Applicability of Clearance Automatic Laser Inspection System to Clearance Measurement of Concrete Segments -9129

M. Sasaki, H. Ogino, T. Hattori
Nuclear Technology Research Laboratory, Central Research Institute of Electric Power Industry (CRIEPI)
2-11-1, Iwado-kita, Komae-shi, Tokyo, Japan

ABSTRACT

In the decommissioning of a nuclear power plant, large amounts of metal scrap and concrete segments require disposal when dismantling the nuclear reactor and surrounding facilities. When their activity level is negligible or sufficiently small, they can be regarded as general industrial waste. To distinguish between radioactive and nonradioactive materials, the clearance level for each radionuclide has been determined in units of activity concentration. These clearance levels are indicated in the International Atomic Energy Agency (IAEA) Safety Standards Series No. RS-G-1.7. The Japanese regulatory authority decided in 2005 to adopt the values given in RS-G-1.7 as clearance levels in legislation. Recently, a Clearance Automatic Laser Inspection System (CLALIS) has been developed utilizing gamma-ray measurement, automatic laser shape measurement and Monte Carlo calculation. CLALIS comprises four laser scanners and eight large plastic scintillation detectors surrounded by a 5-cm-thick lead shield. Using three-dimensional (3D) laser scanning, a measurement target, which is placed on the measuring tray, is represented as a dot image. The dot image is converted into voxels after noise removal, and is written in MCNP (A General Monte Carlo N-Particle Transport Code System) input files. When the gamma-ray measurement is carried out, the MCNP calculations are also carried out to obtain the calibration factor and background (BG) correction factors. For the clearance measurement of concrete segments, the effect of gamma -rays from natural radionuclides in the measurement target, such as K-40 and the radioactive decay products of Th-232 and U-238, should be taken into account to ensure adequate waste management. Since NE102A plastic scintillation detectors are used for gamma-ray measurement in CLALIS, it is impossible to distinguish between the count rates of natural radionuclides and contaminants on the basis of gamma-ray energy information. To overcome this problem, the activity evaluation flow of CLALIS was improved to allow for the BG count rate due to natural radionuclides. The accuracies of calibration and BG correction for concrete segments were estimated using a number of mock -concrete waste samples and varying their position, number and size. As a result, it was found that CLALIS can estimate the calibration factor and BG correction factor within good accuracies of +/- 20% and +/- 2%, respectively. The main components of concrete are cement, fine aggregate and coarse aggregate, and it is well known that the natural activity concentration of these components depends on where they are produced. In CLALIS, the BG count rate due to natural radionuclides is estimated using preanalyzed data obtained from the representative sample, the mass of the measurement target and its efficiency calculated by the MCNP code. Here, the effect of an uneven distribution of natural radionuclides between the measurement targets should be considered because it may contribute to the uncertainty of the BG count rate. In this study, to discuss in detail the applicability of CLALIS to the clearance measurement of concrete segments, the standard equation for detection limit estimation was extended to separately treat the ambient BG count rate, the BG count rate due to natural radionuclides and their uncertainties. Consequently, the precise detection limit of CLALIS was evaluated and its range of application was clarified with respect to the natural activity concentration and its uncertainty.

INTRODUCTION

In the decommissioning of a nuclear power plant, large amounts of metal scrap and concrete waste require disposal when dismantling the nuclear reactor and surrounding facilities. These types of waste are categorized with respect to their activity concentration level. In Japan, low-level radioactive waste is classified into three types, L1, L2 and L3, in decreasing order of activity concentration, i.e., specific activity. L3 waste, which has comparatively low specific activity, can be disposed of by a trench method inside a nuclear power plant site, and L1 and L2 waste should be disposed of at purpose-built facilities. On the other hand, when the activity concentration level is negligible or sufficiently small, waste can be regarded as general industrial waste and it can be reused or recycled. To distinguish between radioactive and nonradioactive waste, the clearance level for major nuclear facilities was determined in units of activity concentration ($Bq.g^{-1}$) in Japan in 1999 [1]. These clearance levels are also given in the International Atomic Energy Agency (IAEA) Safety Standards Series No. RS-G-1.7 [2]. Table I shows the clearance levels of ten important radionuclides cited from RS-G-1.7. In 2005, the Japanese regulatory authority decided to adopt the values given in RS-G-1.7 as clearance levels in legislation.

Table I. Clearance levels for ten important radionuclides [2].

Radionuclide	**Clearance level** [$Bq.g^{-1}$]
H-3	100
Mn-54	0.1
Co-60	0.1
Sr-90	1
Cs-134	0.1
Cs-137	0.1
Eu-152	0.1
Eu-154	0.1
Pu-239	0.1
Am-241	0.1

With this background, the first actual clearance measurements have started in Tokai Power Plant, which was the first commercial nuclear power plant, and also the first commercial plant to be decommissioned in Japan. Currently, the main target for clearance measurements is 2000 tons of scrap metal [3]. The decommissioning is scheduled to finish in 2017, and it is presumed that the clearance measurement of building materials will soon start in parallel with the demolition of the facilities. For the clearance of concrete blocks, various methods have been considered for estimating the activity concentration of concrete waste: by a manual survey using a conventional GM survey meter and by an adherence survey using a large surface detector. The activation calculation method has also been considered for estimating the activity collectively [4].

Furthermore, in the decommissioning of a nuclear power plant, it is expected that not only concrete blocks but also concrete segments will require disposal during the dismantlement of the facilities. The necessity of their disposal will also arise at the nuclear power plants in operation as a result of construction and the renovation of equipment. Therefore, a low-level activity measurement technique for the clearance measurement of concrete segments is necessary.

Recently, the authors have developed a new technique for activity evaluation that can estimate a very low activity level and can be used for clearance measurement [5-10]. This new technique consists of 3-dimensional (3D) noncontact laser shape measurement and MCNP (A General Monte Carlo N-Particle Transport Code System) calculation [11] in addition to conventional radiation measurement. The calibration factor and BG correction factor due to the self-shielding effect [12] of the measurement target are automatically determined by Monte Carlo calculations. As a practical system using the new technique, the Clearance Automatic Laser Inspection System (CLALIS) has been developed. The detection ability of CLALIS for metal waste has been clarified in a large number of experiments using mock metal waste samples and standard radioactive sources.

In this study, to discuss in detail the applicability of CLALIS to the clearance measurement of concrete segments, the standard equation for detection limit estimation was extended to separately treat the ambient BG count rate, the BG count rate due to natural radionuclides and their uncertainties. The potential detection limit of CLALIS for the clearance measurement of concrete segments was estimated by the newly -extended equation and the uncertainties evaluated in previous studies.

METHODS

Outline of CLALIS

Figure 1 shows the outline and an internal 3D schematic view of CLALIS. CLALIS comprises four laser scanners for shape measurement, eight large NE102A plastic scintillation detectors with dimensions of 45 cm * 47 cm * 5 cm (thickness) for gamma-ray measurement and a weighing machine for weight measurement. The laser scanners, which are manufactured by Pulstec Industrial Co., Ltd., are located above the measurement area, as shown in Fig. 1. The measurement target, which is placed on the measurement tray of 80 cm * 80 cm area on the weighing machine, is scanned by the four laser scanners. After laser scanning, the shape of the measurement target can be used for Monte Carlo calculation.

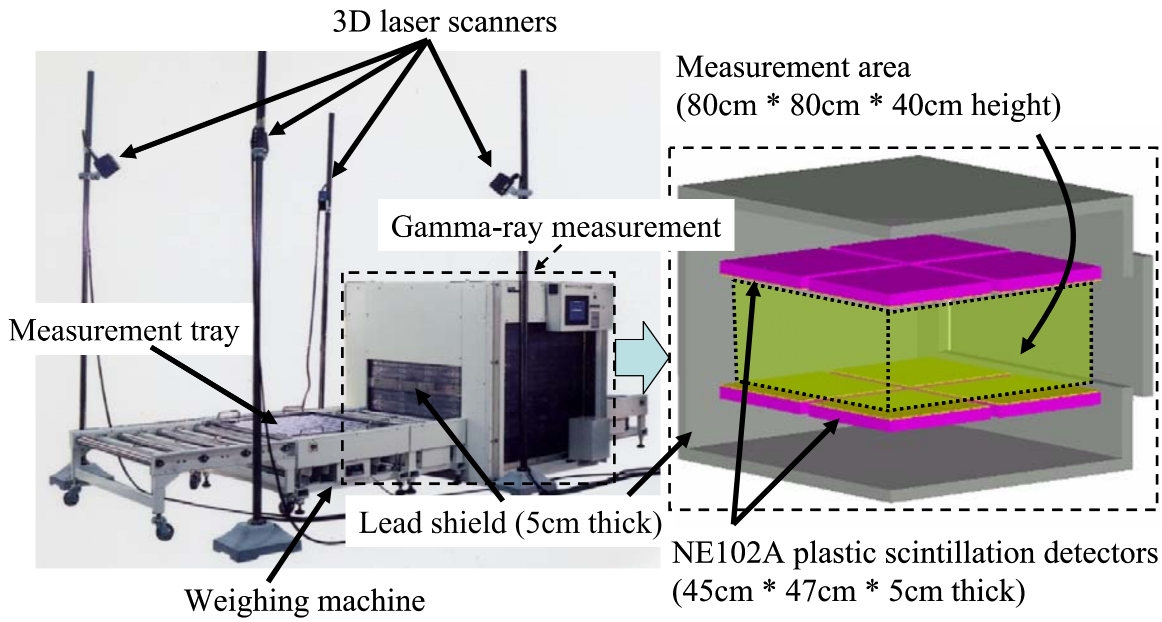

Fig. 1 Outline and an internal 3D schematic view of CLALIS.

For the gamma-ray measurement, the eight scintillation detectors are located face-to-face at the upper and lower sides of the measurement area. There are no detectors on left and right sides of the measurement area because they increases the detection limit by eventually considering the affect on the BG count rate and the efficiency. The size of the measurement area is 80 cm * 80 cm (surface) and 40 cm in height. This area, including the plastic scintillation detectors, is surrounded by a 5-cm-thick lead shield; in addition, the entrance and exit doors are also made of lead so as to reduce ambient BG radiation. The total weight of CLALIS including the lead shield is approximately 6.6 tons.
Figure 2 shows the flow of activity evaluation used in CLALIS. Dotted lines indicate the flow for estimating the count rate due to natural radionuclides.

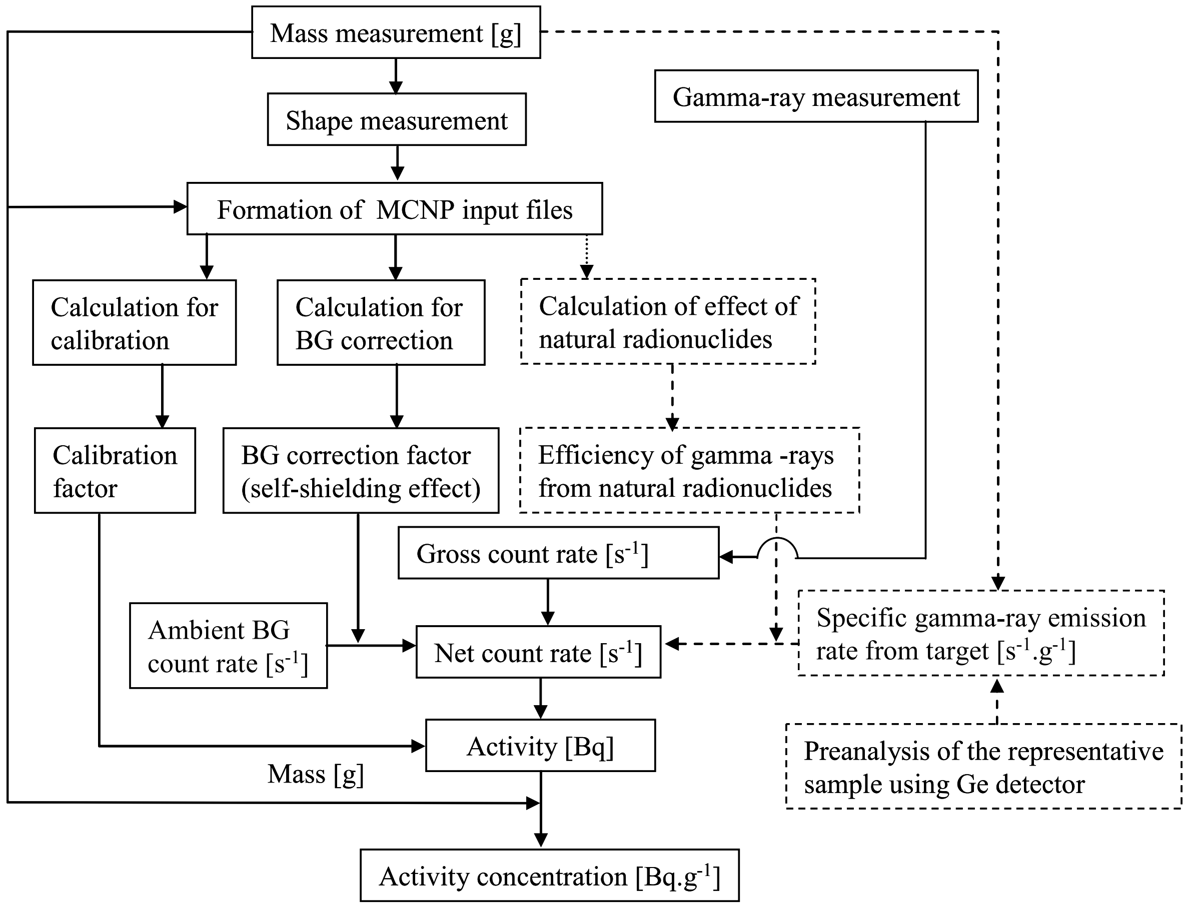

Fig. 2 Flow of activity evaluation used in CLALIS. Dotted lines indicate the flow for estimating the count rate due to natural radionuclides.

Preanalysis of Specific Gamma-Ray Emission Rate

Concrete normally consists of cement, fine aggregate, coarse aggregate, water and admixtures. The cement and aggregates mainly contain natural radionuclides, such as K-40 and the radioactive decay products of Th-232 and U-238, which emit gamma -rays and increase the BG count rate during radiation measurement. Because the activity concentrations of such natural radionuclides depend on where the materials were produced [13], it is necessary to estimate their specific gamma-ray emission rates individually. Moreover, since plastic scintillation detectors are used for gamma-ray measurement in CLALIS, it is impossible to distinguish between the count rates of natural radionuclides and contaminants on the basis of gamma-ray energy information. Therefore, in this flow, the specific gamma-ray emission rate is preanalyzed using a Ge semiconductor detector and the representative sample. Although this measurement should be carried out at least once prior to the clearance measurement, the preanalyzed data can be continuously used as long as the activity concentrations of the natural radionuclides correspond to those of the targets under sequential measurement.

Laser Shape Measurement

In sequential clearance measurements, the mass of the target is first measured. Next, the laser shape measurement is started and the ambient BG count rate is measured at the same time. Since it takes approximately 15 s for each laser scan, it takes approximately 1 min to finish the consecutive scans of the measurement target using the four laser scanners. The instant the shape measurement is finished, the

measurement target is conveyed to the gamma-ray measurement area automatically and the gamma-ray measurement is started.
The data obtained using laser scanners are combined to reveal the shape of the measurement target as a dot image. The dot image is converted into voxels through noise removal. A voxel corresponds to a cube with 2 cm sides in this method. Laser scanning cannot extract information on the internal structure of the measurement target because it can only recognize the surface of the measurement target. For example, when a pipe is measured by laser scanning, it is impossible to determine whether it is a cylinder or a column. In this method, the space under the voxels converted from the surface dot image is filled by voxels to express the shape of the measurement target.

Monte Carlo Calculations

The shape of the measurement target, recognized as an assembly of voxels, is converted into the description used in MCNP [11] input cards. In the input cards, the overestimation of the shielding effect is corrected using the substitutional density evaluated from the volume obtained from voxel numbers and the measured actual mass. When the gamma-ray measurement is carried out, three MCNP calculations are also carried out to obtain the calibration factor and BG correction factors.
In the case of the calibration factor calculation, it is necessary to choose the type of contamination for the measurement target, ‘surface contamination’ or ‘activation’. When surface contamination is chosen, the point source is defined at the most inefficient position on the surface of the measurement target estimated on the basis of the sensitivity distribution of the measurement area so as to evaluate the activity level conservatively. On the other hand, when activation is chosen, the source is defined to be distributed uniformly in the voxels of the measurement target. In the case that Co-60 is the main radionuclide for radiation measurement, gamma-rays of 1.25 MeV are used in the calculation, and the calibration factor is determined using the efficiency and emission rate of the gamma-rays of Co-60.
In the case of calculating the correction factor for decreasing in ambient BG count rate due to the self-shielding effect [12], the BG gamma-ray source is defined as the source at the internal surface of the 5-cm-thick lead shield. The rate of reduction of the ambient BG count rate is estimated on the basis of the ratio of two efficiencies; one is calculated with the measurement target and the other is calculated without the measurement target.
In calculating the efficiency of gamma-rays from the natural radionuclides in the measurement target, the source is defined to be distributed uniformly in the voxels of the measurement target, similarly to in the calibration factor calculation for activation. Then, the gamma-ray energies and their relative emission rates in the MCNP simulation are defined on the basis of preanalyzed data obtained from the representative sample.

Evaluation of Activity Concentration

The activity of the target can be evaluated on the basis of the weight, gross count rate, corrected BG count rate and calibration factor as

$$
\begin{aligned}
&A = C(n_t - n_B f - \varepsilon_{nat} DM) \\
&C = 1/(\gamma\varepsilon) \qquad \text{(Eq. 1)} \\
&f = \varepsilon_{BG} / \varepsilon_{BG0},
\end{aligned}
$$

where

A : activity of the measurement target (Bq)
C : calibration factor (Bq.s)
n_t : gross count rate (s^{-1})
n_B : BG count rate before measurement (s^{-1})
f : BG correction factor for the self-shielding effect

□$_{nat}$: efficiency of gamma-rays from natural radionuclides

D : preanalyzed specific gamma-ray emission rate from the natural radionuclides of the representative

sample ($s^{-1}.g^{-1}$)

M : mass of the measurement target (g)

□ : sum of the emission rates of gamma-rays from radionuclides used for clearance measurement

(for example, 2.000 for Co-60 and 0.851 for Cs-137)

□ : efficiency

□$_{BG}$, □$_{BG0}$: efficiencies of ambient BG gamma rays with and without the measurement target.

The net count rate can be obtained by subtracting the corrected ambient BG count rate, $n_B f$, and the increase in BG count rate due to natural radionuclides, □$_{nat}DM$, from the gross count rate, n_t. Finally, the activity level of the measurement target can be obtained by multiplying the net count rate and the calibration factor. The activity is divided by the mass of the measurement target undergoing clearance level inspection.

Accuracy of Calibration and BG Correction

The accuracy of calibration was experimentally estimated in previous studies using mock metal and mock concrete waste and standard radioactive sources of Co-60, Cs-137 and Eu-152 [5-10]. The mock metal waste samples used in the experiment were JIS 6A, 20A, 40A, 65A, 100A and 250A SUS pipes with lengths of 10, 20, 30, 40 and 80 cm, SUS 20 x 10 cm plates with thicknesses of 1, 2, 3, 4, 5 and 6 cm, SUS 80 x 80 cm plates with thicknesses of 0.5, 1.0, 1.5, 2.0, 2.5, 3.0, 3.5 and 4.0 cm and SUS valves with complex shapes. The mock concrete waste samples were right triangular prisms of two sizes: small triangular prisms of 10 cm height with a right-angled, isosceles triangle cross section with sides of 10, 10 and 14.14 cm, and large prisms of 20 cm height with a right-angled, isosceles triangle cross section with sides of 20, 20 and 28.28 cm.

In total, 268 samples of mock metal waste and 105 samples of mock concrete waste were tested in which the type, position and number of samples were varied. A standard source was attached on the surface of the measurement target and its activity was compared with the CLALIS result. As a result, it was found that the accuracy of calibration was satisfactory; CLALIS can evaluate activity within an accuracy of +/- 20% for metal and concrete waste.

The accuracy of BG correction due to the self-shielding effect was estimated using mock metal waste. In this experiment, we chose 50 samples with a high self-shielding effect from the above 268 samples. The actual reduced BG count rate was easily obtained by carrying out two measurements: one with the measurement sample and the other without it. The accuracy of BG correction due to natural radionuclides was also estimated using mock concrete waste of 60 sample cases by varying its amount. Consequently, the relative uncertainties in calibration, BG correction of the self-shielding effect and BG correction of natural radionuclides were estimated to be 7%, 1% and 1.6%, respectively [6,10]. Table 2 summarizes the results of previous experiments.

Table 2 Summary of results obtained in previous experiments.

Experiment for estimation of uncertainty	Sample material	Type of sample	Number of sample cases	Result	Reference
Calibration	Metal(SUS)	Plate, Pipe, Valve	268	7%	[6]
	Concrete	Segment	105	7%	[10]
BG correction due to the self-shielding effect	Metal(SUS)	Plate, Pipe, Valve	50	1%	[6]
BG correction due to	Concrete	Segment	60	1.6%	[10]

natural radionuclides (Including the self-shielding effect)					

DISCUSSION ON DETECTION LIMIT

Standard Equations for Detection Limit Estimation

Since the activity level that we are concerned with in clearance inspection is very low, the detection limit, which corresponds to the minimum detectable level of a measurement system, is important for characterization. Regarding the detection limit, Currie describes it well in a previous paper [14]. In the United States, the minimum detectable amount (MDA), defined in Currie's paper as the 'detection limit', is usually used as the detection limit [15]. The MDA of the count rate is given by

$$n_{MDA} = \frac{2.71 + 4.65\sqrt{n_B t_B}}{t_t} \,, \qquad \text{(Eq. 2)}$$

where n_{MDA} is the detection limit of the count rate (s^{-1}), t_t is the measurement time (s), t_B is the measurement time for BG (s) and n_B is the BG count rate (s^{-1}).

Generally in Japan, the detection limit of the count rate, n_D, is considered to be three times of the uncertainty of count rate,

$$n_D = 3\sigma_n \,. \qquad \text{(Eq. 3)}$$

Using Eq. 3 and the following equations, which define the net count rate, n_n, and the propagation of uncertainties in radiation measurements,

$$n_n = n_t - n_B \,, \qquad \text{(Eq. 4)}$$

$$\sigma_n = \sqrt{{\sigma_t}^2 + {\sigma_B}^2} \,, \qquad \text{(Eq. 5)}$$

$$\sigma_t = \sqrt{\frac{n_t}{t_t}} \,, \qquad \text{(Eq. 6)}$$

$$\sigma_B = \sqrt{\frac{n_B}{t_B}} \,, \qquad \text{(Eq. 7)}$$

the detection limit of the count rate is given as

$$n_D = \frac{3}{2}\left\{\frac{3}{t_t} + \sqrt{\left(\frac{3}{t_t}\right)^2 + 4n_B\left(\frac{1}{t_t} + \frac{1}{t_B}\right)}\right\}. \qquad \text{(Eq. 8)}$$

Equation 8 has conventionally been used as the detection limit of the count rate in Japan and is qualitatively equivalent to Eq. 2.

Equations for Detection Limit of Activity with Uncertainties in Calibration and BG Correction Taken into Account

Although these equations are essentially correct in many cases, it is impossible to apply them directly to the estimation of the detection limit of the activity of CLALIS because of the uncertainties in the calibration and BG correction. When the uncertainty of the BG count rate is considered, Eq. 6 can be rewritten as

$$\sigma_t = \sqrt{\frac{n_t}{t_t} + r_1^2 n_B^2} \,, \qquad \text{(Eq. 9)}$$

where r_1 is the relative uncertainty in the correction of the decrease in the BG count rate. On the other hand, when the detection limit of the activity, A_D, is given as

$A_D = Cn_D$, (Eq. 10)

$\sigma_{A_D} = \sqrt{\lambda^2 n_D^2 + C^2 \sigma_{n_D}^2}$, (Eq. 11)

where C is the calibration factor and □is the uncertainty in the calibration. When the detection limit of activity is also considered to be three times the uncertainty of the activity, as in Eq. 3, the detection limit of activity can be given as

$A_D = 3\sigma_{A_D}$. (Eq. 12)

Using Eqs. 3, 4, 5, 6, 9, 10, 11 and 12, the detection limit of the count rate is expressed as

$$n_D = \frac{\frac{3^2 C^2}{t_t} + \sqrt{(\frac{3^2 C^2}{t_t})^2 + 4(C^2 - 3^2 \lambda^2)3^2 C^2 \{n_B(\frac{1}{t_t} + \frac{1}{t_B}) + r_1^2 n_B^2\}}}{2(C^2 - 3^2\lambda^2)} . \quad \text{(Eq. 13)}$$

Then, the relative uncertainty in the calibration is defined as r_2,

$r_2 = \frac{\lambda}{C}$, (Eq. 14)

and the detection limit of activity is finally expressed as [4,5]

$$A_D = C \cdot \frac{\frac{3^2}{t_t} + \sqrt{(\frac{3^2}{t_t})^2 + 4(1 - 3^2 r_2^2)3^2 \{n_B(\frac{1}{t_t} + \frac{1}{t_B}) + r_1^2 n_B^2\}}}{2(1 - 3^2 r_2^2)} . \quad \text{(Eq. 15)}$$

Using this equation and uncertainties shown in Table 2, the detection limit of CLALIS can be estimated to be approximately 100 Bq for metal scraps and concrete segments [6, 10].

Extension of Detection Limit Equation

Although, Eq. 15 is essentially adequate for estimating the detection limit when the uncertainly of the BG count rate should be considered, it is desirable to improve the uncertainty of the BG count rate to estimate the detection limit of CLALIS for the clearance measurement of concrete segments with high accuracy. In this case, the BG count rate, which corresponds to "*the count rate to be subtracted from the total count rate*", is composed of the contributions of ambient BG radiation and the natural radionuclides in the measurement target, as indicated in Eq. 1.

The uncertainty of the BG count rate due to ambient BG radiation was estimated experimentally in the previous study as mentioned before. When we consider the uncertainty of the BG count rate due to natural radionuclides, there may be two types of uncertainty that require correction. One is the systematic uncertainty, which is caused by the use of the MCNP correction method shown in Fig. 2, and this uncertainty was estimated to be 1.6% in a previous study [10].

On the other hand, if there is an uneven distribution of natural activity concentration among the concrete waste, the value of *D,* which is the preanalyzed specific gamma-ray emission rate from natural radionuclides of a representative sample, will differ between the representative sample and the measurement target. Thus, Eq. 15 can be extended to treat these two uncertainties separately as

$$A_D = C \cdot \frac{\frac{3^2}{t_t} + \sqrt{(\frac{3^2}{t_t})^2 + 4(1 - 3^2 r_2^2)3^2 \{n_B(\frac{1}{t_t} + \frac{1}{t_B}) + r_1^2 n_B^2 + n_{nat}(\frac{1}{t_t} + \frac{1}{t_B}) + r_3^2 n_{nat}^2\}}}{2(1 - 3^2 r_2^2)} , \quad \text{(Eq. 16)}$$

where r_3 is the relative uncertainty due to the uneven distribution of the natural activity concentration and n_{nat} is the count rate due to the natural radionuclides in concrete segments.

Uncertainty due to Uneven Distribution of Natural Activity Concentration

When we use Eq. 16 to estimate the detection limit, it is necessary to evaluate the r_3, which strongly depends on the size of region considered when defining a representative sample. If a representative sample is defined as a representative sample of all concrete material produced in Japan, the value of r_3 increases because the natural activity concentration varies with the location of production and the type of aggregate [13]. Therefore, in this study, we referred to the data cited from reference [17] as typical values of r_3. Table 2 shows the average natural activity concentration and the range of fluctuation, i.e., the uncertainty due to the uneven distribution of natural activity concentration in the concrete waste of Japanese nuclear power plants. The concrete waste resulted from the demolition of buildings when the steam generators were replaced. This data was investigated to determine whether the concrete waste was "nonradioactive waste" and whether the activity concentration by activation with neutrons was sufficiently small compared with the actual range of fluctuation of the natural activity concentration. Using this data, the relative uncertainties for the uneven distribution of natural activity concentration for the concrete waste at three different nuclear power plants were estimated to be 15%, 6.8% and 8.1%, as shown in Table 3.

Table 3 Examples of uneven distribution of natural activity concentration of concrete waste.

Nuclear Power Plant (NPP)	Average [Bq.g^{-1}]	Range of fluctuation [Bq.g^{-1}] (3□)	Relative Uncertainty [](□)
Mihama NPP Unit 1	0.768	0.342	15
Mihama NPP Unit 3	0.901	0.183	6.8
Takahama NPP Unit 1	0.855	0.207	8.1

Detection Limit of CLALIS for Clearance Measurement of Concrete Segments

The detection limit of CLALIS for concrete segments was estimated using Eq. 16 and the uncertainties shown in Tables 2 and 3. Figure 3 shows the detection limit of CLALIS for Co-60 as a function of the mass of the concrete segment in the measurement. Open circles indicate the detection limit when the uncertainty due to the uneven distribution of natural activity is negligible. In the previous experiment on 105 samples [10], we used mock concrete waste that has a mortar constituent. Since the mortar is composed of cement and fine aggregate, the uncertainty due to the uneven distribution of natural activity is negligible because it is expected that this uncertainty which is shown in Table 3, principally depends on the content rate of coarse aggregate. As shown in this figure, even though the masses of the samples are the same, the detection limits vary since the calibration factor or BG correction factors depend on the position and size of the measurement target.
In Fig. 3, closed circles indicate the average values of the detection limit when $r_3 = 0$. On the other hand, closed and open triangles indicate the detection limit when r_3 is 0.08 and 0.15, respectively. As shown in the figure, the detection limit increases with r_3 and it becomes closer to the clearance level of Co-60. Since these detection limits are estimated in the case that the key radionuclide is Co-60, the detection limits will further increase when the key radionuclide is Cs-137 or Eu-152.

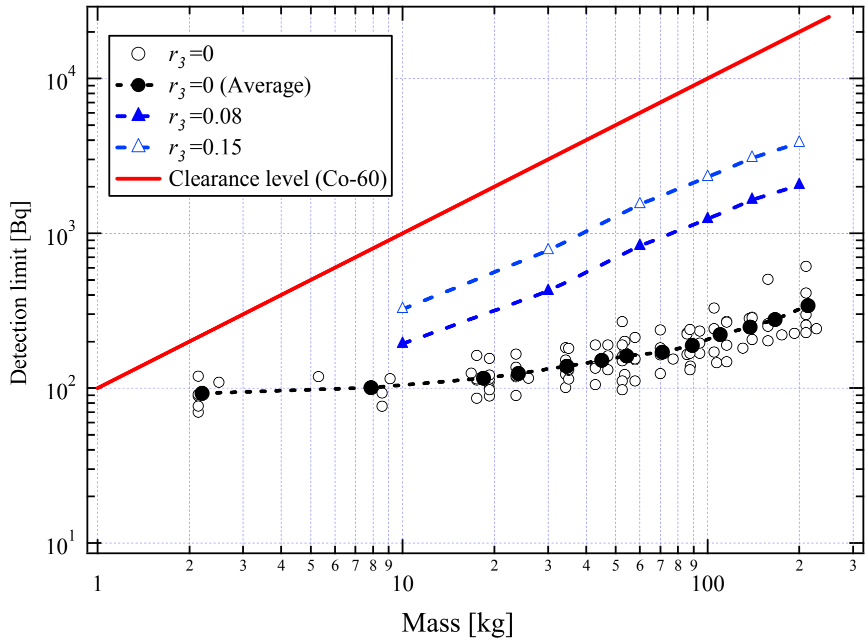

Fig. 3 Detection limit of CLALIS for Co-60 as a function of mass of the measurement target.

Additionally, the relative importance of the key radionuclide is not considered in this estimation. In actual clearance inspections, it is necessary to prove that the sum of the ratios of the activity concentration for each radionuclide (D_i) to its individual clearance level (C_i), must be less than 1 [16], i.e.,

$$\sum_i (Di/Ci) \leq 1. \qquad \text{(Eq. 17)}$$

The activity concentrations for Co-60, Cs-137 and Eu-152 can be estimated by gamma-ray measurement. The activities of alpha -emitters, such as Am-241 and Pu-239, may be estimated using the radionuclide component ratio, i.e., a scaling factor [18] or nuclide vector. In this case, if the relative importance of the key radionuclide is taken into account, the activity concentration level used as the judgment criterion for the key radionuclide becomes smaller than the original clearance level.

Applicability of CLALIS

By considering the relative importance of Co-60, the BG count rate due to natural radionuclides and uncertainty due to the uneven distribution of natural activity concentration, the range of application of CLALIS was roughly estimated using Eq. 16 as indicated in Fig. 4. In this figure, the X axis indicates the value of r_3 and the Y axis indicates the natural activity concentration of K-40. Although, the natural radionuclides include not only K-40 but also the radioactive decay products of Th-232 and U-238, we choose the natural activity concentration of K-40 as a parameter since K-40 is the dominant radionuclide contributing to the BG count rate. The BG count rate due to natural radionuclides was considered to be 4550 $(s^{-1})/(Bq.g^{-1})$ when the mass of the measurement target was 200 kg on the basis of the results in the previous study. Because the clearance level of Co-60 is $(0.1\ Bq.g^{-1})$ as shown in Table 1, it is necessary to confirm whether the activity of the measurement target is less than 20000Bq in this case. However, actually in Japan, the detection limit of such clearance monitors should be less than one tenth of the judgment activity level. Thus, in this study, one tenth of the clearance level of Co-60 was used for estimating the applicability of CLALIS.

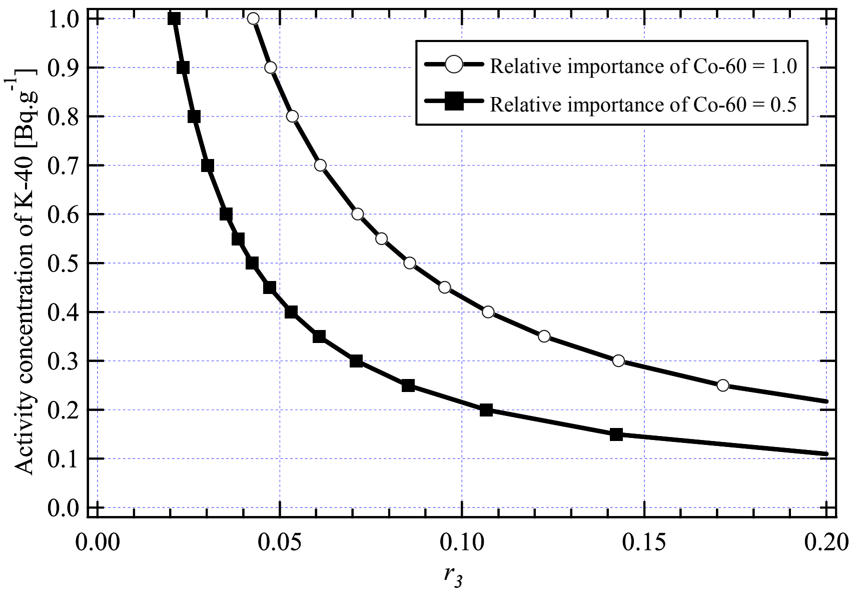

Fig. 4 Range of application of CLALIS for clearance measurement of concrete segments.

In the figure, there are two curves drawn as the boundaries of applicability, one is for the case that the relative importance of Co-60 is 1.0 and the other is for the case that the relative importance of Co-60 is 0.5. Regarding the concrete segments to be measured, when the natural activity concentration of K-40 and the uncertainty due to the uneven distribution of the activity concentration remain under these curves, it is possible to apply CLALIS for clearance measurement. However, it is impossible when these parameters are above these curves. According to reference [19], Eu-152 is also an important radionuclide in the clearance measurement of concrete waste. In the case that the main target radionuclide is Eu-152, the detection limit increases and becomes similar to that for when the relative importance of Co-60 is 0.5. Thus, the range of application of CLALIS strongly depends on the following four factors.

(1) Natural activity concentration
(2) Relative uncertainty due to unevenness of distribution of natural activity, r_3
(3) Relative importance of key radionuclide
(4) Type of key radionuclide

From a practical standpoint, to decrease the natural activity concentration and to change the relative importance or the key radionuclide are impossible. Therefore, the user should assess the appropriateness of the value of r_3 by considering the range of application of representative sample ahead of the sequential clearance measurement.

CONCLUSIONS

For the clearance measurement of decommissioning waste, CLALIS, which enables the measurement of very low level activity by gamma-ray measurement, laser scanning and Monte Carlo calculation techniques, has been developed. CLALIS comprises four laser scanners, eight large NE102A plastic scintillation detectors and a weighing machine.

In this study, the effect of the uneven distribution of natural radionuclides between the measurement targets was considered and the applicability of CLALIS to the clearance measurement of concrete segments was discussed minutely. The standard equation for detection limit estimation was extended to separately treat the ambient BG count rate, the BG count rate due to natural radionuclides and their uncertainties. Consequently, the precise detection ability of CLALIS was evaluated and its range of application was demonstrated. It was found important to evaluate the uncertainty sufficiently carefully by considering the natural activity concentration, the relative uncertainty of unevenness of distribution of natural activity and the relative importance of the key radionuclide.

In this system, taking account of the opening and closing times of lead doors and conveying time, it takes approximately 200 sec to accomplish the inspection of one target. Since shape measurement and gamma-ray measurement can be carried out independently and consecutively, both clearance level and surface contamination density limit inspections can be achieved in a 100 sec cycle. This indicates that 5 tons of metal waste per day (1,000 tons per year) can be measured in batches of more than 20 kg at that process speed. Furthermore, since all processes are fully automatic, it is possible to eliminate errors caused by the human factor and to evaluate activity objectively.

We hope that CLALIS will be a helpful tool that can provide reliability and quality assurance in the rational clearance of nuclear plant waste materials.

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