Report: Depletion Model

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May 23, 2016

1 Introduction

In the operation of a nuclear reactor, macroscopic cross sections are of major importance in order to assert the multiplication factor and to maintain steady state condition.A macroscopic cross section can be defined for each nuclear reactor x and each isotope i and it is composed by the product of microscopic cross sections and of the concentrations of the considered isotopes:

$$\Sigma(E, \vec{r}, t) = N(\vec{r}, t)\sigma(E, \vec{r}, t)$$

where $\vec{r}$ is the position vector, E is the energy of the incident neutron and t underlines the dependence of the two physical quantities from the time.
The time, indeed, plays an important role in reactor physics: while the reactor is operating the isotopes concentrations vary and this leads in a variation of the operation conditions, which results in the variation of the microscopic cross sections too. These variations are related to local and global parameters that are not necessarily the burnup, as:

- Fuel temperature
- Moderator or coolant temperature
- Moderator or coolant density
- Poison load
- Moderator purity

In this report the attention is focused in the variations of the isotope concentrations.The parameter of interest is the burnup. The Batemann equations, or Depletion equations, must be introduced in order to guarantee an appropriate discussion.

1.1 Depletion Equations

The variation of an isotope is equal to the difference between the rates of its production and of its transmutation, due an absorption reaction or to a spontaneous decay. The source term is:

$$S_k(t) = \sum_{l=1}^{L} Y_{k,l} < \sigma_{f,l}(t)\Phi(t) > N_l(t) + \sum_{l=1}^{k} \lambda_{l\to k}(t)N_l(t)$$

and the transmutation term is equal to

$$\Lambda_k(t) = \lambda_k + < \sigma_{a,k}\Phi(t) >$$

where the cross section and the flux are integrated all over the energy domain and space. Their product, being related to the reaction rate, is considered to variate linearly in the calculation stage. The first order system of K equation is

$$\frac{dN_k}{dt} + \Lambda_k(t)N_k(t) = S_k(t)$$

for k=1,...,K.
The normalization of the flux Φ is of major importance in our calculations. The **constant power depletion** case is assumed. The power released will be kept the same at the beginning and at the end of the calculation stage. It's not guaranteed to remain the same in every point of the stage.
In order to solve this system a 4^{th} order Kaps-Rentrop algorithm is used. This algorithm is implemented in the EVO: module of the DRAGON nuclear code. This code is a lattice code developed at the Polytechnique de Montréal. Further calculations performed with this lattice code will be defined as "reference" calculations. It is worth, so, spending few words on it in the next paragraph.

1.2 DRAGON lattice code

The DRAGON code solves the Boltzmann transport equations. The Boltzmann equation is the following:

$$\vec{\Omega} \cdot \vec{\nabla}\Phi(\vec{r}, E, \vec{\Omega}) + \Sigma(\vec{r}, E)\Phi(\vec{r}, E, \vec{\Omega}) =$$
$$Q^{scatt}(\vec{r}, E, \vec{\Omega}) + Q^{fiss}(\vec{r}, E, \vec{\Omega}) = Q(\vec{r}, E, \vec{\Omega})$$

where $\Phi(\vec{r}, E, \vec{\Omega})$ is the neutron angular flux.
In the assumption of isotropic materials:

$$Q^{scatt}(\vec{r}, E, \vec{\Omega}) = \int_{4\pi} d^2\Omega' \int_0^{+\infty} dE' \Sigma_s(\vec{r}, E \leftarrow E', \vec{\Omega} \cdot \vec{\Omega}')\phi(\vec{r}, E', \vec{\Omega}')$$

where a Legendre polynomial expansion can be performed on scattering cross section.
The fission source term is considered isotropic:

$$Q^{fiss}(\vec{r}, E, \vec{\Omega}) = \frac{1}{4\pi k_{eff}} \sum_{j=1}^{J} \chi_j(E) \int_0^{+\infty} dE' \nu\Sigma_{f,j}(\vec{r}, E')\phi(\vec{r}, E')$$

where k_{eff} is a factor used to divide the fission source term and maintain steady-state condition.
The unknowns of the Boltzmann equation are the three component of the space vector $\vec{r}$, the energy and the two component of the solid angle $\vec{\Omega}$. In order to reduce the number of unknowns, a multigroup approach is used. This approach results in the utilization of energy averaged quantities. Consequently, G energy groups create a system of G Boltzmann equations whose properties are energy condensed. The G transport equations are:

$$\vec{\Omega} \cdot \vec{\nabla}\Phi_g(\vec{r}, \vec{\Omega}) + \Sigma_g(\vec{r})\Phi_g(\vec{r}, \vec{\Omega}) = Q_g(\vec{r}, \vec{\Omega})$$

for $1 \leq g \leq G$.
Defining the lethargy $u_g = ln\left(\frac{E_0}{E_g}\right)$, averages are performed. The quantities of interest are

$$\Sigma_{x,g}(\vec{r}, \vec{\Omega}) = \frac{1}{\Phi_g(\vec{r}, \vec{\Omega})} \int_{u_{g-1}}^{u_g} \Sigma_x(\vec{r}, u, \vec{\Omega})\Phi(\vec{r}, u, \vec{\Omega})$$
$$\Sigma_{s,g\leftarrow h} = \frac{1}{\Phi_h} \int_{u_{g-1}}^{u_g} du \int_{u_{h-1}}^{u_h} du' \Sigma_s(\vec{r}, u \leftarrow u', \vec{\Omega} \cdot \vec{\Omega}')\phi(\vec{r}, u, \vec{\Omega})$$
$$\Phi_g(\vec{r}, \vec{\Omega}) = \int_{u_{g-1}}^{u_g} du\Phi(\vec{r}, u, \vec{\Omega})$$

In the next calculations 172 groups have been used (JEFF2.2 DRAGLIB, cross section library).
Once that the flux is evaluated over the region of interest, that can be a cell or an assembly, it is used to evaluate spatial averaged quantities. The idea it is to preserve the reaction rate and to create a list of only one value for each quantity in the unit geometry (Space Homogenization).

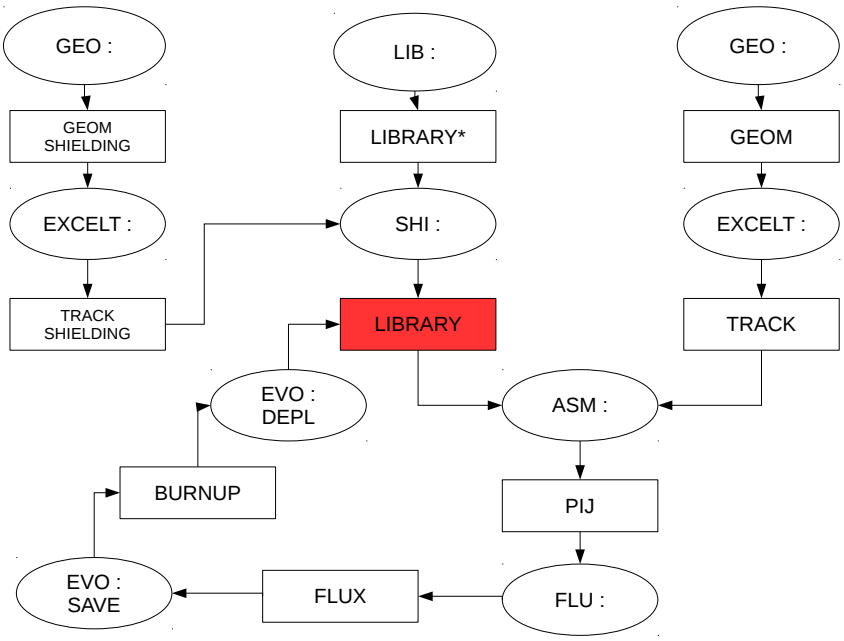

Figure 1: Scheme for reference calculations

This idea is at the base of full core codes as DONJON. This codes use a nodal approach.
Along with the space homogenization energy condensation is performed. In fact, in full core codes less energy groups are used.
This condensed and homogenized quantities are stocked in MULTICOMPO. They are stocked for different local and global parameters. In the calculation presented are stocked for different burnup conditions. **Depletion equations are solved in lattice calculations and the results stocked in the MULTICOMPO.** It will possible to notice that all the models treated in this paper will take into account the MULTICOMPO to extract microscopic cross section data, but, whereas macrodepletions models extract even concentration data, microdepletion models do not and solve depletion equations in a separated way.
The models presented use two MULTICOMPO: one for the moderator which does not stock burnup information, the other for the fuel which does.

1.3 Reference Calculations

In figure 1 it is possible to observe the scheme for reference calculations. Two geometries are created with the GEO: module and then tracked using EXCELT: . In the meantime a microscopic cross section library is created using the LIB: module. Self-shielding is performed using the SHI: module. The library and the tracked geometry is used to assembly the collision probability matrix (ASM:) and then the flux is evaluated (FLU:). The EVO: module is then used. At first it is used to save the proper data in the BURNUP LCM object, then is used to actually solve the depletion equation and to modify the MICROLIB. This modification allows to evaluate again the flux for the following time step.

1.4 DONJON full-core code

The models presented use DONJON. This code solves the diffusion equation:

$$-\vec{\nabla} \cdot (D_g(\vec{r})\vec{\nabla}\Phi_g(\vec{r})) + \Sigma_g(\vec{r})\Phi_g(\vec{r}) = Q_g(\vec{r})$$

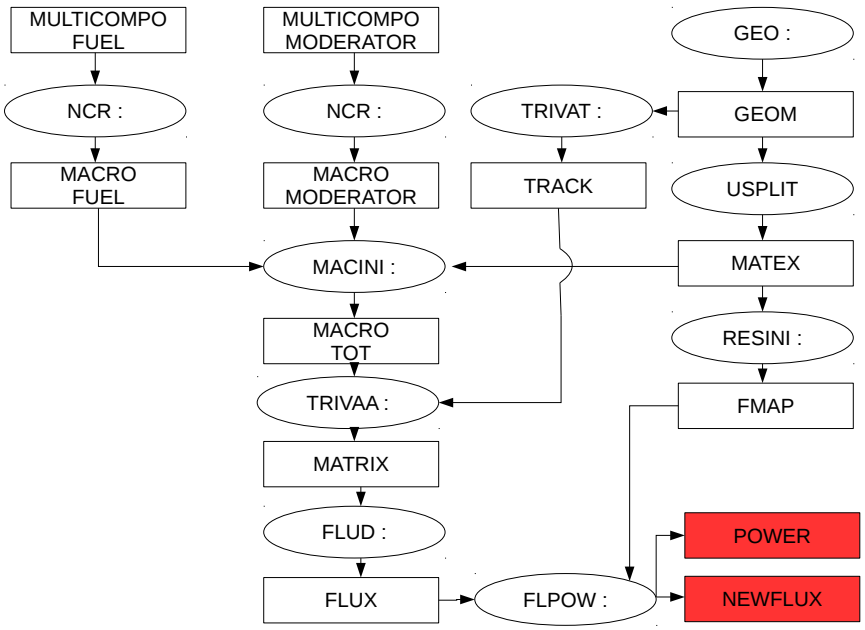

Figure 2: Common general scheme for presented models

where $D_g(\vec{r})$ is the diffusion coefficient. This coefficient is introduced once that this heuristic relationship, called Fick's Law, is used to relate the integrated flux to the integrated current density:

$$\vec{J}_g(\vec{r}) = -D_g(\vec{r})\vec{\nabla}\Phi_g(\vec{r})$$

As already discussed, a nodal approach is used. Also an energy condensation is performed. In the presented calculations two groups have been used. The energy condensation and the space homogenization are applied in the following equations

$$\bar{\Sigma}_{x,i,g} = \frac{\int_{V_i} dV \int_{E_g} dE N_i(\vec{r})\sigma_{x,i}(\vec{r},E)\Phi(\vec{r},E)}{\int_{V_{merg}} dV \int_{E_g} dE \Phi(\vec{r},E)}$$

$$\bar{N}_i = \frac{\int_{V_i} dV N_i(\vec{r})}{\int_{V_{merg}} dV}$$

$$\bar{\sigma}_{x,i,g} = \frac{\bar{\Sigma}_{x,i,g}}{\bar{N}_i}$$

defining V_{merg} the volume after the space homogenization, and V_i the actual volume occupied by isotope i. These definitions highlight how the macroscopic cross sections of the isotope i depend upon the concentration N_i on the transport calculations ($DRAGON$) and on the concentration $\bar{N}_i$ in the diffusion calculations ($DONJON$).

In figure 2 it is possible to observe the scheme for the common part of the models that will be introduced in the paper.

Two MULTICOMPO, one for the fuel and one for the moderator, are used to extract MACROLIB with the aid of the NCR: module. In the meantime a geometry is described using the GEO: module. Using the USPLIT: module a material map (MATEX) is created. This map contains all the information to merge together the two MACROLIB in a single one using the MACINI: module. With the TRIVAA: module, this new MACROLIB is used to assembly a proper matrix with the tracking information. The flux can be evaluated using FLUD: . Once that the flux is evaluated

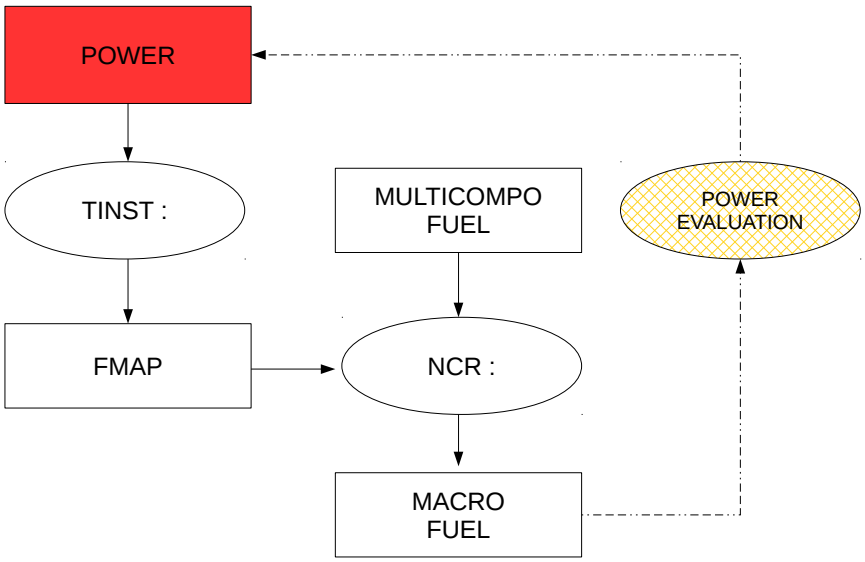

Figure 3: Scheme for TINST macrodepletion model

is normalized using the FLPOW: module. It takes as input a fuel map (FMAP) evaluated with the RESINI: module. This map contains the burnup information for each bundle and will be of major importance. The information about the power and the normalized flux will be used in the presented models.

2 Macro Depletion models

These models do not solve the depletion equations. They extract the needed information from the MULTICOMPO created using DRAGON lattice code. Both the concentrations and the microscopic cross sections are extracted. Two models will be presented: the TINST macrodepletion model and the XENON macrodepletion models.

2.1 TINST macro depletion model

In figure 3 the scheme for TINST macrodepletion model is presented. The TINST: module perform an **interpolation** from the MULTICOMPO database. The Burnup at time $t+\Delta t$, used to extract the information of interest, is evaluated, for fuel bundle j in channel i, with the formula

$$B_{i,j}(t+\Delta t) = B_{i,j}(t) + \frac{P_{i,j}(t)\Delta t}{m_{i,j}}$$

where $m_{i,j}$ is mass of the initial heavy elements.
The fuel map contains the burnup information for each considered bundle and allows to extract the information of interest from the MULTICOMPO of the fuel using the NCR: module. The MACROLIB of the fuel is then merged using the MACINI: module and the flux is evaluated again as already described.

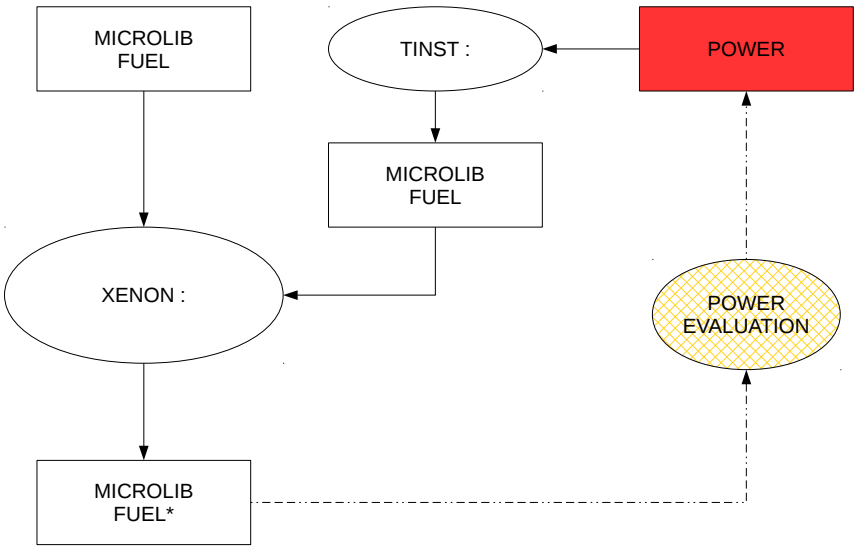

Figure 4: Scheme for XENON macrodepletion model

2.2 XENON macro depletion model

The model in figure 4 is similar to the previous one with the only difference that the MICROLIB of the fuel is modified by XENON: module which uses the power and the fuel map objects. This module computes the equilibrium concentrations using the considered bundle flux ϕ:

$$N_{Xe_{eq}} = \frac{(Y_I + Y_{Xe})\Sigma_f \phi}{\lambda_{Xe} + \sigma_{Xe}\phi}$$

where Σ_f is the fission total macroscopic cross section, Y_I and Y_{Xe} are respectively the fission yield for I135 and Xe135, λ_{Xe} is the decay constant of Xe135 and σ_{Xe} is the capture microscopic cross section.
At each time the flux is modified and the Xe135 concentration is evaluated again. The flux is modified again. An iterative procedure is set. Its convergence is assumed if

$$|k_{eff}^{i+1} - k_{eff}^{i}| < 0.01$$

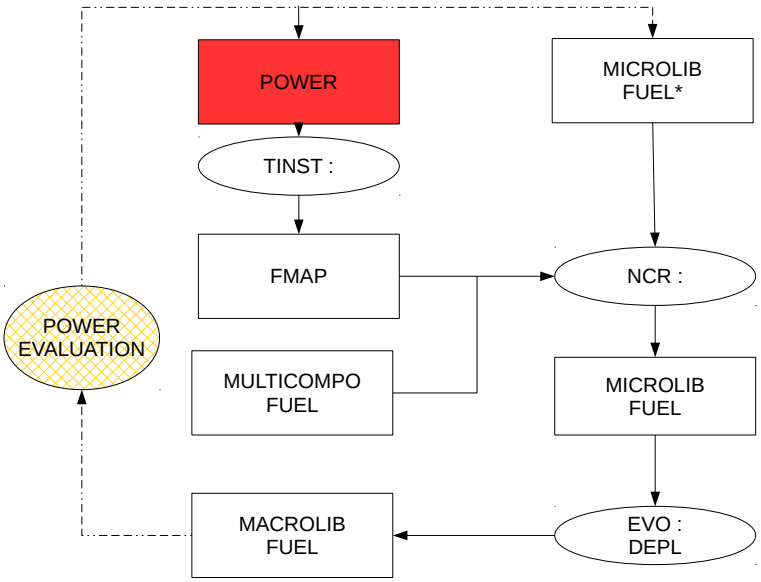

Figure 5: Scheme for FMAP micro depletion model

3 Micro Depletion models

Depletion equations can be solved also in full core diffusion calculations (DONJON). In this case, assuming $\vec{L}$ to be a state vector for the fuel bundle:

$$\Sigma_x(\vec{L}) = \Sigma_x(\vec{L}_0) + \sum_{l_c} \Delta\Sigma_x^l + \sum_k (N_k - N_{0,k})\sigma_{x,k}(t)$$

The variations of the concentrations due to burnup are separated from other causes. The other components of the state vectors can be supposed independent one from the other. Depletion equations are solved and their results are integrated with the MULTICOMPO database. Once that the parameters are defined for each fuel bundle, the depletion can be solved, even in DONJON, using the 4^{th} order Kaps-Rentrop algorithm implemented in the EVO : module.
Two models will be presented: the FMAP micro depletion model and the PICK micro depletion model.

3.1 FMAP micro depletion model

In the FMAP micro depletion model represented in figure 5 the POWER LCM object is used to create a fuel map (FMAP) with the module TINST: . This map is used with the NCR: module to extrapolate the right information from the MULTICOMPO and to modify the MICROLIB. Once that the MICROLIB is modified the EVO: module is applied to evaluate the new isotope concentration and to create the MACROLIB of the fuel to be used in the power calculations. A new FMAP will be created for the following time step.

3.2 PICK micro depletion model

In the PICK micro depletion model represented in figure 6 the fuel map is not used. The microlib is modified using the EVO: module in order to evaluate the new isotope concentrations. From the EVO: module an average burnup value is extracted and using it the MACROLIB is formed

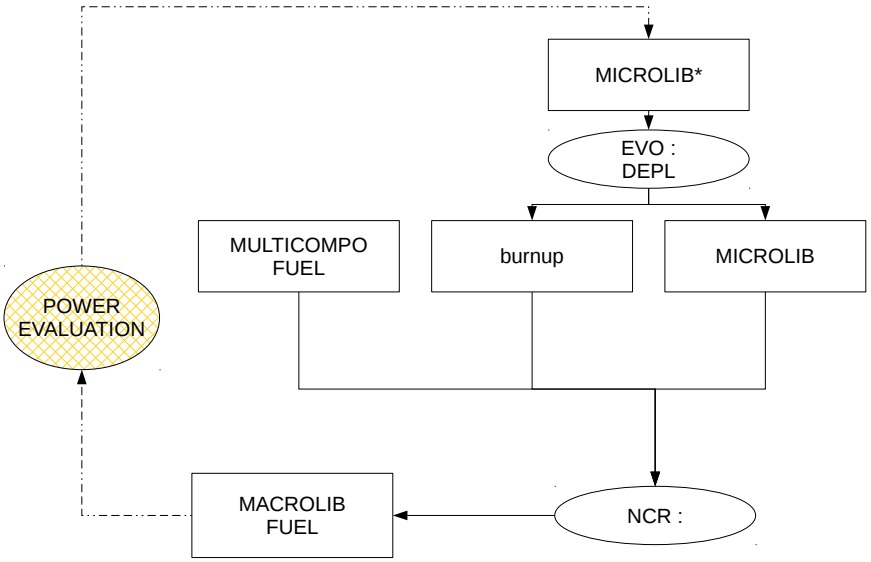

Figure 6: Scheme for PICK micro depletion model

extracting the microscopic cross section with the NCR: module.
Not having a fuel map this module has got a great limitation in case of power heterogeneities. In fact it extracts the cross section for every bundle using an average value of the burnup. However, the simplicity is its major quality. Now, the results of the four models will be presented for different geometries.

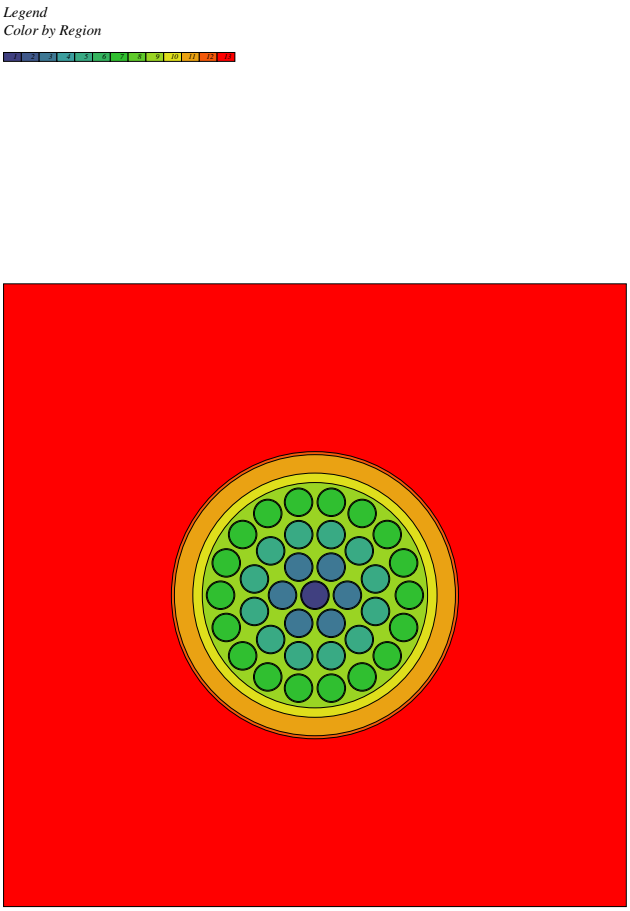

Figure 7: Comparison of keff at 100%

4 Single Cell

4.1 keff

In the figure 7 the cell to be studied is shown. Dragon calculation for a single cell is performed. Two MULTICOMPO are created, one for the fuel (BURNUP data are present) and one for the moderator (no BURNUP data). The model studied are the macrodepletion one, using the TINST: module, the XENON one, which is a macrodepletion model correcting the xenon effect, using the XENON: module and two microdepletion model: both of them use the EVO: module to evaluate the isotope concentration but while one uses a FMAP to describe the fuel irradiation the other uses the keyword PICK in the EVO: module to extract an overall burnup. In the case of a single cell the two model are equivalent.
After 150 days at nominal power (**31.9713 kW/kg ; 634 kW per cell**) - figure 8:

	keff 100%	dkeff [pcm]
reference	1.037356	–
micro FMAP	1.037575	21.9
micro PICK	1.036494	86.2
macro TINST	1.037423	6.7
macro XENON	1.038041	68.5

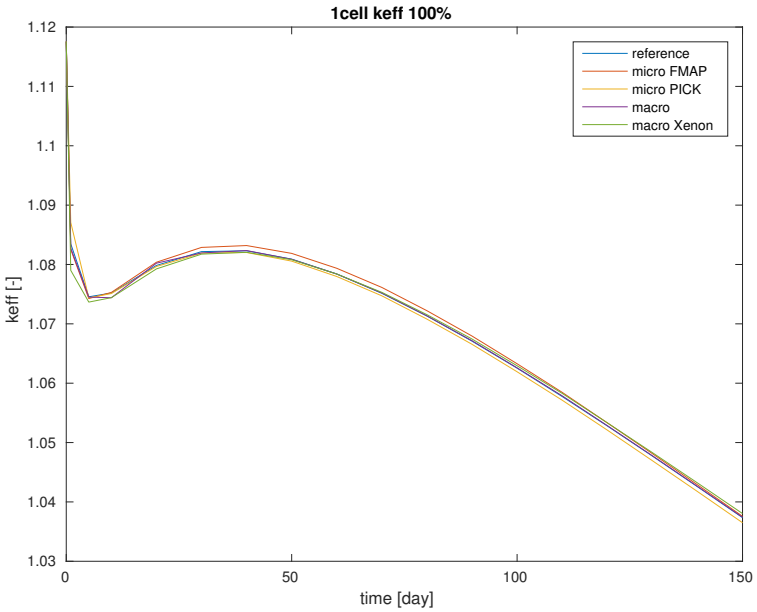

Figure 8: Comparison of keff at 100%

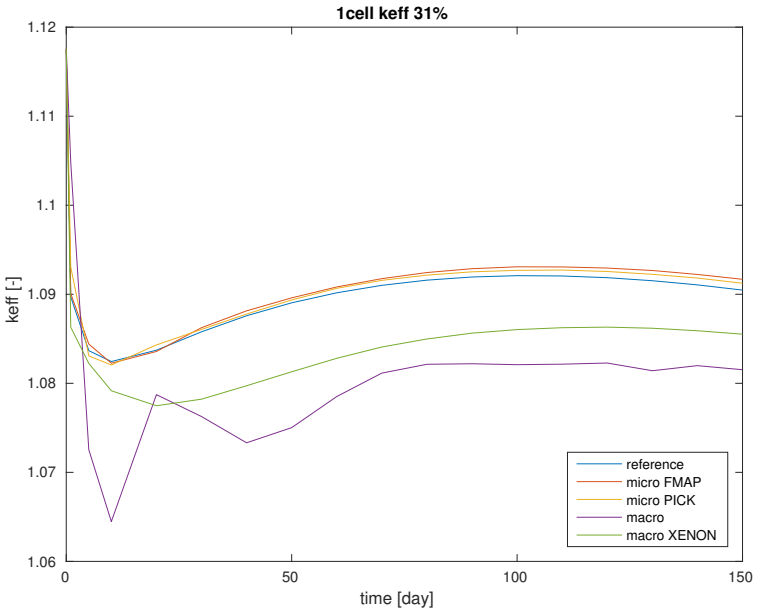

Figure 9: Comparison of keff at 31%

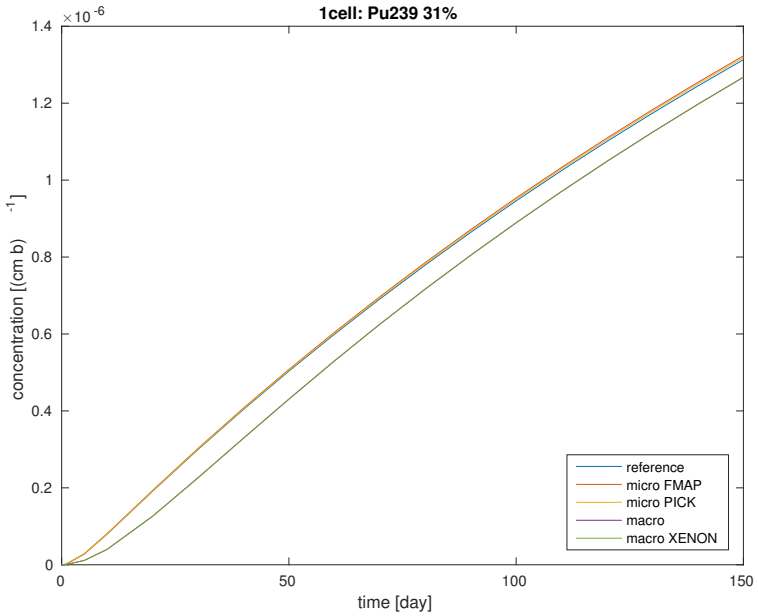

Figure 10: Pu239 concentration at 31%

after 150 days at 31% power - figure 9:

	keff 31%	dkeff [pcm]
DRAGON	1.090470	–
micro FMAP	1.091684	121.4
micro PICK	1.091241	77.1
macro TINST	1.081527	894.3
macro XENON	1.085518	495.2

where it is possible to notice the importance of solving the Batemann equations in non nominal condition. In nominal condition the TINST module is satisfying.
The two microdepletion models are not exactly the same, this is due to the fact that different calculation are performed.

4.2 Concentration

At nominal condition all the modules describes well the evolution of the 4 isotopes considered - figures 16, 17, 15 and 14. In 31% power condition the TINST module shows oscillation and not seem adequate because of a deviation - figures 12, 11, 10 and 13. TINST and XENON module do not describe properly the concentrations of Pu239 and Sm149 in 31% power condition -figures 10 and 11. However the TINST: module is ok in all condition to describe U235 concentration - figure 13. Indeed this isotopes is one of the most important in reactor physics and so, owe to its importance, the TINST: module treats it properly. The deviation is well compensated by the XENON module.

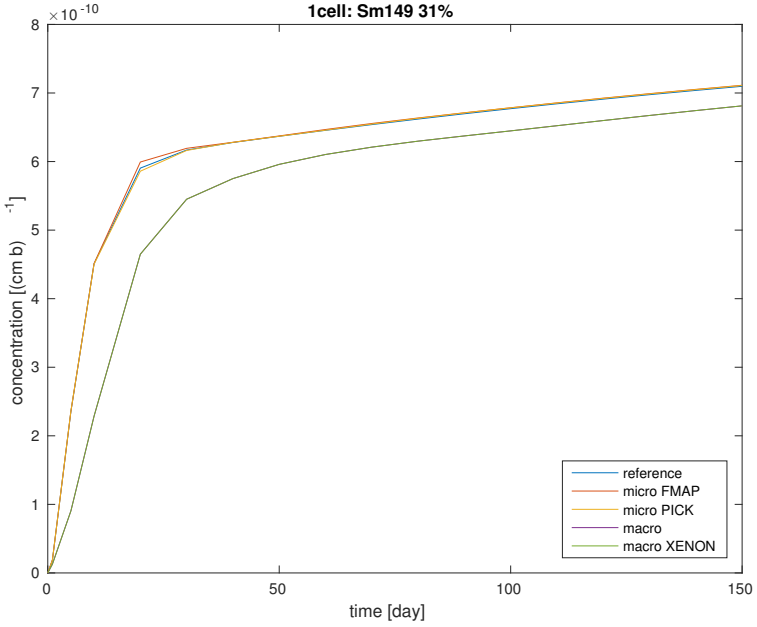

Figure 11: Sm149 concentration at 31%

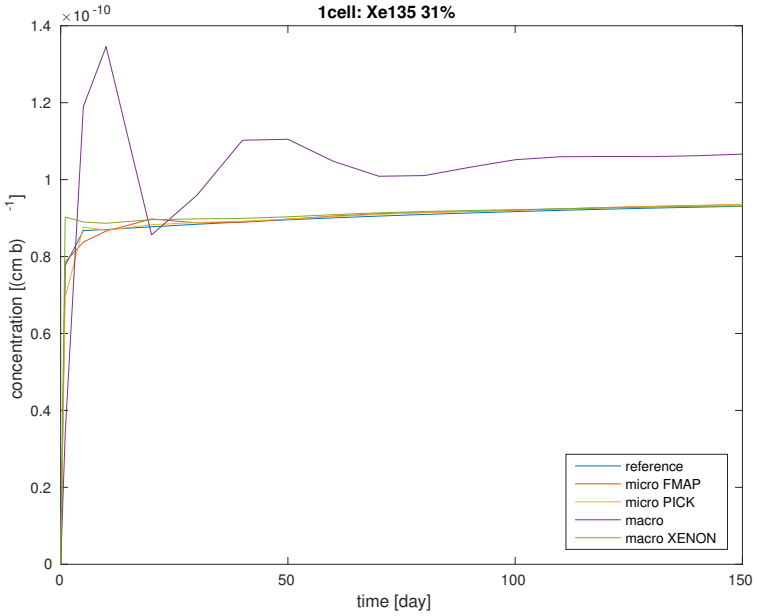

Figure 12: Xe135 concentration at 31%

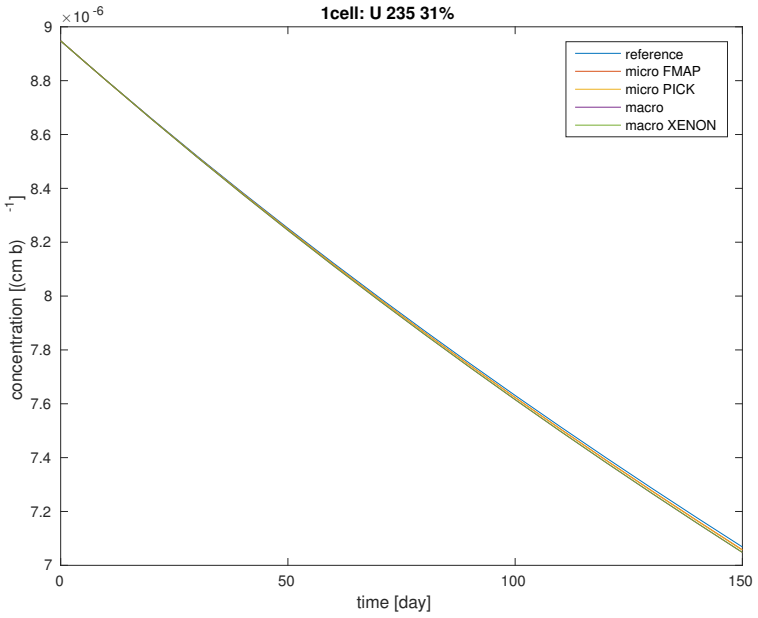

Figure 13: U235 concentration at 31%

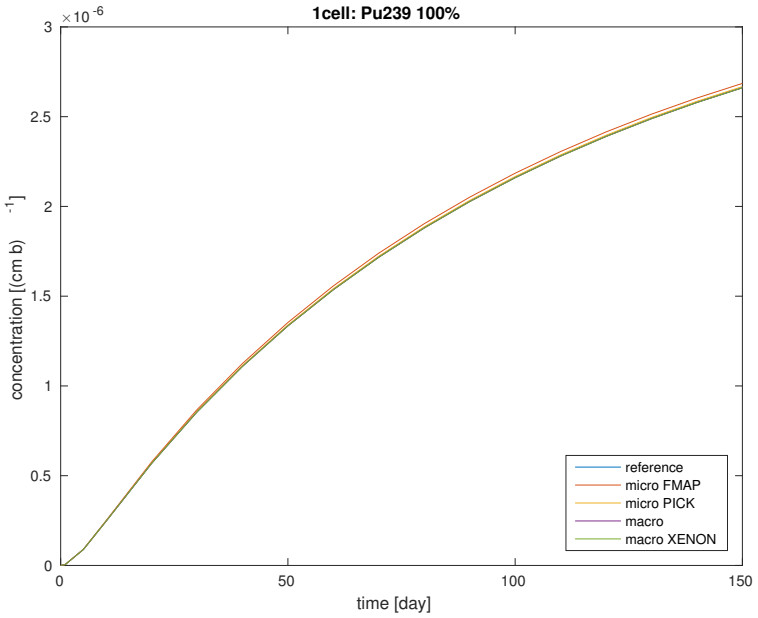

Figure 14: Pu239 concentration at 100%

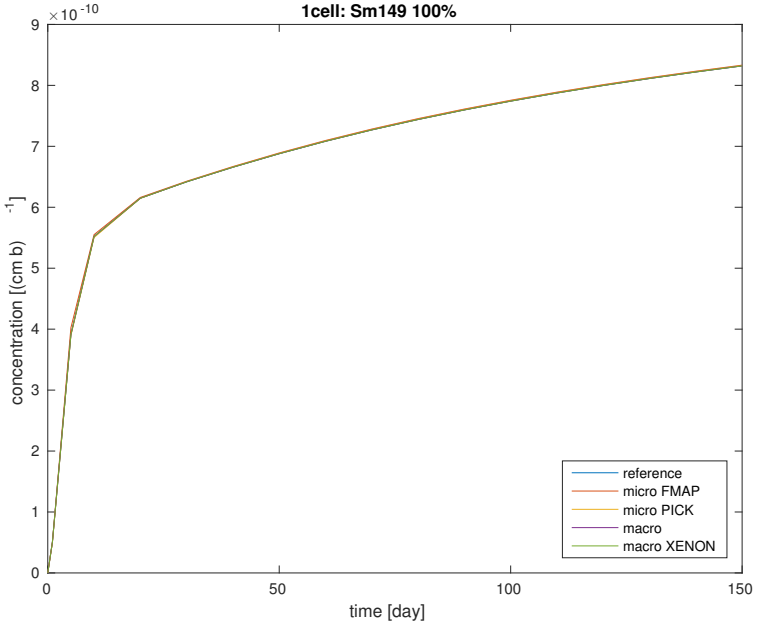

Figure 15: Sm149 concentration at 100%

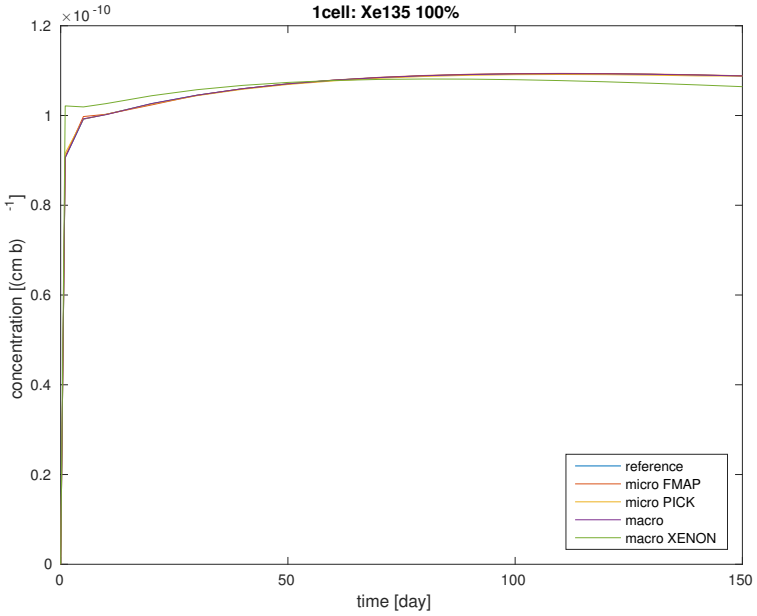

Figure 16: Xe135 concentration at 100%

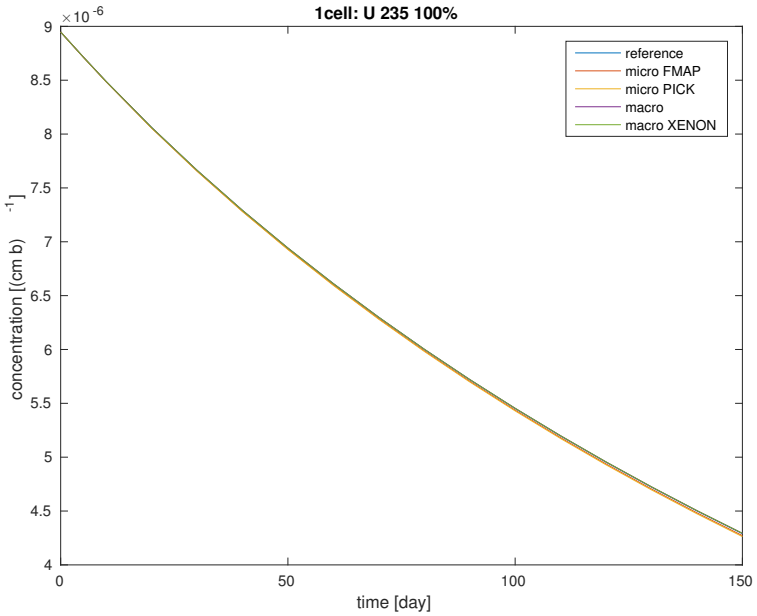

Figure 17: U235 concentration at 100%

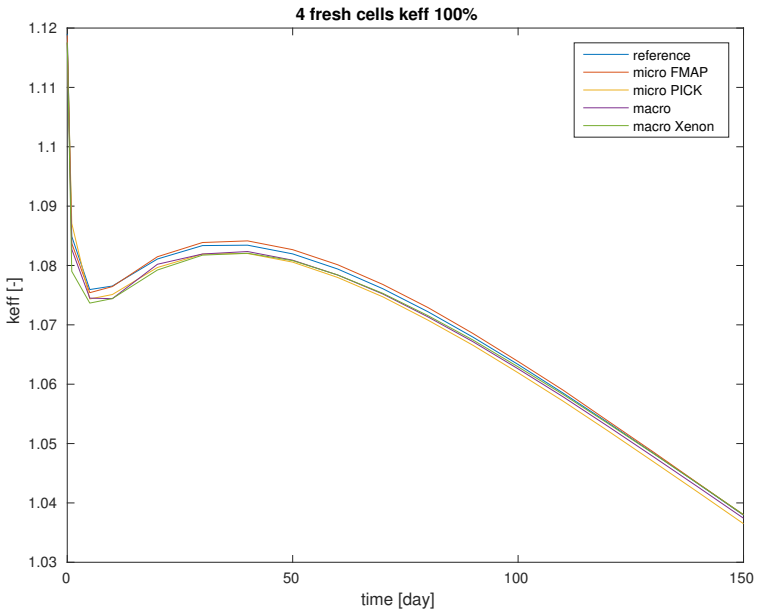

Figure 18: Comparison of keff at 100%

5 4 Cell no power heterogeneity

5.1 keff

The implemented geometry now is the one of 4 identical unit cell, all composed by fresh fuel. DRAGON calculations are compared to the two models of macrodepletion (TINST and XENON) and of microdepletion (FMAP and PICK), as a reference.
After 150 days of exploitation at nominal power - figure 18:

	keff 100%	dkeff [pcm]
DRAGON	1.037356	–
micro FMAP	1.037990	63.4
micro PICK	1.036495	86.1
macro TINST	1.037420	6.4
macro XENON	1.038036	68

and after 150 days of exploitation at 31% power - figure 19:

	keff 31%	dkeff [pcm]
DRAGON	1.090470	–
micro FMAP	1.092516	204.6
micro PICK	1.091243	77.3
macro TINST	1.081525	894.5
macro XENON	1.085509	496.1

It is possible to notice that the results are similar to the one already discussed in section 4.1. The models show the same trends already discussed. The microdepletion with the PICK keywords seems the best model in not nominal conditions.

5.2 Concentration

At nominal condition all the modules describes well the evolution of the 4 isotopes considered - figures 24, 25, 26 and 27. In 31% power condition the TINST module shows oscillation and not

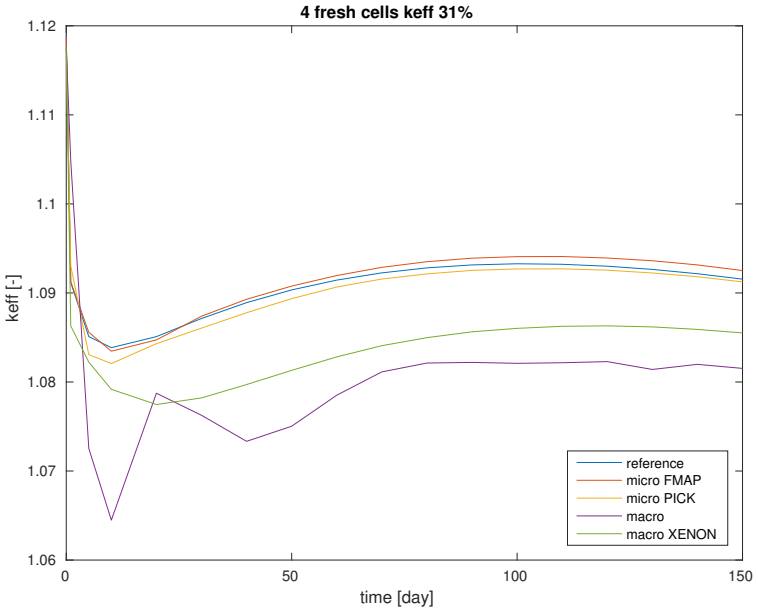

Figure 19: Comparison of keff at 31%

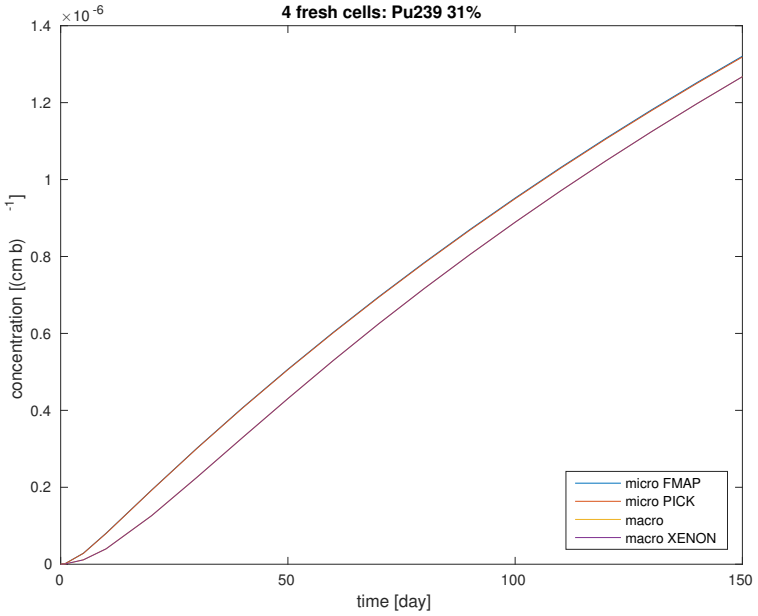

Figure 20: Pu239 concentration at 31%

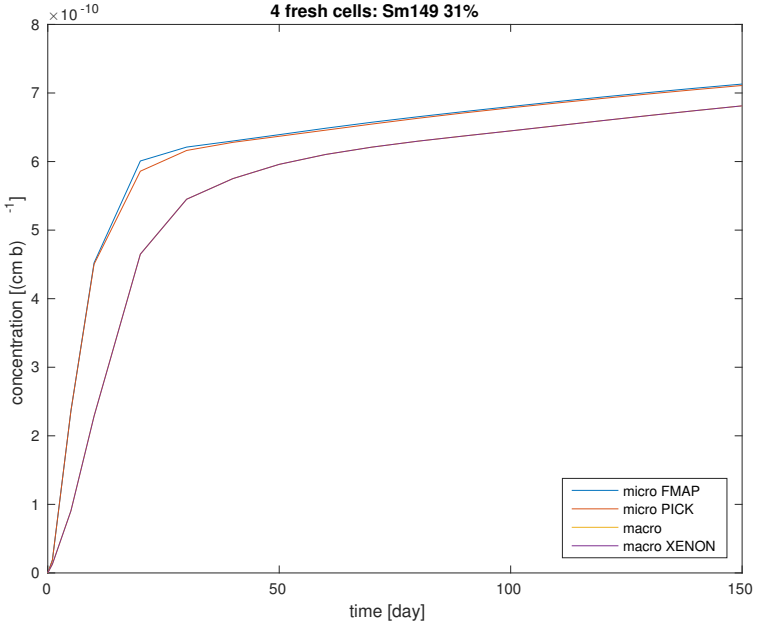

Figure 21: Sm149 concentration at 31%

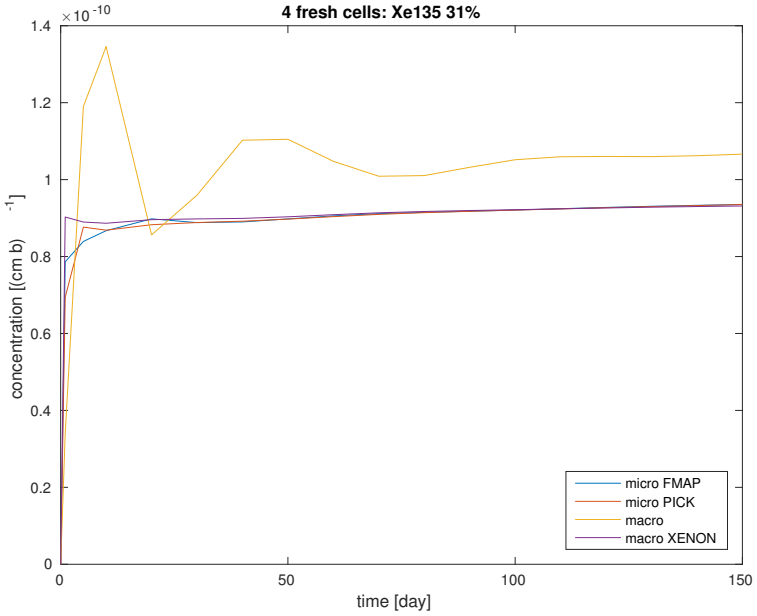

Figure 22: Xe135 concentration at 31%

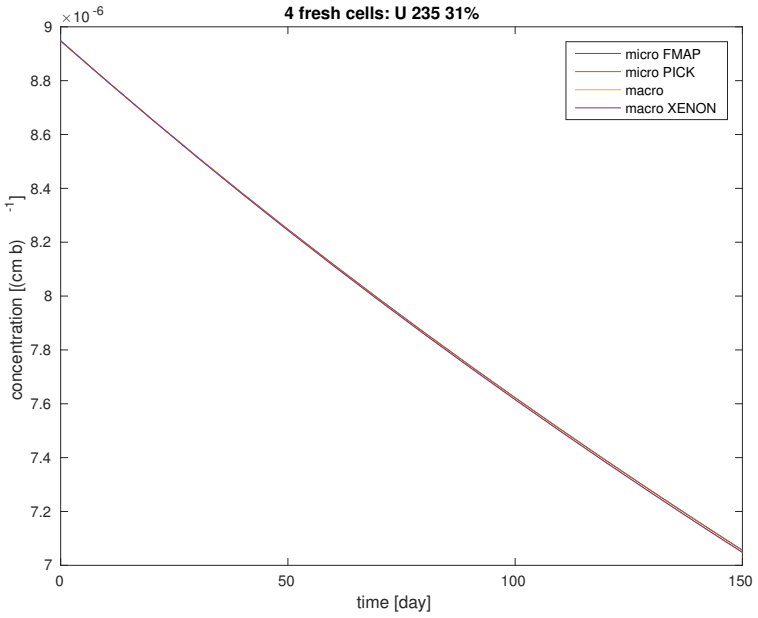

Figure 23: U235 concentration at 31%

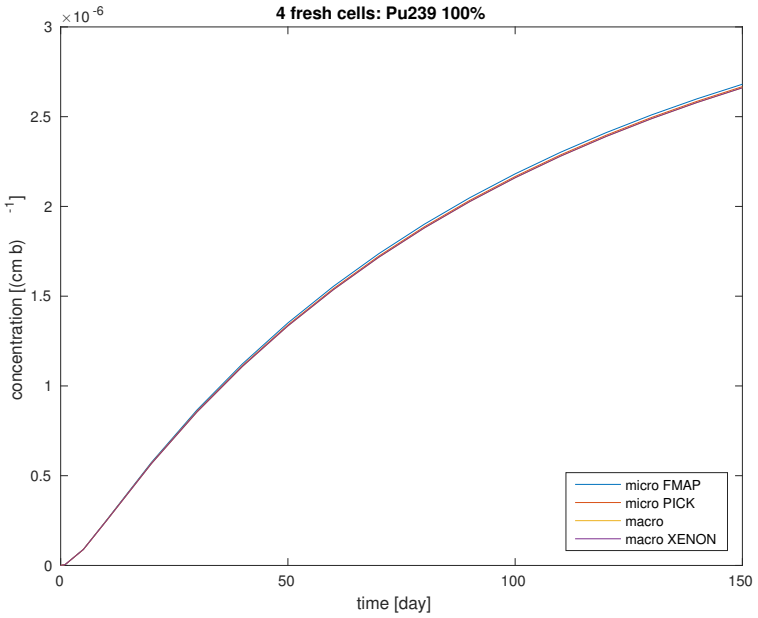

Figure 24: Pu239 concentration at 100%

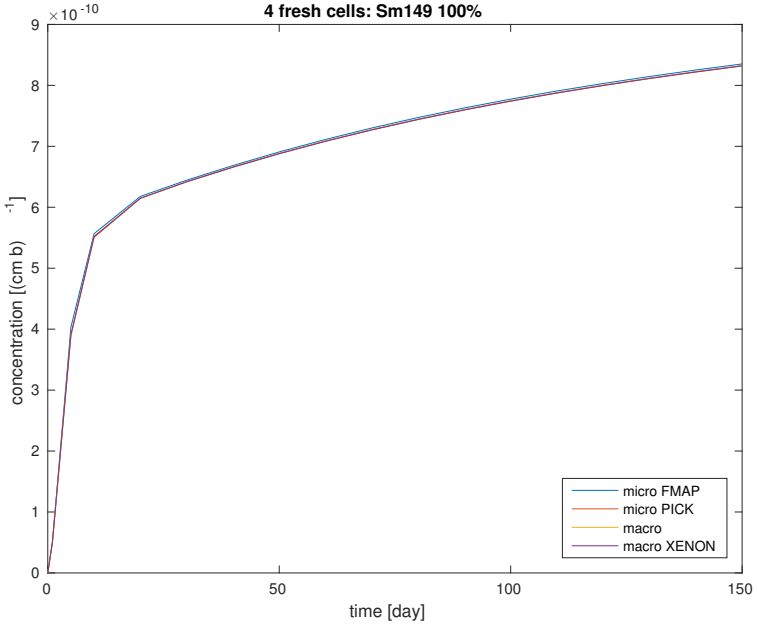

Figure 25: Sm149 concentration at 100%

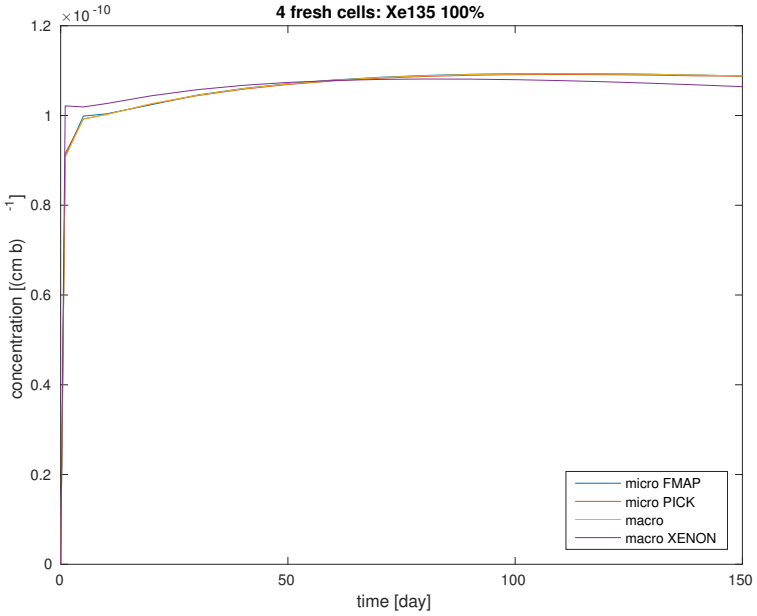

Figure 26: Xe135 concentration at 100%

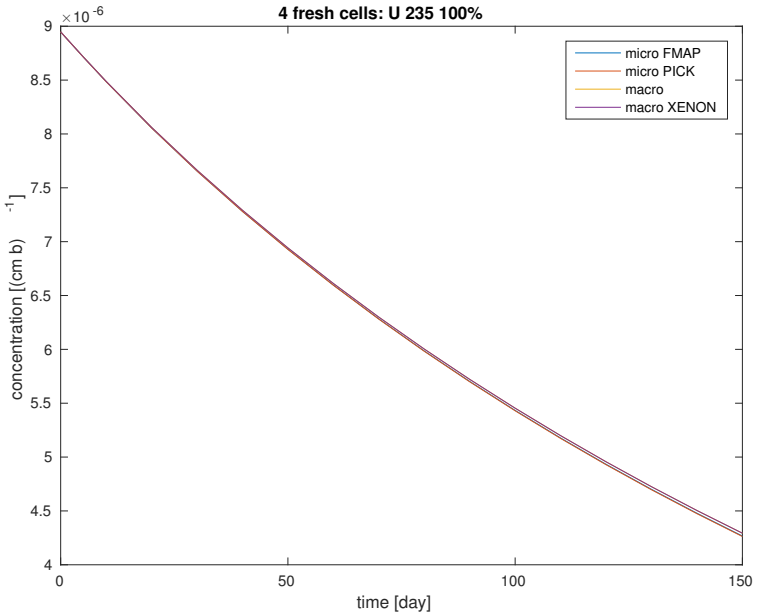

Figure 27: U235 concentration at 100%

seem adequate because of a deviation -figure 22. However the TINST: module is ok in all condition to describe U235 concentration - figure 23. Indeed this isotopes is one of the most important in reactor physics and so, owe to its importance, the TINST: module treats it properly. The deviation is well compensated by the XENON module for the Xe135, not for the Sm149 and Pu239 -figures 22, 21 and 20.

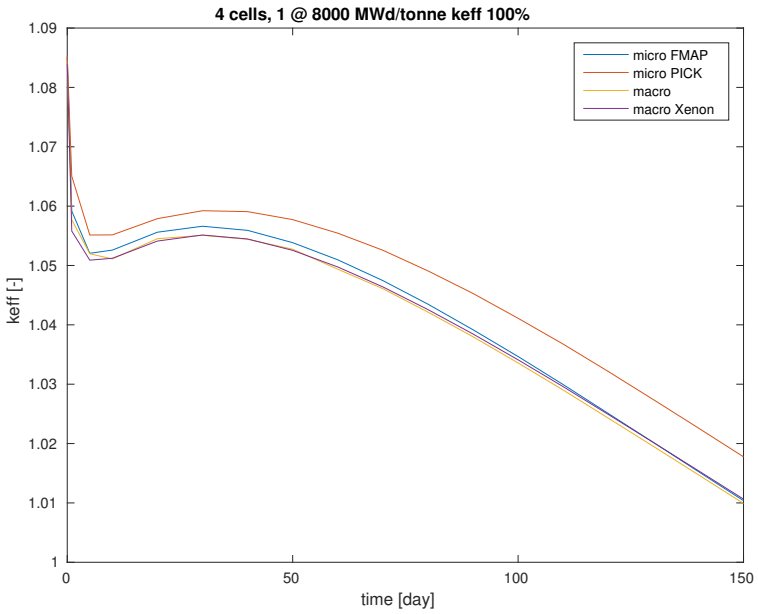

Figure 28: Comparison of keff at 100%

6 4 Cell Power heterogeneity

6.1 keff

The geometry implemented in this case is the one 4 unit cell. 3 of them are fresh, 1 is at 8000 Mwd/tonne. All the models are applied. No reference is present for this case.
After 150 days of operation at nominal condition - figure 28:

	keff 100%
micro FMAP	1.010378
micro PICK	1.017766
macro TINST	1.009912
macro XENON	1.010642

After 150 days of operation at 31% power - figure 29:

	keff 31%
micro FMAP	1.064239
micro PICK	1.068537
macro TINST	1.052582
macro XENON	1.057086

It is possible to notice that that the difference between the two microdepletion models becomes smaller and smaller as soon as the irradiation diminishes. The microdepletion model using the PICK keyword is not adapt to describe the evolution of the isotopes. In fact it overestimates the multiplicative factor. Not having a FMAP with the Burnup of each fuel bundle, in fact, is a limitation, because it means to interpolate the cross section data from a global value of the burnup.

6.2 Concentration

The figures 37, 36, 34 and 35 describe the isotope evolution in nominal condition. It is possible to notice how the PICK microdepletion model underestimate the production of the plutonium and the

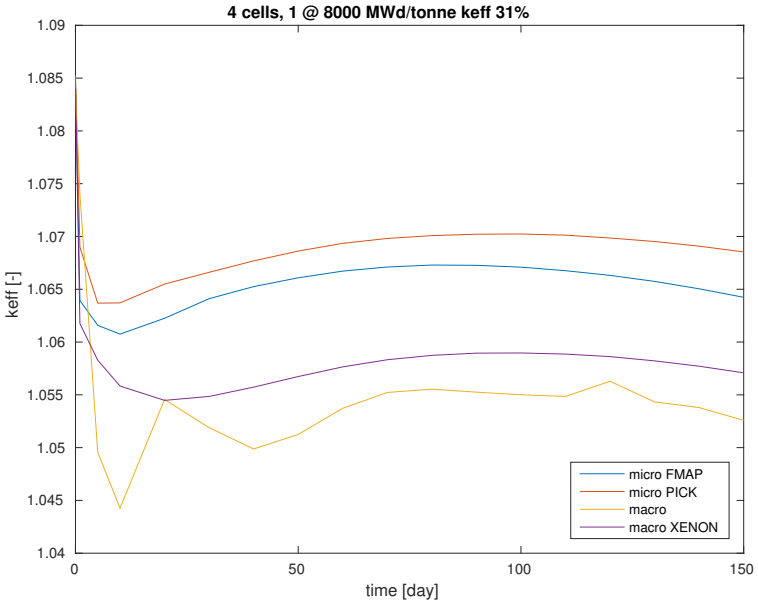

Figure 29: Comparison of keff at 31%

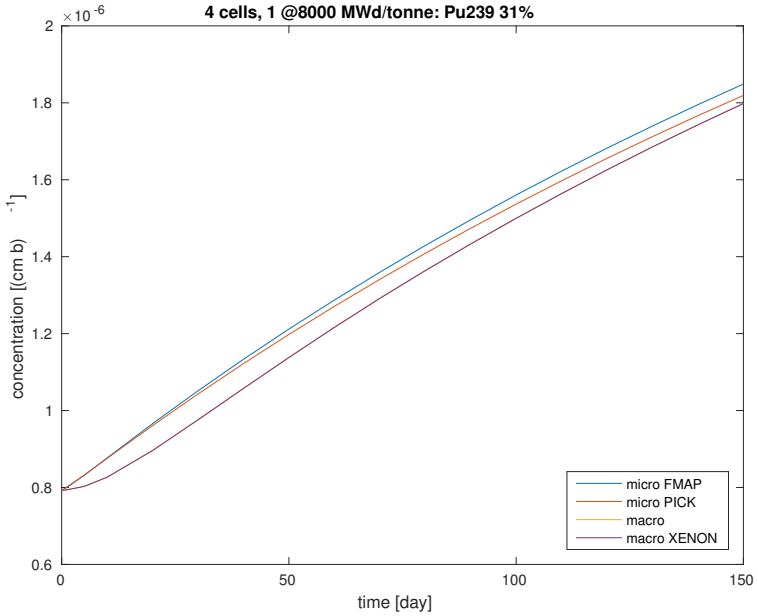

Figure 30: Pu239 concentration at 31%

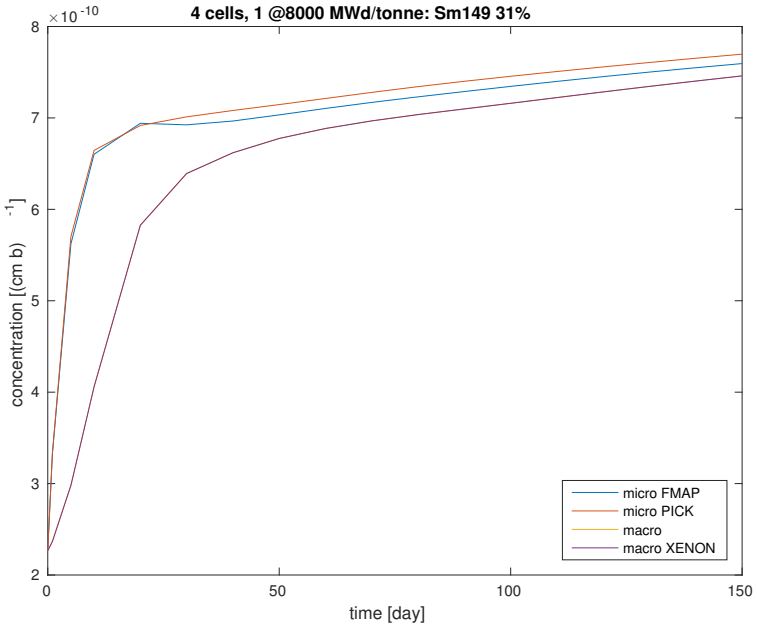

Figure 31: Sm149 concentration at 31%

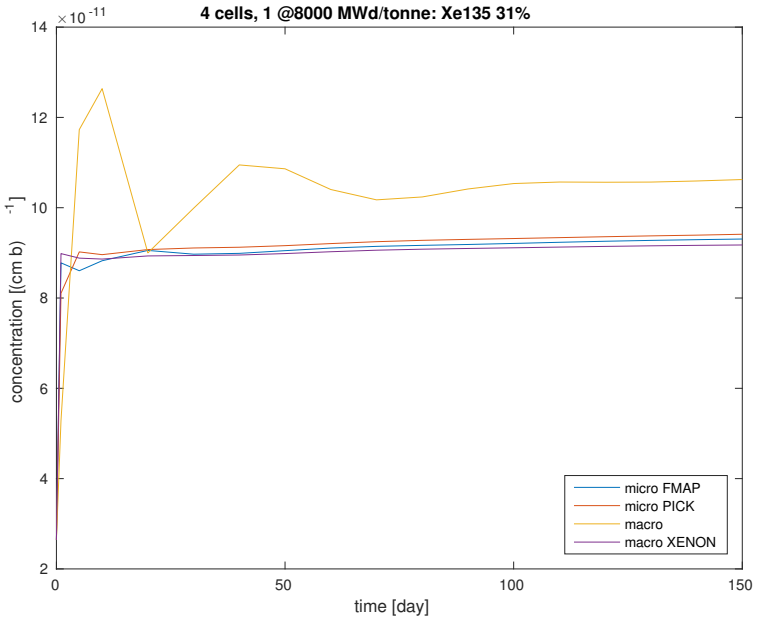

Figure 32: Xe135 concentration at 31%

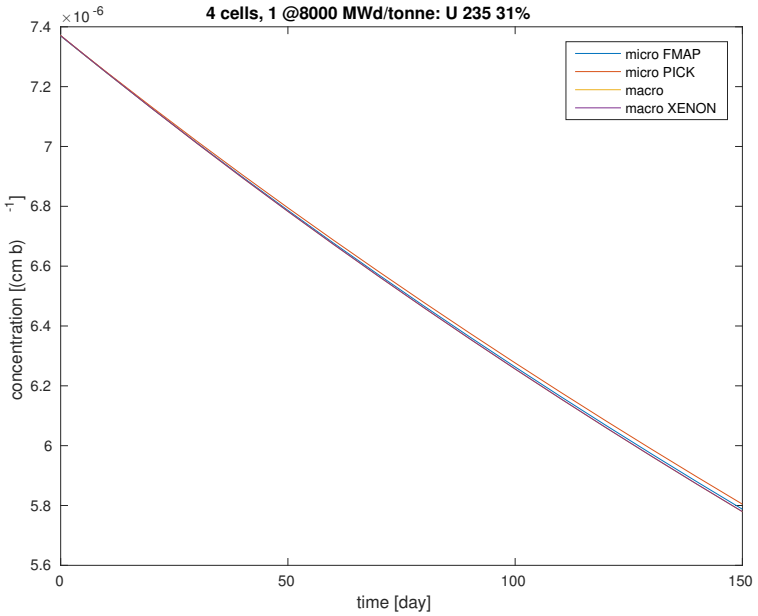

Figure 33: U235 concentration at 31%

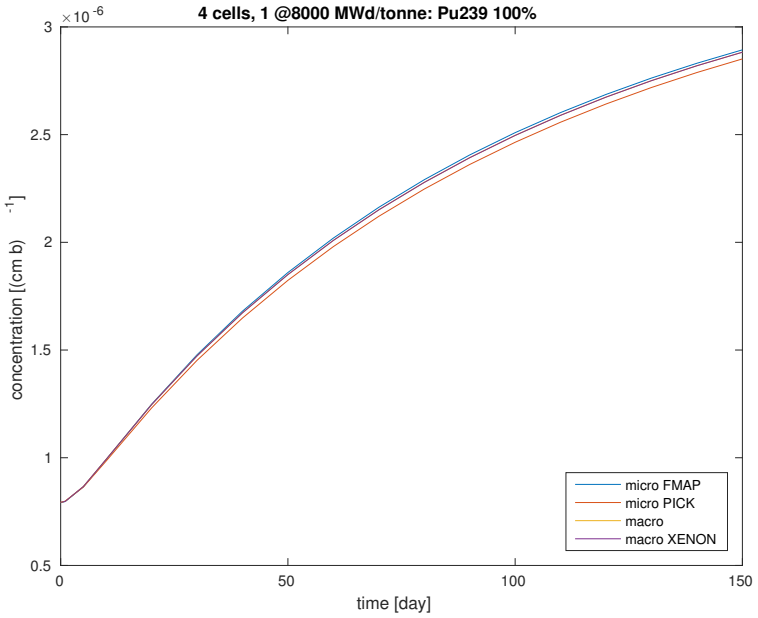

Figure 34: Pu239 concentration at 100%

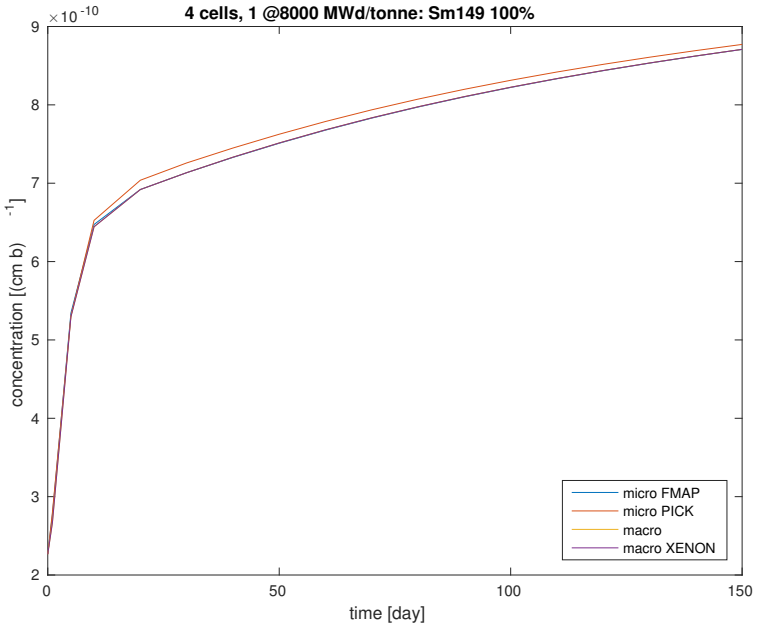

Figure 35: Sm149 concentration at 100%

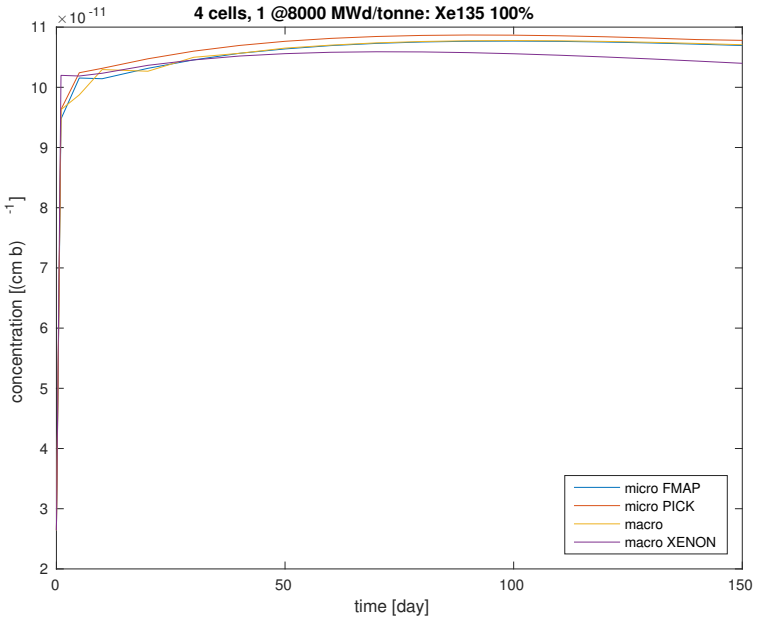

Figure 36: Xe135 concentration at 100%

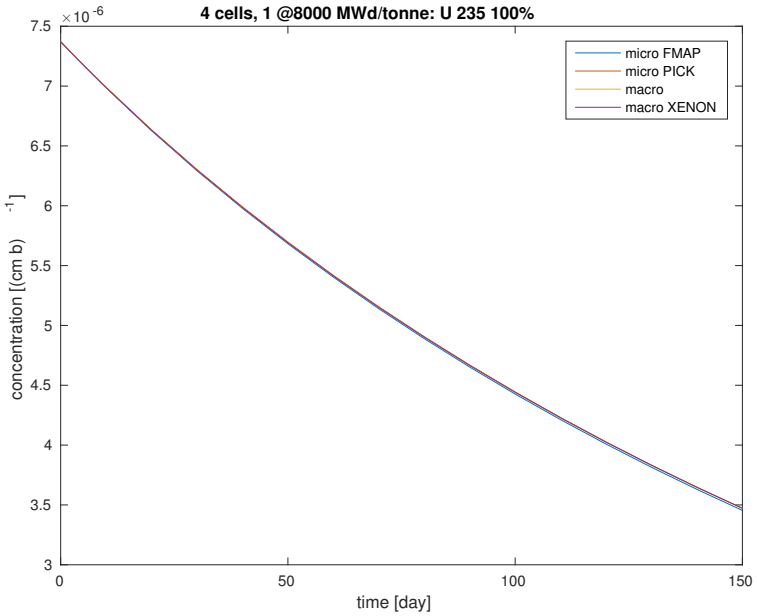

Figure 37: U235 concentration at 100%

consumption of the uranium. Then, concerning the difference between the FMAP microdepletion model and the macrodepletion ones, the same consideration already done applied.
The evolution of the isotopes in not nominal condition is described in figures 33, 32, 30 and 31.

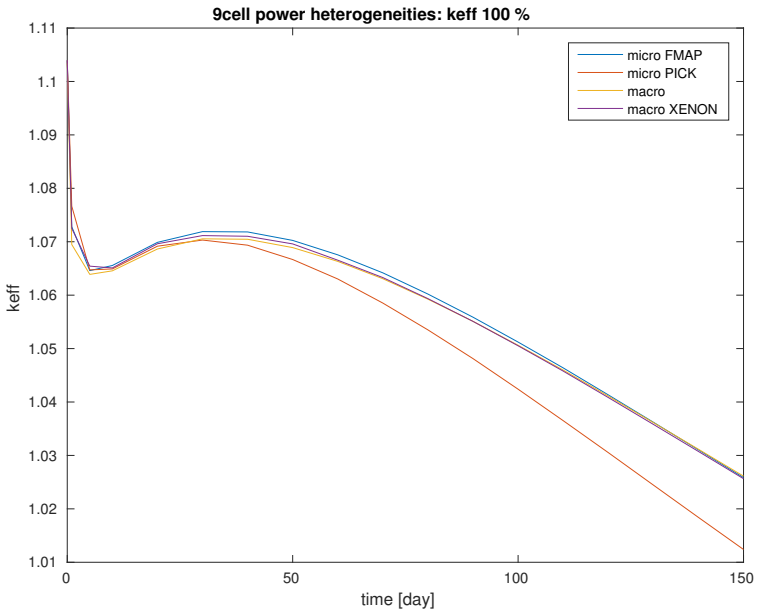

Figure 38: Comparison of keff at 100%

7 9 cells

7.1 keff

An assembly of 9 cell is defined. This assembly has got the right down corner bundle at 8000 MWd/tonne. Power heterogeneities are present. The two macrodepletion models and the microdepletion one using the FMAP are coherent one to each other. The microdepletion one using PICK appears smaller with respect to the others. No reference calculation with Dragon are present. After 150 days of operation at nominal condition - figure 38:

	keff 100%
micro FMAP	1.025864
micro PICK	1.012398
macro TINST	1.025645
macro XENON	1.026120

After 150 days of operation at 31% power - figure 39:

	keff 31%
micro FMAP	1.080246
micro PICK	1.081568
macro TINST	1.069130
macro XENON	1.073540

The difference between the two microdepletion model lowers when the irradiation lowers. Another consideration to be done is the fact that diminishing the relative importance of the power heterogeneity means diminishing the difference between the FMAP macro depletion model and the PICK one.

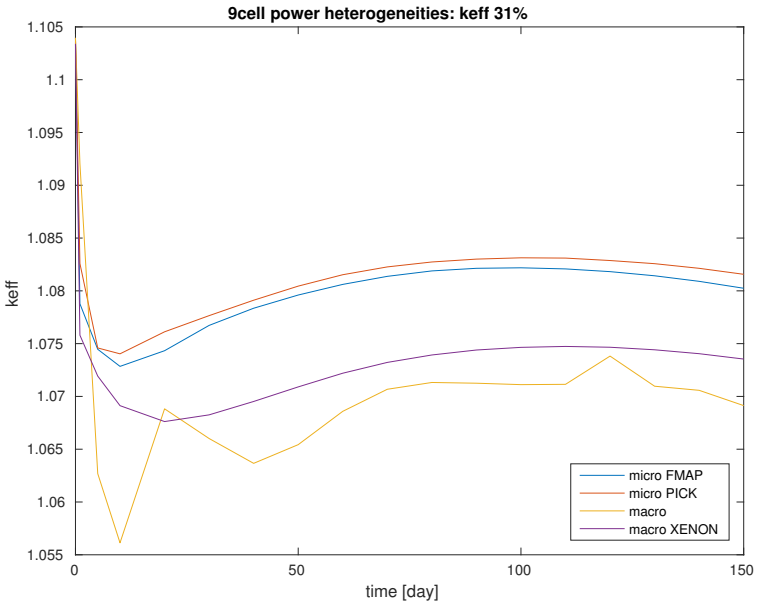

Figure 39: Comparison of keff at 31%

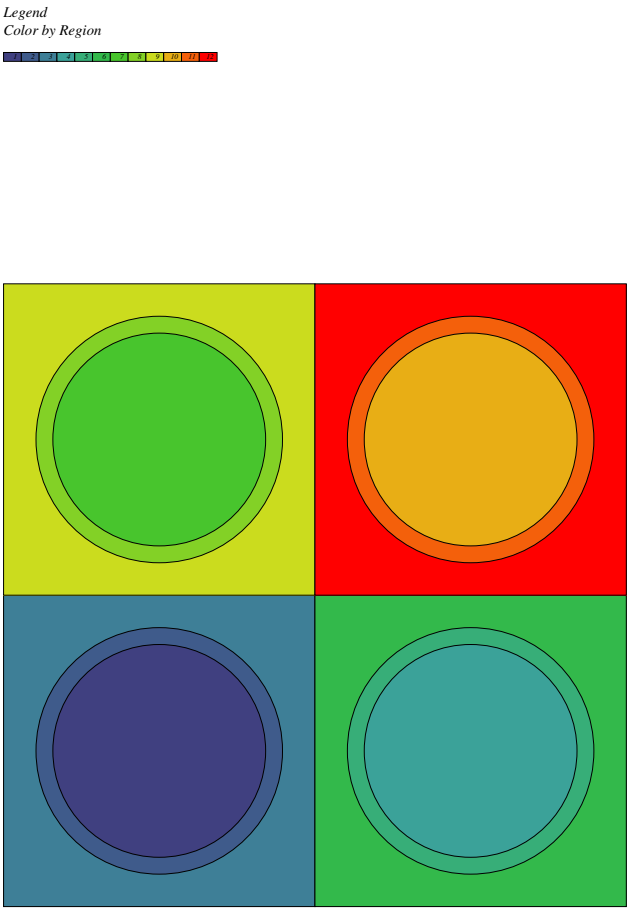

Figure 40: Geometry: UO2 in upper left and down right corner; MOX in upper right and down left corner

8 4 cell, 2 UO2 & 2 MOX

8.1 Geometry

The geometry is set as in figure 40. The data for the fuel are taken from the Rowland test case. The dimension of the geometry too. The MULTICOMPO is created with an overall homogenization. 4 DONJON models are evaluated: 2 macrodepletion (one using the XENON: module to correct the Xenon effect) and 2 microdepletion one (one using a FMAP and the other using only the PICK keyword to extract the burnup).

8.2 keff

In this case, the MULTICOMPO is obtained throughout the homogenization of the whole assembly and not of the single cells.
After 150 days of operation at nominal condition (figure 41):

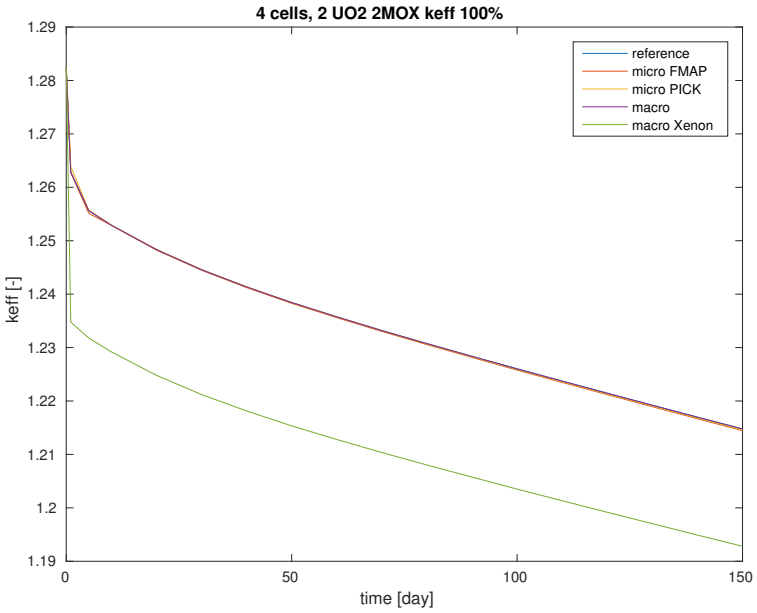

Figure 41: Comparison of keff at 100% @assembly homogenization

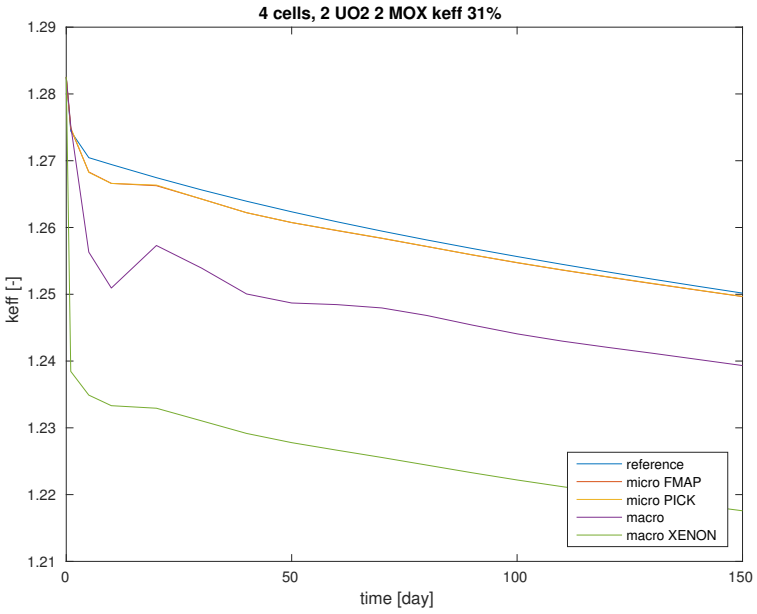

Figure 42: Comparison of keff at 31% @assembly homogenization

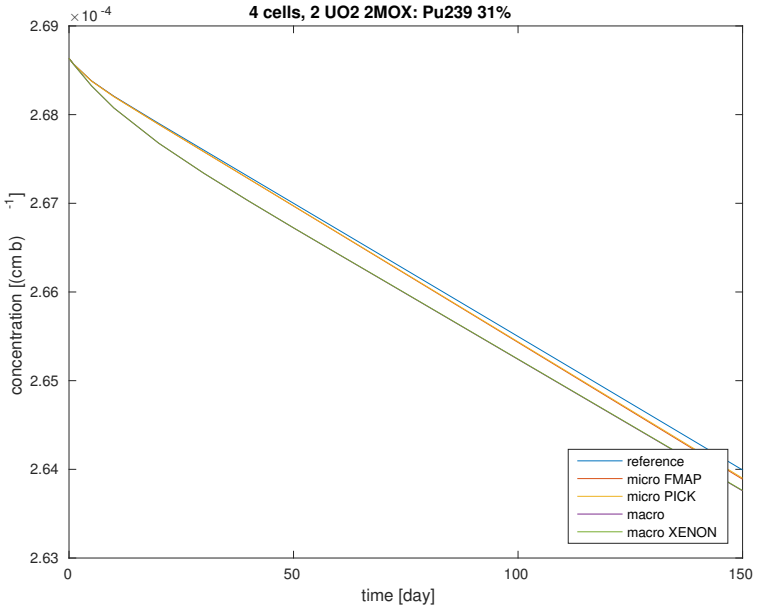

Figure 43: Pu239 concentration at 31% @ assembly homogenization

	keff 100%	dkeff [pcm]
reference	1.214764	–
macro TINST	1.214769	0.5
macro XENON	1.192811	2195.3
micro FMAP	1.214456	30.8
micro PICK	1.214593	17.1

After 150 days of operation at 31% power (figure 42):

	keff 100%	dkeff [pcm]
reference	1.250174	–
macro TINST	1.239325	1085
macro XENON	1.217579	3260
micro FMAP	1.249653	52.1
micro PICK	1.249679	49.5

In this case the TINST macrodepletion model seems adequate to describe the burnup of the assembly at nominal condition. Even in this case the XENON: corrected macrodepletion model underestimates the multiplicative factor.Better results are obtained for the microdepletion models also.
At 31% power condition, the macrodepletion model results does not suit the reference ones. The results of the microdepletion models are, even in this case, more consistent with the reference ones.

8.3 concentration

In this case at nominal condition the TINST macrodepletion model estimates correctly the isotope concentrations (figures 47,48, 50 and 49). The XENON: corrected model overestimates the Xe135 concentration (figure 49).
At 31% power condition the U235 is well estimated (figure 46). There is an overestimation of the Xe135 for the macrodepletion models (figure 45). Sm149 is overestimated too (figure 44). Pu239 is well represented in all the models (figure 43).

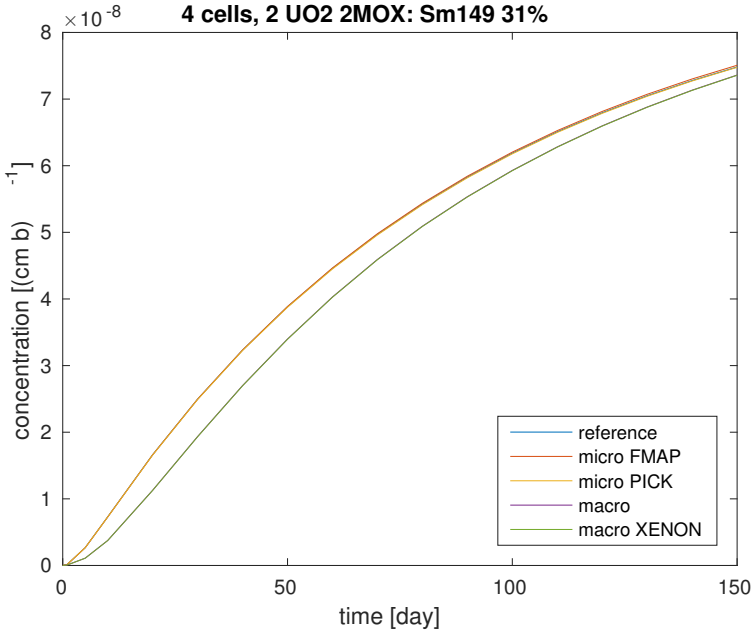

Figure 44: Sm149 concentration at 31% @assembly homogenization

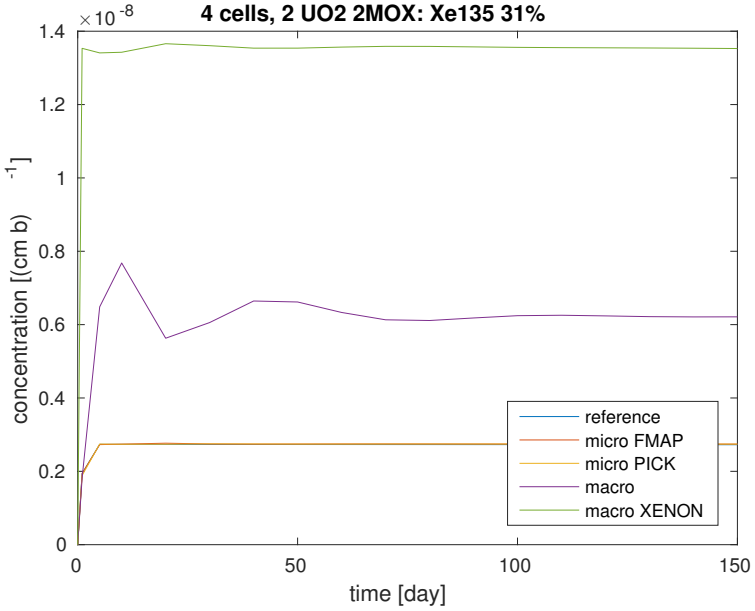

Figure 45: Xe135 concentration at 31% @assembly homogenization

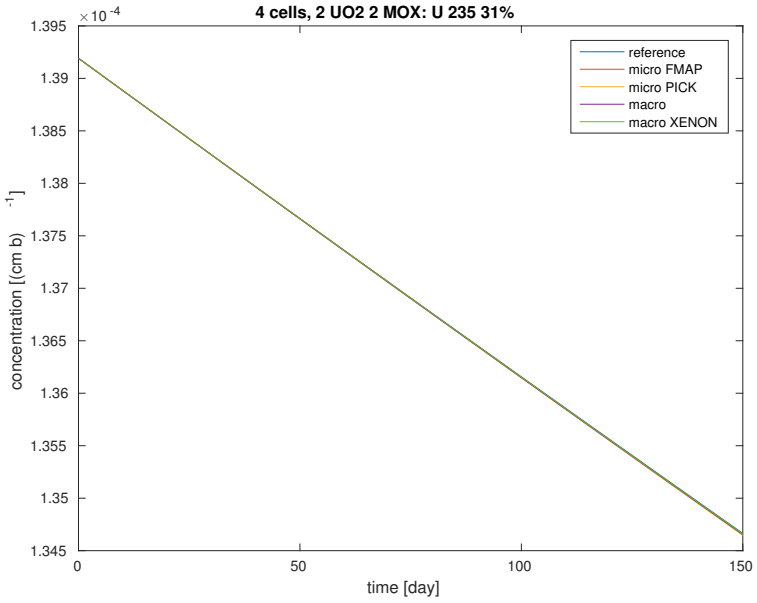

Figure 46: U235 concentration at 31% @assembly homogenization

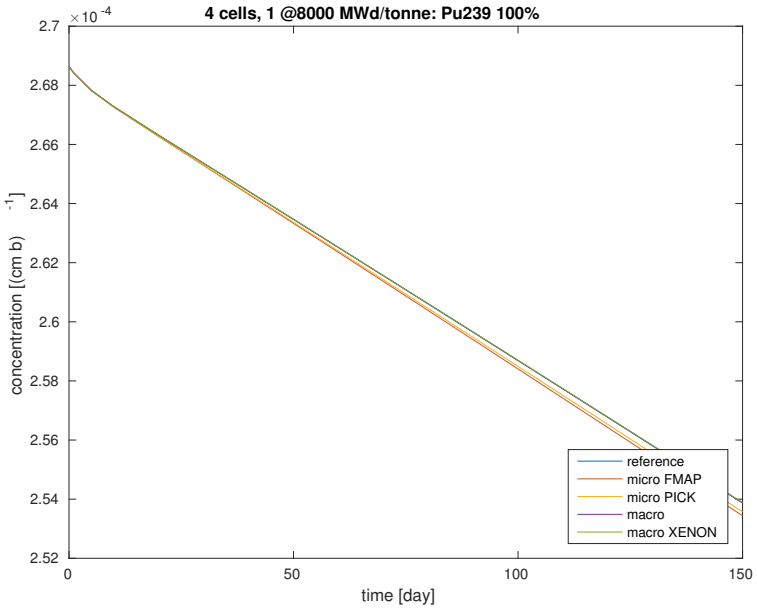

Figure 47: Pu239 concentration at 100% @assembly homogenization

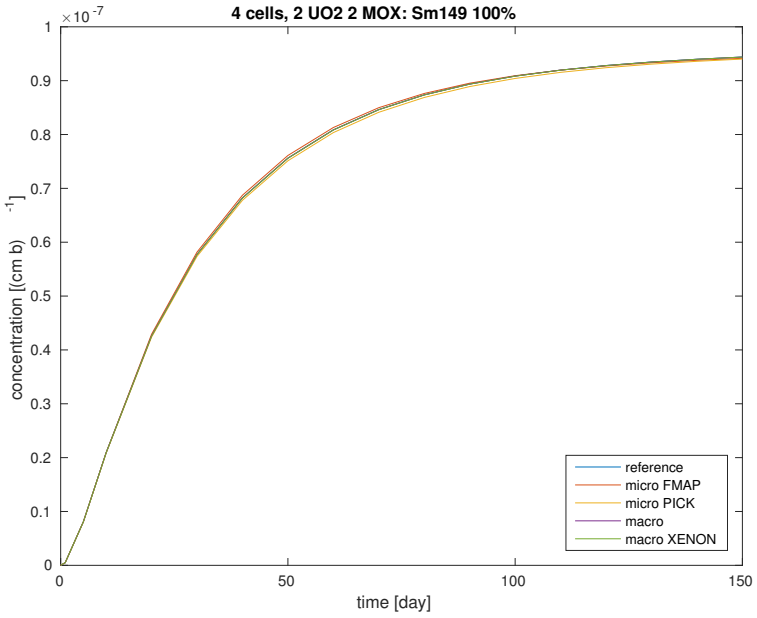

Figure 48: Sm149 concentration at 100% @assembly homogenization

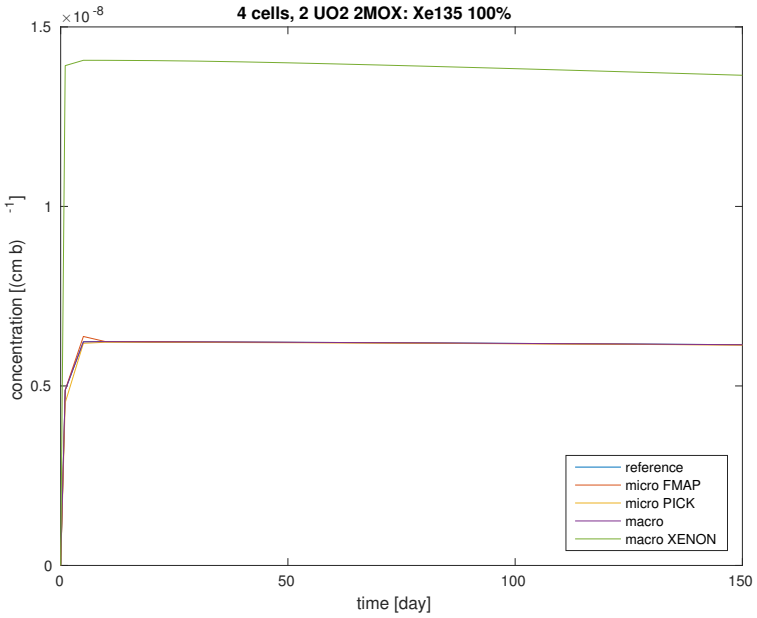

Figure 49: Xe135 concentration at 100% @assembly homogenization

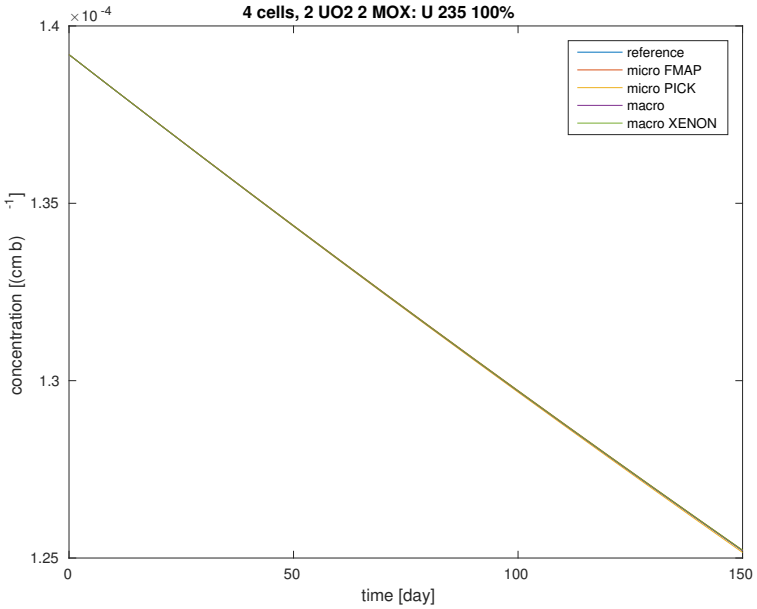

Figure 50: U235 concentration at 100% @assembly homogenization

9 Conclusion

The aim of this paper is to introduce 4 depletion models. Two of them are macro depletion models and they extract from the MULTICOMPO, a reactor database created with a lattice code, DRAGON in this particular case, both the microscopic cross section and the isotope concentrations as long as the cell is irradiated. The other two are micro depletion model and they extract from the database only the microscopic cross sections. General considerations can be made and one of them is that macro depletion methods work fine in nominal conditions, but they do not reach satisfying results in not nominal ones. Micro depletion methods are more time expensive but they supply at this problem. It is possible to add, then, that the fuel map is of major importance in micro depletion methods, if power heterogeneities are present.