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Performance of a Large Hyperpure Germanium Detector Array for In-Vivo Detection of Low-Energy Photon and X-Ray Emitters: Analytical Procedure and Current Capabilities

C. D. Berger
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PERFORMANCE OF A LARGE HYPERPURE GERMANIUM DETECTOR ARRAY FOR IN-VIVO DETECTION OF LOW-ENERGY PHOTON AND X-RAY EMITTERS: ANALYTICAL PROCEDURE AND CURRENT CAPABILITIES

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Abstract

The ORNL Whole Body Counter is one of the few in the country that is capable of assessment of body burdens of low-energy photon and x-ray emitters. This requires detectors and electronics specific for the task. Isotope identification and quantification capabilities are crucial at the facilities which deal with quantities of many radionuclides, and the ORNL Whole Body Counter is such an installation.

The standard procedure for whole body counting done at ORNL employs a 13.34 cm-dia. phoswich (positioned over the left lung field), a hyperpure germanium (HPGe) array consisting of six detectors totaling 80 cm^2 (over the right lung field), and a 23 cm by 23 cm NaI detector positioned under the subject's back.

Isotope identification and quantification capabilities for detection of the actinides using the HPGe array are greatly improved over the phoswich system, making the HPGe detector an integral part of lung counting operations at ORNL, and the subject of this report. The HPGe array has been fully operational since May 1980. Since that time, calibrations have been performed with various sources, a calibration curve derived, and minimum significant measured activity (MSMA) and minimum detectable true activity (MDTA) for various radionuclides have been determined. This report includes a discussion of the current analysis techniques, gives examples of MSMA and MDTA for various isotopes, and discusses the derivation of a universal calibration curve for preliminary estimation of body burden, using data acquired by the HPGe array.

Introduction

The ORNL Whole Body Counter is one of a few that is capable of assessment of organ burdens of low-energy photon and x-ray emitters. Such capability requires detectors and electronics specific for the task. Isotope identification and quantification capabilities are needed at facilities which deal with quantities of many radionuclides, and the ORNL Whole Body Counter suits that need.

In the standard procedure for whole body counting at ORNL, a 13.34 cm-dia. NaI(Tl)-CsI(Na) phoswich (positioned over the left lung field), a hyperpure germanium (HPGe) array consisting of six detectors totaling 80 cm^2 (over the right lung field), and a 23 cm by 23 cm NaI(Tl) detector positioned under the subject's back (see Fig. 1) are used.

Isotope identification and quantification capabilities for detection of the actinides with the HPGe array are greatly improved over those with the phoswich system,[1] thus, the HPGe detector is an integral part of lung counting operations at ORNL, and the subject of this report. The HPGe array has been fully operational since May 1980. Since that time, calibrations have been performed with various sources, a calibration curve derived, and minimum significant measured activity (MSMA) and minimum detectable true activity (MDTA) for various radionuclides have been determined. This report includes a discussion of the current analysis techniques, gives examples of MSMA and MDTA for various isotopes, and discusses the derivation of a universal calibration curve for preliminary estimation of organ burden, using data acquired by the HPGe array.

The entire ORNL whole body counting system is under continuing development, expansion and improvement. The information contained herein, although not static, may be useful to those who use or would like to use the whole body counter services. ORNL/TM-7477 contains a detailed review of the whole body count procedure.[2]

Detection System

The HPGe system consists of six individual detectors in a fixed, closely packed array. The signal from each detector passes through a preamplifier and research amplifier into a gated analog router where all

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Figure 1.

six signals are multiplexed to function as a single counting unit. The router accepts the first qualified analog input, rejecting simultaneous inputs. The accepted signal is then digitized by an analog-to-digital converter which is linked to a minicomputer system and multichannel analyzer. A block diagram of these electronics is in Figure 2. Routine count time is 2600 seconds ($\sim$ 43 minutes) for each subject.

Analysis Procedures

The data handling procedure for the HPGe detector has been described previously in two reports.[1,2] To date, there are three basic computer programs involved: a radionuclide identification program; a least-squares fitting program for isotope quantification; and a routine identification and quantification program for ^{239}Pu and ^{241}Am. The first two programs are called up separately when non-routine peaks are noted in a subject's spectrum and are under continuing expansion and development.

The third program mentioned above is run for every routine whole body count. After acquisition of gross spectral data of a subject, competing background radiations (from a 50,000 sec phantom count) are subtracted channel-by-channel resulting in a net subject spectrum. A typical example is shown in Figure 3. Note that the energy range of this spectrum is 0 to approximately 130 keV. In a normal human spectrum, counts in this region are the result of scattered radiation from whatever high-energy photon emitters are in the subject's body (e.g., ^{40}K, ^{137}Cs, etc.). In the case of internal deposition of low-energy photon or x-ray emitters (e.g., ^{239}Pu, ^{125}I, ^{241}Am, etc.), excess counts in specific channels will be noted.

At this point the ^{239}Pu and ^{241}Am energy regions are observed to determine if there are statistically positive counts. Direct comparison of total net counts in these regions to those in an experimentally determined standard would be difficult due to the variable amounts of ^{40}K found among individuals, resulting in variable levels of low-energy scattering. The presence of a non-naturally occurring radionuclide that also emits high-energy photons would confuse the situation further. We therefore chose to compare what we will call the "Pu-ratio" and "Am-ratio" of an individual subject with the ratios for our standard. These are the ratios of integrated

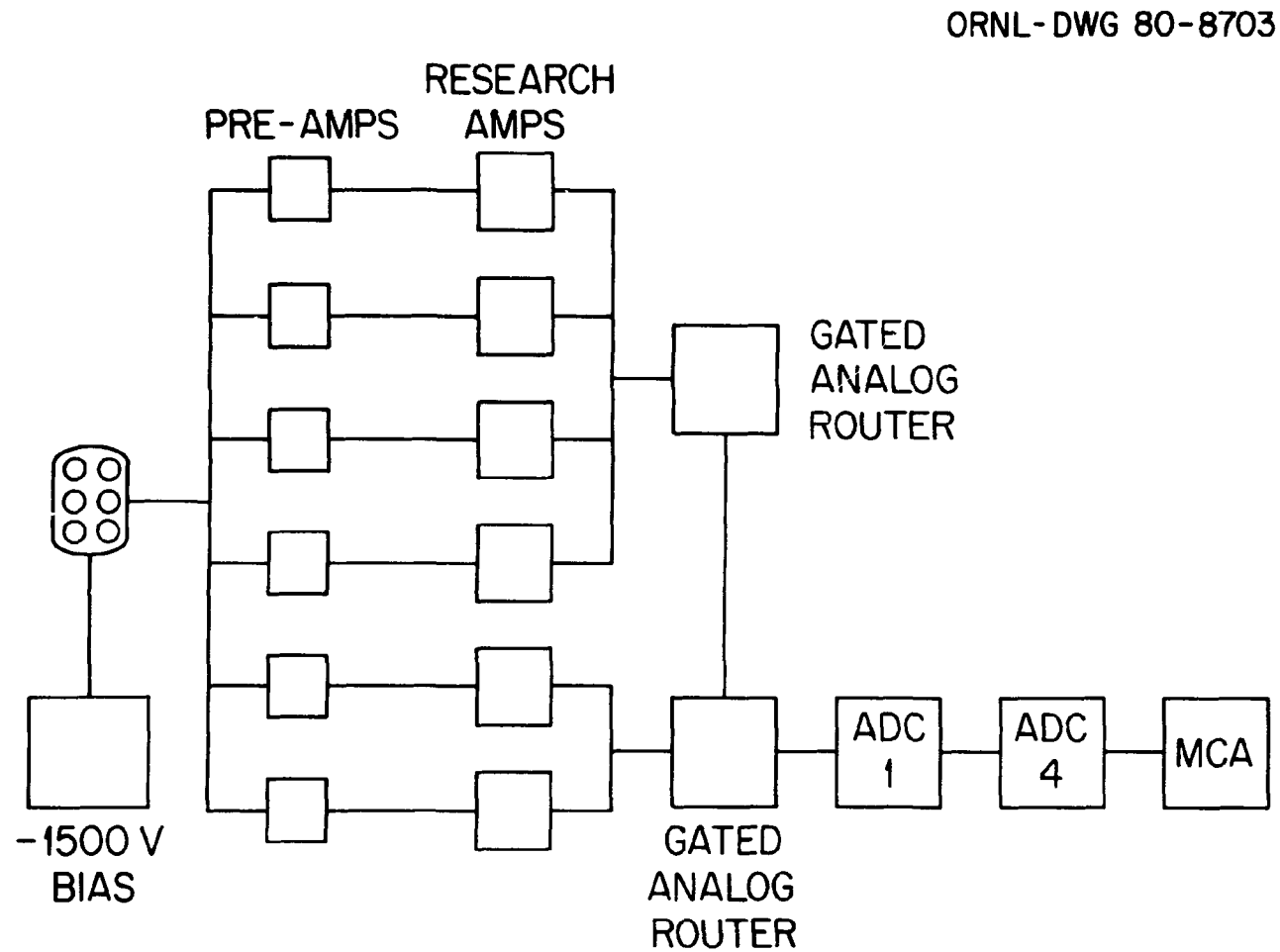


Figure 2. Block diagram of HPGe electronics system. Two ADCs are used: ADC 1 converts data from each detector individually; ADC 4 converts and displays a single multiplexed spectrum.

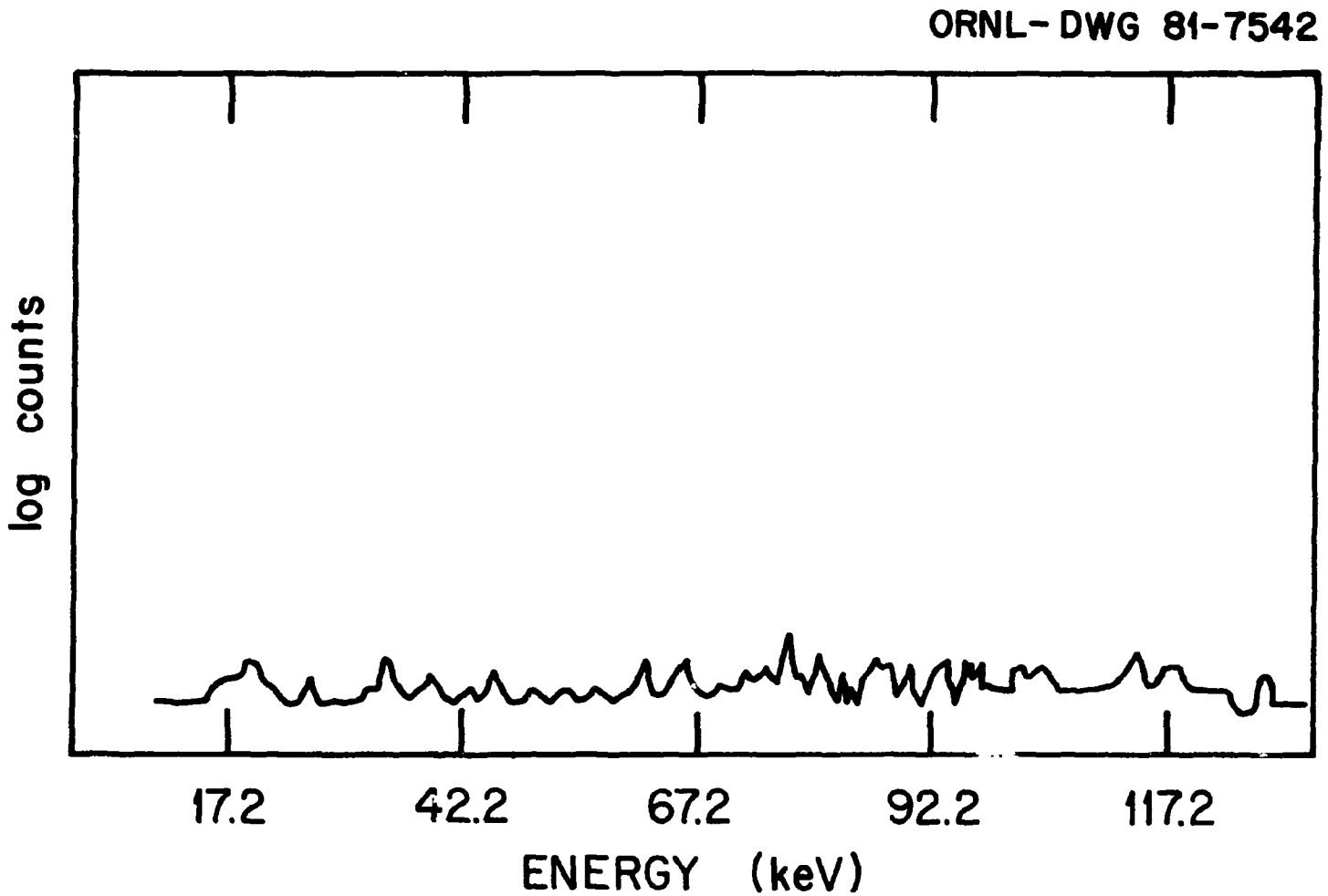


Figure 3. Typical net subject spectrum.

counts in the ^{239}Pu or ^{241}Am energy regions to the integrated counts in some control region (see Figure 4). That is,

$$\text{Am-ratio} = \frac{\int_{\varepsilon_1}^{\varepsilon_2} (\text{Counts in Am region})\, d\varepsilon}{\int_{\varepsilon_3}^{\varepsilon_4} (\text{Counts in control region})\, d\varepsilon}$$

where, ε_1 and ε_2 are chosen to be 1.18 FWHM of the energy peak, which has been demonstrated to maximize the ratio of signal-to-noise.[1] ε_3 and ε_4 were chosen arbitrarily as 20 channels to the right of the full-energy peak. These ratios (Pu and Am) were found to be constant to within ± 1.03% for 138 adult male, non-radiation workers and ± 2.62% for 52 adult female, non-radiation workers. It has also been found that the presence of other higher-energy photons (> 80 keV) does not result in wider fluctuations in the ratios than those noted above.

The routine ^{239}Pu, ^{241}Am analysis program uses this information to predict each subject's background count rate under the ^{239}Pu and ^{241}Am energy peaks. If there are excess counts in this region for any given subject, their statistical significance is determined using a combination of Bayes and Nieman-Pearson criteria,[3] and converted to becquerels (nanocuries) or % organ burden using predetermined calibration factors.

Minimum Detectable Activity

To explain the significance of threshold determinations and the capabilities of the ORNL Whole Body Counter low-level counting techniques, the statistical aspects of determining a threshold value or quantity of some radioisotope deposited inside a human chest will be discussed in this section. The lower limit of detection for the ORNL Whole Body Counter may be expressed statistically in terms of two minimal activities determined by the maximal acceptable risks of making a Type I error (of concluding there is internally deposited activity when there is none) and a Type II error (of concluding that there is no activity in the body when there actually is some). These threshold activities are designated "the minimum significant

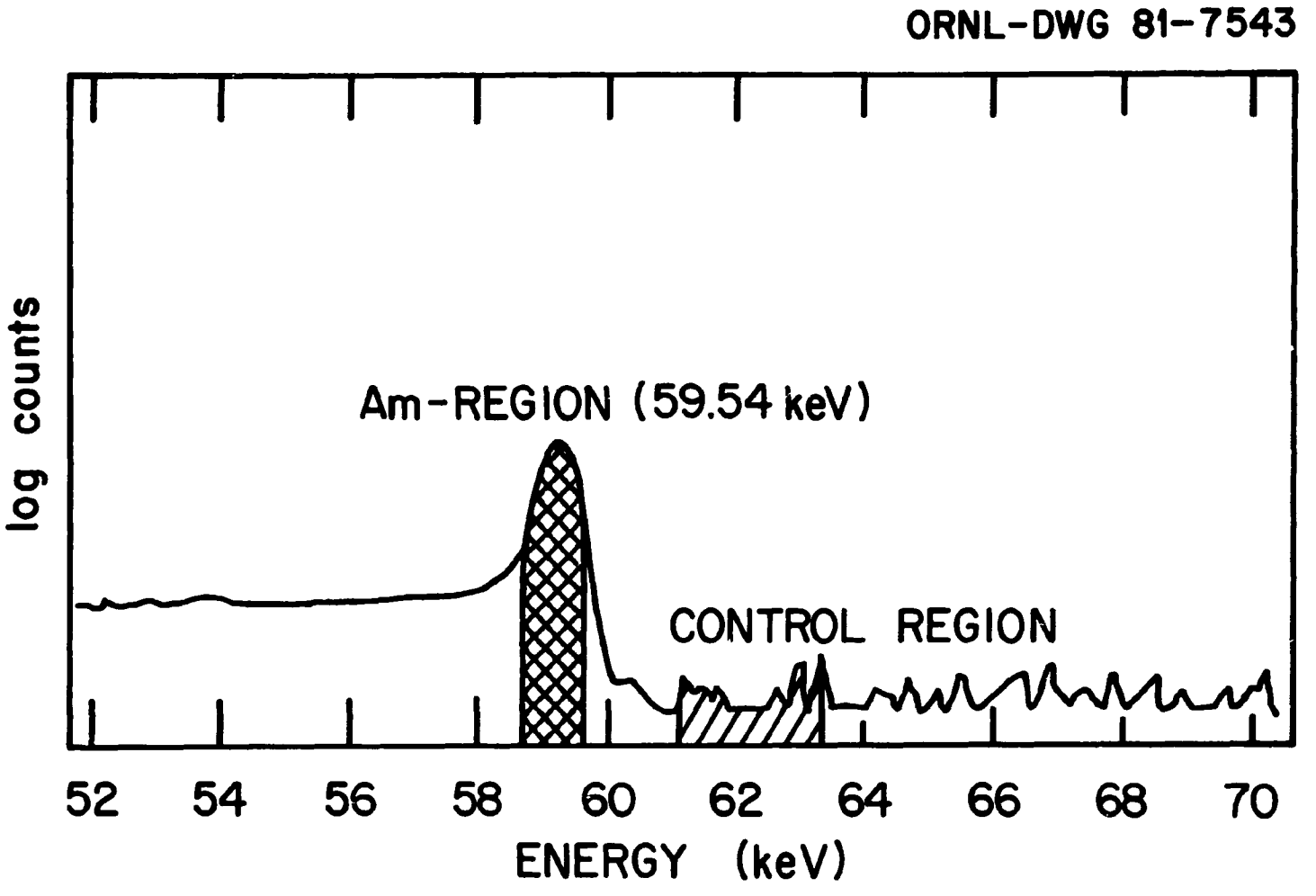


Figure 4. Regions of integration for ^{241}Am and control.

measured activity" (MSMA), defined as the smallest measurement which is interpreted as meaning there is activity in the body, and "the minimum detectable true activity" (MDTA), defined as the smallest amount of activity required to be in the body in order that a measurement can be expected to imply correctly the presence of activity with a predetermined degree of confidence. Note that the former refers to a measurement, while the latter refers to the actual activity present.

The statistical framework involved is derived from the basic theory presented by Currie.[4] The background count rate and the background-plus-activity count rate in a human are random variables having certain probability distributions. A test procedure is established for deciding whether or not true activity is greater than zero. For the problem of lung counting in humans, the appropriate test procedure is obtained by using measured activity as the test statistic and establishing its critical value such that true activity is judged to be zero or greater than zero according to whether the observed measured activity is less or not less than this critical value. The probability of making a Type I error, denoted by α, depends only on the test procedure, whereas the probability of making a Type II error, denoted by β, also depends on the amount of true activity deposited in the body, denoted by A.

Let: cal = calibration factor (Bq/count)
B = predicted background counts
N = predicted value of the activity or sample counts
N' = actual value of the activity or sample counts
A = cal•N = true activity
C_B = measured background counts
C_{B+N} = measured background plus sample counts

If x denotes a standardized normal variable with mean = 0 and variance = 1, its probability distribution function is given by:

$$f(x) = \frac{1}{\sqrt{2\pi}} \exp\ (-x^2/2) \qquad \text{(Eq. 1)}$$

The constant k_α is defined by:

$$P\left\{x \geq k_\alpha\right\} \equiv \int_{k_\alpha}^{\infty} f(x)\, dx = \alpha \qquad \text{(Eq. 2)}$$

and k_β by

$$P\left\{x < -k_\beta\right\} \equiv \int_{-\infty}^{-k_\beta} f(x)\, dx = \beta \qquad \text{(Eq. 3)}$$

Note that $k_\alpha = k_\beta$ when $\alpha = \beta$. Values for k_α and k_β are given in tables of the normal distribution function, some of which are listed in Table 1.

Table 1

TYPICAL VALUES FOR K_α AND K_β

α, β	k_α, k_β
0.005	2.576
0.010	2.326
0.025	1.960
0.050	1.645
0.100	1.282

Consider the case where the background counts in some energy region is accurately known before the count begins. The measured number of counts due to some level of activity is taken to be

$$C_{B+N} - B \qquad \text{(Eq. 4)}$$

The expected value of $C_{B+N} - B$ is N and its standard deviation is $\sqrt{B+N}$. The measured value of the true activity in that person, A, is: $cal \cdot (C_{B+N} - B)$, with standard deviation of $cal \cdot \sqrt{B+N}$.

Therefore, for a Type I error, α, the MSMA, A_I, is as follows. Assuming N=0:

$$\alpha = P_{N=0}\left\{\frac{C_{B+N} - B}{\sqrt{B}} \geq k_\alpha\right\} \quad \text{(Eq. 5)}$$

$$= P_{N=0}\left\{C_{B+N} - B \geq k_\alpha \sqrt{B}\right\} \quad \text{(Eq. 6)}$$

Thus, the minimum significant sample count = $k_\alpha \sqrt{B}$, and the MSMA is:

$$A_I = \text{cal} \times k_\alpha \sqrt{B} \quad \text{(Eq. 7)}$$

The MDTA, A_{II}=calxN, is determined by the condition that the probability of a Type II error is not greater than a chosen value of β:

$$\beta = P_{N=N'}\left\{C_{B+N} - B < k_\alpha\sqrt{B}\right\} \quad \text{(Eq. 8)}$$

From equation 3,

$$\beta = P_{N=N'}\left\{\frac{C_{B+N} - B - N}{\sqrt{B+N}} < -k_\beta\right\} \quad \text{(Eq. 9)}$$

$$= P_{N=N'}\left\{C_{B+N} - B < N - k_\beta \sqrt{B+N}\right\} \quad \text{(Eq. 10)}$$

Comparing these two equations, N' is determined by:

$$k_\alpha \sqrt{B} = N' - k_\beta \sqrt{B+N} \quad \text{(Eq. 11)}$$

Solving for N':

$$N' = \sqrt{B}\left\{k_\alpha + k_\beta \sqrt{1 + k_\alpha/\sqrt{B} + k_\beta^2/4B} + k_\beta^2/2\sqrt{B}\right\} \quad \text{(Eq. 12)}$$

$$= (k_\alpha + k_\beta)\sqrt{B} \text{ when } (k_\alpha + k_\beta)/\sqrt{B} \ll 1 \quad \text{(Eq. 13)}$$

Therefore, the true MDTA is:

$$A_{II} = \text{cal} \times (k_\alpha + k_\beta)\sqrt{B} \text{ when } (k_\alpha + k_\beta)/\sqrt{B} << 1 \qquad \text{(Eq. 14)}$$

Equations 7 and 14 are useful in the case where a subject has had a baseline count before an incident involving inhalation or ingestion of a radionuclide occurs, but it is not always possible to acquire a baseline count. The ORNL facility is also under continuing instrumentation and procedure development, therefore, counts taken 2 or 3 months apart may not yield similar data.

We must, therefore, consider the case where background counts are <u>not</u> well known. In this situation, the measured value of the sample count, N, is the statistic $C_{B+N} - C_B$, which has an expected value of N, and standard deviation of $\sqrt{2(B+N)}$. The measured value of the true sample activity, $A = \text{cal} \cdot N$, is then $\text{cal} \cdot C_{B+N} - C_B$ with the standard deviation, $\text{cal} \cdot \sqrt{2B+N}$.

The quantity $C_{B+N} - C_B - N / \sqrt{(2B+N)}$ has been assumed to be distributed as a standardized normal variable. Since the best estimate of 2B+N is $C_B + C_{B+N}$, the quantity $C_{B+N} - C_B - N / \sqrt{(C_B+C_{B+N})}$ will be used as the test statistic and taken to be distributed as a standardized normal variable.

To determine the MSMA, A_I, for a given α, assume N=0. Then by Equation 5,

$$\alpha = P_{N=0}\left\{ \frac{C_{B+N} - C_B}{\sqrt{(C_{B+N} + C_B)}} \geq k_\alpha \right\} \qquad \text{(Eq. 15)}$$

$$= P_{N=0}\left\{ C_{B+N} - C_B \geq 1/2 \left[k_\alpha^2 + \sqrt{(k_\alpha^4 + 8k_\alpha^2 C_B)} \right] \right\} \qquad \text{(Eq. 16)}$$

Thus, the minimum significant count difference, $\Delta = C_{B+N} - C_B$ is

$$\Delta = 1/2 \left\{ k_\alpha^2 + \sqrt{(k_\alpha^4 + 8k_\alpha^2 C_B)} \right\} \qquad \text{(Eq. 17)}$$

and

$$A_I = \text{cal} \cdot \Delta \cong \text{cal}\ k_\alpha \sqrt{2C_B} \quad \text{when } k_\alpha / \sqrt{C_B} \ll 1 \qquad \text{(Eq. 18)}$$

Therefore, MDTA is

$$A_{II} = \text{cal} \cdot N' = \text{cal}\ \sqrt{2C_B} \left\{ k_\alpha \left[\sqrt{(1+k_\alpha^2/8C_B)} + k_\alpha / \sqrt{8C_B} \right] + k_\alpha \left[1 + k_\alpha^2/4C_B + \frac{k_\alpha}{\sqrt{2C_B}} \sqrt{(1 + k_\alpha^2/8C_B)} \right]^{\frac{1}{2}} \right\} \qquad \text{(Eq. 19)}$$

$$\cong \text{cal} \cdot (k_\alpha + k_\beta) \sqrt{2C_B} \quad \text{when } k_\alpha / \sqrt{2C_B} \ll 1 \qquad \text{(Eq. 20)}$$

Comparison of these two cases shows that the limits of detection, A_I and A_{II}, are increased by an approximate factor of $\sqrt{2}$ when the background is not accurately known beforehand. This indicates that a degree of improvement can be achieved by acquiring valid baseline counts whenever possible.

Table 2 presents the current MSMA and MDTA for various radionuclides, using the HPGe array for data acquisition. MSMA and MDTA are calculated for the case of a 70 kg, adult male with 2.58 cm chest wall thickness. The assumption is made that the background is not well known. Point sources were used to derive calibration factors for all of the radionuclides listed except for ^{239}Pu, in which case a simulated distributed source was used. Note that human variability, distribution of the source in the chest area, etc., will interject a further source of error into these numbers; therefore, they are only used as reference values.

An Attempt at Universal Calibration

The nuclide identification libraries in the ORNL computer-based whole body counting system consist of 18 source entries to date. These entries are radionuclides commonly found in the various health physics areas around the laboratory. Should an individual show some unsuspected level of internally deposited radioactivity not-listed in the libraries after

Table 2

CURRENT MSMA AND MDTA VALUES FOR SELECTED ISOTOPES
USING THE HPGe ARRAY

Nuclide	MPOB (Bq)	Critical Organ	Ref. Energy* (keV)	MSMA (Bq)	MSMA as % MPOB	MDTA (Bq)
^{239}Pu	592	Lung	17-26	398	67.3	797
^{241}Am	592	Lung	60	24.8	4.2	40.7
^{125}I	2.00×10^{5}	Thyroid	27, 31	0.74	3.7×10^{-4}	1.48
^{137}Cs	7.59×10^{4}	Lung	32	370	0.5	609
^{153}Gd	3.15×10^{5}	Lung	41, 97	19.6	0.01	32.2
^{152}Eu	9.25×10^{4}	Lung	40, 45, 121	15.9	0.02	26.3
^{233}U	629	Lung	74, 77	320	50.9	52.7
^{57}Co	5.92×10^{4}	Lung	122	21.1	0.04	34.8

*MSMA and MDTA can be lower for some of these radionuclides when a higher reference energy is chosen and another detector is used [e.g., MSMA for ^{137}Cs is 2.96 Bq when calculations are based on the 662 keV line as detected by a large NaI(Tl) crystal].

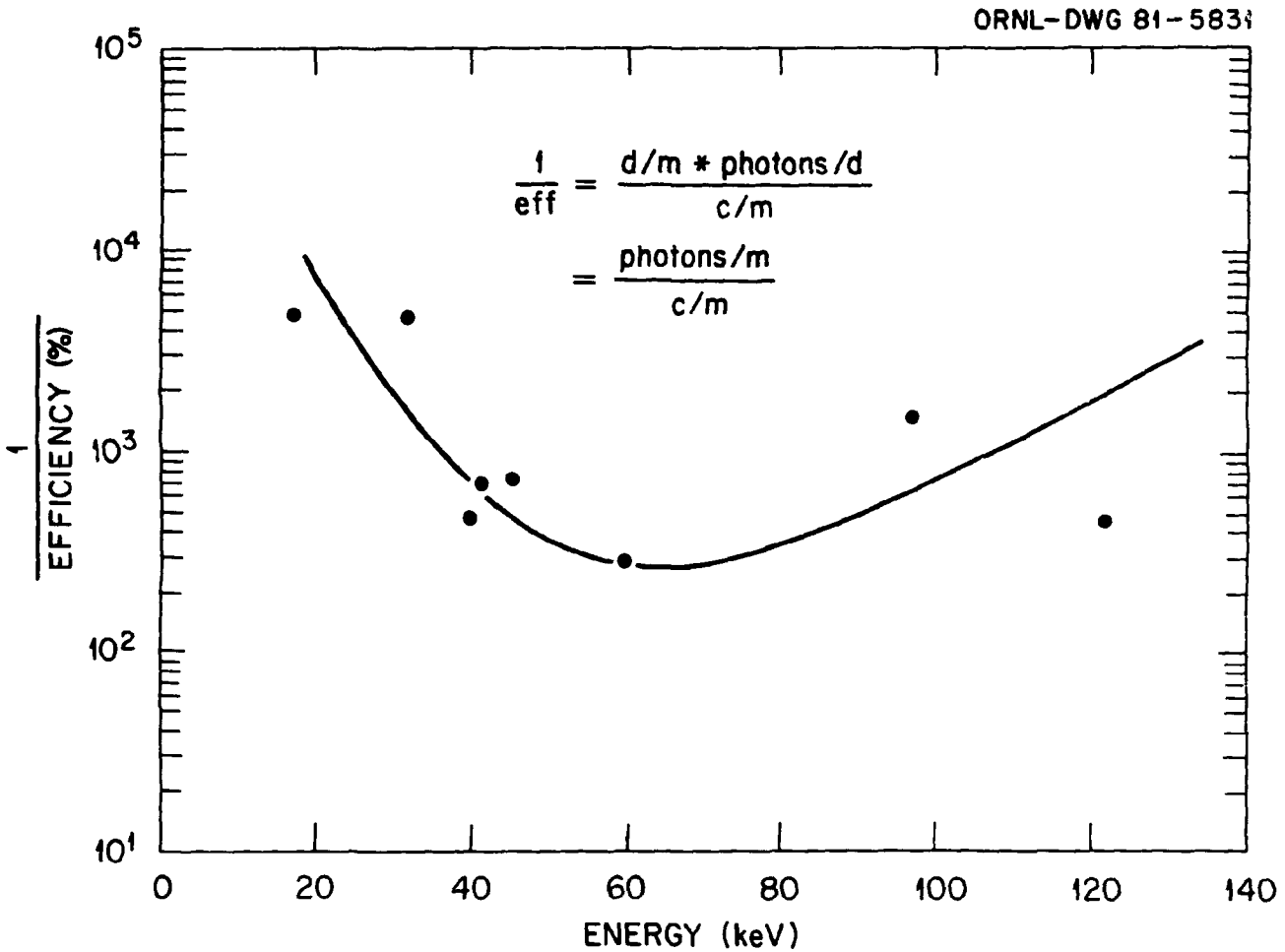
his/her count, the procedure is to obtain the appropriate standardized radiation sources, acquire calibration information (which then becomes a new library addition) and finally quantify the activity noted in that subject's spectrum.

This procedure works out quite well, except in an emergency, when it can take as long as two days after a count to obtain the appropriate calibration information and arrive at a final estimate of organ burden. Since the advantage of a whole body counter is its ability to provide rapid assessment of body burden, we use a simple technique for estimating the quantity of any radionuclide that emits photons or X rays in the energy range of zero to three MeV. While awaiting acquisition of detailed calibration information, a preliminary indication of whether an action level may have been reached or exceeded can be obtained.

This "universal" calibration does not account for absorption and attenuation effects for very low-energy photons and X rays, therefore, estimations of organ burden can be in error by as high as a factor of 10 in some cases, depending on the radionuclide. This can be a critical situation when dealing with transuranics but is considered to be an acceptable estimation for most beta-gamma emitters, whose maximum permissible organ burdens are high compared to those for alpha-emitters. (Estimation of lung burdens of the actinides is never done in this manner, as appropriate calibration data has been obtained.)

This technique is based on derivation of a relative efficiency curve, through the use of a series of radiation sources whose photon energies cover the energy range set for each detector. This is considered to be a "relative" efficiency curve since the data was acquired with standardized sources placed in a phantom, and no attempt was made to correct for absorption. Figure 5 shows one such curve for the HPGe detector array, covering an energy range of zero to 130 keV. Efficiency was calculated by:

$$\text{eff} = \frac{\left\{\frac{\text{counts}}{\text{minute}}\right\}}{\left\{\frac{\text{disintegrations}}{\text{minute}}\right\} * \left\{\frac{\text{photons}}{\text{disintegration}}\right\}} = \frac{\left\{\frac{\text{counts}}{\text{minute}}\right\}}{\left\{\frac{\text{photons}}{\text{minute}}\right\}}$$



Inverse Relative Efficiency Curve-Multiplexed HPGe Array

Figure 5. Energy vs. relative efficiency for HPGe detector array. (Inverse efficiency is plotted for calculational ease.)

Figure 5 is a plot of the photon energy vs. the inverse of the efficiency for the purpose of simplifying calculations. Note that efficiency begins to drop above 70 keV, due to the small active thickness (~ 7 mm) of the HPGe detectors. Efficiency is also lower below 40 keV due to significant attenuation of these low-energy photons by the chest of the phantom. Counts per minute are determined by integrating the area under each energy peak at full width half maximum, and dividing by the count time.

To estimate the activity of an internally deposited radionuclide that is not listed in the nuclide identification libraries, one calculates the counts per minute in the highest energy peak on the spectrum corresponding to the radionuclide in question (to reduce the addition of Compton events) and reads the inverse efficiency off the graph, corresponding to that energy. After determining the photon or X ray yield of that energy line (in photons/disintegration), activity in becquerels is calculated by:

$$\text{Activity} = \frac{1}{\text{eff}} \cdot \frac{\text{counts}}{\text{minute}} \cdot \frac{1}{\text{yield}}$$

Although there is some error associated with this number due to the addition of ^{40}K in a human as opposed to a phantom, and due to differential attenuation variability in humans, this information is useful for assessing "relative" dose commitment status as a basis for making appropriate decisions on rapid acquisition of refined calibration data and repeat subject counts.

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