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Spent Fuel Pool Dose Rate Calculations Using Point Kernel and Hybrid Deterministic-Stochastic Shielding Methods

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ABSTRACT

This paper presents comparison of the Krško Power Plant simplified Spent Fuel Pool (SFP) dose rates using different computational shielding methodologies. The analysis was performed to estimate limiting gamma dose rates on wall mounted level instrumentation in case of significant loss of cooling water. The SFP was represented with simple homogenized cylinders (point kernel and Monte Carlo (MC)) or cuboids (MC) using uranium, iron, water, and dry-air as a bulk region materials. The pool is divided on the old and new section where the old one has three additional subsections representing fuel assemblies (FAs) with different burnup/cooling time (60 days, 1 year and 5 years). The new section represents the FAs with the cooling time of 10 years. The time dependent fuel assembly isotopic composition was calculated using ORIGEN2 code applied to the depletion of one of the fuel assemblies present in the pool (AC-29). The source used in Microshield calculation is based on imported isotopic activities. The time dependent photon spectrum with total source intensity from Microshield multigroup point kernel calculations was then prepared for two hybrid deterministic-stochastic sequences. One is based on SCALE6.2b3/MAVRIC (Monaco and Denovo) methodology and another uses Monte Carlo code MCNP6.1.1b and ADVANTG3.0.1. code. Even though this model is a fairly simple one, the layers of shielding materials are thick enough to pose a significant shielding problem for MC method without the use of effective variance reduction (VR) technique. For that purpose the ADVANTG code was used to generate VR parameters for the MCNP fixed-source calculation using continuous energy transport. ADVATNG employs a deterministic forward-adjoint transport solver Denovo which implements CADIS/FW-CADIS methodology. Denovo uses a structured, Cartesian-grid SN solver based on the Koch-Baker-Alcouffe parallel transport sweep algorithm across x-y domain blocks. This was our first application of ANDVANTG/MCNP hybrid sequence for this type of calculation and the results where compared to SCALE/MAVRIC sequence which we regularly use for similar calculations. The comparison of gamma dose rates on different point detector locations (central above pool and at the top of pool periphery) showed a good agreement between Microshield (point-kernel) and deterministic-stochastic shielding methodologies for the cylindrical approximation of the pool geometry. More complicated cases for model with multi-source option and for cuboids showed very good agreement between SCALE/MAVRIC and ANDVANTG/MCNP calculations.

Keywords*: pool dose rate, point kernel, hybrid deterministic-stochastic, Microshield, SCALE/MAVRIC, MCNP, ADVANTG, FW-CADIS.*

1 INTRODUCTION

Routine shielding calculations are typically performed using deterministic (SN) or Monte Carlo (MC) algorithms when the objective is to find well converged fluxes on point detectors or inside small localized regions. With a fast computer development the MC method was shifted from expensive and complicated approach to routinely used engineering tool. Modern shielding problems are frequently addressing geometry-large problems of obtaining the global radiation field which is a challenging task for a MC method with standard VR techniques. The hybrid shielding methodology, combining SN and MC approach, was developed especially for that purpose. It is based on forward and adjoint SN transport calculation to generate space-energy based variance reduction (VR) parameters for final (optimized) MC calculation.

This paper presents evaluation of spent fuel pool (SFP) radiation dose rates of the Krško Power Plant using conservative assumptions, simplified geometry and different computational shielding methodologies. Auxiliary structures such as transfer canal and cask loading area for underwater nuclear fuel handling and transport were not modeled at the moment. The SFP was represented with homogenized bodies, such as cylinders and cuboids, preserving original material fractions of uranium, iron, water, and dry-air. Such computational model is a fairly simple one, but effective layers of shielding materials are thick enough to pose a significant shielding problem for the MC method in general, without the use of advanced VR techniques. Comparison of such MC calculations using SN mesh-based VR parameters to more traditional point-kernel calculations was investigated for this simplified SFP.

The general objective was to compare dose rates on point detectors using modern hybrid shielding sequence with more traditional approach using point-kernel method. The specific task was to identify limiting gamma dose rates on wall mounted level instrumentation in case of significant loss of cooling water with presumption that the integrity of the fuel rods in not compromised. The comparison of gamma dose rates on different point detector locations (central above pool and at the top of pool periphery) showed a good agreement between Microshield (point-kernel) and deterministic-stochastic shielding methodologies for the cylindrical approximation of the pool geometry. More complicated cases for a model with multi-source option and for cuboids showed very good agreement between SCALE/MAVRIC and ANDVANTG/MCNP calculations.

This paper is organized as follows. Section 2 gives the description of the modern computational tools SCALE/MAVRIC and ADVANTG/MCNP implementing hybrid shielding methodologies. Section 3 gives the description of the Microshield point-kernel code. Section 4 shows SFP computational model with computational parameters. Section 5 gives shielding analysis of the SFP dose rates with different computational approaches. Section 6 gives discussion and conclusions while the referenced literature is given at the end of the paper.

2 HYBRID SHIELDING COMPUTATIONAL TOOLS

Both hybrid shielding tools SCALE/MAVRIC and ANDVANTG/MCNP use a deterministic forward-adjoint transport solver Denovo as a mean for automatic, adjoint and mesh-based VR generation. With such VR parameters the final (optimized) MC calculation is done. Denovo is implemented with Consistent Adjoint Driven Importance Sampling (CADIS) and Forward Weighted CADIS (FW-CADIS) methodologies, developed at Oak Ridge National Laboratory (ORNL). Denovo is a very fast and robust SN solver, working over a structured Cartesian-grids and based on the Koch-Baker-Alcouffe parallel transport sweep algorithm across x-y domain blocks. The most important feature is a multigroup flux positivity when using Step Characteristic (SC) spatial differencing option, which is paramount factor for numerical stability of the MC calculation.

2.1 The SCALE/MAVRIC shielding sequence

The SCALE6.2b3 code package is the beta version of the latest ORNL's computing software platform developed in support for the U.S. NRC needs. In the present form the code has versatile ability to perform a whole spectrum of different calculations pertinent for nuclear engineering activities in wide areas. Some of the possibilities are: criticality, shielding, radiation source term, burnup/depletion and nuclear decay, reactor physics, and sensitivity/uncertainty analyses using well established analytical sequences. The main shielding sequence is MAVRIC, based on the CADIS and FW-CADIS methods utilizing SN solver Denovo for VR calculation and subsequent accelerated MC Monaco particle transport. When one looks for a solution in a form of multiple point detectors or over millions of spatial mesh cells, it is necessary to use FW-CADIS which demands for extra forward SN run. Such forward solution is used for preparing inversely weighted adjoint source, placed in the region of users interest. For both CADIS and FW-CADIS the particle average weight is inversely related to adjoint flux value throughout phase-space, so locations of high importance (i.e. adjoint flux) will have low-weighted particles and vice versa. This implies that adjoint source location with optimized MC results will represent spatial attractor for the source particles, giving "reasonable" MC results in-between regions. This approach was used in SFP shielding calculations presented in the following chapters.

2.2 The ADVANTG/MCNP sequence

The ADVANTG3.0.1 is an automated tool for generating variance reduction parameters for fixed-source continuous-energy Monte Carlo simulations with MCNP code, based on approximate 3-D multigroup discrete ordinates adjoint transport solutions generated by Denovo. The VR parameters generated by ADVANTG consist of space-energy dependent weight-window bounds (WW) and biased source distributions (SB), which are output in formats that can be directly used with unmodified version of MCNP. ADVANTG has been applied to neutron, photon, and coupled neutron-photon simulations of real-world radiation detection and shielding scenarios. ADVANTG is compatible with all MCNP geometry features and can be used to accelerate cell tallies (F4, F6, F8), surface tallies (F1 and F2), point-detector tallies (F5), and Cartesian mesh tallies (FMESH). ADVANTG implements CADIS/FW-CADIS methods for generating VR parameters which provide a prescription for generating space-energy dependent WW targets and a consistent biased source distribution (SB cards in SDEF and WWINP file). The CADIS method was developed for accelerating individual tallies, whereas FW-CADIS can be applied to multiple tallies and mesh tallies. The MCNP6.1.1b [6] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The MCNP treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-VI) are accounted for. Thermal neutrons are described by both the free gas and $S(\alpha,\beta)$ models. 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 techniques; a flexible tally structure; and an extensive collection of cross-section data. Energy ranges are from 10-11 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. Pointwise cross-section data were used within MCNP. Auxiliary program MAKXSF prepares cross-section libraries with Doppler broadening.

3 POINT-KERNEL COMPUTATIONAL TOOL

MicroShield is a photon/gamma ray shielding and dose assessment program developed by Grove Software [14]. It is interactive, easy to use, and utilizes extensive input error checking.

Integrated tools provide material and source file creation, dose calculation and plotting of the results. The code has capability to model simple three-dimensional geometries, and the most simple situation is when specific source material is within cylindrical body oriented such that the axial length of the cylinder is in the positive y-axis direction, with the center of the cylinder's bottom placed at the origin. The appropriate layers of side shielding with their respective thicknesses can surround the cylinder in the radial and z-axis direction, or planar shields can be placed at cylinder top (y-axis direction).

MicroShield uses a method called Gauss quadrature for point-kernel numerical integration for integration calculations. In this method, the source is separated into a number of kernels determined by the quadrature order. In general, the greater the quadrature order, the more precise results will be output. The default values of quadrature orders used in the integrations are [14]: radial 10, circumferential 10, and axial 20. In this calculation the number of subdivisions is increased two times in each direction without significant change in calculated dose rates. The code calculates dose rates without and with build-up factors. Dose buildup factors depend on photon energy, the mean free path traveled by a photon in the material of consideration, geometry of the source, and geometry of the attenuating medium. Build-up is calculated automatically by the code for a shield material specified by the user (closest to the detector point). All results used for further comparison are results with build-up included.

4 THE SFP COMPUTATIONAL MODEL

The realistic SFP walls are made of concrete (1.83 m thick) with stainless steel liner plates (0.6 cm thick). Normal water level in the pool is at elevation 115 m which is a 0-level on the water level indicator. The FA is standard 16x16 Westinghouse PWR with cross section 19.718x19.718 cm^2. Maximum design enrichment is ≤ 5 w/o in ^{235}U and design licensed burnup is 60 GWD/MTU. The FAs are stored in the racks divided in old section (left) made of stainless steel and new section (right) made of borated high density stainless steel. The old and new sections accommodate 621 and 1073 cells, respectively. The total usable capacity of all racks installed in the SFP is 1694 cells. The detailed information about geometry configuration and other dimensions can be found in reference [13]. The MAVRIC model of the SFP with heterogeneous racks is depicted in Figure 1 while the Figure 2 gives the cross sectional view of the different unit cells in the old and the new section.

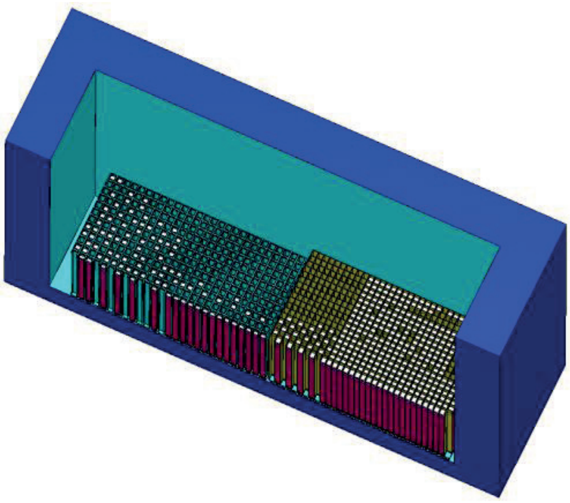

Figure 1: Front view of realistic Krško SFP model with old and new racks (water removed)

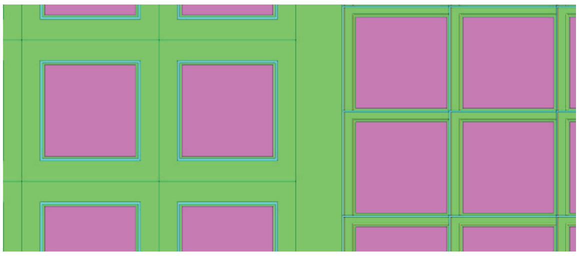

Figure 2: Unit cells of the old (left) and the new (right) section

Preliminary calculations were performed using described realistic model, but because still some important data are missing (mostly related to pool inventory), we have decided to first perform calculations using simplified model of the pool, and compare results against available point-kernel code results.

Simplified SFP computational model uses homogenized regions representing fuel assemblies (source), top shielding layer (homogenized FA nozzles and water), and varying layer of water above it. Locations of interest are at top pool elevation, in the middle of the pool and at pool periphery (potential location of wall mounted level detector). The old section has three additional subsections representing fuel assemblies (FAs) with different burnup/cooling periods of 60 days, 1 year and 5 years. The new section represents the FAs with the average cooling time of 10 years. The time dependent FA isotopic composition was calculated using ORIGEN2 code applied to the depletion of one representative FA present in the SFP (AC-29). The obtained isotopic activities were then used for the source preparation in the point-kernel code Microshield. The source in a form of time dependent, multigroup photon spectrum with known intensity was easy to implement in subsequent hybrid deterministic-stochastic sequences using gamma line spectra with known emission probabilities.

Average density of old SFP racks active part (621 rack locations occupied), with active height of 365.76 cm, was calculated to be 2.5 g/cm^3. This section is composed of water (84.67%), UO_2 (14.83%) and other heavy metals (0.5%). The material selected in the model is uranium. Average density of new SFP racks active part (1073 rack locations occupied) with active height of 365.76 cm was calculated to be 4 g/cm^3. This section is composed of water (73.2%), UO_2 (24.4%) and other heavy metals (8.4%). Above the fuel active height there is an additional 10 cm thick homogeneous layer (top nozzle and water) with an average density of 1.4 g/cm^3 (material selected in model is iron). Depending on calculation scenario there is a layer of water of varying thickness above it (results for 0 and 1 m of waters are presented). Rest of the model is filled with air (density 0.00122 g/cm^3). The bottom cross section of the model is depicted in Figure 3.

Figure 3: Bottom view of the simplified SFP sections: old (S1, S2, S3) and new section

In the middle of the old section there are 64 FAs (8x8) with the cooling time of 60 days (S1). They are surrounded by 80 FAs with the cooling time of 1 year (S2), which are again surrounded by

432 FAs with the cooling time of 5 years (S3). For these three sections the gamma intensities were $2.38\cdot10^{18}$ phot/s, $1.04\cdot10^{18}$ phot/s, $1.91\cdot10^{18}$ phot/s, respectively. This gives total of 576 FAs in total of 621 rack positions. The new section gamma intensity was $2.64\cdot10^{18}$ phot/s. Dose rates were calculated at the center and the edge of the old and new SFP sections at the elevation 1193.76 cm. The point-kernel model uses cylinder source having the same volume as homogenized cuboid. MC runs are performed for the same type of the geometry in order to be able to compare results. Additional MC runs using cuboids are performed to check influence of geometry change. Final multisource MC runs were beyond capabilities of point-kernel code and they are used for comparison between two MC codes. For simple geometry two detector points were used, one at the middle of the section and one at the side of the model, both at the top of the pool (11.93 m above bottom). For MC codes 4 point detectors were used with the following coordinates: new section middle Pd1 (0, 0, 1193.76), new section side Pd2 (0, 355.68, 1193.76), old section middle Pd3 (735.36, 0, 1193.76) and old section side Pd4 (735.36, 384.2, 1193.76). These point detectors represent point adjoint sources in hybrid shielding sequences with spectrum of ANSI-ANS-1977 gamma dose rates (built in function ID=9504 in rem/hr).

5 THE SFP DOSE RATES

Series of shielding calculations were performed for two basic cases: no water above the top FA nozzle and with 1 m of water layer above the top FA nozzle. Separate calculations were performed for each subsection of the old SFP section (S1, S2, S3) and for the new SFP section using cuboidal and cylindrical geometry representation of homogenized regions with the same volumes (i.e. material mass). Obtained MC dose rates with different geometry (cylinders vs. cuboids) and different MC codes showed small difference compared to Microshield. As an example, gamma dose rates for the SFP new section (cylindrical geometry) without and with water layer above the FA nozzle (Figure 4) are shown in Table 1. Gamma dose rates for the smallest old pool section (Figure 5) are shown in Table 3. The differences are reasonable, but it should be mentioned that point-kernel code predictions are not always conservative in terms of calculated gamma dose rate. MC code results for cuboid-based geometry are given in Table 2 and Table 4. The differences in calculated dose rates are small, usually giving lower doses in the middle and higher in the periphery of the model compared to cylinder case.

Table 1 SFP dose rates (rem/h) in the new section

Cylindrical geometry	**ADVANTG/MCNP**	**SCALE/MAVRIC**	**Microshield**
	0 m water layer	0 m water layer	0 m water layer
Pd1	$1.14\cdot10^3 \pm 0.15\%$	$1.17\cdot10^3 \pm 0.08\%$	$1.39\cdot10^3$
Pd2	$8.48\cdot10^2 \pm 0.19\%$	$8.68\cdot10^2 \pm 0.10\%$	$7.81\cdot10^2$
	1 m water layer	1 m water layer	1 m water layer
Pd1	$3.22 \pm 0.19\%$	$3.72 \pm 0.24\%$	5.99
Pd2	$2.07 \pm 0.25\%$	$2.38 \pm 0.27\%$	2.56

Table 2 SFP dose rates (rem/h) in the new section (MC geometry)

Cylindrical geometry	**SCALE/MAVRIC**	**SCALE/MAVRIC**
	0 m water layer	1 m water layer
Pd1	$1.17\cdot10^3 \pm 0.08\%$	$3.72 \pm 0.24\%$
Pd2	$8.68\cdot10^2 \pm 0.10\%$	$2.38 \pm 0.27\%$
Cubical geometry	0 m water layer	1 m water layer
Pd1	$1.15\cdot10^3 \pm 0.08\%$	$3.63 \pm 0.22\%$
Pd2	$9.17\cdot10^2 \pm 0.09\%$	$2.59 \pm 0.27\%$

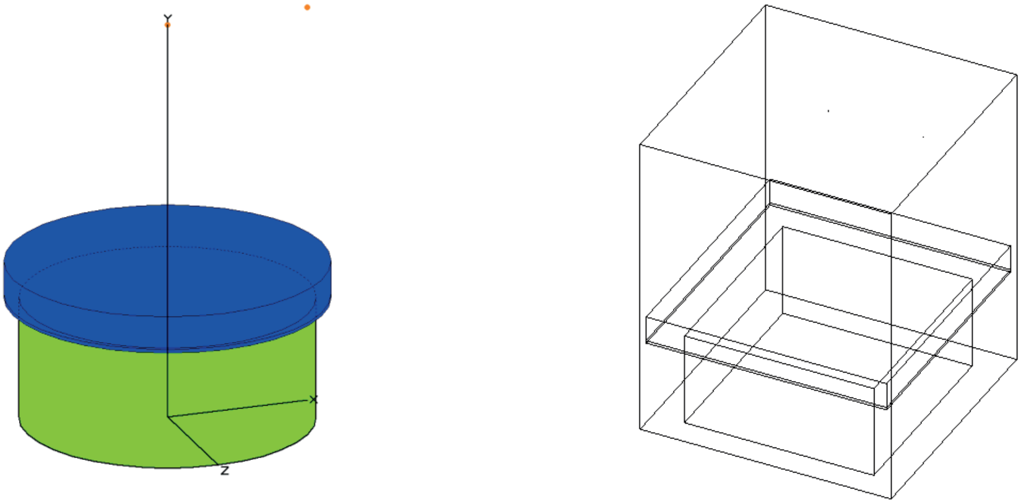

Figure 4: Microshield (r_{eq} = 4.25 m) and SCALE geometries for the new SFP section

Table 3 SFP dose rates (rem/h) in the inner part of the old section

Cylindrical geometry	**ADVANTG/MCNP**	**SCALE/MAVRIC**	**Microshield**
	0 m water layer	0 m water layer	0 m water layer
Pd1	$3.14 \cdot 10^3 \pm 0.22\%$	$3.15 \cdot 10^3 \pm 0.08\%$	$3.11 \cdot 10^3$
Pd2	$2.61 \cdot 10^3 \pm 0.18\%$	$2.63 \cdot 10^3 \pm 0.09\%$	$2.79 \cdot 10^3$
	1 m water layer	1 m water layer	1 m water layer
Pd1	$1.53 \cdot 10^1 \pm 0.19\%$	$1.75 \cdot 10^1 \pm 0.15\%$	$1.66 \cdot 10^1$
Pd2	$8.80 \pm 0.19\%$	$1.02 \cdot 10^1 \pm 0.34\%$	8.48

Table 4 SFP dose rates (rem/h) in the inner part of the old section (MC geometry)

Cylindrical geometry	**SCALE/MAVRIC**	**SCALE/MAVRIC**
	0 m water layer	1 m water layer
Pd1	$3.15 \cdot 10^3 \pm 0.08\%$	$1.75 \cdot 10^1 \pm 0.15\%$
Pd2	$2.63 \cdot 10^3 \pm 0.09\%$	$1.02 \cdot 10^1 \pm 0.34\%$
Cubical geometry	0 m water layer	1 m water layer
Pd1	$2.72 \cdot 10^3 \pm 0.08\%$	$1.49 \cdot 10^1 \pm 0.23\%$
Pd2	$2.63 \cdot 10^3 \pm 0.08\%$	$1.40 \cdot 10^1 \pm 0.25\%$

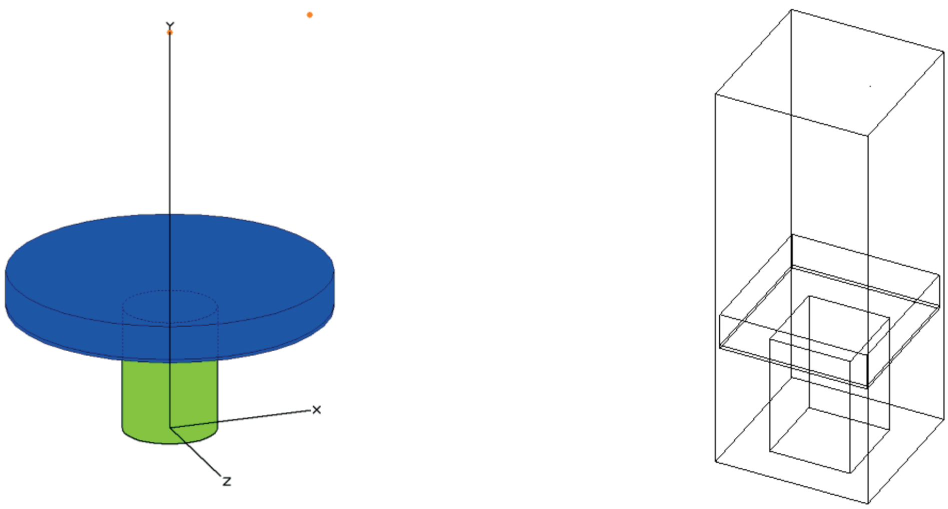

Figure 5: Microshield (r_{eq} = 1.36 m) and SCALE geometries for the inner part of the old SFP section

The final, multisource model (Figure 6 and Figure 7) was prepared implementing all SFP sections and corresponding gamma sources. The obtained gamma dose rates are shown in Table 5 and Table 6. The difference in calculated dose rates is generally small. For the case with 1 m water layer SCALE predictions are always higher than MCNP predictions. For the case without water MCNP calculates higher dose rates for new section (old fuel) and SCALE for old section of the pool (new fuel). As expected, overall difference is larger when water layer is present.

The distribution of gamma dose rates and relative uncertainties in the MCNP model are shown in Figure 8 and Figure 9.

Table 5 SFP dose rates (rem/h) with ADVANTG/MCNP using multisource

Cuboidal geometry	**SFP total dose rates (rem/h) with rel. error**	
	0 m	1 m
Pd1	$5.81 \cdot 10^3 \pm 0.31\%$	$15.28 \pm 0.55\%$
Pd2	$4.52 \cdot 10^3 \pm 0.30\%$	$9.66 \pm 0.81\%$
Pd3	$2.92 \cdot 10^3 \pm 0.33\%$	$4.55 \pm 0.52\%$
Pd4	$2.36 \cdot 10^3 \pm 0.30\%$	$3.28 \pm 0.51\%$

Table 6 SFP dose rates (rem/h) with SCALE/MAVRIC using multisource

Cuboidal geometry	**SFP total dose rates (rem/h) with rel. error**	
	0 m	1 m
Pd1	$5.50 \cdot 10^3 \pm 0.07\%$	$16.35 \pm 0.08\%$
Pd2	$4.28 \cdot 10^3 \pm 0.04\%$	$10.33 \pm 0.08\%$
Pd3	$3.06 \cdot 10^3 \pm 0.06\%$	$5.65 \pm 0.16\%$
Pd4	$2.46 \cdot 10^3 \pm 0.05\%$	$4.06 \pm 0.13\%$

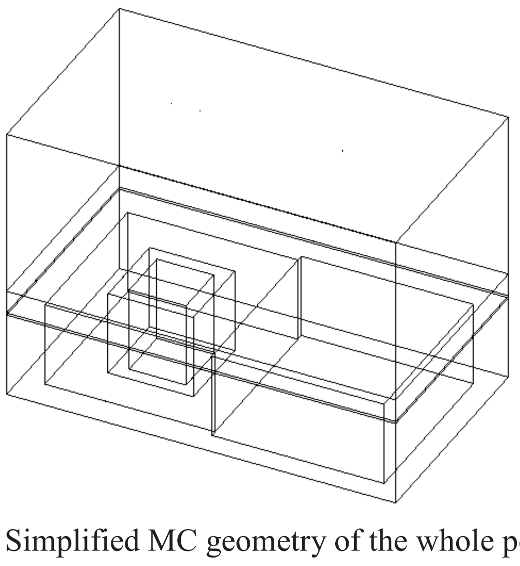

Figure 6: Simplified MC geometry of the whole pool

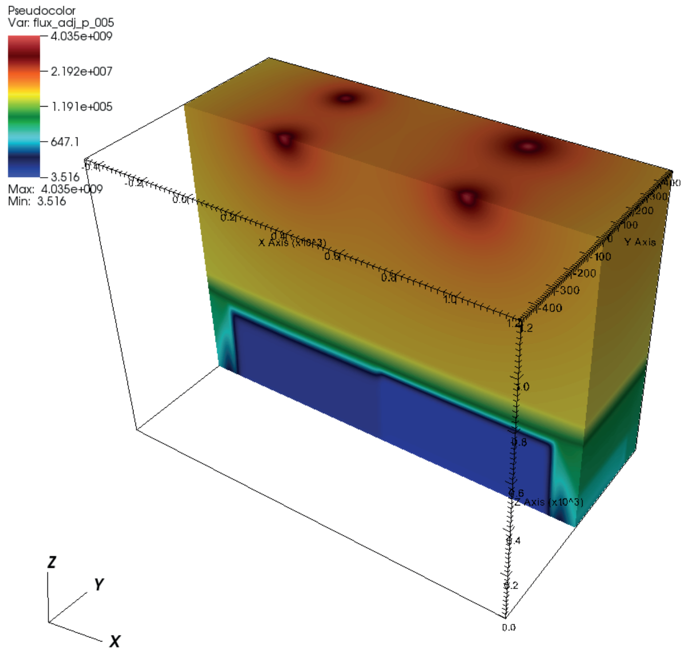

Figure 7: Point detectors as adjoint sources in hybrid shielding methodology using ADVANTG

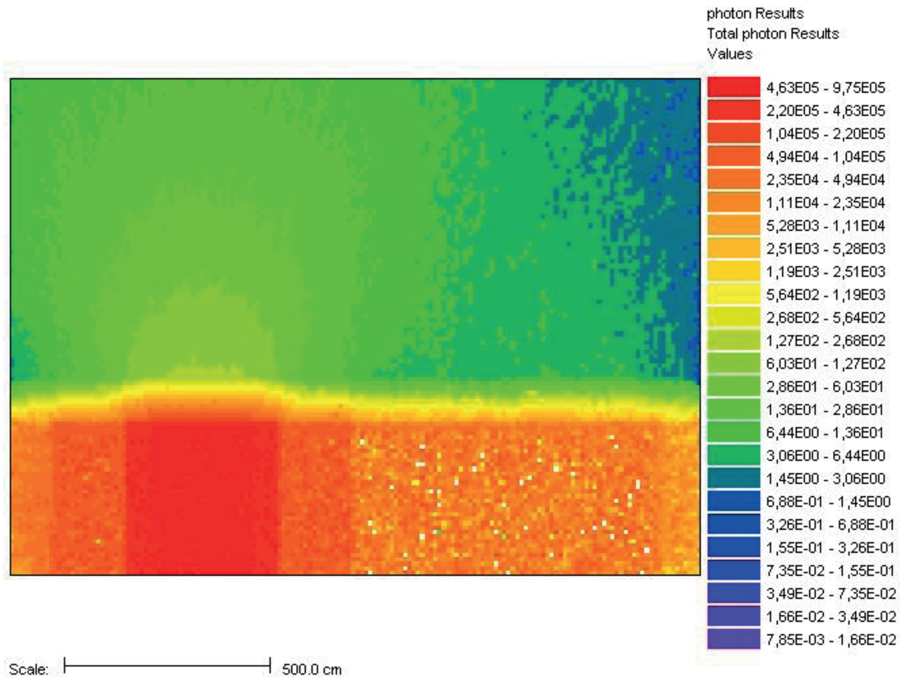

Figure 8: Mesh tally of gamma dose rates (rem/hr) calculated with MCNP6.1.1b

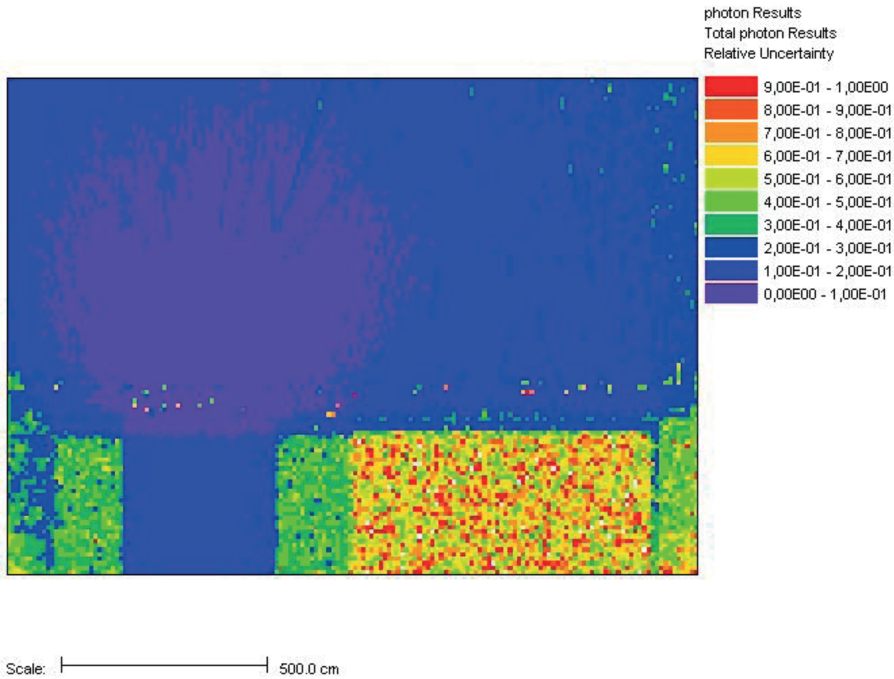

Figure 9: Mesh tally of gamma dose rates relative errors calculated with MCNP6.1.1b

6 DISCUSSION AND CONCLUSIONS

The shielding model of the simplified Krško Power Plant SFP was prepared based on homogenized regions representing old and new FA racks. Every section of the SFP was modeled with different gamma source spectrum based on ORIGEN2 depletion calculations of representative FA AC-29. The hybrid shielding sequences (SCALE/MAVRIC and ADVANTG/MCNP) were used for point detector calculations at center and edge of pool sections at elevation 1193.76 cm. Obtained gamma dose rates were compared to a more traditional approach using point-kernel code Microshield. A good agreement between point-kernel and MC calculations is obtained for cylindrical mono source cases, where both type of the codes should perform similarly. Still, it should be mentioned that point-kernel code predictions in terms of calculated does rates are not always conservative. Different geometrical representation with cylinders or cuboids, gives similar results (slightly lower in the middle and slightly higher at the cuboid periphery compared to cylinder) if equivalence is performed properly. Both MC calculation sequences calculated similar dose rates for multisource whole SFP model. As expected the difference is larger when water layer is present. Monte Carlo codes with implemented VR are reasonable alternative to engineering point-kernel calculation if the geometry is starting to be more complicated.

REFERENCES

[1] "SCALE: A comprehensive Modeling and Simulation Suite for Nuclear Safety and Design", ORNL/TM-2005/39, Version 6.1, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.

[2] G. I. Bell, S. Glasstone, "Nuclear Reactor Theory", Van Nostrand Reinhold Company, New York, 1970.

[3] E. E. Lewis, W. F. Jr. Miller, "Computational Methods of Neutron Transport", American Nuclear Society, Illinois, 1993.

[4] J. C. Wagner, A. Haghighat, "Automated Variance Reduction of Monte Carlo Shielding Calculations Using the Discrete Ordinates Adjoint Function", Nuclear Science and Engineering, 128, 1998, pp. 186-208.

[5] T. M. Evans, A. S. Stafford, R. N. Slaybaugh, K. T. Clarno, "Denovo: A New Three-Dimensional Parallel Discrete Ordinates Code in SCALE", Nuclear Technology, 171, 2010, pp. 171-200.

[6] M. Matijević, D. Pevec, K. Trontl, "Modeling of the ORNL PCA benchmark using SCALE6.0 hybrid deterministic-stochastic methodology", Science and Technology of Nuclear Installations, 2013, DOI:10.1155/2013/252140.

[7] J. C. Wagner, E. D. Blakeman, D. E. Peplow, "Forward-Weighted CADIS Method for Global Variance Reduction", Transactions of American Nuclear Society, 97, 2007, pp. 630-633.

[8] J. C. Wagner, D. E. Peplow, S. W. Mosher, "FW-CADIS Method for Global and Regional Variance Reduction of Monte Carlo Radiation Transport Calculations", Nuclear Science and Engineering, 176, 2014, pp. 37-57.

[9] J. C. Wagner, D. E. Peplow, S. W. Mosher, T. M. Evans, "Review of Hybrid (Deterministic/Monte Carlo) Radiation Transport Methods, Codes, and Applications at Oak Ridge National Laboratory", Progress in Nuclear Science and Technology, 2, 2011, pp. 808-814.

[10] M. Matijević, D. Pevec, K. Trontl, "Dose rates modeling of pressurized water reactor primary loop components with SCALE6.0", Nuclear Engineering and Design, 283, 2015, pp. 175-192, DOI: 10.1016/j.nucengdes.2014.07.013.

[11] M. B. Chadwick, et al, "ENDF/B-VII.0: next generation evaluated nuclear data library for nuclear science and technology", Nuclear Data Sheets 107(12), 2006, pp. 2931-3060.

[12] "VisIt: An End-User Tool For Visualizing and Analyzing Very Large Data", Lawrence Livermore National Laboratory, VisIt Getting Started Manual, February 2003, UCRL-MA-148506-REV-1, Version 2.7.3.

[13] "SFP DOSE RATE EVALUATION", NEK Report number NEK ESD-TR-11/11, Rev.0.

[14] MicroShield User's Manual, Version 10, Grove Software, 2014.