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ASSESSING SHIELDING THICKNESS IN AM-241 NUCLEAR BATTERY

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1. Introduction

Radioisotope Thermoelectric Generators (RTGs) are crucial power sources for a variety of space missions and remote terrestrial applications, utilizing heat from radioactive decay to generate electricity. Despite their exceptional reliability, the use of RTGs requires careful safety considerations due to the inherent risks of radiation exposure and the potential for accidents resulting in the release of radioactive material into the environment. [1].

Safeguarding the operation of RTG devices mandates a thorough comprehension of potential hazards and the implementation of robust mitigation strategies. The importance of safety analyses is to evaluate the dosimetric impact of potential accidents and incidents during RTG operations, alongside the development of stringent safety protocols and engineering controls [2].

Monte Carlo simulations are crucial for precise radiation dose estimation in RTG devices and similar applications. By modeling radiation transport accurately, including interactions with various materials and generating secondary particles, these simulations enable a thorough assessment of radiation fields and dose distributions within and around the device. They accommodate the complex geometries of RTG devices, accurately modeling their architecture and interaction with the environment. These simulations accurately represent the composition of materials within RTG devices, refining dose estimates by considering their specific radiation interaction properties. Moreover, Monte Carlo simulations incorporate variability and uncertainty into dose estimates, providing insights into potential dose ranges and associated uncertainties. Overall, integrating Monte Carlo simulation techniques enhances the accuracy and reliability of radiation dose estimation, ensuring the safe and effective utilization of RTG technology [1, 3].

2. Methodology

Ionizing radiations are governed by three key physical parameters: time, distance, and shielding. This means that the accumulated dose for an individual working in an area exposed to a specific dose rate is directly proportional to the time spent in that area and inversely proportional to the distance maintained from the radiation source, as well as the effectiveness of any shielding employed. In essence, dose control can be achieved by limiting exposure time, increasing distance from the source, and incorporating appropriate shielding materials. The selection of shielding materials depends on several factors including the type of radiation, the activity of the radiation source, and the acceptable dose rate outside the shielding material. When considering the type of radiation, specific criteria apply [4]. Shielding against both electrons and photons in radiation environments is essential for safety. Electrons, being charged particles, necessitate materials with low atomic number (Z) and structural integrity, like acrylic, Teflon, PVC, and polyethylene. They lose energy through collisions with the shielding material, restricting their travel distance. Effective shielding ensures that the electron's range is shorter than the absorbing material's thickness, containing its energy. On the other hand, photons require materials with high Z and high mass density, such as lead, concrete, tungsten, and iron. Photons may interact with the medium's atoms or pass through unchanged, making effective shielding crucial for reducing or absorbing photons as they pass through the material. The composition and thickness of the shielding material primarily determine its efficacy.

Americium-241 (Am-241) is a radioactive isotope utilized in Radioisotope Thermoelectric Generators (RTGs), where its decay emits alpha particles, convertible into heat through thermoelectric converters for electricity production. This feature ensures a stable power output over its approximately 432-year half-life, making it ideal for continuous or long-term power generation without frequent maintenance. Am-241 is readily available as a byproduct of nuclear reactions, particularly in spent nuclear fuel, making it cost-effective compared to isotopes with more complex acquisition processes. Its applications span space exploration missions by agencies like NASA, Russia, and ESA, powering spacecraft such as Voyager probes and ESA's Rosetta mission. Countries like Japan have also employed Am-241 in nuclear batteries for space exploration and other endeavors [1, 2, 5].

To evaluate the thermal output resulting from the decay of Am-241 within the battery core, the proposed configurations were subjected to simulation using the MCNP (Monte Carlo N-Particle) code, version 6.1. Developed in the 1970s by researchers at the Los Alamos National Laboratory (LANL), MCNP is distributed to LANL external users through the Radiation Safety Information Computational Center (RSICC). It comprises subroutines compliant with the ANSI Fortran 90 standard and is platform-independent. Although MCNP can transport various particles, in this study, only the decay products of Am-241, consisting of alpha particles and photons, were simulated [6].

For the Am-241 battery, the emission photon spectrum was obtained from the National Nuclear Data Center (NNCD). The spectrum shows peaks at various energy levels, with the most prominent peak occurring at 59.54 keV, where the emission intensity is highest at 35.90%. Other notable peaks include those at 26.34 keV with an emission intensity of 2.270% and at 33.19 keV with an intensity of 0.126%. Macroscopic quantities such as energy deposited within the battery, electron and photon fluxes were estimated using the MCNP6.1 code through statistical analysis of a sufficient number of histories. Photons and electrons were simulated using a cut-off energy of 1 keV in all simulation. The simulations conducted allowed for the estimation of the thermal powers emitted by the different evaluated geometric configurations. The results obtained in various regions of the battery, with statistical uncertainties below 0.1%. Figure 1 shows the simulated geometry.

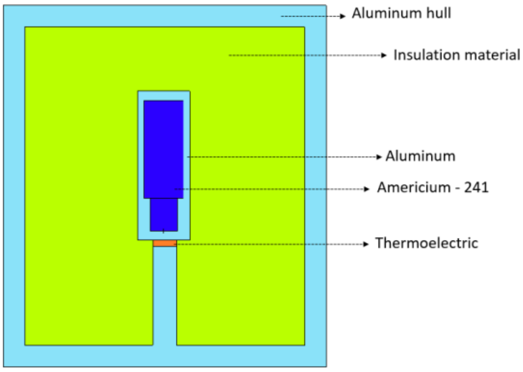

Figure 1: Geometry used in MCNP simulation.

3. Results and Discussion

An evaluation of shielding against radiation emitted by the radioactive core of a radioisotope thermoelectric generator (RTG) battery was conducted. The assessment revealed that electrons emitted have a short range, depositing energy locally within the core. These electrons interact with the core material, emitting bremsstrahlung radiation. Two stages of shielding were established: internal shielding within the RTG near the core and external shielding for transportation/assembly, both adhering to the ALARA principle. Materials such as stainless steel, lead, and tungsten were evaluated for internal shielding, with lead being deemed most effective due to its low melting point, despite tungsten and lead demonstrating lower external dose values. The external shielding around the battery aims to align the entire assembly with basic radiological protection guidelines. Therefore, the material and thickness of the shielding were calculated based on data obtained with the MCNP.6.1 code, aiming to meet limit values established by the NN-3.01 and CNEN NE 5.01 standards. For external shielding, in addition to tungsten, stainless steel, and lead were also evaluated due to their low cost, high ductility, and temperature variations, solely for comparison purposes.

Using the MCNP.6.1 code, we estimated the thickness of external shielding required around the battery based on photon spectra exiting the battery and mass attenuation coefficients of chosen materials. This shielding is crucial for limiting both the quantity and energy of photons exiting the battery. Tungsten, for example, effectively blocks photons with energies greater than 60 keV, as illustrated in Figure 2. By analyzing photon spectra emitted from the battery and mass attenuation coefficients, one can determine the thickness of external shielding needed to reduce radiation intensity to one-tenth of its initial value, known as the tenth-value layer (TVL). The required thicknesses for effective shielding of various materials surrounding the proposed Am-241 battery configuration are: stainless steel necessitates a recommended thickness of 0.6 mm, while lead and tungsten require 0.4 mm and 0.3 mm, respectively. These specifications are critical to attenuate radiation intensity to safe levels, ensuring the secure utilization of the Am-241 battery configurations. The maximum annual dose for an occupationally exposed individual (OEI) is 20 mSv. However, it is considered that an individual works up to 2000 hours per year (8 hours per day, 5 days a week, and 50 weeks per year) and assembles a maximum of one battery per year. Therefore, it is estimated that an exposed individual may be subjected to up to 0.01 mSv/h. Based on the maximum permitted dose for an OEI, one can determine the maximum thicknesses of the respective evaluated shielding. Among the materials evaluated, lead is commonly used for the transportation of radioactive material, resulting in external shielding around the battery to ensure compliance with current regulations.

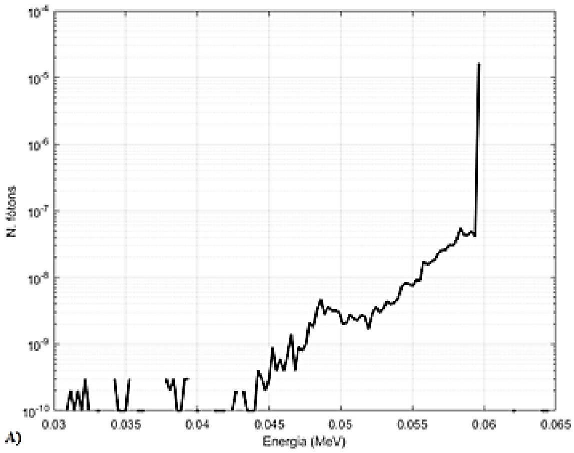

Figure 2: Photon spectra emitted from the proposed Am-241 battery (y axis is number of counts and x axis is Energy in MeV).

4. Conclusions

The evaluation of shielding for the proposed Am-241 battery involved an intricate analysis of electron and photon flow emitted by the radioactive core. Through this assessment, it was determined that while the electron range is limited within the battery core. As a result, radiation shielding was devised in two stages: internal shielding near the core and external shielding for transportation and assembly, adhering to the ALARA principle. Various materials such as stainless steel, lead, and tungsten were evaluated for their effectiveness in reducing external dose to the battery, with lead emerging as a more viable option for internal shielding due to its low melting point. The design of external shielding, based on MCNP.6.1 data and mass attenuation coefficients, aimed to meet established regulatory limits. In essence, this comprehensive evaluation underscores the importance of meticulous shielding design to mitigate radiation exposure, ensuring compliance with safety regulations in the transportation and assembly of radioactive materials.

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