STUDIECENTRUM VOOR KERNENERGIE
CENTRE D'ÉTUDE DE L'ÉNERGIE NUCLÉAIRE

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# MCNPX 2.6.C vs. MCNPX & ORIGEN-S: State of the Art for Reactor Core Management

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# Objectives

- Comparison of depletion capabilities between MCNPX 2.6.C and MCNP(X)&ORIGEN-S combined method
- Testing of MCNPX 2.6.C on the whole core criticality calculations of complex heterogeneous reactor system
- Validation of MCNPX 2.6.C on reactivity measurements at the rector BR2
- Application of MCNPX 2.6.C for reactor core management

# MCNPX 2.6.C depletion methodology

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- **Automatic, internally linked depletion calculations&steady-state flux calculations**
- **MCNPX: steady – state calculations**
  - continuous energy reaction rates for (n,γ), (n,f), (n,2n), (n,3n), (n,alpha) and (n,p) only for the requested materials; $k_{eff}$
  - 63 – energy group fluxes, used by CINDER90 to determine the rest of the interaction rates, which are not calculated by MCNPX
  - calculates the fission rate for thermal, fast and high – energy regions and selects automatically the appropriate fission yield corresponding to the energy range containing the majority of fissions at each time step
- **CINDER90: 1 – D depletion code**
  - Uses one – group constants from MCNPX to generate new number densities for the requested time step
  - Tracks the time-dependent reactions of 3456 isotopes using its own intrinsic 63 – energy group cross sections and decay data for daughter transmutation products when the information is not specified by MCNPX
  - Offers a thermal, fast and high – energy fission yield for each fissile isotope

# MCNP(X)&ORIGEN-S depletion methodology

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- **Externally linked depletion&steady – state flux calculations**

- **MCNP(X): steady – state calculations**
  - continuous energy reaction rates for (n,γ) and (n,f) of main fissile nuclides and (n,γ) of dominant F.P.
  - Calculation of 3 – D power distribution in the core; $k_{eff}$

- **ORIGENS: 1 – D depletion code**
  - Uses one – group constants from MCNP(X) to update the existing cross sections for LWR
  - Input is the fission power or neutron flux calculated by MCNP(X) in the spatial cells where the burnup calculations are needed
  - Evaluates the evolution of the isotopic fuel densities for the desired number depletion time steps
  - The isotopic fuel composition for a given time step is introduced back into the MCNP model and distributed in the core using the detailed 3–D power peaking factors

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# MCNP(X) calculation of effective thermal microscopic cross sections

| Nuclide | **ORIGENS** | **MCNP(X)** | Nuclide | **ORIGENS** | **MCNP(X)** |
|---|---|---|---|---|---|
| $^{235}U(n,\gamma)$ | 98 | 68 | $^{103}Rh(n,\gamma)$ | 150 | 113 |
| $^{235}U(n,f)$ | 520 | 400 | $^{105}Rh(n,\gamma)$ | 1.8E+04 | 1.2E+04 |
| $^{238}U(n,\gamma)$ | 2.73 | 2 | $^{135}Xe(n,\gamma)$ | 3.6E+06 | 2.2E+06 |
| $^{238}U(n,f)$ | 0 | 8E-06 | $^{147}Pm(n,\gamma)$ | 235 | 127 |
| $^{237}Np(n,\gamma)$ | 170 | 153 | $^{149}Sm(n,\gamma)$ | 4.15E+04 | 5.5E+04 |
| $^{237}Np(n,f)$ | 0.019 | 0.013 | $^{150}Sm(n,\gamma)$ | 102 | 72 |
| $^{239}Pu(n,\gamma)$ | 632 | 360 | $^{151}Sm(n,\gamma)$ | 1.5E+03 | 8.3E+03 |
| $^{239}Pu(n,f)$ | 1520 | 750 | $^{152}Sm(n,\gamma)$ | 210 | 150 |

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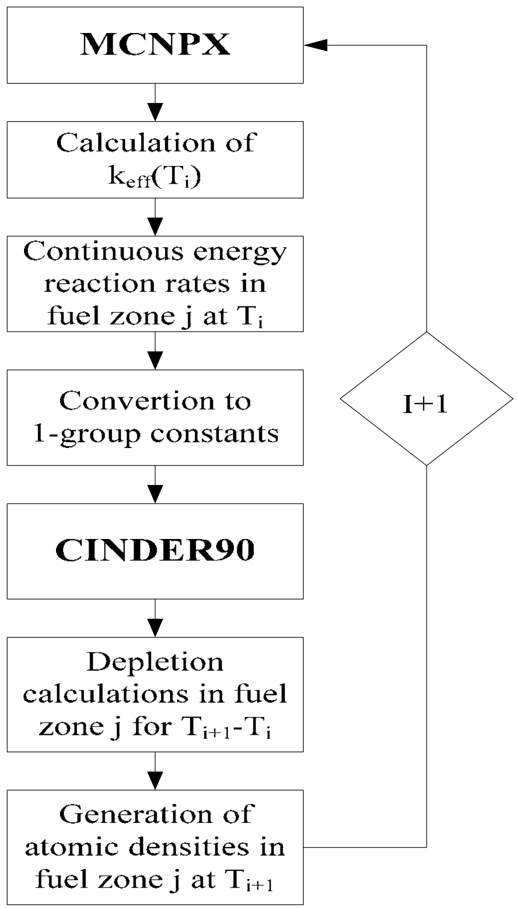
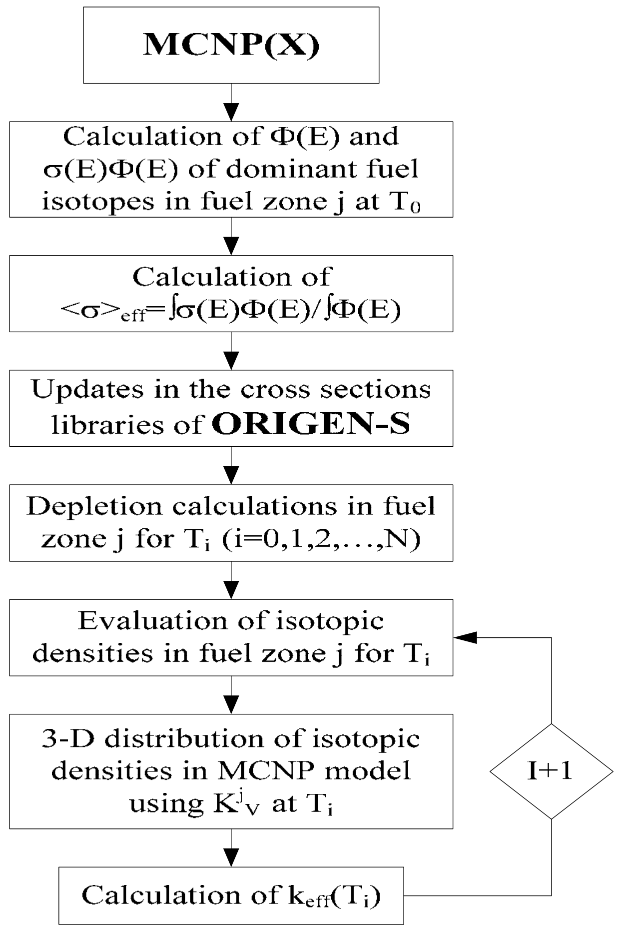
# Comparison of depletion methodologies

## MCNPX 2.6.C




## MCNPX&ORIGEN-S

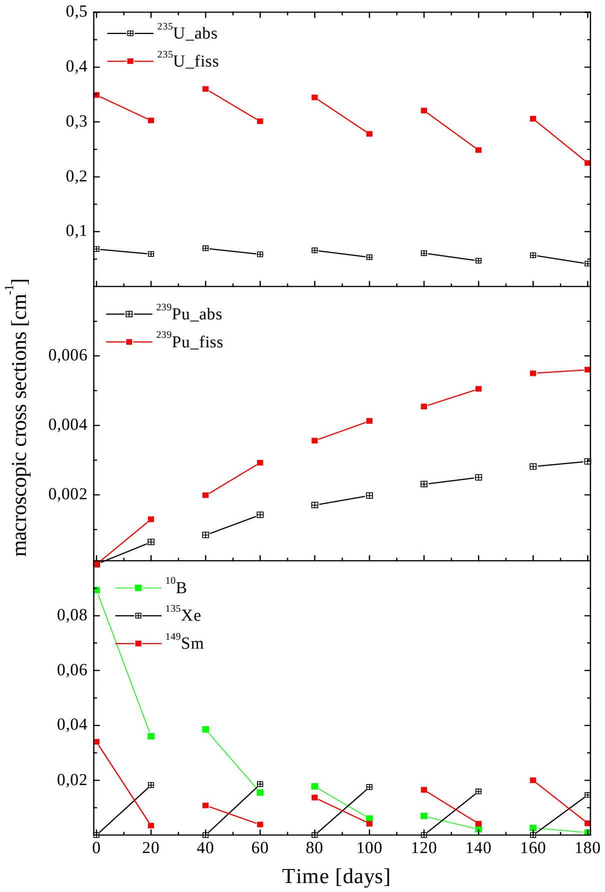


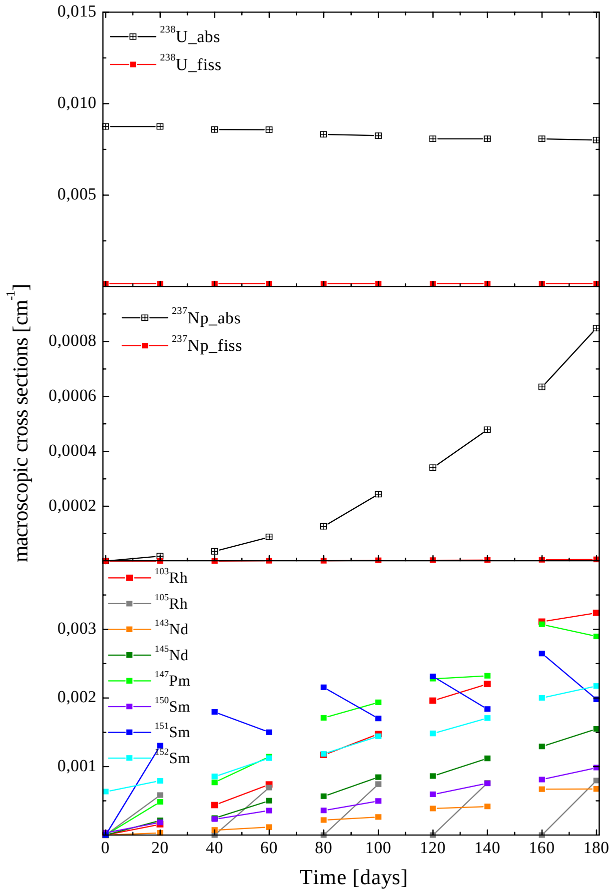
# Testing of MCNPX 2.6.C on the Research Reactor BR2

- Evolution of macroscopic cross sections by MCNPX 2.6:

Automatic calculations at each time step:

- reaction rates
- power distribution
- isotopic fuel densities
- burnup
- activity



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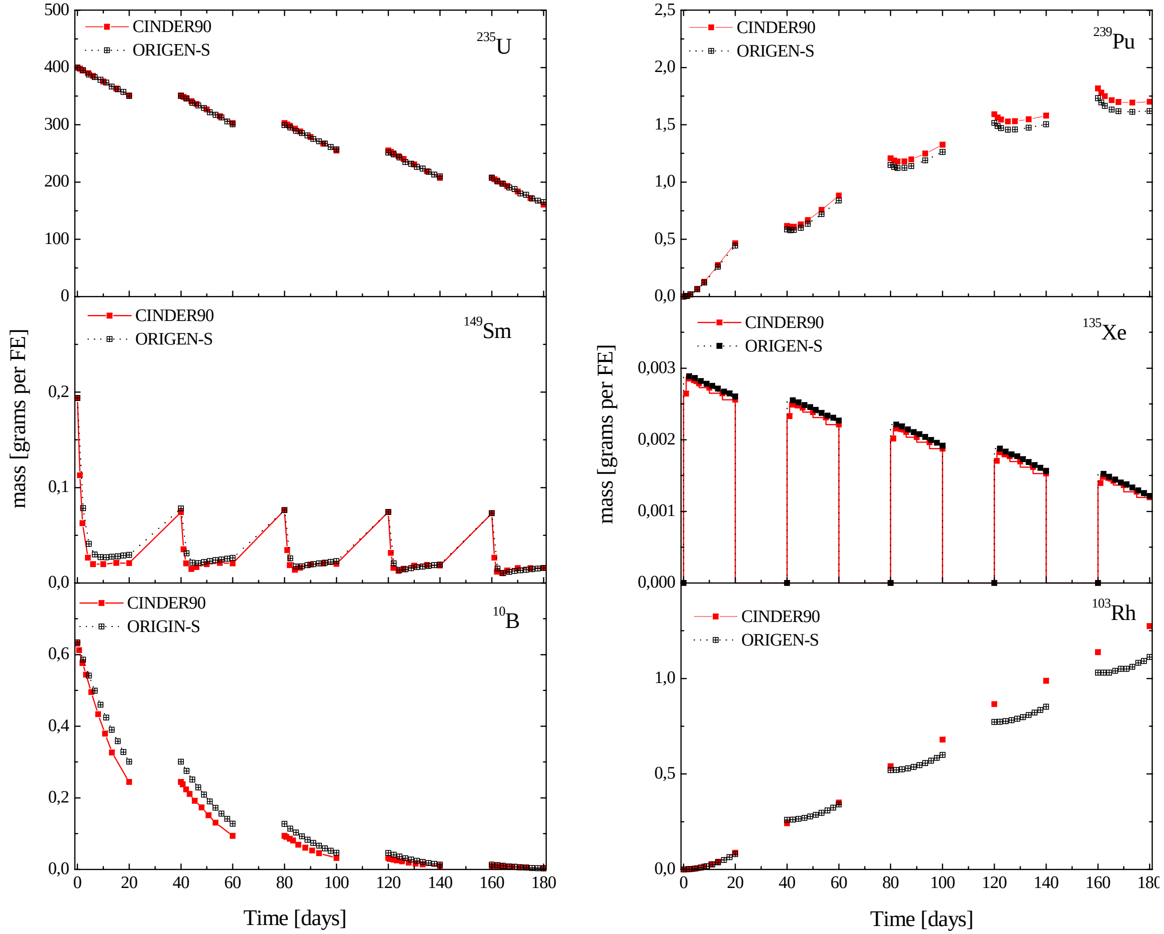
# Testing of MCNPX 2.6.C on the Research Reactor BR2

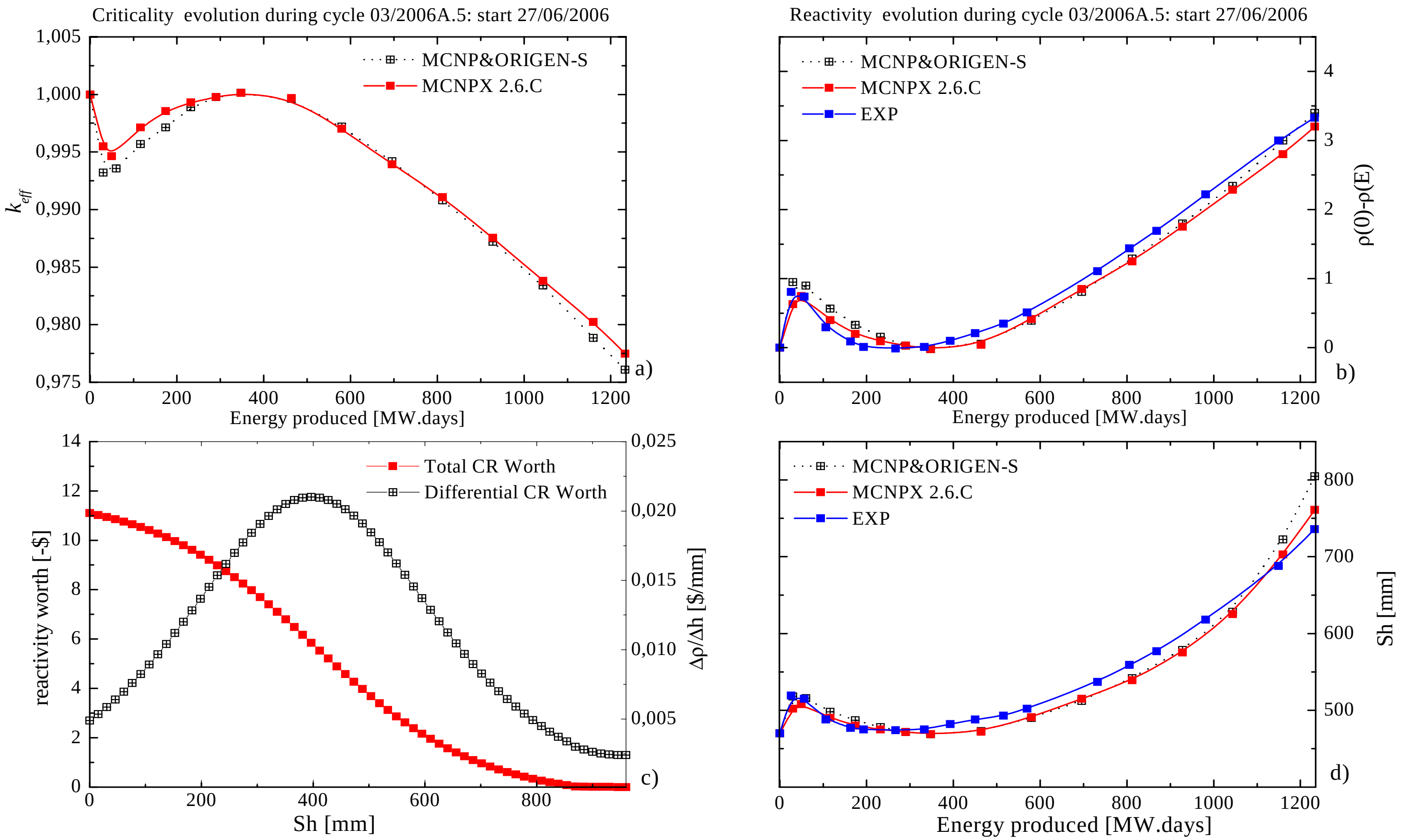
- **Comparison of the atomic densities by CINDER90 and ORIGEN-S:**

Good agreement for main fissile nuclides and dominant F.P.



# Testing of MCNPX 2.6.C on the Research Reactor BR2

- **Criticality calculations by MCNPX 2.6 and MCNP&ORIGEN-S**



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# Conclusions – common features

- The capabilities for depletion and criticality reactor core analysis of the new burnup Monte Carlo code MCNPX 2.6.C are compared the MCNPX&ORIGEN-S method

- Both methods use the same Monte Carlo code, which is linked with a 1 – D depletion code: CINDER90 in MCNPX 2.6 and ORIGEN-S in the MCNP(X)&ORIGEN-S method.

- In the both methods the reaction rates are calculated by MCNP(X) and the one – group constants are introduced into the depletion equation.

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# Conclusions – differences

- ***MCNPX 2.6.C***: the whole process is ***automatic*** and the steady – state flux calculations by MCNPX are ***internally linked*** with the depletion calculations by CINDER90
  - the reaction rates are updated for each time step in the requested fuel region during the irradiation period

- ***MCNP(X)&ORIGEN-S method***: the reaction rates are calculated by MCNP(X) ***once*** – at BOC and introduced into ORIGEN-S, which performs the depletion calculations for ***all desired time steps***
  - Then the isotopic fuel composition for a given time step is introduced back into the MCNP geometry model and distributed in the core using the calculated earlier 3 – D power peaking factors

# Conclusions –
# advantages and disadvantagess

- *MCNPX 2.6.C*:
  - the number of the spatial fuel zones, which can be depleted is still limited by the allowed computer memory
  - Easy for use

- *MCNPX&ORIGENS*:
  - The number of the fuel depletion zones used in the MCNP&ORIGEN-S method is unlimited (used about 4000 fuel cells)
  - Higher accuracy in criticality calculations vs. MCNPX 2.6.C