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DEVELOPMENT OF NEW MONITORING SYSTEM FOR ACTIVITY DISTRIBUTION ESTIMATION IN LARGE WASTE CONTAINER FOR LSA-II MATERIAL

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ABSTRACT

Low-level radioactive waste generated from the operation of nuclear power plants is currently being stored in 200-L metal drums and transported to a disposal site in Japan. On the other hand, a great amount of low-level radioactive waste with larger dimensions will be generated from the decommissioning of nuclear power plants. When large radioactive waste is to be disposed of, it is desirable to package such waste in large containers instead of drums to reduce the cost of cutting it into small pieces, and also to protect workers and the public from additional radiation exposure. According to the IAEA Safety Standards Series Requirements No. TS-R-1 published in 2005, most of these wastes will be categorized as ‘low specific activity (LSA)-II material’, whose uniformity in the distribution of activity must be certified before transport. The following simple criterion of uniform distribution is suggested in the IAEA Safety Guide of ST-2:226.14, 15: differences between the specific activities of portions, which are defined as one-fifth or one-tenth of the volume, should be within a factor of less than 10. Thus, we are investigating techniques for evaluating the activity distribution in large waste containers. In this study, metal pipes with different filling rates were taken to be radioactive waste in a large rectangular waste container (100 cm (W), 100 cm (H) and 300 cm (L)). First, a simple gamma-ray measurement method was examined using the Monte Carlo N-Particle Transport Code System (MCNP). The uncertainty of the activity distribution estimation of one-tenth the volume was calculated for various source positions and measurement positions. As a result, it was found that applying a simple gamma-ray measurement is inadequate to determine the distribution of activity when the filling rate is higher than approximately 10 % and the measurement position is 100cm from the center of the container. To overcome this limitation on the filling rate, a new conceptual design, which utilizes three-dimensional shape measurement, Monte Carlo calculation and radiation measurement, is proposed for the activity distribution monitoring system for waste with a higher filling rate.

INTRODUCTION

Low-level radioactive wastes are classified into three types, L1, L2 and L3, in order of descending activity concentration, i.e., specific activity, in Japan. The L3 waste, which contains a comparatively small amount of specific activity, can be disposed of by a trench method inside a nuclear power plant site. On the other hand, since the L1 and L2 wastes should be disposed of at their exclusive facilities, these wastes should be transported in compliance with the transport regulations imposed by the Japanese regulatory authority.

In Japan, the L2 wastes generated from nuclear power plants are currently stored in 200-L metal drums, and transported to the disposal site in Rokkasho Village in Aomori prefecture. Here, the wastes include metal pipes, concrete debris, waste cloths, plastic sheets and resins. They are generated during ordinary operation, the construction of a relevant facility and the regular periodical inspection of a nuclear power plant. These wastes are stored in a container and the solidification process is carried out using a binding agent, such as cement or plastic materials. In this paper, whole waste placed within a container is called a ‘waste package’, including the container itself. Then, eight solidified waste packages, i.e., 200-L drums, are shipped using a special packaging, which is designed to endure possible accidents and to suppress surface dose rate.

On the other hand, in the decommissioning of a nuclear power plant, a great amount of low-level radioactive waste with larger dimensions will be generated, such as building blocks near the reactor core and reactor internal structures. Since these types of waste will be categorized as L1 or L2 waste, they should be transported to the disposal site. When large radioactive wastes are to be disposed of, it is preferable to package them in large waste containers instead of the current 200-L drums to reduce the cost of cutting large wastes into small pieces. In this case, if a waste package can be treated as ‘low specific activity (LSA)-II material’, the transport cost can be reduced using the industrial packaging instead of the type B transport cask. Furthermore, increasing the filling rate of a large waste package is also important for rational waste transport.

According to the IAEA Safety Standards Series Requirements No. TS-R-1 published in 2005 (IAEA, 2005), the LSA-II material, whose uniformity in the distribution of activity must be certified before transport. In the IAEA Safety Guide of ST-2:226.14 and 15 (IAEA, 2002), the following simple criterion of uniform distribution is suggested: *differences between the specific activities of portions, which are defined as one-fifth or one-tenth of the volume, should be within a factor of less than 10*.

Thus, in this study, we have begun to develop a measurement system, which can evaluate the specific activities of portions of a large waste package. First, simple gamma-ray measurement methods were supposed and their detection abilities were investigated using the Monte Carlo N-Particle Transport Code System (MCNP) (Briesmeister, 2000). In the calculation, metal pipes with different filling rates were considered to be radioactive waste. The uncertainties in the specific activity evaluation of a portion of the waste package, which depend on the measurement method and filling rate of waste, were estimated.

CALCULATIONS

Figure 1 shows the calculation model of dismantling waste, a large container, a portion, radioactive sources, radiation detectors and measurement methods. The details are as follows:

Waste: Although, concrete debris, blocks and metal scraps with complex geometries and various sizes and shapes can be projected as large dismantling waste in actual practice, SUS metal pipes of Japanese Industrial Standard (JIS) 40A (48.6 mm in outer diameter, and 3 mm in thickness), 100A (114 mm in outer diameter, and 4 mm in thickness) and 250A (267 mm in outer diameter and 6.5 mm in thickness) type were assumed as dismantling waste in the MCNP calculation. The filling rates of waste packages were to be 19 %, 10 % and 6 %, respectively, when these SUS metal pipes were placed closer together, as shown in Fig. 1. The solidification material was not considered in the calculation.

Large container: In the current transportation of L2 waste, a packaging with dimensions of 160 cm (W), 110 cm (H) and 320 cm (L) is used in Japan. As described before, eight solidified waste packages, i.e., 200-L drums are shipped using this type of packaging. Assuming that a similarly sized packaging is used, the size of a large inner container may be slightly smaller than that of the current packaging. Hence, the outer dimensions of a large container were assumed to be 100 cm (W), 100 cm (H) and 300 cm (L).

The thickness of a 200-L drum wall is approximately 0.1 - 0.16 cm. It is expected that the thickness of a large container may be thickened in order to reduce the surface dose rate, and to

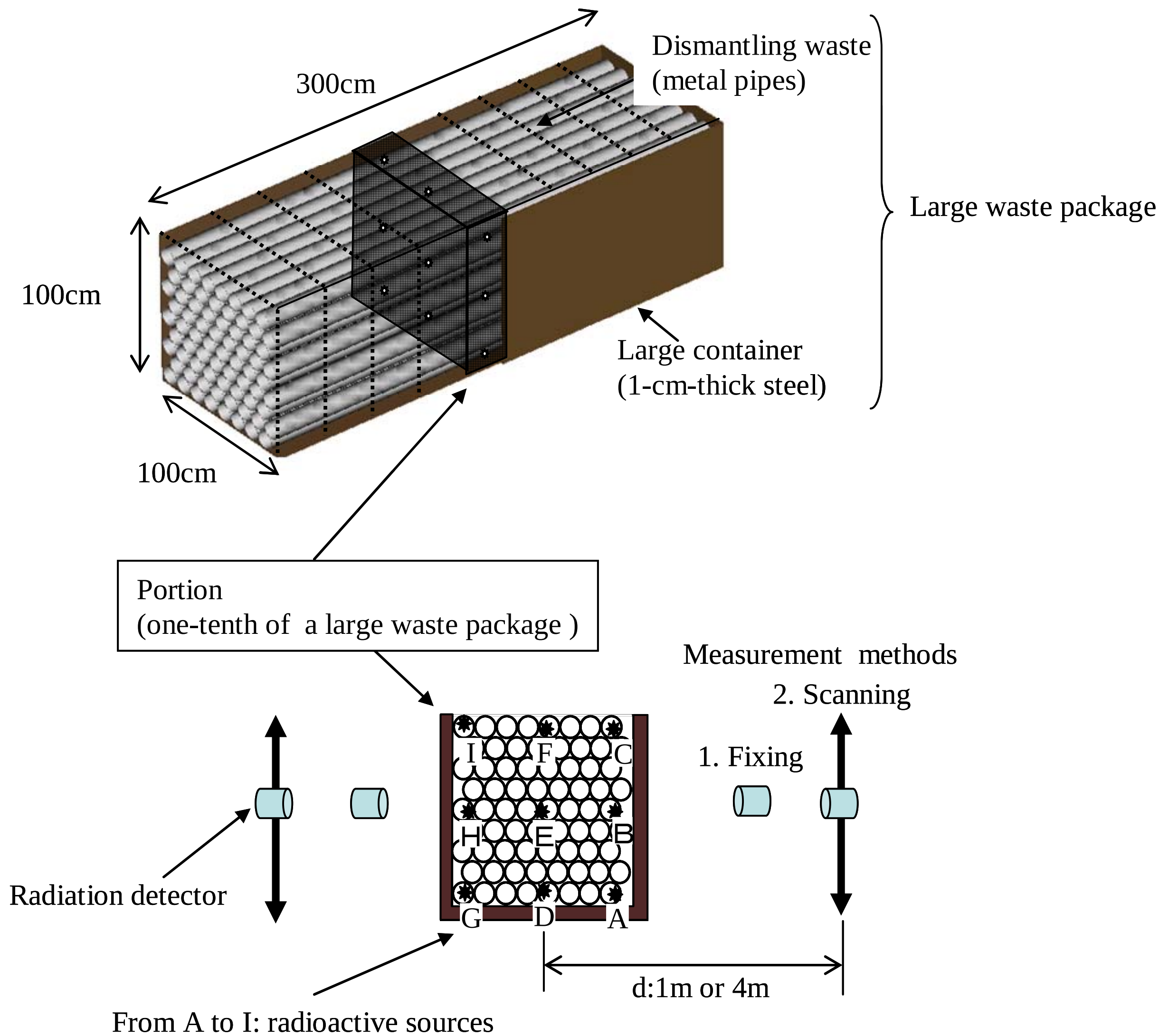


Figure 1. Calculation model of a large container, dismantling waste, positions of radioactive sources and radiation detectors. The dimensions of a portion are 100 cm (W), 100 cm (H) and 30 cm (L) since one-tenth of a large waste package is regarded as a portion when its volume is greater than 1.0 m^3.

increase the mechanical strength. The thickness of a large container wall was assumed to be 1 cm.

Portion: According to the IAEA Safety Guide of ST-2:226.14 and 15, it is described that *a volume greater than 1.0 m^3 should be divided into ten parts of approximately equivalent size*. Thus, in this calculation, the size of a portion was considered to be 100 cm (W), 100 cm (H) and 30 cm (L).

Radioactive source: Because the main radionuclide is Co-60 in the radiation measurement of dismantling waste, gamma-rays with an energy of 1.25 MeV were generated from a point source in the MCNP calculation. The source positions from 'A' to 'I' are defined at nine distinct positions in a portion since the position and distribution of radioactive sources are essentially unknown as long as it is certain that there are no secondary contaminations or hot spots on the measurement target.

Radiation detector, assumed measurement methods and analysis conditions: In the actual gamma-ray monitoring, NaI(Tl) scintillation or HPGe detectors are often utilized because they have good characteristics in terms of energy resolution. In the MCNP calculation, the gamma-ray energy spectra at detector locations were estimated by the F5 tally to compare the advantages of peak analysis and gross analysis conditions in the specific activity evaluation. It was assumed that the measurement methods were 'fixing' and 'scanning', and radiation measurements were carried out at both sides of a large waste package. Although it is expected that a large waste package may be filled with a binding agent in the actual process as in the case of typical L2 wastes, such solidification is not considered in the calculations. The distance between the center of a large waste package and the surface of a radiation detector, *d*, was assumed to be 1 m or 4 m.

RESULTS AND DISCUSSION

Results of MCNP

Figure 2 shows the (a) peak gamma-ray fluxes and (b) gross gamma-ray fluxes for different source positions obtained using the MCNP code, assuming that the measurement method is scanning, *d* is 4 m and the waste materials are 100A pipes. In the following results, the lower-level discrimination of measurable gamma-rays is assumed to be 50 keV and the flux is normalized by the Co-60-equivalent source. Since the gamma-ray flux can be converted into the count rate of a detector using appropriate response functions, it is possible to regard the difference in gamma-ray flux as the difference in count rate. As shown in Fig. 2 (a) or (b), even if point sources with the same activities exist in a portion, the count rate would be different

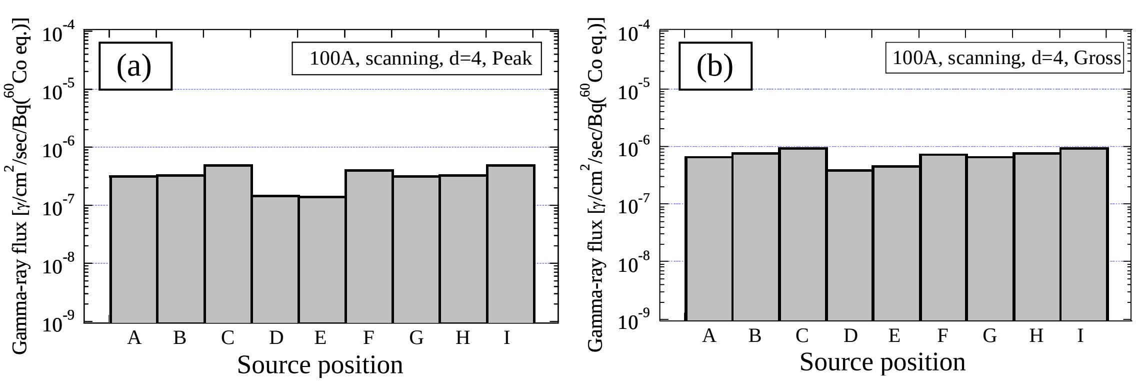


Figure 2. Peak gamma-ray fluxes, (a), and gross gamma-ray fluxes, (b), at a detector location for different source positions when the waste is 100A SUS pipes, measurement method is scanning and *d* is 4m.

depending on the source position because of the unknown degree of the shielding effect. This difference causes an uncertainty in the specific activity evaluation of a portion accordingly. Hence, it is preferable that the difference in gamma-ray flux be rather small regardless of the source positions. From this point of view, gross gamma-ray flux analysis has an advantage over peak gamma-ray flux analysis, as shown in Fig. 2.

Figure 3 shows the gross gamma-ray fluxes for different source positions when the distance *d* is (a) 1 m and (b) 4 m, as indicated in Fig. 1. Since the efficiency of gamma rays strongly depends on the distance *d*, the difference of the average gamma-ray fluxes between (a) and (b) is larger than that in Fig. 2. In Fig. 3 (b), the gamma-ray fluxes for different source positions are roughly constant within a factor of two, which is comparatively better than the case in Fig. 3 (a). This indicates that the uncertainty of activity estimation, which is caused by unknown source positions, can be reduced by increasing the distance between a waste package and a detector.

Figure 4 shows the gross gamma-ray fluxes for different source positions for the cases in which the waste materials are (a) 40A SUS pipes and (b) 250A pipes. As described previously, their filling rates were to be 19 % and 6 %, respectively. Because the filling rate directly affects the degree of the shielding effect, in the case that the waste materials are 40A SUS pipes, the count rate may significantly decrease when the sources exist near the center of a large waste package.

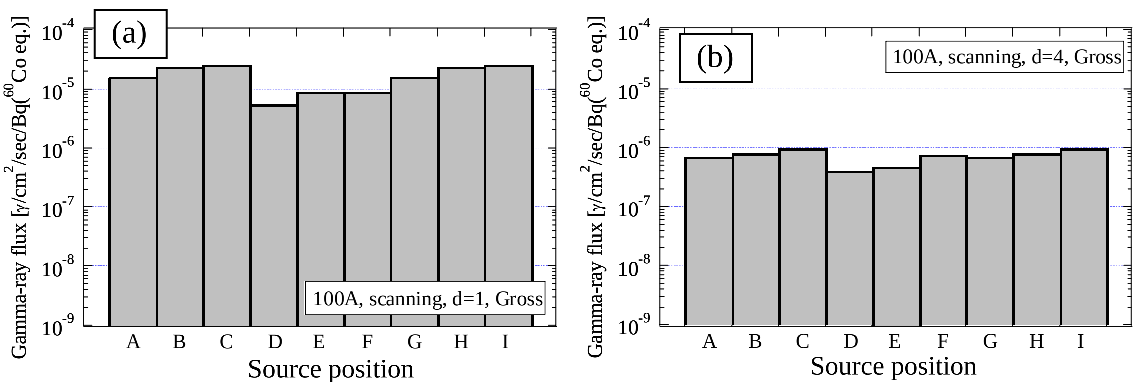


Figure 3. Gross gamma-ray fluxes at detector locations when (a) *d* is 1m and (b) *d* is 4m for different source positions. The waste materials are 100A SUS pipes, the measurement method is scanning and *d* is 4m.

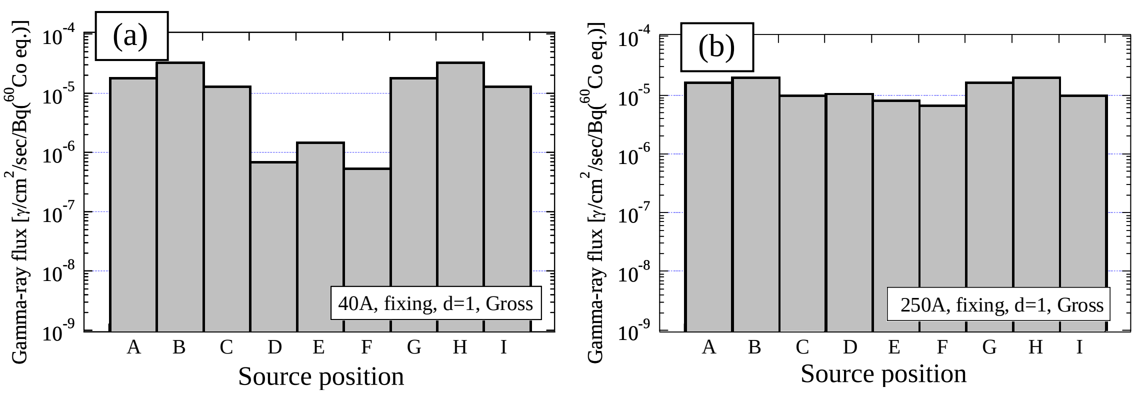
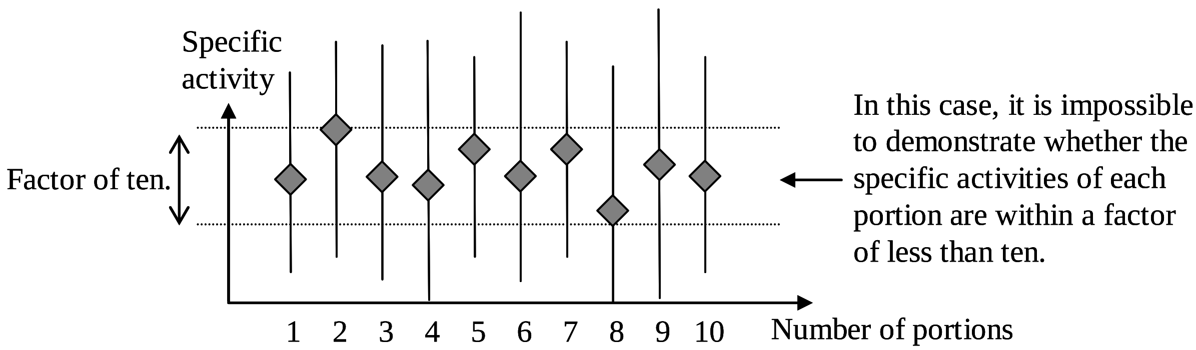


Figure 4. Gross gamma-ray fluxes at detector location for different source positions when the measurement method is fixing and *d* is 4m. The left graph (a) shows the results when the waste materials are 40A pipes and right one (b) shows those when the waste materials are 250A pipes.

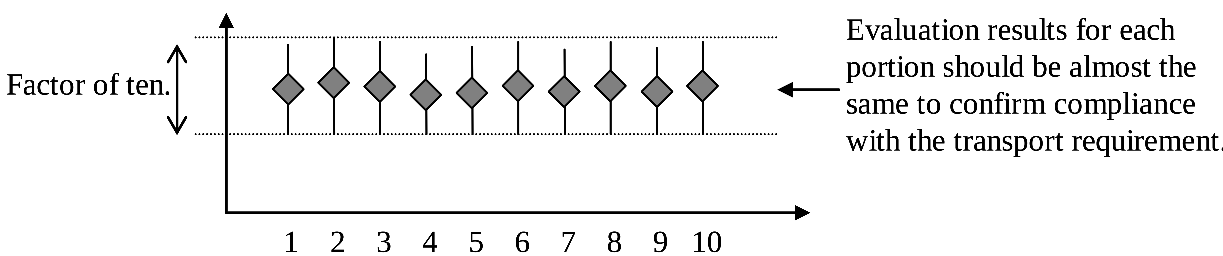
Upper limit of uncertainty in a specific activity evaluation method

To confirm that the specific activities of each portion are within a factor of less than 10, there may be an upper limit of uncertainty in a specific activity evaluation method. When an evaluation method gives an uncertainty larger than the square root of 10, it may be impossible to demonstrate whether the specific activities of each portion satisfy the transport requirement, as shown in Fig. 5 Case (i). In the case that the uncertainty is approximately equal to the square root of 10, the specific activities of each portion should be almost the same, as shown in Fig. 5 Case (ii). Thus, it is desirable to decrease the uncertainty as much as possible, as shown in Fig. 5 Case (iii). In this study, prospective uncertainty (PU) was estimated to roughly compare the detection ability for each measurement method and analysis condition. The PU was defined as

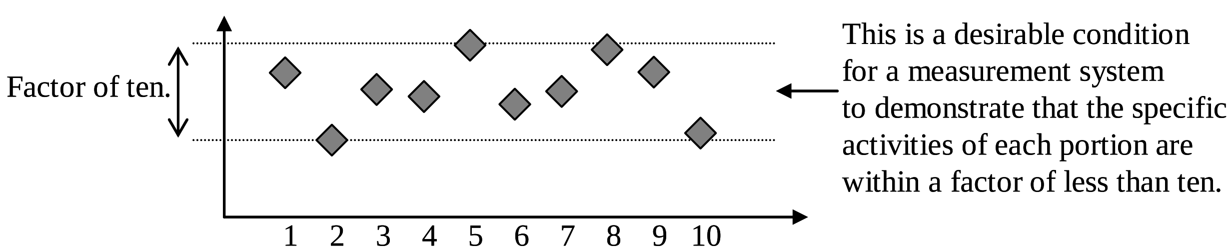
PU=ϕ_{gm}/ϕ_{min},



Case (i) Example of specific activity comparisons between each portion of a large waste package when the uncertainty of evaluation result is relatively large.



Case (ii) Example of specific activity comparisons between each portion of a large waste package when the uncertainty of evaluation result is approximately 3.



Case (iii) Example of specific activity comparisons between each portion of a large waste package when the uncertainty of evaluation result is relatively small.

Figure 5. Examples of specific activity comparisons in terms of the degrees of uncertainty in evaluation result.

where ϕ_{gm} and ϕ_{min} are the geometric mean and minimum values of gamma-ray flux estimated by MCNP calculations. Using this equation, the PUs, which depend on the measurement methods, analysis conditions and detector locations, were estimated as a function of filling rate. As mentioned earlier, when PU is larger than the square root of 10, it is impossible to demonstrate whether the specific activities of each portion are within a factor of less than ten. Thus, here, we chose the reference value of PU to be 3, which is barely applicable to the inspection of the transport requirement.

Table 1 shows the maximum filling rates required to satisfy the reference value of PU for each specific activity evaluation method. As shown in this table, increasing *d* is effective in decreasing the PU for higher filling rates, which would be helpful for rational radioactive waste transport. However, from the view of actual radiation measurement at a nuclear power plant, it may be impossible to keep *d* as large as 4 m because of space limitations. When *d* is 1 m, the simple radiation measurement technique assumed in the calculation can be applied to the inspection necessary for the transport requirements for a large waste package with a filling rate of up to 13 %.

Table 1. Measurement method and filling rates, where we can barely verify whether the specific activities of each portion are within a factor of less than ten.

Radiation measurement and analysis methods		Filling rate when the PU is 3	
		d=1	d=4
Scanning	Gross	13 %	17 %
Fixing	Gross	12 %	20 %
Scanning	Peak	10 %	15 %
Fixing	Peak	Less than 6 %	17 %

Conceptual design of specific activity evaluation system for a large waste package

Since it was revealed that the simple measurement method cannot be used to confirm compliance with the transport requirements when the filling rate is higher, we have begun to develop a new specific activity evaluation system for rational waste transportation. Recently, a very low level activity evaluation technique has been developed, which utilizes three-dimensional laser shape measurement, Monte Carlo calculation and radiation measurement (Hattori, 2003) (Sasaki, 2005, 2007). We are now trying to apply this technique to the new specific activity evaluation system by utilizing the stereo matching technique using a pair of digital cameras instead of employing the laser scanning technique in consideration for the size of a large waste package. This system can become a real-time monitoring tool that can measure the specific activity of large waste as soon as it is placed in a large waste container, and can eventually evaluate the specific activities of portions with good accuracy.

CONCLUSIONS

Using MCNP calculation techniques, a simple gamma-ray measurement method was examined for a supposed large waste package. The point radioactive source was assumed to be located at various positions in a portion of the waste package. Consequently, it is found that the simple radiation measurement method cannot be used to achieve the inspection necessary for the transport requirement when the filling rate of a large waste package is higher than approximately 10 %.

Meanwhile, an amendment is considered (IAEA, 2007) in the safety guide of DS 346: *there is no need to assess and compare 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.* However, the transport requirement is still in force for waste when the specific activity of a portion may exceed the regulation limit. Moreover, the absolute value of the activity of a waste package should be evaluated with good accuracy to confirm that the total activity satisfies the limitation of the disposal site. We hope the newly developing system will enable the evaluation of the activity distribution of a large waste package at a high filling rate with good accuracy for the rational waste transport of large dismantling waste.

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