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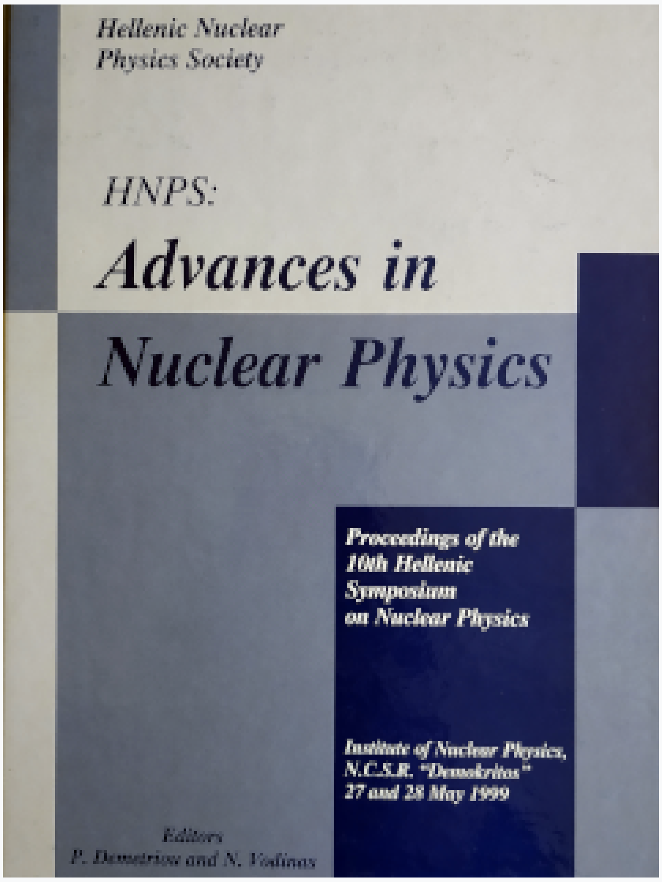

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Evaluation of external doses from exposure to gamma sources in the soil using the MCNP code

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Abstract

Monte Carlo Neutron, Photon transport code (MCNP) was utilized to calculate the external dose from exposure to gamma emitting radionuclides on the ground. The results of the simulation were compared with the experimental data and the results of calculations performed using other analytical methods of estimating external doses from the ground and was found to be in a very good agreement.

1 Introduction

The dose rate from exposure to gamma radionuclides on the ground can be estimated using various techniques. These include in situ measurements or theoretical calculations based on the radioactive concentrations in the soil. It is suggested that two or more approaches should be used simultaneously in order to verify the results and the techniques themselves [1, 2].

The present work is concerned with radiation dose equivalent due to exposure from radionuclides found on the ground. In particular, the external radiation doses from the radionuclides commonly found in the natural soil environment were estimated by the application of two methods: Gamma-radiometry and Monte Carlo calculations using the MCNP code, taking into consideration the concentrations of gamma emitters in the soil. The calculated results are compared with the experimental data and the results of calculations performed with other analytical methods.

Field measurements and the respective soil sampling points were located in the Ikaria island. Ikaria island, an area of 267 km^2, is located in the eastern Aegean Sea. The natural radioactive status of the island which is mainly characterized by its highly radioactive springs is of particular interest in relation to abiotic and biotic environment. Sampling areas were chosen on the basis of a geological map [3, 4].

2 Experimental procedure

2.1 Gamma radiometry

To determine the natural gamma radiation status in the areas of interest, as well as in some reference areas, gamma radiometry has been applied using a 1″x 1″ NaI (Tl) cylindrical detector. The detector was calibrated using a ^{226}Ra standard source. With this technique, values of photon fluence density and dose rate were obtained at several measurement points.

2.2 Gamma spectrometry

To determine the inventory of the external environmental radiation levels, soil samples were also analysed by gamma spectrometry. The above samples were analyzed using a high resolution gamma spectrometry system with a HpGe detector of 20% relative efficiency to 3″x 3″ NaI detector. Samples were collected from each one of the respective gamma radiometry measurement points.

Activity concentrations of ^{40}K, ^{238}U, ^{226}Ra, ^{228}Th and ^{228}Ra were determined. It was assumed that the gaseous nuclides of radon and thoron do not escape from the soil. The received dose rate from external irradiation was calculated from this data.

3 Theoretical calculations

3.1 MCNP code

MCNP-4B code is a general purpose, continuous energy, generalized geometry, coupled neutron, photon, electron Monte Carlo transport code system [5]. MCNP-4B code was used to perform a simulation of the soil-layer and NaI(Tl) detector configuration. The code was used in conjuction with a cross-section library based on ENDL and ENDF cross-sections data.

Based on literature data [6, 7, 8, 9, 10, 11] and environmental field conditions (soil layer, homogeneity, soil slope, subject rock layer) the calculation geometry adapted for this problem is shown in Fig. 1. A detector-receptor situated 1m above the soil was assumed. The soil acts as a uniform gamma source of radiation.

The input materials in the studied geometry are soil, air, NaI and Al. Air is assumed to consist of N_2, O_2 and Ar, while soil is basically assumed to consist of SiO_2, Al_2O_3 CaO, K_2O and H_2O. Table 1 shows densities and chemical compositions of air and soil.

A cell tally was utilized. A cell tally calculates the track length estimate of the particle flux over a specified cell. A cylindrical NaI(Tl) cell detector of 2.54 x 2.54 was used. The detector in the above geometry has the same characteristics with the detector used in the field measurements and detects photon fluence density (photon/ cm^2 . source photon). The photon fluence conversion to dose rate is performed using the ICRP51 [12] photon fluence to dose rate conversion factors.

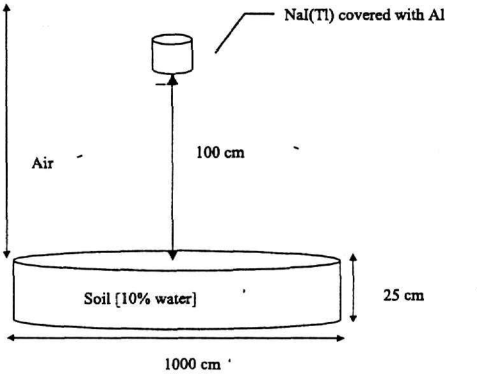

Fig. 1. Simulation geometry

3.2 Analytical formulae

The results of the activity concentrations of the radionuclides in the soil (Bq kg^{-1}) as derived by gamma spectrometry were used to determine the external dose rates at the reception point.

Table 2 shows the analytical formulae utilized to calculate the external dose rates.

4 Results and Discussion

The minimum, maximum and mean values of annual dose equivalent rate (μSv y^{-1}) obtained from the application of the different techniques are given in Table 3. The results allowed ratification of the two dose rate measurement techniques used in this study.

The dose estimations performed in the present study using the MCNP code are found in excellent agreement with the corresponding experimental values. Comparing the dose equivalent rates derived from MCNP code and the other analytical formulae for the same soil activity concentrations, a general agreement was found.

Table 1
Composition and density of soil (10% water) and air assumed in the calculation [13].

Soil		Air	
	Abundance (% by weight)		Abundance (% by weight)
Si	26.60	N	75.5
Al	7.84	O	23.2
Fe	4.83	Ar	1.3
Ca	3.47		
K	3.47		
O	53.60		
H	1.11		
Density ($g.cm^{-3}$)	1.0	Density ($g.cm^{-3}$)	1.205×10^{-3}

Table 2
Analytical formulae for external dose rate calculation

D1(μSv y^{-1})	=	2.618 As(^{238}U) + 2.093 As(^{232}Th) + 0.264 As(^{40}K) [14, 15, 1, 16]
D2(μSv y^{-1})	=	3.017 As(^{238}U) + 2.409 As(^{232}Th) + 0,294 As(^{40}K) [17,18]
D3(μSv y^{-1})	=	3,550 As(^{226}Ra) + 2.416 As(^{232}Th) + 0.235 As(^{40}K) [19, 20]
D4(μSvy^{-1})	=	0.061As(^{238}U)+3.1As(^{226}Ra)+2.582(^{228}Th)+1.741As(^{228}Ra) +0.276As(^{40}K) [11]

where Di =exposure dose rate in μSv y^{-1}, As(i)=activity concentration of the radionuclide i considered in soil in Bq kg^{-1}.

However, the values obtained using the MCNP code are lower than the respective values obtained using the other analytical formulae. These differences can be attributed to a better representation of the environmental parameters of the specific area in the MCNP simulation, rather than the application of the other formulae. It is noted that dose equivalent rate calculated in relation D4 assumes 100% outdoor exposure and therefore approaches better the MCNP code simulation.

Table 3
Annual dose equivalent rates derived from gamma radiometry measurements and theoretical approaches applied in 33 samples of the present work (μSv y^{-1}).

Min	Max	Mean Value	Calculation method
157	3604	643	Theoretical estimation [D1]
178	4129	734	Theoretical estimation [D2]
237	3752	704	Theoretical estimation [D3]
202	3312	635	Theoretical estimation [D4]
228	3000	449	Experimental value [This study]
123	3130	484	MCNP calculation [This study]

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