Nondestructive Assay System for Radioactive Waste Steel Boxes Based on Tomographic Gamma Scanning Emission Measurement

Authors: Huang, Dr. Xi, yuan, Dr. kaiming, Dr. Hua Chen, Qu, Dr. Jin-Hui, YUAN, Yonggang, Tan, Dr. Zhaoyi, Tan, Dr. Zhaoyi

Date: 2025-02-25T00:00:00+00:00

Abstract

Nondestructive assay (NDA) plays a pivotal role in radioactive waste management. It can accurately evaluate the activity distribution of target nuclide without destroying the original form of the waste, which can provide the crucial basis for subsequent safe classification, proper storage, and scientific disposal. In this paper, a nondestructive assay prototype of radioactive waste steel box is introduced. Given the large size and thickness of the steel box, the emission measurement method in tomographic gamma scanning (TGS) is employed. The Monte Carlo method is utilized to achieve the passive efficiency calibration of the detection system. In the designed prototype, three high-purity germanium (HPGe) detectors mounted on the measurement support platform are used to efficiently measure the full-energy peak counting rate of target nuclides. Based on the Boosted-Gold algorithm, the reconstruction of the radioactive waste activity distribution in steel box is accomplished. The errors of the actual activity measurement are less than 50% and 80% in uniform medium (air, water and sand) and nonuniform medium (aluminum parts), respectively. Moreover, the reconstructed activity distribution shows a high degree of consistency with the actual activity distribution. These results validate the capability of the proposed prototype and can fully meet the requirements of nondestructive assay of radioactive waste steel box.

Full Text

Preamble

Nondestructive Assay System for Radioactive Waste Steel Boxes Based on Tomographic Gamma Scanning Emission Measurement

Xi Huang[1], Kaiming Yuan[2], Hua Chen[1], Jinhui Qu[3], Yonggang Yuan[1],†, and Zhaoyi Tan[1],‡
[1]Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics, Mianyang, Sichuan 621000, China
[2]Northwestern Polytechnical University, Xi' an, Shaanxi 710129, China
[3]East China University of Technology, Nanchang, Jiangxi 330014, China

Nondestructive assay (NDA) plays a pivotal role in radioactive waste management, enabling accurate evaluation of target nuclide activity distributions without altering the waste' s original form. This capability provides crucial basis for subsequent safe classification, proper storage, and scientific disposal. This paper introduces a nondestructive assay prototype system for radioactive waste steel boxes. Given the large dimensions and thickness of steel boxes, the emission measurement method from tomographic gamma scanning (TGS) is employed. The Monte Carlo method is utilized to achieve passive efficiency calibration of the detection system. The designed prototype uses three high-purity germanium (HPGe) detectors mounted on a measurement support platform to efficiently measure the full-energy peak counting rates of target nuclides. Based on the Boosted-Gold algorithm, the reconstruction of radioactive waste activity distribution within steel boxes is accomplished. For uniform media (air, water, and sand), the errors in actual activity measurement are less than 50%, while for nonuniform media (aluminum components), errors are less than 80%. Moreover, the reconstructed activity distribution shows high consistency with the actual distribution. These results validate the capability of the proposed prototype and demonstrate that it fully meets the requirements for nondestructive assay of radioactive waste steel boxes.

Keywords: Nondestructive assay, tomographic gamma scanning, radioactive waste steel box

Introduction

Nuclear energy has served as a clean substitute for fossil fuels, meeting approximately 11% to 20% of global energy demands [1, 2]. However, nuclear energy development generates copious amounts of radioactive waste through decommissioned nuclear facilities as well as from the production, processing, storage, transportation, and utilization of nuclear materials [3, 4]. Improper handling of radioactive waste constitutes a critical environmental health concern [5]. Prior to final disposal, precise identification of contained nuclides and quantitative measurement of their activity distribution are critical safety prerequisites. While conventional destructive chemical analysis methods rely on representative sampling followed by laboratory examination [6–8], current nondestructive assay primarily employs two gamma-ray spectroscopic approaches: Segmented Gamma Scanning (SGS) and Tomographic Gamma Scanning (TGS).

SGS is mainly used for quantitative measurement of radionuclides and their content in unevenly distributed, low-density media [9, 10]. It scans samples through axial segmentation and radial uniform rotation, obtaining the linear attenuation coefficient of each segmented sample medium via transmission measurement [11, 12]. The γ attenuation of the measured sample is then corrected using this linear attenuation coefficient. The accuracy of SGS measurement results depends on the attenuation constant of the radioactive medium, the volume of the radioactive sample, and the detector locations. To reduce measurement error and improve detection accuracy, several improvement methods based on traditional SGS have been proposed [13, 14]. However, SGS remains unsuitable for radioactive samples in non-drum packaging and high-density media cases.

To overcome these limitations, TGS prototypes have been successfully developed [15–17]. TGS represents an improvement and extension of SGS, scanning target objects in three dimensions rather than two. In addition to axial segmentation scanning, TGS performs translation and rotation scanning for each layer [18–20]. In TGS, the distribution of linear attenuation coefficients is obtained through transmission measurements. The actual distribution of nuclide activity in the sample can then be determined by using these coefficients to correct the activity distribution measured via emission measurement. Combining TGS with the virtual point detector method has verified the accuracy of TGS measurement results, showing good agreement between actual and TGS-measured activity distributions [21, 22]. Based on simulated detection systems for radioactive waste drums, models for estimating voxel effects in TGS imaging technology have been presented, confirming the possibility of measuring non-uniform medium waste drums using TGS technology [23–25].

Currently, research on nondestructive assay of radioactive distribution in nuclear waste drums focuses primarily on TGS technology improvements [26–29]. A novel reconstruction algorithm for transmission image reconstruction in tomographic gamma scanning has been proposed based on the conventional transmission equation and equivalent gamma-ray track length modified by Monte Carlo methods, with implementation through simulation [30]. The Neighborhood Homogeneous Measurement (NHM) method has been proposed to process salt-pepper noise in TGS image systems, avoiding shortcomings of traditional median filtering algorithms [31]. A new algorithm based on total variation minimization reconstructs voxel transmission images in greater detail with the same number of measurements, precisely describing the attenuation process. Smaller voxel size decreases the effect of inhomogeneity and improves accuracy [32]. Furthermore, to save time, small amounts of data are measured by dividing the drum into several large voxels, which leads to inaccurate TGS images. Therefore, an improved algebraic reconstruction technique (IART) has been proposed, applying total variation minimization and a self-adaptive relaxation factor to improve the iterative process of traditional algebraic reconstruction technique (ART) [33]. Additionally, an improved NMO-OSEM (Non-minimization optimization OSEM) method has been designed as an iterative algorithm with corrected initial values optimized by non-minimization optimization methods [34].

Traditional reconstruction algorithms require the same amount of projection data as the pixel values of the transmission image, necessitating long scan times and severely limiting industrial TGS applications. Recent methods combine reconstruction algorithms with artificial intelligence to optimize TGS measurement processes and accuracy. Sparse angle scanning improves TGS system efficiency, but the limited data generated makes traditional algorithms difficult to apply, leading to blurs and artifacts in reconstructed images. An algebraic reconstruction technique (ART) combined with Residual Network (ResNet) has been designed to reconstruct TGS transmission images [35]. Moreover, when measuring a waste drum with uniformly distributed medium, the counting rates from three detectors at different positions can be input into a trained neural network to directly output the equivalent annular source, ultimately achieving accurate reconstruction of total radionuclide activity inside the waste drum [36].

However, for actual nondestructive measurement of radioactive waste packaged in steel boxes, the size and thickness are much larger than those of steel drums, making transmission measurement in TGS difficult. In previous work, only one case' s radioactivity was reported, the steel box was assumed to be a special "image" divided into 2$×5×$3 voxels, and reconstruction was performed based on the simulated efficiency matrix and experimentally measured characteristic γ-ray counting rates [37]. However, this voxel division was coarse and the experimental verification method was simplistic, preventing adequate explanation of the method' s performance and reliability.

Therefore, this paper introduces a nondestructive assay system prototype for radioactive waste steel boxes. Considering the large size and thickness of steel boxes, the emission measurement method from tomographic gamma scanning (TGS) is utilized. The Monte Carlo method is used to achieve passive efficiency calibration of the detection system. The prototype employs three high-purity germanium (HPGe) detectors mounted on an automatic carrier platform to efficiently measure the full-energy peak counting rates of target nuclides. Based on the Boosted-Gold algorithm, reconstruction of radioactive waste activity distribution within steel boxes is completed. Actual activity measurement tests using both uniform and nonuniform media are carried out to verify the capability of the proposed prototype.

II. Method

As illustrated in Fig. 1 [Figure 1: see original paper], the target steel box model, with actual dimensions of 1573 mm × 1565 mm × 1331 mm, is divided into three equal layers. Each layer is further subdivided into 16 voxels, resulting in a total box model composed of 48 voxels. Voxel labeling follows a sequential three-dimensional ordering convention: left-to-right in the horizontal plane, inferior-to-superior in the vertical dimension, and anterior-to-posterior in the depth direction, as graphically represented in Fig. 1 [Figure 1: see original paper]. The designed nondestructive assay system can obtain the activity of target nuclides in each voxel through nondestructive measurement around the steel box, for both point source and volume source cases.

The system diagram is shown in Fig. 2 [Figure 2: see original paper]. For the primarily considered high-energy gamma nuclides (^{60}Co, ^{152}Eu) in radioactive waste, three HPGe detectors are used to simultaneously measure gamma spectra. The detector group comprises three vertically arranged detectors. Activity reconstruction is performed based on the full-energy peak counting rate induced by the highest-energy gamma photon emitted from the radioactive waste. By partitioning the steel box as depicted in Fig. 1 [Figure 1: see original paper], each side is subsequently divided into 3 layers and 12 two-dimensional regions, with the center point of each region designated as a spectrum measurement point.

The detector array incorporates a lead collimator with three square apertures, enabling precise three-layer radiation measurements. To minimize voxel interference during collimator design, critical parameters including aperture geometry, collimation depth, and shielding thickness must be optimized following distance determination. This optimization ensures spatial alignment between each detector’ s field-of-view and its corresponding two-dimensional region on the steel box side, while effectively suppressing background noise and cross-talk from adjacent voxels. By conducting four measurements at designated positions, the system achieves efficient full-coverage measurements for a single side. As depicted in Fig. 3 [Figure 3: see original paper], a total of 16 measurement points are arranged, with each sampling location visually indicated by gray dots.

The measurement procedure involves sequential scanning of the steel box through rotational positioning. Upon completing measurements on one side, the detector array retracts to allow a 90-degree rotation of the steel box, after which scanning proceeds to the adjacent side. The entire procedure requires 16 measurements with three intermediate rotations, ultimately achieving comprehensive coverage of all four sides. Twelve energy spectra from each side are spatially organized in left-to-right and bottom-to-top configuration, with the four surfaces numbered counterclockwise from the initial scanning face. Notably, the sequential arrangement of these 48 measurement points (distributed across 48 two-dimensional regions on 4 sides) differs from the three-dimensional voxel arrangement of the steel box. Figure 3 [Figure 3: see original paper] illustrates the vertical measurement positions with gray dots, where the leftmost number corresponds to the lowest detection point and the rightmost to the uppermost one. Subsequent analysis focuses on quantifying the full-energy peak counting rate contributions from these 48 measured spectra to the steel box’ s radioactive activity distribution.

For TGS, transmission measurement is performed in advance. The gamma-ray emitted by the transmission source passing through a material follows Beer’s law, which satisfies Nk = N0e (cid:80)I i=1 −µixki, where Nk indicates the full-energy peak counting rate in the k-th transmission measurement and N0 indicates the full-energy peak counting rate of a certain energy, which is attenuated by air only, µi is the linear attenuation coefficient of the i-th voxel, j [1, 48], xki is the transmitting track length of the gamma-ray passing through the i-th voxel when the detector array is at the k-th transmission measurement. Each voxel is assumed to be uniform in both transmission and emission measurements. Let ck = − ln(Nk/N0), from (1) one can have (cid:88) −µixki = ck, where ck and xki can be measured and calculated, respectively. By a statistical iterative method, the linear attenuation coefficient matrix can be obtained. Then the gamma-ray emitted by the radioactive waste in the steel box passing through the medium is considered in the emission measurement.

Nk = α (cid:88) AnEne (cid:80)I i=1 −µixn where Nk indicates the full-energy peak counting rate in the k-th emission measurement. α is the branching ratio of the gamma ray, An is the activity of the concerned nuclide in the n-th voxel, En is the detection efficiency when a gamma photon with certain energy is emitted from the n-th voxel without attenuation. µi is the linear attenuation coefficient of the i-th voxel which is obtained from the transmission measurement. ki is the transmitting track length of the gamma-ray emitted in the n-th voxels passing through the i-th voxel when the k-th measurement is performed. Under the uniform medium conditions, one can have Nk = α (cid:88) AnEneµxn let Rkn = αEneµxn k, (4) can be converted into (cid:88) RknAn, and its matrix form can be obtained as (60Co:1332 keV, 152Eu: 1408 keV) which are emitted during the decay of a nuclide atom.

As can be seen from Fig. 4 Figure 4: see original paper, when a gamma photon is emitted from the first voxel, the full-energy peak counting rates of the first measurement point and the 40th measurement point both exhibit maximum values, since these two measurement points are equidistant and closest to the emitting voxel. The full-energy peak counting rates of the energy spectra at the 13th and 28th measurement points both show local maximum values, since these two measurement points, which can cause detector response under the presence of the lead collimator, have the local minimum distance from the emitting voxel. The local minimum distance is the width of three voxels. Figure 4(d) shows a similar situation, in which the gamma photon is emitted from the 48th voxel. In Fig. 4(c), gamma photons are emitted from the 24th voxel. The full-energy peak counting rates of the energy spectra decrease from the nearest 22nd measurement point to the 12th, 33rd, and 47th measurement points, which are 0, 1, 2, and 3 voxels away from the emitting voxel, respectively.

With the response function matrix, the Boosted-Gold algorithm is adopted to perform activity reconstruction by obtaining the least squares solution of (6) [38], which can be defined as Ak+1 = Ak + δk(RT N − RT RAk), where δk is an introduced relaxation coefficient vector, whose element δk i is defined as N = R · A, l=1(RT R)ilAk (cid:80)N where N is the full-energy peak counting rate vector, R is the system response function matrix, and A is the activity vector of the 48 voxels.

The matrix R contains detection efficiency parameters and the linear attenuation coefficient, which can be obtained by passive calibration methods and transmission measurement, respectively. However, due to the large size of the steel box, calibration sources and transmission sources with high activity would be needed, posing significant challenges for radiation safety and protection. Therefore, the passive calibration method is adopted. The system response function matrix R is obtained through Monte Carlo software. Response function matrix libraries for volume sources and point sources have been computed separately. Each library contains multiple response function matrices corresponding to material densities ranging from that of air (0.001 g/cm^3) to aluminum (2.7 g/cm^3), with a density increment of 0.3 g/cm^3 between successive matrices. Figure 4 [Figure 4: see original paper] shows an example response function matrix for a point source in air. Fig. 4(a) plots the entire response function matrix as a bar chart. Each column in Fig. 4(a), corresponding to a voxel number, shows the expected contribution of gamma rays emitted from that voxel to the full-energy peak counting rates in the 48 measured energy spectra. The full-energy peak counting rates are obtained from characteristic energy gamma-rays where (RT R)il indicates the element in the i-th row, j-th column of the matrix RT R, N is the number of response functions that constitute the response matrix. With (7) and (8), the iterative formula can be derived as (RT N)Ak l=1(RT R)ilAk (cid:80)n After a given number of iterations, shown in (9), a nonlinear enhancement process is performed, which is given as follows: i = (Ak i)p, where p > 1 is the power exponential enhancement coefficient. The enhancement result is taken as the initial value of (9). After a given number of enhancements, the solution that satisfies the convergence condition or the iteration and enhancement number conditions is obtained. The convergence condition is shown below ||Ak+1 − Ak|| ||Ak|| where Ak is the result of the k-th iteration, and Ak+1 is the k+1-th iteration, is the given iteration residual precision. || · || indicates the Euclidean norm.

III. Experiment

The algorithm iteration process can be systematically described through four sequential stages. In the initialization phase, critical parameters including the initial matrix Ak, iteration threshold k, enhancement threshold K, and termination criterion are established. The process then executes two key operations: (1) applying the recursive formula (9) for matrix updating, and (2) evaluating the convergence condition (11). When the enhancement condition is triggered, the algorithm implements an enhancement mechanism by adjusting factor p to accelerate convergence. This process continues until the termination criterion or iteration threshold is satisfied, ultimately yielding reconstructed activity distributions and total activity of radioactive waste within the steel box.

In the nondestructive assay of radioactive waste activity within steel box packaging, gamma nuclides with high energy and long half-life are the primary concern. Therefore, response function matrices for the highest-energy gamma photons of ^{60}Co and ^{152}Eu are calculated in this paper for both point source and volume source cases. Based on these response function matrices, simulation experiments and actual experiments in uniform and non-uniform media were carried out. In these experiments, the maximum-energy full-energy peak counting rates of 48 measurement points were obtained through Monte Carlo simulation or actual measurement. Using the method designed in Section II, the reconstructed total activity and activity distribution of the steel box are determined, and relative errors are evaluated.

A. Simulation Experiment

To verify the effectiveness and robustness of the designed method, a simulation experiment was conducted. For the 48 voxels in the steel box model, pulse, linear, and random distributions of radioactive waste cases were simulated (^{60}Co@1332 keV). The full-energy peak counting rates at 48 measurement points for different distributions were simulated using Monte Carlo software. Gaussian white noise was added to the simulated vector to characterize influences from Monte Carlo model errors and full-energy peak counting rate calculation algorithms for actual measured spectra. The noise data was input into the algorithm, and reconstructed results were calculated and compared with actual simulation data. The reconstructed activity distributions for the three cases are shown in Fig. 5 [Figure 5: see original paper]. The simulated total activities, reconstructed total activities, and relative errors are shown in Table 1 .

In the pulse distribution, illustrated in Fig. 5(a), the activity of the 21st voxel was set as 1.00$\times 10^{9}$ Bq. Reconstruction results, including the relative error between simulated and reconstructed total activities and the reconstructed distribution, show good conformance. In the linear distribution, illustrated in Fig. 5(b), activities of the first to 48th voxels were set to increase linearly from 1.00$\times 10^{9}$ Bq to 4.80$\times 10^{10}$ Bq. The reconstructed activity distribution shows a linear distribution with certain fluctuations, and the reconstructed total activity exhibits low relative error. In the random distribution, illustrated in Fig. 5(c), activities of the 2nd, 9th, 21st, 27th, 28th, and 32nd voxels were set to 3.00$\times 10^{9}$ Bq, 5.00$\times 10^{9}$ Bq, 1.00$\times 10^{9}$ Bq, 4.00$\times 10^{9}$ Bq, 6.00$\times 10^{10}$ Bq, and 8.00$\times 10^{10}$ Bq, respectively.

B. Actual Experiment Under Uniform Medium

The self-developed nondestructive assay system prototype is shown in Fig. 6 [Figure 6: see original paper], consisting of three main parts: a radioactive waste steel box, a mechanical control platform, and a detection module. The steel box, marked (1) in Fig. 6, is a FA-IV steel box measuring 1573 mm × 1565 mm × 1331 mm.

The mechanical control platform, marked (3) in Fig. 6, consists of a steel box support platform and a measurement support platform. The steel box support platform measures 1600 mm × 1600 mm × 830 mm and is primarily used for 360° rotation and weighing of the steel box. Four weighing meters are installed at the platform' s four corners, with an industrial computer collecting the actual steel box weight. By subtracting the steel box weight from collected data, density information of the radioactive waste can be obtained. The steel box can be transported to the support platform via forklift. The support platform is equipped with four protruding snap devices to ensure box stability during rotation. It rotates at 2 degrees per second with a maximum weighing capacity of 10 tons.

The measurement support platform has three degrees of freedom (X, Y, and Z axes), constructed by integrating three individual platforms corresponding to each axis. The overall platform dimensions are 2190 mm × 3660 mm × 2280 mm.

The detection module, marked (2) in Fig. 6, is precisely positioned at the Z platform. This module comprises a detector array for measuring gamma spectra and a shield module designed in Section II to protect against external interference from other voxels. The distance between the steel box and detector front ends is set at 40.6 cm, and the collimator hole side length in front of the detectors is 8.3 cm to ensure axial coverage of one voxel side.

Based on this measurement system, experimental validation utilized three uniform media (air, water, and sand) to assess system performance. ^{60}Co (2.22$×10^{7}$ Bq) and ^{152}Eu (5.17$×10^{7}$ Bq) sources were used for activity reconstruction tests. In the sand medium case, two additional sources with different activities (^{60}Co at 4.32$×10^{7}$ Bq and ^{152}Eu at 1.02$×10^{8}$ Bq) were used to further verify system effectiveness. To rigorously evaluate robustness, three distinct source configurations were implemented: (1) solitary ^{60}Co placement, (2) solitary ^{152}Eu placement, and (3) simultaneous dual-source deployment.

Figure 7 [Figure 7: see original paper] shows the experimental site and reconstructed activity distributions under different sources and media. Figures 7(a), 7(e), and 7(i) show experimental sites under air, water, and sand media, respectively. In air and water media, the radioactive source was placed at the voxel center by hanging from the steel box top. In sand medium, the source was inserted into the sand using an aluminum rod. Figures 7(b), 7(c), and 7(d) show reconstructed activity distributions when single ^{60}Co source, single ^{152}Eu source, and both sources were placed in the steel box under air medium. Red indicates ^{60}Co activity, blue indicates ^{152}Eu activity, and voxel color depth is proportional to activity. Figures 7(f)-7(h) and 7(j)-7(n) follow the same principle. Actual voxel placement of radioactive sources in each case is shown in Table 2 . To verify method reliability, two additional tests were conducted by changing source activities in Figs. 7(m) and 7(n), with ^{60}Co and ^{152}Eu activities changed to 1.02$×10^{8}$ Bq and 4.32$×10^{7}$ Bq, respectively. Table 2 shows measured medium density, radioactive source locations, reconstructed activity, standard activity, and relative errors. In the three media, the first two rows are single-source cases, and the remaining row is the dual-source case. Relative error ranges are -27.3% to 26.6% for air, -37.5% to 48.5% for water, and -45.6% to 32% for fine sand. Maximum values of reconstructed nuclide activity distribution are consistent with actual source placement voxels, showing significant differences from other voxels.

C. Actual Experiment Under Nonuniform Medium

The first two subsections demonstrated successful reconstruction of radioactive waste activity in steel boxes using simulated and measured data under uniform medium conditions. However, achieving perfectly uniform media is impractical in real steel box applications even with meticulous arrangement. To address this realistic challenge, irregularly shaped aluminum components were used to simulate nonuniform measurement media within the steel box, enabling comprehensive evaluation of the proposed algorithm’ s stability and robustness in nonuniform environments.

Figure 8 [Figure 8: see original paper] shows the experimental site and reconstructed activity distribution under aluminum medium, where the radioactive source was inserted into the aluminum by mounting an aluminum rod. Figures 8(b), 8(c), and 8(d) demonstrate reconstructed activity distributions for single ^{60}Co source, single ^{152}Eu source, and both sources placed in the steel box, following the same format as Fig. 7. Table 3 follows the same format as Table 2. The relative error range for aluminum medium is -12.4% to 73.9%. Maximum activity voxels in the reconstructed distribution are consistent with actual source placement voxels, though differences with adjacent voxels are less significant, as shown in Fig. 8(d).

Based on the aforementioned experiments under nonuniform and uniform media, Fig. 9 [Figure 9: see original paper] provides comparison results between response functions and measured full-energy peak counting rates after normalization for four representative cases. Figure 9(a) illustrates the full-energy peak counting rate vector and corresponding response function for the ^{60}Co source in Fig. 7(b). Figures 9(b), 9(c), and 9(d) were obtained with the ^{152}Eu source in Fig. 7(g), ^{152}Eu source in Fig. 7(c), and ^{60}Co source in Fig. 8(d), respectively.

IV. Results and Discussion

Section III utilized Monte Carlo models to calculate full-energy peak counting rate vectors under three given radioactive source distributions, with Gaussian noise superimposed based on counting rates. Simulation results showed that activity reconstruction errors were less than 5% after noise superposition, fully demonstrating the reconstruction algorithm’ s robustness and stability. Relative errors in analytical outcomes primarily derive from two factors: (1) preconfigured noise levels in vectors, and (2) intrinsic limitations of the analytical algorithm’ s mathematical approximations.

In actual experiments under uniform media, single and dual-source tests were conducted in air, water, and sand media. Due to the hanging basket inside the steel box and errors in medium loading and sizing, water and fine sand medium densities are smaller than standard values. Various source locations were attempted, including central and boundary voxels. Reconstruction results show that errors when sources are placed in central voxels are generally smaller than those in boundary voxels, primarily due to the hanging basket and steel box modeling errors.

The activity assay error range is -19.9% to 28.8% in air medium, -37.5% to -41.9% in water medium, -45.6% to 32% in fine sand medium, and -12.4% to 73.9% in aluminum medium, all less than 100%. In uniform media, relative error values primarily originate from geometric position errors of radioactive sources, analytical algorithm errors, differences between theoretically calculated response functions and actual measurements, and errors in energy spectrum characteristic peak area calculations.

In air medium, reconstruction noise is more pronounced for two reasons. First, large source activity results in significant detector dead time (especially in boundary voxels), which still introduces errors in full-energy peak counting rates even after calibration. Second, a black hanging basket installed in the steel box for installation convenience was not modeled in the Monte Carlo simulation. The density gap is relatively small in water and sand media, making the impact less obvious. Meanwhile, in air medium, the radioactive source is suspended by iron wire, which shakes during support platform movement and steel box rotation, causing the full-energy peak counting rate to follow Monte Carlo calculation trends but with significantly different ratios between detection points, as shown in Fig. 9(a).

From Fig. 9(b), in water medium, the actual full-energy peak counting rate is closest to Monte Carlo simulation results, primarily because the steel box Monte Carlo model in water medium most closely matches the actual model. In sand medium, bagged sand creates gaps between bags and water content differences, making the medium less uniform and resulting in greater noise in inversion results. Meanwhile, water medium is denser than air medium, and the iron wire suspension provides more stability during steel box measurement and rotation. This makes the full-energy peak counting rate agree with Monte Carlo simulation results in both trend and proportion, as shown in Fig. 9(b).

In the final medium, irregular aluminum components commonly found in radioactive waste were used to simulate nonuniform media. Compared with cement-solidified waste packages, this method creates greater nonuniformity. Activity reconstruction results show a maximum relative error of 73.9%, representing a performance difference compared with uniform media. The main reason is that measured full-energy peak counting rates differ significantly from response functions. For example, in Fig. 9(d), counting rate data at the 17th and 18th measurement points differ considerably, resulting in two voxels with similar activity in Fig. 8(d). However, it is noteworthy that even in this extremely heterogeneous medium, total activity errors do not exceed an order of magnitude, demonstrating system robustness.

Additionally, experimental results across all media indicate no significant difference between measurement errors for single and dual sources. Two main reasons account for this: first, the designed method utilizes only the full-energy peak counting rate of the maximum gamma energy for target radionuclides, rather than full-spectrum counting rates. Second, the prototype is equipped with three HPGe detectors with excellent energy resolution, ensuring that count results for full-energy peaks of different energies do not interfere. Thus, simultaneous measurement of multiple radionuclide activities is achieved.

V. Summary

This paper introduced a nondestructive assay system prototype for radioactive waste steel boxes employing the emission measurement method from tomographic gamma scanning (TGS). The Monte Carlo method was used to achieve passive efficiency calibration of the detection system. The prototype utilized three HPGe detectors to efficiently measure full-energy peak counting rates of target nuclides. Based on the Boosted-Gold algorithm, reconstruction of radioactive waste activity distribution within steel boxes was completed. The system' s actual activity measurement error is less than 50% in uniform media (air, water, and sand) and less than 80% in non-uniform media (aluminum components). Moreover, the reconstructed activity distribution shows high consistency with the actual distribution. These results verify the capability of the proposed prototype and demonstrate that it fully meets requirements for nondestructive assay of radioactive waste steel boxes.

References

[1] H. S. Kim, Comparison of cost efficiencies of nuclear power and renewable energy generation in mitigating CO2 emissions. Environ. Sci. Pollut. Res. 28, 789-795 (2021). doi:10.1007/s11356-020-10537-1

[2] N. K. Katiyar, S. Goel, Recent progress and perspective on batteries made from nuclear waste. Nucl. Sci. Tech. 34, 33 (2023). doi:10.1007/s41365-023-01189-0

[3] G. A. Wen, J. H. Wu, C. Y. Zou, et al., Preliminary safety analysis for a heavy water-moderated molten salt reactor. Nucl. Sci. Tech. 35, 106 (2024). doi:10.1007/s41365-024-01476-4

[4] D. Deng, L. Zhang, M. Dong, et al., Radioactive waste: A review. Water Environ. Res. 92, 1818-1825 (2020). doi:10.1002/wer.1442

[5] S. A. Darda, H. A. Gabbar, V. Damideh, et al., A comprehensive review on radioactive waste cycle from generation to disposal. J. Radioanal. Nucl. Chem. 329, 15-31 (2021). doi:10.1007/s10967-021-07764-2

[6] Y. W. Chen, J. L. Wang, Removal of cesium from radioactive wastewater using magnetic chitosan beads cross-linked with glutaraldehyde. Nucl. Sci. Tech. 27, 43 (2016). doi:10.1007/s41365-016-0033-6

[7] M. F. Attallah, S. E. Rizk, S. A. Shady, Separation of 152+154Eu, 90Sr from radioactive waste effluent using liquid-liquid extraction by polyglycerol phthalate. Nucl. Sci. Tech. 29, 84 (2018). doi:10.1007/s41365-018-0423-z

[8] A. O. Adeola, K. O. Iwuozor, K. G. Akpomie, et al., Advances in the management of radioactive wastes and radionuclide contamination in environmental compartments: a review. Environ. Geochem. Health. 45, 2663-2689 (2023). doi: 10.1007/s10653-022-01378-7

[9] J.W. Zhao, W.N. Su, An iterative image reconstruction algorithm for SPECT. Nucl. Sci. Tech. 25, 030302 (2014). doi: 10.13538/j.1001-8042/nst.25.030302

[10] J. K. Sprinkle, S. T. Hsue, Recent advances in segmented gamma scanner assay. Trans. Am. Nucl. Soc. 55 (1987).

[11] E. Laloy, B. Rogiers, A. Bielen, et al., Bayesian inference of 1D activity profiles from segmented gamma scanning of a heterogeneous radioactive waste drum. Appl. Radiat. Isot. 175, 109803 (2021). doi: 10.1016/j.apradiso.2021.109803

[12] H. Zheng, X. Tuo, W. Zhao, et al., An efficiency function model of segmented gamma scanning for measuring radioactive waste drum. Appl. Radiat. Isot. 199, 110895 (2023). doi: 10.1016/j.apradiso.2023.110895

[13] S. Patra, C. Agarwal, S.Chaudhury, Full energy peak efficiency calibration for the assay of large volume radioactive waste drums in a segmented gamma scanner. Appl. Radiat. Isot. 144, 80-86 (2019). doi: 10.1016/j.apradiso.2018.11.014

[14] Y. F. Bai, E. Mauerhofer, D.Z. Wang et al., An improved method for the non-destructive characterization of radioactive waste by gamma scanning. Appl. Radiat. Isot. 67, 1897-1903 (2009). doi: 10.1016/j.apradiso.2009.05.017

[15] T. Q. Dung, N. D. Thanh, L. A. Tuyen, et al., Evaluation of a gamma technique for the assay of radioactive waste drums using two measurements from opposing directions. Appl. Radiat. Isot. 67, 164-169 (2009). doi: 10.1016/j.apradiso.2008.08.008

[16] T. Roy, J. Ratheesh, A. Sinha, Three-dimensional SPECT imaging with LaBr3: Ce scintillator for characterization of nuclear waste. Nucl. Instrum. Methods Phys. Res., Sect. A. 735, 1-6 (2014). doi: 10.1016/j.nima.2013.08.037

[17] T. Roy, M. R. More, J. Ratheesh, et al., Active and passive CT for waste assay using LaBr3 (Ce) detector. Radiat. Phys. Chem. 130, 29-34 (2017). doi: 10.1016/j.radphyschem.2016.07

[18] T. Q. Dung, A simple gamma technique for the assay of radioactive waste drums. Int. J. Nucl. Energy Sci. Technol. 5, 290-297 (2010). doi: 10.1504/ijnest.2010.035539

[19] R. J. Estep, T. H. Prettyman, G. A.Sheppard, Tomographic gamma scanning to assay heterogeneous radioactive waste. Nucl. Sci. Eng. 118, 145-152 (1994). doi: 10.13182/NSE94-A19380

[20] R. Venkataraman, M. Villani, S. Croft, et al., An integrated Tomographic Gamma Scanning system for non-destructive assay of radioactive waste. Nucl. Instrum. Methods Phys. Res., Sect. A. 579, 375-379 (2007). doi: 10.1016/j.nima.2007.04.125

[21] H. D. Chuong, T. M. Sang, L. G. Thien, et al., Combination of the gamma emission tomography and the virtual point-detector methods for calculating radioactive waste drum by non-destructive assays. Sci. Tech. Develop. J. Nat. Sci. 2, 214-221 (2018). doi: 10.32508/stdjns.v2i6.884

[22] J. Suran, P. Kovar, J. Smoldasova, et al.. New high-throughput measurement systems for radioactive wastes segregation and free release. Appl. Radiat. Isot. 130, 252-259 (2017). doi: 10.1016/j.apradiso.2017.09.043

[23] T. Q. Dung. Estimation of systematic error of the gamma tomography for correction of self-attenuation in assay of radioactive waste. 47, 155 (2013). doi: 10.54607/hcmue.js.0.47.891

[24] D. C. Camp, H. E. Martz, G. P. Roberson, et al., Nondestructive waste-drum assay for transuranic content by gamma-ray active and passive computed tomography. Nucl. Instrum. Methods Phys. Res., Sect. A. 495, 69-83 (2002). doi: 10.1016/S0168-9002(02)01315-3

[25] S.Tortajada, F. Albiol, L. Caballero, et al., A portable geometry-independent tomographic system for gamma-ray, a next generation of nuclear waste characterization. Sci. Rep. 13, 12284 (2023). doi: 10.1038/s41598-023-39405-x

[26] Y. Wang, Y. Liu, X. Yao, et al., Design of segmented gamma scanning and tomographic gamma scanning automatic detection system for barreled nuclear waste. J. Radiat. Res. Radiat. Proc. 37, 61-66 (2019). (in Chinese)

[27] H. Yang, X. Y. Zhang, W. G. Gu, et al., A novel method for gamma spectrum analysis of low-level and intermediate-level radioactive waste. Nucl. Sci. Tech. 34, 87 (2023). doi: 10.1007/s41365-023-01236-w

[28] Y. C. Yan, M. Z. Liu, X. Y. Li, et al., Improved Cohen-Sutherland algorithm for TGS transmission imaging. Nucl. Sci. Tech. 34, 98 (2023). doi: 10.1007/s41365-023-01238-8

[29] M. Belzunce, C. Verrastro, L. M. Garbino, et al., An attenuated projector for iterative reconstruction algorithm of a high sensitivity tomographic gamma scanner. IEEE Trans. Nucl. Sci. 61, 975-984 (2014). doi: 10.1109/TNS.2014.2302454

[30] Q. H. Zhang, W. H. Hui, D. Wang, et al., A novel algorithm for transmission image reconstruction of tomographic gamma scanners. Nuc. Sci. Tech. 21, 177-181 (2010). doi: 10.13538/j.1001-8042/nst.21.177-181

[31] K. X. Peng, X. G. Tuo, B. Cai, et al., An Algorithm of Reduction Salt-Pepper Noise in TGS Images. 2012 International Conference on Computer

Science and Electronics Engineering. IEEE, 416-419 (2012). doi: 10.1109/iccsee.2012.15

[32] K. Wang, Z. Li, W. Feng, Reconstruction of finer voxel grid transmission images in tomographic gamma scanning. Nucl. Instrum. Methods Phys. Res., Sect. A. 755, 28-31 (2014). doi: 10.1016/j.nima.2014.04.046

[33] H. Zheng, X. G. Tuo, S. Peng, et al., An improved algebraic reconstruction technique for reconstructing tomographic gamma scanning image. Nucl. Instrum. Methods Phys. Res., Sect. A. 906, 77-82 (2018). doi: 10.1016/j.nima.2018.07.095

[34] A. J. He, X. G. Tuo, R. Shi, et al., An improved OSEM iterative reconstruction algorithm for transmission tomographic gamma scanning. Appl. Radiat. Isot. 142, 51-55 (2018). doi: 10.1016/j.apradiso.2018.09.001

[35] C. M. Wang, R. Shi, X. G. Tuo, et al., Reconstruction of tomographic gamma scanning transmission image from sparse projections based on convolutional neural networks. Nucl. Instrum. Methods Phys. Res., Sect. A. 1039, 167110 (2022). doi: 10.1016/j.nima.2022.167110

[36] M. X. Shu, C. Y. Shan, W. G. Gu, et al., A neural network-based method for measuring the activity of waste drum. Nucl. Tech. 46, 120501 (2023). in Chinese

[37] Z. J. Guo, P. Nie, Direct measurement method and verification of γ nuclides in boxed waste. Ato. Ener. Sci. Tech. 58, 913-921 (2024). in Chinese

[38] M. Jandel, M. Morhac, J. Kliman, et al., Decomposition of continuum γ-ray spectra using synthesized response matrix. Nuc. Inst. Meth. Phys. Res. A. 516, 172-183 (2004). doi: 10.1016/j.nima.2003.07.047

Note: Figure translations are in progress. See original paper for figures.