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Predicting Ex-core Detector Response in a PWR with Monte Carlo Neutron Transport Methods

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***Abstract*—An approach for calculating ex-core detector response using Monte Carlo code MCNP was developed. As a first step towards ex-core detector response prediction a detailed MCNP model of the reactor core was made. A script called McCord was developed as a link between deterministic program package CORD-2 and Monte Carlo code MCNP. It automatically generates an MCNP input from the CORD-2 data. A detailed MCNP core model was used to calculate 3D power distributions inside the core. Calculated power distributions were verified by comparison to the CORD-2 calculations, which is currently used for core design calculation verification of the Krško nuclear power plant. For the hot zero power configuration, the deviations are within 3 % for majority of fuel assemblies and slightly higher for fuel assemblies located at the core periphery. The computational model was further verified by comparing the calculated control rod worth to the CORD-2 results. The deviations were within 50 pcm and considered acceptable. The research will in future be supplemented with the in-core and ex-core detector signal calculations and neutron transport outside the reactor core.**

I. Introduction

Knowing the reactor power level and its distribution inside the reactor core at any moment is of utmost importance for a safe operation of a nuclear power plant. Neutron detectors positioned outside the reactor core enable continuous power reading. As a first step towards predicting their response, a detailed model of a typical pressurized water reactor (PWR) core was developed with the Monte Carlo neutron transport code MCNP [1]. As a typical pressurized water reactor, a Krško nuclear power plant (NPP) was chosen. The Monte Carlo method was considered because it enables simulation of a more complex and detailed geometry, compared to the deterministic methods, which are more commonly used. However, its downside is its calculation time, which drastically increases for reactor simulations, where a large number of neutron histories and low stochastic errors are needed. Nowadays, with the general increase of the availability of computational resources, also Monte Carlo simulations of a sophisticated reactor became feasible.

To enable safe and continuous operation of the nuclear power plant it is important to accurately control the reactivity. Neutron flux profile inside the reactor core strongly depends on the control rod position [2], [3]. This redistribution alter the reading of ex-core detectors, which can lead to the non-linear power detector response and distort subsequent control rod worth determination from the ex-core detectors reading.

II. Krško Nuclear Power Plant

The Krško NPP is a Westinghouse design and has a two loop pressurized water reactor. Currently, the thermal rating is 1994 MWt with 727 MWe gross electric production. The core is composed out of 121 fuel assemblies (see Fig. 1).

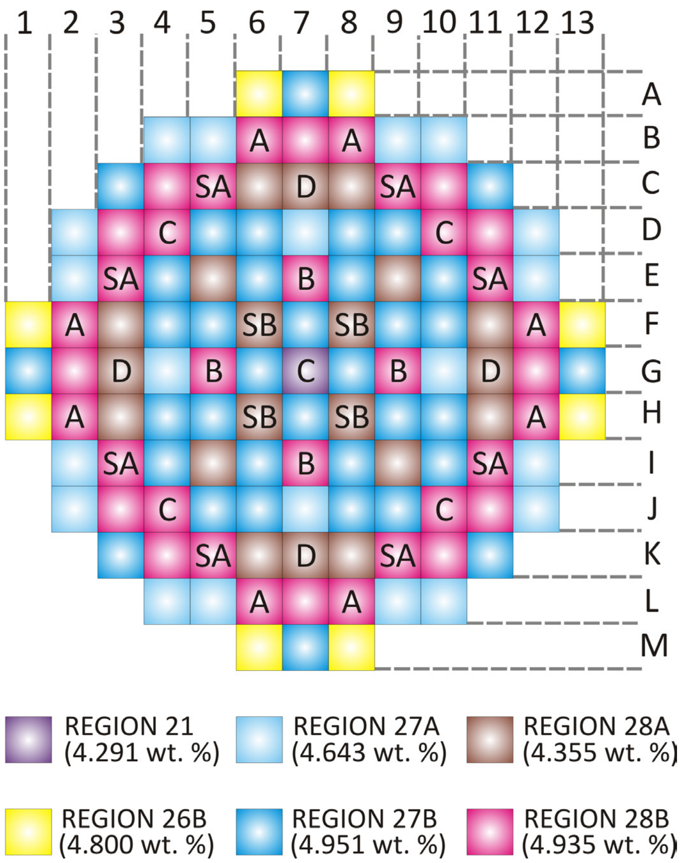


Fig. 1: Core configuration for Cycle 26 with marked fuel regions in different color and control rod cluster positions.

Fuel assembly has a 16×16 lattice filled with 235 fuel rods, 20 guides for control rods and 1 guide for in-core instrumentation as presented in Fig. 2. Integral Fuel Burnable Absorbers (IFBA) are added to some fuel rods to enable long term reactivity control. Typically six different IFBA patterns are used, one of them with 32 IFBA rods is presented in Fig. 2.

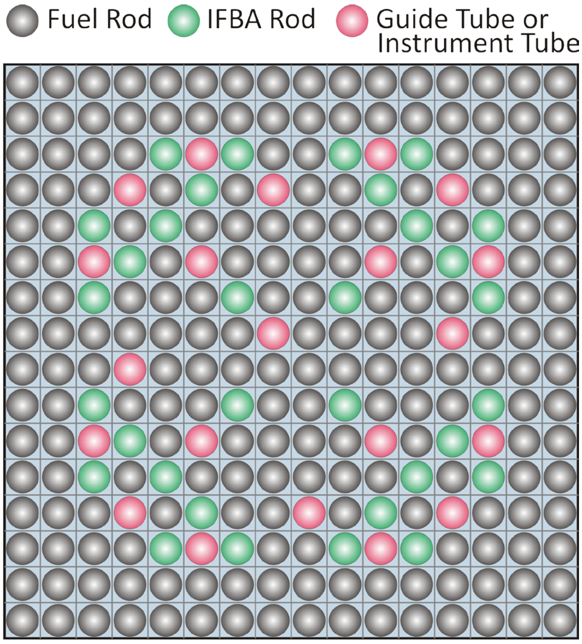


Fig. 2: Schematic drawing of fuel assembly with 32 IFBA rods.

A. Neutron Detectors

The ex-core detector system monitors neutron flux from shutdown conditions to 200 % of full power. This represents neutron flux variations from 10^{-1}-10^{11} n/(cm^2 s), therefore three types of neutron detectors are utilized: BF_3 counter (source range), compensated ionization chamber (intermediate range) and uncompensated ionization chamber (power range). The ex-core detectors are placed in wells, which are located on the cavity wall. Power range detectors are positioned in four equally spaced locations around the core. There are 4 power range channels with 2 vertical detectors per channel.

The in-core neutron instrumentation consist of 4 movable neutron detectors, which can be positioned in 36 in-core positions. The movable detectors are placed in the instrumentation guide thimbles selected so as to monitor a significant number of representative assemblies [4]. Taking into account the reflection of data across the core centrelines, a complete 3D core flux map may be obtained.

B. Control Rod Calibration

Inside the reactor core are 33 rod cluster control assemblies (RCCA) [4] distributed as presented in Fig. 1. A RCCA consist of 20 individual Ag-In-Cd absorber rods fastened at the top to a common spyder assembly. To enable safe operation of the nuclear reactor it is important to accurately know the control rod worth. The control rod worth can be measured using different techniques (i.e. boron dilution and rod swap method). A newer method, called the rod insertion method [5], [6] was developed at the Jožef Stefan Institute (JSI) Reactor Physics Department. It relies on the analysis of the reactor signal, which is recorded during continuous insertion of the control rod bank. Its major advantage is high execution speed (approximately 15 minutes per control rod bank) in contrast to the boron dilution method, which takes about 4 hours. During the insertion of the RCCA the spatial distribution of neutron population is changed [2], [3]. Since the detector measures local neutron flux at the detector location, this can lead to the non-linear power reading. To account for those redistributions, the neutron flux redistribution factors as a function of RCCA axial position are introduced. Neutron flux redistribution factors currently used in the NPP Krško have been determined with a single adjoint flux distribution calculation for the first operational cycle [7]. The calculation was performed using 2D deterministic code DOT [8]. In future research those factors will be updated using Monte Carlo models described in this paper.

III. Calculation Procedure

The calculation procedure designed to obtain ex-core detector response in a typical PWR NPP is schematically presented in Fig. 3. Firstly, the deterministic code package CORD-2 [9] is used to obtain fuel temperature, water temperature, water density, fuel isotopic composition due to the burn-up, burnable absorber layer (IFBA) and boron concentration for each fuel assembly in 10 axial layers. In next step a newly developed subroutine McCord is used to generate MCNP input from the CORD-2 output data. Using MCNP core input power and neutron flux distribution inside the reactor core can be analysed and redistribution due to the control rod movement can be studied. Moreover, in-core detector response can be anlyzed and compared to measurements to validate the MCNP core model. The MCNP core input will be used to calculate the neutron source for the MCNP ex-core model. The MCNP ex-core model includes ex-core structures, such as: reactor vesel, ex-core neutron detectors and all the surrounding concrete structures. Using hybrid code ADVANTG [10] weight windows for neutron transport will be generated and used in the MCNP simulation to study the ex-core detector response and neutron dose field in the containment.

A. CORD-2 calculation

Deterministic CORD-2 system [9] was developed at the JSI Reactor Physics Department and is routinely used for core design verification calculations of the Krško NPP. It consists of two basic reactor physics codes: WIMS-D [11] and GNOMER [12]. WIMS-D is a widely-used lattice code and GNOMER solves the neutron diffusion equasion in three-dimensional Cartesian geometry by Green's function NOdal MEthod [13]. As other deterministic codes, also CORD-2 includes cross-section homogenization. The thermal feedback is taken into account with a simple thermohydraulic module from the CTEMP [14] code. The CORD-2 calculation results were thoroughly verified with core design calculations and its output data (water temperatures, water densities, fuel temperature, fuel isotopic and IFBA layer) will be used as input parameters for MCNP core model.

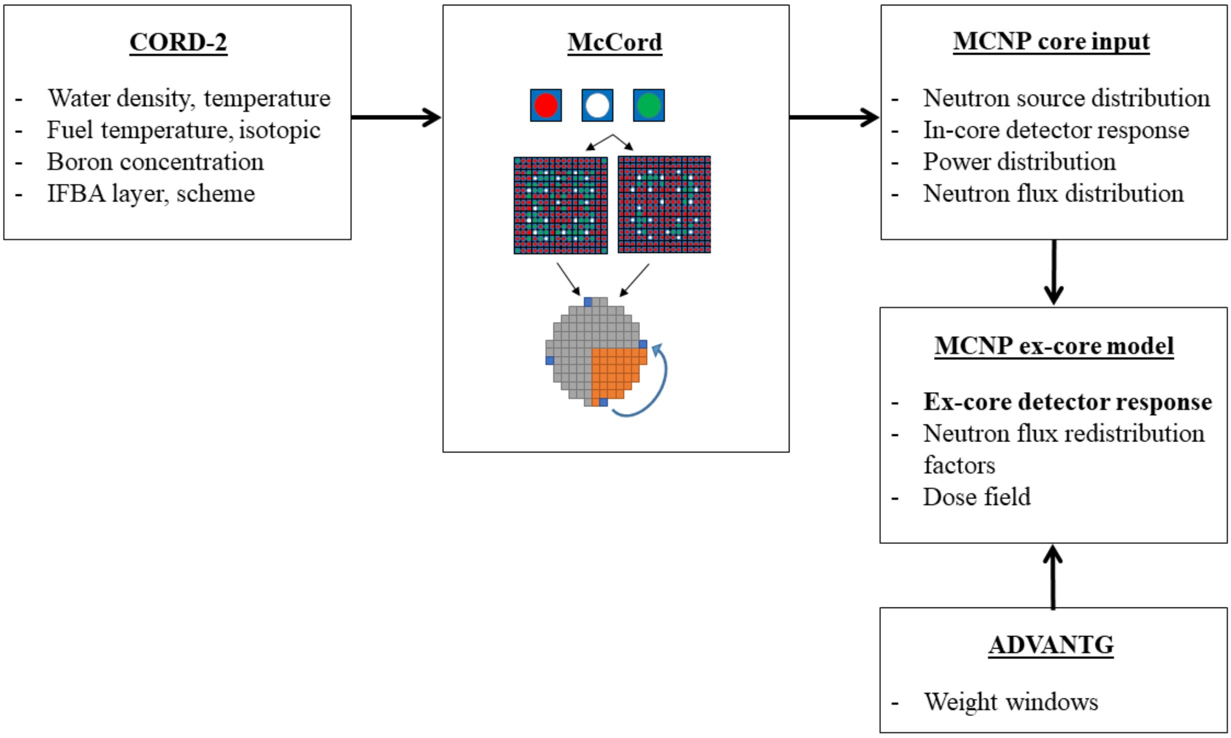


Fig. 3: Calculation procedure scheme to obtain ex-core detector response.

B. Subroutine McCord

The MCNP PWR core model is very complex and detailed (e.g. it includes more than 30.000 cells and approximatelly 1.000 materials, see Figs. 4 and 5) and is currently composed out of approximately 100.000 input lines.

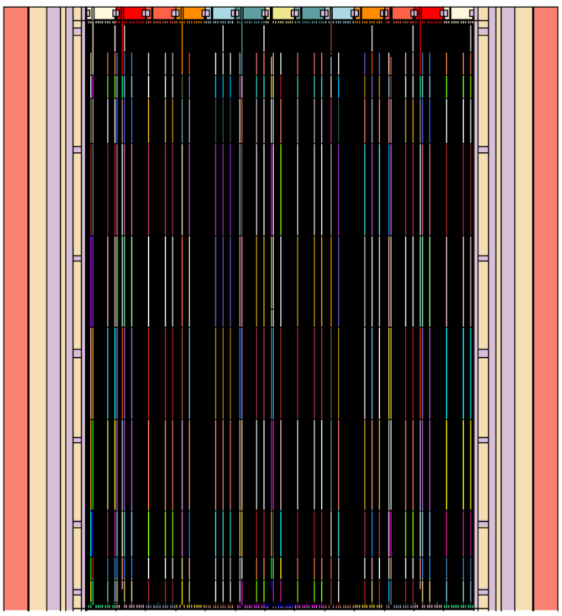


Fig. 4: xz view of the MCNP reactor core model.

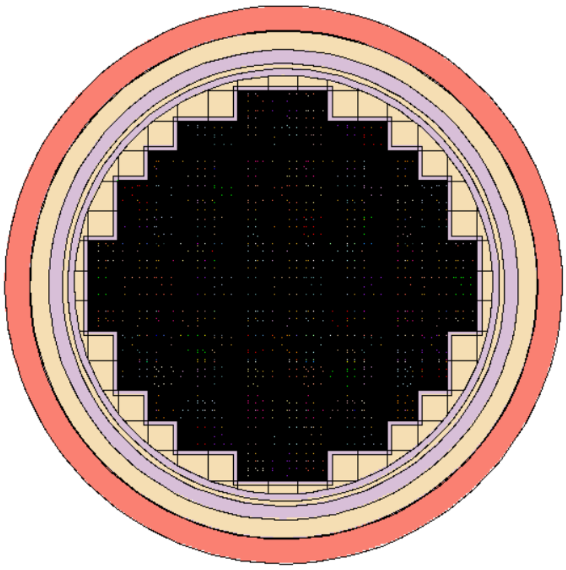


Fig. 5: xy view of the MCNP reactor core model.

To enable prompt generation of MCNP core inputs for different cycles and parts of cycle (e.g. beginning of life or end of life) the process must be completely automated. A subroutine McCord was developed at the JSI Reactor Physics Department and is capable of automated generation of MCNP core input from the CORD-2 output data. It serves as a link between deterministic CORD-2 and Monte Carlo MCNP calculations.

In the McCord code the generation of MCNP input must be very systematic [15]. The basic cell is a fuel pin divided into 10 axial layers and surrounded with water. Different fuel pin cells are described depending on their position inside the fuel assembly and presence of IFBA coating. Each fuel pin layer has dedicated its own water density, fuel density (calculated from fuel isotopic) and IFBA coating, all of which are output from the CORD-2 calculation.

Each axial layer of fuel pin has its own fuel and water material specified taking into account their temperature and fuel isotopic from the CORD-2 output. To describe the materials at desired temperature two neutron cross section libraries at nearest temperatures were selected and atomic fractions of isotopes were calculated using pseudo materials method [16]. Thermal scattering cross sections $S_{\alpha,\beta}$ for fuel (U-UO_2 and O_2-UO_2) were taken at the nearest given temperature. $S_{\alpha,\beta}$ cross sections for water (H-H_2O) have higher impact on the results and were therefore generated in 5 K intervals using MAKXSF code [17] to interpolate pre-calculated thermal scattering libraries, then the $S_{\alpha,\beta}$ cross section at nearest temperature was taken.

The fuel assembly lattice is formed from fuel pin cells and guide tubes for control rods and instrumentation. When forming the fuel assembly lattice the IFBA pattern (number of IFBA coated fuel pins in fuel assembly) must be taken into account. Information about the IFBA pattern of individual fuel assembly is also described in the CORD-2 output file. In total 37 fuel assemblies are generated and translated taking into account quadrant symmetry to form a core lattice. Control rods are added in banks and their axial position can be easily altered. Above the active fuel height top nozzles are placed and below active fuel region bottom nozzles are positioned. Around the active core part the baffle, core barrel, former plates, thermal shield, inner reactor vessel cladding and reactor vessel are modelled as can be observed in Fig. 4 and 5. Between those structures the water with inlet water characteristics is added.

C. MCNP core model calculation and neutron source generation

The MCNP core model produced with McCord subroutine can be used for various purposes, e.g. to calculated 3D power distribution within the core, in-core detector response or to evaluate the control rod worth from the change in the calculated multiplication factor. The MCNP core model will also be used to generate neutron source, which will be implemented in the MCNP containment model and then used for ex-core neutron transport calculations. Fission rate and neutron spectrum will be calculated for each fuel rod in multiple axial layers [18].

D. MCNP containment model calculation

A simplified MCNP model of containment building including reactor pressure vessel and souring structures was already developed [19] and will be supplemented with explicitly modelled ex-core neutron detectors. This model will be used for ex-core neutron transport calculations. To speed up the neutron transport from core to the detectors a hybrid code ADVANTG [10] will be used.

IV. Monte Carlo Computational Model of the Reactor Core

The detailed Monte Carlo model of the reactor core was made using Monte Carlo Neutron Transport Code MCNP [1] and is presented in Fig. 4 and Fig. 5. A great effort was put into designing the core model in a systematic way to enable quick generation of inputs for desired cycle configuration [15] using specially developed subroutine. In the MCNP core model elemental cell is fuel pin and different fuel pins form a fuel assembly lattice. Fig. 6 shows closer view of the MCNP fuel assembly model. For core reconstruction 37 fuel assemblies are modelled and translated taking into account quadrant symmetry. Core is divided into 10 axial layers, which are smaller at the top and bottom of the core and larger in the middle. Individual fuel assembly is specified its own water temperature, water density, fuel temperature, fuel isotopic and IFBA layer for each axial layer.

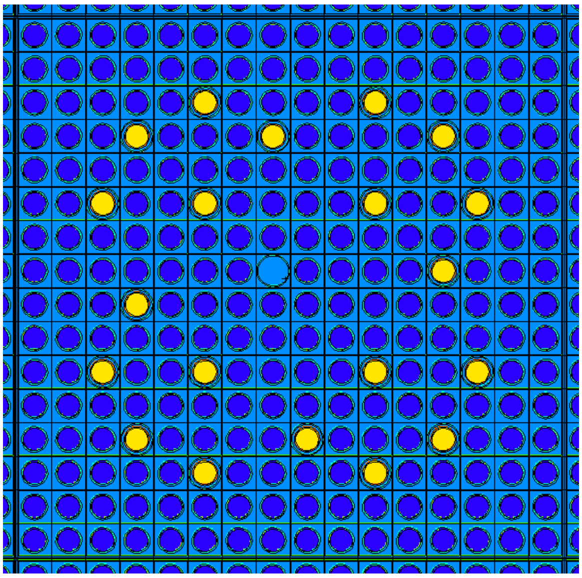


Fig. 6: Closer view of the MCNP fuel assembly model.

The detailed MCNP core model includes several geometric details, such as upper and lower nozzle, simplified model of axial grids, plenum spring, Zircaloy cladding, fuel pin plugs, control rods, etc. Fig. 7 shows closer view of the MCNP top nozzle model and Fig. 8 closer view of the completely inserted control rods. Around the core surrounding structures were added to enable more accurate calculations. In Figs. 4 and 5 baffle, core barrel, former plates, thermal shield ad reactor vessel surrounding the core can be observed.

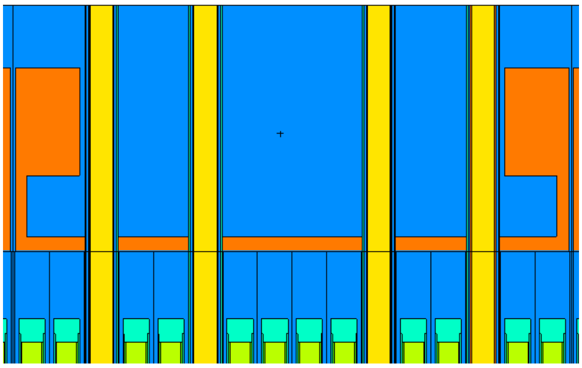


Fig. 7: Closer view of the MCNP top nozzle model.

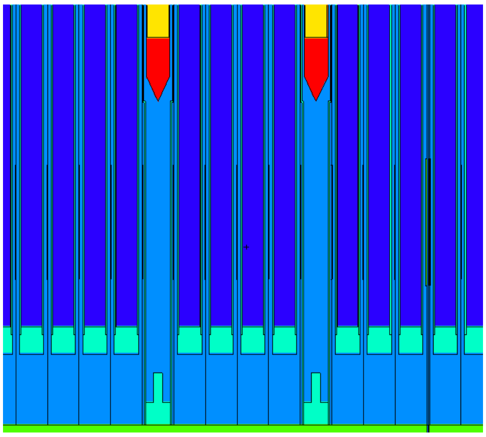


Fig. 8: Closer view of the completely inserted control rods in the MCNP model.

V. Results

A complexity of the MCNP core model makes room for mistakes, which are hard to notice. Therefore, the model has to be thoroughly validated before further use. Its results were validated with extensively verified and validated deterministic code package CORD-2. Specific power (reactor power per unit of volume) distribution inside the reactor core per fuel assembly and axial distributions were analysed. Verification continued with control rod worth determination through the calculation of multiplication factor.

A. Reactor Power Distribution

First step toward validation of the MCNP core model was comparing specific reactor power distribution to the verified and validated CORD-2 results. In Fig. 9 the calculated specific power for cycle 28 hot zero power (HZP) configuration in xy plane normalized to the average value over the entire reactor core is presented. The specific reactor power was calculated through the entire reactor core using fine geometry mesh of 250×250×1 voxels, with one volume being equivalent to approximately one fuel pin. The dips in the specific power within the core are in position of guide tubes for control rod and instrumentation.

To compare results with CORD-2 the calculated specific power over individual fuel assembly was normalized to the total specific power inside the reactor core. The MCNP results for the entire core are presented in Fig. 10, where numbers reported on top stand for normalized specific power to the total specific power and numbers reported on bottom represent stochastic 1σ uncertainty in %. The 1σ statistical uncertainty originating from Monte Carlo method is <0.2 % and therefore considered negligible. Normalized specific power (p_{MCNP}) was compared to the normalized CORD-2 results (p_{CORD}) and deviations for cycle 28 HZP configuration are presented in Fig. 11, where top number represents absolute deviation and bottom number represents relative deviation in term of $p_{CORD}/p_{MCNP} - 1$ in %. It can be observed that the deviations are systematically increasing toward core edge, this can be explained with different treatment of neutron leakage in both codes. The maximum deviation being ∼6.5 %, however the majority of fuel assemblies show good agreement between both codes with deviations <3 %.

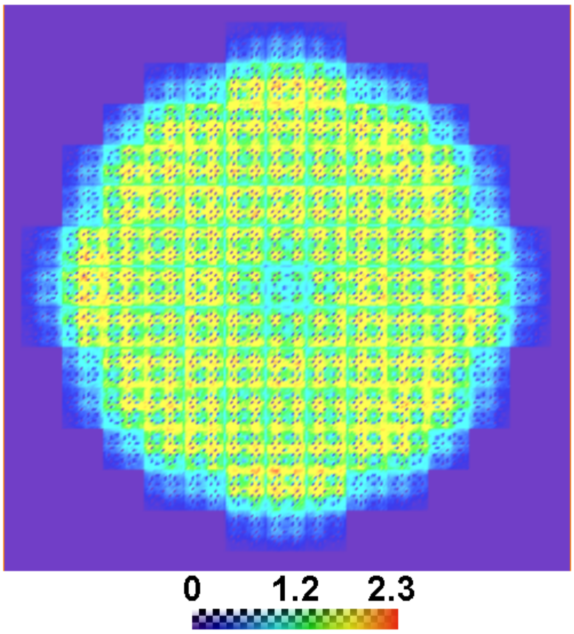


Fig. 9: Specific reactor power in xy plane for cycle 28 HZP configuration calculated using MCNP core model. The specific power values are normalized to the average over the entire core and are presented with colors ranging from low values in blue to high values in red.

To further verify the MCNP core model axial specific power distributions were compared to the CORD-2 results. Specific powers for axial profile analysis were calculated in 10 axial layers, which coincide with geometrical layers in computational model. The MCNP computational relative uncertainty of individual axial layer was below 1 % and considered negligible.

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						0.3226 0.1333	0.4486 0.1176	0.3223 0.1334					
B				0.3884 0.1264	0.6145 0.1007	1.1113 0.0819	1.2400 0.0769	1.1133 0.0818	0.6149 0.1007	0.3891 0.1262			
C			0.5101 0.1121	1.0683 0.0828	1.2052 0.0758	1.2017 0.0732	1.2425 0.0720	1.2051 0.0730	1.2066 0.0756	1.0710 0.0826	0.5161 0.1112		
D		0.3836 0.1272	1.0680 0.0828	1.2570 0.0729	1.2423 0.0731	1.1902 0.0739	1.2127 0.0736	1.1971 0.0738	1.2438 0.0730	1.2551 0.0728	1.0718 0.0826	0.3893 0.1263	
E		0.6094 0.1013	1.2074 0.0758	1.2469 0.0730	1.2781 0.0711	1.2591 0.0728	1.3225 0.0712	1.2581 0.0728	1.2777 0.0711	1.2406 0.0731	1.2077 0.0757	0.6145 0.1007	
F	0.3173 0.1343	1.1108 0.0820	1.2093 0.0731	1.2003 0.0737	1.2593 0.0727	1.2134 0.0730	1.1109 0.0764	1.2163 0.0729	1.2599 0.0727	1.1929 0.0738	1.2037 0.0732	1.1062 0.0820	0.3209 0.1338
G	0.4462 0.1179	1.2445 0.0769	1.2520 0.0718	1.2230 0.0733	1.3194 0.0712	1.1049 0.0766	0.8912 0.0812	1.1067 0.0766	1.3151 0.0714	1.2114 0.0736	1.2366 0.0721	1.2335 0.0771	0.4459 0.1180
H	0.3202 0.1339	1.1159 0.0818	1.2129 0.0730	1.2010 0.0738	1.2582 0.0728	1.2118 0.0731	1.1030 0.0766	1.2070 0.0732	1.2480 0.0730	1.1864 0.0740	1.1976 0.0733	1.1039 0.0821	0.3211 0.1338
I		0.6132 0.1010	1.2113 0.0757	1.2488 0.0730	1.2806 0.0711	1.2575 0.0728	1.3170 0.0714	1.2536 0.0729	1.2707 0.0714	1.2359 0.0733	1.2026 0.0759	0.6112 0.1010	
J		0.3833 0.1274	1.0691 0.0829	1.2587 0.0729	1.2475 0.0731	1.1970 0.0739	1.2171 0.0735	1.1956 0.0739	1.2414 0.0732	1.2481 0.0732	1.0663 0.0829	0.3892 0.1265	
K			0.5060 0.1127	1.0692 0.0829	1.2063 0.0759	1.2048 0.0732	1.2481 0.0720	1.2086 0.0732	1.2052 0.0759	1.0632 0.0830	0.5063 0.1125		
L				0.3833 0.1275	0.6061 0.1015	1.1046 0.0823	1.2360 0.0771	1.1102 0.0821	0.6068 0.1015	0.3806 0.1277			
M						0.3160 0.1348	0.4424 0.1185	0.3175 0.1344					

Fig. 10: Specific reactor power per fuel assembly for cycle 28 HZP configuration calculated using MCNP core model. Top number represents specific power normalized to the one over reactor core and bottom number is relative uncertainty in %.

	7	8	9	10	11	12	13
G	0.0342 3.8387	0.0263 2.3757	0.0248 1.8862	0.0177 1.4574	0.0055 0.4443	−0.0340 −2.7432	−0.0233 −5.2294
H	0.0252 2.2724	0.0290 2.3914	0.0269 2.1394	0.0189 1.5828	0.0064 0.5308	−0.0313 −2.8211	−0.0200 −6.2577
I	0.0223 1.6925	0.0254 2.0195	0.0208 1.6302	0.0140 1.1247	−0.0029 −0.2407	−0.0122 −1.9946	
J	0.0200 1.6451	0.0189 1.5781	0.0145 1.1679	0.0077 0.6109	−0.0157 −1.4727	−0.0118 −3.0540	
K	0.0045 0.3626	0.0030 0.2452	−0.0048 −0.3942	−0.0164 −1.5352	−0.0140 −2.7552		
L	−0.0330 −2.6661	−0.0331 −2.9797	−0.0146 −2.3910	−0.0129 −3.3520			
M	−0.0228 −5.1123	−0.0207 −6.4634					

Fig. 11: Deviation in normalized specific reactor power between CORD-2 (p(CORD)) and MCNP (p(MCNP)) for cycle 28 HZP configuration. Deviations are presented per fuel assembly for one quarter of the core. Top number represents absolute difference (p(CORD)-p(MCNP)) and bottom number relative difference (p(CORD)/p(MCNP)-1) in %.

Results for individual axial layer (i) of individual fuel assembly (j) obtained with MCNP ($p_{i,j}$) were normalized as:

$$p_{z,i,j} = \frac{p_{i,j} p_{xy,j} H}{\sum_{i=1}^{10} p_{i,j} H_i}, \tag{1}$$

$$p_{z,i} = \sum_{j=1}^{121} \frac{p_{z,i,j}}{121}, \tag{2}$$

where $p_{xy,j}$ is the normalized specific power for fuel assembly (j) from the xy power profile, H is a total active fuel hight and H_i is a hight of individual axial layer. Averaging is performed through all fuel assemblies within the core to obtain average axial profile (p_z). Results are presented in Fig. 12, where a black line represents MCNP results and a red line CORD-2 results. In Fig. 13 absolute deviations between both calculations are presented. Even though the deviations can be up to 10 %, the shape of the MCNP axial profile is similar to the shape of axial profile calculated with CORD-2 program package, which indicates that there is no sever flaw in the model.

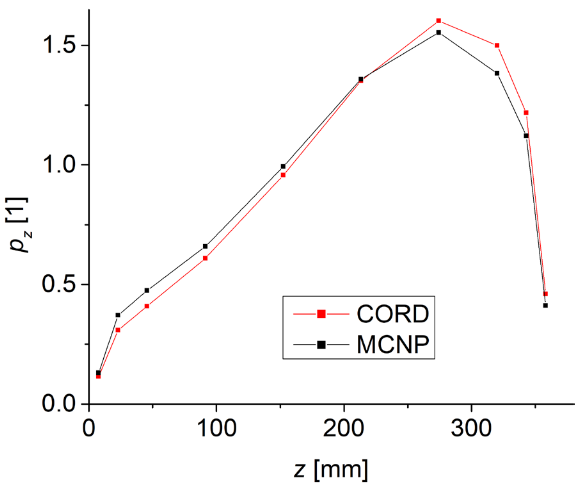


Fig. 12: Normalized specific power average axial profile calculated with CORD-2 (red) and MCNP (black) for fuel cycle 28 HZP configuration.

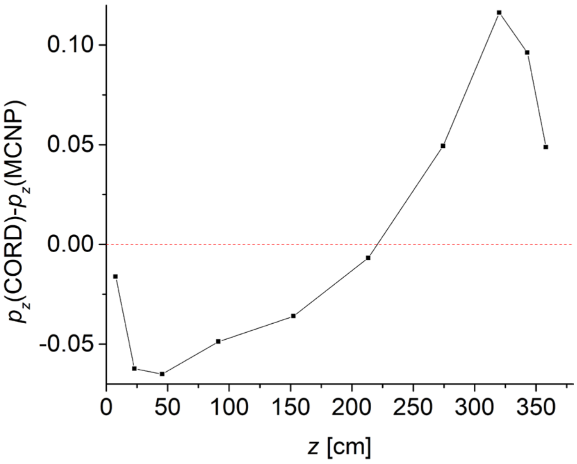


Fig. 13: Absolute deviation between normalized CORD and MCNP average axial profiles for fuel cycle 28 HZP configuration.

B. Control Rod Worth

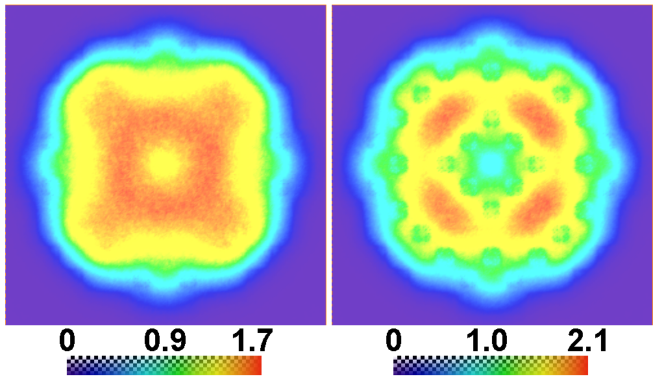
Control rod worth for different RCCA banks (see Fig. 1) was determined by comparing multiplication factor of calculation with all banks completely withdrawn and calculation with bank of interest fully inserted. The results are presented in Table I, where W_{RI} represents the measured RCCA worth through the rod insertion method and W_M and W_C represent the calculated MCNP and CORD-2 RCCA worth respectively. It can be observed that calculated values between the MCNP and CORD-2 deviate for less then 50 pcm. It can be concluded that the calculations agree well with each other, which additionally confirms the MCNP core model for the HZP configuration. It should be taken into account that the measured values represent the ex-core detector response and cannot be directly compared to the currently calculated values, which take into account only neutron transport within the core. The RCCA worth through calculation of ex-core detector response will be studied in future research and is out of the scope for this paper.

TABLE I: Control Rod Worth for Cycle 28.

Bank	W_{RI} [pcm]	W_M [pcm]	W_C [pcm]	W_M-W_C [pcm]
D	741	703	702	1
C	868	818	833	-15
B	706	721	741	-20
A	929	856	840	16
DCBA	3244	3098	3116	-18
SA	1385	1227	1235	-8
SB	591	638	608	30
SA+SB	1976	1866	1843	23
Total	5220	4964	4959	5

C. Neutron flux redistribution due to the RCCA movement

The neutron flux redistribution due to the RCCA movement was studied and is presented in Fig. 14.



(a) A bank completely inserted. (b) All banks completely inserted.

Fig. 14: Neutron flux in xy plane, normalized to the average value of 1 through the entire core for cycle 28 HZP configuration calculated using MCNP core model. Normalized neutron flux values are represented with colors, ranging from low values in blue to high values in red.

The neutron flux was calculated through the entire reactor core using fine geometry mesh of 250×250×1 voxels, with one volume being equivalent to approximately one fuel pin. Flux was normalized to get 1 as average value through the reactor core. It can be observed that RCCA movement can significantly affect the neutron flux distribution and therefore alters the reading of ex-core neutron detector, which is also used for RCCA worth determination. To account for this effect the neutron flux redistribution factors are introduced when evaluating RCCA worth with rod insertion method. Their determination will be a subject of further studies and is out of the scope for this paper.

VI. Future Research

After validation and verification of the MCNP core model the focus of the future research will be neutron transport outside the reactor core, with the aim to determine ex-core detector response. For those calculations simplified MCNP model of the reactor pressure vessel, surrounding structures including ex-core detectors and containment building presented in Fig. 15 will be used [19]. Firstly, the core model will be used to generate neutron source, which will then be used inside the ex-core model. Fission rate and spectrum will be determined for each fuel rod in multiple axial layers [18]. To speed up neutron transport from core to the e.g. neutron detectors, a hybrid code ADVANTAG [10] will be used. The study will be supplemented with the effect of the control rod movement on the ex-core detector response, which will enable the calculation of neutron flux redistribution factors for each fuel cycle individually, therefore significantly improving the control rod worth determination, which is a safety related parameter.

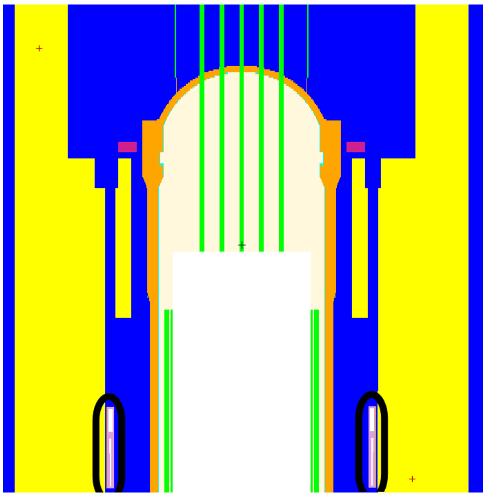


Fig. 15: Simplified reactor vessel and surrounding structure MCNP model with ex-core neutron detectors marked within black ellipse.

VII. Conclusion

The ex-core neutron detectors are in nuclear power plant used to monitor the reactor power during normal operation or when performing different physical tests, e.g. rod insertion for control rod worth determination. Therefore, it is important to know their response as accurately as possible. To predict their response a calculation procedure using Monte Carlo neutron transport was developed. Currently, the majority of core design and control rod worth calculations are performed using deterministic programs, which include geometry simplifications. Using Monte Carlo method would enable simulation of more complex geometry and more accurate prediction of the detector response. The detailed MCNP model of a typical PWR reactor core was developed and automated generation of inputs performed by newly developed subroutine McCord. Hot zero power configuration was verified with deterministic CORD-2 results and the agreement was considered acceptable. In future research the MCNP core model will be validated by in-core measurements and neutron transport outside the reactor core will be performed to study ex-core detector response.

Acknowledgment

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