DEVELOPMENT OF A SPECIFIC ACTIVITY DISTRIBUTION ESTIMATION METHOD FOR LARGE LOW-LEVEL RADIOACTIVE WASTE USING SHAPE MEASUREMENT TECHNIQUE

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ABSTRACT

When large items of low-level radioactive waste are to be transported, it is desirable to pack them into a large container instead of a drum to reduce the cost of cutting them into small pieces and also to protect workers from unnecessary radiation exposure. According to IAEA Safety Standards No. TS-R-1, this type of waste is regarded as a ‘low specific activity (LSA)-II material’, and it is possible to transport it as an industrial package (IP) when there is no extreme nonuniformity in the distribution of its activity. The simple criterion of a uniform distribution is suggested in IAEA Safety Standards TS-G-1.1 as “*the differences in specific activity between portions of a factor of less than 10 would cause no concern*”. In our previous study, it was clarified that a conventional nondestructive radiation measurement method is inapplicable for judging the transport requirement of a completed large waste container because the uncertainty is too large when the filling rate is above approximately 10%. Thus, the authors have developed a new specific activity distribution estimation method for large low-level radioactive waste. In this method, the specific activity estimation of a segment, which refers to a certain amount of radioactive waste placed in a large container, is repeated until a waste package is completed, by the use of gamma-ray measurement, mass measurement, shape measurement by photogrammetry and Monte Carlo calculation techniques. The specific activities of the portions of the large waste package can be estimated as a summation of the specific activities of the segments. The degree of standard uncertainty in the specific activity estimation of segments was evaluated experimentally by using standard radioactive sources and mock-metal-waste samples and varying the filling rate, the distance between the detector and the waste, and the thickness of the segment. The combined standard uncertainty of the specific activity of portions was also estimated by a Monte Carlo simulation. As a result, the applicable scope of this method was clarified in terms of the parameters of segment size and filling rate.

INTRODUCTION

In Japan, low-level radioactive waste generated from nuclear power plants is currently stored in 200L metal drums and transported to the disposal site in Rokkasho Village in Aomori prefecture. The waste includes metal pipes, concrete debris, clothing, plastic sheets and resins, which are generated during the ordinary operation, the construction of facilities and the periodic inspections of nuclear power plants. The waste is stored in containers and a solidification process is carried out using a binding agent, such as cement or a plastic material. In this paper, the waste placed within a container, including the container itself, is referred to as a ‘waste package’. Then, eight solidified waste packages, i.e., eight 200L drums, are shipped to Rokkasho using special packaging.

On the other hand, in the decommissioning of a nuclear power plant, a huge amount of low-level radioactive waste with larger dimensions will be generated, such as building blocks near the reactor core and internal structures. When large items of radioactive waste are to be disposed of, it is preferable to package them in large waste containers instead of the current 200L drums to reduce the cost of cutting large pieces of waste into small pieces. In this case, if a waste package can be treated as a ‘low specific activity (LSA)-II material’, the transport cost can be reduced by using industrial packaging (IP) instead of type B transport casks. Furthermore, increasing the filling rate of large waste packages is also important for rational waste transport and the effective use of waste packages and repository sites.

According to IAEA Safety Standards No. TS-R-1 published in 2009 [1], for an LSA-II material, the uniformity in the distribution of specific activity must be certified before transport. In IAEA Safety Standards No. TS-G-1.1 (Rev.1) [2], the following criterion for a uniform distribution is suggested: “*the differences in specific activity between portions of a factor of less than 10 would cause no concern. However, there is no need to assess and compare the specific activity of each of these portions, provided that the estimated maximum average specific activity in any of these portions does not exceed the specific activity limit for solids”*.

In our previous study, the authors found that a simple gamma-ray measurement method cannot be used to verify the compliance with the transport requirement when the filling rate of a large waste package is higher than approximately 10% [3]. Thus, the development of a new specific activity distribution estimation system using shape measurement and Monte Carlo (MC) calculation techniques is underway. In this system, a clearance measurement technique [4] is applied to estimate specific activity with high accuracy.

METHODS

Activity Distribution Estimation System

Figure 1(a) shows a photograph of the specific activity distribution estimation system deployed in our laboratory. This system consists of a photogrammetry system (Fig. 1(b)), a radiation measurement system and a mock-disposal container with a weighing machine on the platform (Fig. 1(c)).

The photogrammetry system is composed of two CCD cameras (XCD-SX910CR, Sony Corporation) and lenses (SV-0514MP or SV-03514, V.S. Technology) with ten laser markers, four lamps for illumination and dedicated software for shape recognition. The laser markers and lamps are used to obtain the surface shape of the radioactive waste clearly. The radiation measurement system is composed of two NaI(Tl) scintillation detectors, whose crystal sizes are 2” in diameter and 2” in length. These collimators are used with respect to the distance between the waste and the detector surface. The positions of these radiation detectors can be tuned manually.

The mock disposal container with outer dimensions of 165 cm x 165 cm x 120 cm (height) was deployed on the platform for load dispersion. The dimensions of the mock disposal container were chosen with reference to an actual disposal container with outer dimensions of 160 cm x 160 cm x 120 (160) cm (height), which is currently under consideration by the Federation of Electric Power Companies in Japan (FEPC).

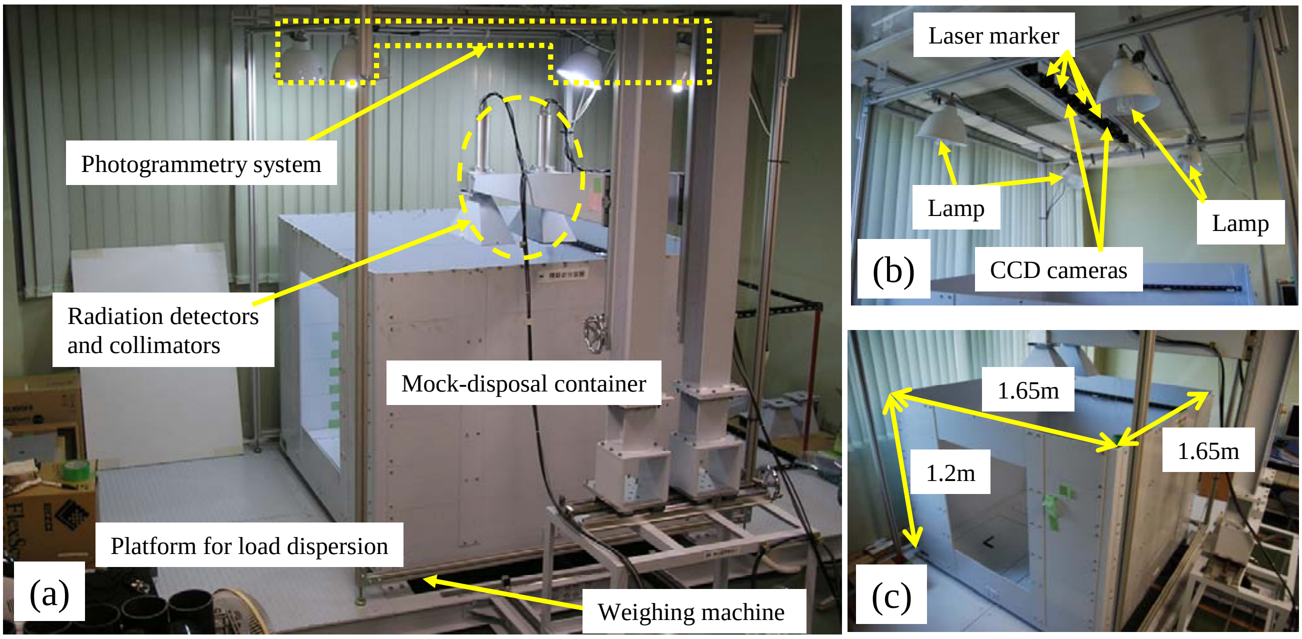

Figure 1. Appearance of the activity distribution estimation system.

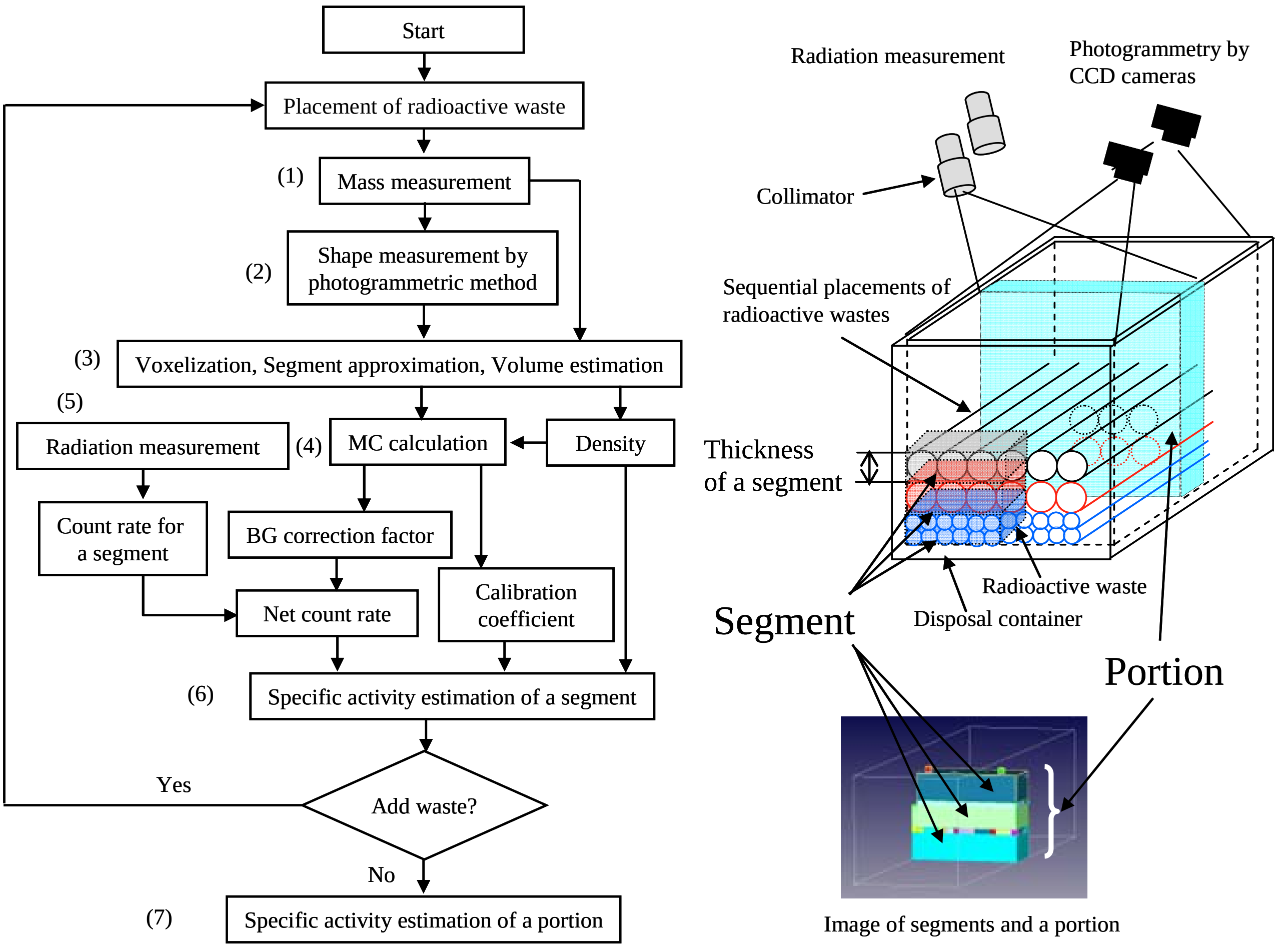

Figure 2. Flow of specific activity estimation in a large container. A segment, which is a part of a portion, is regarded as the unit in specific activity estimation.

Flow of Specific Activity Estimation

The flow of the specific activity estimation method indicated in Fig. 2 is as follows:

(1) Mass measurement is carried out after the placement of a certain amount of radioactive waste.

(2) By the photogrammetry method, the surface shape of the radioactive waste is obtained as a dot image.

(3) The dot image is voxelized after noise removal. A voxel corresponds to a cube of side 5 cm. Considering the size of the portion, voxels are assembled into a 'segment'. A segment, which is a part of a portion, is regarded as the basic unit in specific activity estimation in this system. The volume and density for each segment is calculated approximately.

(4) The shape of the radioactive waste is written in the Monte Carlo N-Particle Transport Code System (MCNP) input files [5]. MCNP calculations for the calibration factor and the BG correction factor are carried out.

(5) Radiation measurements are carried out for each segment.

(6) Using the count rate, calibration factor and BG correction factor, the specific activity of each segment can be estimated. Then, the user determines whether or not additional radioactive waste can be placed in the disposal container.

(7) When the container is filled with radioactive waste, or placement is stopped after considering other criteria (surface dose rate, total activity, etc.), the specific activity for each portion can be calculated by summing the specific activities of the segments comprising in each portion. Then, compliance with the transport regulation can be judged.

EXPERIMENTS

Degree of Uncertainty in Calibration

In the MCNP Monte Carlo calculation for the calibration factor, the user should define the source type (surface contamination or activation) according to the log of the radioactive waste. When there is no possibility of surface contamination, the activation source can be defined and the calibration process can be carried out with comparatively high accuracy. On the other hand, when there is a possibility of surface contamination, a point source is assumed at the center of gravity of a segment. In this case, it is necessary to consider the uncertainty in calibration due to the unknown shielding effect, which is associated with the unknown source position.

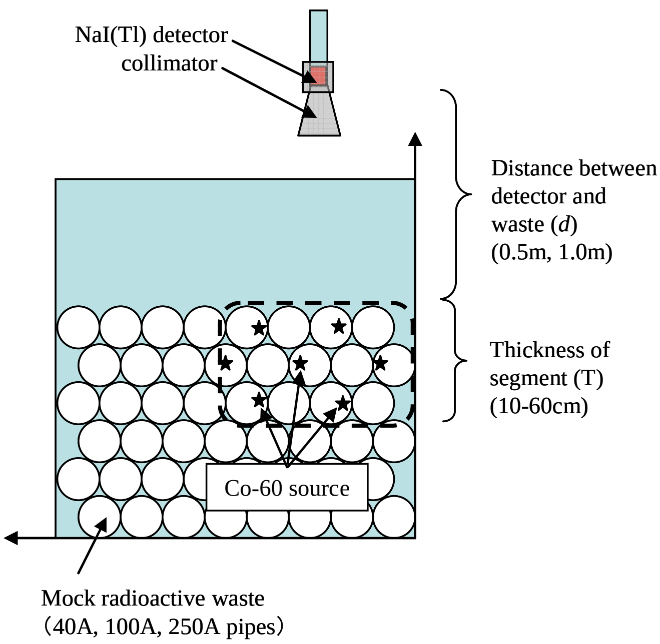

Figure 3. Experimental setup for estimation of calibration uncertainty. A Co-60 source was placed at various positions.

Hence, the degree of this type of uncertainty was experimentally estimated by varying the position of a Co-60 standard radioactive source, the distance between the radiation detector and the surface of the radioactive waste, *d*, the filling rate of the mock radioactive waste, the thickness of a segment, T, and the component of the pulse height distribution (Fig. 3). SUS metal pipes satisfying Japanese Industrial Standard (JIS) 40A (48.6 mm outer diameter and 3 mm thickness), 100A (114 mm outer diameter and 4 mm thickness) and 250A (267 mm outer diameter and 6.5 mm thickness) were used as mock dismantling waste. The filling rates of these pipes were 7.5%, 12% and 21%, respectively.

RESULTS AND DISCUSSION

Ratios between estimated and actual activity

Some experimental results are shown in Fig. 4. The X-axis indicates the source position and the Y-axis indicates the ratio of estimated activity to the actual activity.

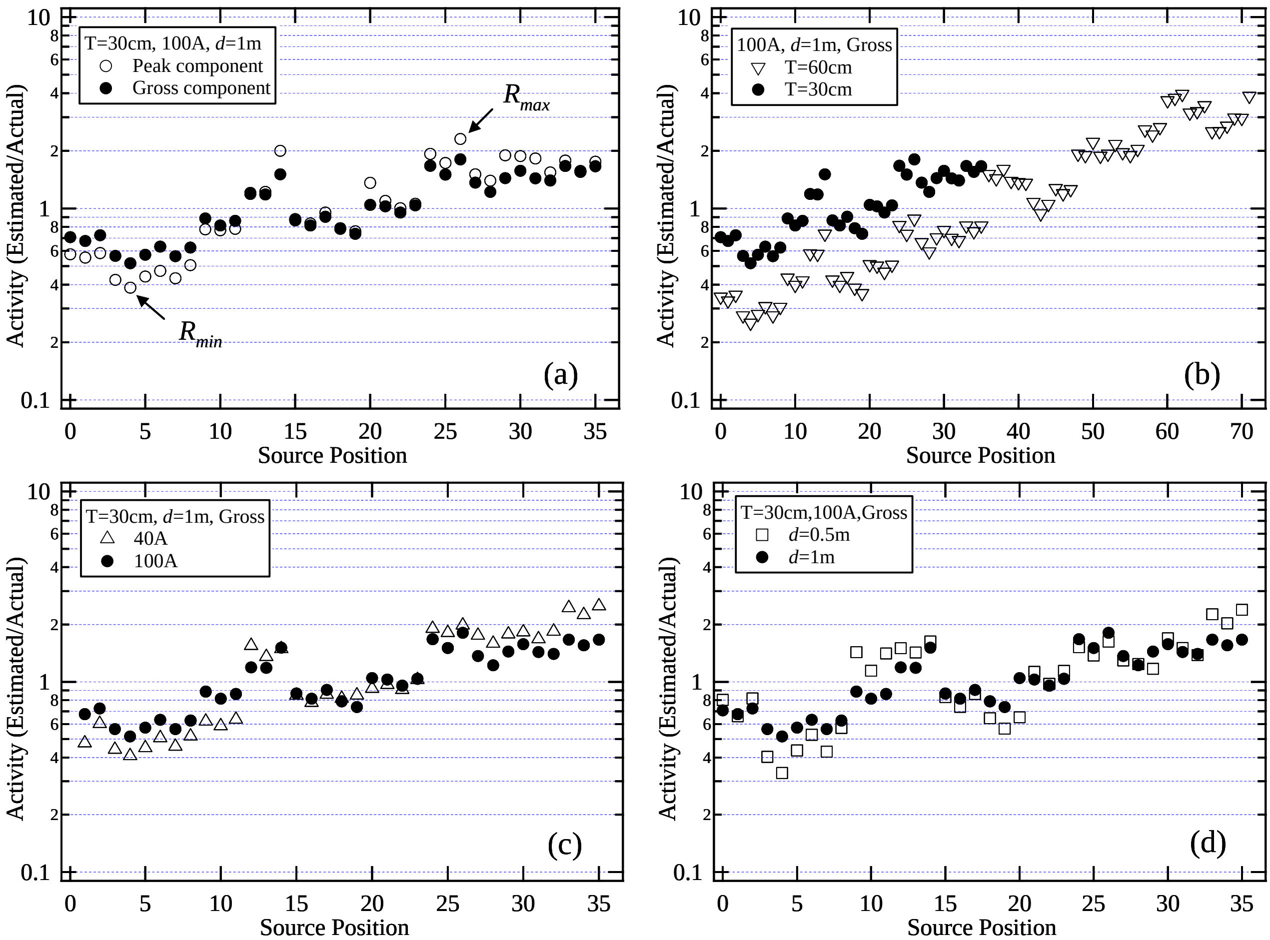

Figure 4. Ratio of activity estimated by the system to the actual activity.

The results in Fig.4 (a) show that analysis of the gross component of the pulse height spectrum is preferable to analysis of the peak component, because the degree of discrepancy between the estimated activity and actual activity is comparatively small. This is because scattered gamma rays can compensate the attenuation of the peak components and smoothly ensure high overall efficiency. The results in Fig. 4(b) show that a thinner segment is preferable for suppressing the uncertainty in calibration. This is due to the decrease in the degree of the shielding effect. For the same reason, Fig. 4(c) shows that a lower filling rate is preferable even it reduces the effectiveness of the use of radioactive waste disposal containers. The result in Fig. 4(d) shows that a greater distance between the radioactive waste and the detector can also suppress the uncertainty in calibration.

Using these results, the calibration uncertainty factor for the activity estimation of a segment, U_f, was estimated by the following equation:

$$U_f = \frac{\sqrt{R_{\max} \cdot R_{\min}}}{R_{\min}}, \quad (1)$$

where R_{max} and R_{min} are the maximum and minimum values for each set of experimental conditions, respectively, as shown in Fig. 4(a). This uncertainty factor is used in the analysis described below.

Estimation of Standard Uncertainties in Activity Estimation of a Segment

As mentioned in the specific activity estimation flow, since the specific activity of a portion can be estimated by summing the specific activities of the included segments, it is necessary to consider the propagation of uncertainties to estimate the uncertainty in the specific activity of a portion. Here, since contamination exists at various positions in actual radioactive waste, its existence probability is assumed to be spatially homogeneous among a segment. Moreover, because the size of a segment can be selected in the placement process, over/underestimates beyond R_{max} and R_{min} never occur.

Given that the attenuation of radiation can be expressed using exponential functions, a log-uniform distribution can be used as the probability distribution for the calibration uncertainty of a segment.

Here, according to the Guide to the Uncertainty in Measurement, GUM [6], the variation for a uniform distribution, $u^2(x_i)$, is expressed as

$$u^2(x_i) = a^2/3, \quad (2)$$

where, a is the difference between the maximum (minimum) and its arithmetic mean value (Fig. 5(a)). By adopting this relationship in the log-uniform distribution, the following equation can be regarded as the standard uncertainty in the log-uniform distribution:

$$u(x_i) = \exp(\ln(a)/\sqrt{3}). \quad (3)$$

Figure 5(b) shows the probability density of the log-uniform distribution, and Fig. 5(c) shows the probability density when d is 1.0 m, T is 60 cm, the mock radioactive waste is a 40A pipe

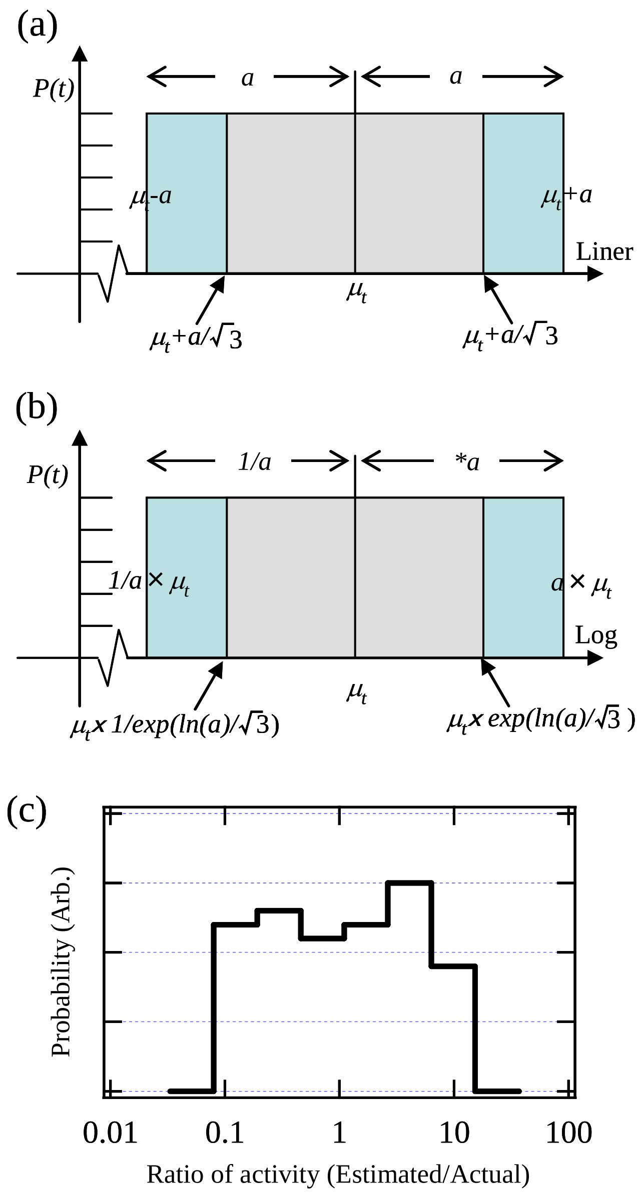

Figure 5. Probability densities for uniform distribution and log-uniform distribution.

and gross components are analyzed. As shown in these figures, it is found that the adoption of the log-uniform distribution for the uncertainty in calibration is appropriate and that U_f corresponds to the parameter *a* in the log-uniform distribution.

Table 1 summarizes the standard uncertainties, $u(x_i)$, in the calibration of the activity estimation for a segment. Because of the size of our laboratory, the values of *d* were restricted to 0.5 and 1.0 m in the experiments. For actual nuclear power plants, it is predicted that *d* will increase with the size of the measurement room. In this study, MCNP Monte Carlo simulations were also used to estimate the uncertainty in calibration when *d* is 0.5, 1.0, 2.0, 3.0 and 4.0 m. The calculated results are also summarized in Table 1.

Table 1. Experimental and calculated standard uncertainties in calibration of activity estimation for a segment.

Experimental results, *d* = 0.5m

Filling rate	Thickness of segment				
	10cm	20cm	30cm	40cm	60cm
0.075	1.49	1.49	1.62	1.62	2.98
0.121	1.52	1.58	1.77	2.13	2.97
0.214	1.43	1.87	2.35	2.97	4.72

Experimental results, *d* = 1.0m

Filling rate	Thickness of segment				
	10cm	20cm	30cm	40cm	60cm
0.075	1.20	1.25	1.25	1.54	1.88
0.121	1.17	1.36	1.44	1.71	2.21
0.214	1.14	1.47	1.69	2.29	3.78

Calculated results, Filling rate = 0.214

d	Thickness of segment				
	10cm	20cm	30cm	40cm	60cm
0.5m	1.52	1.86	2.29	2.83	4.32
1.0m	1.31	1.59	1.96	2.39	3.61
2.0m	1.25	1.50	1.81	2.20	3.23
3.0m	1.22	1.46	1.75	2.11	3.07
4.0m	1.21	1.44	1.72	2.07	3.00

Estimation of Combined Standard Uncertainty in Activity Estimation of a Portion

To obtain the combined standard uncertainty in the activity estimation of a portion, the propagation of uncertainties should be considered when summing the probability density of the log-uniform distribution. In this case, the degree of uncertainty in the activity estimation of a portion cannot be estimated using an analytical method. Thus, the authors conducted Monte Carlo simulations to estimate the combined standard uncertainty.

Meanwhile, to confirm that the specific activities of each portion are within a factor of less than 10, there may be an upper limit of the degree of the uncertainty. When an estimation method gives an uncertainty larger than the square root of 10, it may be impossible to demonstrate whether the specific activities of the portions satisfy the transport requirement. Thus, here, the authors chose the reference value of combined standard uncertainty to be 2, which is applicable to the inspection of the transport requirement.

Figure 6 shows the boundaries where the combined standard uncertainty in the activity estimation of a portion is 2, as functions of segment thickness, filling rate and the distance between the radiation detector and the surface of the radioactive waste. As shown in the figure, the effect of distance is small when *d* is larger than 3.0 m. When *d* is 3.0 m, and the user designates that the filling rate of a radioactive waste package is 30 %, the thickness of a segment should be less than approximately 36 cm. Using these results, rational radioactive transport can be achieved because an appropriate filling rate can be estimated while taking the uncertainties into account.

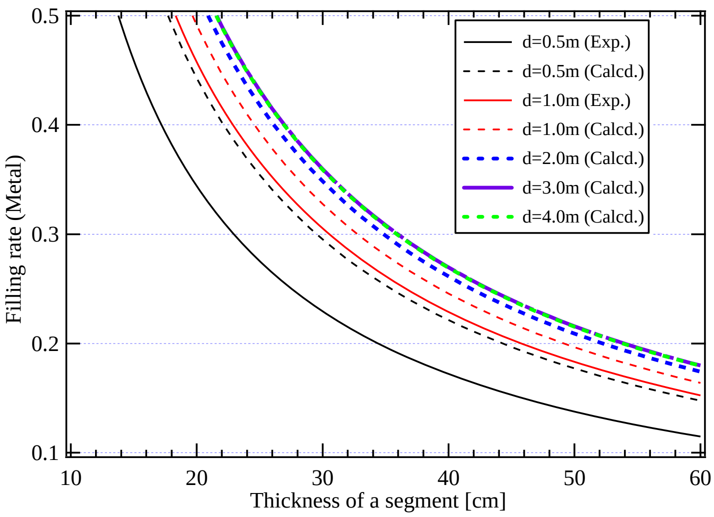

Figure 6. Range where combined standard uncertainty in activity estimation of a portion is less than 2 as functions of segment thickness, filling rate and the distance between the radiation detector and the surface.

CONCLUSIONS

A new specific activity distribution estimation method for large items of low-level radioactive waste has been developed by utilizing gamma-ray measurement, mass measurement, shape measurement by photogrammetry and MCNP Monte Carlo calculation techniques. The standard uncertainty in the activity estimation of a segment was experimentally obtained and that of a portion was estimated by Monte Carlo simulation by assuming that the probability density of uncertainty in the activity estimation of a segment is a log-uniform distribution. The authors hope that the new system will be a helpful tool for the transport of large waste packages with a high filling rate.

REFERENCES

[1] International Atomic Energy Agency, Regulations for the Safe Transport of Radioactive Material 2009 Edition: IAEA Safety Standards Series, No. TS-R-1, IAEA, Vienna (2009).

[2] International Atomic Energy Agency, Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.1 (Rev. 1), IAEA, Vienna (2008).

[3] M. Sasaki and T. Hattori, 'Development of New Monitoring System for Activity Distribution Estimation in Large Waste Container for LSA-II Material', Proceedings of the 15th International Symposium on the Packaging and Transportation of Radioactive Materials PATRAM 2007, October 21-26, 2007, Miami, Florida, USA (2007).

[4] M. Sasaki and T. Hattori, 'Clearance Automatic Laser Inspection System (CLALIS) for Radioactivity Evaluation of Metal Waste from Decommissioning Nuclear Power Plants', Proceedings of ICEM'05: The 10th International Conference on Environmental Remediation and Radioactive Waste Management, September 4-8, 2005, Scottish Exhibition & Conference Centre, Glasgow, Scotland (2007).

[5] J. F. Briesmeister, 'MCNP^{TM} - A General Monte Carlo N-Particle Transport Code, Version 4C', LA13709M, Los Alamos National Laboratory (2000).

[6] International Standards Organization, Guide to the Expression of Uncertainty in Measurement: GUM, ISO (1995).