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Evaluation and benchmarking of gamma dose rate employing different nuclear data libraries for MCNP code at the decommissioning stage of Ignalina NPP

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Abstract. A comparative study was performed to reveal the differences of three nuclear data libraries for gamma dose rate calculations when applied to heterogeneous environment in the case of decommission of the Ignalina Nuclear Power Plant (INPP). The following libraries were investigated by employing the Monte Carlo n-particle transport code (MCNP): ENDF/B-VII, JEFF-3.1 and JENDL-3.3, based on the experiments performed for gamma radiation dose rate measurements inside the emergency core cooling system (ECCS) tank with surface radioactive contamination up to 54 Bq/cm^2. MCNP precise simulation and the benchmark between the libraries highlighted the differences of results for the selected case of this investigation. The results revealed that the ENDF library is trustworthy for various dose and shielding calculations and similar applications since it showed a statistically satisfied agreement between the simulation results and experimental data.

Key words: dose rate • Monte Carlo (MC) • nuclear data libraries • decommissioning

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Introduction

The INPP Unit 1 was shutdown in 2004. In the early 2002, the Government of the Republic of Lithuania adopted the resolution on Unit 1 decommissioning by means of immediate dismantling in order to ensure that this process does not lead to serious social, economic, financial and environmental consequences. This is the first project in Lithuania designed for the decommissioning of nuclear power plant (NPP). The first decommissioning activities are linked to the dismantling and decontamination of ECCS of reactor Unit 1 installed in the INPP building no. 117/1. During the decommissioning of any type of nuclear equipment, the most important aspect is to ensure safety of the personnel, population and the whole process. Radiation protection means should be established to protect the workers from the hazardous influence of ionizing radiation during such a type of activities. In this relation, human exposure assessment must be performed in advance to comply with ALARA (as low as reasonably achievable) objectives.

One of the means to evaluate radiation exposure in nuclear application is MCNP code based on the Monte Carlo (MC) method. The origin of MC method was attributed to scientists working on the development of nuclear weapons in Los Alamos during the 1940s. The first Los Alamos general-purpose particle transport MC code was MCS, written in 1963 and used for radiation transport problems. MCS was followed by MCN in 1965, which could solve the problem of neutrons interacting with matter in a three-dimensional (3D) geometry and used physics data stored in separate nuclear data libraries. In 1973, MCN was merged with MCG (a MC gamma code that could use higher energy photons) to form MCNG, a coupled neutron-gamma code. Furthermore, after few years MCNG was merged with MCP (a MC photon code with detailed physics usage down to 1 keV) to accurately model neutron-photon interactions. The code has been known as MCNP ever since. However, at first MCNP stood for Monte Carlo neutron photon, whereas now it stands for MCNP [11].

MCNP, and, in general, the MC method, does not solve an explicit equation, but rather achieves answers by simulating individual particles and recording some physical aspects of their average behavior. MCNP "solves" a transport problem by simulating particle histories rather than by solving an equation: no transport equation needs to be written for solving a transport problem by MCNP. MC is well dedicated for solving complicated 3D, time-dependent problems and as a result, it supplies information only about specific aspects or information requested by the user. The results presented in the literature show that MC simulations can be used as a predictive tool of dose rate measurements in an irradiation plant [7].

The ENDF formats were originally developed for use in the United States national nuclear data files called ENDF/B (the evaluated nuclear data files) [3]. ENDF--format libraries are commonly used for a wide variety of applications that require modeling of the transport of neutrons, photons and charged-particles through materials, also the calculations of the interactions of the radiation with various materials and their surroundings, and the time evolution of the radioactivity associated with nuclear processes. In this particular case, the usage of ENDF-based libraries is grounded on shielding and radiation protection calculations.

The joint evaluated fission and fusion (JEFF) [6] is a European joint project of NEA Data Bank member countries. The JEFF library contains sets of evaluated nuclear data, mainly for fission and fusion applications; it contains a number of different data types, including neutron and proton interaction data, radioactive decay data and photo-atomic interaction data. The JEFF-3.1 Nuclear Data Library is the latest version of the Joint Evaluated Fission and Fusion Library. JEFF-3.1 contains special purpose files with radioactive decay data, activation data and fission yield data, i.e. JEFF Radioactive Decay Data and Fission Yield Libraries.

The Nuclear Data Evaluation Center of Japan Atomic Energy Agency with the aid of Japanese Nuclear Data Committee (JNDC) prepared the Japanese Evaluated Nuclear Data Library (JENDL): an evaluated Nuclear Data Library [9], which contains a data file with recommended nuclear data. The latest version JENDL-3.3 released in 2002 contains various kinds of data in the wide energy range. In addition to that, the JENDL special purpose files, which contain only the data needed in their application field, are used for calculations of particular cases. JENDL Photonuclear Data File, photon-induced reaction data up to 140 MeV for 68 nuclides were employed for solving problems described further in this article.

These three nuclear data libraries contain essential experimental data for solving problems related to radiation transport in various environments. Theoretical models and relevant calculations of photon production and photonuclear reaction data are developed in all three libraries. Over the last few years, the benchmark between these libraries for various neutronic calculations has been performed and represented in the literature. Nuclear data libraries were compared through the analysis of a shutdown dose rate experiment in a mock up of the ITER vacuum vessel. The differences were detected between the data packages of the order of 20% at a maximum in the dose rate [1]. Moreover, the benchmarks of nuclear data libraries for various fission products were performed, i.e. the estimation better the present status of the fission-product data were performed by a similar analysis of the ENDF/-B-VI, JEFF-3.0, JENDL-3.3, and FOND-2.2 evaluations [4]. In spite of all available data, nuclear data libraries still need to be analyzed more, including the comparison with performed measurements emphasizing gamma dose rate calculations. Radiological safety, the decrease of gamma radiation effects and correct estimation of gamma doses on operating personnel, as well as shielding design and installation are the major problems during the decommission stage of NPP.

In this article a comparative study and revelation of distinctions of three different nuclear data libraries, i.e. ENDF/B-VII, JEFF-3.1 and JENDL-3.3, for gamma dose rate calculations when applied to heterogeneous environment in the case of decommission of INPP are presented. To conduct such research, the estimation of radiation gamma doses and risks from residual radioactive materials using MCNP code for the worst case scenario (based on where maximal contamination level was defined) were performed and a comparison of the numerical simulation results with experimental data is presented in this article.

Experimental data

As it was outlined earlier in this article, one of the first decommissioning stages was dismantling ECCS INPP located in 117/1 building. ECCS includes 16 tanks and their schematic view is shown in Fig. 1. The main function of those tanks is to provide the short term cooling of the reactor in the event of a loss of coolant from the reactor cooling system. The volume of one tank equals 25 m^3. Each tank was filled with water and pressurized with nitrogen above the water surface. ECCS tank consists of a welded vertical cylinder with two ellipsoidal bottoms, support structure, nozzle and man-hole, with the following characteristics: height – about 14 m, mass (without water) about 48 t, outside and internal diameters – 1.76 m and 1.60 m, respectively. Dominating material is carbon steel.

In order to constrain the calculation model within MCNP code and to perform a comparative analysis, a set of gamma dose rate measurements were employed. These experiments were performed by INPP personnel using a dosimeter Teletector [8]. The main purpose of these measurements were to define the radioactive contamination at the bottom and upper part of ECCS

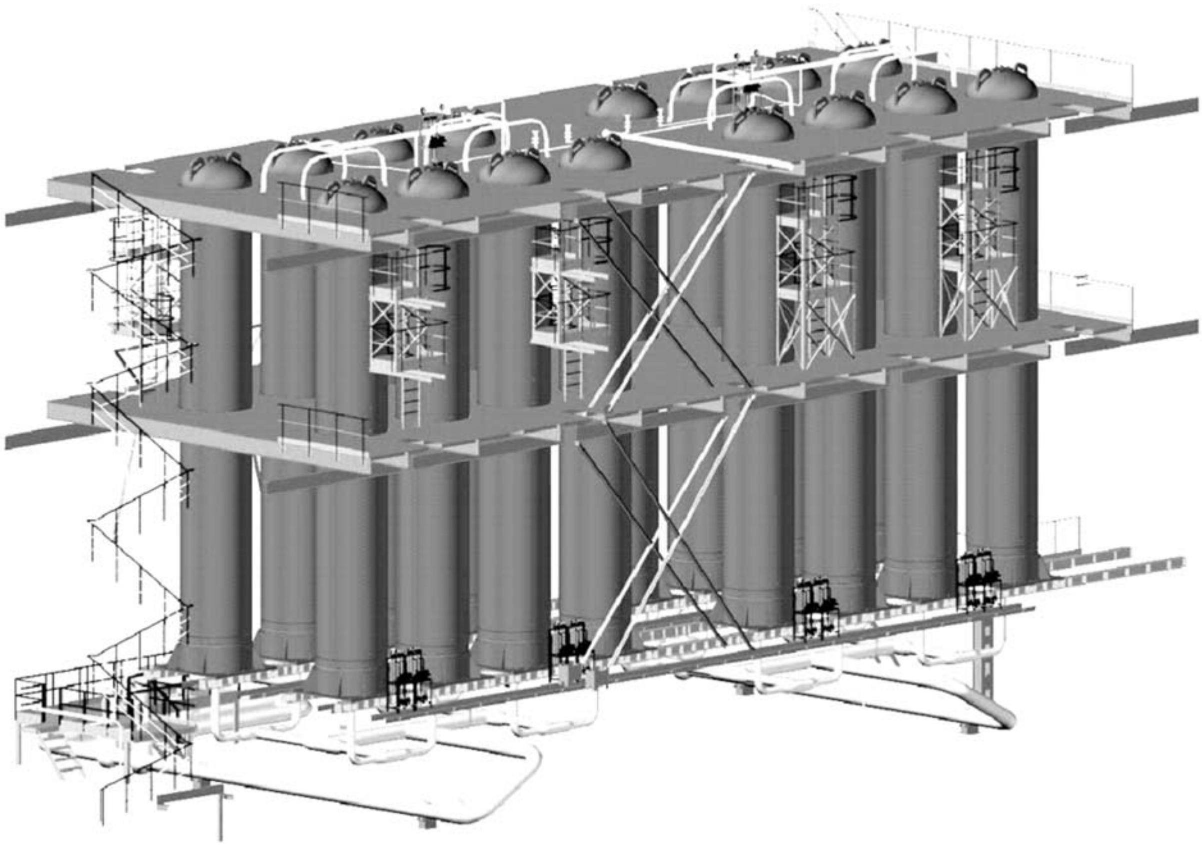

Fig. 1. Schematic view of the ECCS tank set [8].

tank located in building 117/1 of INPP. In addition, the determined nature of contamination, identified radionuclides and their contribution to the total activity provided essential information on model source definition and numerical simulation.

Equally, for the following investigation the experimental data and required information were taken from the performed direct measurements of the surface contamination. Gamma radiation dose rate measurements were carried out inside at the bottom of the tank with water level of 0.1 m and without water. These data, represented in Tables 1 and 2, became the main reference points for the study. The decontamination procedure by flushing out inside ECCS was performed with the aim to achieve a free release level and thereby from the point of radiological safety considerations to reduce gamma dose rate on operating personnel. After the decontamination, the water level of 0.1 m was formed at the bottom of the ECCS tank.

Measurement results, obtained by wet smear, indicated that the surface contamination of the tank internal surface was from 0.4 Bq/cm^2 (upper part of the tank) to 54 Bq/cm^2 (bottom of the tank). The radioactive contamination of the vertical surface (wall of the tank) appeared strongly fixed. The radioactive contamination of the horizontal surface (bottom of the tank) consisted of poorly fixed contamination (silt sediments) and the fixed one, which was confirmed by direct measurements (Table 1). Gamma dose rate measurements showed that the increase of the distance to 1 m from the bottom of ECCS tank to the measuring point can reduce dose rate by a factor of 3. This statement is true for both the

Table 1. Contamination measurements at the bottom of ECCS tank [8]

Distance from the bottom of ECCS to the measuring point (m)	Averaged gamma radiation dose rate, experimental data (μSv/h)	
	Before decontamination (without water)	After decontamination (with water)
0.1	12.5	9.5
1.0	3.6	2.7

Table 2. Radionuclide content of contamination at the bottom of ECCS tank defined by wet smear [8]

Radionuclide name	Activity (Bq/cm^2)	
	Before decontamination	After decontamination
^{54}Mn	8.0	0.85
^{60}Co	34.7	4.35
^{137}Cs	11.3	4.06
Total surface activity	54.0	9.30

measured cases, i.e. before and after decontamination. However, the impact of decontamination procedure reduced the gamma dose rate up to 25%. In addition, radionuclide content measurements of contaminated inventory and total surface activity at the bottom of ECCS tank before and after decontamination revealed the main contributors of the actual remaining activity. The experimental data shows that the main contribution to the total activity was influenced by ^{60}Co isotope, while the contribution of ^{54}Mn and ^{137}Cs radionuclides activity was low, but not negligible (for more details, see Table 2).

MCNP model

MCNP is a general-purpose Monte Carlo n-particle code which can be employed for neutron, photon and electron transport. Radiation protection and dosimetry, radiation shielding, nuclear criticality safety, fission and fusion reactor design, decontamination and decommissioning are specific areas of application. The code uses an arbitrary 3D configuration of materials in geometric cells. The MCNP code utilizes the latest nuclear cross--section libraries and uses physics models for particle types and energies.

MC can be used to describe theoretically a statistical process, such as the interaction of nuclear particles with materials, and it is especially valuable for complex situations which are difficult to be modeled by computer codes that use deterministic methods. In particle transport, the MC technique is treated as a numerical experiment. It is composed of every following particle originated from a source during its life and transportation to its death, i.e. particle capture, absorption, escape, etc.

In this gamma dose rate study, the MC multi-particle transport code MCNP4C2 was employed. The dose rate estimation was performed from the cylindrical object: an ECCS tank with inner surface contamination (see Table 2) located in the 117/1 building of INPP. The geometry of the simulation object (Fig. 2) is described with the following parameters: outer radius – 0.88 m, thickens of the pipe wall – 0.08 m, height –13.87 m and mass – 47 650 kg. Gamma dose rate evaluation was performed from the cylinder with closed ends. The distance between radiation source and point detector were 0.1 m and 1.0 m. In addition, a case with 0.1 m water level inside the tank was analyzed. The gamma transport calculations were performed by MCNP for all non-void geometry cells. MCNP photon tally f5 was used as a particle flux at the point detector followed by the modification by a dose function to calculate gamma dose rate.

The case scenario data provided in Table 2 were used as gamma radiation source from the inner surface of the cylinder. The gamma energy spectrum of the ^{54}Mn, ^{60}Co and ^{137}Cs sources used in investigation and in MCNP simulations to estimate the gamma dose rate is represented in Table 3.

Simulation results

In order to evaluate gamma dose rate, as well as the contribution of the main radionuclides to the total emitted radioactivity, three nuclear data libraries, i.e. ENDF/B-VII, JEFF-3.1 and JENDL-3.3, were used during the investigation. Experimental data were taken for comparison with the benchmark results and for the representation in the figures of each studied case. The estimated gamma dose rates had shown an acceptable agreement with experimental results using all nuclear data libraries. Nevertheless, some results need a more detailed analysis for explaining the differences of the results.

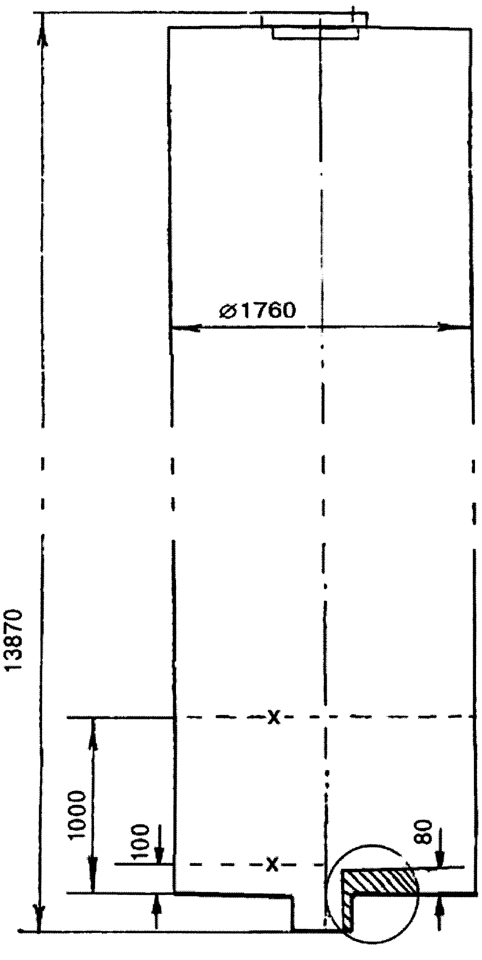

Fig. 2. Schematic view of the ECCS tank model for MCNP simulation (the sign "X" marks measuring points), (mm).

Table 3. Gamma lines and emission probabilities [2, 5, 10]

^{54}Mn		^{60}Co		^{137}Cs	
Gamma line, *Eg* (keV)	Probabilisty, *Ig* (%)	Gamma line, *Eg* (keV)	Probabilisty, *Ig* (%)	Gamma line, *Eg* (keV)	Probabilisty, *Ig* (%)
		1173.237	99.974		
834.848	99.976			661.657	85.120
		1332.501	99.986		

Simulation and comparison with experimental data were made for four cases using separate library, i.e.:

– Distance from the bottom surface – 0.1 m, the tank without water.
– Distance from the bottom surface – 1.0 m, the tank without water.
– Distance from the bottom surface – 0.1 m, the tank with water.
– Distance from the bottom surface – 1.0 m, the tank with water.

In addition, the contribution of each nuclide, i.e. ^{54}Mn, ^{60}Co and ^{137}Cs, obtained from MCNP simulation, is presented in each figure. The contribution of other radionuclides was not considered due to their negligible impact on the total activity.

The MC simulation of the first two cases was performed without water inside the ECCS tank. The simulation results allowed us to reproduce experimental data with acceptable agreement for the cases without water using ENDF and JEFF nuclear data libraries (Figs. 3 and 4). The MC simulation of the second two cases with water confirmed the marginal decrease of the influence of water inside the ECCS tank to the total dose value (Figs. 5 and 6). Simulation results obtained using all three libraries indicated that the increase of the distance to 1 m (from the bottom of ECCS tank to the point detector) can reduce dose rate by a factor of 3 for both the cases of simulation, i.e. with and without water level, which was confirmed by experimental data outlined earlier in this paper.

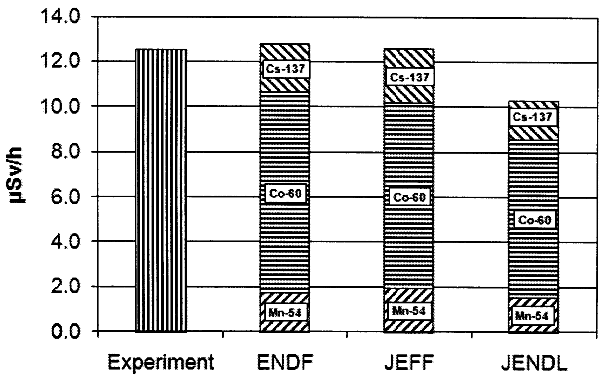

Fig. 3. Comparison of gamma dose rates (distance from the surface 0.1 m, the tank without water) before decontamination.

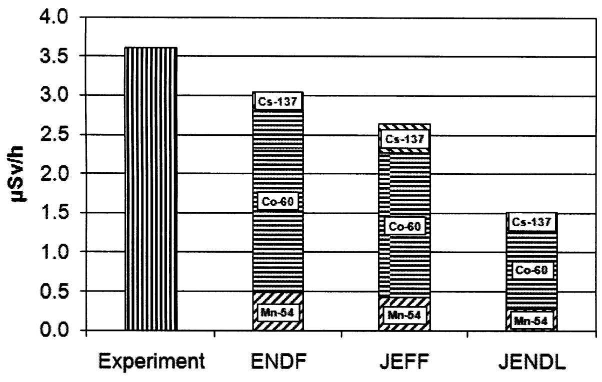

Fig. 4. Comparison of gamma dose rates (distance from the surface 1.0 m, the tank without water) before decontamination.

For the first case, when the point detector is placed 0.1 m above the bottom of the ECCS tank before the decontamination (without water inside the tank), all three libraries showed a very good agreement with experimental measurement (Fig. 3). Note that the computer running time was chosen in such a way that the statistical error of MC simulation results were always less than 10%. As it is clearly shown in Fig. 3 and further, the main contribution to the total dose rate is generated by ^{60}Co, while the impact of ^{54}Mn and ^{137}Cs varies differently for all cases of simulation. Nevertheless, their impact is not negligible and should be taken into account during the determination of the shielding efficiency performing the decommissioning activities in INPP.

The JENDL library showed apparently worse results in the benchmark between the European and American libraries in the case of no water inside ECCS tank. The difference between JENDL and test data is 38%, on the average, while ENDF and JEFF showed 9% and 14% discrepancy, respectively (Figs. 3 and 4). This can be explained by the missing specification data for widely used elements as silicon and iron isotopes in the chemical characterization of stainless steal and concrete materials, which are dominating in the constructions of the experiment environment. In addition, numerical simulation using JENDL library confronted with the lack of gamma-ray production cross-section for ^{137}Cs, which appeared to be a significant contributor to the total value of estimated gamma dose rate. This incompleteness of experimental data makes simulation with JENDL library highly model

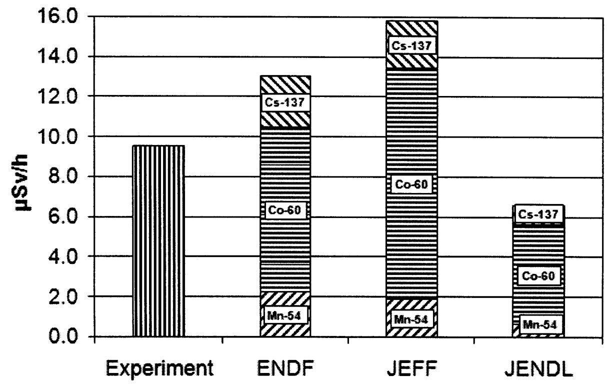

Fig. 5. Comparison of gamma dose rates (distance from the surface 0.1 m, the tank with water) after decontamination.

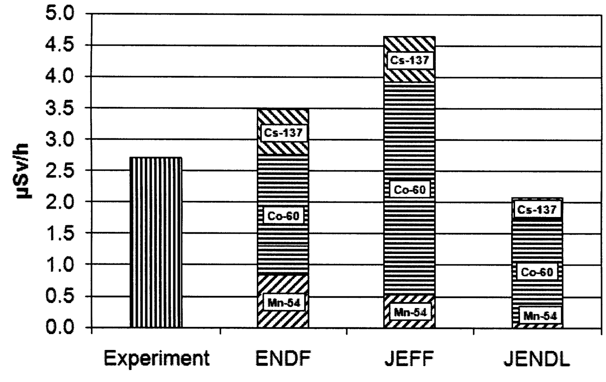

Fig. 6. Comparison of gamma dose rates (distance from the surface 1.0 m, the tank with water) after decontamination.

dependant due to the employment of user-defined parameters.

One should note separately that the total gamma dose rate values obtained employing European JEFF nuclear data library gives maximal values in comparison with ENDF and JENDL libraries, and especially with experimental measurements in case of water inside ECCS tank. The discrepancy between JEFF and experimental data is 68%, on the average, while ENDF and JENDL demonstrated 27 and 26%, respectively (Figs. 5 and 6). Such phenomena can be related to the fact that the simulation results showed the enhanced ^{137}Cs contribution, while ^{54}Mg and ^{60}Co gave relatively similar input to the total dose rate value. In general, JEFF library provided results with average agreement to experimental data and the discrepancy exceeded up to 41%, on the average, while JENDL underestimates the experimental data nearly by 32%.

Finally, the best agreement with experimental data was obtained when the MC simulation was performed using ENDF nuclear data libraries. The average difference between numerical calculations and experimental measurements did not exceed more than 18% (Figs. 3–6). Such achievement allowed to reproduce perfect measurement results on gamma dose rates during decommission stage of ECCS at INPP.

Conclusions

Valuable information and additional knowledge were obtained during this study concerning the estimation of gamma dose rates, which has a great importance during the dismantling operation of the installations of nuclear reactors and for the studied case during dismantling of ECCS at INPP. One of the most important phases during decommissioning of such facilities is the estimation of safety related factors; thus, it is very important to ensure safe operation with radioactive materials under hazardous conditions. For the above reasons, the benchmark study for gamma dose rate was performed using three different nuclear data libraries: ENDF/B-VII, JEFF-3.1 and JENDL-3.3. The theoretical model was developed and the employed MCNP code allowed to simulate the experiment environment with high precision.

The benchmark between the three nuclear data libraries highlighted the individual lacks of specific data for the analyzed cases with and without water inside ECCS tank for the different distances from the radiation source to the measuring point. The benchmark of the investigated nuclear data libraries revealed that the usage of ENDF is the most efficient for various dose or shielding calculations since it showed statistically satisfying agreement between simulation results and experimental data.

MC simulation results for gamma dose rate inside the ECCS tank with surface radioactive contamination showed that the increase of the distance from 0.1 to 1 m from the bottom of the ECCS tank to the point detector can reduce dose rate by a factor of 3 for the both cases of simulation, i.e. with and without water level. This statement was confirmed by experimental measurements as well. Furthermore, MCNP simulation confirmed the marginal decrease (up to 25%) in the influence of 0.1 m level of water inside the ECCS tank to the total dose value.

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