Electronic Dose Conversion Technique Using a NaI(Tl) Detector for Assessment of Exposure Dose Rate from Environmental Radiation

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Abstract
An electronic dose conversion technique to assess the exposure dose rate due to environmental radiation especially from terrestrial sources was developed. For a 2x2 inch cylindrical NaI(Tl) scintillation detector, pulse-height spectra were obtained for gamma-rays of energy up to 3 MeV by Monte Carlo simulation. Based on the simulation results and the experimentally fitted energy resolution, dose conversion factors were calculated by a numerical decomposition method. These calculated dose conversion factors were then, electronically implemented to a developed dose conversion unit (DCU) which is a microprocessor-controlled single channel analyzer (SCA) with variable discrimination levels. The simulated spectra were confirmed by measurement of several monoenergetic gamma spectra with a multichannel analyzer (MCA). The converted exposure dose rates from the implemented dose conversion algorithm in the DCU were also evaluated for a field test in the vicinity of the nuclear power plant at Kori as well as for several standard sources, and the results were in good agreement with separate measurement by a high pressure ionization chamber (HPIC) within a 6.4 % deviation.

II. MATERIAL AND METHOD
A. Electronic Dose Conversion System
The exposure dose rate can be calculated from pulse-height spectrum of any gamma-ray spectroscopy system by the following simple equation in principle;

\[
\hat{X}_{air} = \int_{E_{min}}^{E_{max}} S(E) \times G(E)dE
\]

where \(S(E)\) is the measured spectrum and \(G(E)\) is the dose conversion factor specific to the detector in use. Also \(E_{min}\) and \(E_{max}\) are the low and the upper bound of the pulse-height spectrum, respectively. Figure 1 (a) shows a typical spectroscopy system using a NaI(Tl) scintillation detector and an MCA. However this method of dose conversion is not practical in the real-time environmental radiation monitoring because it requires a step of a numerical calculation and it is hard to maintain an MCA in good condition when used outdoors.

I. INTRODUCTION
Since the gamma-ray exposure is proportional to the amount of ionized charges in air by definition, usually a high pressure ionization chamber (HPIC) is used as the radiation sensor in the environmental radiation monitoring system (ERM) around nuclear power plants. However, since an HPIC is operated in current mode, it is impossible to access the contribution of man-made radionuclides to the total measured exposure. In addition, the uncertainty becomes significant in measurement of low-level radiation with an HPIC.

Therefore highly efficient solid-state detectors such as a NaI(Tl) scintillation detector have recently been adapted to ERM in several countries[1] because of its simplicity in operation and maintenance and, more importantly, because of its capability of distinguishing man-made radiation from natural background radiation[2], which will not be discussed in this paper.

Generally NaI(Tl) scintillation detectors are used in gamma-ray spectroscopy system with a multichannel analyzer (MCA), but, in this study, we proposed a simple electronic dose conversion method, where a NaI(Tl) detector can be used to measure the exposure dose rate in air without using an MCA.

Figure 1: Configurations of typical spectroscopic system for the gamma-ray spectrum (a) and of a developed, in this study, measurement system for exposure dose rate (b)

S. Moriuchi et al. developed a simple method to estimate the exposure dose rate for the real-time monitoring using a NaI(Tl) detector, a single channel analyzer (SCA) and a specific pulse generator[3]. By repetitively modulation of the low-level discriminator of the SCA in accordance with the waveform from the pulse generator, the integration of (1) can be accomplished.

While keeping the upper level discriminator (ULD) to the
level equivalent to 3 MeV which covers the energy range of environmental radiation for the exposure dose in air and modulating the low level discriminator (LLD) as a waveform of the inverse $G(E)$ function, we can convert the series of analog signal pulses into a count rate that is directly proportional to the exposure dose rate in air, $\dot{X}_{air}$. The proportionality constant is set to $G(E)$ value where $E$ is 3 MeV.

S. Maurich also calculated $G(E)$ by a simple successive integration assuming the flat Compton continuum, which is rather a poor approximation. Also their waveform produced by an uni-junction transistor circuit did not simulate the dose conversion factor very accurately[4].

Therefore in order to utilize the $G(E)$ function more exactly, a dose conversion unit (DCU) is designed and manufactured in this study, of which schematic is shown in Figure 1 (b). The main elements of DCU are an SCA, a read-only-memory (ROM), a digital-to-analog converter (DAC) and a microprocessor. The calculated data of the inverse $G(E)$ function are stored in the ROM and are converted into a modulation waveform through the DAC and fed into the LLD terminal of the SCA.

B. Dose Conversion Factor, $G(E)$

In gamma-ray spectroscopy, the measured spectrum can be expressed by the following integral form

$$S(E) = \int \phi(E') \times R(E,E')dE'$$

(2)

where the kernel $R(E,E')$ is the detector response-function or a pulse-height spectrum for a single incident gamma-ray with energy $E'$, and $\phi(E')$ is the incident gamma-ray energy spectrum. If we substitute (2) into (1) and rearrange the integrands, then the exposure dose rate is presented as

$$\dot{X}_{air} = \int_{E_{min}}^{E_{max}} \int \phi(E') \times G(E)dE'$$

(3)

where the integral in the bracket can be written as $\dot{X}_{air}(E')$ and is interpreted as the exposure dose rate due to a single incident gamma-ray with the energy $E'$. It can be calculated with its definition as shown as

$$\dot{X}_{air}(E') = \frac{\mu_{en}}{\rho} \times \frac{q}{W} \times E'$$

(4)

where $(\mu_{en}/\rho)_{air}$ is the mass energy-absorption coefficient of air, $q$ is the electronic charge ($=1.6 \times 10^{-19}$ Coulomb) and $W$ is the average energy spent to produce an ion pair in air ($\sim 34$ eV). Since the response-function, $R(E,E')$ cannot be obtained in an analytic form over all energies of interest practically, if we take discrete data set, the dose conversion factor, $G(E)$, can be determined by

$$G_i = R_{ij}^{-1} \times \dot{X}_{j}$$

(5)

where $i$ is the index of pulse height interval and $j$ is the index of the gamma-ray energy discrete interval. In principle, $G_i$ is dependent upon the several measurement conditions, such as the energy range of interest, the intensity of dose rate and the directionality of incident gamma-ray etc. Practically $E_{min}$ and $E_{max}$ are set to the electronic noise level and the maximum detectable energy of gamma-ray due to the finite size of the detector respectively. In this study, they are set to 50 KeV and 3 MeV respectively.

C. Monte Carlo Calculation

In 1995, J. S. Jun et al. has calculated $G(E)$ for 2x2 inch and 3x3 inch cylindrical NaI(Tl) scintillators using matrix inversion method based on the gamma-ray spectra obtained by Monte Carlo simulation[5]. But Jun's calculation assumed that the gamma-rays are incident onto the top surface of the cylindrical NaI(Tl) crystal. By extensive Monte Carlo calculation, however, R. L. French has proved that the major part of the gamma-rays is incident onto the wall when the cylindrical NaI(Tl) detector is positioned upright at 3 ft above an infinite planar source[6].

Therefore, we have recalculated the $G(E)$ of a 2x2 inch NaI(Tl) detector enclosed by an aluminum housing and exposed to gamma-rays incident in parallel with the earth surface. The NaI(Tl) scintillation detector was simplified as shown in Figure 2. Due to the symmetry of the detector shape, only a quarter of parallel beam planar source was considered.

Figure 2: Monte Carlo geometry used in determining the gamma-ray response-function for a 2 x 2 inch cylindrical NaI(Tl) scintillation detector. Because of the symmetry, a quarter of source is used. Described numbers are in centimeters.

First, in order to obtain the response-function matrix, $R_{ij}$, for gamma-ray of energy up to 3 MeV with 10 keV energy bin, we used Monte Carlo N-Particle Transport Code, version 4A (MCNP 4A)[7]. Then to consider the spectrum broadening effect due to the gain function in photo-multiplication tube and the electronic noise of amplifier in use, we corrected the spectra using a Gaussian shape broadening function. The standard deviation for each energy bin was obtained from the linear fitting of the measured value for several standard sources and it became

$$\sigma(E) = 10.2 + 0.0134 \times E$$

(6)

where $\sigma$ and $E$ have the dimension of keV. Secondly, the
values of dose conversion coefficient, $x_{air}$, were obtained from the ICRU (International Commission on Radiation Units and Measurements) report 47[8]. Finally, based on these calculated $x_{air}$ for an arbitrary energy of gamma-ray and the numerically inversed response-function, $R_y^{-1}$, we calculated the dose conversion factor, $G(E)$ by (5).

D. Experimental Verification

We performed several measurements in order to assure the accuracy of the calculated dose conversion factor and the performance of the new DCU-based system.

First, in order to check the accuracy of the step of $G(E)$ calculation indirectly, the pulse-height spectra were measured and analyzed for several monoenergetic gamma-ray sources such as $^{85}$Sr (0.514 MeV), $^{54}$Mn (0.854 MeV), $^{60}$Co (1.173 & 1.332 MeV) with the system as shown in Figure 1 (a). The source-detector distance was set to 1 meter, which approximately produces a broad parallel beam. For the spectrum analysis and the direct comparison between measurement and simulation, the channel of measured spectra were reset to the same channel of simulation and background spectra were subtracted from them.

Secondly, the operation of DCU was checked by comparison with calculation as well as separate measurements by HPIC (Reuter-Stokes RSS-1013). In order to see the dose rate effect, we varied the source-detector distance of the 92.571 $\mu$Ci $^{60}$Co source. Before the field test, indoor operation of the developed system for couple of days was performed to check the long-term stability.

Finally, we set the developed DCU system in the Kori nuclear power plant site and measured over 10 days for the field test. In this experiment, $^{137}$Cs sample source was introduced temporarily as an artificial fluctuation against background radiation.

III. RESULTS AND DISCUSSION

A. Spectrum and Dose Conversion Factor

Figure 3 shows the simulated and measured spectra of gamma-ray from $^{85}$Sr, $^{54}$Mn and $^{60}$Co sources. The photopeak efficiencies of the simulated and measured spectra agree quite well with each other, but there are non-negligible discrepancies especially at the low energy region of the spectrum. It seems due to the following reasons: (i) the back scatter peak in the vicinity of 0.2 – 0.25 MeV originated from the large-angle scattering of gamma-rays with surrounding materials of the detector crystal[9], (ii) the external bremsstrahlung photons emitted from the interaction of the accompanied beta-rays with the gamma decay of $^{85}$Sr, $^{54}$Mn and $^{60}$Co[10], and (iii) the pulse pile-up effect around the photo-peak due to high incidence rate of gamma-ray in comparison with the finite resolving time (~27 usec) of the multichannel analyzer used in measurements[9].

However these effects are not significant when the proposed system is used around nuclear power plants because it uses no MCA and the environmental radiation dose is normally low level and the gamma-emitting sources are widely spread over the hemisphere.

Figure 3: Comparisons between the measured spectrum by NaI(Tl) scintillation detector with MCA and the simulated spectrum by Monte Carlo method for (a) $^{85}$Sr (0.514 MeV), (b) $^{54}$Mn (0.854 MeV) and (c) $^{60}$Co (1.173 & 1.332 MeV) sample sources.

The calculated $G(E)$ for the horizontal incidence case is plotted in Figure 4 together with that for the vertical incidence case[11]. The horizontal $G(E)$ is a little lower at low energy and a little higher at high energy of the spectrum.
than that for the vertical incidence case, and it is due to the geometrical dependency of the detection efficiency.

Figure 4: Dose conversion factor, $G(E)$, as a function of incident gamma-ray energies from 10 keV to 3 MeV, which practically covers the gamma energy range of artificial and terrestrial nuclides.

B. Exposure Dose Rate

Figure 5 shows the measured and calculated exposure dose rates due to a $^{60}$Co source located at several source-detector distances. The data measured by the NaI(Tl)-based DCU system show good agreement with the calculation up to the level of 100 μR/hr. Also when compared with the measured value by an HPIC, there is a constant deviation of 0.23 μR/hr, which is considered as the contribution from the ionizing component of cosmic-rays over 3 MeV. This result is consistent with others[2, 5]. After subtraction of the cosmic-ray contribution, the maximum difference is only 6.4 % when the distance is 4 meters.

Figure 5: Comparison between calculated and measured dose rates as a function of source-detector distance for $^{60}$Co (1.173 & 1.332 MeV) with the activity, 92.571 μCi.

Figure 6 shows the measured exposure dose rate for 10 hours with and without $^{60}$Co source at several different locations. Finally, Figure 7 is the result of the field test for 10 days in the Kori nuclear power plant site. The new system seems quite stable and reasonably sensitive to show the increase of radiation level due to the rain-fall and due to introduction of $^{137}$Cs test source of which radioactivity produces ~0.58 μR/hr.

Figure 6: Measured dose rates for indoor stability test.

Figure 7: Measured dose rates in the Kori nuclear plant site.

IV. CONCLUSION

We have calculated the dose conversion factor, $G(E)$, for a 2×2 inch NaI(Tl) detector based on the incidence of monoenergetic gamma-rays parallel to the earth surface. And an electronic dose conversion unit was developed to estimate the environmental dose rate directly without using an MCA. The experimental test of the new system with standard gamma sources showed good agreement with the calculated exposure dose rate. It also yielded consistent results to HPIC measurement except for the cosmic ray contribution to the total exposure in HPIC. Therefore the developed system using a NaI(Tl) detector and the electronic dose conversion unit can be used to monitor the low-level range of environmental radiation accurately and reliably.

V. REFERENCES


