# MCNP CALCULATION OF NEUTRON SHIELDING FOR RBMK-1500 SPENT NUCLEAR FUEL CONTAINERS SAFETY ASSESMENT

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***Abstract:*** The RBMK-1500 spent nuclear fuel (SNF) composition determine the radiation characteristics of the CONSTOR and CASTOR containers, where SNF from Ignalina NPP (Lithuania) is temporarily stored now. The extension of containers storage time depends on the knowledge of the characteristics of SNF and resulting dose rates. The fuel containers safety assessment can be performed using powerful modelling tools. Modelling of nuclide composition and it axial distribution along the fuel channel is done using the Monteburns code systems. MCNP5 code was used for the evaluation of radiation shielding and the surface dose rate of the CASTOR cask taking into account the heterogeneity of the different burnup of SNF assemblies' distribution. The simulation results confirm that neutron flux distribution is extremely sensitive to the fuel burnup and influence the neutron dose rate calculation results. The modelling results are compared with existing experimental data of the representative container leading to satisfactory agreement. This result proves the validation of calculation methodology for the radiation shielding safety assessment of CASTOR and similar containers used for the storage of RBMK-1500 SNF.

## Introduction

Part of Ignalina Nuclear Power Plant (INPP) SNF is stored in the CASTOR and CONSTOR containers [1]. The metallic CASTOR cask is designed for transportation and long term storage purposes, while heavy concrete based CONSTOR cask is usually used for the long term immobile storage. Both of these containers are designed for 50 years with possible extension up to 100 year depending on their safety conditions. To assess the radiation doses outside the containers and to evaluate their reliability after expire time, the composition of the SNF must be known and the transport of resulting source neutrons and $\gamma$ quanta modeled properly. One has to note that the extension of storage time using these containers depends on the knowledge of the characteristics of SNF and resulting dose rates. Unfortunately, there are very little published data on RBMK SNF characteristics. Even the information on neutron fluxes in the fuel elements is not known precisely. Therefore, experimental measurements of dose rates outside the representative storage containers are indispensable despite of powerful modeling tools available today for radiation transport estimates.
In this work we consider a typical CASTOR cask dedicated for RBMK-1500 SNF storage, for which radiation dose rates are calculated and compared with experimental results.

## Modeling procedure of the RBMK-1500 SNF storage cask CASTOR

A 3D model of the RBMK-1500 spent nuclear fuel storage container CASTOR has been created using the 3D MCNP5 geometry set-up (see Fig. 1 for details). The geometrical parameters, material properties and spent nuclear fuel filling history of the container were taken from the data of representative CASTOR container No.0067-14. The 2.08 m of diameter, 4.4 m of height and 30 cm of wall thickness cylindrical metallic container CASTOR is produced from cast iron. The container is filled with "32M" stainless steel basket where SNF assemblies are placed. This "32M" basket contains 51 fuel assemblies, each of them being divided into two parts (341 cm long each). The burnup of 2 % $^{235}$U enrichment spent nuclear fuel in the container varies from 13 MWd/kg to 22 MWd/kg. In order to simplify our approach only 6 types of different fuel burnup assemblies were modeled as presented in Table 1. 102 different half-assemblies were located in the specified places using MCNP5 hexagonal lattice option filled with appropriate assemblies (see Fig. 1) as reported in the CASTOR container.

To perform the calculations several computational codes had to be combined. Modeling of SNF nuclide composition was done using Monteburns code [2]. Afterwards, the gamma and neutron source spectra were calculated with the ORIGEN-ARP code from the SCALE 5 codes package. Finally, MCNP5 was used for neutron and gamma transport and evaluation of radiation dose rates at the surface of CASTOR cask. For the input data MCNP5 requires detailed 3D geometry and particle source description, material composition, their densities and specific nuclear data. In this case, ENDF/B-VI data library was used for the fuel and structure materials, while JENDL-3.2 data files were employed for fission products.

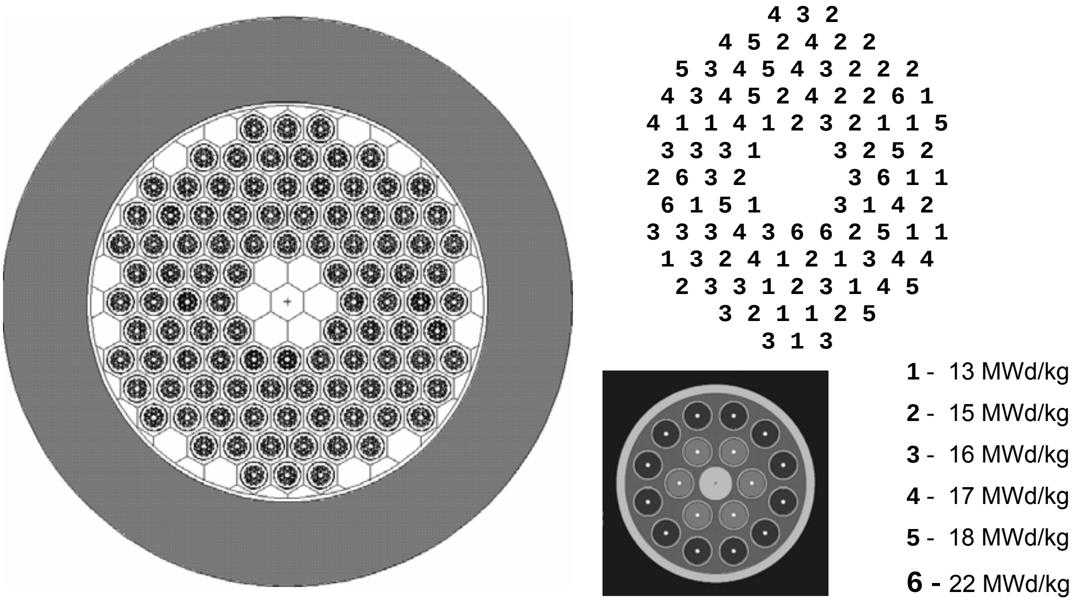


***Figure 1***. Left: cross-section of the SNF container CASTOR modeled with MCNP5; Right: the arrangement of the different burnup fuel assemblies (from 13MWd/kg (label **1**) to 22MWd/kg (label **6**)) (on the right) and cross-section of RBMK fuel assembly (lower part).

This particular container No.0067-14 was also examined experimentally with standard thermo-luminescence (TLD) and “Series 1000 MIN-RAD” dosimeters. The dose in the TLD attached at 8 points around the CASTOR container was accumulated during the period of 167 hours, the total measurement uncertainty is below 10 % [3]. In addition, $\gamma$ dose rates were also measured with MIN-RAD at 4 points at the CASTOR cask surface. The precision of MIN-RAD was somewhat worse compared to TLD but should be below 15 % [3].

## Results

Gamma source, resulting from ORIGEN-ARP calculations, for different burnup of RBMK-1500 SNF is presented in Fig. 2 (left). The energy distribution of $\gamma$ rays is very similar for every fuel burnup considered, but on the other hand, the $\gamma$ rays intensity is changing with fuel burnup. The difference of total $\gamma$ source intensity between 13 MWd/kg and 22 MWd/kg fuel is about 50 %.

The neutron spectra are also similar for different cases of fuel burnup as shown in Fig. 2 (right). The difference between 13 MWd/kg and 22 MWd/kg fuel burnup results in the increase in the neutron source intensity nearly by one order of magnitude (~8 times).

The total $\gamma$ source intensity in the cask is about $1.76{\cdot}10^{16}$ $\gamma$/s, and the total neutron source intensity is ~$7.62{\cdot}10^{7}$ n/s. These values were used in our further calculations for normalization of gamma and neutron doses calculated with MCNP5. The average intensity of the $\gamma$ source from one fuel assembly is about $3.45{\cdot}10^{14}$ $\gamma$/s, which corresponds to 16 MWd/kg fuel burnup. The averaged intensity of the neutron source from one fuel assembly is $1.49{\cdot}10^{6}$ n/s corresponding to the fuel burnup between 16 MWd/kg and 17 MWd/kg.

The dose rate calculations were performed with MCNP5 using ring detector option for two different particle source descriptions: homogeneous gamma and neutron source distribution of 16 MWd/kg SNF in all 102 half-assemblies and heterogeneous distribution taking into account different burnup SNF assemblies from 13 MWd/kg to 22 MWd/kg according to the existing irradiation history.

Due to $\gamma$ rays attenuation in CASTOR wall the photon flux decreases nearly by 10 orders of magnitude at the surface. In order to obtain a statistically significant result in a reasonable computing time specific variance reduction techniques were applied to increase the statistics of Monte Carlo calculations: the photon tracking transport importance in MCNP5 was gradually increased from 1 to $10^6$ starting from the initial to the outside wall of the cask.

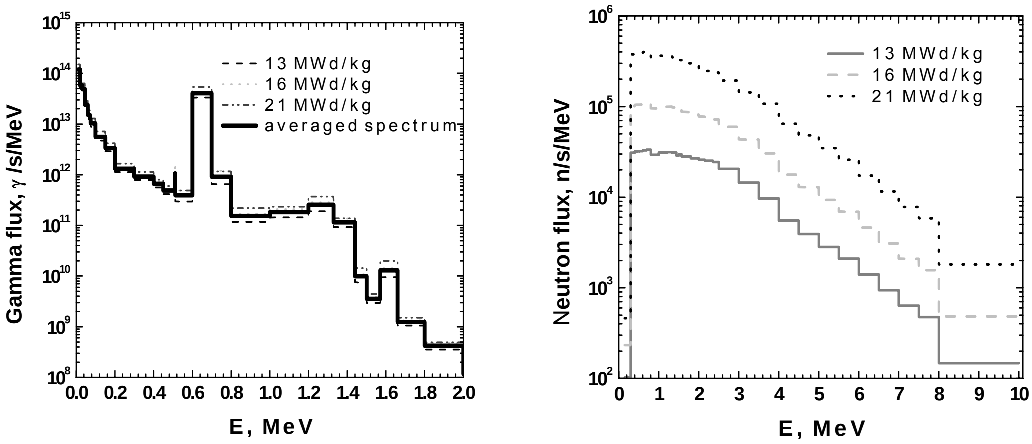


***Figure 2***. Energy spectra for different burnup SNF modeled with ORIGEN-ARP. Left: $\gamma$ source case; Right: neutron source case.

The results of $\gamma$ dose rates at the surface of CASTOR cask are presented in Fig. 3. The experimental data measured with TLD and MIN-RAD dosimeters are presented in circles and triangles, respectively. The MCNP5 modeling results are also compared to SCALE 4.3 predictions from Ref. [1]. We note that $\gamma$ dose rates obtained from Monte Carlo simulations even taking into account the neighboring containers (we assumed the $\gamma$ dose rate from the similar container surface at 1 m distance (11.8 μSv/h) and added to the calculated surface dose rate (26.8 μSv/h)) give a somewhat smaller average value of 38.6±0.4 μSv/h compared with measurements. All experimental $\gamma$ dose points are ~1.6 times above the MCNP5 calculated average, especially for MIN-RAD measurement at 180º angle, where difference is almost 2.5 times. In our opinion, this result is rather good taking into account other uncertainties of our model (spent nuclear fuel composition, irradiation and cooling history, Co contents in the fuel cladding, etc.). In addition, this difference might also be due to the CASTOR container surface contamination with fission products from the cooling pool. In addition, the TLD are sensitive to the thermal neutron flux, which also might explain the increase in the measured $\gamma$ dose.

On the other hand, the Monte Carlo calculations agree quite well with independent calculations performed with SCALE 4.3, which resulted in the 52 μSv/h dose rate after 10 years of cooling time [1]. The difference in the $\gamma$ dose after ~13 years (our work) and 10 years [1] is determined mainly by $^{106}$Rh (daughter of $^{106}$Ru) high energy gammas, which have great influence on the $\gamma$ dose due to the higher penetration through the cask walls, and by the $^{60}$Co with its relatively short half-life ($T_{1/2}$ = 5.27 years).

Neutron dose rate at the surface of the CASTOR cask calculated with MCNP5 is presented in Fig. 4. The two cases were considered: a) the case of a homogeneously distributed neutron source of 16 MWd/kg fuel burnup, and b) the case of a heterogeneously distributed neutron source taking into account fuel assemblies of higher burnup. The average neutron dose rate is about 75 μSv/h, and for the homogeneously distributed neutron source of 16 MWd/kg fuel burnup the neutron dose rate values vary from 68 μSv/h to 85 μSv/h. In addition, one sees that for the neutron source even a few assemblies with high burnup can cause observable anisotropy of neutron dose rates. This can be explained by lower neutron attenuation rates compared to gammas both in the absolute value and changes in energy spectra. For the heterogeneously distributed neutron source the average neutron dose rate is about 85 μSv/h, and the neutron dose rate values vary within the range from 76 μSv/h to 104 μSv/h. The local superposition of half-assemblies of higher burnup (16 MWd/kg - 22 MWd/kg) at the angles of 270º and 220º resulted in about 28 μSv/h increase in the neutron dose rate. This result confirms what neutron flux distribution is extremely sensitive to the fuel burnup and influences the neutron dose rate calculation results. In conclusion we would like to emphasize that neutron dose rate calculation should be performed with real SNF neutron source distribution in the container, because averaged burnup fuel (in our case 16 MWd/kg) causes incorrect results. The measurements of neutron dose rates would be very useful to validate our predictions.

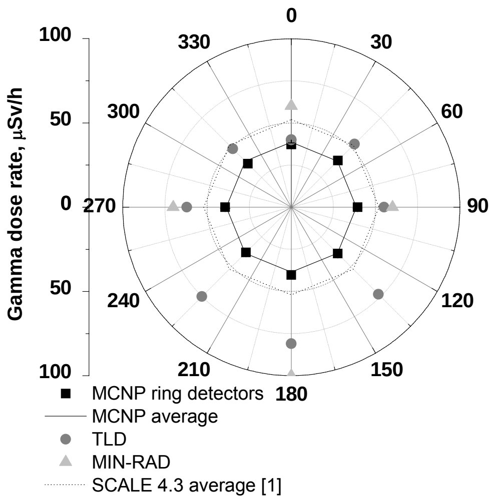


*Figure 3*. Angular gamma dose rates on the surface of CASTOR cask calculated with MCNP5 against SCALE 4.3 predictions from Ref. [1] and experimental data measured with TLD and MIN-RAD dosimeters (see the legend for details).

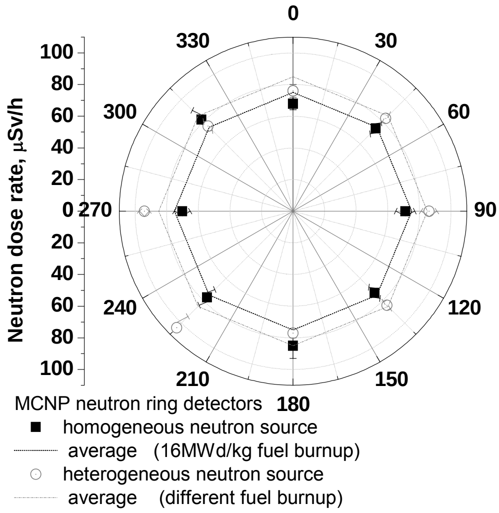


*Figure 4*. Angular neutron dose rates on the surface of CASTOR cask calculated with MCNP5 in case of homogeneously distributed neutron source and in the case of heterogeneously distributed neutron source (see the legend for details).

## Conclusions

The neutron and gamma ray sources in the SNF storage cask CASTOR have been estimated using different radiation transport and material evolution code packages as ORIGEN-ARP, MCNP5 and Monteburns. The neutron dose rate from the surface of the cask varies within the range between 76 μSv/h and 104 μSv/h. The anisotropy of dose rates is observed for neutrons and is explained by heterogeneous distribution of different burnup fuel assemblies inside the container. The estimated averaged gamma dose rate is 38.6±0.4 μSv/h, and is determined mainly by $^{137}$Cs and $^{60}$Co. This result underestimates the corresponding measurements performed with TLD and MIN-RAD dosimeters by ~50 %. The obtained discrepancies might be due to possible surface contamination, the partial gamma dosimeters sensitivity to neutron flux from the container, etc. Some additional investigations on this subject will be performed in the near future, although in general the agreement between our predictions and measurements is satisfactory. This work validates the methodology of the radiation shielding and dose rate calculations for RBMK SNF storage facilities. However, more experimental data are needed (in particular for other types of storage containers) in order to provide more precise uncertainties of the predicted values.

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