CRITICALITY SAFETY AND DETECTOR RESPONSE CALCULATIONS OF THE VVER-440/V213 REACTOR

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1. Introduction

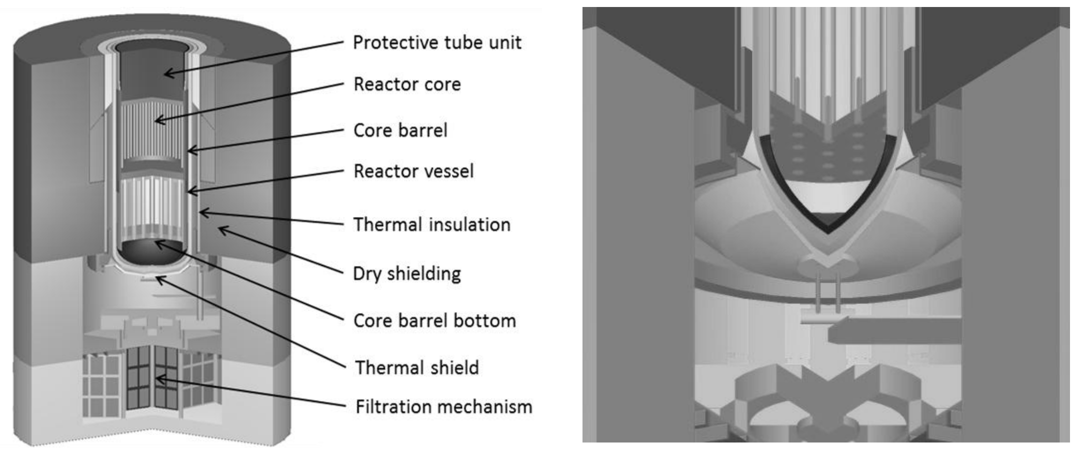
The safe operation of a nuclear power plant requires intensive examination of criticality states in all areas of the fuel cycle. The key issue is to estimate and to predict the deviation of calculation from reality. The fundamental assumption is that, the calculation bias is mostly caused by errors in the cross-section data. In addition, the use of random variables (e.g. Monte Carlo techniques) in the calculation introduces a non-random bias in the computed result as well. The computational bias of criticality safety calculations must be established through the validation of the applied methods to critical experiments.

A nuclear reactor is a prolific source of potentially dangerous nuclear radiation. This radiation is unavoidable, since most of the radiation released originates with the fission process itself. In addition to the energetic neutrons, gamma rays are emitted simultaneously with the fission event. To enable personnel to work in the vicinity of an operating reactor, it is necessary to absorb the nuclear radiation released in a thick shield surrounding the core. Even for regions where human access in not permitted during reactor operation, the shielding may be necessary to limit the activation and possible destruction of construction materials or electronic devices. Although many types of nuclear particles are released, the essential shielding problem is the attenuation of penetrative fast neutrons and high-energy gamma rays that are released in the reactor core and the reactor shield. Moreover, these neutrons are detected by ex-core detectors, to give reliable information about the core condition, and are widely used in noise techniques monitoring vibrations of fuel and other structural materials.

To fulfill the present needs in the Slovak nuclear industry detailed and precise KENO 3D model of the VVER-440/V213 reactor has been developed for criticality, shielding and detector response calculations. The model was partially validated by the criticality calculation of the real operational conditions reached on the 310th effective day in Slovak NPP Bohunice unit 4 during cycle 30. This paper investigates several modelling issues associated with VVER-440 criticality and shielding calculations using the SCALE computational system [1].

2. VVER-440/V213 model description

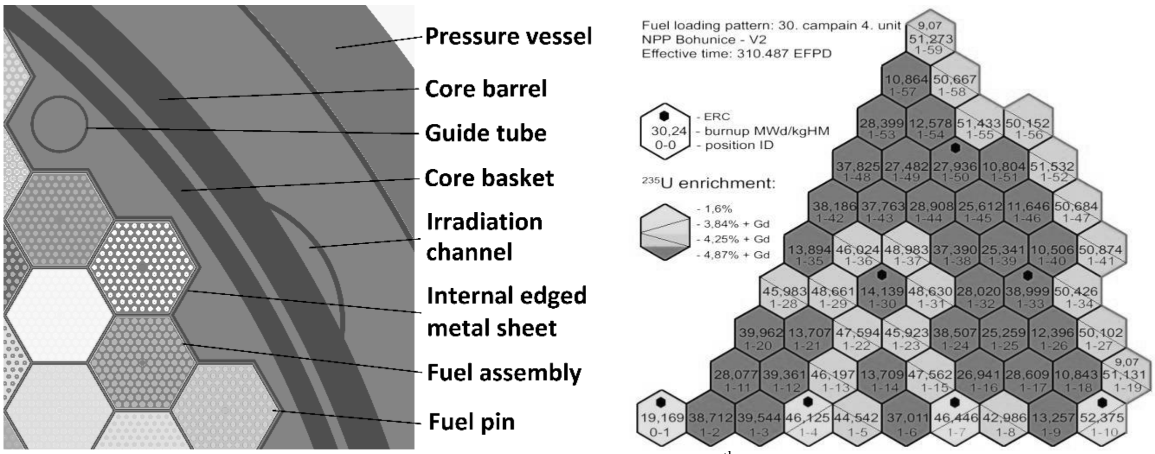
The whole-core 3D model shown in Fig. 1-a consists of the reactor in-vessel components, such as fuel assemblies (including fuel rods, upper spacer grid, intermediate spacer grids, supporting grid, mixing grid, central tube and fuel endings), emergency reactor control assemblies (ERC - absorber and fuel part), core basket, barrel and the reactor pressure vessel. The thermal shield and associated technology model can be found in Fig. 1-b. The boundaries of the created VVER-440 whole-core model are given by the outer surface of the dry shielding, the level of hot-leg piping and the basement of filtration mechanism.



a) 3D model geometry b) detail of thermal shield

Fig. 1: *KENO 3D model of the VVER-440/V213 and fuel loading pattern.*

As can be seen in Fig. 2-a, all the VVER-440 fuel assemblies (FAs) are hexagonal and the fuel rods are placed in the assembly in a triangular grid pattern. The fuel rod bundle of the assembly is enclosed in a hexagonal wrapper with the width across the flat equal to 145 mm (the 2nd gen. FA). The FA and ERC are positioned in a hexagonal grid with a spacing of 147 mm. The wrappers of FAs and ERCs are made of the E125 zirconium alloy.



a) peripheral region of the core b) 30th fuel loading pattern of NPP EBO unit 4

Fig 2: *Fuel loading pattern and adopted numbering system.*

The outside diameter of fuel rod cladding is 9.1 mm (in same cases 9.07 mm), the inside diameter is 7.75 mm. The cladding houses a fuel column assembled of UO_2 pellets. The rods are filled with helium and the fuel pellet density is 10.55 g/cm^3. Several types of profiled fuel assemblies are used to maintain power peaking factors under the design limits. The Gd_2O_3 absorber is integrated with a mass content of 3.35% into FAs to aid fuel profiling. The fuel loading pattern in the representative sixth of NPP Bohunice unit 4 during cycle 30 is shown in the Fig. 2-b. The profiling diagrams with various initial enrichments can be found elsewhere [2]. The FA with the average enrichment of 4.25 wt % ^{235}U exists in two modifications differing in rod claddings outer diameters. The fuel rod cladding outer diameter of the first modification is equal to 9.07 mm and the second modification diameter is identical to other FAs. In addition, another type of fuel rod bundle with the uniform enrichment of 1.6 wt % ^{235}U is used as a fuel part of central ERC.

3. Criticality safety calculation methodology

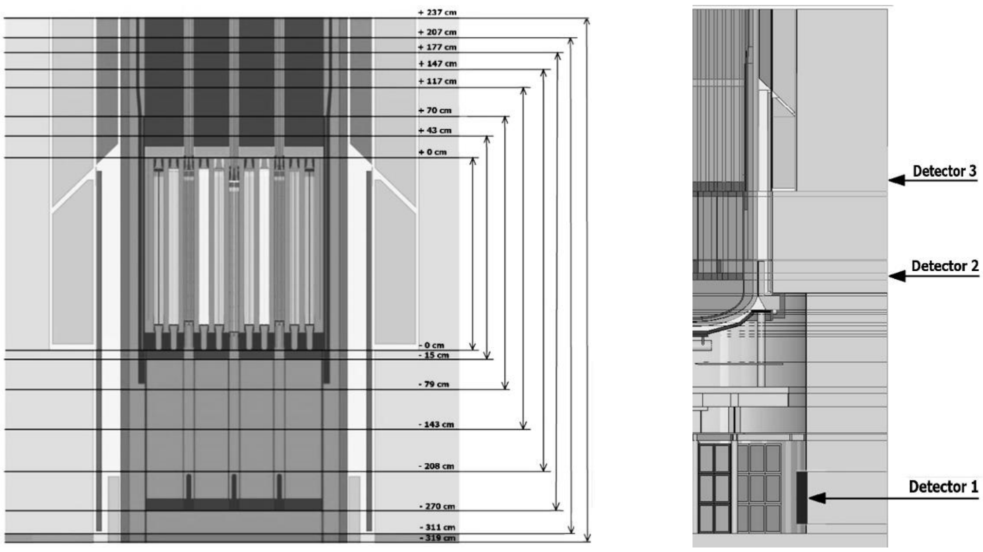
Burnup calculation

An accurate treatment of neutron transport and depletion in modern fuel assemblies characterized by heterogeneous, complex designs, such as the VVER assembly configuration, requires the use of advanced computational tools capable of simulating 3D geometries. The depletion module TRITON, which is part of the SCALE system, was used to perform depletion simulations of FA and fuel part of ERC in 2D models. The isotopic compositions of the FAs and FP ERCs following burnup levels shown in Fig. 2-b, were calculated by SCALE 6.1.3 system - TRITON depletion sequence and NEWT flux solver. Our modelling philosophy was based on parametric study of burnup modelling issues associated with VVER-440 fuel and on the best modelling approach extensively described in [2]. In this work the effects of variations in the depletion parameters, operation history, assembly type, Gadolinium presence, used energy group structure and time steps are investigated and graphically illustrated.

The isotopic vectors calculated for each FA and ERC and associated burnup value were used in the following criticality calculation.

Criticality calculation

The core material composition was based on the real data available in the NPP documentation. Following the burnup calculation, 59 isotopic vectors of fuel material were defined and homogeneously distributed in the individual FA's and ERC's definitions. All of these FAs and ERCs were placed to the model of core based on the real loading pattern shown in the Fig. 2-b. The temperature specification for criticality calculation was the same as in burnup calculation; i.e. 933 K for fuel, 555 K for incore structural materials and coolant, 541 K respectively 571 K for coolant and structural materials depending on the inlet or outlet coolant site of model. The boric acid concentration (0.09 g/kg) and the position of sixth group of ERCs (232 cm) correspond to the values achieved on the 310th efpd of the real operation for which the fuel depletion was determined.



a) truncation planes b) detector positions

Fig 3: *Truncation planes and detector positions.*

To minimize the costs of criticality and fission source calculations, it was desirable to truncate the new developed whole reactor model just to the region relevantly influencing the core neutron balance. In our approach, just the axial dimension was optimized. The radial dimension was kept intact, due to the model requirements of further shielding analyzes. 8 variations of axial model truncation were studied, where white boundaries where used at the new model peripheries. The 27 group neutron cross section (XS) library based on the ENDF/B-VII evaluated data was used in related KENO 6 calculations. Fig. 3-a gives visual and numerical information about the distances from the top and the bottom of fuel assemblies used for the model truncation. These truncation planes were chosen on the basis of the real technological and structural parts of the VVER 440 reactor design.

The criticality calculations were carried out with the chosen truncated model and with four XS libraries; Continuous Energy, 27 group and 238 group ENDF/B-VII [3], and 44 group ENDF/B-V. In case of multigroup libraries the self-shielding calculation was necessary to be performed, which means that 59 cell calculations using the BONAMI/CENTRM/PMC codes were invoked within each multigroup criticality calculation.

In case of 44 and 238 group libraries default parameters were retained during cell calculation. In the case of 27 group library the re-evaluation by CENTRM code was extended to the area of ^{238}U resonances, due to the course energy structure and library optimization for shielding transport applications [4]. To achieve acceptable statistical uncertainty of the investigated parameters in peripheral regions of the core, from 80 to 900 mill. neutron histories were calculated.

4. Detector response calculation

Fission source distribution

The KENO 6 cannot easily calculate the detector responses during its eigenvalues calculation. Therefore, KENO must be used to calculate the fission source (the spatial and energy dependent distributions of neutrons created by fission), with is then used as a fixed source by the Monaco [5] module along with appropriate variance reduction to calculate requested detector response. Since Monaco is a fixed-source Monte Carlo code, the source and particle responses can be biased regardless the fission source convergence. For this purpose, the FW-CADIS [6] automated variance reduction technique available in the SCALE system was used in our approach. To create weight windows and consistent biased source distribution the FW-CADIS calculations take advantage of two determinist calculations (one adjoint and one forward). To tally the fission distribution, very fine 3D mesh was defined in KENO calculation. The smallest mesh step reached 2 cm in the core center and the largest step equal to 15 cm was used outside the core region. The total mesh had a total of 2.5 mill. cells, which are enough cell such that the cell sizes are not very large compared to the mean-free path of neutrons in the system. To gain the relevant statistic and converge of the fission distribution, 900 mill. neutron histories were simulated. The fission source strength was normalized to represent the reactor power reached on the 310. efpd (1448.48 MWth).

Decay gamma and spontaneous fission neutron source

The gamma source, the spontaneous fission neutron source strengths and the energy spectra for each subassembly in the core were computed by the ORIGEN-S code assuming the same conditions as stated in the burnup paragraph. These data were collected and transferred to the Monaco calculation in the form of F71 binary concentration files.

Specification of the detector locations and responses

To demonstrate the capabilities of the SCALE system and the developed VVER-440 model, two point detectors were placed just behind the dry shielding. The detector No. 3 is placed on the level of bottom fuel pin endings and gives possibility to monitor the core status or can be used for activation analysis. The detector No. 2 is placed on the level of core barrel bottom and can assess the possible modifications of reactor thermal shield. The most remote detector from the core is located at the outer side of the door leads into the room A004. This region is important from the point of personnel exposure. The exact point detector locations can be found in Fig. 3-b. The detector responses were calculated according to SCALE manual and based on ICRU-57 to neutron response ID 9030 and gamma response ID 9510. The first response stands for the effective neutron dose in units of (Sv/h)/(neutrons/cm^2/s) and the second one represents the effective photon dose in units (Sv/h)/(photons/cm^2/s).

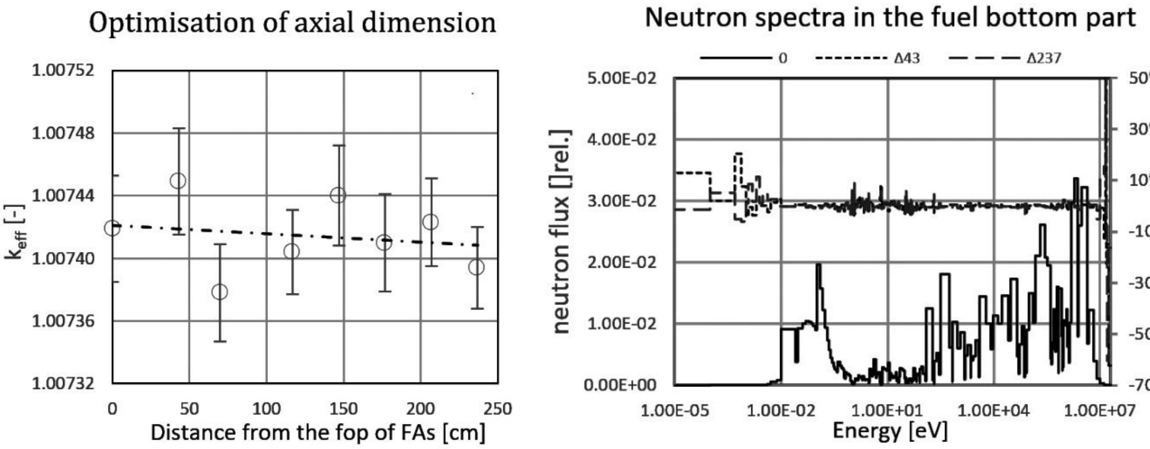
Calculation conditions

The final calculation using the CADIS-FW variance reduction technique was performed in total with using 3 billion source particles and took about 27 days. Just for the sake of curiosity, also a calculation without variance reduction techniques was carried out with 100 mill. neutron histories.

5. Results

The impact of truncation planes on the results of the effective multiplication factor are shown in Fig 4-a. The presented values are burdened with the associated Monte Carlo uncertainty. Although the trend line indicates a slight decrease in k_{eff} values towards to higher distances from the fuel, the differences between values are comparable to these uncertainties. Therefore, with respect to the computational accuracy and calculation time we can judge the closest truncation plane as satisfactory. This judgement is supported by comparison of neutron spectra in the bottom part of fuel presented in Fig.4-b. As can be seen, the neutron spectra deviations presented on the secondary axis do not differ significantly, the deviations can be mostly seen in the very low energies (about 10%) and in the fast energy range above 1.8 MeV (80%). If we assume statistical behaviour of calculated results, we can consider these differences insignificant.

The calculated fission source uncertainty was always kept below 3% inside the core region and below 7% at the core peripheral parts. These results can be considered acceptable and promise a wide application of the fissions source in the following fix-source calculations.



a) effective multiplication factors b) deviations of neutron spectra

Fig 4: *Optimization of axial dimension results.*

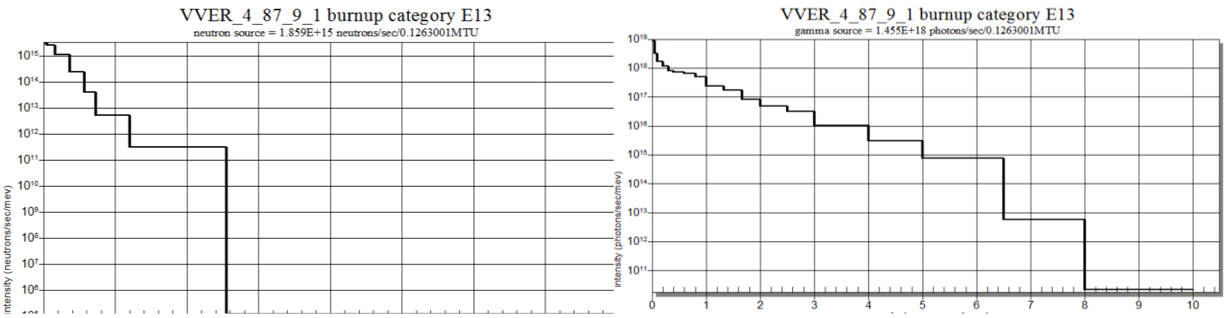
The results of the criticality calculation in case of the closest truncation plane for different energy structures are presented in Tab. 1. The lowest computational bias was achieved in case of the CE calculation. The best result using multigroup approach was calculated with 27 group library, which can be explained by the applied cell calculation not used for the finer group structures. The almost identical k_{eff} values calculated with 44 and 238 group libraries give a possibility to relevantly decrease the calculation time with preservation of accuracy.

Table 1: *Comparison of k_{eff} for different energy structures.*

Energy structure	CE	27 group	44 group	238 group
k_{eff}	1.00465±0.00006	1.00681±0.00003	1.00769±0.00006	1.00751±0.00007

The maximum deviation from real criticality is just 769 pcm, what can be in the frame of analysis provided judged as satisfactory. More detailed results of computation bias with associated uncertainties can be found elsewhere [7].

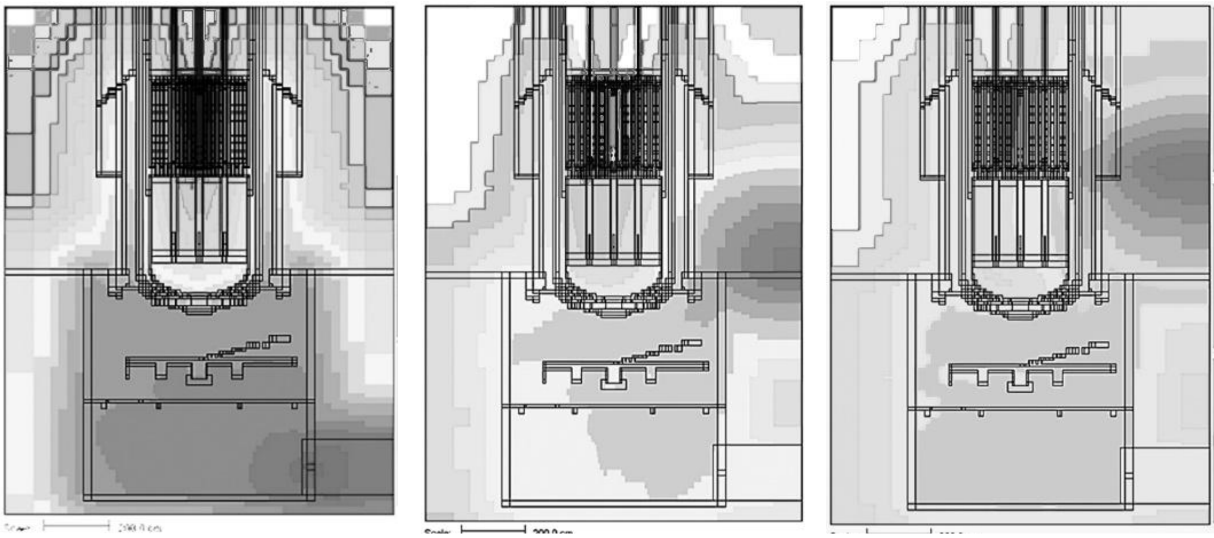
The results of gamma and spontaneous fission neutron source calculations revealed that the gamma source strengths and the spectra are for different burnups almost identical. In contrary, the spontaneous fission neutron source strengths get smaller with burnup. The difference between 10 MWd/kgU and 40 MWd/kgU for FA with average enrichment of 4.87% ^{235}U equals to almost 15 %. The spontaneous fission neutron spectra and the decay gamma spectra for the same FA and the burnup 40 MWd/kgU can be seen in Fig. 5.



a) spontaneous fission neutron source b) decay gamma source

Fig 5: *Spontaneous fission neutron and decay gamma spectrum of FA 4.87% ^{235}U.*

The results of the detector responses calculation can be split into two categories. First, problem dependent adjoint functions (importance maps) calculated by deterministic code DENOVO shown in Fig. 6 bring an interesting insight how neutrons and gamma rays contribute to the detector responses. As can be seen in Fig. 6-a, the region relevantly contributing to the detector No. 3 is the room A004 situated under the core. More surprising fact is, that neutrons do not come to this region directly through core barrel bottom, but travel through *the concentric channel* between the reactor pressure vessel and the thermal insulation.



a) detector No. 1 b) detector No. 2 c) detector No.3

Fig 6: *DENOVO problem adjoint functions.*

From the radiation point of view, the serpentine concrete used around the thermal shield serves as a barrier to neutrons, protecting the personnel from radiation overdoses near the door into room A004. The adjoint functions for detector No. 2 and No. 3 just emphasize the influence of *concentric channel* to their responses. It should be noted that the response of detector No. 3 is more sensitive to neutrons born in the opposite site of the core than to neutrons born in the core center.

The second category of results contains absolute values of calculated responses with associated uncertainties.

Table 2: *The absolute detector responses and their uncertainty.*

Effective dose (Sv/h)	detector No. 1		detector No. 2		detector No. 3	
	value	uncertainty	value	uncertainty	Value	uncertainty
photons	1,15E-04	10,99%	1,28E-09	5,50%	1,66E-07	6,35%
neutrons	2,18E-03	9,84%	7,51E-11	25,75%	2,38E-11	25,89%
combined	2,29E-03	9,55%	1,28E-09	5,50%	1,66E-07	6,35%

* Uncertainty equals to the standard deviation of calculated response.

The results shown in Tab. 2 confirm the effectiveness of the reactor dry shielding (also known as biological protection). The combined responses of detectors No. 2 & 3 situated behind this shielding are at the level of nSv/h. The combined response values are driven mainly by photons, where neutron contributions are almost negligible. The different results can be found for detector No. 1 where the combined response reaches the level of several mSv/h. Here neutrons are more relevant than photons where the difference is in order of magnitude. These quite high responses just confirm the function of *the concentric channel* already discussed above. We can consider that any construction change of thermal shield and serpentine concrete can have relevant influence to the radiation situation under the core.

6. Conclusion

The detailed model of the VVER-440/213 reactor was developed for criticality and shielding analyses including reactor core, core basket, core barrel, pressure vessel with all internals in an appropriate level of accuracy. To minimize the costs of criticality and fission source calculations, several truncation planes were introduced to the geometry model. Based on the k_{eff} values and associated neutron spectra, the truncation planes closest to the FAs were chosen as acceptable geometry simplification. The applied simplifications resulted in low computational bias of k_{eff} which did not exceed 0.8% in computational cases. Special attention was given to the methodology applied to the determination of the fuel isotopic vectors modelled in one-sixth symmetry core configuration what is one of the reasons of small calculation bias. The results of gamma and spontaneous fission neutron source calculations revealed that the gamma source strengths and the spectra are for different burnups almost identical. In contrary, spontaneous fission neutron source strengths decreased as function of burnups.

To demonstrate the capabilities of the SCALE system and the developed VVER-440 model and by using CADIS-FW variance reduction technique, detector responses placed in different locations were successfully calculated. From the statistical point of view, the analog calculation without advanced variance reduction techniques gave no answer to asked questions, therefore the use and understanding of variance reduction techniques in Monte Carlo codes is a must and cannot be avoided. The results of the detector responses confirm the effectiveness of reactor biological protection and revealed that neutrons travel through the concentric channel between the reactor pressure vessel and the thermal insulation to the room A004 situated under the core barrel bottom. We can consider that any construction change of the thermal shield and the serpentine concrete should be carefully investigated in advance, due to the possible impact to the radiation situation under the core.

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