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A Radiation Shielding Analysis for 500W Radioisotope Thermo-Photo-Voltaic (RTPV) System with MCNP6 and MONACO/MAVRIC

Seong Jae Cheon and Ser Gi Hong*
Dept. of Nuclear Enginnering, Kyung Hee University, 1732 Deokyoungdaero, Giheung-gu, Yongin, Gyeonggi-do 446-701

Corresponding author: sergihong@khu.ac.kr

1. Introduction

Radioisotope Thermoelectric Generators (RTG) has been used as a power supply system for long-lived operation of spacecraft. RTG is an essentially nuclear battery that reliably converts the heat resulted from the radioactive decay into electricity. However, it is known that RTG systems have relatively low thermoelectric energy conversion efficiency, which makes the advantage of high energy density of the radioisotope small. On the other hand, a RTPV (Radioisotope Thermo-Photo-Voltaic) system has typically much higher energy conversion efficiency of ~20% than RTG system because RTPV energy conversion is a direct conversion process from heat differentials to electricity via photons.

Recently we have studied a RTPV system using alpha decay heat from $^{238}PuO_2$ as a long-term power supplier. However, the radioactive decays are accompanied by radiations such as neutron, alpha, beta and gamma. Therefore the shielding of these radiations is very important for radiation safety of the workers. The objective of this work is to perform the shielding design and analysis for the RTPV system. In this work, the RTPV system is introduced and then, the shielding design and analysis are performed using the MCNP6 and MONACO/MAVRIC codes for the radioisotope heat source block for the RTPV system.

2. Radiation Safety Analysis

2.1 Selection of radioisotopes of RTPV source.

When designing a RTPV system, the radioisotope for heat source should be selected by considering several factors such as specific power and volumetric power density because they can determine the volume and mass of the system, respectively for a given power. Fig. 1 compares the energy and power densities for several candidate radioisotopes. From Fig. 1, it is shown that Sr-90(Y-90) and Pu-238 have higher energy and power densities than the other ones considered. So, these two isotopes have been popularly considered as the radioisotopes for heat source. $^{90}SrTiO_3$ and $^{238}PuO_2$ are the most widely used compounds for Sr-90 (Y-90) and Pu-238, respectively [1]. Table I compares the main features of Sr-90 and Pu-238. As shown in Table I, Sr-90 undergoes β-decay into Y-90 with half-life of 28.9 years while Pu-238 undergoes α-decay into U-234 with half-life of 87.7years. The average decay energies for Sr-90 and Pu-238 are 0.1958 and 5.5MeV, respectively. Their specific activities of their compounds (i.e., $SrTiO_3$ and PuO_2) are 133 and 15 Ci/g, respectively. $^{238}PuO_2$ has lower specific power of 480W/kg but higher volumetric power of 5.52 W/cm^3 than $^{90}SrTiO_3$. So, the $^{238}PuO_2$ can play a role as a compact heat source over much longer time period than $^{90}SrTiO_3$. So, $^{238}PuO_2$ is selected as the radio-compound for the RTPV system.

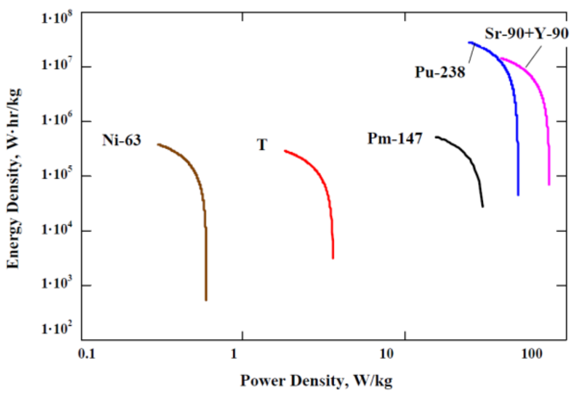


Fig. 1. Ragone curves for NB (T, Pm-147, Ni-63, Sr-90+Y-90) and RTG (Pu-238)

Table I : Unit volume of power of radioactive isotopes used in the source

Parameters	Values	
	$^{238}PuO_2$	$^{90}SrTiO_3$
Half-life of isotope ($T_{1/2}$, year)	87.7	28.9
Specific activity (Ci/g)	15.0	133.0
Specific power (μW/Ci)	32000.0	6665.92
Type of decay	α	β
Average decay energy (MeV)	5.5	Sr : 0.196 Y : 0.93
Initial power density (W/kg)	480.0	889.2
Energy density over full life (W·hr/kg)	5.3x10^7	3.2x10^7
Density (g/cm^3)	11.5	5.11
Initial volumetric power density (W/cm^3)	5.52	4.538
Melting point (^{o}C)	2400	2080

Also we selected $^{238}PuO_2$ as the heat source because its chemical and material stabilities have been well verified in nuclear engineering industry. With the volumetric power density given in Table I, the volume of the heat source region can be determined for a given thermal power. In this study, we assumed the thermal power of 500W which leads to a heat source region of 90.5796 cm^3.

2.2 Radiation shielding analysis procedure

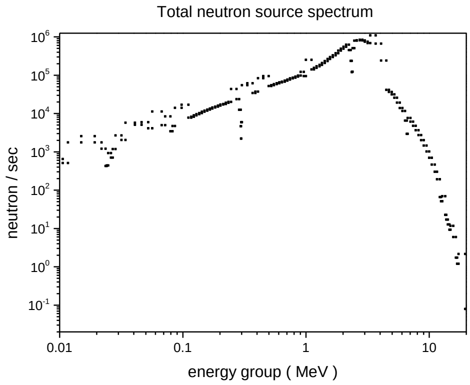
The radiation safety analysis consists of three steps : 1) identification of the radiation sources and estimation of radiation source intensities and spectra, 2) setup of geometrical model for RTPV device, and 3) radiation transport simulation for evaluating the radiation dose. In this work, the ORIGEN-S code was used to evaluate the intensities and spectra of this neutrons and gamma rays emitted from the $^{238}PuO_2$ [2]. Then, the radiation fluxes and doses are estimated by performing neutron and gamma transport simulations using two Monte Carlo codes and the source intensities and spectra estimated with ORIGEN-S above. In this work, we used MCNP6 and MONACO/MAVRI to perform the neutron and gamma transport calculations [4,5].

2.3 Source Term Evaluation with ORIGEN-S

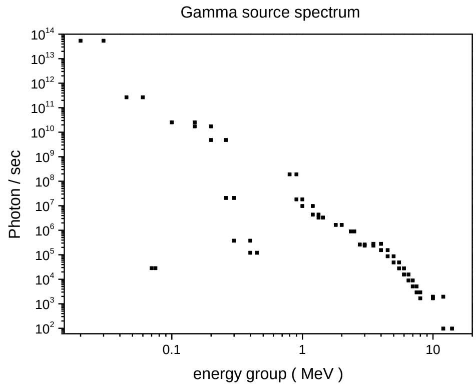
ORIGEN-S which is a part of SCALE6.1 is a computer code for evaluating the radioactive decay of radionuclide and the amount of radionuclide after the neutron irradiation and cooling. The radiation source terms should be described prior to the detailed radiation safety analysis [2]. The alpha decay of ^{238}Pu accompanies 5.5MeV energy and also additional neutrons are released by the reactions (i.e., (α,n) reactions) of ^{17}O and ^{18}O with the alpha particles. In addition, the spontaneous fission of ^{238}Pu generates the neutrons. For gamma source, the most dominant contribution is from the accompanied gamma by the decay of ^{238}Pu while the spontaneous fission of ^{238}Pu and (α,n) reactions generate additional gammas [3]. The gammas from the decay of ^{238}Pu have peaks at 43 keV, 99 keV, and 152 keV energies. We estimated these gamma and neutron source intensities and spectra by using the ORIGEN-S code. Table II summarizes the neutron source intensities estimated using ORIGEN-S for a 500W heat generation. As shown in Table II, the main contribution to the neutron source is from the reactions of ^{18}O. On the other hand, the gamma rays are accompanied by the alpha decays of ^{238}Pu, spontaneous fissions, decays of the fission products, and the (α,n) reactions. Fig. 3 shows the spectra of the total neutron and gamma sources. These spectra are obtained by using ORIGEN-S with the 200 and 47 energy group structures for neutron and gamma, respectively. The total gamma and neutron source intensities were estimated to be 5.413×10^{13} photons/sec and 1.492×10^{7} neutrons/sec, respectively, for 500W heat generation.

Table II: Neutron source intensities (neutrons/sec) for 500W $^{238}PuO_2$ source

Components	Values
Spontaneous fission	2.39×10^{6}
Delayed neutrons	Negligible
(α,n) reactions - (mainly from ^{18}O)	1.253×10^{7}
(α,n)reactions - (from ^{17}O)	9.996×10^{5}
Total (α,n) reactions	1.253×10^{7}
Total	1.492×10^{7}



(a) Neutron source spectrum



(b) Gamma source spectrum

Fig. 3. Comparison of the total neutron and gamma source spectra

2.4 Radiation safety analysis with MCNP6 and MONACO/MAVRIC (SCALE6.1)

In this section, we performed radiation safety analysis by using MCNP6 with ENDF/B-VII.r1 cross sections and MONACO/MAVRIC codes with the specified intensities and spectra of the radiation source in Sec. 2.3. Monaco is a new 3-D Monte Carlo code being developed within SCALE for shielding calculations. It is a fixed-source, multi-group Monte Carlo transport code for shielding applications. And the MAVRIC sequence is completely automated [5]. The multi-group transport calculations were performed with the cross section library of 'v7-200n47g' in SCALE6.1. The size of the source is fixed because the release of heat source is fixed to 500W. RTPV device consists of a 4.4910cm x 4.4910cm x 4.4910cm central cubic source region and its two surrounding regions. The central source region is first surrounded by tungsten (W) radiation shielding region which is followed by a 0.5cm thick outer tantalum (Ta) emitter region. The radiation dose is estimated in a 10cm thick spherical water shell which is located at 100cm distance from the center of the RTPV device. The region between RTPV device and the water spherical shell is filled with air. Fig. 4 shows MCNP6 modeling of the RTPV device for radiation safety analysis. The radiation dose is calculated using the ICRP-21 dose conversion factors that are provided in MCNP6.

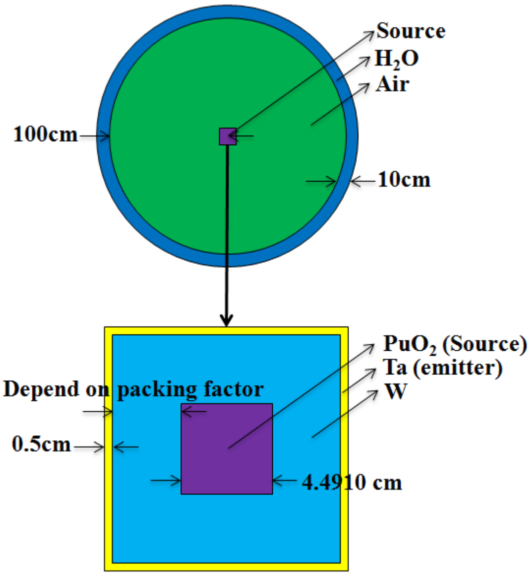


Fig. 4. Geometrical model of RTPV device for radiation safety analysis

The efficiency of RTPV strongly depends on the packing factor which is defined as the ratio of the $^{238}PuO_2$ source region volume to the total RTPV device volume. So, in this work, we analyzed the effects of the packing factor on the radiation dose.

Table III : Results of the radiation safety analysis for 500W $^{238}PuO_2$ RTPV device with MCNP6

Items	Packing factor (%)			
	10.0	30.0	50.0	70.0
Neutron dose (mSv/hr)	0.128	0.149	0.157	0.161
Gamma dose (mSv/hr)	9.06×10^{-5}	6.54×10^{-4}	1.36×10^{-3}	2.13×10^{-3}
Total dose (mSv/hr)	0.128	0.149	0.158	0.163
Working hours (days)	391 (16)	335 (14)	317 (13)	307 (13)

Table III summarizes the calculation results of neutron and gamma doses in the spherical water shell as function of packing factor. The results are obtained with MCNP6. From this Table III, it is shown that an increase of packing factor leads to only a small increase of total radiation dose, the total radiation dose is mainly contributed from the neutron dose, and the total radiation dose for 70% packing factor is estimated to be 0.163mSv/hr. In Korea, the annual legal limit of radiation dose for workers is 50mSv/year and so the permissible working hour for 70% packing factor was estimated to be about 307 hours (i.e., ~13 days) for one year [6].

Next, the radiation shielding analysis is performed using MONACO/MAVRIC in SCALE 6.1 for comparison with those using MCNP6. For this calculation, the source intensities and spectra are the same as those used in the MCNP6 calculation. The radiation dose was converted from the radiation fluxes using the ICRU-57 dose conversion factors provided in SCALE6.1 [7]. Fig. 5 shows the geometric modeling of the RTPV device which was visualized using KENO3D in SCALE 6. Fig. 6 shows the total neutron dose distribution that was obtained with MONACO/ MAVRIC and the mesh file viewer.

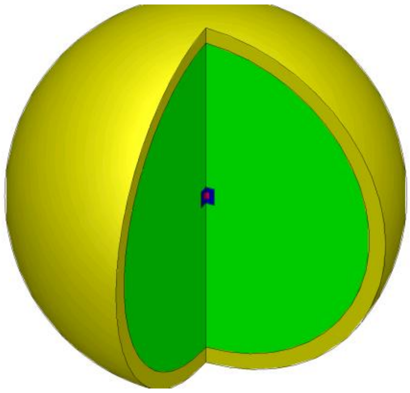


Fig. 5. Geometric model for the RTPV device shielding analysis using MONACO/MAVRIC

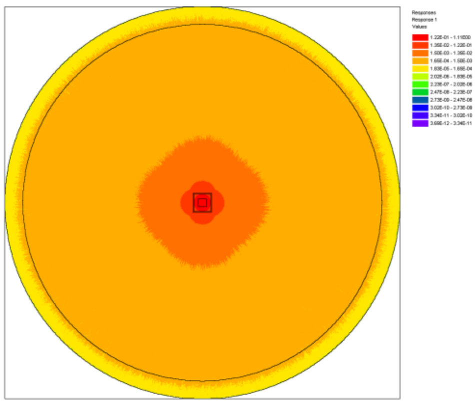


Fig. 6. Distribution of the total neutron dose (Sv/hr)

Table IV: Total fluxes in the water shell (#/cm^{2}sec)

Packing factor (%)	Neutron source				Photon source	
	Neutron flux		Photon flux		Photon flux	
	MCNP6	[b]MAVRIC	MCNP6	[b]MAVRIC	MCNP6	MAVRIC
10	6.54E+02	7.10E+02 ([a]7.9)	2.62E+02	2.82E+02 (7.2)	1.11E+01	9.62E+00 (13.2)
30	6.32E+02	6.80E+02 (7.2)	2.45E+02	2.63E+02 (6.6)	7.88E+01	8.50E+01 (7.4)
50	6.17E+02	6.64E+02 (7.0)	2.34E+02	2.51E+02 (6.7)	1.82E+02	1.87E+02 (2.7)
70	6.07E+02	6.52E+02 (6.9)	2.25E+02	2.41E+02 (6.5)	2.89E+02	2.98E+02 (2.9)

[a] Discrepancies (%) between the flux values of MCNP6 and MONACO/MAVRIC, [b]MONACO/MAVRIC

Table V: Doses in the water shell (Sv/hr)

Packing factor (%)	Neutron source				Photon source		Total dose value	
	Neutron dose		Photon dose		Photon dose			
	MCNP6	MAVRIC	MCNP6	MAVRIC	MCNP6	MAVRIC	MCNP6	MAVRIC
10	1.23E-04	1.16E-04 ([a]5.9)	4.91E-06	4.94E-06 (0.6)	9.06E-08	7.60E-08 (16.1)	1.28E-04	1.21E-04 (5.7)
30	1.44E-04	1.36E-04 (5.6)	4.58E-06	4.57E-06 (0.0)	6.54E-07	6.20E-07 (5.1)	1.49E-04	1.41E-04 (5.5)
50	1.52E-04	1.44E-04 (5.3)	4.38E-06	4.37E-06 (0.0)	1.36E-06	1.35E-06 (0.7)	1.58E-04	1.50E-04 (5.2)
70	1.57E-04	1.49E-04 (5.1)	4.23E-06	4.22E-06 (0.2)	2.13E-06	2.14E-06 (0.4)	1.63E-04	1.55E-04 (4.9)

[a] Discrepancies (%) between the dose values of MCNP6 and MONACO/MAVRIC

Table IV compares the total group integrated neutron and gamma (photon) fluxes calculated with MCNP6 and MONACO/MAVRIC in the water shell. As shown in Table IV, MONACO/MAVRIC gives similar results with MCNP6 within the discrepancies less than 8% except for the gamma flux for the 10% packing factor. Finally, Table V compares the neutron and gamma doses in the water shell. Table V shows that MONACO/MAVRIC and MCNP6 give good agreements in the radiation doses and specifically very small discrepancies between MONACO/MAVRIC and MCNP6 are observed in the gamma doses. The discrepancies in the total doses are less than 6.0%. In particular, it is shown that MCNP6 gives more conservative values of the radiation doses than MONACO/MAVRIC and the high packing factor of 70% gives higher total radiation dose by 27~28% than the small packing factor case of 10%.

3. Conclusions

In this work, the radiation shielding analysis for a RTPV device using $^{238}PuO_2$ was performed to estimate the radiation dose distribution and radiation doses in a specified tally zone. The radiation source intensities and spectra were evaluated with ORIGEN-S and the detailed shielding analyses were performed with MCNP6 and MONACO/MAVRIC (in SCALE 6.1). The analysis using MCNP6 showed that a worker under the regulation limit of 50mSv/year can be allowed for 307 hours and 391 hours with 70% and 10% packing factors, respectively in the region which is located at a 100cm distance from the center of the RTPV device. The comparative shielding analysis using MCNP6 and MONACO/MAVRIC (in SCALE 6.1) showed that MONACO/MAVRIC gives good agreements in the total radiation doses with MCNP6 and the discrepancies between the total doses estimated with these codes are less than 6%. So, MONACO/MAVRIC can be effectively used for shielding analyses for RTPV device with reduced computing times.

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