Validation of MCNP&ORIGEN-S 3-D Computational Model for Reactivity Predictions During BR2 Operation

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A B S T R A C T

The Belgian Material Test Reactor (MTR) BR2 is strongly heterogeneous high flux engineering test reactor at SCK-CEN (Centre d'Etude de l'energie Nucléaire) in Mol at a thermal power 60 to 100 MW. It deploys highly enriched uranium, water cooled concentric plate fuel elements, positioned inside a beryllium reflector with complex hyperboloid arrangement of test holes. The objective of this paper is the validation of a MCNP&ORIGEN-S 3D model for reactivity predictions of the entire BR2 core during reactor operation. We employ the Monte Carlo code MCNP-4C for evaluating the effective multiplication factor k_{eff} and 3D space dependent specific power distribution. The 1D code ORIGEN-S is used for calculation of isotopic fuel depletion versus burn up and preparation of a database (DB) with depleted fuel compositions. The approach taken is to evaluate the 3D power distribution at each time step and along with DB to evaluate the 3D isotopic fuel depletion at the next step and to deduce the corresponding shim rods positions of the reactor operation. The capabilities of the both codes are fully exploited without constraints on the number of involved isotope depletion chains or increase of the computational time. The reactor has a complex operation, with important shutdowns between cycles, and its reactivity is strongly influenced by poisons, mainly ^{3}He and ^{6}Li from the beryllium reflector, and burnable absorbers ^{149}Sm and ^{10}B in the fresh UAl_x fuel. Our computational predictions for the shim rods position at various restarts are within 0.5\$ (β_{eff}=0.0072).

I. INTRODUCTION

In the last two decades many efforts have been directed toward development of methods for performance of criticality credit analysis. Generally, all methods combine a code, which calculate the evolution of the isotopic fuel depletion with a code that evaluates the reactivity k-eigenvalue, using Monte Carlo or deterministic methods. An advantage of the deterministic transport and diffusion methods, which are successfully applied for reactivity predictions in many commercial (PWR, BWR, LWR) and test reactors all over the world (ATR, HIFR, BR2, OSIRIS, JHR, etc.) is the simplicity of use, at the same time the disadvantages are the model approximations and the limited application to non-regular array of fuel cells. The BR2 core configuration is composed from big number skew beryllium prisms, loaded with fresh and burnt fuel elements, control rods and experimental devices in test holes, which are arranged in a twisted hyperboloid bundle. The detailed modeling of the hyperboloid core and especially the explicit modeling of the heterogeneous fuel elements only with a precise Monte Carlo code like MCNP can eliminate any homogeneous effects.

Recently, various schemes of coupling Monte Carlo codes (KENO, MCNP) with depletion codes (ORIGEN2, ORIGEN-S, CASMO, WIMSD4) have been used for reactor physics analysis [1], [2], [3] (see Fig. 1). The developed at INEEL combined method MCWO [4] and other similar methods as MCB [5], can be successfully implemented for solving of local space dependent problems, for example for detailed power distribution in MOX irradiated fuel rods. However the large increase of computational time, due to the introduced depletion chains into the 3D space dependent Monte Carlo analysis, constraints the application of such methods for criticality calculations of the entire core. Such methods are more suitable for calculation of k-infinity, as it is done in [6], [7], [8] (in latter the developed in INEL combined method MOCUP [9] was implemented) or of k_{eff} of critical experiments [10].

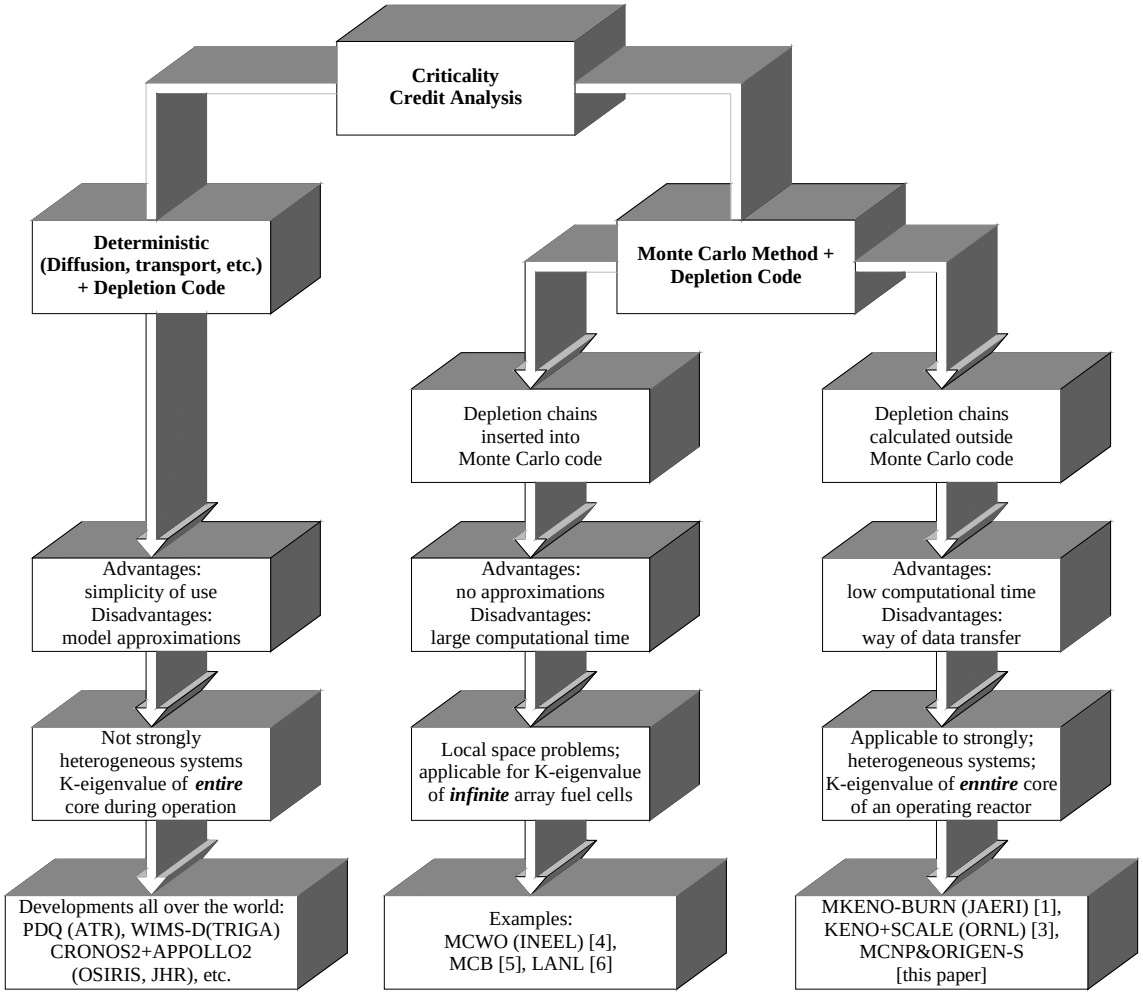


Fig. 1. A general overview of linking Monte Carlo and deterministic methods with depletion codes.

The objective of this paper is the validation of a calculation model for performance of criticality calculations of the entire core of an operating, strongly heterogeneous test reactor

like BR2. The main method for BR2 core management uses a perturbation model based on diffusion method [11]. This method is found satisfactory concerning reactivity evolution during operating cycle. However, strong accurate criticality calculations of the full 3D heterogeneous core have never been performed previously. The methods for shim rods predictions of the stationary core at BOC are based on comparison with measured reactivity components in a previous cycle.

Our purpose is development of a model, which allows:

(1) To evaluate theoretically reactivity evolution of entire reactor core.

(2) To evaluate the shim rods predictions during operation with accuracy within 0.5$.

(3) To perform quickly (in a couple of days) our evaluations before the start of the operating cycle.

The necessity of (3) originates from the requirements for irradiation of the different engineering devices and therefore dynamic changing of the load before start of the reactor.

To reach this goal we utilize a combination of MCNP and ORIGEN-S. The depletion chains are calculated outside the Monte Carlo analysis, which allows saving significant computational time.

Our methodology includes the following steps:

Preparation stage:

(1) MCNP modeling of 3D actual twisted hyperboloid heterogeneous reactor core.

(2) Evaluation with ORIGEN-S of isotopic fuel depletion versus fuel burn up and preparation of a database (DB) with large number isotopic fuel compositions, each containing 80 depleted isotopes.

Executive stage – performed before start of an operating cycle:

(1) Calculation with MCNP of the average fission power in each fuel element, which along with ORIGEN-S predicts the depletion step.

(2) Calculation with MCNP of 3D power peaking factors K_V in each fuel element for a few positions of the control rods. One K_V –distribution is used for evaluation of 3D power distribution in several depletion steps during the cycle.

(3) Along with DB – evaluation of 3D isotopic fuel depletion for each depletion step, using the 3D power distribution in the previous step.

(4) Shim rods predictions at various depletion and restart steps of reactor operation.

II. DESCRIPTION OF BR2 REACTOR

The Belgian Material Test reactor (MTR) BR2 is strongly heterogeneous high flux engineering test reactor at SCK-CEN in Mol. The reactor was designed in 1957 by Nuclear Development Corporation of America – White Plains (NY-USA). Routine operation of BR2 started in January 1963 [12]. The reactor is cooled and moderated by light water in a compact HEU core, positioned in and reflected by a beryllium matrix. The beryllium matrix is an assembly of a big number of irregular hexagonal prisms, each with a cylindrical test hole which forms the channel in the core region. The central 200 mm hole (H1 channel) containing the central beryllium plugs is vertical. All other beryllium assemblies (84 mm hole or 200 mm hole) are skew and form a twisted hyperboloid bundle around the central H1 channel (Fig. 2). The main design parameters of BR2 are presented in Table I.

BR2 REACTOR

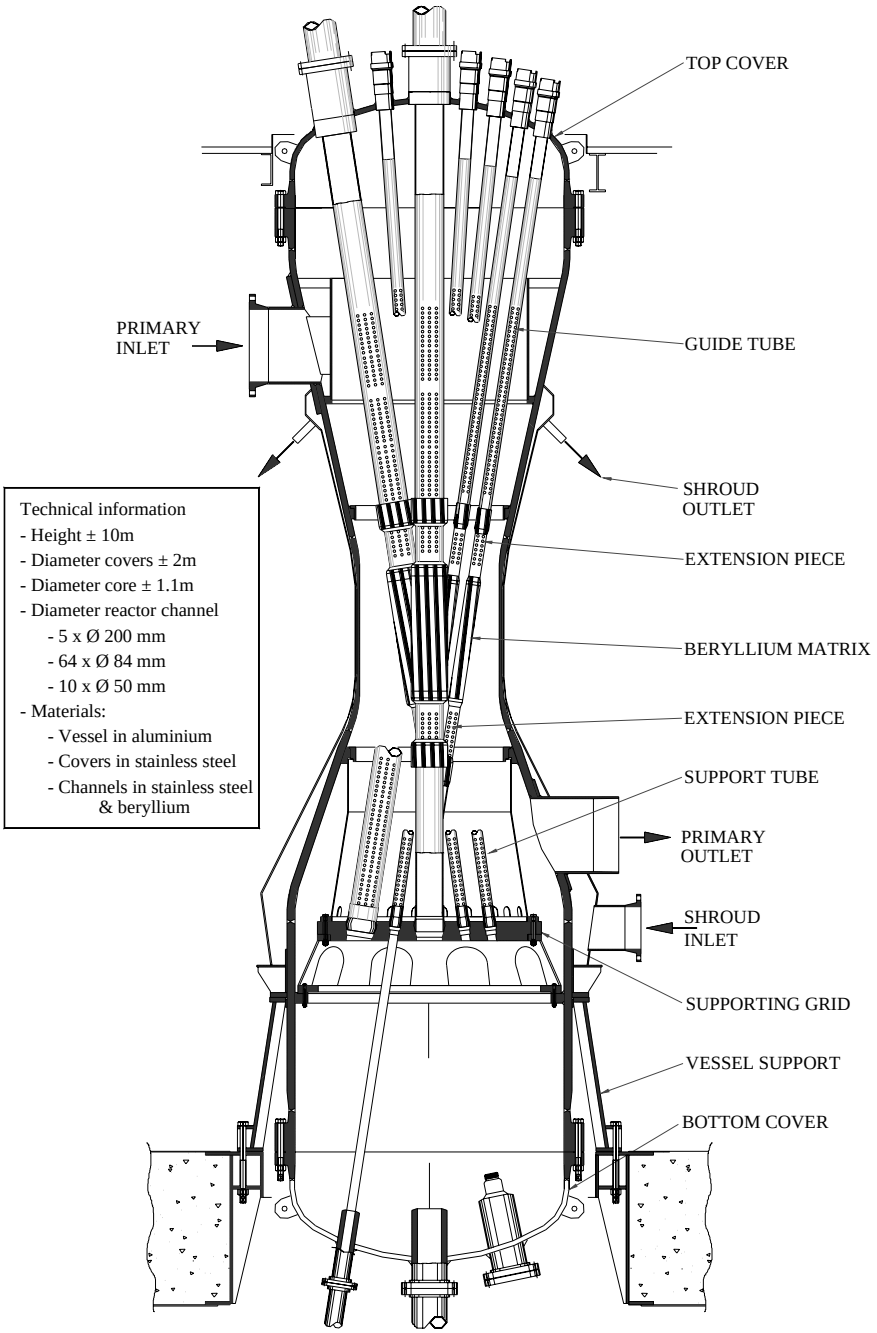


Fig. 2. BR2 Reactor.

Table I. Main design parameters of BR2.

Reactor type	Tank-in-pool		
Primary flow	Water in closed loop		
Speed between fuel plates, m/s	10.4		
Temperature , T of inlet – T of outlet, °C	40 – 57		
Fuel plate temperature, °C	150		
Power level (MW)	50 to 100		
Maximum heat flux (W/cm^2)	470		
Maximum neutron flux ($n.cm^{-2}.s^{-1}$)	1.2×10^{15} (E_n<0.5 eV), 8.4×10^{14} (E_n>0.1 MeV)		
Number of fuel elements in the equilibrium core	32÷33, fresh and burnt fuel elements, with varied mean fuel depletion between 0% and 50%		
Lattice hexagon pitch (cm)	9.64438		
Parameters of a standard fuel element (FE) (Common characteristics for all used fuels)			
Number of MTR-type fuel plates	18 plates (6 concentric tubes)		
Channel diameter (cm)	8.42 ± 2		
Plate length (cm)	97.0		
Fuel length (cm)	76.2		
Fuel meat thickness (cm)	0.051		
Plate thickness (cm)	0.127		
Al clad (cm)	0.038		
Water gap between plates (cm)	0.3		
Fuel types	UAl_x	UAl_x	UAl_x
^{235}U enrichment (wt%)	93%	72%	73%
^{235}U mass per FE (grams)	400	330	400
Total U mass per FE (grams)	430	458	553
Fuel density, gU/cm^3	1.3	1.3	1.7
Burnable poisons, homogeneously mixed into the fuel meat:			
Natural boron in B_4C (grams)	3.8	1.8	3.2
Natural samarium in Sm_2O_3 (grams)	1.4	1.3	1.4
Fuel consumption			
Mean depletion at elimination ($^{235}U+^{239}Pu$)	50%	42%	55%
Cycle length (day)	21÷28	21	21÷28
Number of batches	4÷3	2.8	5÷4
Control Rods			
Type	Shim safety and regulating rods		
Absorber material	Cadmium		
Number of safety shim rods	6 operational +2 reserve		
Number of regulating rods	2 shims (only 1 operational)		

III. METHODOLOGY

We use a combination of the 3D Monte Carlo code MCNP-4C [13] with the 1D depletion code ORIGEN-S [14]. Our methodology includes two stages: preliminary (preparation) stage and executive stage for performance of routine reactivity calculations before start of an operating cycle.

III.A. Preparation stage.

III.A.1. Development of the full scale 3D heterogeneous geometry model with MCNP.

The full-scale 3-D heterogeneous geometry model is developed using the Monte Carlo code MCNP-4C and presented at Fig. 3. The model includes a sophisticated 3D description of the actual twisted hyperboloid reactor core, formed from skew beryllium prisms with individual orientation of the loaded fuel elements and engineering devices inside test holes. Each one of the 6 fuel plates of all 32 fuel elements is divided into axial zones by each 6 cm. The hot plane of the outer fuel plate of each element is divided into azimuth sectors by each 5°. The total number of the spatial cells with varied fuel depletion in the model is 4600.

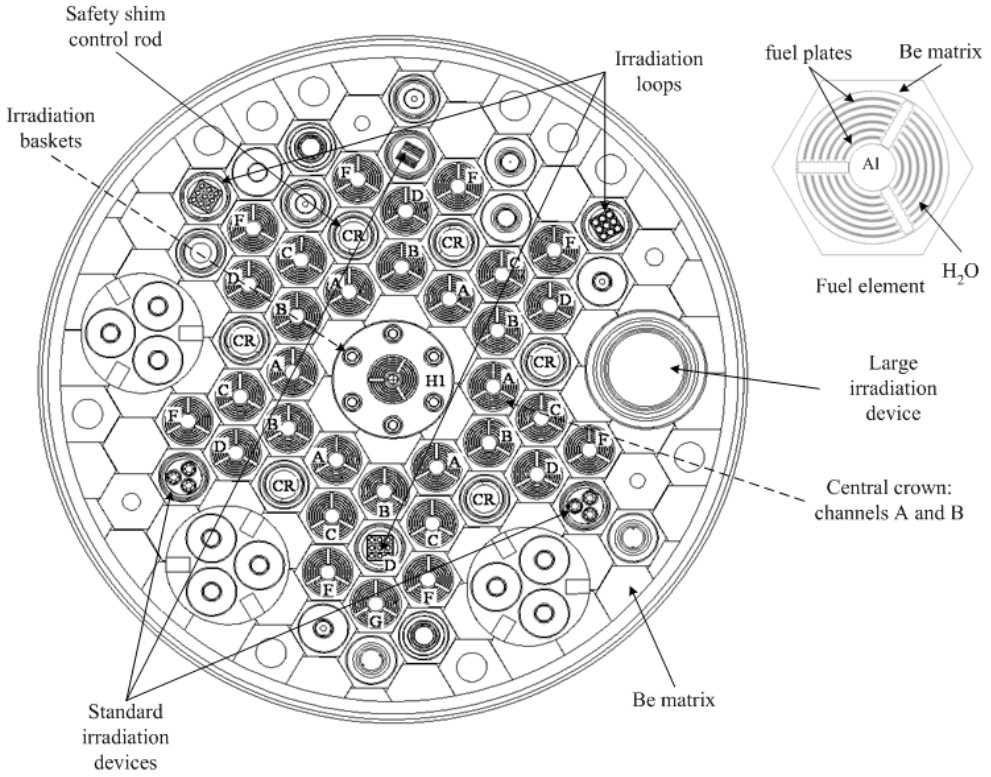


Fig. 3. MCNP whole core geometry model of BR2 – cross section at Z=+25 cm. A typical reactor core load contains 6 fresh and 27 burnt fuel elements with mean fuel depletion and mean fission power in each fuel type channel as follows: A – 12% (2.5 MW), B – 30% (2.1 MW), C– 0% (1.8 MW), D – 42% (1.4 MW), F,G (1.0 MW) – 44%, H1/Central – 34% (2.8 MW). Total thermal reactor power $P_r \sim 57$ MW.

III.A.2. Evaluation of isotope fuel depletion versus fuel burn up with ORIGEN-S.

The initial composition of a fresh, highly enriched uranium fuel (90% ^{235}U content) and the average fission power of a fuel element we introduce into ORIGEN-S and evaluate the depletion of various isotopes versus fuel burn up. The fissions in HEU fuel are caused mainly by ^{235}U, the contribution from all other U and Pu isotopes is negligible (0.2÷0.4% from the total fissions). Therefore in this paper we use the following definition for a fuel burn up: a fuel burn up is equivalent to the produced fission energy by 1 kg ^{235}U in fresh HEU fuel [MW.days/kgU5]. With β^5 [X%] we notate *isotopic* fuel composition, corresponding to X% ^{235}U depletion (i.e. a number of X atoms ^{235}U per cent removed from the initial fuel). The dependence of ^{235}U depletion, β^5 [X%] versus the fuel burn up is linear (Fig. 4):

$$\beta^5[\%] = \frac{M^5(0) - M^5(T)}{M^5(0)} \approx const \times \frac{T \times P}{M^5(0)} = \frac{const \times T}{c^5(0)} \times \frac{P}{V} \qquad (1)$$

where: P [MW] is the average fission power of a fuel element, V [cm^3] – volume of the fuel meat, T [days] is the irradiation time; $M^5(0)$ [kg] is the initial mass of ^{235}U in a fresh fuel element; $M^5(T)$ [kg] – the residual ^{235}U mass after irradiation at time T; $c^5(0)$ [g.cm^{-3}] is the initial ^{235}U density in a fresh fuel element; the $const=tg\alpha$ [kgU5.MW^{-1}.days^{-1}] is determined from the slope of the graph. Using ORIGEN-S we evaluate the isotopic fuel depletion by steps, equivalent to 1% ^{235}U depletion. Each fuel composition β^5[X%] for different ^{235}U depletion: β^5[1%], β^5[2%], …., β^5[89%], β^5[90%] contains about 100 fissile, 30 non-fissile isotopes (light elements) and fission product inventory of about 800 isotopes. From this we select the dominant 80 nuclides to form the isotope depleted fuel compositions, which we store in a database (DB).

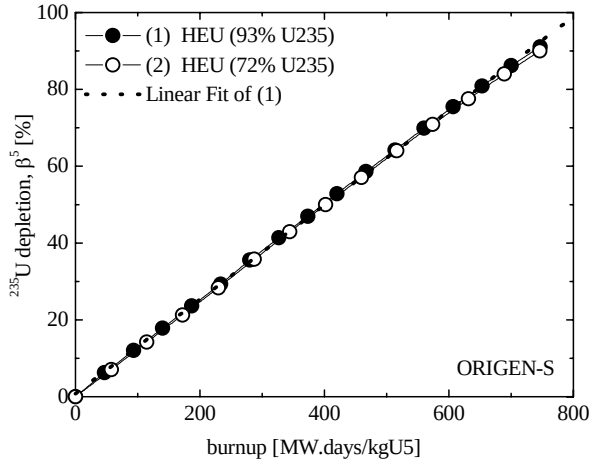


Fig. 4. ^{235}U depletion versus fuel burn up of HEU fuel.

Equation (1) is used to predict the mean isotopic fuel depletion $\overline{\beta_j^5}$ in a fuel element, the average fission power of the fuel element, being determined by MCNP calculation. The 3D space distribution of isotopic fuel depletion in a fuel element j in the fuel zone (r_m, z_n) is simply calculated as:

$$\beta_j^5(T_i, r_m, z_n) = \overline{\beta_j^5}(T_i) \times K_V^j(T_i, r_m, z_n) \tag{2}$$

Where:

$$K_V^j(T_i, r_m, z_n) = \frac{\int_{r_m} dr \int_{z_n} dz \int_{\varphi=0}^{2\pi} p_j(T_i, r, z, \varphi) d\varphi \Big/ \int_{r_m} dr \int_{z_n} dz \int_{\varphi=0}^{2\pi} d\varphi}{P_j(T_i)/V_j} \tag{3}$$

is the power peaking factor in fuel zone (r_m, z_n); 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; $p_j(T_i, r_m, z_n, \varphi)$, is the fission power in the spatial segment (r_m, z_n) inside the fuel element and $P_j(T_i)$ is the total power in the fuel element j; V_j - total volume of the fuel element; φ is the azimuth angle in the fuel element. The 3D isotopic fuel depletion in a fuel element j, located in any BR2 channel at arbitrary time step T_i^k during operating cycle k can be calculated, using the information about the isotopic fuel depletion and the power distribution in the previous time step $T_{i\text{-}1}{}^k$ and at $T_{eoc}{}^{k\text{-}1}$ in previous cycle k-1:

$$\begin{aligned} &\beta_j^5(T_i^k, r_m, z_n) = \overline{\beta_j^5}(T_{eoc}^{k-1}) \times K_V(T_{eoc}^{k-1}, r_m, z_n) + \\ &+ \left[\overline{\beta_j^5}(T_{i-1}^k) + \frac{tg\alpha \times (T_i^k - T_{i-1}^k)}{c_0^5} \times \frac{P_j(T_{i-1}^k)}{V} \right] \times K_V(T_{i-1}^k, r_m, z_n) \end{aligned} \tag{4}$$

III.B. Executive stage – reactivity evolution and shim rods predictions.

The approach taken is to evaluate the 3D relative power distribution at each depletion step and in combination with DB with stored isotopic depleted fuel compositions to evaluate the 3D isotopic fuel depletion at next time step, allowing for the previous irradiation history of the fuel element. Parallel to this we evaluate the poisoning of the beryllium matrix, following the full previous irradiation history of the matrix from 1997.

III.B.1. Evaluation of time and 3D space dependent isotopic fuel depletion.

Using MCNP we calculate the average fission power P [MW] in each fuel element at the end of a previous operating cycle k-1 and from (1) determine the mean isotopic fuel depletion $\overline{\beta_j^5}(T_{eoc}^{k-1})$ in each FE. We evaluate with MCNP the 3D space dependent specific power distribution (P/V) in each fuel element for the height of the shim rods $Sh(T_{eoc}^{k-1})$ and along with (2), (3) and DB evaluate the 3D space isotopic fuel depletion at EOC k-1. This distribution is used as initial for the beginning of the next cycle k, but taking into account the accumulation of ^{149}Sm and the disappearance of ^{135}Xe after the shutdown. We evaluate/predict the critical height of the shim rods at $Sh(T_{boc}^k)$ for the chosen core loading by adjusting their position in the core for k_{eff} = 1.0.

Inside cycle k we calculate the average fission power in each fuel element at T_{boc}^k and predict the average isotope fuel depletion at next time step T_i^k with (1). The calculated with MCNP 3D power peaking factors $K_V(T_{boc}^k)$ for $Sh(T_{boc}^k)$ we use in (4) and in combination with the database (DB) evaluate/predict the 3D isotopic fuel depletion in each

FE at T_i^k, which is used as initial distribution for the next time step $T_{i+1}{}^k$. We calculate the critical height of the shim rods $Sh(T_i^k)$ at depletion step T_i^k. Further we evaluate the average power/3D power distribution at the next step and along with our database (DB), prepared with ORIGEN-S, evaluate the 3D isotopic fuel depletion and the shim rods predictions at the various depletion and restart steps of reactor operation. The following time steps are chosen for evaluation of the 3D isotopic fuel depletion and the critical heights of the shim rods: T=0, 0.5 day, 1 day, 2, 3, 4, 5, 6, 8, 10, 12, 14, 16, 20, 24, 28 days.

The 3D power peaking factors are calculated for a few heights of the shim rods during operating cycle, e.g. for Sh(BOC)=400/500 mm, Sh(MOC)=600/700 mm and Sh(EOC)=800/900 mm, depending on the travel of the Control Rods bank during the operating cycle. The position of the shims influences on the axial symmetry of the specific fission power and the 3D profile of isotopic fuel depletion. Calculations of distributions of the power peaking factors K_V, following typical irradiation history of a fresh fuel element, used in several operating cycles, are performed with MCNP and demonstrated at Fig. 5.

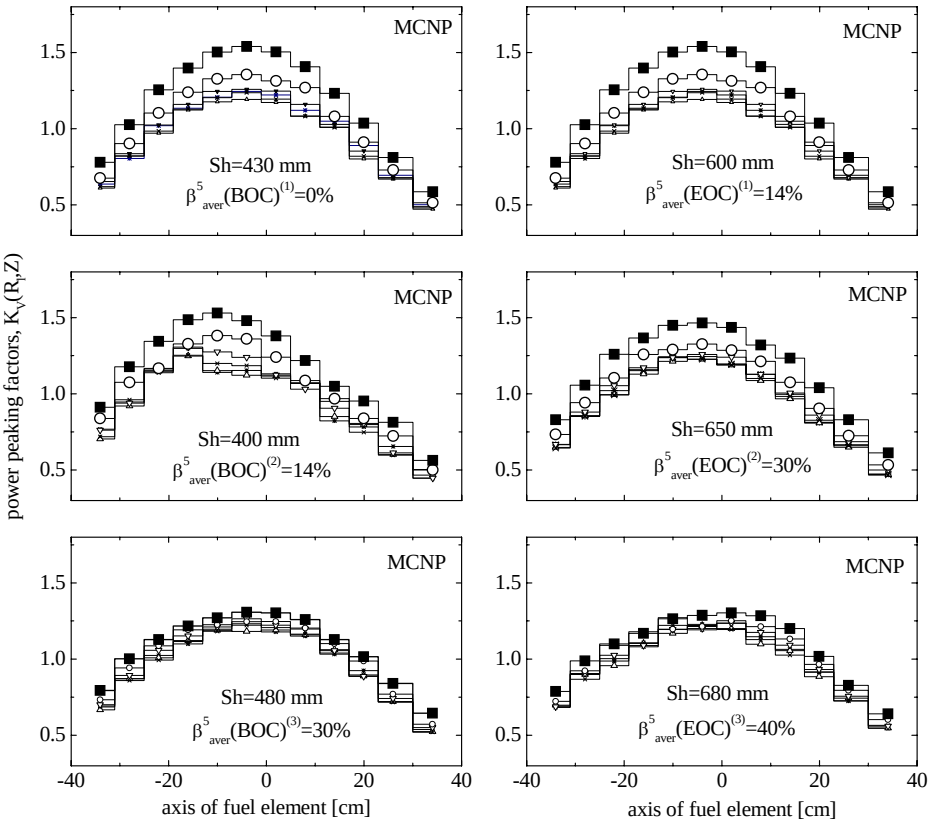


Fig. 5. Power peaking factors in typical irradiation history of a fresh fuel element, used in several BR2 operation cycles: $K_V(R_l,Z)$ are axial power peaking factors in the fuel plates R_l of the fuel element, l=1,..6; $\beta^5_{aver}(BOC)^k$, $\beta^5_{aver}(EOC)^k$ are the mean fuel depletion in the fuel element at BOC and EOC of cycle k, k=1,2,3; Sh – critical height of shim rods. Notation for the curves:

—✳— 1st (inner) fuel plate —△— 2nd fuel plate —×— 3rd fuel plate
—▽— 4th fuel plate —○— 5th fuel plate —■— 6th (outer) fuel plate

III.B.2. Evaluation of time and 3D space dependent poisoning of beryllium matrix/reflector

For various restarts and depletion steps we evaluate the time and 3D space distribution of the products of the poisoning of the beryllium reflector ^{3}He, ^{6}Li and ^{3}T. We calculate with MCNP the 3D space dependent reaction rates (n,α) on ^{9}Be, (n,t) on ^{6}Li and (n,p) on ^{3}T at EOC k-1. These reaction rates are introduced into a system of differential equations for the time evolution of ^{6}Li, ^{3}He and ^{3}T atomic concentration during shutdown, which are used as initial concentrations for BOC of the next operating cycle k. At BOC k we calculate with MCNP again the 3D reaction rates for the new core load and evaluate the burning of ^{3}He and the concentrations of the remaining poisons ^{6}Li and ^{3}T during the cycle and include into the MCNP model. Along with step III.B.1 we evaluate the multiplication effective factor k_{eff} at various depletion and restart steps. Schematically the preparation and the executive stages are presented at Fig. 6, 7 and 8.

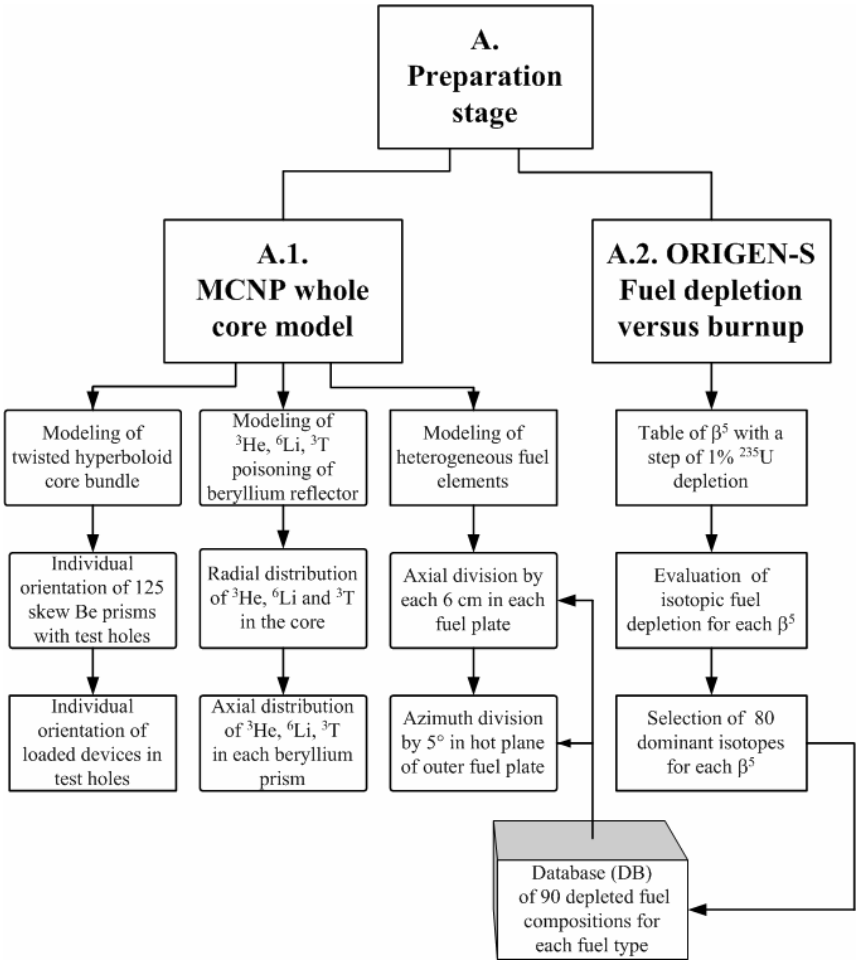


Fig. 6. Preparation stage: evaluation of isotopic fuel depletion with ORIGEN-S and preparation of a database (DB) with depleted isotopic fuel compositions.

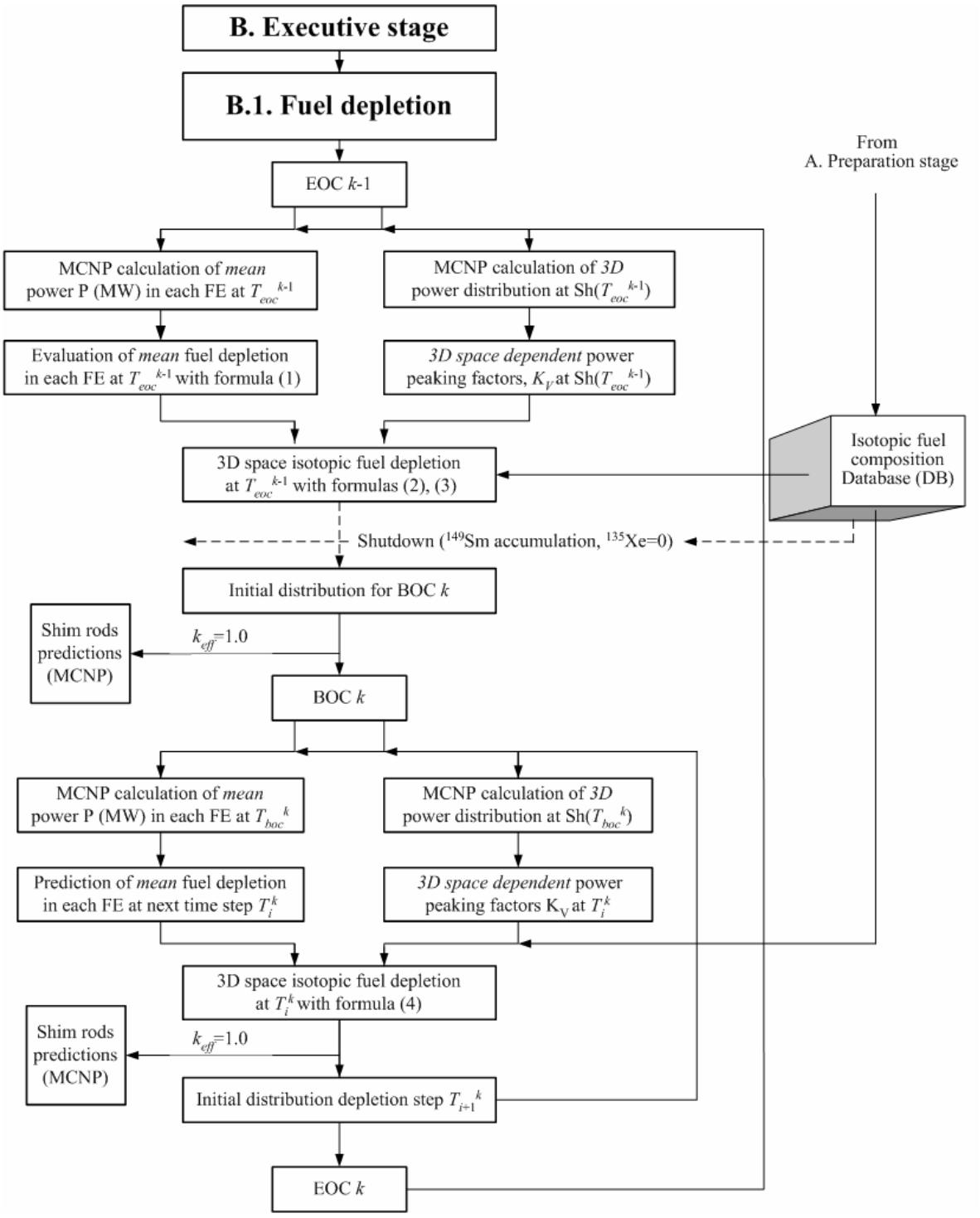


Fig. 7. Executive stage B.1.: evaluation of time and 3D space dependent fuel depletion.

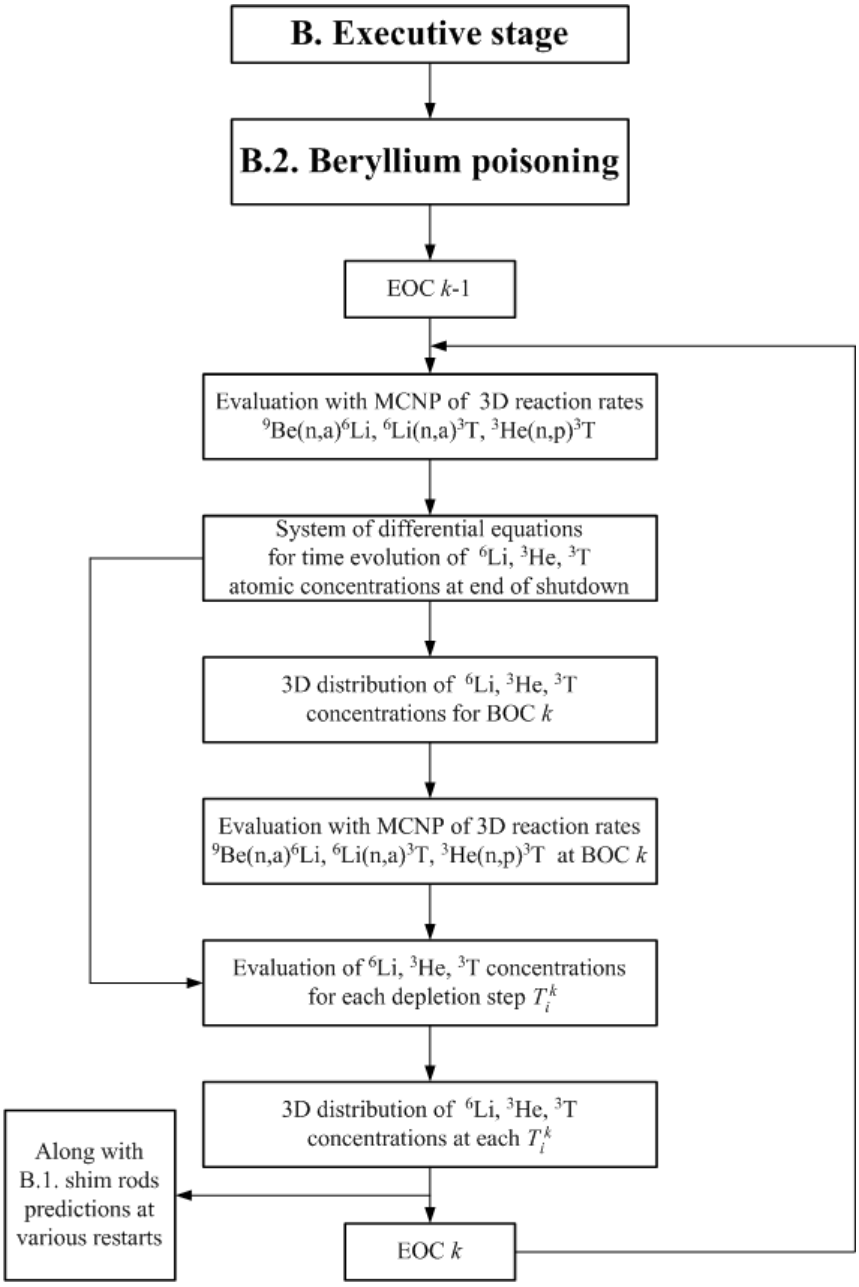


Fig.8. Executive stage B.2.: evaluation of time and 3D space dependent beryllium poisoning.

IV. CRITICALITY CALCULATIONS

IV.A. REACTIVITY-CONTRIBUTING COMPONENTS

The main reactivity-contributing components in BR2 for the reactor core load, given at Fig. 3, are listed in Table II. All considered contributors are included into net reactivity evaluations, presented in this paper.

Table II. Reactivity-contributing components in BR2. MCNP calculations.

Reactivity-contributing component (fuel cells)	BOL	EOL
^{10}B as burnable poison in 6 fresh FE and 24 burnt FE	–8.2 $	–3.5 $
Samarium as burnable poison in 6 fresh FE	–1.3 $	0.0
^{149}Sm as burnable poison and fission product (included saturation after shutdown, T~20 days)	–4.0 $	–1.1 $
(^{151}Sm+^{152}Sm) as fission product and burnable poison	–0.8 $	–1.7 $
^{135}Xe (saturated)	0	–3.8 $
The rest of F.P.	–1.6 $	–3.7 $
Fuel depletion (U+Pu+Np)	0	–2.8 $
Absorption in H_2O (fuel cells)	–10.5 $	–10.0 $
Reactivity-contributors (remaining, non fuel cells)		
^{3}He – product of the poisoning of the beryllium matrix	–4.5 $	–0.9 $
^{6}Li – product of the poisoning of the beryllium matrix	–2.5 $	–2.8 $
Ir-192 samples	–3.2 $	–2.8 $
Absorption in H_2O in the remaining reactor	–2.0 $	–2.0 $
Experimental devices	–0.7 $	–0.6 $
Control Rods worth of 6 Shim Rods[1] for Sh(BOC)=510 mm	–4.2 $	–4.5 $

Reactivity dollars

For BR2 one dollar (1$) is equal to a reactivity of β_{eff}=0.0072, including the fraction of delayed and photo neutrons. This value has been determined experimentally [15]. The fraction of photo neutrons in the reactor core is β_{ph}(core)≈0.00018 and in the beryllium reflector β_{ph}(Be)≈0.00045, thus the total photo neutron source is β_{ph}(total)≈0.00063 according to [15].

Control rods position

All control rods travel separately, the minimum height of the 6 shim rods fully inserted is Sh=0 mm, the maximum height of the rods fully withdrawn is Sh=900 mm. A typical negative reactivity worth of the six shims for ΔSh=0÷900mm is R0=–(13±1) $. The curves of the total control rods worth for different reactor core loads are calculated with MCNP

[1] The worth between Sh=510 mm and Sh=900 mm.

and present at Fig. 9. The calculated with MCNP values of the total worth of 6 shim rods for ΔSh=0÷900 mm were compared with the experiment: Δρ(EXP.)–Δρ(MCNP) ≤ (0.4±0.1) $ for different measurements at BOC.

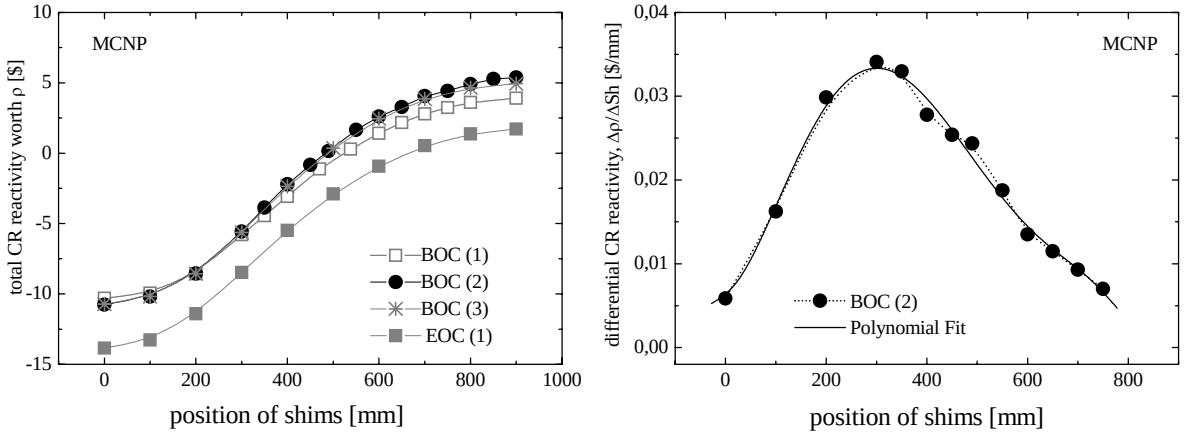
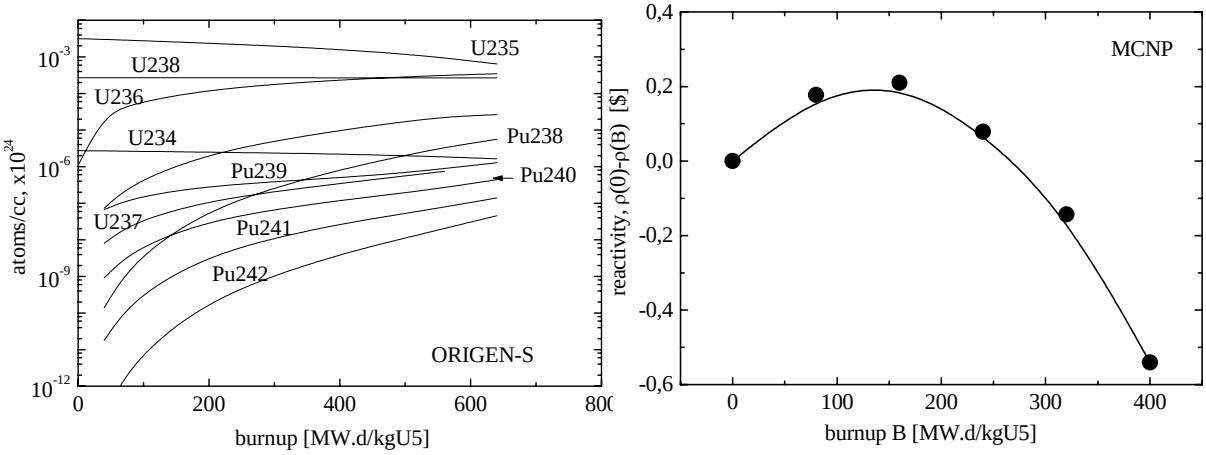


Fig. 9. Total and differential reactivity worth of 6 shims for typical reactor core loads.

Fuel depletion.

The actinide libraries of ORIGEN-S include depletion chains for 100 fuel, transplutonium and decay daughter nuclides. From all depleted actinides we selected 10 fissile PU and U isotopes to include into the MCNP model. The evolution of the atomic concentrations of the used isotopes versus burn up is evaluated assuming continuous irradiation in the reactor. The sensitivity of reactivity of a standard fuel element versus ^{235}U depletion in a fuel element is given at Fig. 10b. The specific profile is due to the presence of the burnable poisons B_4C and Sm_2O_3 in the fuel meat of a fresh fuel element.



a) b)

Fig. 10. a) Depletion of all fissile actinides in HEU fuel versus fuel burn up; b) Sensitivity of reactivity of a standard fuel element versus fuel burn up.

Burnable poisons B_4C and Sm_2O_3 (^{149}Sm as B.P. and F.P.) in the fuel meat

A standard BR2 fuel element contains HEU under the form of UAL_x matrix with burnable poisons B_4C and Sm_2O_3 homogeneously mixed into the fuel meat of a fresh fuel element. ^{149}Sm is used to reduce the control rod motion at the start-up until ^{135}Xe and ^{149}Sm have reached equilibrium concentrations [16]. The time evolution of the atomic concentration of the burnable poison ^{10}B in a standard fresh fuel element is evaluated with ORIGEN-S, assuming continuous irradiation in the reactor without shutdown and presented at Fig. 11a. The reactivity effect due to ^{10}B absorption versus fuel burn up during an operating cycle for the reactor core load, given at Fig. 3 is calculated with MCNP and shown at Fig. 11b.

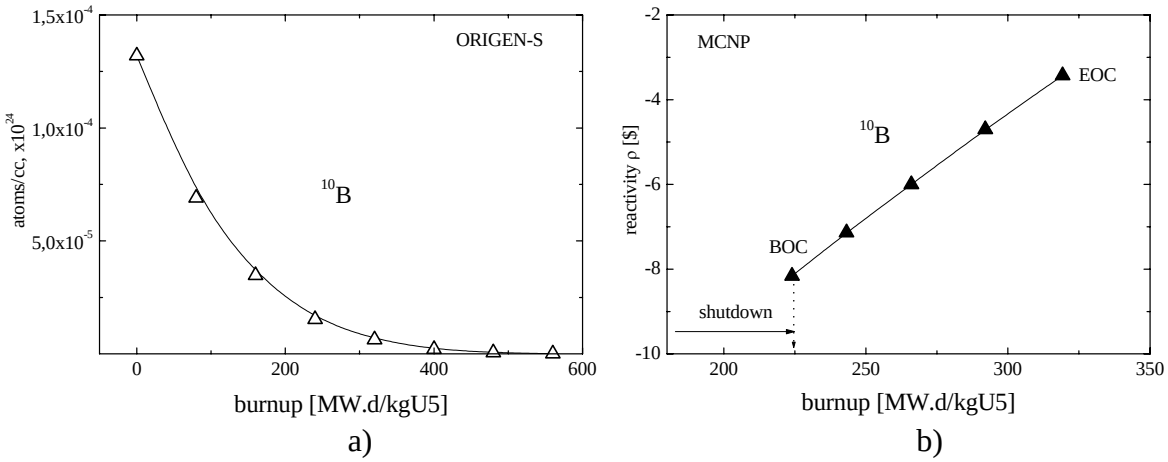
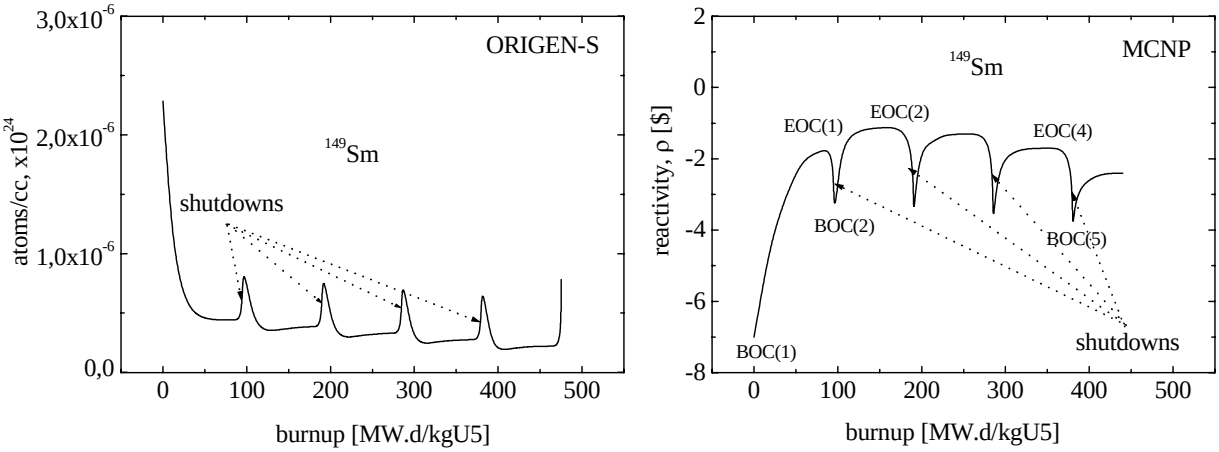


Fig. 11. a) Evolution of the atomic concentration of the burnable poison ^{10}B in a standard fresh fuel element versus fuel burn up, assuming continuous irradiation in the reactor without shutdown; b) reactivity effect in the whole core due to ^{10}B absorption during typical BR2 cycle for the core loading, presented at Fig. 3.

During shutdown important reactivity losses occur due to accumulation of ^{149}Sm in the fuel meat, which is consumed in the first couple of days after the start of the reactor operation. The evolution of the ^{149}Sm atomic concentration is evaluated with ORIGEN-S, following typical irradiation history of a fresh fuel element, used in several operating cycles and taking into account the accumulation of ^{149}Sm during shutdown (see Fig. 12a). The effect of ^{149}Sm absorption on reactivity variations in the whole fresh and burnt cores at various restarts is calculated with MCNP and shown at Fig. 12b. The evaluated atomic concentrations of ^{149}Sm during shutdown are included into the fuel depleted compositions, used for the reactivity calculations at BOC and stored in DB.

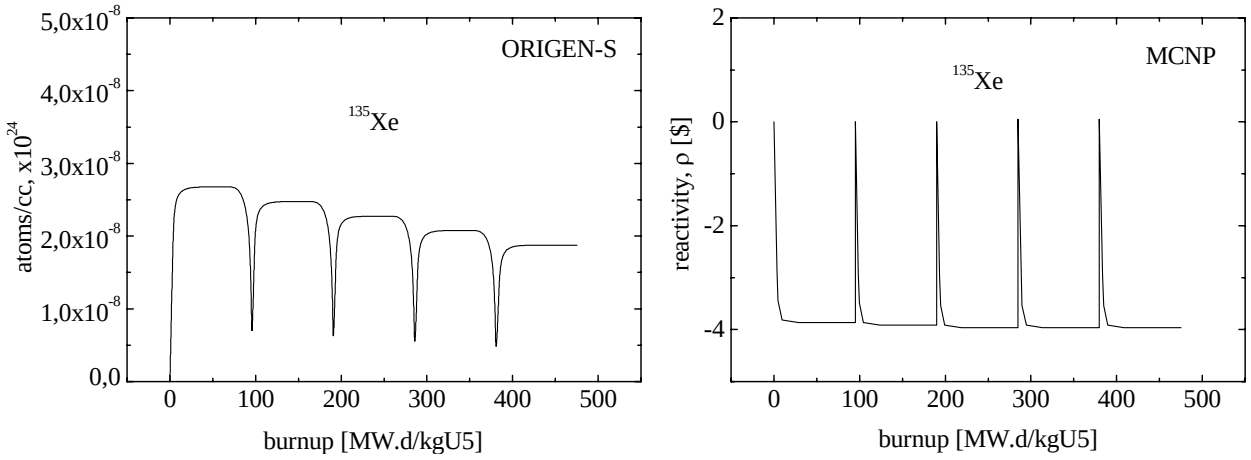


a) b)

Fig. 12. a) Evolution of the atomic concentrations of the burnable poison and fission product ^{149}Sm, following the irradiation history of a fresh fuel element, used in several BR2 cycles and taking into account the duration of shutdown between cycles; b) Reactivity effect due to ^{149}Sm absorption in the whole fresh and burnt cores at various restarts of the reactor and taking into account accumulation of ^{149}Sm during the shutdown.

Fission products poisoning.

The fission product library of ORIGEN-S contains depletion chains for about 800 nuclides. We select the dominant 70 from the output of ORIGEN-S and include into each fuel depleted composition, notated as β^5[X%], which is used in the MCNP model. The evolution of the atomic concentration of ^{135}Xe in a FE is evaluated versus burn up, allowing for the disappearance of ^{135}Xe during the shutdown (see Fig. 13a). The reactivity effect in the whole core due to ^{135}Xe absorption is evaluated using MCNP and presented at Fig. 13b.



a) b)

Fig. 13. a) Evaluation of the atomic concentration of ^{135}Xe in a FE , taking into account the shutdowns between the operating cycles; b) Evaluation of the reactivity effect in the whole core due to ^{135}Xe absorption versus burn up.

The atomic concentrations of the remaining fission products (excluding all samarium isotopes and ^{135}Xe) are evaluated versus burn up with ORIGEN-S, assuming continuous fuel irradiation in the reactor (see Fig. 14).

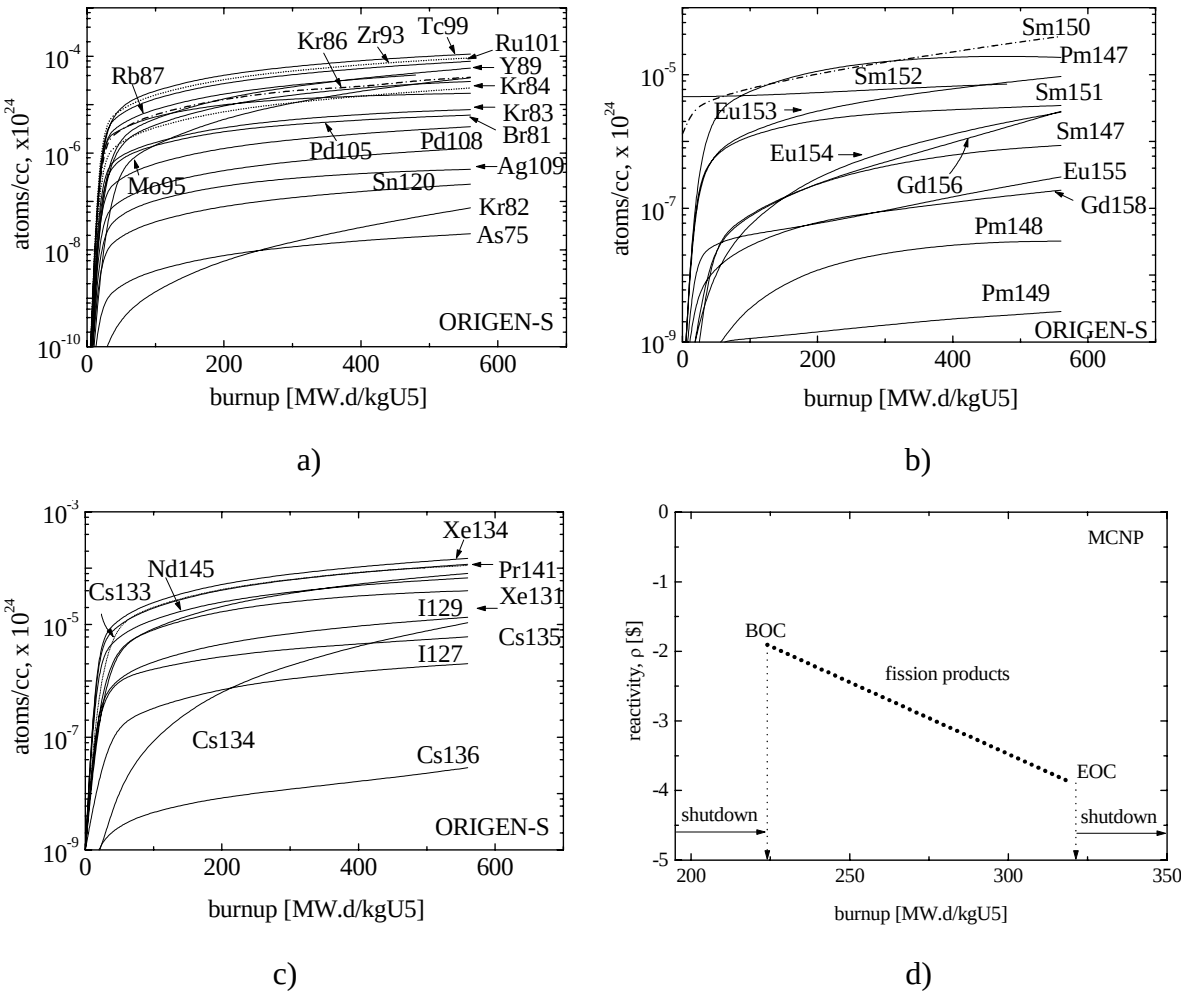


Fig. 14. (a,b,c) Evaluation of the atomic concentrations of the fission products, assuming continuous fuel irradiation in the reactor. d) Reactivity effect of fission products (excluding all samarium isotopes and ^{135}Xe) during typical operating cycle.

Poisoning of the beryllium matrix (reflector)

The poisoning effect of beryllium [17] appears after irradiation of Be-matrix with fast neutrons due to (n,α) – reaction on ^{9}Be and following transmutations into nuclides of ^{6}Li and ^{3}T according to the reactions:

$$n+{}^{9}Be\rightarrow{}^{6}He+{}^{4}He;\quad {}^{6}He\rightarrow{}^{6}Li+e^{-}+\nu_{e^{-}}\quad (T_{1/2}^{{}^{6}He}=0.808s) \tag{6}$$

$$n+{}^{6}Li\rightarrow{}^{4}He+{}^{3}T;\quad {}^{3}T\rightarrow{}^{3}He+e^{-}+\nu_{e^{-}}\quad (T_{1/2}^{{}^{3}T}=12.33years) \tag{7}$$

$$n+{}^{3}He\rightarrow{}^{3}T+p \tag{8}$$

The 3D space dependent reaction rates: (n,α) on ^{9}Be, (n,t) on ^{6}Li and (n,p) on ^{3}He are calculated with MCNP-4C. After that we introduce the reaction rates into the following

system of differential equations [18] and evaluate the time evolution of ^{6}Li, ^{3}He and ^{3}T atomic concentrations:

$$\frac{dN_{Be}(r,t)}{dt} = -N_{Be}(r)\int \varphi(r,E,t)\sigma_{n,\alpha}^{Be}(E)dE \ ,$$
$$\frac{dN_{Li}(r,t)}{dt} = N_{Be}(r,t)\int \varphi(r,E,t)\sigma_{n,\alpha}^{Be}(E)dE - N_{Li}(r,t)\int \varphi(r,E,t)\sigma_{n,\alpha}^{Li}(E)dE \ ,$$
$$\frac{dN_{T}(r,t)}{dt} = N_{Li}(r,t)\int \varphi(r,E,t)\sigma_{n,\alpha}^{Li}(E)dE + N_{He}(r,t)\int \varphi(r,E,t)\sigma_{n,p}^{He}(E)dE - \lambda_T N_T(r,t) \ , \qquad (9)$$
$$\frac{dN_{He}(r,t)}{dt} = -N_{He}(r,t)\int \varphi(r,E,t)\sigma_{n,p}^{He}(r,E)dE + \lambda_T N_T(r,t) \ ,$$

Here N_{Li}, N_T, N_{He} are atomic concentrations of ^{6}Li, ^{3}T and ^{3}He in Be matrix, σ_r^n is the microscopic cross section of nuclide *n* for reaction *r*, φ(r,E,t) is a neutron flux density, and λ is a decay constant for ^{3}T.

During reactor operation ^{3}He is burning out according to the reaction (8). The tritium concentration $N_T >> N_{Li} >> N_{He}$ and N_T increases linearly with the energy produced [MW.days]. After reactor shut down, the concentrations of ^{3}He increases due to the decay of ^{3}T, thereby causing losses in the reactivity of the reactor core. The accuracy of the calculated poisoning was verified on 3 measured values of the reactivity loss due to build-up of ^{3}He in shut down (see Table III). The reactivity effect due to the ^{3}He poisoning during reactor operation history is evaluated with MCNP and presented at Fig. 15.

Table III. Comparison between measured and calculated by MCNP–4C reactivity losses $\Delta\rho(^3\text{He})$ due to neutron absorption by He-3 during 3 different shut down periods.

Time of reactivity measurement in shut down	MCNP–4C [$/day]	EXPERIMENT [$/day]	MCNP/EXP
T1=30/09/1999	–(0.0143±0.0008)	–(0.0160±0.0010)	0.89
T2=15/05/2001	–(0.0242±0.0015)	–(0.0280±0.0019)	0.86
T3=09/10/2002	–(0.0308±0.0020)	–(0.0380±0.0023)	0.81

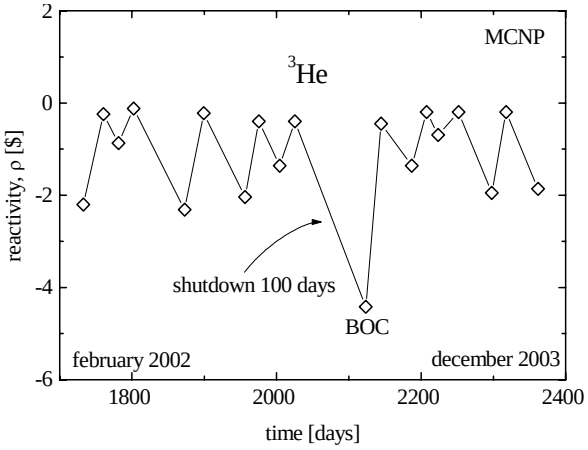


Fig. 15. Reactivity effect due to accumulation ^{3}He in shutdowns.

Sensitivity of net reactivity to burn up

The evolution of the net reactivity during operation, influenced by the different reactivity-contributing components is demonstrated at Fig. 16. In practice, before start of an operating cycle we evaluate only the net isotopic fuel depletion and the burning of ^{3}He during operation. The detailed analysis for the contribution from the different fission products for each cycle is difficult, because of the used Monte Carlo technique.

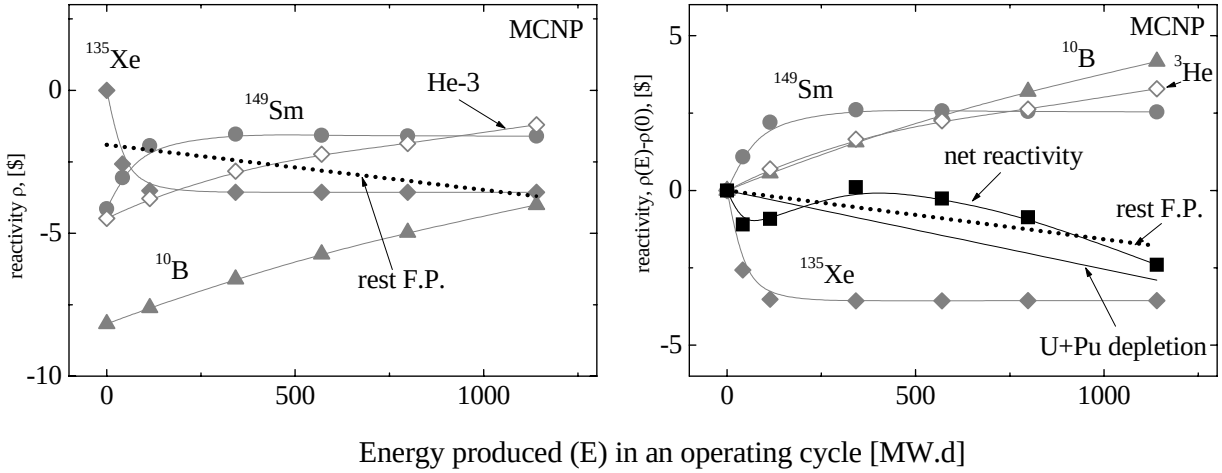


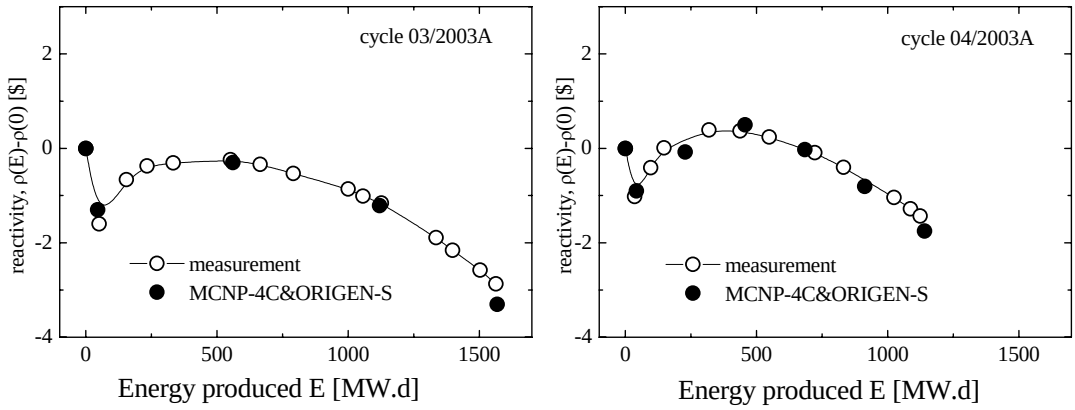
Fig. 16. Evolution of reactivity-contributing components in BR2 versus energy produced in typical operating cycle for the core loading at Fig. 3.

IV.B. VALIDATION ON REACTIVITY MEASUREMENTS

The validation of the model started with comparison on reactivity measurements of a set reactor core loads at BOC of various BR2 operating cycles. The results of the reactivity calculations at BOC for 11 operation cycles of BR2 having different reactor core load are summarized in Table IV. The measured heights of the Control Rods in the critical reactor are given in column 2 of Table IV. The deviations between calculated with MCNP-4C reactivity and measurement are summarized in column 4. After that, the validation of the model was extended to comparison between calculated and measured curves of the reactivity evolution during the operating cycle. Examples of such comparisons are shown at Fig. 17. A difference of about 0.5 $ between calculation and measurement is demonstrated at Fig. 17d. The reason for such deviation was an attempt to simplify the calculations, replacing the 3D space dependent isotopic fuel depletion in part of the fuel elements with uniform distribution. After that the calculations were repeated with the detailed 3D isotopic fuel depletion profile in all fuel elements and the curve was corrected (Fig. 17e). The curves of the reactivity evolution during the operating cycle are evaluated for the critical height of the CR at BOC. The critical heights of the CR during the operating cycle are estimated from the curve of the differential CR reactivity (see Fig. 9), which is calculated before start of an operating cycle. Using the estimated critical positions, the calculations are repeated for the CR motion during the cycle, adjusting their positions at each time step for k_{eff}=1.0.

Table IV. Comparison of reactivity calculations using MCNP-4C with the measurement for BOC of different BR2 operation cycles. The critical height of the Control Rods bank is signed as Sh.

Operation cycle of BR2	Sh [mm] measured (Experiment)	Calculated k_{eff} for measured Sh (MCNP-4C)	ρ(MCNP)−ρ(measured) [$]
05/1999A.7	411.5 ±1.0	0.9994 ± 0.0001	–(0.083±0.014)
01/2002A.2	546.0 ±1.0	1.0017 ± 0.0004	+(0.236±0.055)
02/2002A.3	459.0 ±1.0	1.0010 ± 0.0004	+(0.139±0.055)
03/2002A.4	454.0 ±1.0	0.9990 ± 0.0002	–(0.139±0.028)
04/2002A.7	591.0 ±1.0	1.0007 ± 0.0002	+(0.097±0.028)
05/2002A.2	520.0 ±1.0	0.9981 ± 0.0002	–(0.264±0.028)
01/2003A.4	633.0 ±1.0	0.9995 ± 0.0002	–(0.070±0.028)
02/2003A.6	443.0 ±1.0	1.0000 ± 0.0002	(0.000±0.028)
03/2003A.5	462.0 ±1.0	0.9998 ± 0.0002	–(0.028±0.028)
04/2003A.3	529.0 ±1.0	1.0011 ± 0.0003	+(0.153±0.042)
05/2003A.1	510.0 ±1.0	0.9993 ± 0.0002	–(0.097±0.028)



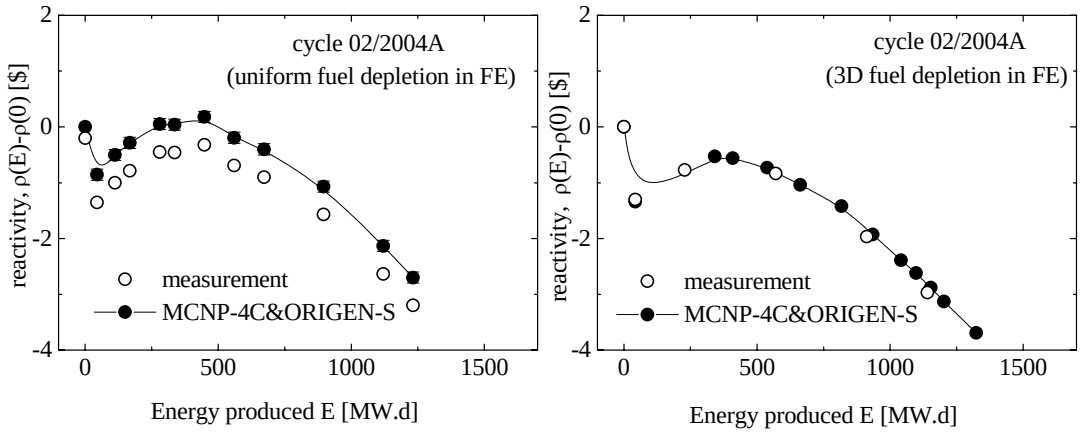


Fig. 17. (a,b,c) Comparison between measured and calculated with MCNP&ORIGEN-S curves of net reactivity evolution during BR2 operating cycles, using 3D isotopic fuel depletion profile in FE. The last two curves demonstrate the effect of 3D modeling of fuel depletion in FE: (d) is calculated with uniform fuel depletion in half of the fuel elements; (e) is calculated with 3D fuel depletion profile in all fuel elements.

IV.C. ACCURACY OF CRITICALITY CALCULATIONS

The computational time for a Monte Carlo simulation by MCNP-4C for the described model, containing total more than 6000 spatial cells is 10000 histories per minute on work station PC PENTIUM-4/2GHz/LINUX Red Hat 7.2. A number of N=2.000.000 (T=3÷4 hours) is sufficient for assessment of k_{eff} with a standard deviation ±0.0005.

Cross sections data

Our purpose in this paper is not to investigate the impact of various cross sections libraries on the value of the multiplication coefficient k_{eff}. One set continuous-energy cross sections data for the selected nuclides was used in all MCNP calculations, presented in this paper. Most of used data are those from ENDF60, based on ENDF/B-VI file, the absent nuclides are taken from ENDF/B-V [19]. The libraries for decay data, used with ORIGEN-S, are based on ENDF/B-VI file and the fission-product yields for all fissionable nuclides are based on ENDF/B-V.

To account for the unresolved resonance range cross sections data for the following nuclides were taken from the library URES, which is distributed with the MCNP package: ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{149}Sm, ^{150}Sm, ^{152}Sm, ^{153}Eu, ^{154}Eu. However, due to the highly enriched ^{235}U fuel content (90%), the perturbation in reactivity values due to the unresolved resonance range is less than –0.10 $. To allow for the Doppler broadening of the cross sections during operating cycle, cross sections data, evaluated at 400 K for ^{235}U, ^{238}U and ^{239}Pu from file ENDF62MT [19] were used in calculations of the hot core, which temperature is about 150 °C.

3D modeling of fuel depletion in the whole core

The main impact on the criticality calculations of BR2 is due to the modeling of the fuel depletion in FE and modeling of the poisoning of the beryllium reflector. The detailed axial and radial 3D fuel depletion modeling in each fuel element is important because of the high peaking factor in BR2, K_V~1.5. The effect of the detailed 3D space distribution of the fuel depletion in the core was analyzed on comparison with uniform distribution in each fuel element. The typical reactivity effect due to the 3D fuel depletion profile is –1.0 $ in comparison with average fuel depletion in the fuel elements in the case of no loaded fuel element in the central channel H1/C. For a reactor core with loaded FE in H1/C the effect is –1.4 $.

Perturbation effect of CR position on 3D power peaking factors

The perturbation effect of the control rods position on the axial symmetry of the 3D power peaking factors was investigated. A demonstration of this effect on the axial distribution of the power peaking factors in the outer fuel plate of a fuel element is presented at Fig. 18 (such curves were calculated for the remaining 5 fuel plates, too).

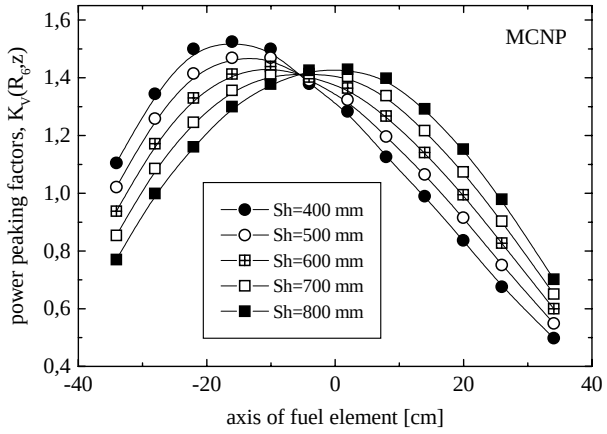


Fig. 18. Perturbation effect of control rods position (Sh) on the axial symmetry of power peaking factors in the outer (6^{th}) fuel plate of a fuel element, located in one of the channels A in the central crown of the BR2 reactor core.

With the motion of the control rods, the maximum of the curves is shifting from Z=–10 cm for Sh=400 mm to Z=0 cm for Sh=800 mm and the curves become more symmetrical. However, the analysis has shown that the perturbation in the reactivity values due to the CR motion for a fuel element, located in one of the channels A or B in the central crown, is very small: Δρ=0.0021$ per ΔSh=100 mm (or $\Delta k_{eff}/k_{eff}$ =0.0015% per 100 mm movement of CR) per 1 FE. The typical travel of the control rods inside the cycle varies between 400 and 700 mm, depending on the fuel loadings. This means that if we choose one average position Sh=550 mm, for which we evaluate K_V, the maximum uncertainty in the reactivity value is: $\Delta k_{eff}/k_{eff}$ =0.00225% per 1 fuel element and for 12 elements in the central crown $\Delta k_{eff}/k_{eff}$ =0.027%, which is within the statistical errors of the Monte Carlo calculations. This allowed us to simplify significantly the calculations of the 3D power peaking factors. Practically, the 3D power peaking factors K_V are calculated only for 1÷2 positions of CR and therefore the number of the evaluated K_V–distributions is reduced to 1÷2 during the

cycle. K_V – distributions, calculated for one position of the CR, are used for evaluation of 3D power distribution in several depletion steps, which are simply predicted using the average power of the fuel element, evaluated with MCNP and formula (1). A special treatment is made only for a fuel element, located in the central channel H1/C, for which $\Delta\rho$=0.1 \$ per ΔSh=100 mm (or $\Delta k_{eff}/k_{eff}$ =0.072% per 100 mm CR movement), for which we calculate K_V for three positions Sh of the CR during the cycle.

Poisoning of Be-matrix

The present beryllium matrix has replaced the previous, poisoned one, at 1997. From that time more than 45 cycles were operated at BR2, each with different core load and different duration of shutdown, forming very complicated time and 3D space dependent distribution of the poisons in the whole reactor core. Our efforts in reducing of the correlated uncertainties were directed into following the full irradiation history from 1997 and calculation of the 3D space dependent reaction rates on beryllium almost for all 45 core loadings. This significantly (by 15%) improved our predictions in the poisoning by helium-3, but still a work which has to be done, is the accurate evaluation of the contribution into the poisoning from the photo neutrons from (γ,n)-reactions on beryllium.

V. SUMMARY

The validation of the MCNP-4C&ORIGEN-S model for the criticality calculations of the 3D full heterogeneous core during operation of the BR2 reactor is discussed. The specific HEU reactor core configuration in a form of a twisted hyperboloid bundle and the strong heterogeneousity of the fuel elements make any others than Monte Carlo method non-applicable for reactivity calculations of such core. The 3D Monte Carlo code MCNP is used for the precise modeling of the heterogeneous core and for evaluating the 3D power peaking factors. The fuel element is modeled from 144 spatial cells (6 fuel plates×12 axial zones in each plate×72 azimuth sectors in the hot plane of the outer plate) with varied fuel depletion. The isotopic depletion chains are calculated outside Monte Carlo analysis with 1D code ORIGEN-S, which allows to safe large computational time. The 3D isotopic fuel depletion at each depletion step is evaluated, using the 3D power distribution in a previous step in combination with a database with isotopic depleted fuel compositions, prepared earlier with ORIGEN-S.

The accuracy of the reactivity predictions is affected by the following factors:

(1) The effect of 3D isotopic fuel depletion distribution on the reactivity values is $\Delta k_{eff}/k_{eff} \sim$ (–0.7%÷–1.0%) in comparison with uniform modeling of the fuel depletion in the fuel elements.

(2) The perturbation effect of the control rods position on the axial symmetry of the 3D power peaking factors K_V was investigated. The analysis has shown that the uncertainty in the reactivity values due to the CR motion is within the statistical errors: $\Delta k_{eff}/k_{eff} \leq 0.03\%$, which allows to simplify the calculations and to evaluate 3D K_V–distributions only for 2÷3 positions of the control rods during the cycle.

(3) The isotopic fuel depletion is evaluated versus burn up, assuming continuous irradiation in the reactor, which is reasonable taking into account the high ^{235}U content in the fuel (90%) and the linear dependence of the atomic concentrations of ^{235}U and the fission products versus fuel burn up. A special treatment is made for ^{149}Sm and ^{135}Xe, which are

depleted, allowing for the shutdown between operating cycles and the further use of the burnt fuel in the reactor. However, a detailed analysis of the effect of the core reload on the variation of the remaining isotopic atomic fuel concentrations must be performed in the future.

(4) The main uncertainty, $\Delta k_{eff}/k_{eff} \sim 0.40\%$ is associated with the modeling of the ^{3}He poisoning of the beryllium reflector after long shutdown ($T_{shut} \geq 40$ days).

The model is tested on reactivity measurements of more than 15 operating cycles at BOC and on reactivity evolution curves during operation. Our computational predictions are within $\Delta k_{eff}/k_{eff} \sim(0.40\pm0.05)\%$ at various restarts and $\sim(0.80\pm0.05)\%$ after long shutdown.

NOMENCLATURE

B.P. – burnable poison
F.P. – fission product
DB – database with 90 depleted isotopic fuel compositions for each fuel type
FE – fuel element
HEU – highly enriched uranium
k_{eff} – effective multiplication factor
K_V – power peaking factor (r.u.)
Sh – position of shim rod (mm)
Greek
β_{eff} –fraction of delayed neutrons
β^5 [X%] – isotopic fuel depleted composition, corresponding to X% ^{235}U depletion and containing 80 depleted nuclides, including actinides, fission product inventory and light elements
$\Delta k_{eff}/k_{eff}$ – reactivity (%)
ρ – reactivity (\$); $\Delta\rho$ – reactivity change (\$)

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