The Monte Carlo Method: Versatility Unbounded In A Dynamic Computing World
Chattanooga, Tennessee, April 17–21, 2005, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2005)

OVERVIEW OF THE MONTE CARLO CALCULATIONS OF THE BELGIAN MTR BR2

Vadim Kuzminov, Silva Kalcheva and Edgar Koonen
SCK•CEN, Belgian Nuclear Research Centre
Boeretang 200, B-2400 MOL-Belgium

Corresponding Address
vadim.kuzminov@sckcen.be; silva.kalcheva@sckcen.be; edgar.koonen@sckcen.be

ABSTRACT

This paper gives an overview of the various applications of the Monte Carlo method for the reactor calculations of the Belgian Research Reactor BR2. The R&D related to the various irradiation programs require a precise description of the environment of the experimental facilities. The MCNP-4C code was used for development of the full-scale 3D-heterogeneous model of BR2. Since 2001 the model is systematically used for evaluation of the irradiation conditions of the major experimental programs and covers various areas of the nuclear and non-nuclear reactor applications. The model is validated on the multiple comparisons of the neutron fluxes with the measured foil reaction rates; on benchmark calculations of the thermal power in MTR advanced fuel plates, MOX rods and in comparison with thermal balance/γ-spectroscopy methods; on the reactivity measurements.

Key Words: MCNP, Research Reactor, reactor applications

1 INTRODUCTION

BR2 is one of the major high flux MTR type reactors in the world, operated by SCK-CEN at the Mol site in Belgium. The reactor has a wide and multipurpose irradiation program, related to different aspects of national and international R&D, concerning the safety of nuclear reactors, plant lifetime evaluations, the use of MOX fuel, medical and industrial applications.

The irradiation programs require a very precise description of the environment of the experimental facilities. This has lead to the necessity of the use of the Monte Carlo method as the most sophisticated tool for evaluation and prediction of the irradiation conditions for ongoing and future experiments. A detailed full-scale 3-Dimensional heterogeneous geometry model of the whole reactor core, which describes the real hyperbolic form of the reactor core, has been developed using MCNP-4C.

Since 2001 the model is systematically used for evaluation of the irradiation conditions of the major experimental programs and covers various areas of the reactor applications. The model is validated on benchmarks and experiments for testing of advanced fuel plates, for irradiation of structural materials from LWR and candidate structural materials for the ITER fusion reactor and ADS. An assessment of the accuracy of the developed calculation model is done by analysis of the calculated and measured reaction rates in activation foils.

An extended model of BR2 including detailed axial and radial fuel burn-up distributions in the core has been developed using MCNP-4C&ORIGEN-S codes and validated on the reactivity measurements of more than twenty BR2 operation cycles. The accurate calculations of the axial and radial distributions of the poisoning of the beryllium matrix by ^{3}He, ^{6}Li and ^{3}T were verified on the measured reactivity loss and used to predict the reactivity behavior during the future ten or more years.

2 DESCRIPTION OF BR2 REACTOR

BR2 is a heterogeneous high flux, tank-in-pool type reactor, cooled and moderated by light water. The HEU (93% ^{235}U) reactor core is positioned in a beryllium matrix − a big number of prismatic hexagonal prisms which are skew and form a twisted hyperbolic bundle around the central 200 mm channel H1 containing beryllium plugs. The reactor can be operated at the power level of 50÷100 MW, currently about 120 full power days per year. The maximum admissible heat flux in the fuel plates of the standard BR2 fuel elements is 470 W/cm2. The maximum neutron flux is 1.2×10^{15} cm^{-2}s^{-1} (E_n<0.5 eV) and 8.4×10^{14} cm^{-2}s^{-1} (E_n>1.0 MeV).

The reactor BR2 has more than 100 irradiation positions, including a large number of 84 mm test holes/channels in the Be-matrix – traps for fast neutron fluxes in the axis of fuel elements, 4 peripheral 200 mm channels for large irradiation devices and a 200 mm central trap for high thermal neutron flux. A dedicated CALLISTO in-pile loop (CApabiLity for Light water Irradiation in Steady state and Transient Operation) provides a representative environment for the irradiation of LWR fuel, studies of the structural performances of materials for LWR irradiated to high neutron fluence, for fusion reactor. The loop comprises three in-pile sections (IPS); each IPS can receive a square basket containing nine 9.5mm outer diameter fuel rods, each 1m in length.

3 CALCULATION CODES

MCNP [1-2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The unique ability of the Monte-Carlo method to treat arbitrary irregular geometries, supplemented with the high flexibility of utilization, makes MCNP particularly useful for reactor calculations of the BR2 core. MCNP uses continuous-energy nuclear and atomic data libraries, based on ENDF/B data evaluation file. For processing of dpa cross sections data and neutron cross sections at various temperatures the nuclear data processing system NJOY [3] is used.

Fuel depletion, actinide transmutation and fission product buildup and decay are calculated using ORIGEN-S, the 1-D depletion module of the SCALE system [4]. The actinide libraries of ORIGEN-S include depletion chains for 100 fuel, trans-plutonium and decay daughter nuclides. The fission product library of ORIGEN-S contains depletion chains for about 800 nuclides.

Another code used for complete burn-up calculations in a one single run is MCB – Monte-Carlo Continuous Energy Burn-up Code [5], which integrates MCNP-4C with a novel Transmutation Trajectory Analysis code.

4 NEUTRONIC MODELLING OF BR2

4.1 MCNP whole core 3-D heterogeneous model [6-7]

The full-scale 3-D heterogeneous geometry model was developed using the Monte-Carlo code MCNP-4C and presented at Fig. 1. The model includes a sophisticated description of the actual twisted hyperboloid reactor core, formed from skew beryllium prisms with individual orientation of the loaded fuel elements, control rods and engineering devices inside the test holes.

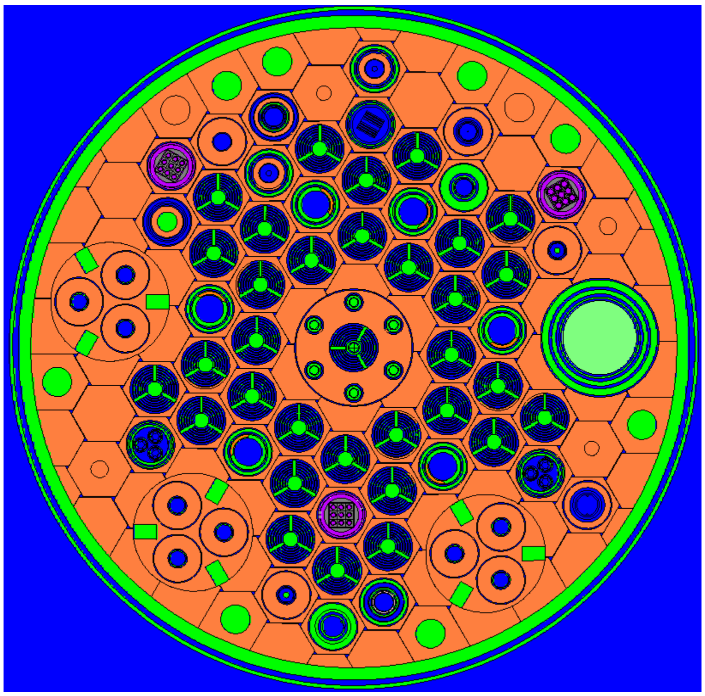


Figure 1. MCNP model of BR2 core in the middle plane.

4.2 Simulation of the isotopic fuel depletion using MCNP and ORIGEN-S

A combination of the 3-D Monte Carlo code MCNP with 1-D depletion code ORIGEN-S was used for the modeling of the 3-D space dependent isotopic fuel depletion in the core [7]. The depletion chains are calculated outside the Monte-Carlo analysis, allowing to save significant computational time. MCNP is used for evaluation of the 3-D space dependent specific power distribution in the fuel elements. For this purpose, each one of the 6 fuel plates of all of 32 fuel elements is divided into axial zones over the fuel height by each 6 cm. The fuel zone in the hot plane of the outer fuel plate of each element is divided in the azimuth direction by sectors of 5°. ORIGEN-S is used for evaluation of the isotopic composition of the depleted fuel versus the fuel burn-up and for preparation of a fuel composition database (DB), which is further used in the MCNP model. Schematically this approach is shown at Fig. 2. The total number of the spatial cells with varied fuel depletion in the model is 4600.

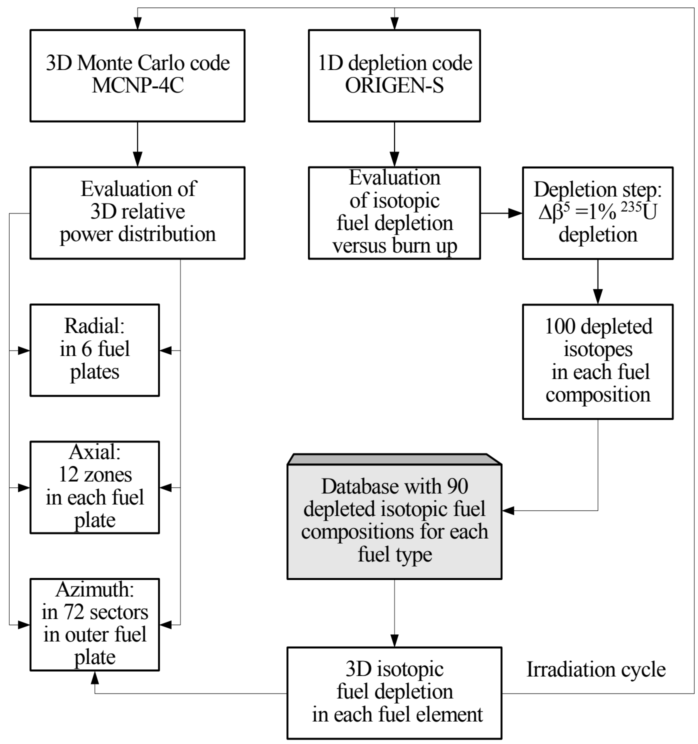


Figure 2. A chart of the modelling of 3-D distribution of the fuel depletion in the BR2 fuel elements using MCNP & ORIGEN-S combined simulation method.

4.3 Delayed gamma source modelling using MCNP & ORIGEN-S codes

MCNP-4C code can only solve neutron-induced photon transport, the contribution from the delayed photons from the fission products can not be computed directly by the code. A separate geometry model of BR2 has been developed for evaluation of the heating from the delayed photons. For this purpose the codes ORIGEN-S or SASH2 [8] of the SCALE system are used for evaluations of the photon spectra and the photon intensity from fission products accumulated in the fuel elements during irradiation in BR2 reactor. Using the calculated with MCNP power peaking factors in each fuel element, the axial and radial distributions of the intensity of the delayed photon sources in the core are prepared and used as an external source in the MCNP model of BR2 for the supplementary photon transport calculation.

5 MCNP CALCULATIONS OF FUNCTIONALS

5.1 Neutron fluxes and reaction rates

The main part of Monte-Carlo reactor calculations of BR2 is related to evaluations of neutron fluxes and comparison with the experimental results obtained in dosimetry measurements of different activation foils.

5.1.1 Conventional thermal flux

The conventional thermal flux is determined in the experiment using the measured activity of ^{59}Co in AlCo foil dosimeter. The activity of the thin ^{59}Co foil can be expressed through the rate of neutron capture reaction on ^{59}Co as:

$$R_i = \int_0^{0.5eV} \sigma_{n,\gamma}^{Co59}(E)\varphi(E,r_i)dE, \tag{1}$$

where E_{th}~ 0.5 eV is the upper boundary of energy for thermal neutrons, $\sigma_{n,\gamma}^{Co59}(E)$ is the activation cross-section for ^{59}Co in (n,γ) reaction; r_i is the co-ordinate of the dosimeter position; $\varphi(E,r_i)$ is the neutron flux density in the foil normalized to the nominal reactor power. Using an assumption of 1/υ behavior of the cross section $\sigma_{n,\gamma}^{Co59}(E)$ for thermal neutrons, Eq.(1) can be rewritten:

$$R_i = \int_0^{0.5eV} \sigma_{n,\gamma}^{Co59}(E_0)\frac{\upsilon_0}{\upsilon} n(E,r_i)\ \upsilon\ dE = \sigma_{n,\gamma}^{Co59}(E_0)\upsilon_0 \int_0^{0.5eV} n(E,r_i)dE, \tag{2}$$

where υ_0=2200 m/s and the cross-section $\sigma_{n,\gamma}^{Co59}(E_0)$= 37.18 barns are defined at the energy E_0=0.0253 eV, $n(r_i,E)$ is the neutron density. The conventional thermal flux, φ_0, is determined as a ratio of the reaction rate R_i (2) to the cross-section $\sigma_{n,\gamma}^{Co59}(E_0)$:

$$\varphi_0(r_i) = \frac{R_i}{\sigma_{n,\gamma}^{Co59}(E_0)} = \upsilon_0 \int_0^{0.5eV} n(E,r_i)dE \tag{3}$$

where the integral defines the total number of neutrons in the thermal energy range. Re−writing the expression (3), the conventional thermal flux φ_0 can be calculated as:

$$\varphi_0(r_i) = \upsilon_0 \int_0^{0.5eV} \frac{\varphi(E,r_i)}{\upsilon} dE, \quad \upsilon = \sqrt{2E/m} \tag{4}$$

where υ is the neutron velocity; E is the neutron energy; m is the mass of neutron.

Two methods are used to calculate the conventional thermal flux: using the density of thermal neutrons Eq.(4), and the reaction rate in ^{59}Co dosimeter Eq.(3). Generally, the conventional flux is lower than the real thermal neutron flux density by the factor of difference between the mean velocity of neutron spectrum and the most probable velocity in neutron spectrum. The temperature dependence of the thermal neutrons spectrum may influence the conventional flux.

5.1.2 Equivalent fission neutron flux

In the experiment the equivalent fission flux is determined from the measured activity of various foil dosimeters. The reaction rates

$$R_i = \int_{E > E_{S,n\to x}^{^kY}} \sigma_{n,x}^{^kY}(E)\varphi(E,r_i)dE \tag{5}$$

are calculated from the activity of the nuclide ^kY, induced by fast neutrons after the threshold reactions ${}^kY \xrightarrow{(n1,x)} {}^kZ$, e.g. such as

$$^{54}Fe \xrightarrow{(n,p)} {}^{54}Mn\,;\quad {}^{58}Ni \xrightarrow{(n,p)} {}^{58}Co\,;\, {}^{46}Ti \xrightarrow{(n,p)} {}^{46}Sc\,;\, {}^{93}Nb \xrightarrow{(n,n')} {}^{93}Nb \qquad (6)$$

where $E^{^kY}_{S,n\to x}$ are the threshold energies and $\sigma^{^kY}_{n,x}$ are the corresponding microscopic neutron cross section of the reactions (6) for the nuclide ^kY; $\varphi(E,r)$ is the neutron flux density in the foil.

The equivalent fission neutron flux is defined as:

$$\phi(r_i) = \frac{\int_{E^{^kY}_{S,n\to x}}^{20} \sigma^{^kY}_{n,x}(E)\varphi(E,r_i)dE}{\left\langle \sigma^{Y}_{n,x} \right\rangle}, \qquad \left\langle \sigma^{^kY}_{n,x} \right\rangle = \int_{E^{^kY}_{S,n\to x}}^{20} \chi(E)\sigma^{^kY}_{n,x}(E)dE \qquad (7)$$

where $\chi(E) = C\exp(-E/a)sh(bE)^{1/2}$ is the Watt fission energy spectrum with mean energy <E>=2.03 MeV.

The reaction rates (5) and the effective cross-section $\left\langle \sigma^{^kY}_{n,x} \right\rangle$ in the denominator of Eq.(7) are calculated using MCNP and cross sections data libraries for the nuclide ^kY from ENDF/B-V,VI and from the dosimetry files. The calculated with MCNP effective cross section of the reaction ^{54}Fe(n,p)^{54}Mn with threshold energy $E^{^{54}Fe}_{S,n\to p}$=1MeV was equal to $\langle \sigma^{^{54}Fe}_{n\to p} \rangle$=81.1mb. The calculated effective cross sections for the reactions $^{58}Ni \xrightarrow{n,p} {}^{58}Co$ were equal to 106.6 mb, for the reaction $^{46}Ti \xrightarrow{n,p} {}^{46}Sc$ - 11.2 mb, for $^{93}Nb \xrightarrow{n,n'} {}^{93}Nb$ - 12.6 mb. Usually the equivalent fission neutron flux, defined using Eq. (7) is less than the fast neutron flux density for E>1MeV.

5.1.3 Effective DPA cross section

Effective DPA cross-section for cutting energy E_c=1 MeV is used to determine the dpa in the irradiated materials using the measured fast fluxes. Calculations are performed using MCNP and continuous energy dpa cross-sections for the irradiated materials from IRDF-90 file [9] and NJOY [3] code:

$$\sigma_{DPA} = \frac{\int_0^{20MeV} \sigma_{DPA}(E)\varphi(E)dE}{\int_{1MeV}^{20MeV} \varphi(E)dE} \qquad (8)$$

5.2 Fission rate and thermal power characteristics

The number of fission reactions $R_{fis,i}$ which are generated by the fissionable isotope i in a volume V of the fuel meat:

$$R_{fis,i} = \rho_i \int_E dE \int_V dr\, \sigma_{f,i}(E)\varphi_n(E,r) \qquad [fiss/n] \qquad (9)$$

where ρ_i and $\sigma_{f,i}$ are atomic density and microscopic fission cross section of the fissionable isotope i; φ_n is the neutron flux [cm^{-2}] per source neutron.

The intensity of neutrons generated per second in the whole BR2 at the total reactor power P_{BR2} is

$$N_n = \frac{P_{BR2} * \nu_f}{E_{fis,BR2}} \qquad [n/s] \tag{10}$$

where ν_f =2.43 is the average number of fast neutrons emitted per one fission event, $E_{fis,BR2} \approx$(193±2)MeV is the effective energy released per fission reaction of ^{235}U nucleus in BR2 (the standard fuel used in BR2 contains 93% ^{235}U).

The power deposited in a volume V of the fuel meat of a fuel plate/rod is

$$P_{fis}^{dep} = N_n E_{fis}^{dep} \sum_i R_{fis,i} \qquad [W] \tag{11}$$

where $E_{fis,i}^{dep}$ is the effective energy deposited in the fuel meat and normalized per fission event in this fuel. This value is calculated using MCNP-4C and ORIGEN-S taking into account the absorption of prompt, delayed and captured gamma rays in the fuel meat of a fuel plate.

The specific power (heat flux) on the cooling surface S [cm^2] of the fuel plate and the linear power density of a fuel rod with a length L [cm] are defined as:

$$Q_S = \frac{P_{fis}^{dep}}{2S} \quad [W/cm^2] \text{ and } Q_L = \frac{P_{fis}^{dep}}{L} \quad [W/cm] \tag{12}$$

5.3 Neutron/gamma heating

The calculations of the neutron/gamma heating take into account the contribution from prompt and delayed neutrons, prompt and delayed γ-rays, as well γ-rays released in neutron capture reactions (n,γ) on various materials in the reactor BR2. The heating in a dedicated volume V of a fuel plate/rod includes contributions from all parts of the reactor core: from itself, from all channels in reactor core, beryllium reflector, experimental devices, cooling water etc.

The heating caused by neutrons or by γ-rays is calculated in MCNP using the formula [1]

$$q_i = \left(\rho_a \big/ \rho_g \right) \iint_{VE} H_i(E) \Phi(r,E) dE \frac{dV}{V} \tag{13}$$

where (i=n, $\gamma-$ rays) is a type of the particle (neutron or photon), ρ_a is an atomic density (atoms per cm^3), ρ_g is the material density (grams per cm^3), $H_i(E)$ are heating response functions taking into account all processes in the energy deposition of neutrons and photon transport [1].

5.4 Determination of the fission energy deposited into the fuel meat of fuel plates/rods

The determination of the effective heating energy E_{fis}^{dep} deposited into the fuel meat and normalized per fission in this fuel is important for the accurate assessment of the thermal power characteristics of fuel plates/rods, such as the heat flux Q_S and the linear power density Q_L. A detailed discussion of the total effective energy released in the fission reaction for fissionable nuclides of U, Pu is given in the paper of M.James [10]. The recommended values of the useful energy released in fissionable materials include the kinetic energy of fission fragments, E_k^{FP}; total kinetic energies of emitted fission neutrons, E_k^n; energy of photons (prompt-E_γ^p, and delayed-E_γ^d) and beta particles, E_β. According to [10] the energy E_{fis}^{gen} released in the fission of one nucleus ^{235}U by neutrons is equal to 192.9 MeV; for ^{238}U – 193.9 MeV, for ^{239}Pu – 198.5 MeV and for ^{241}Pu – 200.3 MeV.

The track length of fission fragments and β-particles is small and they deposit all their energy locally in the fuel meat of the fuel plate/rod. The fission neutrons have higher free path length and they lose their energy during slowing down in elastic and inelastic collisions outside the fuel meat. The prompt and delayed photons may escape from the fuel plate/rod and interact with the structural elements surrounding it. An estimation of the photon leakage from the fuel meat and calculations of the heating energy were performed using the MCNP and the SCALE codes. Detailed calculations have been performed for the deposition of the gamma heating in different type fuel plates/rods. It was obtained that only 2÷3 MeV from the total gamma energy (total ~15÷16 MeV for the main fissionable nuclides U, Pu released in fission) remains in the fuel meat. The rest of the photons leave the fuel meat and deposit their energy in the surrounding structural materials. Consequently, the effective heating energy, E_{fis}^{dep}, deposited into the fuel meat of fuel plates (rods) is less than the total energy released in all events accompanying the fission reaction by 6÷8%. This was taken into account in the evaluations of the deposited thermal power characteristics of advanced MTR fuel plates and MOX rods, irradiated in BR2 under the various experimental programs.

5.5 Criticality calculations

The capability to calculate k_{eff} is a standard feature [1] of the MCNP code, which is particularly useful for calculation of the neutron multiplication factor in highly irregular geometries, such as the core of BR2.

6 MCNP CALCULATIONS OF BENCHMARKS&EXPERIMENTS

6.1 Nuclear reactor applications

The irradiation programs include R&D, related to the safety of nuclear fuels in normal and accidental conditions, the increase of their burn-up and the use of MOX fuels; study of damaging of structural materials in nuclear installations (fission, fusion), evaluations of plant lifetime and ageing of their components.

6.1.1 Safety evaluations of MOX fuel rods and MTR fuel plates

A benchmark calculation for determination of the linear power density in MOX fuel rods has been performed under the BACCHANAL program [11-12]. It concerns a particular irradiation program where the burn-up of nine pre-irradiated MOX fuel rods, all loaded in one IPS of the CALLISTO loop, had to be increased to predetermined values. An important characteristic for irradiation conditions of fuel elements is the power distribution. In the experiment several methods were used to determine the heating energy: thermal balance method, gross γ-spectrometry of measuring the distribution of fission products along the length of fuel rods, etc.

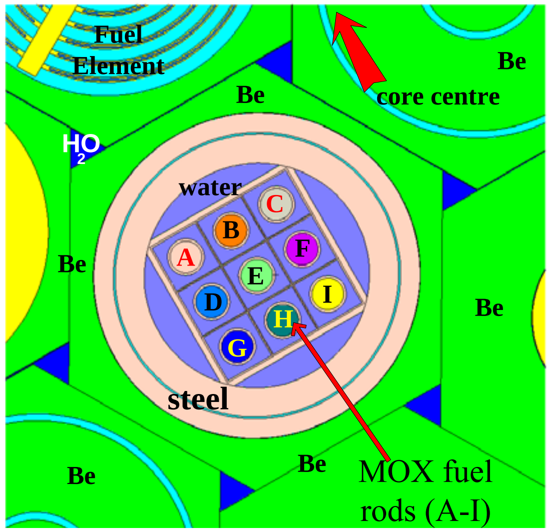
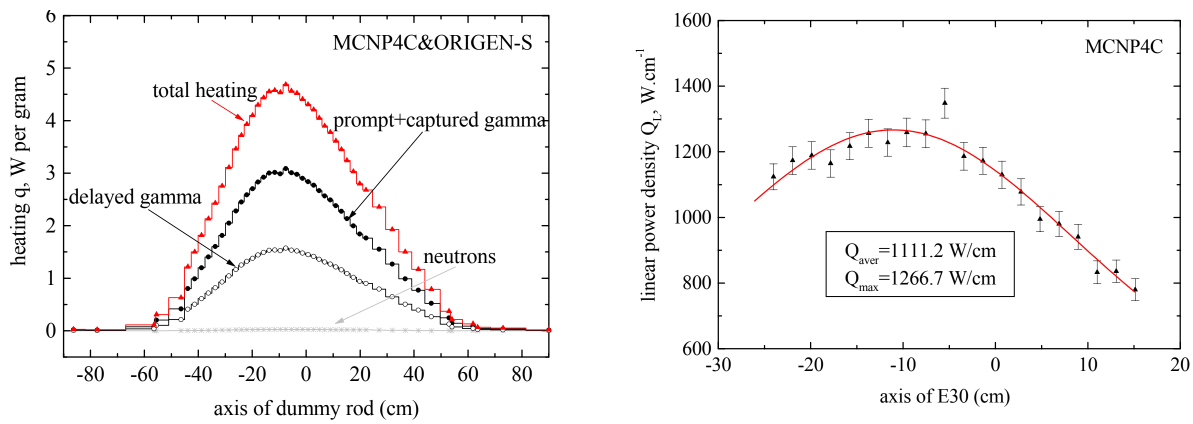


Figure 3. MCNP model of an in-pile section IPS of the CALLISTO loop, loaded with 9 MOX fuel rods in the shroud tube.

The thermal balance method allows for on-line measurement of the total heating energy in an in-pile section of the CALLISTO loop. For determination of the detailed distribution of heating energy in the irradiated fuel rods a preliminary calculation of the power peaking factors is needed. The γ-spectrometric measurement of activity of fission products over the length of fuel rods provides information about the axial peaking power and the fission rate, which can be measured with a high accuracy. The MCNP model of BR2 was used for a simulation of the irradiation of MOX fuel rods (see Fig. 3). Calculations of the effective heating energy per fission in the irradiated MOX fuel rods were performed in order to determine the absolute values of the thermal power in the rods. The gamma-spectrometric measurements of the fission events distribution as well as the results from the thermal balance method were compared with the theoretical calculations. The mean linear power calculated using the MCNP model of BR2 was lower by 2% than the power measured in the thermal balance method. The peak linear power in calculations is higher than expected from the thermal balance method by 4%. The calculated fission rate in MOX rods coincides with the γ-spectrometry results within the error margin ± 5%.

Determination of the linear power in MOX fuel segments, located in a PWC device (Pressurized Water Capsule) during power ramp tests and comparison with the gamma spectroscopy measurements and thermal balance methods were performed under the GERONIMO program [13-14]. The PWC rigs are versatile capsules with pressurized stagnant water in either PWR or BWR conditions. The PWC in-pile section can receive one single fuel rod (fresh or pre-irradiated in a commercial PWR) of about 1 m length, or shorter segments. A CCD (Cycling and Calibration Device) device that incorporates a neutron absorbing screen to perform power cycling or transients surrounds a PWC rig. The PWC and CCD are instrumented with censoring devices to perform precise thermal balance measurements on the fuel rod heating rate. The determination of the power in the fuel elements in such devices requires a model allowing deducing the power deposited in each subassembly.

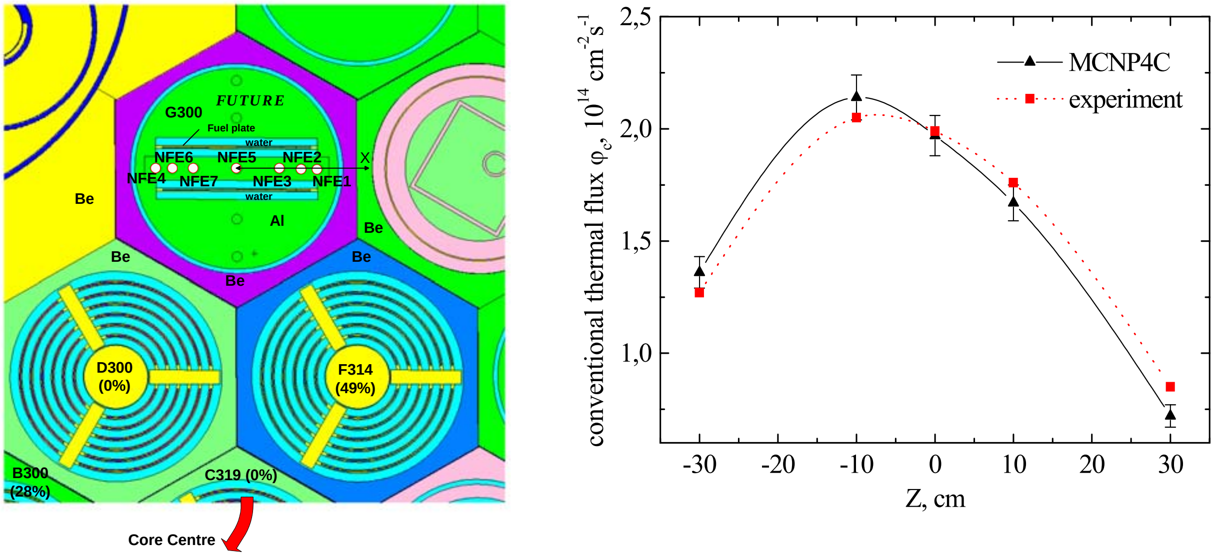


a) b)

Figure 4. a) Total heating in the first stainless steel wall (AISI-321) surrounding the fuel rod in the PWC/CCD device; b) linear power density Q_L, deposited in the fuel rod in GERONIMO experiment. Position of the Control Rods is S_h=500 mm.

The MCNP whole core model was used for evaluation of the total energy deposited by neutrons and photons into the different structure parts of PWC/CCD device and in the fuel rod. The PWC/CCD device was located in channel E30 of BR2, which was surrounded by 3 fuel channels, 2 reflector channels and 1 channel, containing Control Rod. All structure parts and fuel rod of the PWC/CCD device have been modelized in details using the MCNP code. Detailed axial and radial distributions of the neutron, prompt/delayed photon heating in the different structure parts (Fig. 4a) and of the linear power density in the fuel rod (Fig.4b) have been calculated for different positions of the Control Rods. The preliminary MCNP calculations [13] were used together with the thermal balance and γ-spectroscopy methods [14] for the on-line determination of the linear power in the fuel rod during transient ramp tests, which have been performed after cycles 01/2003A and 05/2003A in BR2. The comparison of the predicted by MCNP linear power with the deduced linear power from the measured fission rate by γ-spectroscopy method has shown a difference less than 4%.

The FUTURE (FUel Test Utility for REsearch Reactor) device [6], [15-16] concerns an irradiation program of MTR fuel plates manufactured with UMo (LEU, 8g Utot/cc). The experimental device was instrumented with dosimeters and loaded into the channel G300 of BR2. The general layout of FUTURE fuel plates and of the Al holder is given at Fig. 5a. A benchmark calculation of the conventional thermal neutron fluxes was performed and the results of comparison with the dosimetry results are shown at Fig.5b. The mean statistical deviation of the calculated conventional thermal fluxes from the measured ones for all dosimeters is equal to –2%. Detailed distributions of the heat fluxes over the cooling surface of UMo fuel plates as well as the distribution of the fuel burn-up in the plates were evaluated using the calculated distribution of fission events in the plates.



a) b)

Figure 5: a) Irradiation of 2 UMo fuel plates in channel G300 of BR2: layout of the FUTURE device and NFEi (i=1,..,7) dosimeters; b) Distribution of the conventional thermal flux in dosimeters located in the center of the Al holder.

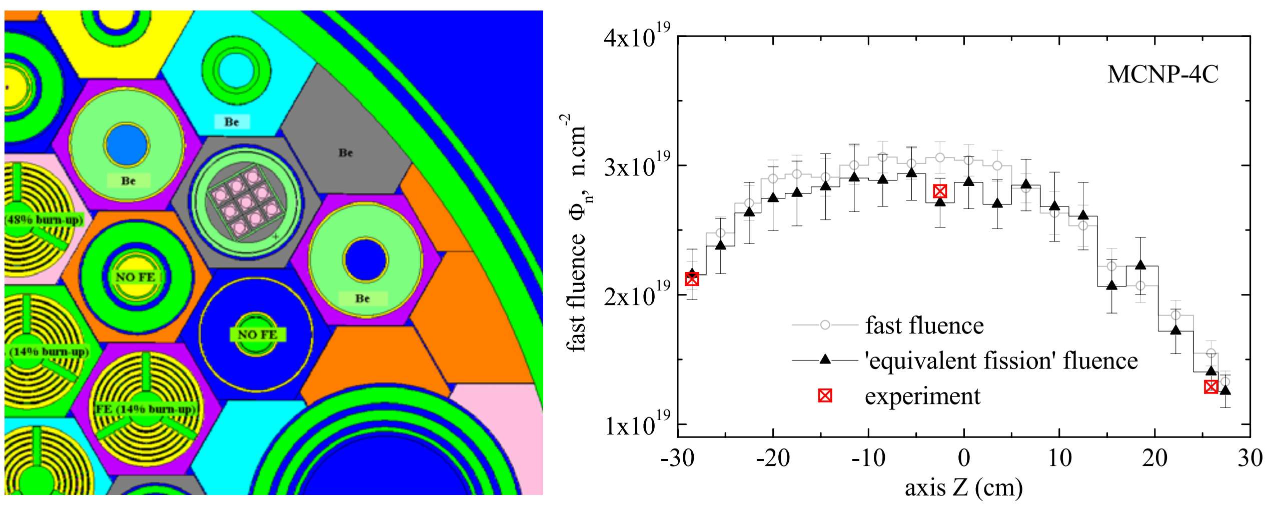
The thermal neutron sensitivities of a set of self-powered neutron detectors (SPND) with Co, Ag and Rh emitters were used for development of a procedure for on-line monitoring of the irradiation conditions in the BR2 reactor. A device with a set of SPNDs is installed in a channel close to the irradiation channel and the SPND data can be continuously recorded to monitor the irradiation process [17]. The absolute thermal neutron flux data from the SPND data can also be used to validate detailed MCNP calculations for the BR2 core, so as to obtain more reliable information on crucial irradiation parameters like fission power. The SPND device can not be introduced in the same channel as fuel MTR test channel, so it was mounted in a neighboring channel containing BR2 driver fuel element. Detailed MCNP neutron calculations have been executed to predict simultaneously the irradiation conditions in the fuel test channel (e.g. for testing of UMo fuel plates in the FUTURE device) and the thermal neutron fluxes at the positions of the SPNDs in the neighboring channel. Data for 6 SPNDs during the BR2 cycle 04/2002A yield an average difference between the experimental and calculated flux data of -7.8 % with a standard deviation of 6.2 %.

6.1.2 Evaluations concerning plant lifetime behavior and aging of structural materials for Fission/Fusion Reactors and ADS

All irradiations performed in the past years dealt with high neutron exposures, e.g. within $(3\div7)\times10^{19}$ n/cm^2 for E > 1 MeV. However, the large irradiation effects are already observed at the fluences below 1×10^{19} n/cm^2, and they tend to saturate at higher fluences. Due to the lower neutron fluences and shorter irradiation time some experiments are very attractive (steel samples from LWR/PWR pressure vessel, other experiments).

Using the MCNP model of BR2, the “equivalent fission flux” and spectrum of fast neutrons were calculated in the RADAMO-10 vessel steel samples, located in rod I of IPS3 (see Fig. 3) for various axial positions of the dosimeters. The comparison between the calculated and measured “equivalent fission flux” has shown a maximum deviation less than 10% [18].

The irradiation of vessel steel charpies from PWR [19], [20] was foreseen to be performed in CALLISTO-loop IPSs during cycle 03/2003A in August, 2003. A series of preliminary optimization calculations for the arrangement of the specimens in IPSs has been performed by MCNP in March 2003. It was shown that the absence of fuel element around IPS (see Fig. 6a) is the needed condition for a maximum ratio: $\Phi_{E<0.4eV}/\Phi_{E>1MeV}$=max and a fast fluence at EOI $\Phi_{E>1MeV} \sim 1.3e10^{19}$ n.cm^{-2}. The irradiation of PWR-vessel steel specimens was performed during cycle 03/2003A at operational power P=56 MW and effective irradiation time $T_{irr}=2.42\times10^{6}$ sec. The arrangement of the specimens was based on the results of preliminary calculations performed using the MCNP code [19]. 31 dosimeters have been located in different axial positions of each of the nine rods in IPS1 and IPS3. The average difference between calculated by MCNP-4C and measured 'equivalent fission' fluxes in 15 dosimeters, located in IPS1 was 2% and 6% in 16 dosimeters, arranged in IPS3 [20]. A detailed axial distribution of the calculated with MCNP fast and fission fluences and comparison with the experimental results is demonstrated at Fig. 6b.



a) b)

Figure 6. a) MCNP model for arrangement of the steel specimens from PWR reactor vessel. The surrounding channels contain no fuel elements in the operating cycle; b) Comparison of calculated fast and "equivalent fission" neutron fluence with the experimental results.

Evaluations of the behavior of structural materials for Fusion and ADS applications, such like various types martenistic steels were carried out under the experimental programs MISTRAL (Multipurpose Irradiation System for Testing of Reactor Alloys), [21] and SPIRE, [22]. The MISTRAL fuel type element contains 5 instead of 6 fuel plates: the inner fuel plate is removed and the free space in the axis is used for irradiation of samples. Calculations of 'equivalent fission' and fast fluxes in the SPIRE steel samples located in the central gap of the MISTRAL fuel elements were performed (see Fig. 7a). The effective DPA cross-sections calculated at various axial positions in two MISTRAL fuel elements (Fig. 7b) are used together with the measured fast fluxes (for E>1MeV) to estimate dpa in the irradiated sample.

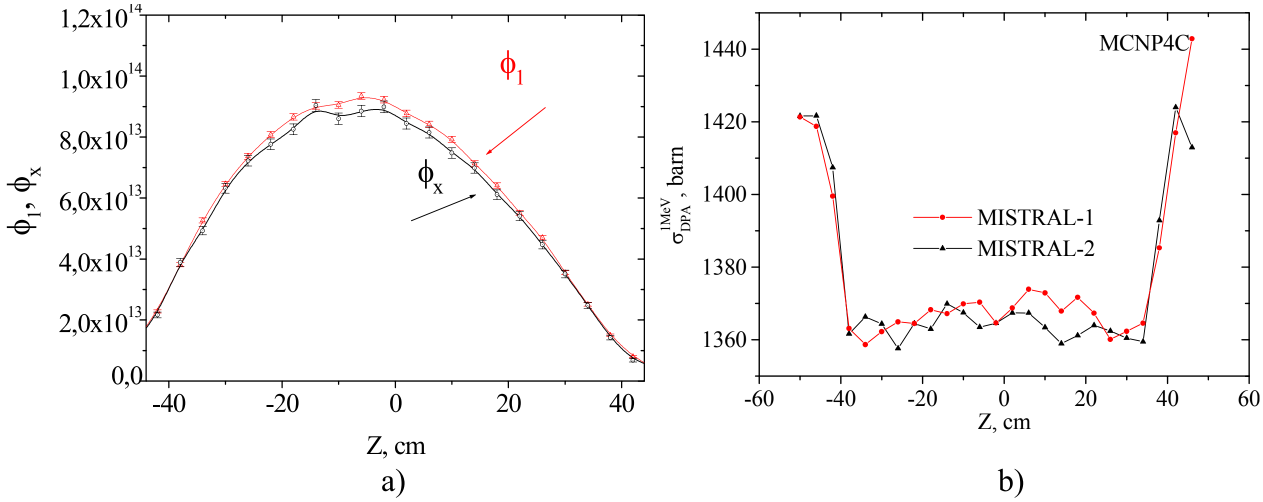
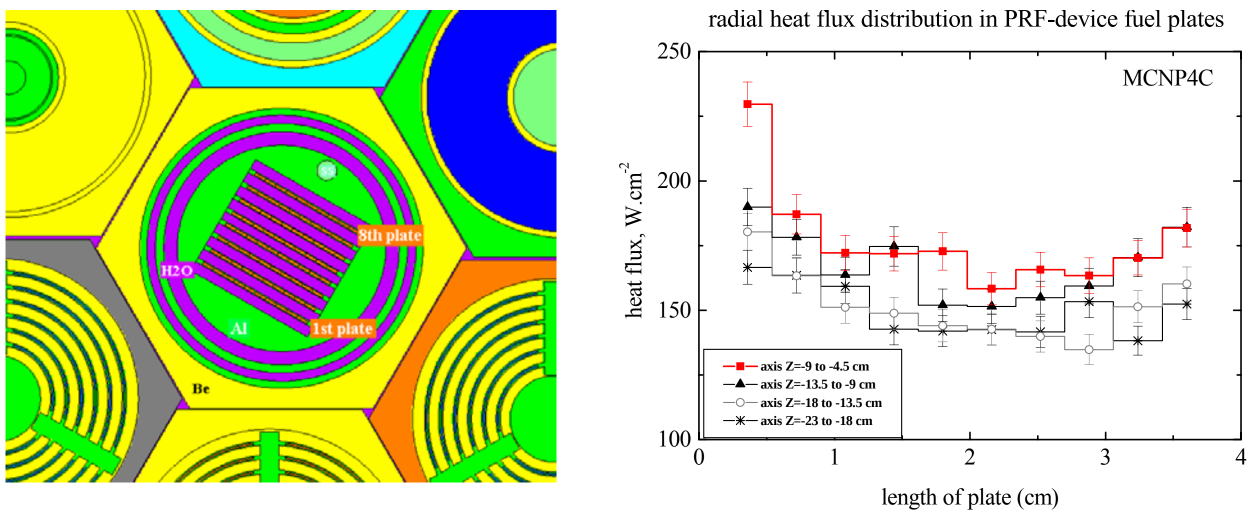


Figure 7. a) Distributions of fast flux ϕ_1 and 'equivalent fission' ϕ_x flux (7), n.cm^{-2}.s^{-1} in stainless steel samples SPIRE, located in the axis of the MISTRAL fuel element; b) Distributions of the effective DPA cross-section versus the axial position in the central gap inside the MISTRAL-1,2 fuel elements.

6.2 Non-nuclear reactor applications

Besides the scientific and technological experiments, the BR2 reactor has always been thoroughly used for the production of radioisotopes for medical and industrial applications. Evaluations of the irradiation conditions for production of isotopes for applications in the nuclear medicine under the irradiation programs PRF were performed using MCNP [23]. The PRF (Primary Reloadable water-cooled devices for Fissile targets) are facilties for production of ^{99}Mo which is separated from the fission products in the irradiated enriched ^{235}U targets. Calculations of the residual activity of Mo-99, the residual activity of the total amount of accumulated fission products and the residual heat in the fuel plates of PRF after EOI were performed with MCNP-4C and SCALE-4.3 (ORIGEN-S). MCNP is used for evaluation of the fission power in the fuel plates of PRF-device, which further is introduced into ORIGEN-S for evaluation of the residual activity. The general layout of the PRF-device, consisted from 2 baskets, each containing 6 or 8 fuel plates, is given at Fig.8a. Detailed radial and axial distributions of the heat flux in the fuel plates of PRF device reactor were performed (Fig. 8b).



a) b)

Figure 8. a) Calculation model by MCNP-4C for PRF device at the plane Z=+10cm; b) Heat flux distributions in the fuel plates of PRF device at various axial levels.

7 MCNP REACTOR CALCULATIONS

7.1 Criticality calculations [7], [24]

The computational time for a Monte Carlo simulation by MCNP-4C for the described model of BR2, containing total ≈ 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 satisfactory assessment of K_{eff} with a standard deviation ±0.0005. The developed with MCNP&ORIGEN-S extended geometry model of BR2 with included detailed axial and radial fuel burn-up distribution in the core was validated on the reactivity measurements of more than twenty BR2 operation cycles [7] and compared with some results obtained by the MCB code [5].

7.2 Azimuth heat flux distributions in the fuel plates and orientation of the fuel element in the core [25-26]

Evaluations of the detailed azimuth heat flux distributions in the "hot" plane of the outer fuel plates of all fuel elements in the BR2 core are performed before each operating cycle with a purpose to predict the maximum value of the heat flux [25]. The importance of an azimuth fuel burn-up modeling in the fuel plates and orientation of the fuel element in the core was discussed in [26]. It was shown that an enhancement up to 20÷25% in the maximum value of the heat flux due to different orientations of the fuel elements in the core is possible.

7.3 Poisoning of Be-matrix [7], [27-28]

The poisoning effect of beryllium 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, ^{3}T and ^{3}He. The

^{6}Li and ^{3}He isotopes have very high absorption cross section of thermal neutrons: $\sigma_a^{^6Li} = 942b$ and $\sigma_a^{^3He} = 5330b$ thereby causing losses in the reactivity of the reactor core. The production of ^{3}T, ^{6}Li and ^{3}He depends on the neutron spectrum in the reactor core. MCNP-4C is used for the calculations of the reaction rates (n,α) on ^{9}Be, (n,t) on ^{6}Li and (n,p) on ^{3}He in all positions of the Be-matrix. The accuracy of the calculated poisoning has been validated on the measurements of the reactivity losses due to the build-up of ^{3}He in the 3rd beryllium matrix of BR2. At the present time evaluations for prediction of various scenarios of the reactivity evolution in the beryllium matrix poisoned up to 2010 and later are conducted using MCNP-4C [28].

8 MCNP OPTIMIZATION CALCULATIONS OF NEW EXPERIMENTAL DEVICES [29]

A model of fast neutron flux booster inside a standard BR2 fuel element has been developed [29]. The purpose was to convert the thermal flux in the water around the axis of the fuel element into a fast flux which can be used for irradiation of samples, located in the axis of the fuel element. Various designs of a fast neutron flux booster have been studied, e.g. an additional uranium fuel ring or basket with uranium fuel pins around the irradiated samples. The considered designs of a fuel element with a booster in the axis allowed enhancement of the fast flux of about 60% compared to a standard fuel element (without a booster).

9 SUMMARY

This paper presents a summary of the most of the published and unpublished results of the Monte Carlo calculations of the BR2 reactor, performed in the period 2001÷2004. The main codes used are MCNP-4C, MCB and the SCALE4.4 system. A short description of the Belgian Research Reactor BR2 and of the main experimental facilities is presented.

The full-scale 3-D heterogeneous model of BR2, describing the real hyperbolical arrangement of the reactor core has been developed with MCNP-4C code in 2001. In 2002, an extended model of BR2, including the detailed 3-D space dependent distribution of the isotopic fuel depletion in the fuel elements, was developed using MCNP-4C&ORIGEN-S.

The utilization of MCNP, MCNP&ORIGEN-S and MCB codes for calculations of the main functionals in the reactor physics like: 'conventional thermal' and 'equivalent fission' neutron fluxes; number of dpa, fission rate and thermal power characteristics (heat flux and linear power density), neutron and photon (prompt and delayed) heating, determination of the fission energy deposited into the fuel meat of fuel plates/rods, neutron multiplication factor, fuel burn up, etc., is discussed.

The MCNP calculations of the reactor BR2 are validated on the benchmarks and experiments related to various irradiation programs, carried out in BR2. A detailed MCNP geometry model of the experimental device and of the environment around it is developed for each reactor irradiation program.

The main conclusions are:

– neutron fluxes are predicted ordinary with accuracy less than ± 10% in comparison with the dosimetry measurements;

– thermal power characteristics (fission rate, heat flux, linear power) in the irradiated advanced MTR fuel plates and spent MOX fuel rods is also predicted within the margin of ± 10% in comparison with the thermal balance and γ - spectroscopic methods.

– predictions of the neutron multiplication factor k_{eff} in the criticality calculations are less than ± 0.5%, compared to the reactivity measurements.

10 REFERENCES

1. MCNP-4C. Monte Carlo N-Particle Transport Code System. Oak Ridge National Laboratory, RSICC Computer Collection, CCC-700/MCNP-4C
2. MCNPDATA. Standard Neutron, Photon, and electron Data Libraries for MCNP-4C. Oak Ridge National Laboratory, RSICC Computer Collection, DLC-200/MCNPDATA.
3. R.E. MacFarlane and D.W. Muir, "The NJOY Nuclear Data Processing System Version 91", LA–12740–M (1994).
4. 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.
5. J. Cetnar, J. Wallenius and W. Gudowski, "MCB: A Continuous Energy Monte Carlo Burn up Simulation Code", In: *Actinide and Fission Products Partitioning and Transmutation*, EUR 18898 EN, (OECD/NEA 1999).
6. V.Kuzminov, E.Koonen and B.Ponsard, "Monte Carlo Simulation of Irradiation of MTR Fuel Plates in the BR2 Reactor using a full-scale 3-D Model with Inclined Channels", *Proceedings of the 6th Int. Topical Meeting on Research Reactor Fuel Management*, Ghent, Belgium, March (2002), p. 206-210.
7. S.Kalcheva, E.Koonen, B.Ponsard, "Accuracy of the Monte Carlo Criticality Calculations During BR2 Operation", *Proceedings on CD-ROM of ANFMIII*, Hilton Head Island, USA, October 5-8 (2003), 18-01.pdf.
8. SASH2: SCALE System Module: A Coupled One-Dimensional Depletion and Shielding Analysis Module, Oak Ridge National Laboratory, NUREG/CR-0200, Revision 6, vol. 1, section S2. ORNL/NUREG/CSD-2/V2/R6, March 1998.
9. IRDF-90/G file. Version 2. IAEA, NUCLEAR DATA SECTION.
10. M.F.James, "Energy Released in Fission", *Journal of Nuclear Energy*, **Vol.23**, pp.517 to 536. Pergamon Press 1969, Printed in Northern Ireland Author(s).
11. V.Kuzminov and E.Koonen, "Power Distribution in MOX Fuel Rods. Benchmark Calculation", Scientific Report SCK•CEN-BLG-951, June, 2003.
12. V.Kuzminov, M.Weber and E.Koonen, "Determination of the Linear Power in MOX Fuel Rods Irradiated at BR2 Reactor", *Proceedings on CD-ROM of PHYSOR-2004*, Chicago, Illinois (U.S.A.), April 2004.
13. S.Kalcheva and E.Koonen, "Total Heating in Dummy Rod (AISI304) and in Fuel Segments for the Experiment GERONIMO: cycles 04/2002A; 01/2003A,B,C,D; 05/2003A,B",

Technical Notes, SCK•CEN, BR2-SCU/SK/D085021/geron_dum/11/11/03, BR2-SCU/SK/D085021/geron_fuel/14/12/03.

14. L.Vermeeren, Ph.Gouat, S.Kalcheva and A.Verwimp, "GERONIMO Transient Tests at BR2", *At Meeting of GERONIMO Third PCM/TOP-GUN Second PCM*, Brussels, November 2003.

15. V.Kuzminov and E.Koonen, "Irradiation of Two UMo Fuel Plates in BR2", *Proceedings of the 8th Int. Topical Meeting on Research Reactor Fuel Management*, Munchen, Germany, March (2004), p. 287-191.

16. V.Kuzminov, "Revision Calculation of Neutron and Heat Fluxes in the FUTURE Experiments: UAl–UMo fuel plates (cycles 02/2001A, 04/2002-05/2002)", Technical Notes, SCK•CEN, BR2-SCU/VK/A001054/FUTURE/07.11.03.

17. L.Vermeeren, V.Kuzminov, B.Coupé, E.Koonen and B.Ponsard, "On-Line Monitoring of Local Irradiation Conditions in BR2", *Proceedings of the 7th Int. Topical Meeting on Research Reactor Fuel Management*, Aix-en-Provence, France, March (2003), p. 104-108.

18. V.Kuzminov, "Fission Flux and Spectrum of Fast Neutrons in the RADAMO-10 Sample, Located in Rod I of IPS3", Technical Note, SCK•CEN, BR2-SCU/VK/F04302/RADAMO-10/05.09.03.

19. S.Kalcheva, E.Koonen, " MCNP-4C calculations of reaction rates, neutron fluxes and spectra in steel specimens in CALLISTO IPS3 (K311) and IPS1 (K49)", SCK•CEN, Technical note: BR2-SCU/SK/F044063/fluxes IPS3_IPS1/9/12/09, 26 March, 2003

20. S.Kalcheva, E.Koonen, " MCNP-4C calculations of fast and fission neutron fluxes in steel specimens from reactor ATUCHA (Project TANGO)", SCK•CEN, Technical note: BR2-SCU/SK/F044063/fluxes TANGO/14/11/03, 14 November, 2003

21. V.Kuzminov, "Fission and Fast Fluxes in the Central Gap of MISTRAL-1 and MISTRAL-2 Fuel Elements", Technical Note, SCK•CEN, BR2-SCU/VK/F04302/fluxes-dpa/23.06.04.

22. V.Kuzminov, "Fission and Fast Fluxes in the SPIRE Experiment in Cycles 01/2003A, 03/2003A, Technical Note, SCK•CEN, BR2-SCU/VK/F04302/SPIRE1/10.02.04.

23. S.Kalcheva and B.Ponsard, "Optimization Calculations of PRF-4 Device Using MCNP-4C&ORIGEN-S", SCK•CEN, Technical Note: BR2-SCU/SKBP/D025028 /prf4_0502A/17/05/2003.

24. S.Kalcheva, "MCNP-4C&ORIGEN-S Calculations of BR2 Operation Cycles", SCK•CEN, Technical Notes, BR2-SCU/SK/A001040/cycle_0304A.2/15/07/04.

25. S.Kalcheva, "MCNP-4C Evaluations of Azimuth Heat Flux Distributions in Hot Plane of Outer Fuel Plates of BR2 Fuel Elements", SCK•CEN, Technical Notes, BR2-SCU/ SK/A001040/safety_0503A/14/11/03.

26. S.Kalcheva, E.Koonen and P.Gubel, "Assessment of Fuel Safety Margins with MCNP", *Proceedings on CD-ROM of Best Estimate 2004*, Washington D.C., USA, November 14-18 (2004), p.288-301.

27. S.Kalcheva, B.Ponsard and E.Koonen, "Reactivity Effects Due to the Beryllium Poisoning of BR2", *Proceedings on CD-ROM of PHYSOR-2004*, Chicago, Illinois (U.S.A.), April 2004.

28. S.Kalcheva, E.Koonen and P.Gubel, "Scenarios of the Reactivity Evolution in the Beryllium Matrix Poisoned up to 2010", SCK•CEN, Technical Note: BR2-SCU/SK/A001044/ Be_matrix_2010/06/22/04.

29. S.Kaltcheva and E.Koonen, "Enhancement of Local Fast Flux in the Axis of a Standard BR2 Fuel Element", *Proceedings of the 7th Int. Topical Meeting on Research Reactor Fuel Management*, Aix-en-Provence, France, March (2003), p. 104-108.