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NEUTRONICS CODES AND METHODOLOGIES APPLIED FOR THE 3-D WHOLE CORE EVALUATIONS OF THE BR2 FUEL CYCLE

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ABSTRACT

This paper presents an overview of the neutronic codes and methodologies which are used for the management of the current BR2 fuel cycle. Similar methodologies will be applied in the feasibility studies for the conversion of the BR2 reactor from HEU to LEU fuel.

The steady – state flux calculations and the evaluation of the effective multiplication factor k_{eff} are performed for the detailed 3-D heterogeneous reactor model, using the steady-state option of any MCNPX version.

The application and comparison of three depletion methodologies are discussed.

In the first two methodologies MCNPX is coupled with a 1-D burn up code (ORIGEN-S, or CINDER90 through MCNPX 2.7.D2), which evaluates the evolution of the fuel composition in an infinite lattice. The method in these methodologies is based on preliminary preparation of databases, containing depleted isotopic fuel compositions with depletion step 2% between 0% and 80% fuel burn up, and power peaking factors, which are calculated following the standard irradiation history of the fuel element in the BR2 fuel cycle. The approach taken is to calculate by MCNPX (any version) the total power and the mean burn up in each fuel element at different time depletion steps and then along with the databases to evaluate the 3-D power and 3-D isotopic fuel distributions in the core. MCNPX simulations are prepared for each requested time depletion step and submitted in parallel on the Fermi4 Cluster at SCK•CEN, which takes a few hours of computational time.

The third methodology represents fully automatic 3-D whole core depletion calculations by the Monte Carlo burn up code MCNPX 2.7.D2.

1. INTRODUCTION

A brief overview of the neutronic codes and methodologies used for the evaluations of the fuel cycle at the BR2 reactor is presented. Some of these methodologies will be applied in the feasibility studies for the conversion of the BR2 reactor from HEU to LEU fuel. The steady – state 3-D flux calculations and the

evaluation of the effective multiplication factor k_{eff} are performed for the detailed full scale 3-D heterogeneous geometry model of the reactor BR2 by the Monte Carlo code MCNPX, using the steady-state option in any of the versions 2.5.0, 2.6.E, 2.6.F, 2.7.A, 2.7.B, 2.7.C, 2.7.D2. In the presented methodology MCNPX is coupled with a 1-D depletion code (ORIGEN-S or CINDER90 through MCNPX 2.7.D2), which evaluates the isotopic fuel composition. MCNPX 2.7.D2 [1] is an extended version of the code MCNPX 2.5.0 with included automatic depletion/burn up capabilities. MCNPX 2.7 is under active development and the latest versions of the code are available to the beta testers under the Beta Test Agreement.

The application and comparison of three depletion methodologies are discussed in this paper. The evolution of the fuel composition in the first two methodologies is performed in ***an infinite lattice***.

(1) MCNPX & ORIGEN-S.

The first methodology couples the 1-D depletion calculations by ORIGEN-S [2] in an infinite lattice with the 3-D flux calculations by MCNPX (any version). The 3-D reaction rates for *selected majo*r actinides and fission products are calculated by MCNPX (any version) and the one – group effective microscopic cross sections are introduced into ORIGEN-S to replace the existing cross sections for the LWR reactor in the ORIGEN-S libraries. The depleted isotopic fuel compositions are introduced back into the whole core 3-D heterogeneous geometric MCNPX (any version) model.

(2) MCNPX & CINDER90 (through MCNPX 2.7.D2).

In the second methodology the depletion is performed by the 1-D depletion code CINDER90 [1], which is automatically linked with the 3-D flux calculations by MCNPX 2.7.D2 [1] in an infinite lattice. In this methodology, the 3-D reaction rates for *all* isotopes are calculated by MCNPX 2.7.D2 and the one-group effective microscopic cross sections are automatically transferred to CINDER90 to perform the depletion calculations. The evaluated depleted isotopic fuel compositions are introduced back into the whole core, 3-D heterogeneous geometric MCNPX (any version) model.

The application of these two methodologies at BR2 is based on preliminary preparation of Data Bases:

- **DFDB** (Depleted Fuel Data Base). The depleted isotopic fuel compositions are evaluated with a 1-D depletion code (ORIGEN-S or CINDER90 in MCNPX 2.7.D2). This database contains several classes of fuel compositions: (a) for conditions at BOC after a shutdown of 30 days (with saturated Sm-149, disappearance of Xe-135, etc.); (b) for conditions at t=0 ÷ 2 days irradiation after BOC, when Xe-135 concentration reaches maximum; (c) at equilibrium conditions during the cycle.
- **PPDB** (Power Peaking factors Data Base). This database contains the axial power peaking factors in the fuel elements with various mean burn up. The power peaking factors are calculated, following a standard irradiation (depletion) history of the fuel element in the standard BR2 operation cycles. This PPDB contains several classes of power peaking factors, calculated at different control rod positions during the BR2 operation cycle.

The approach taken in the first two methodologies is to calculate the total power by MCNPX (any version) and to evaluate the mean fuel burn up in each fuel element at different depletion time steps. Then, using the PPDB along with DFDB, the 3-D isotopic fuel distribution is evaluated in each fuel element at a given time step and introduced back into the MCNPX model. MCNPX (any steady-state version) runs are prepared for the requested number time depletion steps and submitted in parallel on the Fermi4 Cluster at SCK•CEN. The computation time for one single run (or 16 parallel independent runs, corresponding to the different time depletion steps) takes about 3 hours for 2.000.000 neutron histories.

(3) MCNPX 2.7.D2.

The third methodology represents ***fully automatic whole core depletion calculations*** by the Monte Carlo burn up code MCNPX 2.7.D2 in the 3-D heterogeneous geometric model of the BR2 reactor.

2. MCNPX & ORIGEN-S COMBINED METHOD

This section gives a brief description of the first depletion methodology, which has been used up to the present time for evaluation of the BR2 operation cycles. The depletion calculations are performed in an infinite lattice using the 1-D burn up code ORIGEN-S [2]. It is a module of the SCALE system, which can also be used as a stand-alone module. The evaluated depleted isotopic fuel compositions as function of the irradiation time for a given fuel type are stored in a database DFDB (Depleted Fuel Data Base). MCNPX calculates the total power in a fuel element, which is used as input for ORIGEN-S, and the 3-D power distribution in the core. The power peaking factors are calculated, following a standard irradiation (depletion) history of the fuel element in the BR2 operation cycles and stored in a database PPDB (Power Peaking factors Data Base). Using the PPDB along with DFDB, the 3-D isotopic fuel distribution is evaluated in the MCNPX whole core model at a given time step. MCNPX runs are prepared for the different number time depletion steps and submitted in parallel on the Fermi4 Cluster at SCK•CEN to evaluate the k_{eff} for the requested depletion time steps. The following sections describe the assumptions adopted in this method.

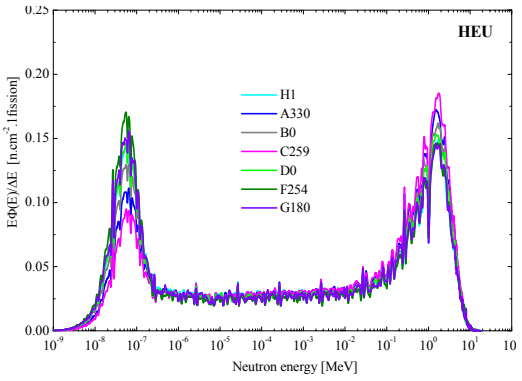
2.1 Preparation of DFDB (Depleted Fuel Data Base)

2.1.1 Calculation of effective microscopic cross sections by MCNPX in the whole core model

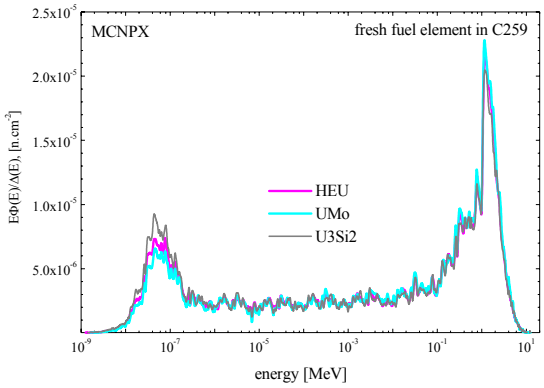
The ORIGEN-S nuclear data libraries contain cross sections and fission yields for LWR. MCNPX is used to evaluate the 3-D continuous energy reaction rates and fluxes in the full core model, which are converted into one – group constants. The effective microscopic cross sections $<\sigma>_{eff}^{MCNPX}$ for selected major actinides and dominant fission products of the HEU fuel, weighted in the BR2 fuel spectrum, are used to update the existing cross sections for the LWR reactor in the ORIGEN-S libraries.

MCNPX calculations of the neutron spectrum in the different channels for the HEU fuel are presented in Fig. 1a. Figure 1b shows comparison of the spectra between different fuel types. Detailed calculations have shown that the reaction rates are independent on the position of the fuel channel in the core (see Fig. 1c). The effective microscopic cross sections, calculated by MCNPX for *major* HEU fuel isotopes are given in Fig. 1d.

The same calculations performed in an infinite lattice have not shown any change in the spectral dependence of the reaction rates and microscopic cross sections.



a.



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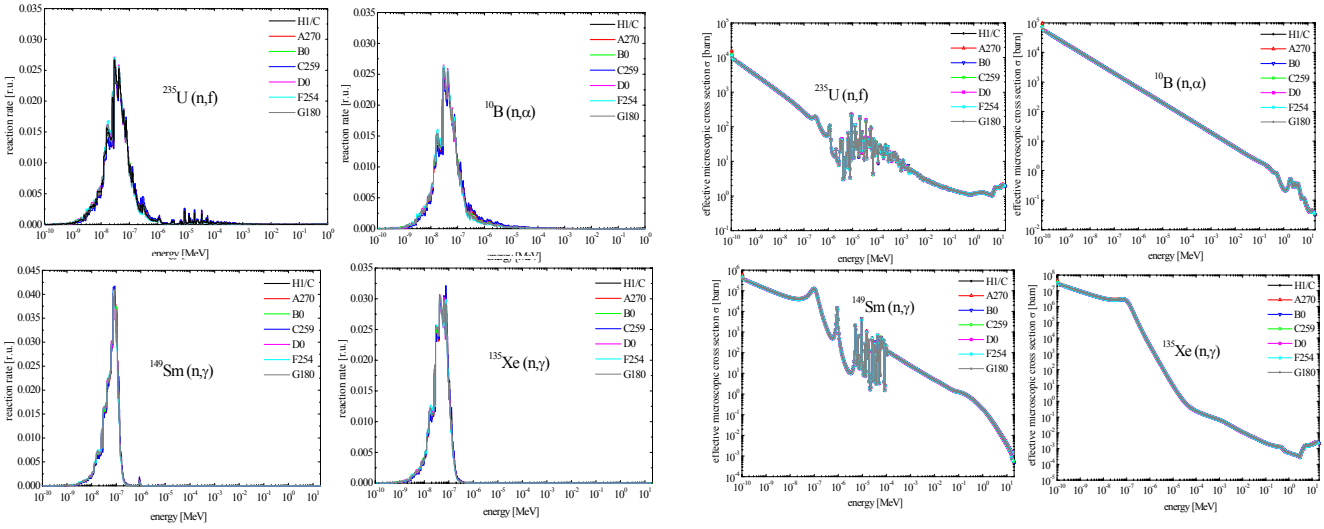


c. d.

Figure 1. a. Neutron spectrum in different channels for HEU fuel; b. Neutron spectrum in different fuel types c. Reaction rates and d. effective microscopic cross sections of major HEU fuel isotopes, calculated with MCNPX in different fuel channels in the full 3-D heterogeneous geometry model.

2.1.2 Evaluation of the isotopic fuel composition by ORIGEN-S in an infinite lattice

The input for ORIGEN-S can be the local fission power or local flux, calculated by MCNPX in the fuel cells where the burn up calculations are needed. However due to the complicated 3-D power and flux distribution in the BR2 core, it would be very time consuming to calculate by ORIGEN-S the fuel evolution for each separated fuel geometric cell. Therefore, the following assumptions have been adopted. The fuel depletion is evaluated by ORIGEN-S in infinite lattice for the average power of the fuel element. However, the local power in the BR2 channel is axially dependent and also depends on the position of the channel in the reactor core. Therefore, several evaluations of the isotopic fuel evolution were performed by ORIGEN-S varying the initial power of the fuel element in the infinite lattice. The purpose of these evaluations is to check how strong is the dependence of the isotopic fuel concentrations on the local power. The comparison of the evolution of the atomic concentrations for some fission products and actinides are shown in Fig. 2. These figures show slight dependence of the transmutation rates of the major fission products and actinides (Sm-149, Xe-135, Pu-239) and larger dependence for the minor fission products (Rh-103, Pm-147) on the value of the initial power.

The evolution of the isotopic fuel composition for the HEU fuel is calculated by ORIGEN-S for initial fuel power 2 MW of the fresh fuel element in the infinite lattice. The depletion calculations are performed with a time step $\Delta t \sim 3.15$ days, which corresponds to $\Delta\beta^5 \sim 2\%$ depletion step in U-235. The evaluated depleted isotopic fuel compositions with varied fuel burn up between 0% and 80% are stored in a database DFDB. This database contains several classes of fuel compositions: (a) for conditions at BOC after operation of 21 days and a shutdown of 30 days (with saturated Sm-149, disappeared Xe-135, etc.); (b) for conditions at ~ 0.8 days irradiation after BOC, when Xe-135 concentration reaches maximum; (c) at equilibrium conditions during the cycle. Each fuel composition contains about 100 fuel isotopes, selected from the ORIGEN-S output.

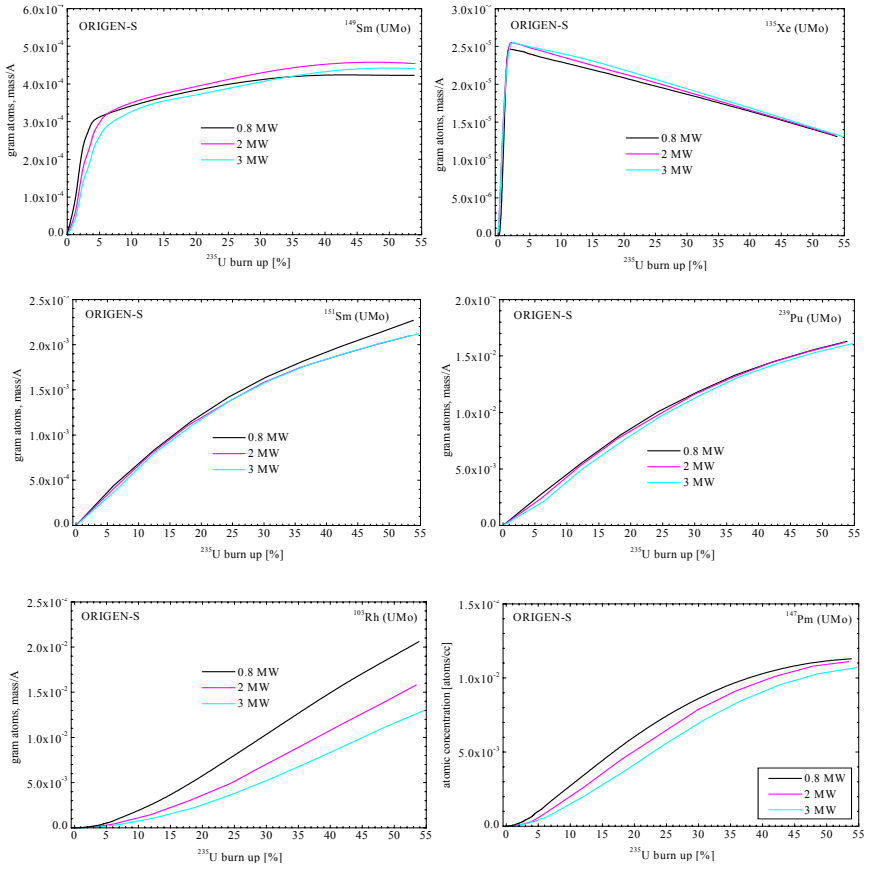


Figure 2. Dependence on the local power of the atomic fuel isotopic concentrations, evaluated with ORIGEN-S.

2.2 Preparation of PPDB (Power Peaking factors Data Base)

2.2.1 Calculation of power peaking factors by MCNPX

The isotopic fuel composition for a given time step is introduced back into the whole core geometric MCNPX model and distributed in the core using the detailed 3−D power peaking factors, which are earlier evaluated with MCNPX and stored in a database PPDB. The typical reloading BR2 strategy consists of an operation cycle with duration of three weeks and a shutdown of 30 days. The axial power peaking factors for a given fuel type are calculated by MCNPX following the reloading strategy, which is schematically shown in Fig. 3a. The axial power peaking factors for different fuel types are given in Fig. 3b. As can be seen, with increase of the fuel burn up, the power distribution in the fuel plates of the UMo fuel element become smoother than for the HEU fuel.

a.

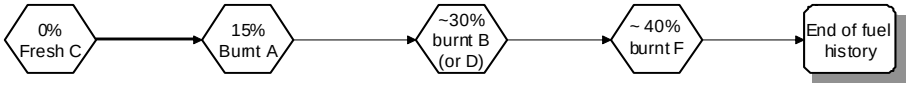
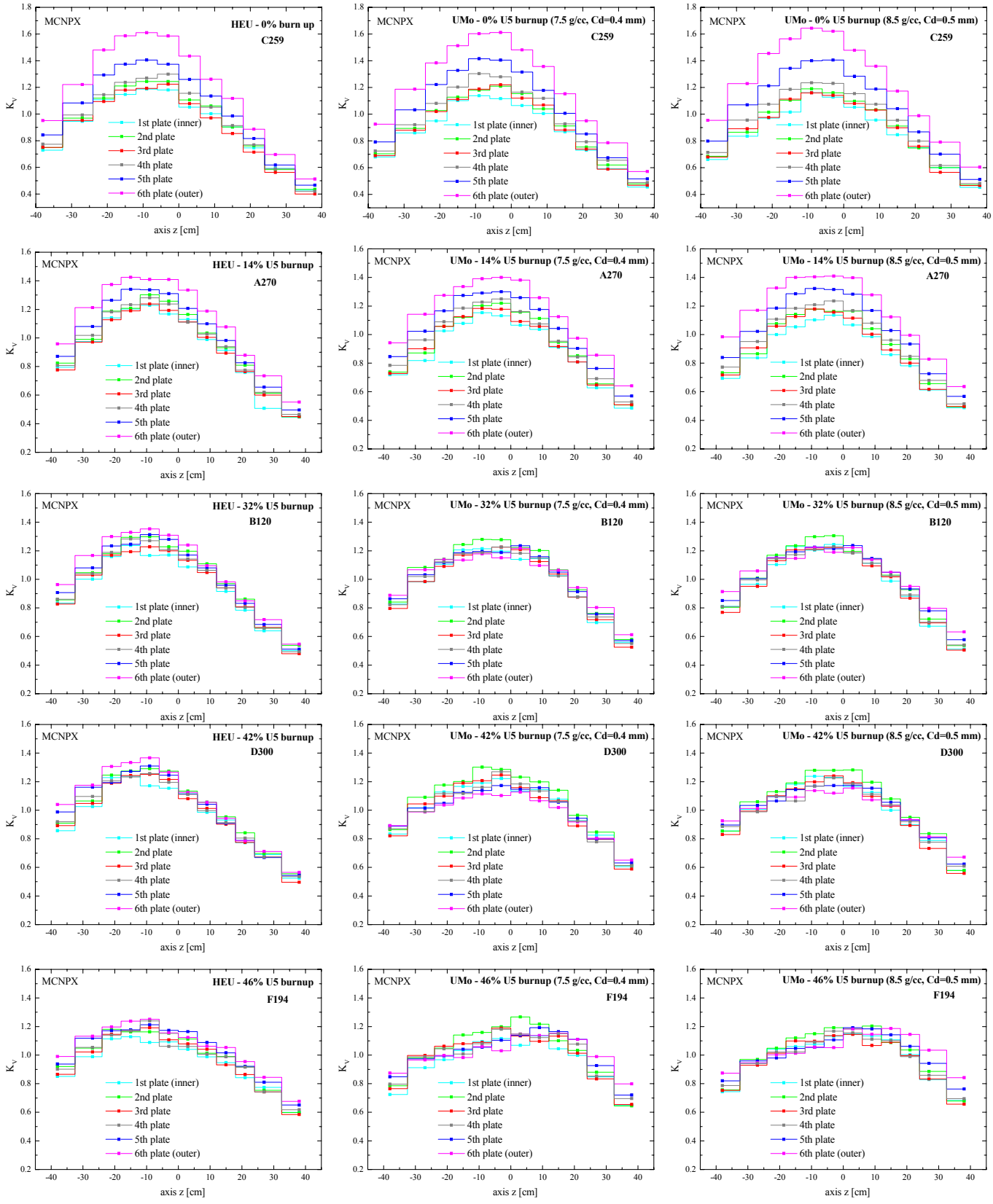


Figure 3. a. Schematic view of irradiation history of a standard BR2 fuel element, following typical reloading BR2 strategy.



3b. Axial power peaking factors for different fuel types calculated in the fuel plates of a fuel element, loaded in different fuel channels.

The power peaking factor in a fuel zone (r_m, z_n, φ_l) of the fuel element j of fuel type k during the irradiation time T_i is

$$K_{V_k^j}(T_i, r_m, z_n, \varphi_l) = \frac{\int_{r_m} dr \int_{z_n} dz \int_{\varphi_l} d\varphi p_k^j(T_i, r, z, \varphi) \Big/ \int_{r_m} dr \int_{z_n} dz \int_{\varphi_l} d\varphi}{\overline{P_k^j}(T_i) \Big/ V_k^j} \tag{1}$$

where: r_m is the radial position of a fuel plate in the fuel element; z_n is the axial coordinate of the fuel zone n in the fuel element; φ_l is the azimuth angle in azimuth fuel sector l in the fuel element; $p_k^j(T_i, r_m, z_n, \varphi_l)$, is the fission power in the spatial segment (r_m, z_n, φ_l) inside the fuel element and $\overline{P_k^j}(T_i)$ is the average fuel power of the fuel element, which are evaluated by MCNPX; V_k^j is the total fuel meat volume of the fuel element j of fuel type k.

2.2.2 3-D isotopic fuel distribution in the whole core MCNPX model

The distribution of the 3-D isotopic fuel depletion in the core is calculated, using the correlation between the fuel depletion and the relative 3-D power distribution. The mean fuel depletion in a fuel element j is calculated with the fission power $\overline{P_k^j}$ of the fuel element j of type k, being determined by MCNPX calculation. The equation for the fuel depletion in the fuel element j for fuel type k, including the formation and the depletion of all TRU – isotopes, is

$$\overline{\beta_k^j}[\%] = \frac{M_k^j(0) - M_k^j(T)}{M_k^j(0)} \approx \text{const}_k^j \times \frac{T \times \overline{P_k^j}}{M_k^j(0)} = \frac{\text{const}_k^j \times T}{c_k^j(0)} \times \frac{\overline{P_k^j}}{V_k^j}, \tag{2}$$

$$\text{const}_k^j = \frac{A_k^j}{N_A \cdot E_k^{\text{eff}}} \cdot \frac{\langle \sigma_f + \sigma_c \rangle_k^j}{\langle \sigma_f \rangle_k^j}, \tag{3}$$

$$\alpha_k^j = \frac{\langle \sigma_f + \sigma_c \rangle_k^j}{\langle \sigma_f \rangle_k^j} \tag{4}$$

is a constant for fuel type k in the fuel zone V_k^j. $c_k^j(0)$ and $M_k^j(0)$ in Eq. (2) are the initial fuel density and the initial fuel mass of the fuel element j of type k and $\overline{P_k^j}$ is the deposited fission energy in the fuel meat. E_k^{eff} in Eq. (3) is the effective released fission energy by all fissile isotopes in fuel type k and A_k^j is the atomic mass number of the fissile isotopes in the fuel element j of type k.

As it was shown in Sect. 2.1.2, the evolution of the atomic fuel densities, and respectively transmutation rates ΔM/M [%], mainly depend on the fuel burn up and produced energy Pt, and almost do not depend on the local flux/power. The assumption is that the isotopic fuel distribution is proportional to the power peaking factor

$$\beta_k^j[\%] \sim \frac{\Delta M_k^j}{M_k^j}[\%] \sim \text{Pt[MW.d]} \sim K_{V_k^j} \tag{5}$$

Then, the 3-D space distribution of the isotopic fuel depletion in the fuel zone (r_m, z_n, φ_l) of the fuel element j of fuel type k at the time T_i is calculated using the mean isotopic fuel depletion $\overline{\beta_k^j}(T_{i-1})$ and the power peaking factors $K_{V_k^j}(T_{i-1}, r_m, z_n, \varphi_l)$ of the fuel element j and type k in the previous time T_{i-1}

$$\beta_k^j(T_i, r_m, z_n, \varphi_l) = \overline{\beta_{k,FE}^j}(T_{i-1}) \times K_{V_k^j}(T_{i-1}, r_m, z_n, \varphi_l) + \frac{const_{k,FE}^j \times (T_i - T_{i-1})}{c_{k,FE}^j(0)} \times \frac{\overline{P_{k,FE}^j}(T_i)}{V_{k,FE}^j} \times K_{V_k^j}(T_i, r_m, z_n, \varphi_l). \quad (6)$$

Following the irradiation history of the fuel element from channel to channel as shown in Fig. 3a, the power peaking factors are evaluated in the fuel element for different mean burn up and for few typical positions of the control rods during the BR2 operation cycles. These power peaking factors are stored in a database PPDB.

3. MCNPX & CINDER90 COMBINED METHOD

The methodology, described in this section has been applied since the 1st cycle with the EVITA loop in channel H1, and RJH fuel element (27% U5, U_3Si_2) in channel H1/Central (cycle 04/2009A.2 operated from 29 July to 17 August, 2009). This methodology is used for the evaluation of the fuel depletion in the RJH fuel element, and in the standard HEU fuel element, as well. Up to now it has been tested on the reactivity evolution and control rods motion of six operation cycles at different conditions in the channels H1 and H1/Central: with fresh RJH and Al&H_2O plug; with burnt RJH&Al&H_2O plug; with fresh RJH and fresh beryllium plug; with burnt RJH and fresh beryllium plug.

3.1 Preparation of DFDB (Depleted Fuel Data Base)

Similarly to the MCNPX&ORIGEN-S method, the evolution of the fuel composition is performed in an infinite lattice (see Fig. 4) using the **depletion option** of MCNPX 2.7.D2. The difference from the MCNPX&ORIGEN-S method is that the 3-D reaction rates for *all* fuel isotopes are calculated by MCNPX and the one-group effective microscopic cross sections are automatically transferred inside MCNPX 2.7.D2 to the 1-D burn up code CINDER90, which performs the depletion calculations. The MCNPX 2.7.D2 calculations are performed for initial fuel element power $P_{FE}^{HEU} = 2\,\mathrm{MW}$ in the HEU infinite lattice with burn up time step $\Delta t \sim 3.15$ days as in the MCNPX&ORIGEN-S method. Similar calculations are performed for the fuel element j in the LEU infinite lattice for the same depletion time steps and for initial fuel element power equal to:

$$P_{LEU}^j[\mathrm{MW}] = \frac{const_{HEU}^j}{const_{LEU}^j} \frac{M_{LEU}^j(0)}{M_{HEU}^j(0)} \cdot P_{HEU}^j[\mathrm{MW}]. \quad (7)$$

The depleted fuel compositions with varied burn up between 0% and 80%, with U-235 depletion step $\Delta\beta^5 \sim 2\%$, are stored in a database DFDB. The database DFDB contains only two classes of evaluated depleted fuel composition: DFDB-**OC** for conditions during the Operation Cycle (for irradiation times t > 2 days) and DDFB-**BOC** for conditions at Beginning Of Cycle (at t=0). The following procedure for evaluation of the databases at different irradiation conditions is adopted.

The database DFDB-**OC** is evaluated for the continuous irradiation of the fresh fuel element in the infinite lattice, from t=0 to t=126 days, with equal time steps $\Delta t_i \sim 3.15$ days; i=1, 2, 3, …., 40. The obtained isotopic fuel compositions with varied fuel burn up between 0% and 80% and with U-235 depletion step $\Delta\beta^5 \sim 2\%$ are stored in the database DFDB-**OC**.

The DDFB-**BOC** is evaluated for conditions at beginning of cycle, after a shutdown 30 days, when the saturation of Sm-149, disappearance of Xe-135, and the changing of the remaining fuel isotopes is taken into account. The methodology for the evaluation of this database uses the same MCNPX 2.7.D2 model of the infinite lattice with the same irradiation time steps $\Delta t_i \sim 3.15$ days. In order to evaluate the different fuel compositions for conditions at BOC (after 30 days shutdown), separate runs are simulated for the following burn up steps:

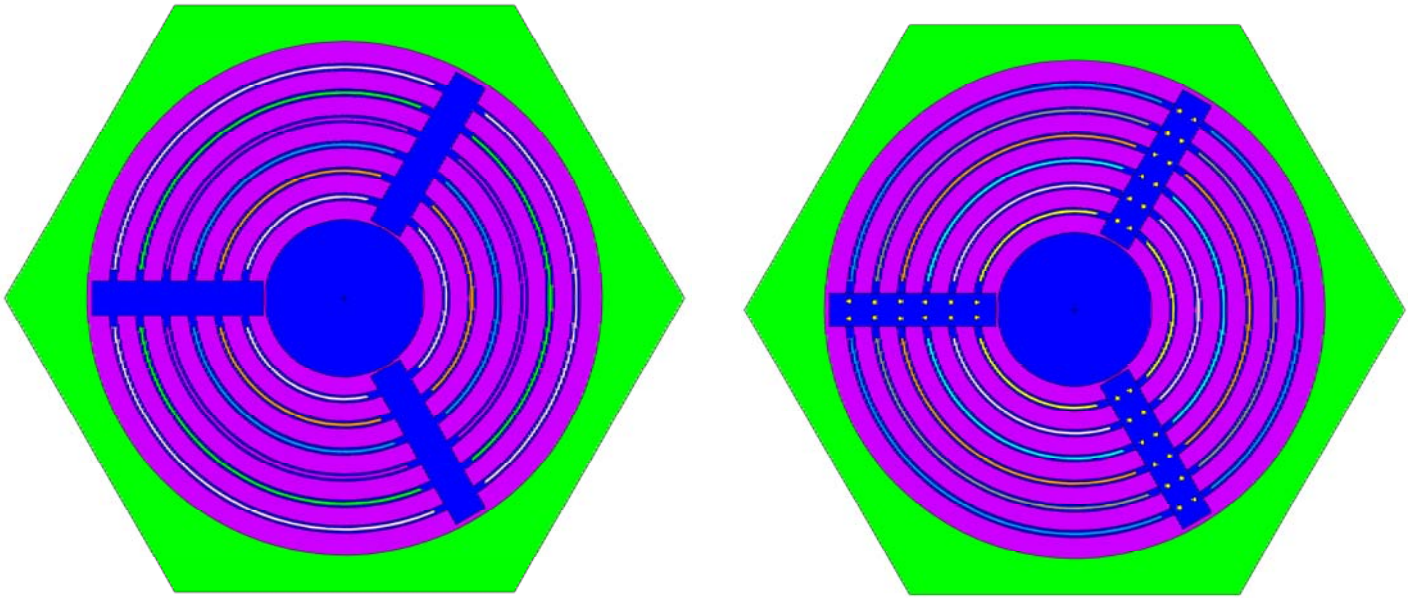
1. for $\Delta t_1 = 3.15 = 1 \times 3.15$ days, followed by shutdown 30 days;
2. for $\Delta t_2 = 3.15 + 3.15$ days $= 2 \times 3.15$ days, followed by a shutdown 30 days;
3. for $\Delta t_3 = 3.15 + 3.15 + 3.15$ days $= 3 \times 3.15$ days, followed by a shutdown 30 days;
4. ……..

40. for $\Delta t_{40} = 3.15 + 3.15 + \ldots\ldots + 3.15 = 40 \times 3.15$ days, followed by a shutdown 30 days

The separate MCNPX 2.7.D2 runs are submitted in parallel on the Fermi4 Cluster in SCK•CEN. The computational time for each run depends on the number of the requested burn up steps. The maximum PC time is needed for the 40^{th} run with 40 burn up steps, which takes about 2.5 days. These runs are performed only once and the fuel compositions are stored in the database DFDB-**BOC**.

For the conditions at $t \sim 0 \div 2$ days after BOC, this methodology does not use database. The evaluation of k_{eff} and the 3-D fuel composition in all fuel elements are performed automatically by MCNPX 2.7.D2 in the full heterogeneous 3-D geometric model of the BR2 reactor for four irradiation time steps:

1. $\Delta t_1 = 0.5$ d.
2. $\Delta t_2 = 0.5$ d. + 0.5 d.
3. $\Delta t_3 = 0.5$ d. + 0.5 d. + 0.5 d.
4. $\Delta t_4 = 0.5$ d. + 0.5 d. + 0.5 d. + 0.5 d.



HEU LEU with Cd-wires

Figure 4. MCNPX models of infinite lattice, represented by a standard BR2 fuel element, loaded in a hexagonal beryllium channel of the Be-matrix.

3.2 Preparation of PPDB (Power Peaking factors Data Base)

Similarly to the MCNPX&ORIGEN-S method, the fuel compositions for a given depletion time step, and for the corresponding operation conditions (at t=0 or for t > 2 days) are derived from the databases (DFDB-**BOC** and DFDB-**OC**) and introduced back into the whole core geometric MCNPX (any version) model. The 3-D isotopic fuel depletion is distributed in the core using the 3–D power peaking factors from the database PPDB (Power Peaking factors Data Base), which are earlier evaluated with MCNPX for the full heterogeneous model. However, the power peaking factors in typical fuel channels A, B, C, D, F and H1/C are re-calculated every time if a special experiment is loaded into the core (e.g., the loading of the EVITA loop in H1/Central, loading of E-FUTURE in a standard fuel channel, loading of new PRF-devices, etc.).

3.3 Evaluation of the reactivity evolution and control rods motion during the cycle

Similarly to the MCNPX&ORIGEN-S methodology, MCNPX (any steady-state version) runs are prepared for the different number time depletion steps and submitted in parallel on the Fermi4 Cluster at SCK•CEN. The calculations of k_{eff} are performed at the same critical position Sh_{BOC} of the control rods bank at the beginning of cycle. After that the reactivity value for each time step along with the calculated by MCNPX differential control rods worth for the considered cycle are used to estimate the positions Sh of the CR bank during the operating cycle. A next series of MCNPX runs is submitted on the Cluster in order to check that k_{eff}=1.0 for the estimated control rods positions at the different time steps.

4. MCNPX 2.7.D2 AUTOMATIC WHOLE CORE DEPLETION

In the third methodology the depletion and reactivity calculations by MCNPX 2.7.D2 are performed automatically in the full 3-D heterogeneous geometric model of the BR2 reactor (see Fig. 5). This methodology does not use any assumptions or simplifications as discussed in the previous chapters. The calculations of 10 depletion steps during the operating cycle takes about a week on the Fermi4 Cluster at SCK•CEN for 2.000.000 neutron histories per step.

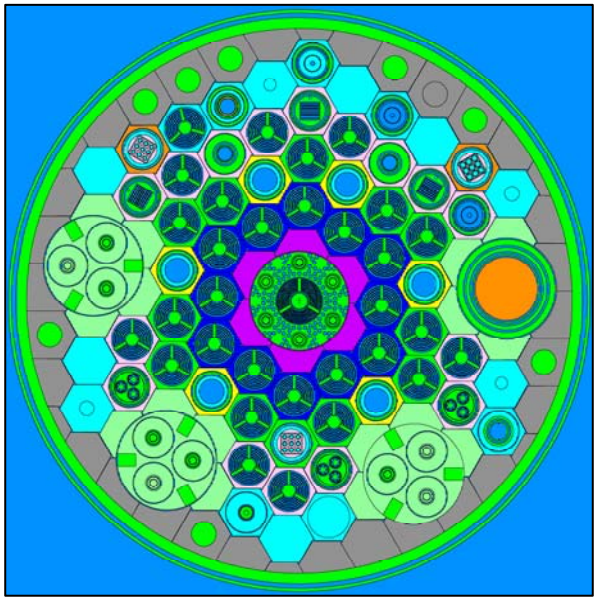


Figure 5. 3-D heterogeneous model used for the whole core depletion calculations by MCNPX 2.7.D2.

The depletion/burn up capability in MCNPX 2.7.D2 is based on the 1 – D burnup code CINDER90 and Monte Burns. The MCNPX depletion process internally links the steady – state flux calculations in MCNPX with the isotopic depletion calculations in CINDER90. MCNPX runs a steady – state calculation to determine the effective multiplication factor k_{eff}, 63 – group fluxes and continuous energy reaction rates for (n,gamma), (n,f), (n,2n), (n,3n), (n,alpha) and (n,p), which are converted into one – group constants and used by CINDER90 to carry out the depletion calculations and to generate new number densities for the next time step. MCNPX takes those new number densities for the corresponding fuel cells and generates another set of fluxes, reaction rates. The process is automatic and repeats itself for each time step until the requested final time step. The calculated MCNPX 63 – energy group fluxes in combination with the inherent 63 – group cross sections of CINDER90 are used to determine the rest of the interaction rates, which are not calculated by MCNPX.

5. BURN UP CAPABILITIES OF PRESENTED METHODOLOGIES

5.1 Evolution of macroscopic cross sections

The evolutions of the macroscopic cross sections of the main actinides, burnable absorbers, and fission products are performed following the irradiation history of a standard BR2 fuel element during 5 standard, 3 weeks operating cycles at nominal power 56 MW (see Fig. 6). For the HEU fuel, the major contributions to the negative reactivity of the core are given by the burnable absorbers ^{10}B, ^{149}Sm and ^{135}Xe. Among the non dominant F.P. the most contributions come from ^{103}Rh, ^{147}Pm, ^{151}Sm, ^{152}Sm.

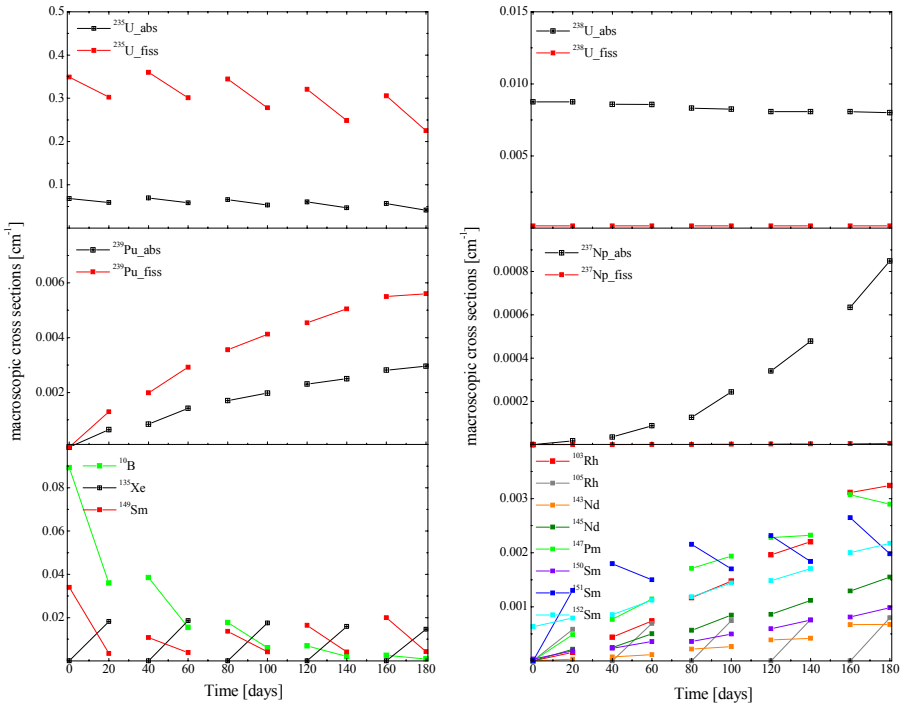


Figure 6. Evolution by MCNPX 2.7.D2 (CINDER90) of the macroscopic cross sections of the main actinides, burnable absorbers and dominant F.P. during 5 operating cycles (shown are data only for BOC and EOC).

The same isotopes are the major contributors to the negative reactivity of the LEU fuel (UMo) except for the burnable poison B-10, which will be replaced by the Cd-wires in the aluminum side plates in the LEU fuel assembly. The comparison of the evolution of the macroscopic cross sections for some 'key' isotopes in the HEU and LEU infinite lattices is given in Fig. 7. It is seen that the burn up rate of the major cadmium isotope, Cd-113 is higher than for B-10, however the macroscopic cross section of Cd-113 is about one order of magnitude less at the beginning of the irradiation.

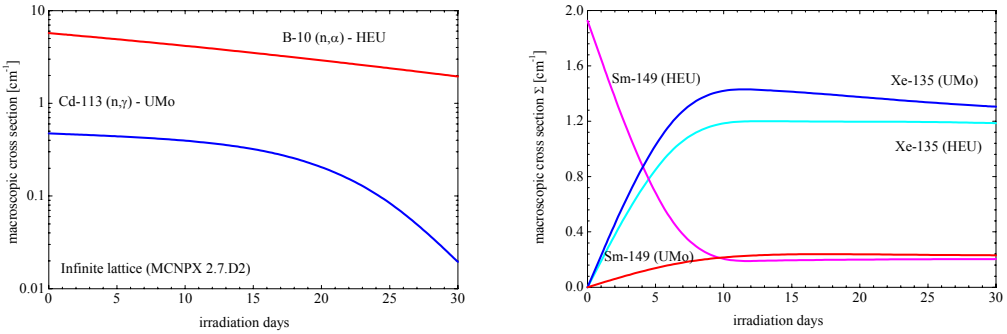


Figure 7. Comparison of the macroscopic cross sections of 'key' isotopes for continuous irradiation of the HEU fuel (90% U5, B_4C and Sm_2O_3 mixed in the fuel meat) and in the LEU fuel (UMo, Cd wires) in the infinite lattice.

5.2 Comparison of isotopic fuel densities

An important goal of this study is to compare the depletion and criticality capabilities of the new burn up Monte Carlo code MCNPX 2.7.D2 with those of the combined MCNPX & ORIGEN-S method. The evolutions of the isotopic fuel densities for irradiation of the HEU fuel element during 5 standard BR2 cycles are evaluated by CINDER90 and by ORIGEN-S and compared at Fig. 8, which shows a good agreement for the major fuel isotopes between the both methods.

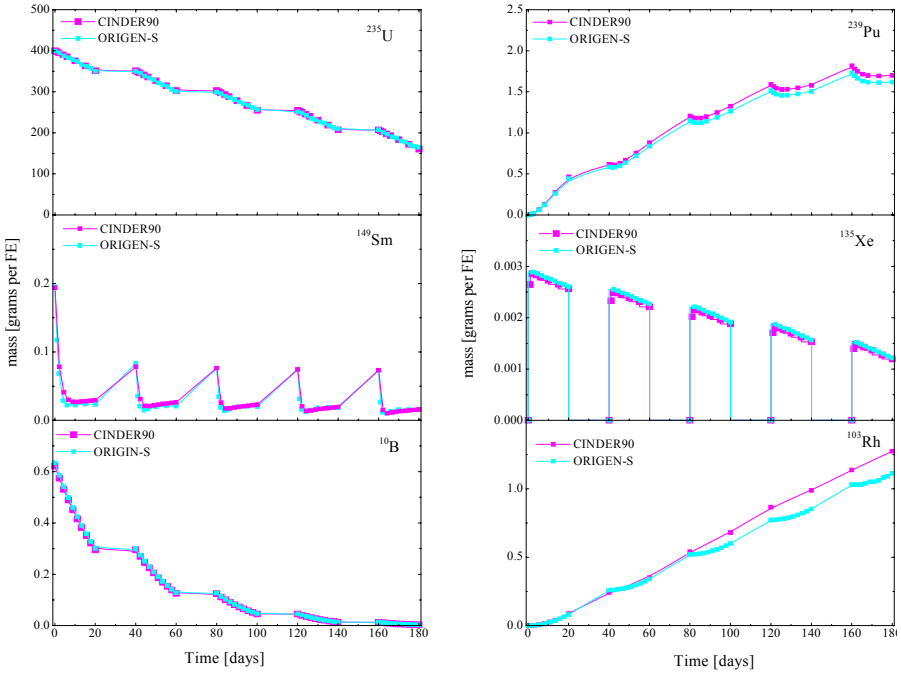


Figure 8. Time evolution of fissile isotopes and dominant F.P. in HEU (90% ^{235}U) fuel.

5.3 Comparison of k_∞ in infinite lattice

Another goal of the study is to check the criticality capabilities of MCNPX 2.7.D2. For this purpose simulation calculations of the k_∞ are performed for the continuous irradiation of the fuel in the infinite lattice. The comparison of the k_∞ in the HEU infinite lattice, calculated by the different methodologies is given in Fig. 9, which shows a very good agreement between MCNPX & ORIGEN-S method and MCNPX 2.7.D2. We should note here, that the calculations of the k_∞ are only performed with the purpose to check the consistency between the different depletion methodologies, however these calculations do not have significant physical meaning. In our studies, we are mostly interested to test the different methodologies on the real BR2 operating cycles, which is discussed in the next chapter.

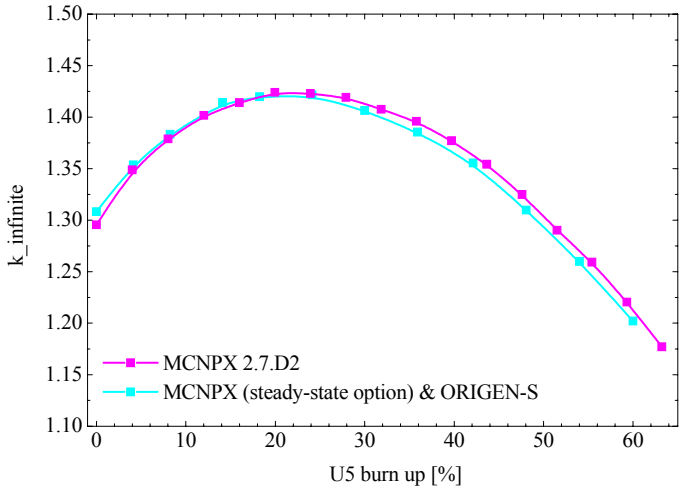


Figure 9. Comparison of k_∞, calculated in HEU infinite lattice by different depletion methods.

6. TESTING AND VALIDATION OF THE DEPLETION METHODOLOGIES ON THE REACTIVITY MEASUREMENTS AT BR2

The final goal of the study is to test the described 3 depletion methodologies on the reactivity measurements and CR motion during BR2 operation cycles. The combined MCNPX & ORIGEN-S method has been tested and validated on more than 30 operating cycles. The testing of the MCNPX & CINDER90 methodology started since the cycle 04/2009A.2 in July 2009 and presently is used for the management of the reactor core loads of the current BR2 cycles with the EVITA loop in the H1 channel. As it was discussed previously in Chapter 3 and Chapter 4, these two methodologies are still preferred for the management of the core load due to the low computational time and flexibility in using of the databases. In the 3rd method, the calculation procedure by MCNPX 2.7.D2 includes automatic calculation of k_{eff} and nuclide inventory by CINDER90 at each depletion time step, which takes about a week to calculate $\sim$ 10 steps in the whole core 3-D heterogeneous model.

The calculations by the described 3 methodologies are performed at the same critical position of the CR bank at BOC. After that the reactivity value for each time step (Fig. 10a, Fig. 10b) together with the calculated earlier differential CR worth (Fig. 10c) are used to evaluate the positions Sh of the CR bank during the operating cycle (Fig. 10d). The comparison of the reactivity evolutions and CR motion during the cycle determined by the 3 described methods has shown an acceptable agreement with the experimental values.

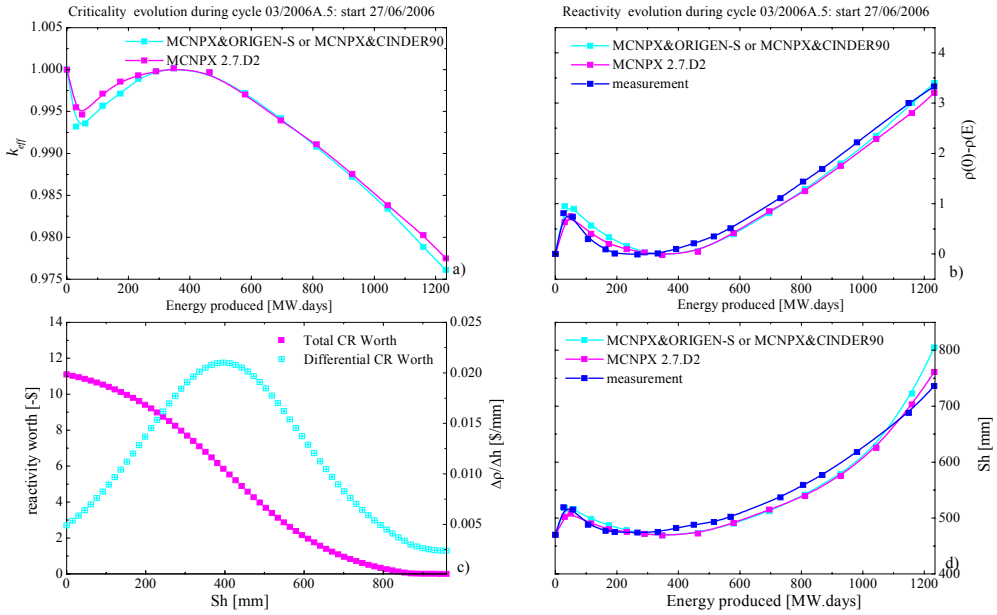


Figure 10. a. Evolution of k_{eff} ; b. Reactivity evolution; c. Total and Differential CR Worth; d. Motion of the Control Rods bank during cycle 03/2006A.5 of BR2.

7. SUMMARY

The applications of 3 depletion methodologies for the core load management of the BR2 reactor are discussed. The combined MCNPX & ORIGEN-S and MCNPX & CINDER90 methods use 2 preliminary prepared databases: (1) a database with depleted fuel in an infinite lattice; (2) a database with the power peaking factors, calculated by MCNPX following the standard irradiation history of the fuel element in the heterogeneous 3-D geometry whole core model. Until the present time and currently, these are the preferred methodologies for the reactor core load management. The third methodology represents fully automatic whole core depletion calculations by the Monte Carlo burn up code MCNPX 2.7.D2. This methodology is still in testing period and it is not used for the routine preparation of the reactor core load.

In the MCNPX & ORIGEN-S method the reaction rates for selected major fuel isotopes are calculated by MCNPX (any version) once – at BOC and introduced into ORIGEN-S to update the existing cross sections data libraries. The other two methods use the Monte Carlo burn up code MCNPX 2.7.D2 in which the steady-state 3-D flux calculations by MCNPX are automatically linked with the 1-D depletion calculations by CINDER90. In MCNPX 2.7.D2 the reaction rates of all fuel isotopes are re-calculated at each requested time depletion step, which takes a lot of computational time if use the whole core model. Therefore, a second MCNPX&CINDER90 method was developed similarly to the MCNPX&ORIGEN-S methodology. In this method, the evolution of the fuel composition is evaluated in an infinite lattice by CINDER90 inside MCNPX 2.7.D2 and stored in a database. Another database contains the power peaking factors in the fuel elements with varied burnup. MCNPX (any version) whole core model is prepared for each requested time depletion step extracting the relevant 3-D isotopic fuel distribution from the databases. The individual MCNPX simulations for the requested number depletion time steps are submitted in parallel on the SCK•CEN Fermi 4 cluster, which takes about 3 hours for 16 parallel jobs.

One major goal of this study was to compare the depletion capabilities of the different methodologies, which have shown very good agreement for the evolution of the isotopic fuel densities of the HEU fuel. The MCNPX & ORIGEN-S method has been validated on many BR2 cycles, the MCNPX & CINDER90 is presently testing on the current BR2 cycles with the EVITA loop and JHR fuel element (LEU, 27% U5, U_3Si_2) in the H1 channel of the BR2 reactor. The preferred method used for the management of the LEU fuel core will be the combined MCNPX & CINDER90 (MCNPX 2.7.D2) method.

8. REFERENCES

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[2] ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Terms, Oak Ridge National Laboratory, NUREG/CR-0200, Revision 5. ORNL/NUREG/CSD-2/V2/R5, March 1997.