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Estimation Of Dose Rates Around Dry Storage Building During Campaign One Loading In Nuclear Power Plant Krsko

Paulina Dučkić, Davor Grgić, Mario Matijević, Radomir Ječmenica
University of Zagreb Faculty of Electrical Engineering and Computing
Unska 3
10000 Zagreb, Croatia
paulina.duckic@fer.hr, davor.grgic@fer.hr, mario.matijevic@fer.hr,
radomir.jecmenica@fer.hr

ABSTRACT

In this paper, neutron and gamma (fuel gamma, neutron induced gamma and hardware activation gamma) dose rates were estimated around the Dry Storage Building (DSB) in Nuclear Power Plant (NPP) Krško during Campaign one loading using MCNP6.2 and ADVANTG3.03 code packages. The Campaign one consists of 16 HI-STORM storage casks filled according to NPP Krsko specific fuel loading plan. The charactersitics of the spent fuel were based on real operating history and their source in terms of neutron and gamma intensity and spectrum were calculated using ORIGEN-S module from SCALE 6.2.4 code package. The annual dose at the closest site boundary and dose rates at the DSB walls are compared with the regulatory limits of 0.05 mSv and 3 μSv/h, respectively.

1 INTRODUCTION

Nuclear Power Plant (NPP) Krško opted for a Spent Fuel Dry Storage (SFDS) solution to increase its storage capabilities necessary for life-time extension of 20 years. The SFDS system consists of a Dry Storage Building (DSB) with a capacity to accept 70 HI-STORM FW [1] overpacks. The Spent Fuel Assemblies (SFAs) will be relocated from the Fuel Handling Building (FHB) to DSB using HI-TRAC VW transfer cask. They will be contained within a Multi-Purpose Container (MPC) which will be sealed and filled with helium. Each MPC can accept 37 SFAs. In total, it is foreseen that 2294 SFAs will have been generated by the 2043 and they will be stored in 62 HI-STORM overpacks. The remaining 8 overpacks might be used for temporary storage of high-level radioactive waste produced during NPP decommissioning. The SFAs will be relocated in four loading campaigns. In the first two campaigns 16 overpacks will be filled, in the third 12 overpacks, and in the last 18 [2].

The construction of the DSB at the NPP Krško site is underway and the expected start of the Campaign one loading is January 2023. Initially, the start of loading was planned for the year 2020, but that changed due to delays in the project. It is important to note that there are dose rate limits at the site boundary and at the DSB walls specific for the NPP Krško which had to be taken into account during project design. These are annual limit at the site boundary of 0.05 mSv and dose rate at the DSB walls of 3 μSv/h.

In this paper, radiation shielding analysis of DSB containing 16 overpacks from the loading Campaign one was performed to check the regulatory limits. The analysis was conducted in a hybrid shielding fashion using MCNP6.2 and ADVANTG3.0.3 codes in order to accelerate the final Monte Carlo (MC) calculations. The charactersitics of the SFAs were based on real operating history and their sources in terms of neutron and gamma intensities and spectrum were calculated using ORIGEN-S module from SCALE 6.2.4 code package. History data were coming from validated plant SFAs FAR data base and ARPLIB libraries were prepared for NPP Krško fuel using optimized depletion parameters in order to increase accuracy of predictions. The MCNP model of the DSB was developed including the HI-STORM overpacks, for which the MCNP model was adopted from our previous work [3] and [4]. In addition, this paper provides comparison of dose rates obtained for different Campaign one start loading dates (years 2023 and 2020).

2 COMPUTER CODES

MCNP6.2 [5] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of Variance Reduction (VR) techniques; a flexible tally structure; and an extensive collection of cross-section data. Energy ranges are from 1e-11 to 20 MeV for neutrons with data up to 150 MeV for some nuclides, 1 keV to 1 GeV for electrons, and 1 keV to 100 GeV for photons.

ADVANTG3.0.3 [6] is an automated tool for generating variance reduction parameters for fixed-source continuous-energy MC simulations with MCNP6.2 code, based on approximate 3D multigroup discrete ordinates forward-adjoint transport solutions generated by Denovo code. The Denovo [7] is a structured, Cartesian grid S_N solver based on the Koch-Baker-Alcouffe parallel transport sweep algorithm across the x-y domain blocks. Denovo is used in forward and adjoint mode to approximate the space-energy dependent flux across the S_N mesh. These solutions are utilized to calculate space-energy dependent biasing parameters, i.e. biased source and transport importance map (so called weight-windows), to be used as VR parameters in the MCNP. CADIS [8] methodology is used to optimize MC results in localized regions of phase-space, while FW-CADIS [9] is applied to obtain global uniform statistical uncertainty by weighting the adjoint source with expected detector response approximated with forward Denovo solution. CADIS and FW-CADIS are based on the adjoint function [10] (i.e., solution of the adjoint Boltzmann equation) which has long been recognized as the importance function for some objective function of interest.

ORIGEN-S module of the SCALE 6.2.4 code package [11] is a tool used for time dependent isotopic concentrations and source terms calculations. It consists of depletion, activation, and decay data from ENDF/B-VII.1 library [12].

MeshView from SCALE 6.2.4 code package was used for visualization of the mesh tally results and VisIt [13] was used for plotting the ADVANTG results.

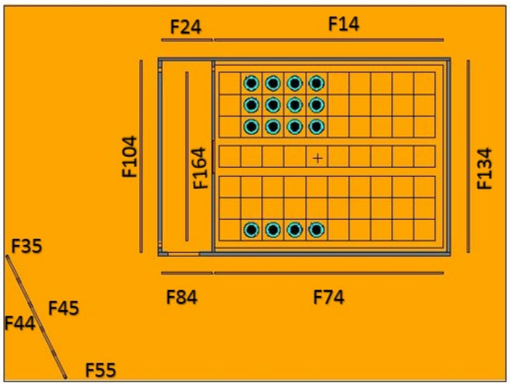
3 ADVANTG/MCNP MODEL OF DSB

3.1 Geometry

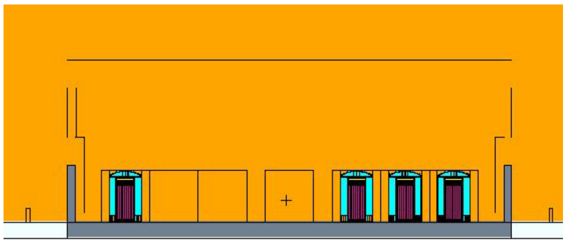
DSB is a reinforced concrete and steel structure with dimensions of 69.8 m × 47.4 m. It has ventilation openings to ensure passive cooling of the overpacks by natural circulation of the air. During the project design phase, an additional shielding was achieved by adding sandwich plates made of carbon steel (CS) and high-density polyethylene (HDPE) on the inner side of the south and west DSB walls. North and east DSB inner walls are covered with 6 mm CS plate. Thicker layers of HDPE are added at the south DSB wall to reduce the dose rates at the site boundary. These were all accredited in the MCNP model and the roof is modelled as a thin steel structure.

As already mentioned, the loading process will be conducted in four campaigns. In this paper, the dose rates after the first campaign containing 16 overpacks have been considered. The MCNP model of DSB containing 16 overpacks with the tally locations is shown in Figure 1. One tally is located in the DSB loading area, six tallies are located close to the external DBS walls and four tallies are located at the site fence. Two types of tallies are used, point and wall (i.e. volume) detectors. Tallies F14 and F24 are located 400 cm north from DBS wall and are wall types of tallies. F14 tallies contributions from storage area and F24 from loading area. Tallies F74 and F84 are of same dimensions as F14 and F24 and are located 400 cm south from DSB wall. Tallies F104 and F134 are also wall types of tallies and are located west and east from DSB wall, respectively. At the fence, there are three point detector tallies and one wall detector tally. All wall detector tallies are 40 cm thick and 150 cm tall consisting of dry air.

Additionally, a mesh tally with 200 × 150 × 90 intervals is used for dose rate visualization over the entire DSB site geometry.



a)



b)

Figure 1: DSB model in MCNP after first loading campaign a) XY view, b) YZ view

3.2 Source

The loading Campaign one consists of 16 HI-STORM overpacks, each containing 37 SFAs. This means that there are 592 SFAs representing independent volumetric sources. Their intensities are accredited in the MCNP model by applying relative weights obtained as the intensity of each SFA divided by the intensity of all 592 SFAs. The total Campaign one neutron, fuel gamma, and hardware gamma source intensities are 7.41×10^{10} neutron/s, 1.34×10^{18} γ/s, and 4.92×10^{15} γ/s, respectively. These source intensities were calculated for each SFA and summed up on the cask-basis. The calculations were performed using ORIGEN-S from the SCALE 6.2.4 code package.

In this work, neutron and gamma sources arising from the SFA are considered, as well as secondary gammas resulting from n- reaction and hardware gammas resulting from SFA hardware activation. For each type of radiation, a separate calculation is performed. Radial source sampling is on assembly-by-assembly basis and axial source sampling is based on burnup dependent source axial distribution. For each overpack, there are three energy spectra for the three spatial regions. More details on the source modelling can be found in our previous work [3].

3.3 Conversion factors and cross sections

ANSI/ANS-6.1.1-1977 [14] Neutron Flux-to-Dose Rate Conversion Factors are used to convert tally results to rem/h/particle. All output and mesh tally results are postprocessed by multiplication with corresponding source intensity and with 10^4 to obtain dose rates in μSv/h. In this analysis, ENDF/B-VII.1 cross section library was used.

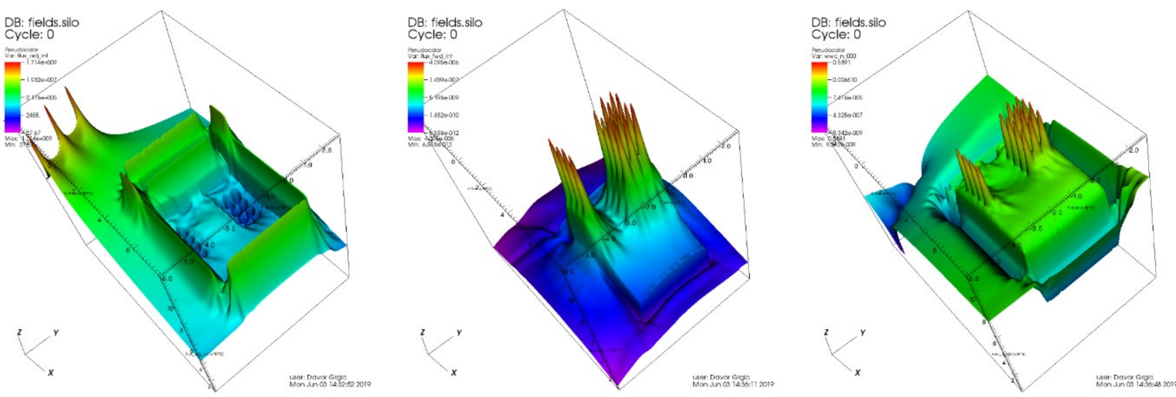
3.4 ADVANTG calculations

Having prepared MCNP input file, ADVANTG calculations can take place. The most important control parameters were: the FW-CADIS method to optimize multiple tallies, the multigroup cross section library ENDF/B-VII.0 47n/20g library, the S_N mesh corresponding to the one used for MCNP mesh tally, the Denovo S_N solver with level symmetric S_4 quadrature and P_1 Legendre expansion order for cross section representation. The maximum number of iterations within group was 100 and the tolerance was set to 10^{-6}.

4 RESULTS

4.1 ADVANTG results

The ADVANTG results in terms of integrated adjoint flux, integrated forward flux, and weight windows are shown in Figure 2 for the highest neutron energy group taken here as an example. Typically for the adjoint flux the source is depressed and tally locations are emphasized, while reverse is true for the forward flux. There is an inverse relationship between adjoint flux and distribution of particle weights, resulting in preferential particle transport (splitting process) towards exterior of the cask in direction of tally objects.



a) b) c)

Figure 2: Results of neutron ADVANTG calculation: a) integrated neutron adjoint flux, b) integrated neutron forward flux, c) neutron weight windows for highest energy group

4.2 MCNP RESULTS

In this section, the most important results of the radiation shielding analysis of DSB loaded with 16 overpacks belonging to Campaign one with a loading date start in 2023 are provided.

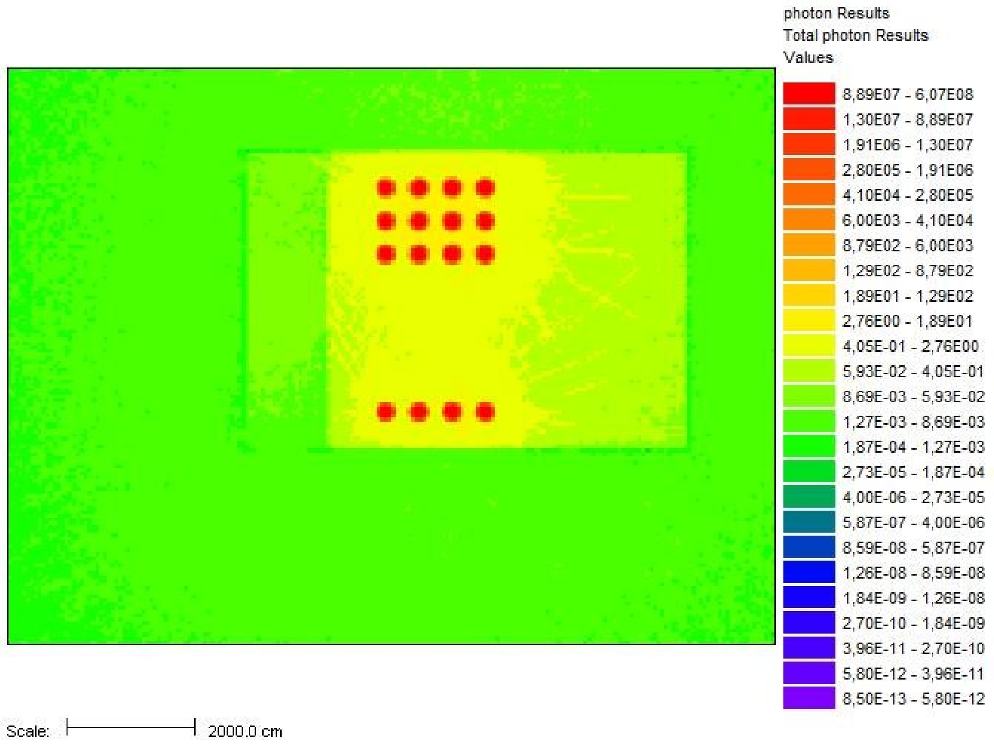
The neutron and gamma dose rates (including fuel gamma, n-γ gamma and hardware gamma) are given in Table 1 for two wall tallies and one site fence tally. Recall, tally locations F14 and F74 are at the north and south DSB wall, respectively, and tall F45 is a point detector tally located at the center of the fence. From these results, it can be seen that the north DSB wall dose rates are higher than at the south. There are two reasons for that. Firstly, from Figure 1 a) it can be seen that 12 overpacks are located at the north of the DSB storage area, while at the south there are 4 overpacks only. Secondly, south DSB wall has additional shielding in form of CS-HDPE sandwich plates which affects more neutron dose rate since the HDPE is a good neutron attenuator. In this table, the total dose rate for each tally location is provided in order to check the regulatory limits. The total dose rate at north and south DSB walls are 6.15×10^{-3} μSv/h and 3.91×10^{-3} μSv/h, respectively, which is significantly lower than the limiting 3 μSv/h. The highest contribution to the total dose rate of 5.61×10^{-4} μSv/h at the fence is from neutrons, followed by fuel gammas, hardware gammas and n-γ gammas. If conservative occupancy time of 8760 hours per year is applied, then the annual dose at the site boundary is 0.011 mSv and the limiting dose is 0.05 mSv.

Table 1: Dose rates at the north and south DSB walls and at the fence

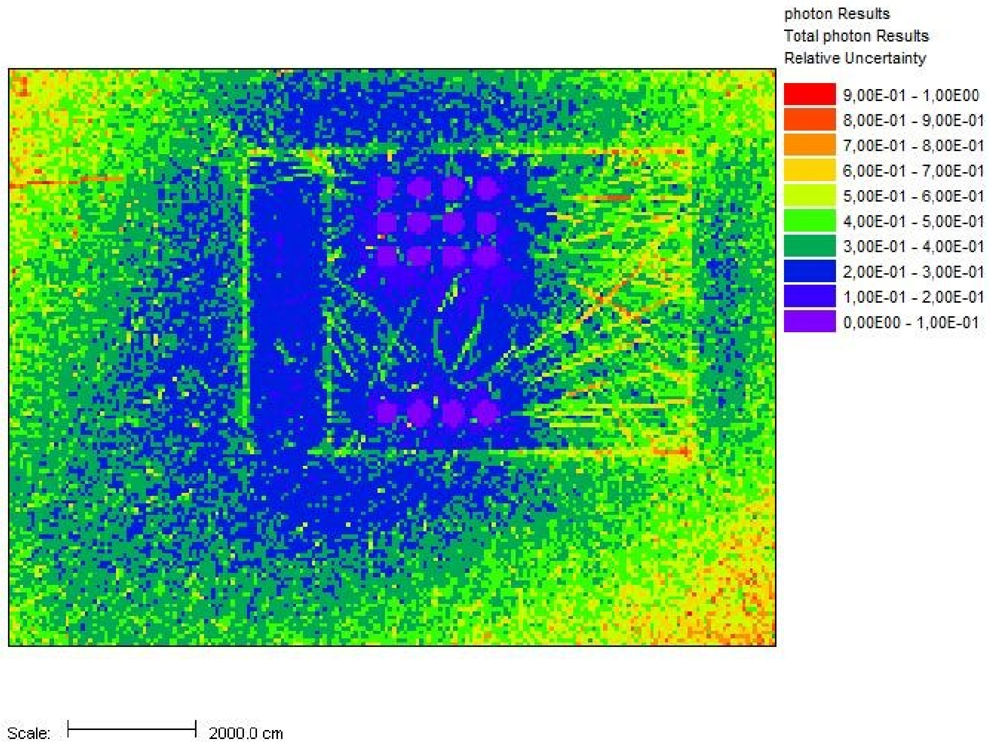
Tally	**Neutron dose rate [μSv/h] (relative uncertainty)**	**Fuel gamma dose rate [μSv/h] (relative uncertainty)**	**n-γ gamma dose rate [μSv/h] (relative uncertainty)**	**Hardware gamma dose rate [μSv/h] (relative uncertainty)**	**Total dose rate [μSv/h] (relative uncertainty)**
F14	3.98×10^{-3} (4.03 %)	1.14×10^{-3} (8.05 %)	4.64×10^{-4} (2.84 %)	5.68×10^{-4} (5.10 %)	6.15×10^{-3} (3.05 %)
F74	2.08×10^{-3} (4.11 %)	9.04×10^{-4} (8.64 %)	4.89×10^{-4} (1.52 %)	4.39×10^{-4} (4.68 %)	3.91×10^{-3} (3.01 %)
F45	5.61×10^{-4} (2.59 %)	3.59×10^{-4} (8.36 %)	9.10×10^{-5} (1.44 %)	1.87E $\times 10^{-4}$ (5.30 %)	1.20×10^{-3} (2.91 %)

To obtain the insight into the dose rates over the entire model, the total dose rate xy distribution at a plane z=143 cm and associated relative uncertainties are presented in Figure 3. Figure 4 shows the total dose rate xz distribution at a plane y=564 cm associated relative uncertainties. The total dose rate distribution along x axis at y=564 cm (center of the middle row in the upper cask section) and along y axis at x=2540 cm (center of the third cask column) are shown in Figure 5.

Furthermore, the comparison of dose rates obtained for the loading start in the year 2023 and 2020 are given in Table 2 for the fence tally. In the reference case (2023), neutron fuel gamma and hardware gamma source intensities were 7.41×10^{10} neutron/s, 1.34×10^{18} γ/s and 4.92×10^{15} γ/s, respectively, whereas for the year 2020, the neutron, fuel gamma and hardware gamma source intensities were 8.05×10^{10} neutron/s, 1.53×10^{18} γ/s, and 2.41×10^{16} γ/s, respectively. Recall, the source intensities were obtained using ORIGEN-S from the SCALE 6.2.4 code package. Three years longer cooling time resulted in dose rate decrease of 17.81 %. The highest relative difference is observed for the hardware gamma dose rates. This is expected since the hardware gamma source intensity for the year 2020 is about 5 times higher than for the year 2023. This is due to short halflife of the Co-60, 5.27 years.

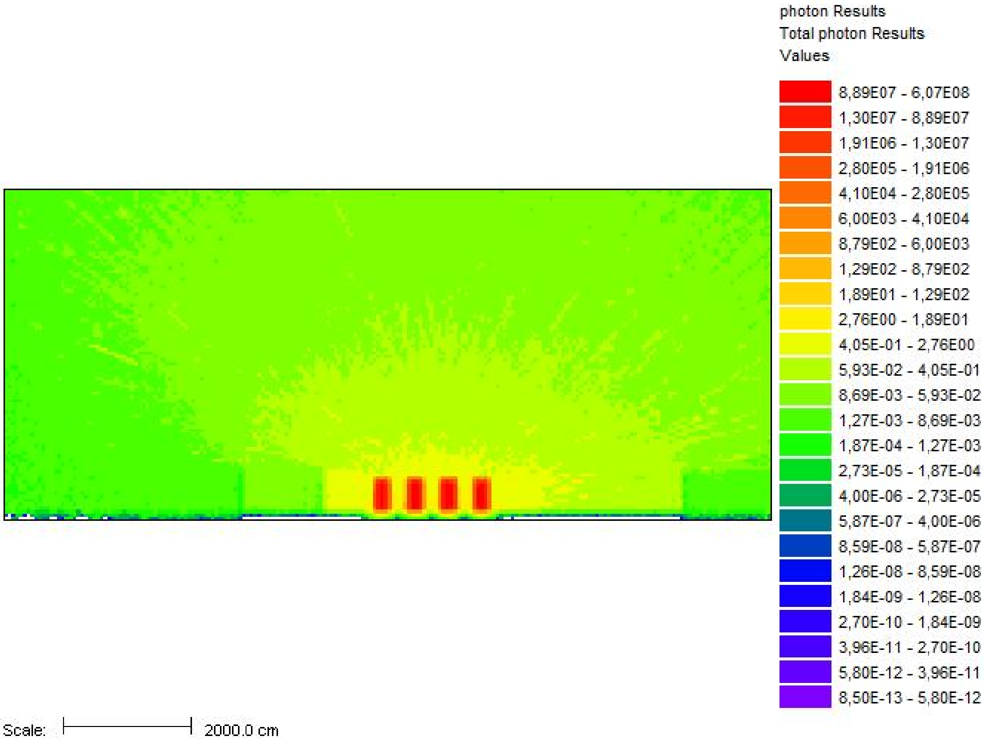


a)



b)

Figure 3: Total dose rate (µSv/h) xy distribution (a) and relative uncertainty (b) at the plane z=143 cm



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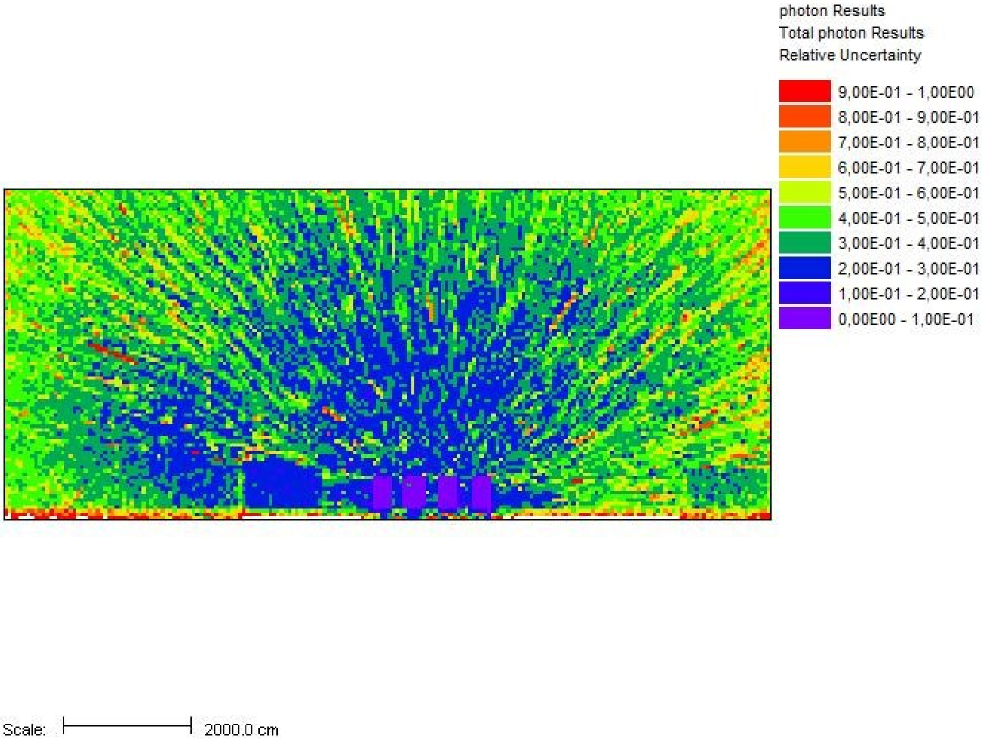
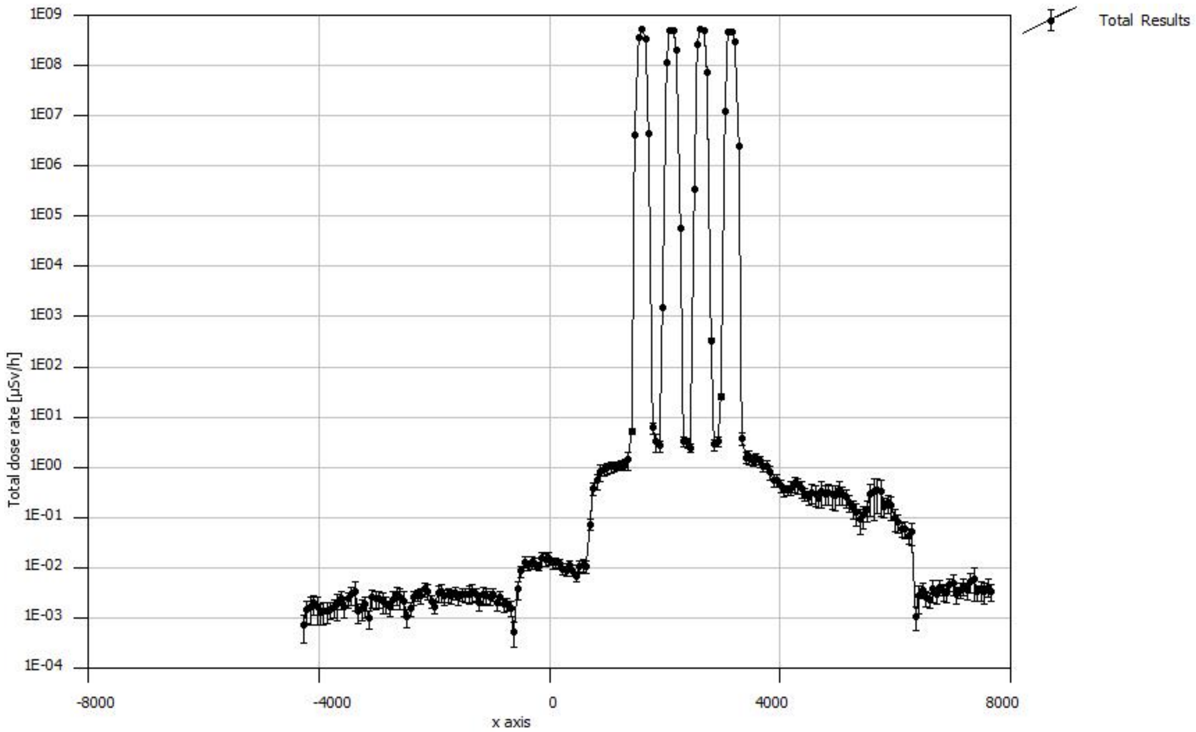
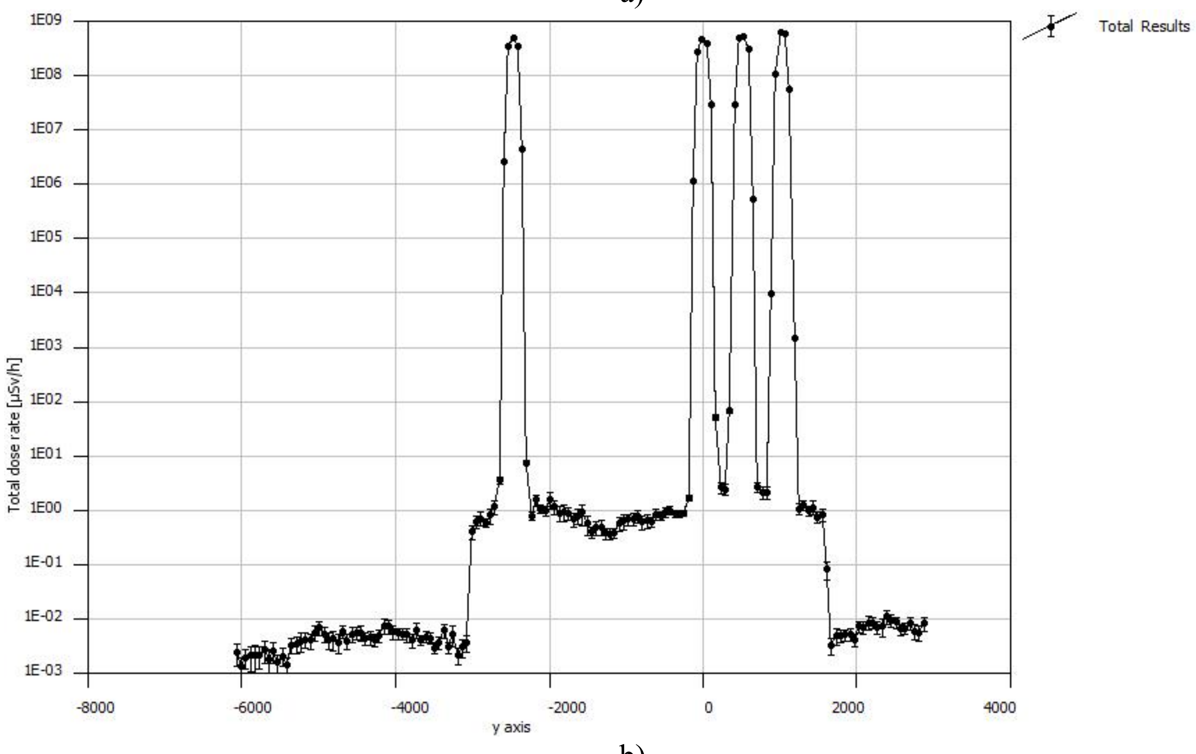


b)

Figure 4: Total dose rate (μSv/h) xz distribution (a) and relative uncertainty (b) at the plane y=564 cm



a)



b)

Figure 5: Total dose rate a) along x axis, b) along y axis

Table 2 Comparison of dose rates for the Campaign one loading date start in 2023 and 2020 at the fence tally F45

Year	2020	2023	Relative difference
Neutron dose rate [μSv/h]	6.22×10^{-4}	5.61×10^{-4}	-9.81 %
Fuel gamma dose rate μSv/h	4.39×10^{-4}	3.59×10^{-4}	-18.22 %
n- γ dose rate [μSv/h]	1.00×10^{-4}	9.10×10^{-5}	-9.00 %
Hardware gamma dose rate [μSv/h]	2.95×10^{-4}	1.87×10^{-4}	-36.61 %
Total dose rate [μSv/h]	1.46×10^{-3}	1.20×10^{-3}	-17.81 %

5 CONCLUSION

In this work, radiation shielding analysis of the DSB loaded with 16 overpacks belonging to the loading Campaign one is presented. This was done in order to check the dose rates limits specific for the NPP Krško site. The calculations were based on the real operating history of the 592 SFAs to be loaded in 16 overpacks. The neutron, fuel gamma and hardware gamma sources were prepared for each SFA using ORIGEN-S from SCALE6.2.4 code package. The dose rate calculations were calculated using hybrid shielding approach involving MCNP6.2 and ADVANTG3.0.3 codes.

The results were presented for the north and south DSB walls and for the fence tally to check the limiting dose rates. DSB wall tallies showed about three orders of magnitude lower dose rates than the limiting, and at the fence the obtained annual dose rate is 78% lower than the limiting.

Furthermore, to obtain the insight into the dose rates over the entire model, the mesh tally results in terms of dose rates and associated relative uncertainties were presented for the total dose rate (sum of neutrons, fuel gammas, n-γ gammas, and hardware gammas).

Finally, the comparison of dose rates for different Campaign one loading start dates showed that three years longer cooling time resulted in dose rate decrease of 17.81 %. The highest relative difference is observed for the hardware gamma dose rates.

Our future work will be focused on the more detailed radiation shielding analysis within DSB and sensitivity calculation needed to identify variables having important influence on dose prediction.

ACKNOWLEDGMENTS

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