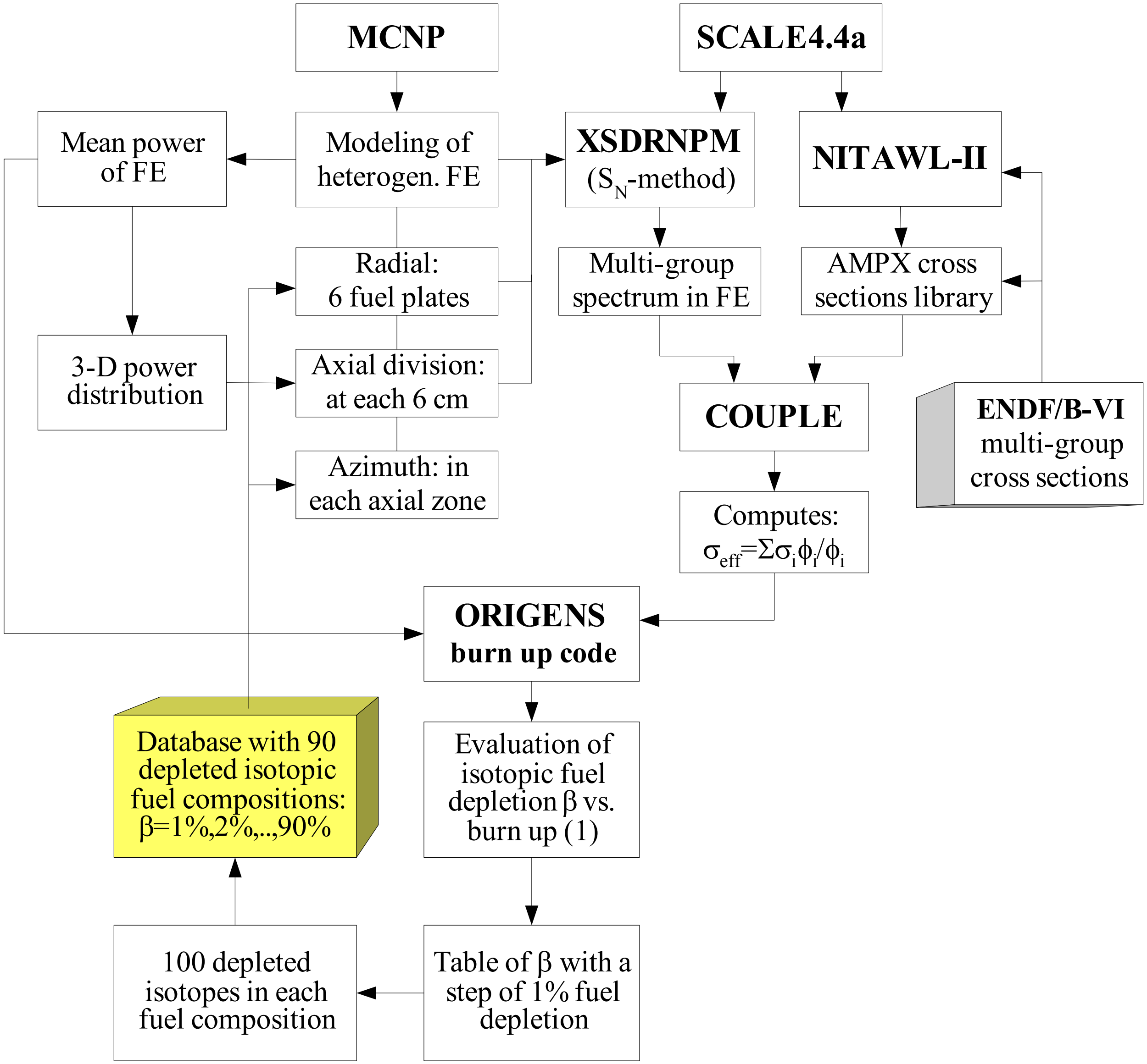
Fuel burn-up in the fuel element containing dominant fissile nuclide ^{235}U

$$\beta_5(T_N) = 1 - \exp\left(\sum_i^N - \left(\frac{2\overline{q}(T_i)}{\delta_f E_{eff} \rho_5(0)\left(1-\beta_5(T_{i-1})\right)} \Delta T_i \right) \frac{\langle \sigma_c + \sigma_f \rangle}{\langle \sigma_f \rangle} \right) 100\%$$

$\Delta T_i \rightarrow$ duration of the irradiation period i
$q(T_i)$ (W/cm^2) $\rightarrow$ mean value of the heat flux at T_i on the surface of the fuel zone with the thickness of δ_f

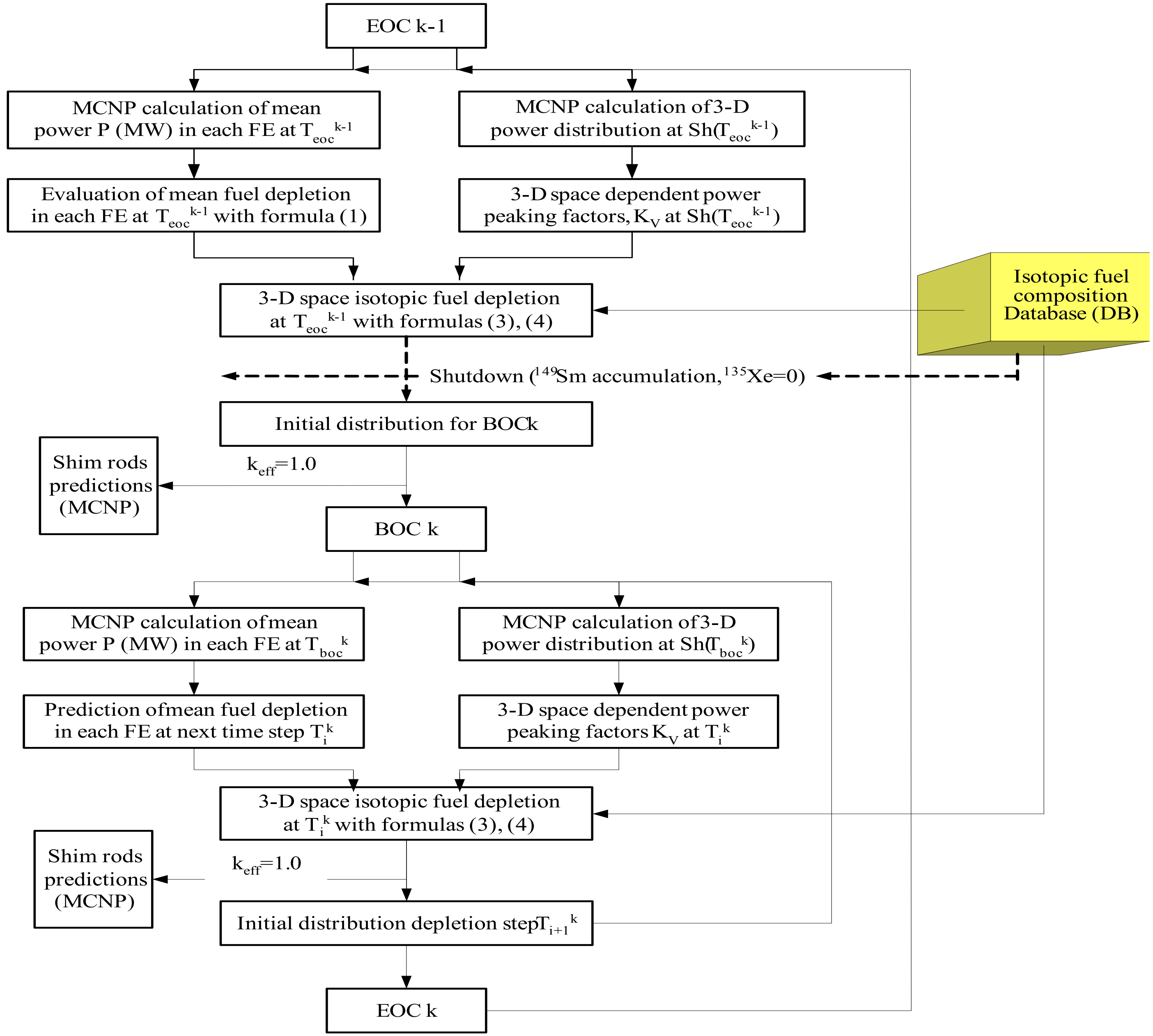
Linking MCNP with modules of the SCALE system

- creation of a Data Base (DB) with stored isotopic depleted fuel compositions, which are →
- easily derived from DB and inserted into spatial registration mesh of MCNP model



Criticality calculations of whole core

- **Number of depletion time steps ~15÷18**
- **Computation time ~ 6÷8 hrs.**
- **3 parallel PC/2GHz**
- **$\Delta k_{eff}=\pm 0.0005$**



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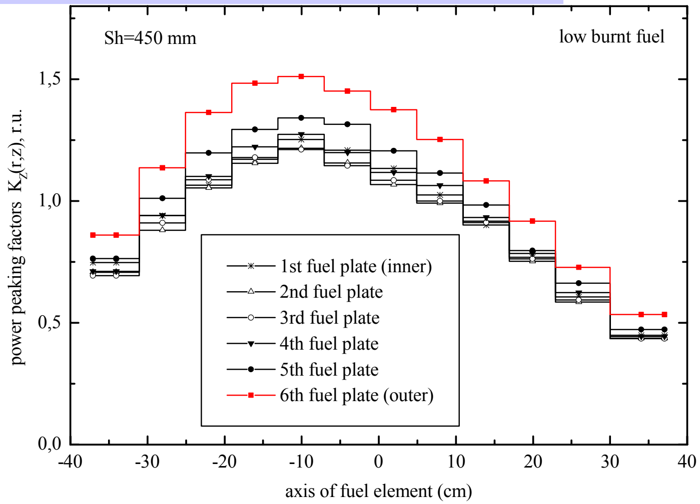
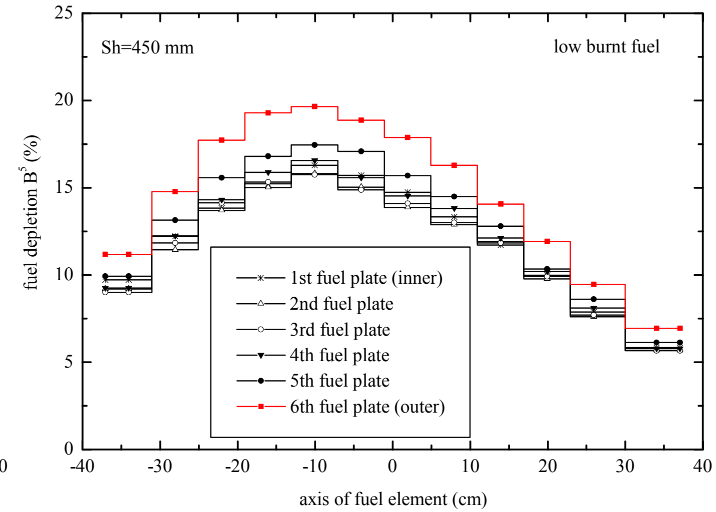
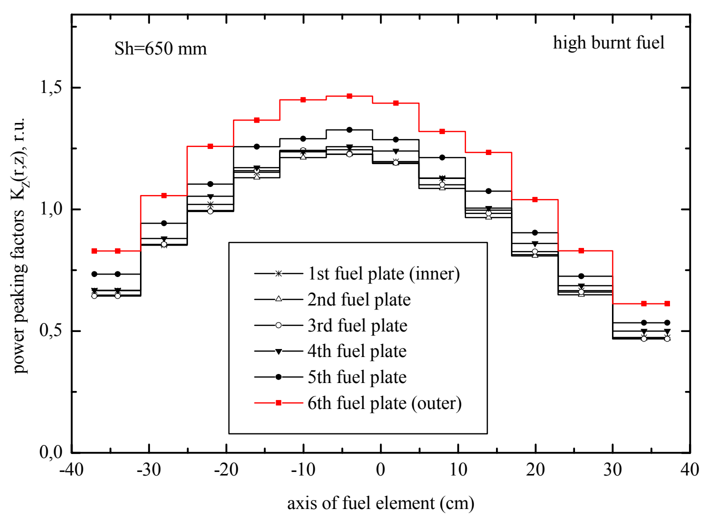
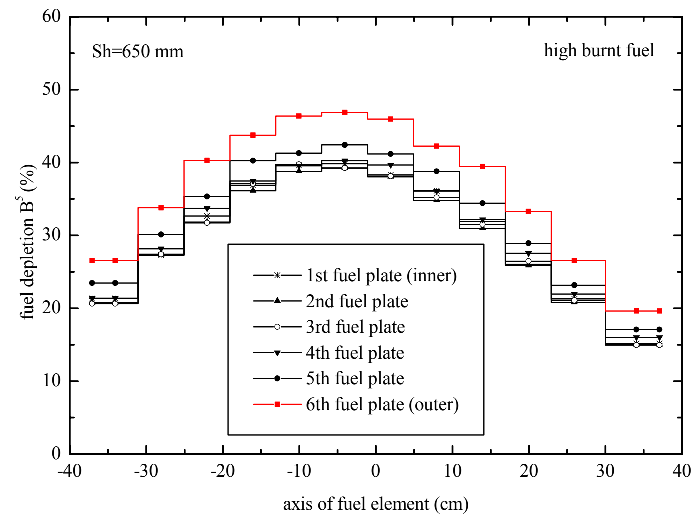
Axial and radial power peaking factors in fuel plates and fuel burn up profile

- **minimum number of spatial cells with varied burn up ~ 5000**









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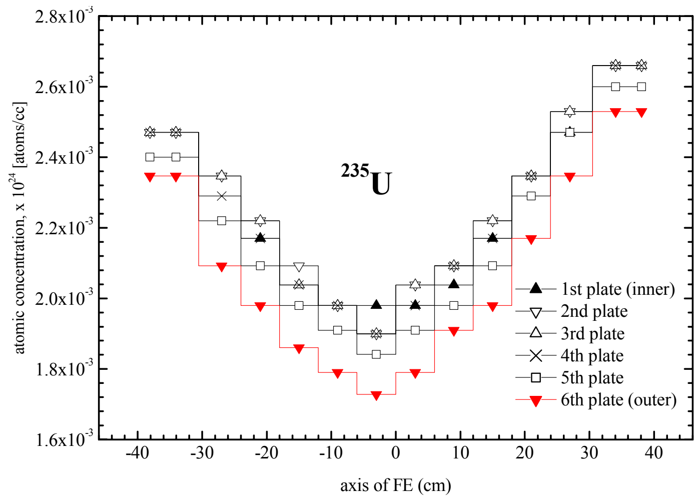
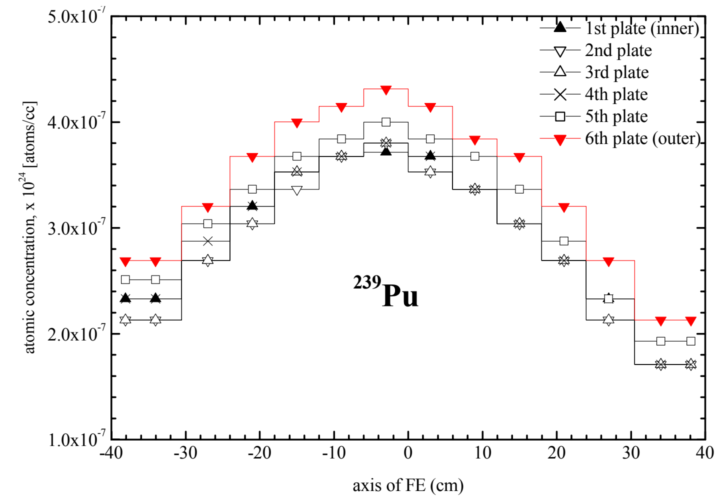
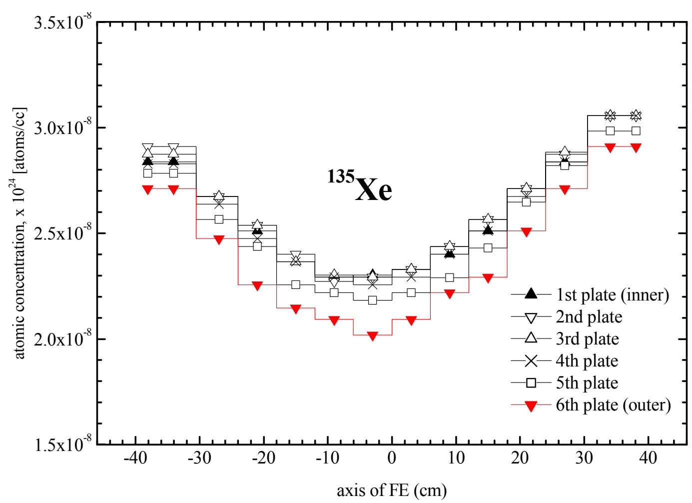
3-D Distribution of isotopic concentrations in fuel plates

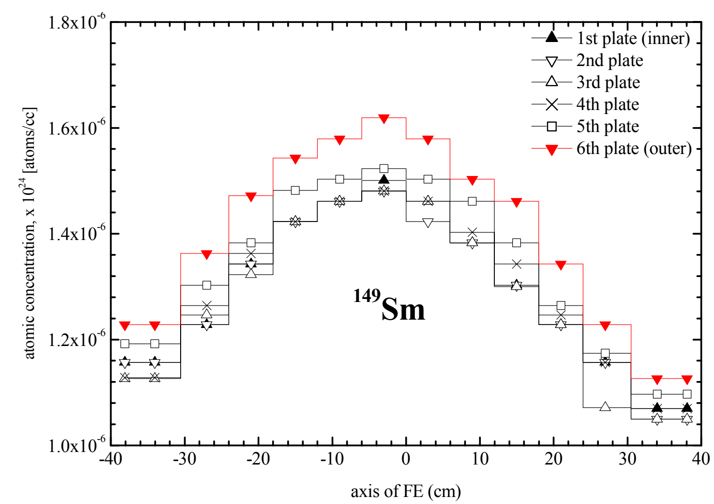
Similar distributions for ~ 100 isotopes:

- light elements
- actinides
- fission products









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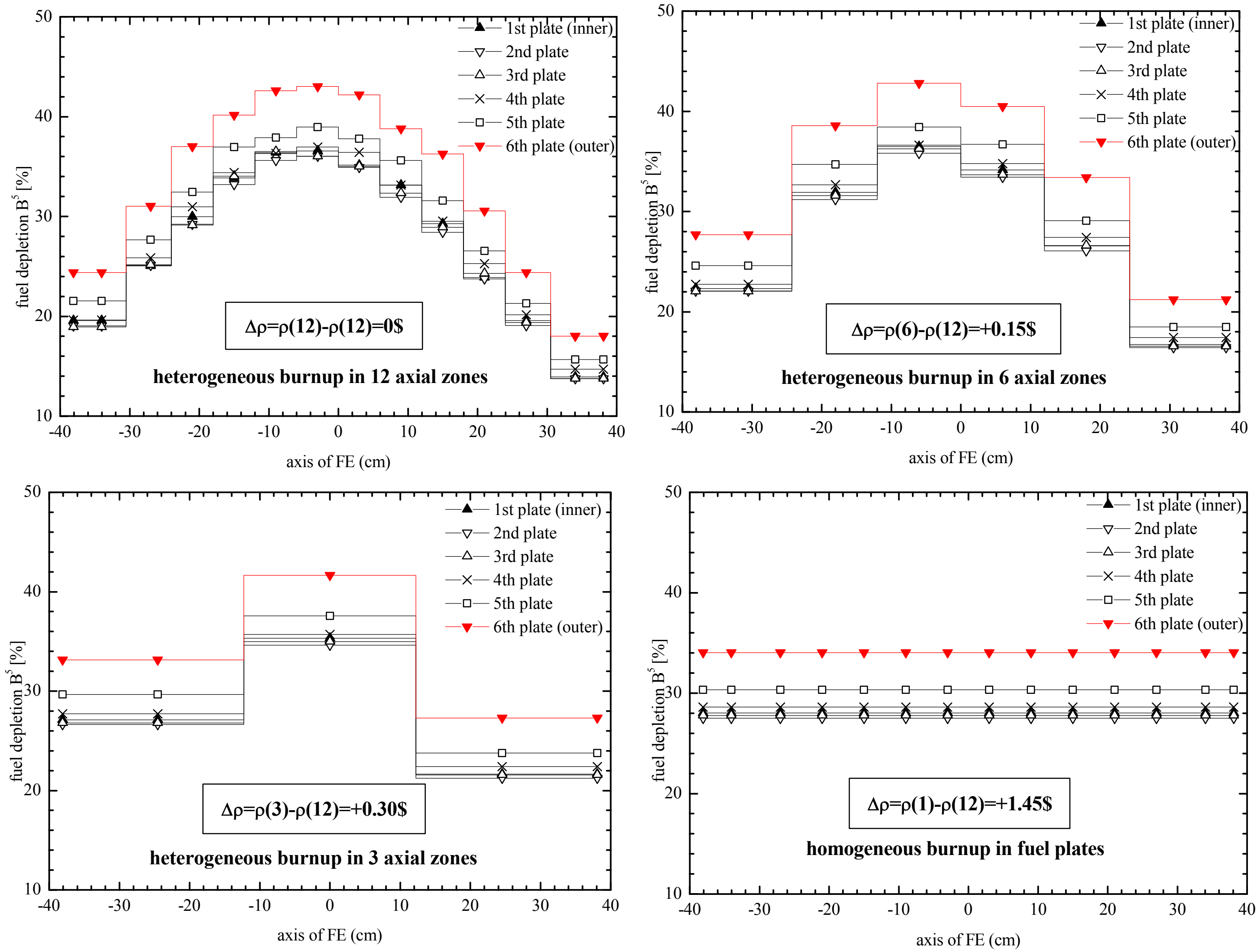
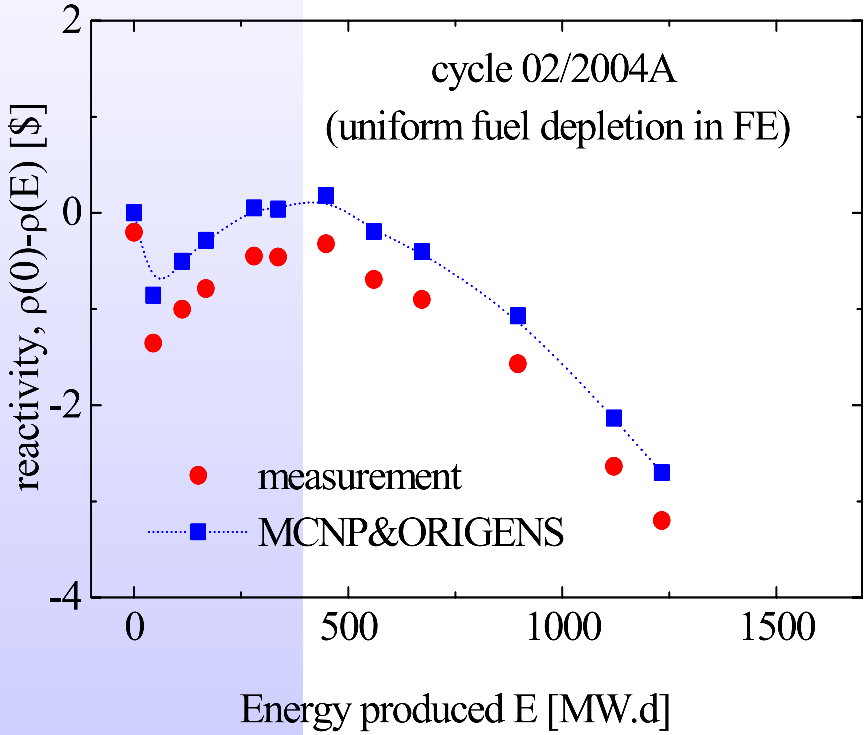
Reactivity effects of 3–D fuel burn up profile

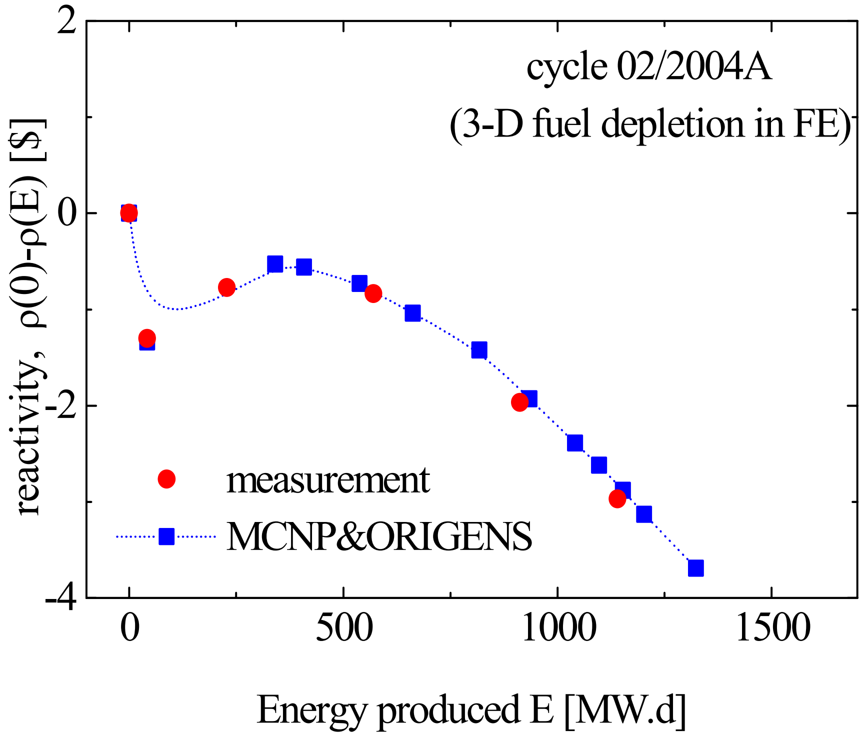
- Reactivity effects of 3–D fuel burn up profile are investigated for different number of axial divisions
- decreasing number of heterogeneous zones → uncertainty in Δk/k ~ 1.0%



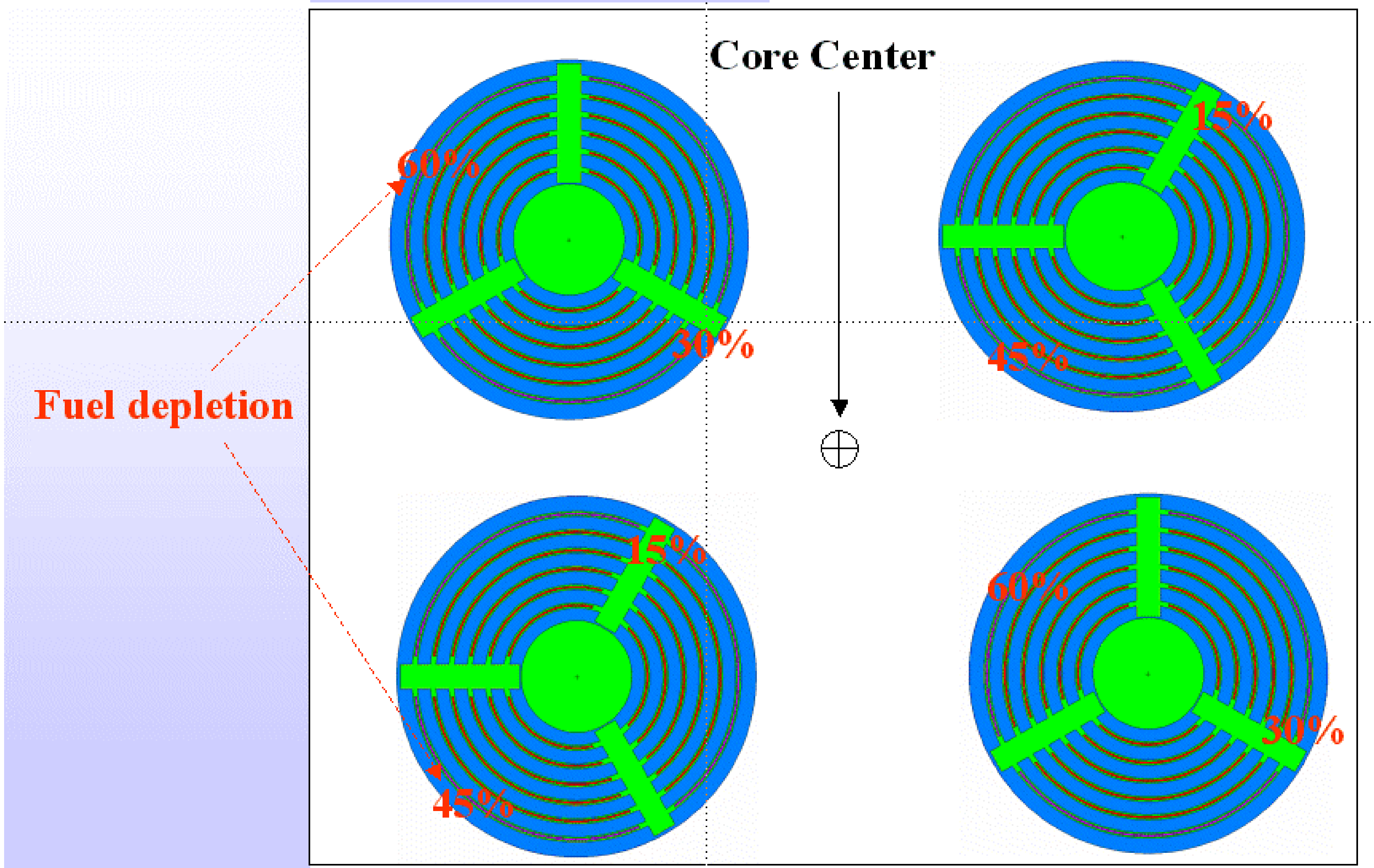
Reactivity effects of 3–D fuel burn up profile during an operation cycle

➢ use of homogeneous fuel depletion in ½ of the fuel elements inserts
uncertainty in reactivity values : Δk/k ~ 0.5$=0.36%

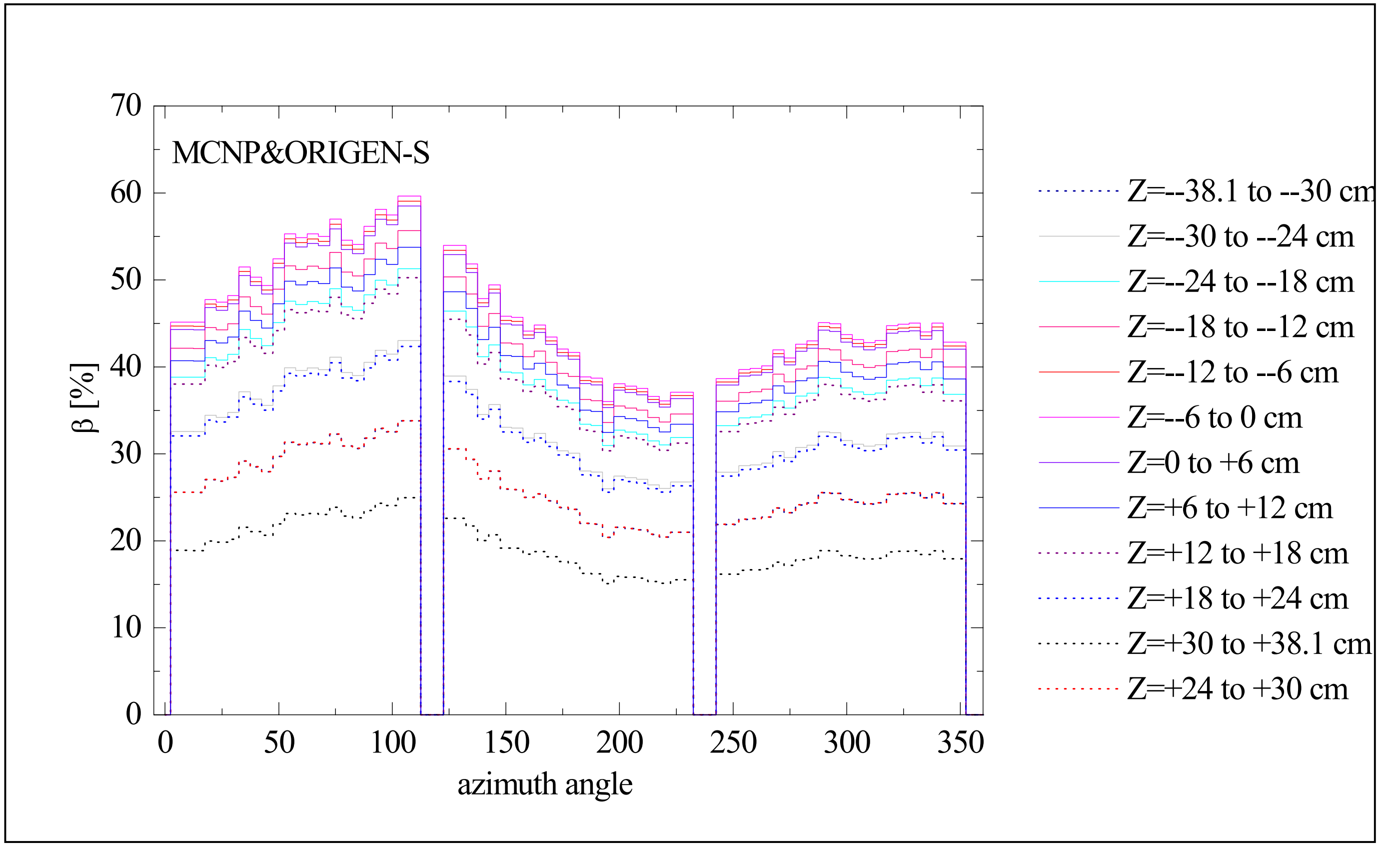




Orientation of FE with azimuthal depletion profile in the fuel plates



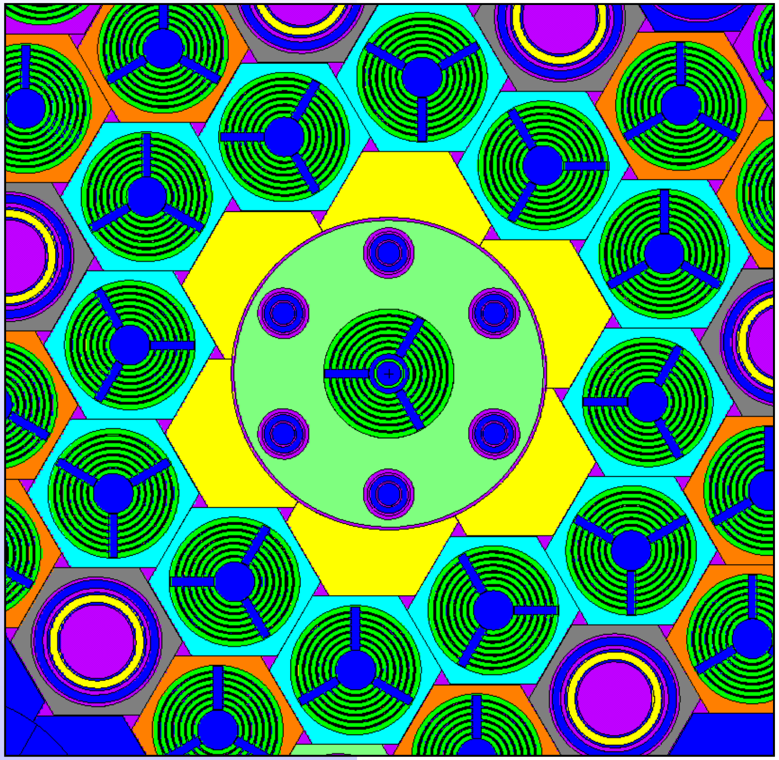
Azimuthal modeling of fuel depletion in different axial segments of fuel plates

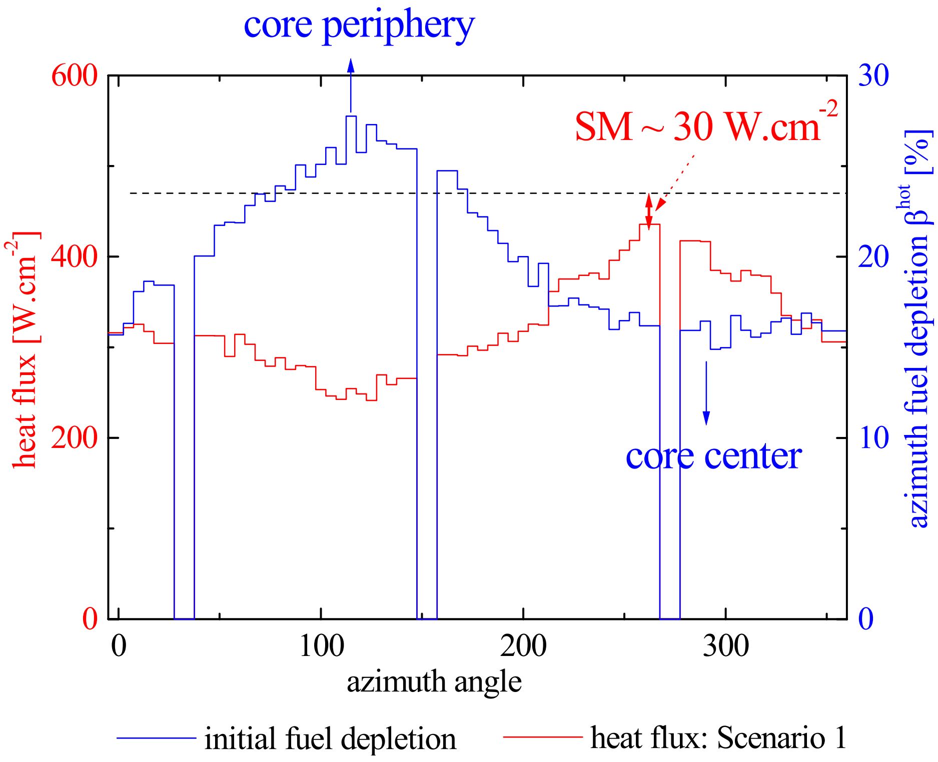


Assessment of safety margins of the heat flux for different azimuthal orientations of FE: Scenario 1

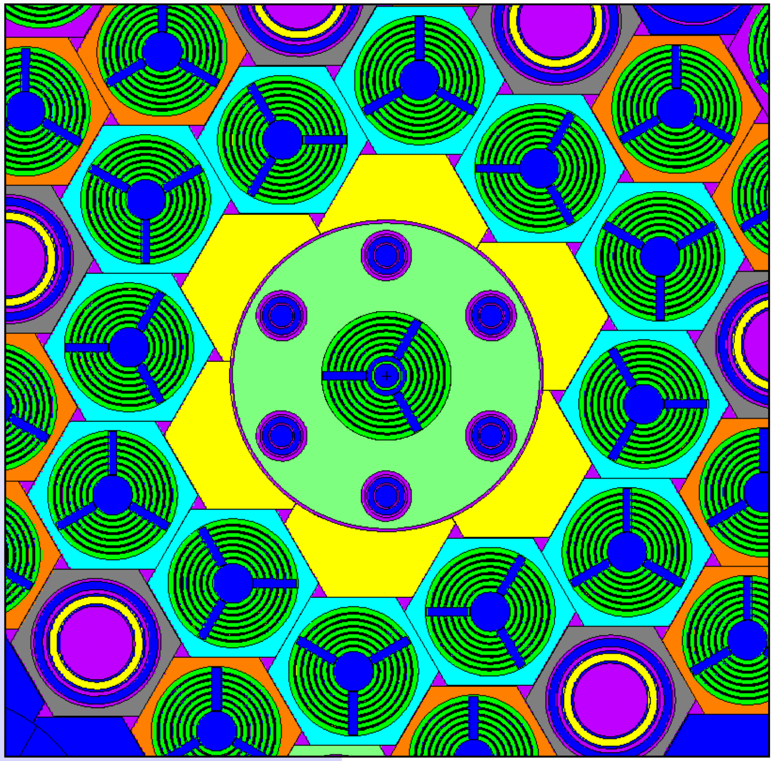
- The safety margin (SM) of the heat flux is defined with the following equation where: Q^{TH}_{SL} – safety operating limit, ΔQ^{TH}_{SL} – uncertainties of thermo-hydraulics calculations of hot spot factors

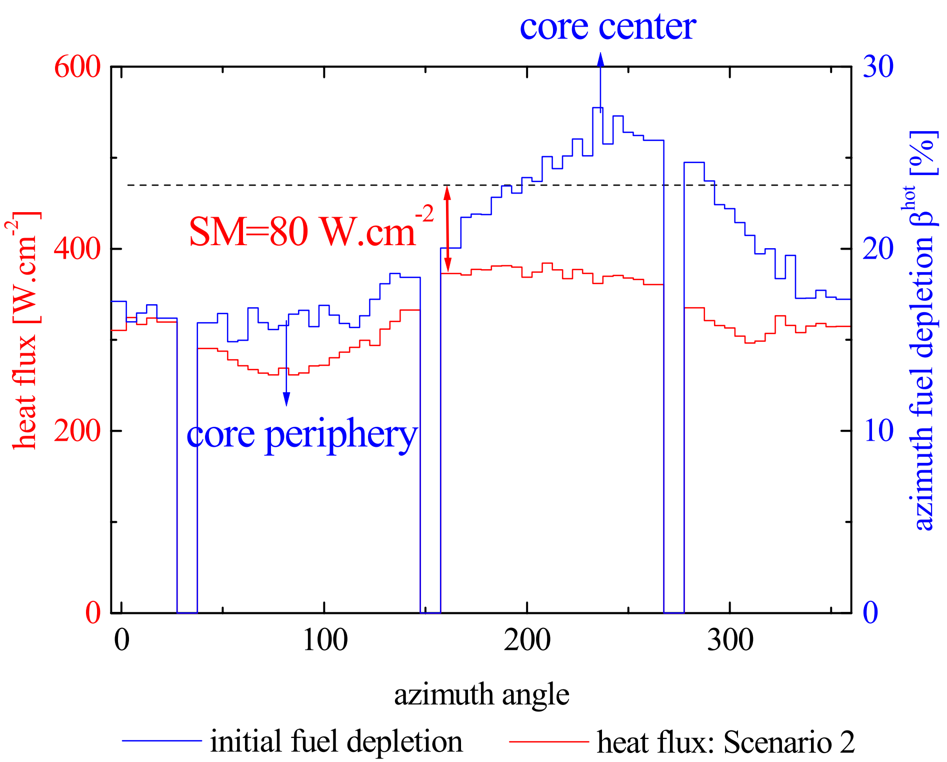
$$SM = (Q^{TH}_{SL} - \Delta Q^{TH}_{SL}) - (Q^{MCNP}_{max} + \Delta Q^{MCNP}_{max}) = 470 - (Q^{MCNP}_{max} + \Delta Q^{MCNP}_{max}), [W.cm^{-2}]$$





Assessment of safety margins of the heat flux for different azimuthal orientations of FE: Scenario 2





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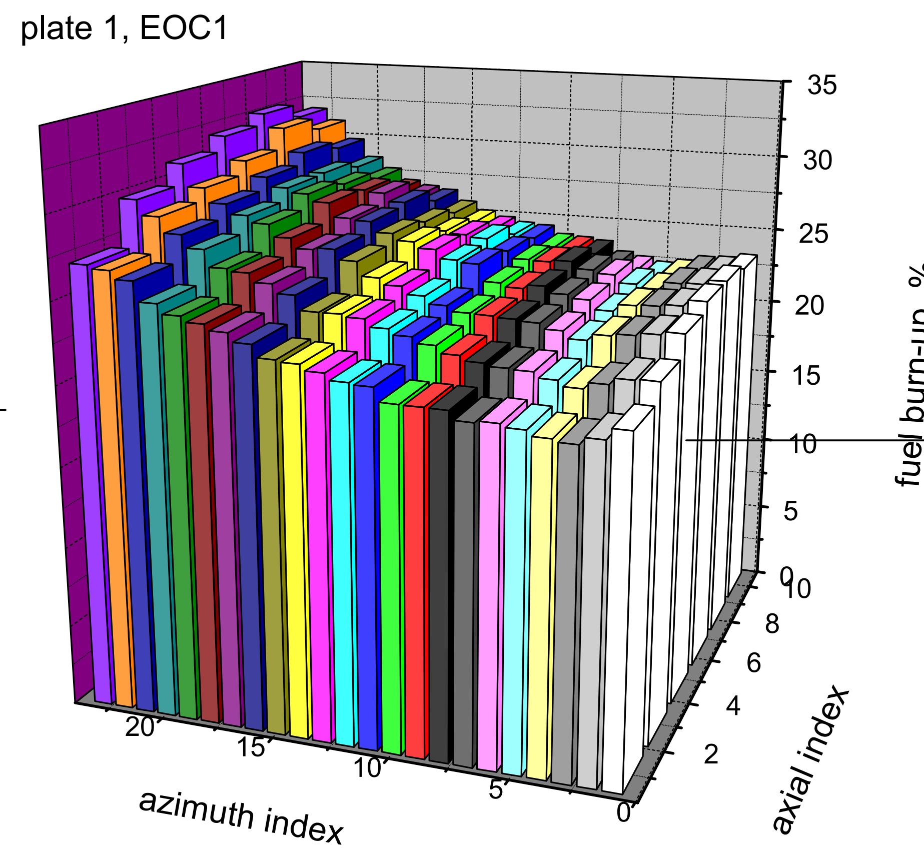
Example of burn-up distribution

- After the 1st irradiation cycle

Core centre



periphery

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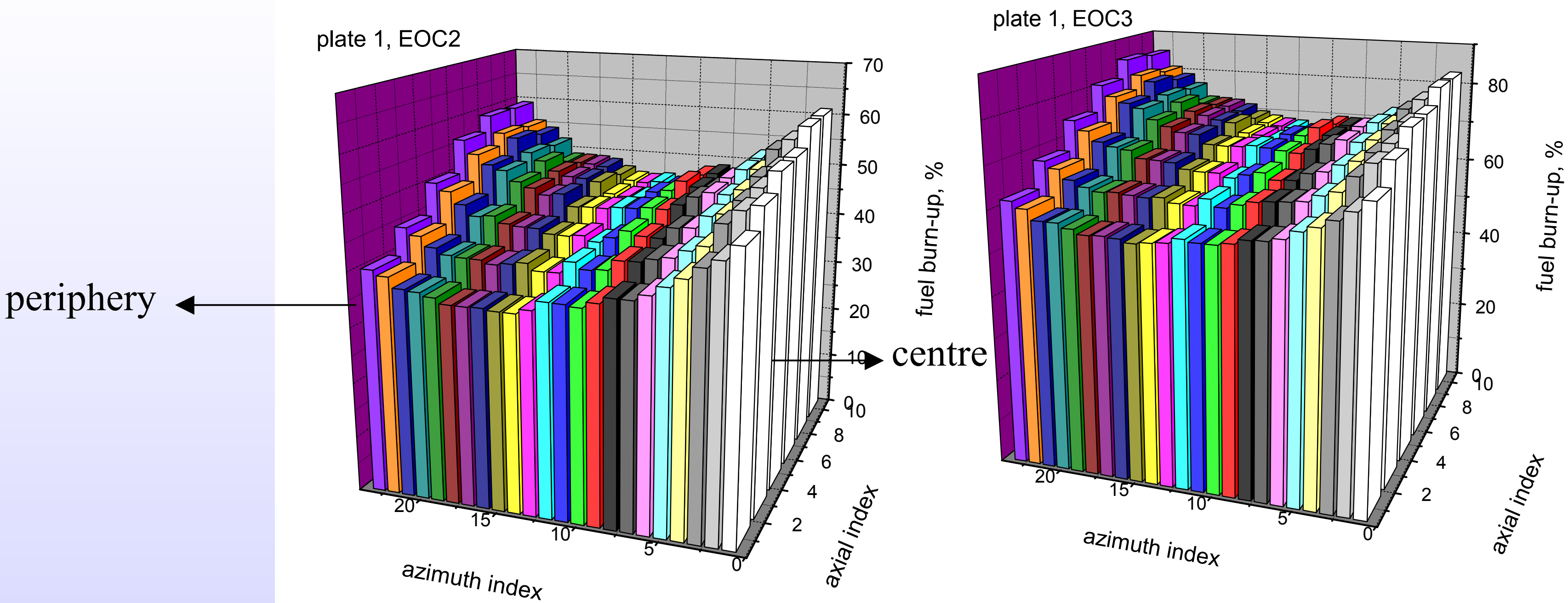
Example of burn-up distribution

- After the 2nd and 3rd irradiation cycles



Variation of the spatial distributions of the fuel burn-up in the central hottest part of one fuel plate due to the change of the azimuth orientation of the fuel element and its position in the reactor core in several successive irradiation cycles

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Summary

- Reactivity effects of burn up profile in 12÷24 axial segments in fuel plates are estimated in comparison with profiles in 6, 3 and 1 axial zones
  - Uncertainties in reactivity values due to **axial and radial burn up** modeling can be up to **1.0%**
  - Uncertainties in reactivity values, resulting from **azimuth fuel profile** in the fuel plates and due to different orientations of the fuel element in the core are within **0.2%**
- Variations of the maximum values of the heat flux due to different orientations of a **low burnt fuel element** in the core are within **10÷12%**
- Variations of the maximum values of the heat flux due to different orientations of a **high burnt fuel element** in the core can be about **20%**