Radiological Analysis of Spent Fuel Transportation Container

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Abstract. The transportation of Kozloduy NPP's WWER-1000 spent nuclear fuel requires design and validation of a special transport equipment. The exploitation of such equipment requires the fulfillment of IAEA Safety Standards Series Safety Guide No. NS-G-1.4 recommendations and compliance with the Bulgarian legislation, which states that the maximal dose rate at the external cask surface shall not exceed 2 mSv/h at any point. The cask TK-13/3 together with the internal basket 37/3 are used for the transportation of spent nuclear fuel from the spent fuel pool in the reactor building to the spent fuel storage facility. The dose rate distribution around the TK-13/3 cask is calculated with the widely used and validated for similar analyses software system SCALE. The analysis is done using the latest version of SCALE – 6.2.3 with the application of modules ORIGAMI → ORIGEN → MAVRIC. The results show that all normative requirements and recommendations are met.

Keywords: dose rate, spent fuel, SCALE 6.2.3, radiological analysis.

1 Introduction

The aim of the present paper is to show the radiological analysis of the on-site spent fuel transportation for WWER-1000 reactor.

The results for the dose rate distribution around the transport container are obtained by calculations for spent fuel, which reached a burnup of 55 MWd/kgU. The time needed for a fuel assembly with such burn-up to decay up to 1.67 kW is 5 years. This average decay heat level allows 12 spent fuel assemblies to be transported together in one container. In addition, an analysis for a 10-year post-irradiation time in the reactor spent fuel pool after the removal from the reactor core is carried out.

The transportation of Kozloduy NPP's WWER-1000 spent nuclear fuel requires design and validation of special transport equipment. The exploitation of such equipment requires fulfillment of IAEA Safety Standards Series Safety Guide No. NS-G-1.4 [1] recommendations:

- Irradiated fuel should be transported in shielded and adequately cooled casks that are either internally dry or partially filled with coolant;
- The casks should have an internal structure to keep the fuel in a well-defined geometric arrangement during transport;
- The fuel may first be placed in a basket which may then be loaded into the cask;
- The casks should meet the applicable requirements;
- In accordance with Bulgarian legislation, the maximal dose rate at the external cask surface shall not exceed 2 mSv/h at any point. The cask TK-13/3 together with the internal basket 37/3 are used for the transportation of spent nuclear fuel from the spent fuel pool in the reactor building to the spent fuel storage facility.

2 Dose Rate Calculations

The dose rate distribution around the TK-13/3 cask is calculated with the widely used and validated for similar analyses software system SCALE – Standardized Computer Analyses for Licensing Evaluation. The analysis is done using the latest version of SCALE – 6.2.3 [2] with application of modules ORIGAMI [3] → ORIGEN [4] → MAVRIC [5], which interact automatically in one run.

ORIGAMI [3] – computes detailed isotopic compositions for light water reactor assemblies containing UO_2 fuel by using the ORIGEN transmutation code with pre-generated ORIGEN libraries, for a specified assembly power distribution. ORIGAMI performs ORIGEN burnup calculations for each of the specified power regions to obtain the spatial distribution of isotopes in the burned fuel.

Multiple cycles with actual burn-times and downtimes are used. ORIGAMI produces stacked ORIGEN binary output data ("ft71 file") for each depletion zone; files with nuclide concentrations at the last time-step for each axial depletion region, a file containing the axial decay heat at the final time-step; and gamma and neutron radiation source spectra.

ORIGEN [4] – calculates time-dependent concentrations, activities, and radiation source terms for a large number of isotopes simultaneously generated or depleted by neutron transmutation, fission, and radioactive decay. A standard decay library is used to perform decay calculations.

ORIGEN saves the results (isotopics and source spectra) in a special ORIGEN binary concentrations file (f71).

MAVRIC [5] – the sequence is used to calculate fluxes and dose rates with low uncertainties. MAVRIC is based on the CADIS (Consistent Adjoint Driven Importance Sampling) and FW-CADIS (Forward-Weighted CADIS) methodologies, which use an importance map and biased source that are derived to work together. MAVRIC uses XSProc to generate cross sections for the defined materials. Using these multigroup cross sections and a user-specified spatial mesh, MAVRIC performs a three-dimensional, discrete-ordinates calculation using Denovo to determine the adjoint flux as a function of position and energy. This adjoint flux information is then used by MAVRIC to construct a space and energy-dependent importance map to be used for biasing during particle transport and a mesh-based biased source distribution. MAVRIC then passes the importance map and biased source distribution to the functional module Monaco, a three-dimensional, fixed-source (multigroup in this case Monte Carlo radiation transport code. In addition to materials input, the geometry description is done with the SCALE General Geometry Package; source description as a function of position, energy, and direction; tally descriptions (fluxes over mesh grids); response functions; planes for the mesh used by Denovo; and tallies to use as the Denovo adjoint source. Output consists of files for 3D mesh tallies for neutrons and photons.

Flux-to-Dose conversion factors for effective dose from ICRU-57 [6] are used.

3 Fuel and Container Models for SCALE

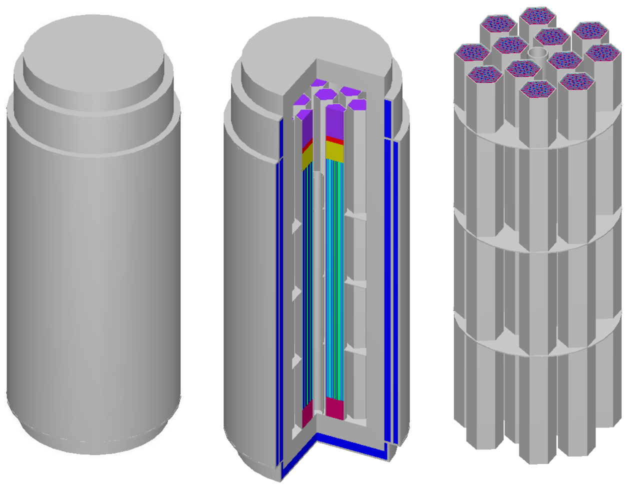
The burn-up calculation is done for fuel which is irradiated over the course of 4 years and has a burn-up of 55 MWd/kgU. The burn-time duration is 325 effective days followed by 35 days downtime.

The average enrichment of ^{235}U is 4.62%. The dependence of uranium isotope content on ^{235}U enrichment is given in Table 1.

Table 1. Dependence of uranium isotope content on ^{235}U enrichment (X stands for ^{235}U enrichment)

Isotop	Isotope wt%
^{234}U	0.0089 X
^{235}U	1.0000 X
^{236}U	0.0046 X
^{238}U	100 – 1.0135 X

The container TK-13/3 [6] is a thick metal cylindrical vessel with a welded bottom and hermetically closed metal lid. The container parts are manufactured from corrosion-resistant steel 08X18H10T or 12X18H10T. A neutron shielding is installed on the external side of the container. It is made of volumes filled with antifreeze solution or water. They provide the neutron shielding around and below the container. The lid of the container is metal only. Figures 1(a) and 1(b) show a 3D model of the container.

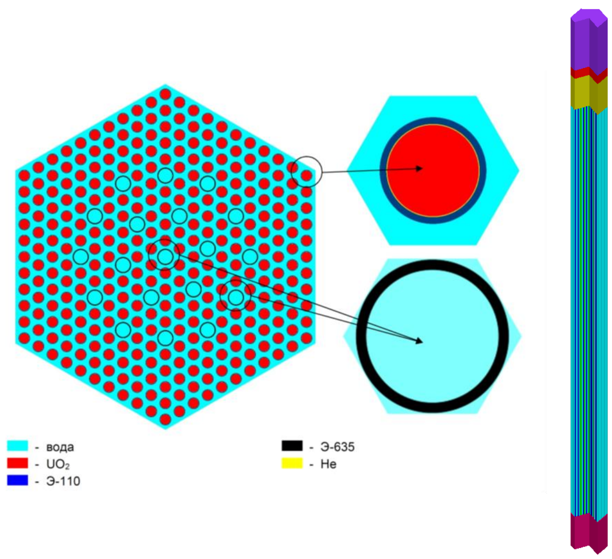


(a) (b) (c)

Figure 1. (a) container general view; (b) container section cut; (c) basket general view.

The basket 37/3 [7] is used together with the container TK-13/3 [7] for the transportation of the spent fuel assemblies from the reactor spent fuel pool to the spent fuel storage facility. The basket is composed of several welded metal elements – base plate, several grids, central guide tube and 12 hexagonal pipes to accommodate WWER-1000 spent fuel assemblies. Figure 1(c) shows the 3D model of the basket.

In the model of the fuel assembly the top and bottom nozzles are homogenized. Figures 2 (a) and 2 (b) show TVSA-12 fuel assembly together with the fuel cell and guide tube and the TVSA-12 3D axial view.



(a) (b)

Figure 2. (a) TVSA-12 fuel assembly together with the fuel cell and the guide tube; (b) TVSA-12 axial view.

4 Results

The results of the calculation of the dose rates around the container loaded with fuel assemblies 5 years after the removal from the reactor core are summarized in Table 2.

Table 2.

Source	1 m above the container	2 m above the container	1 m away from the container	2 m away from the container
neutrons	$\sim$ 40 μSv/h	$\sim$ 12 μSv/h	$\sim$ 5 μSv/h	$\sim$ 2 μSv/h
photons	$\sim$ 15 μSv/h	$\sim$ 2 μSv/h	$\sim$ 25 μSv/h	$\sim$ 15 μSv/h
neutrons + photons	$\sim$ 40 μSv/h	$\sim$ 10 μSv/h	$\sim$ 30 μSv/h	$\sim$ 22 μSv/h

Table 3.

	source	max value
on-lid surface	neutrons	181.022 μSv/h
on side surface	photons	147.423 μSv/h
on-lid surface	neutrons+photons	187.271 μSv/h

Table 3 shows the maximal values of the dose rates on the surface of the container loaded with fuel assemblies 5 years after the removal from the reactor core.

The result of the calculation of the dose rates around the container loaded with fuel assemblies 10 years after the removal from the reactor core are summarized in Table 4.

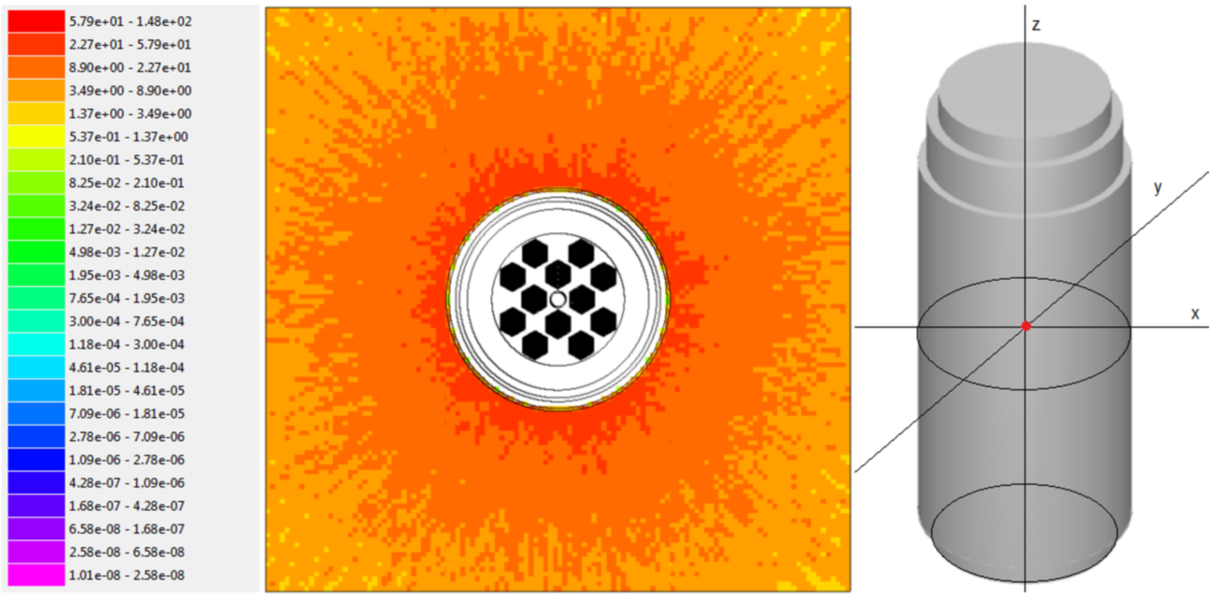
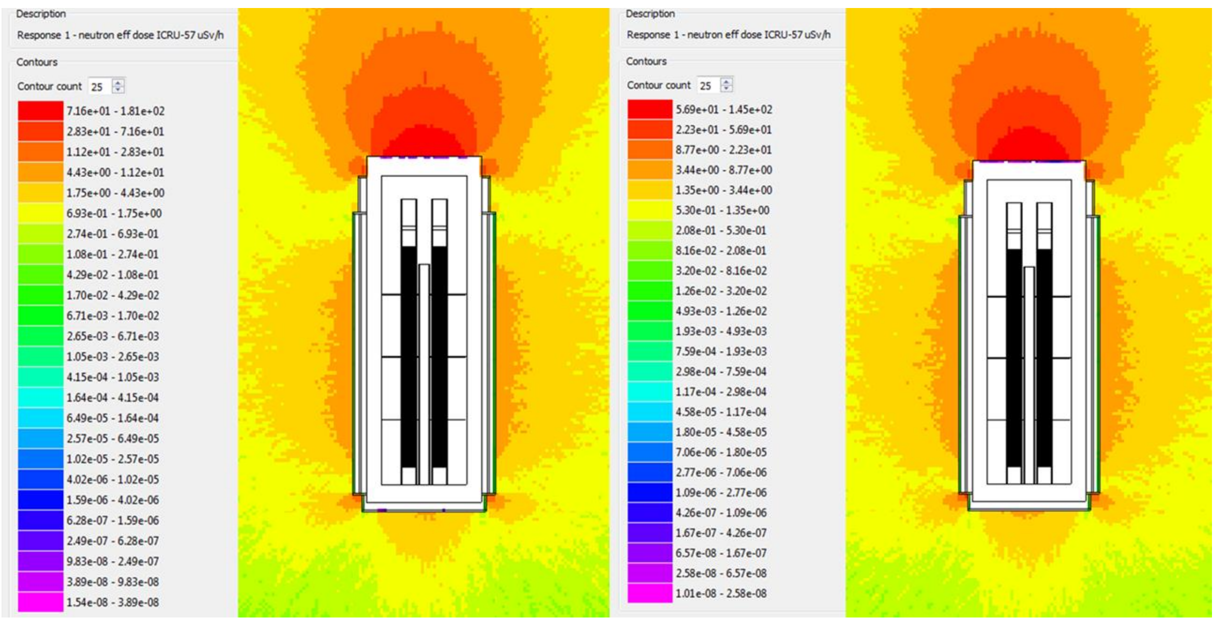


Figure 3. Total (neutron+photons) dose rate distribution along horizontal plane X-Y (Z=0); 5 years decay.



(a) (b)

Figure 4. Neutron dose rate distribution along vertical plane X-Z (Y=0): (a) 5 years decay; (b) 10 years decay.

Table 4.

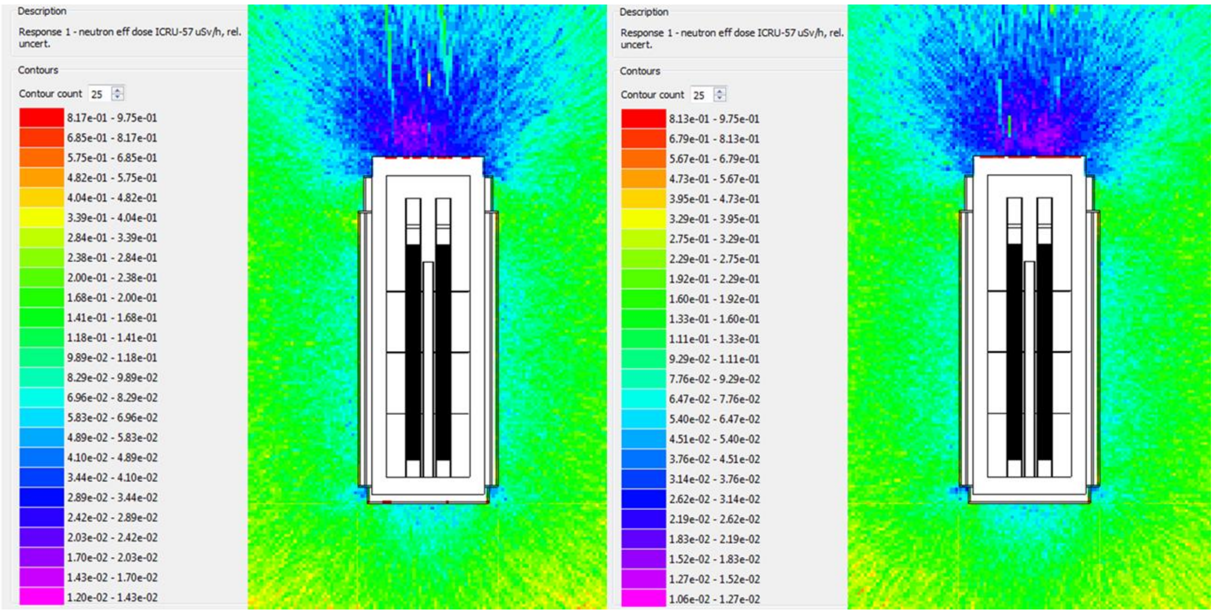
Source	1 m above the container	2 m above the container	1 m away from the container	2 m away from the container
neutrons	$\sim$ 30 μSv/h	$\sim$ 10 μSv/h	$\sim$ 2 μSv/h	$\sim$ 1 μSv/h
photons	$\sim$ 1 μSv/h	$\sim$ 0.5 μSv/h	$\sim$ 14 μSv/h	$\sim$ 5 μSv/h
neutrons + photons	$\sim$ 30 μSv/h	$\sim$ 10 μSv/h	$\sim$ 16 μSv/h	$\sim$ 5 μSv/h

Table 5 shows the maximal values of surface dose rates of the container loaded with fuel assemblies 10 years after the removal from the reactor core.

Figures 3–9 show the dose rate distributions and their relative uncertainties.

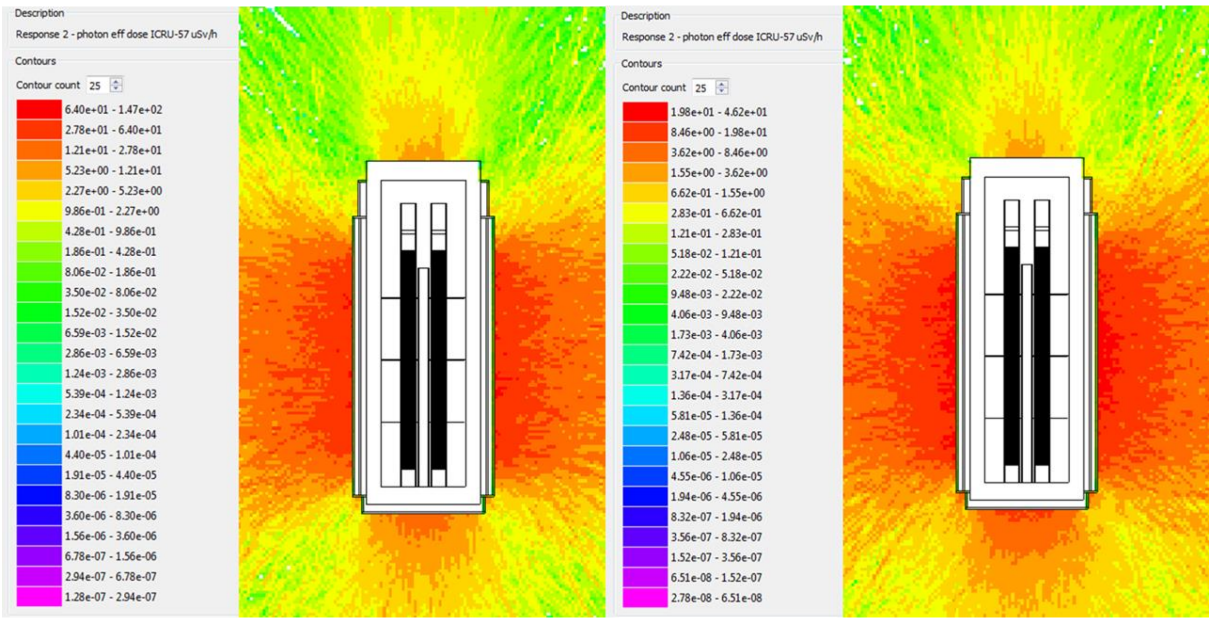
Table 5.

	source	max value
on-lid surface	neutrons	145.088 μSv/h
on side surface	photons	146.226 μSv/h
on-lid surface	neutrons+photons	147.665 μSv/h



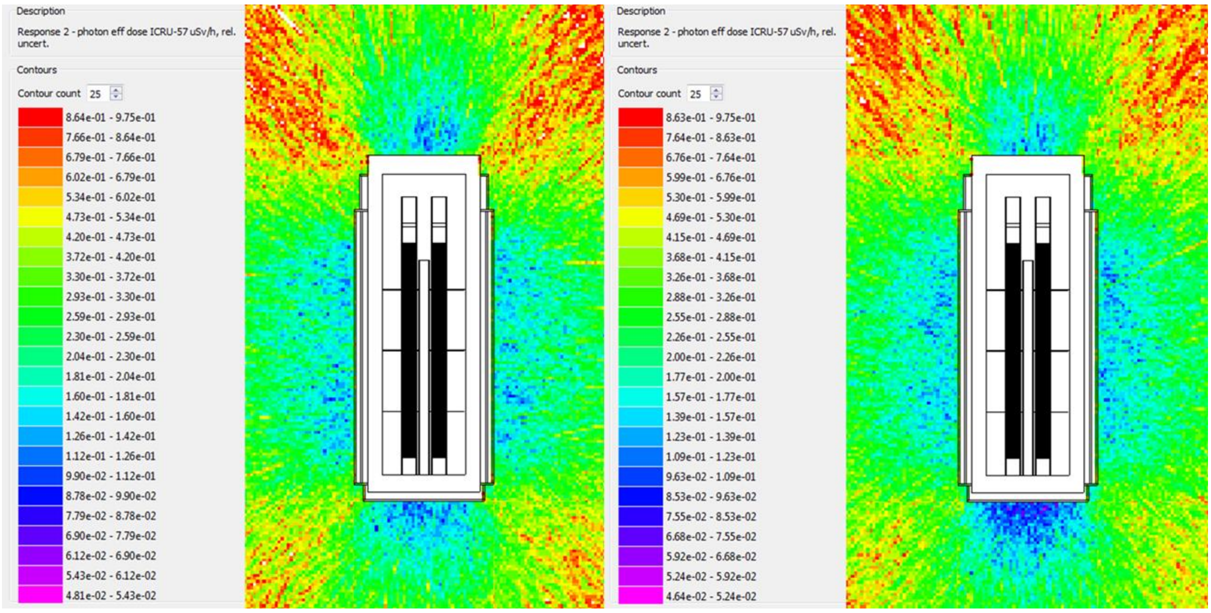
(a) (b)

Figure 5. Relative uncertainty of the neutron dose rate along vertical plane X-Z (Y=0): (a) 5 years decay; (b) 10 years decay.



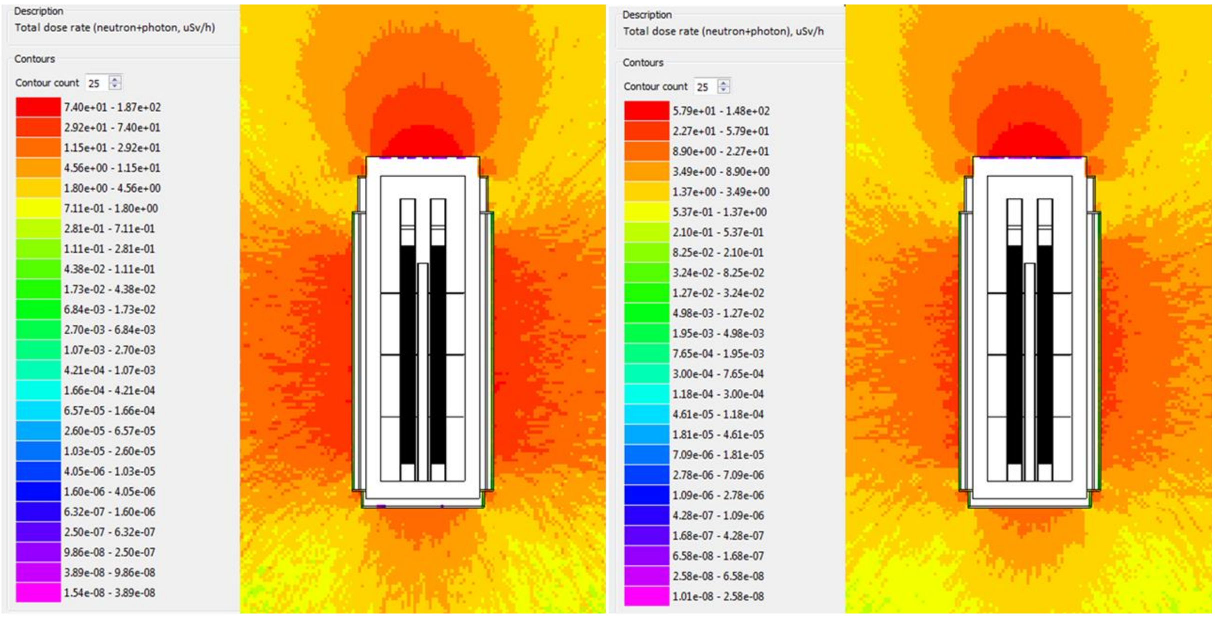
(a) (b)

Figure 6. Photon dose rate distribution along vertical plane X-Z (Y=0): (a) 5 years decay; (b) 10 years decay.



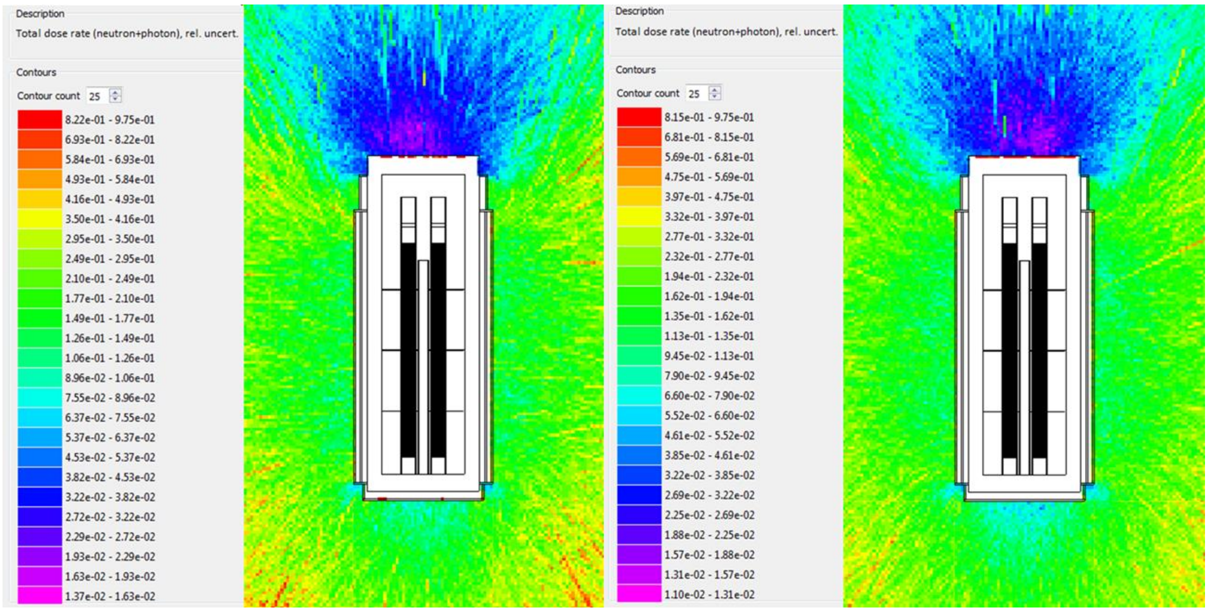
(a) (b)

Figure 7. Relative uncertainty of the photon dose rate along vertical plane X-Z (Y=0): (a) 5 years decay; (b) 10 years decay.



(a) (b)

Figure 8. Total (neutron+photons) dose rate distribution along vertical plane X-Z (Y=0): (a) 5 years decay; (b) 10 years decay.



(a) (b)

Figure 9. Relative uncertainty of the total (neutron+photons) dose rate along vertical plane X-Z (Y=0): (a) 5 years decay; (b) 10 years decay.

5 Conclusion

The results of the calculations show that the used equipment for an on-site transportation provides safe possibilities for operation with spent fuel assemblies according to the present safety requirements.

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