OPTIMIZATION OF THE OPERATION PROCESS AT THE DIFFUSION LENGTH AND FERMI AGE MEASUREMENT FACILITY

ŠTEFAN ČERBA, BRANISLAV VRBAN, JAKUB LÜLEY, FILIP OSUSKÝ, DOROTA FLAMÍKOVÁ and VLADIMIR NEČAS
Slovak University of Technology in Bratislava, Faculty of Electrical Engineering and Information Technology, Institute of Nuclear and Physical Engineering, Bratislava, Slovakia
e-mail: stefan.cerba@stuba.sk

Abstract

After a detailed analysis of events at the Fukushima Daiichi nuclear power plant, it was found out that, besides the reactors, the spent fuel pool storage had significantly contributed to the consequences of the accident. A fundamental requirement for safety is that such a system must remain subcritical under all circumstances, which is however very difficult to be assessed using commonly used simulation methods. In order to justify the simulation results and well determine the physical characteristics of the simulated systems an experimental facility, designed for the measurement of neutron diffusion length, Fermi age and neutron emission rate, is being developed at the Slovak University of Technology in Bratislava. One of the most fundamental parts of the measurements is the loading and unloading of the neutron source to the spherical Plexiglas vessel used for the measurement of neutron emission rate and to the aluminium tank used for the diffusion length and Fermi age experiment. Although the neutron source loading mechanism is remotely controlled and has been designed to minimize the doses of the operation personnel, the management of source loading/unloading and of the whole measurement process are still of a major concern. The present paper deals with the optimization of the operation process by means of the VISIPLAN 3D ALARA and the SCALE6 system. The analysis deals with the development of the detailed model of the measurement facility for dose assessment purpose and with the comparison of achieved results between VISIPLAN 3D ALARA and SCALE6 focusing on gamma radiation of source.

Keywords: *Shielding analysis, Radiation exposure, Dose rate, SCALE6, VISIPLAN 3D*

Introduction

Under a framework of the ongoing research project a new experimental workspace for neutron emission rate, neutron diffusion length and Fermi age measurements is being developed at the Institute of Nuclear and Physical Engineering in Bratislava, Slovakia [1].

For the neutron emission rate measurement, a spherical Plexiglas vessel filled with $MnSo_4.H_2O$ solution, connected with Marinelli baker though a circuit that allows continuous flow of the media, is used. The measurement is based on the activation of manganese isotopes by neutrons emitted from isotope radioactive source and the measurement of the subsequent beta decay. For the measurement of the neutron diffusion length and Fermi age aluminium tank filled with water is used. For both experiments PuBe and AmBe isotope neutron sources, available at our institute, are used. As a result of the unique shielding properties of water, the experimental workspace is sufficiently safe, as far as the neutron source is placed in the Plexiglas vessel or in the water tank.

The most challenging part of the experiment is however the loading of the neutron source, which is commonly performed using the source loading mechanism. In order to perform the measurements in line with the ALARA principle and to minimize the radiation doses collected by the personnel, the measurement process should be optimized in a sense of locating the safest places in the laboratory for the remote handling of the source loading mechanism.

Calculation tools

The analyses presented in this paper were performed using two calculation tools, SCALE6 [2] and VISIPLAN 3D ALARA [3]. The codes used in the analyses are briefly described in this section.

The reference calculation tool used for the presented analyses is the SCALE6 system. It is a comprehensive modelling and simulation suite developed for nuclear safety analyses by the Reactor and Nuclear Systems Division (RNSD) of the Oak Ridge National Laboratory (ORNL) in USA. The dose rate estimation in the preparation phase of the neutron emission rate, neutron diffusion length and Fermi age measurement experiments was performed using the Monaco code of the SCALE6 system. Monaco is a fixed-source Monte Carlo shielding code that calculates neutron and photon fluxes and response functions for specific geometry regions, point detectors, and mesh tallies.

In addition, Monaco has variance reduction capabilities, such as source biasing and weight windows, which can be automated via the MAVRIC sequence (Monaco with Automated Variance Reduction using Importance Calculations). MAVRIC performs radiation transport on problems that are too challenging for standard, unbiased Monte Carlo methods. For the calculations in this analysis 27 neutron and 19 groups library based on ENDF/B-VII.0 [4] evaluated nuclear data library was selected.

Various scenarios bound up with particular performing activities were assessed by using the computer code VISIPLAN 3D ALARA. VISIPLAN code represents analytic tool for external gamma radiation exposure assessment of workers or population. This computer code complies with requested parameters utilizable for the purpose of geometrical model design, materials and sources definition as well as trajectory definition of workers group performing the assigned activities.

The photon fluency rate at a dose point near a volume source can be determined by considering the volume source as consisting of a number of point sources. By adding the contribution of every point source to the dose at the dose point the photon fluency rate at the dose point from the entire source is defined by the equation given below:

$$\Phi = \int_V \frac{S \cdot B \cdot e^{-b}}{4\pi r^2} dV \qquad (1)$$

where S is the source strength per unit volume, r is the distance from a point source, B is the build-up factor, b is a dimensionless term representing attenuation effectiveness coefficient, and V is the volume.

It uses ″point-kernel″ iteration methodology for the calculation of dose rates. This method is based on the fact that bulk radioactive source is modelled by a set of point radioactive sources called ″kernels″, while the resulting dose rate is defined as superposition of partial dose rate from each particular point radioactive source [5].

Calculation models

The model of the experimental workspace was prepared in both calculation codes. In SCALE6 a detailed model was created that consists of all equipment from the laboratory. Due to the limited geometry modelling capability in VISIPLAN, a simplified model was created. Except the simplifications in the geometry, all calculation conditions were the same in both codes. The overview of models can be found in the following.

Detailed model in the SCALE6 system

The detailed model of the experimental workspace created in SCALE6 is shown in Fig.1-a. The model consists of the source loading mechanism with the PuBe source (1), Plexiglas vessel used for the neutron source emission rate measurements (2), crane (3), ladder (4), SUR reactor vessel (5), aluminium tank for the neutron diffusion length and Fermi age measurement (6), IBC tanks used to store the demineralised water (7), ventilation system (8), tables (9) and metal plate wardrobes (10). The model is surrounded by concrete (13) and polystyrene and glass walls (not shown in the picture). The concrete wall is interrupted by two doors, the one from the corridor (11) and the other one from the Mossbauer laboratory.

Simplified model in the VISIPLAN 3D ALARA tool

The simplified geometry model of the experimental workspace created in VISIPLAN is shown in Fig.1-b. The model consists of the source loading mechanism with the PuBe source (1), Plexiglas vessel (2), crane (3), ladder (4), SUR reactor vessel (5), aluminium tank for the neutron diffusion length and Fermi age measurement (6) and 2 IBC tanks used to store the demineralised water (7). In comparison with the detailed model prepared in SCALE6 system, simplifications were made for the source loading mechanism (modelled as concentric cylinders), the crane (modelled as a simple platform), the Plexiglas vessel (modelled as a simple sphere placed in a plastic box) and the IBC containers (modelled as simple cubes filled with water). The SUR reactor vessel and the tank for the diffusion length and Fermi age measurement were modelled identically. The tables, wardrobes and the ventilation system were not modelled in VISIPLAN.

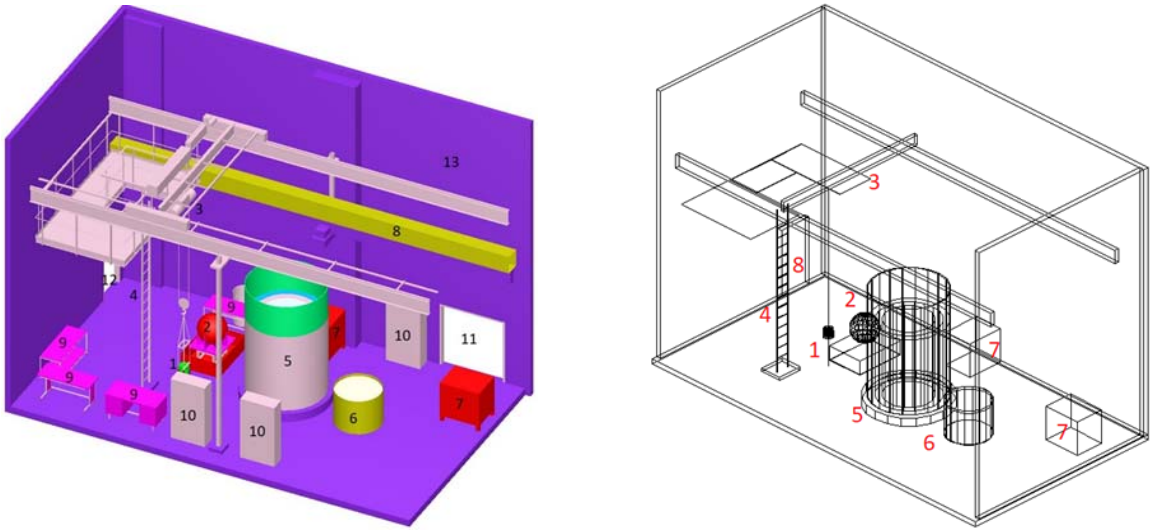


Fig. 1. SCALE 6 detailed model (left) and VISIPLAN simplified model (right) of the experimental site

Definition of the sources

In the SCALE6 system, the PuBe source was modelled as a combined neutron and gamma point source. The spectrum of the neutron source with an emission rate of 5.34E5 ± 1.26E4 n/s was defined based on [6]. The neutron spectrum of the source is shown in Fig.2.

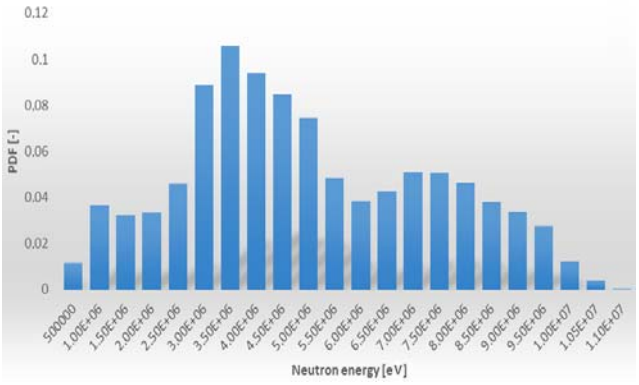


Fig. 2. Probability distribution of the PuBe neutron source

The spectrum of the gamma source with an emission rate of 1.66E+05 ± 1.33E+04 particles/s [7] was defined simply, using only the 4.4 MeV energy, produced as the product of the following reaction.

$$ {}^{9}_{4}Be + \alpha \rightarrow {}^{12}_{6}C^{*} \rightarrow {}^{12}_{6}C + n + 5.71\ MeV \qquad (2) $$

Since VISIPLAN does not support neutron calculations only the gamma source was defined, using the same emission rate as for SCALE model, and the 4.4 MeV photon energy. Considering the small size of the PuBe source, it was justified to model it as a point source, placed at the source loading mechanism. As build-up factor for the calculation, the lead shielding of the source loading mechanism was selected, since it represents the last significant layer for photon attenuation.

Two calculation cases were investigated. In the reference case, the neutron source was placed inside the source loading mechanism, 55 cm above the ground level. In the second calculation case, the source loading mechanism was moved 100 cm upwards, while the source was ejected and remained in position 55 cm from the ground level.

Definition of the responses

To estimate the radiation situation in the SCALE6 system 3D mesh-tallies were used. For the mesh-tallies linear 70x50x70 rectangular mesh was used with two responses defined by ICRU-57 (flux to dose conversion factors). The first calculated response with ID 9036 stands for the effective neutron dose in units of (Sv/h)/(neutrons/cm^2/s) and the second one with ID 9510 represents the effective photon dose in units (Sv/h)/(photons/cm^2/s) [3]. Since the purpose of this paper is to compare the gamma results between SCALE6 and VISIPLAN (does not calculate neutrons), the neutron results are not presented here. The results of analysis primary aimed on the neutron doses can be found elsewhere [8].

In VISIPLAN 2 dimensional rectangular meshes were used through the X-Y and X-Z planes. The X-Y mesh originates in X=315, Y=217 and Z=0 and 18 and 20 mesh elements were used in the X and Z direction. The X-Z mesh originates in X=315, Y=127 and Z=55 and 18 elements were used in both X and Y directions.

SCALE6 results

The gamma dose results of the SCALE6 calculation for both cases are shown in Figs. 3 and 4, where CASE1 represents the shielded and CASE2 the unshielded source. The results for the X-Y and X-Z planes were extracted from the created 3D meshes. For the X-Y plane, Z=55cm and for the X-Z plane, Y=217 cm position was used.

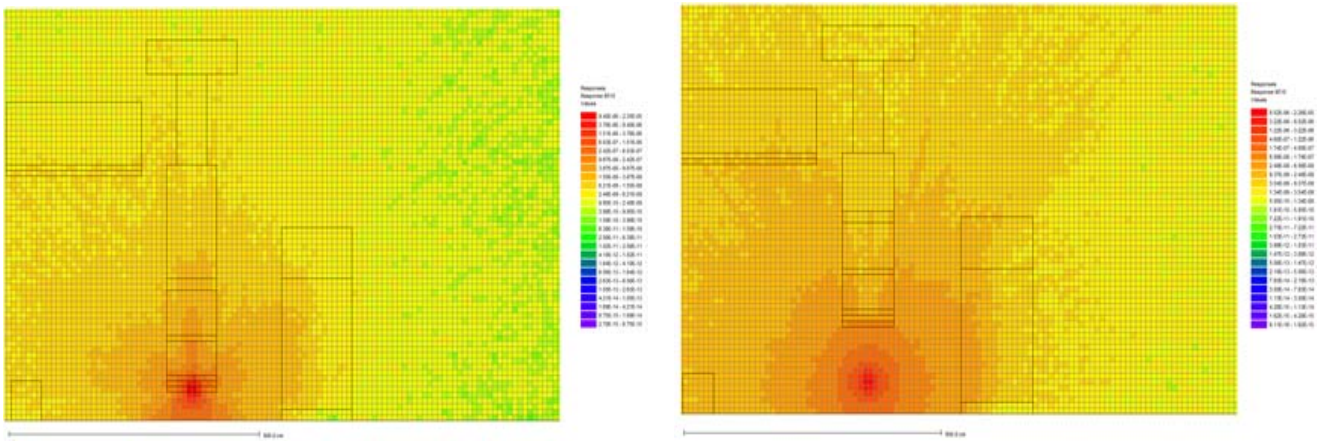


Fig. 3. Results of the X-Z mesh tally in SCALE6 for CASE1 (left) and CASE2 (right)

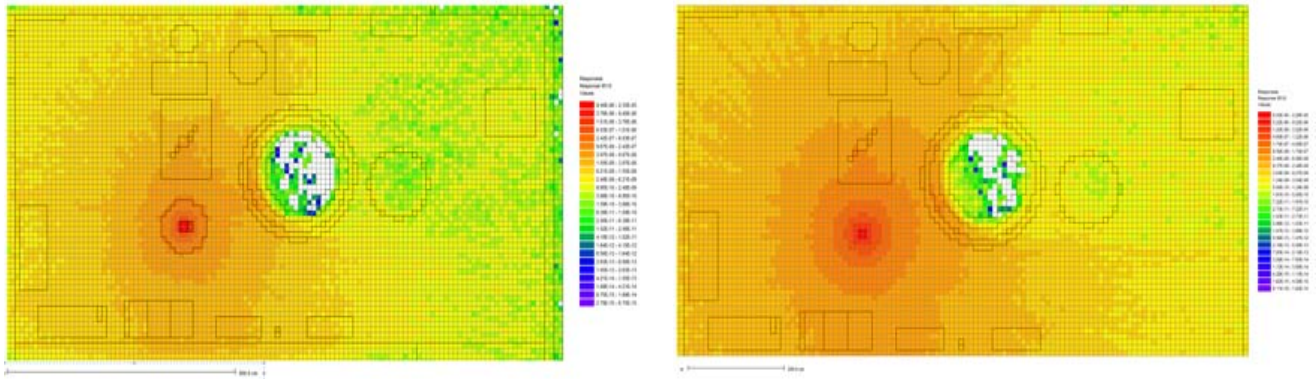


Fig. 4. Results of the X-Y mesh tally in SCALE6 for CASE1 (left) and CASE2 (right)

From the figures above it can be seen that the effective gamma dose, exactly at the source, reaches the level of tens of μSv/h (21.5 μSv/h in CASE1 and 22.5 μSv/h in CASE2).

Due to the shielding properties of the source loading mechanism, in CASE1, the effective gamma doses decreased by more than 2 orders of magnitude at the distance of 50 cm from the source. At the same distance from the source, in CASE2, the effective dose of gamma radiation was approximately 5 times higher. It can be concluded that the area behind the SUR reactor vessel could be an appropriate place for the personnel to hide during source loading and unloading. Another relatively safe place in the laboratory could be the platform of the crane, where the gamma doses decreased by more than 3 orders of magnitude, compared to the ones close to the source loading mechanism.

VISIPLAN results

Since VISIPLAN comes with the capability to calculate doses in mesh elements that collide with solid objects, the size of the calculation mesh was smaller compared to the one considered in SCALE6 calculations.

The results obtained for the two calculation cases over the X-Y and X-Z meshes are shown in Figs. 5 and 6.

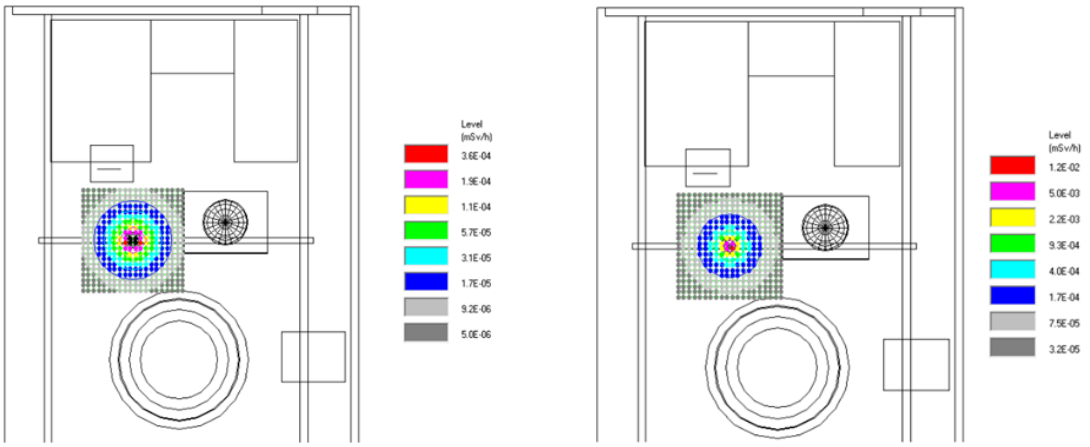


Fig. 5. Results of the X-Y mesh tally in VISIPLAN for CASE1 (left) and CASE2 (right)

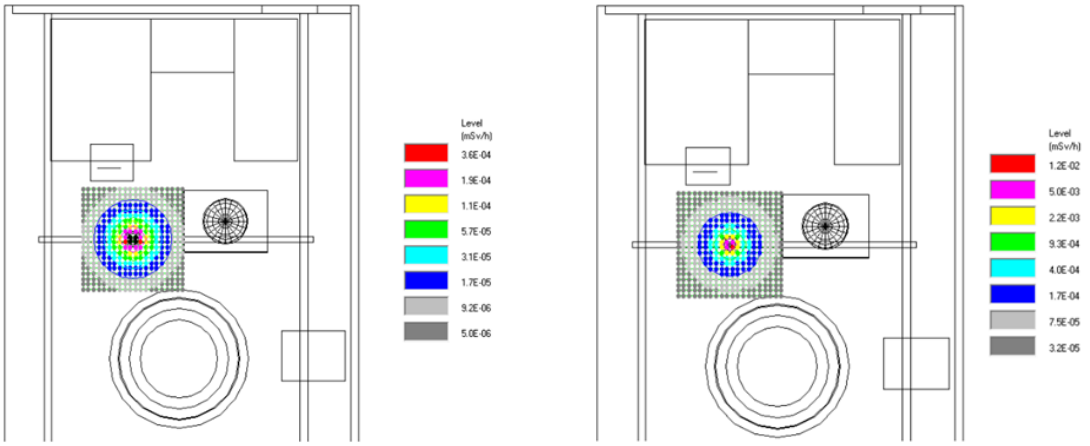


Fig. 6. Results of the X-Z mesh tally in VISIPLAN for CASE1 (left) and CASE2 (right)

Comparison of SCALE6 and VISIPLAN results

For CASE1 a large discrepancy between the SCALE6 and VISIPLAN results can be seen, where in position 1 the difference between the codes exceeds one order of magnitude. Considering the relatively rough averaging of the VISIPLAN mesh results, the differences in the remaining positions are acceptable. The large discrepancy in the first position could have been caused by improper use of build-up factor for the led shielding or by geometry simplifications.

For CASE2 the differences between SCALE6 and VISIPLAN are in good agreement. The differences in positions 2 – 5 could have been caused by the averaging of the scale of the radiation map (only 7 dose bins).

Simulation of movement over a defined trajectory

After the comparison of the mesh results between SCALE6 and VISIPLAN, the analysis of movement of one worker over a defined trajectory was performed in VISIPLAN.

For the trajectory 5 positions were selected, and the effective gamma doses were calculated. The results for both calculation cases (shielded and unshielded source) are shown in Fig. 7.

Positions 1 and 2 represent the movement of the worker as he enters the laboratory and positions 3, 4 and 5 are various scenarios of source loading. For a better visibility, the results summary is given in Table1.

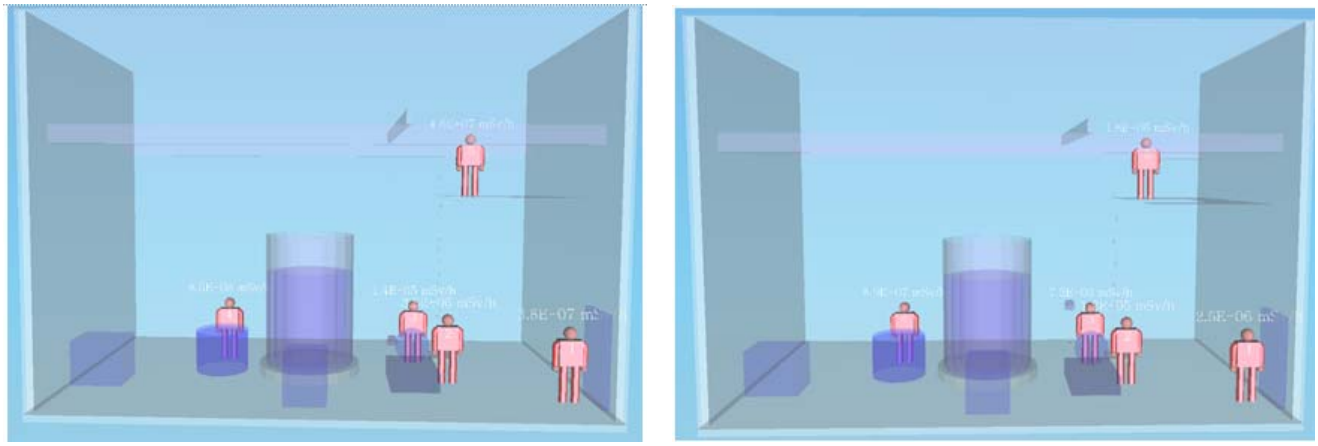


Fig. 7. Results of movement over a defined trajectory in VISIPLAN for CASE1 (left) and CASE2 (right)

Table 1. Summary of the results of movement over a defined trajectory

	CASE1 [µSv/h]	*CASE2* [µSv/h]
Position 1	8.8E-4	2.5E-3
Position 2	3.5E-3	1.2E-2
Position 3	1.4E-2	7.2E-2
Position 4	8.5E-5	8.9E-4
Position 5	4.6E-4	1.8E-3

It can be seen that in positions 1 and 2, the effective doses collected by the personnel were only at the level of the natural background.

The highest doses received were in position 3 (1.4E-2 µ Sv/h). Since this dose is sufficiently low, it is safe to perform the source handling procedure from position 3, close to the shielding, from position 4, behind the SUR reactor vessel, or from the position 5, on the crane platform.

In case of the unshielded source (CASE2), the doses in positions 2 and 3 were sufficiently low. Even though, the gamma dose in position 3 is only (7.2E-2 µSv/h), but considering the fact that in the analyses only the 4.4 MeV photons were modelled and the gamma activity of the plutonium was not taken into account, it would be safer to perform the source handling from positions 4 or 5.

Conclusions

In this paper, the process of source loading in the preparation phase of the neutron emission rate, neutron diffusion length and Fermi age measurement was optimized in a sense of minimizing the gamma doses obtained by the operation personnel.

For the simulations, the SCALE6 system and the VISIPLAN 3D ALARA engineering tool were used. In SCALE6 a detailed and in VISIPLAN a simplified model of the experimental workspace was created.

The simulations were performed using 3D and 2D meshes and responses to calculate the effective dose rate of gamma radiation. Two calculation cases have been investigated, one with shielded and another one with unshielded source. In VISIPLAN also movement of a person over a specified trajectory was investigated.

The comparison of the codes results showed discrepancies in the shielded case, were the difference between the results was greater than one order of magnitude. The discrepancies could have been caused by improper use of build-up factors, geometry simplifications and very rough averaging of the dose rate results in VISIPLAN.

The comparison of the results in the unshielded case showed good agreement between the codes.

The analysis of the movement of the worker over a specified trajectory showed that the safest place for the operation of the source loading mechanism is from the area behind the SUR reactor vessel and from the crane platform.

The analyses also revealed the necessity to model not only the 4.4 MeV gamma energy resulting from the carbon decay, but also the gamma peaks of the ^{239}Pu and ^{241}Am decay.

In the future it will be also necessary to focus more on the selection of proper build-up factors for the VISIPLAN calculations and on the estimation of effective doses due to neutron radiation.

It will be also necessary to perform a sensitivity analysis on selecting different build-up factors for positions close to the source.

Acknowledgements
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