Designation: D7784 – 12

Standard Practice for the Rapid Assessment of Gamma-ray Emitting Radionuclides in Environmental Media by Gamma Spectrometry[1]

This standard is issued under the fixed designation D7784; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This practice covers the quantification of radionuclides in environmental media (e.g., water, soil, vegetation, food) by means of simple preparation and counting with a high-resolution gamma ray detector. Because the practice is designed for rapid analysis, extensive efforts to ensure homogeneity or ideal sample counting conditions are not taken.

1.2 The values stated in SI units are to be regarded as standard. The values given in parentheses are provided for information purposes only.

1.3 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*[2]

C998 Practice for Sampling Surface Soil for Radionuclides
D1129 Terminology Relating to Water
D3370 Practices for Sampling Water from Closed Conduits
D3648 Practices for the Measurement of Radioactivity
D3649 Practice for High-Resolution Gamma-Ray Spectrometry of Water
D7282 Practice for Set-up, Calibration, and Quality Control of Instruments Used for Radioactivity Measurements

2.2 *Other Documents:*

PCNUDAT data files National Nuclear Data Center, Brookhaven National Lab, Upton, NY, USA

3. Terminology

3.1 *Definitions*—for definitions of terms used in this practice, refer to Terminology D1129.

4. Summary of Practice

4.1 Following sample collection, sample material is placed in a suitable container for analysis by a gamma spectrometry system. A suitable container is defined as a container which will both hold the sample in a fixed geometry and for which the gamma spectrometry system has been calibrated. For solid samples, the samples may be ground, sieved, or otherwise prepared for the purpose of volume reduction, homogenization, or conformance to the calibration standard, as desired.

5. Significance and Use

5.1 This practice was developed for the rapid determination of gamma-emitting radionuclides in environmental media. The results of the test may be used to determine if the activity of these radionuclides in the sample exceeds the action level for the relevant incident or emergency response. The detection limits will be dependent on sample size, counting configuration, and the detector system in use.

5.2 In most cases, a sample container which is large in diameter and short in height relative to the detector will provide the best gamma-ray detection efficiency. For samples of water or other low-Z materials (e.g., vegetation), the re-entrant or Marinelli-style beaker may yield the best gamma-ray detection efficiency.

5.3 The density of the sample material and physical parameters of the sample container (e.g., diameter, height, material) may have significant consequences for the accuracy of the sample analysis as compared to the calibration. For this reason, the ideal calibration material and container (often referred to as 'geometry') will be exactly the same as the samples to be analyzed. Differences in sample container or sample matrix may introduce significant errors in detector response, especially at low gamma-ray energies. Every effort should be made to account for these differences if the exact calibration geometry is not available.

5.4 This method establishes an empirical gamma-ray spectrometer calibration using standards traceable to a national standardizing body in a specific geometry selected to ensure that the container, density, and composition of the standard matches that of the samples as closely as possible. However, in some cases it may be beneficial to modify such initial calibrations using mathematical modeling or extrapolations to an

---

[1] This practice is under the jurisdiction of ASTM Committee D19 on Water and is the direct responsibility of Subcommittee D19.04 on Methods of Radiochemical Analysis.

Current edition approved Nov. 1, 2012. Published November 2012. DOI: 10.1520/D7784–12

[2] For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* volume information, refer to the standard's Document Summary page on the ASTM website.

alternate geometry. Use of such a model may be acceptable, depending on the measurement quality objectives of the analysis process, and provided that appropriate compensation to uncertainty estimates are included. The use of such calibration models is best supported by the successful analysis of a method validation reference material (MVRM).

5.5 This practice addresses the analysis of numerous gamma-emitting radionuclides in environmental media. This practice should be applicable to non-environmental media (for example, urine, debris, or rubble) that have similar physical properties. The key determination of "similar physical properties" is the ability to demonstrate that the gamma spectrometry system response to the sample configuration is suitably similar to that for which the system is calibrated.

5.6 For the analysis of radionuclides with low gamma-ray emission energies (<100 keV), self-absorption of the gamma-rays in the sample matrix can have a significant adverse effect on detection and quantification. The user should verify that instrument calibrations appropriately account for any self-absorption that may result from the sample matrix.

5.7 Commonly available energy and efficiency calibration standards cover the energy range of approximately 60 keV to 1836 keV. Results obtained using gamma-ray peaks outside the efficiency calibrated energy range will have greater uncertainty not accounted for in the uncertainty calculations of this practice. Great care should be taken to review the efficiency calibration values and the shape of the efficiency curve outside this range. For greater accuracy in the analysis of radionuclides whose gamma-ray energies are outside this range, a calibration standard which includes radionuclide(s) whose gamma-ray energies span the energy range of radionuclides of interest is advised.

6. Interferences

6.1 A list of some gamma-ray emitting radionuclides with relevant data is provided, for information only, in Table 1. This list includes radionuclides which may be of interest to agencies responding to a large scale radiological event. Through inspection of the list, it becomes apparent that there are numerous opportunities for interferences based on the gamma energy emissions. For this reason, it is important that the determination of the presence of a given radionuclide be supported by all available evidence (e.g., additional gamma-ray emissions).

6.2 The data provided in Table 1, Table 2, and Table 3 are not mandatory and are provided for information only. The composition of the nuclide library used by the laboratory should be matched to the analytical need and the data should be validated using a current reference source (e.g., Laboratoire National Henri Becquerel, http://www.nucleide.org/DDEP_WG/DDEPdata.htm, or NuDAT data files, National Nuclear Data Center, Brookhaven National Lab, Upton, NY, USA)

6.3 Several of the radionuclides listed in Table 1, Table 2, and Table 3 have x-ray emissions which may interfere with gamma-ray emissions, particularly below approximately 40 keV. It is the responsibility of the laboratory to ensure that x-ray and gamma-ray interferences are accounted for in the analytical process.

TABLE 1 Example of most likely radionuclides for emergency response

Nuclide	Gamma Energy (keV)	Gamma Fraction	Half-Life (d)
Ac-227	100	3.17E-04	7.96E+03
Ac-227	83.96	2.21E-04	7.96E+03
Ag-110m	657.75	9.47E-01	2.50E+02
Ag-110m	884.67	7.29E-01	2.50E+02
Am-241	59.54	3.63E-01	1.58E+05
Am-242m	49.3	1.90E-03	5.55E+04
Am-243	74.67	6.60E-01	2.70E+06
Au-198	411.80	9.55E-01	2.70E+00
Au-198	70.82	1.38E-02	2.70E+00
Ba-133	30.97	6.29E-01	3.91E+03
Ba-133	355.86	6.23E-01	3.91E+03
Ba-137m	661.62	9.00E-01	1.77E-03
Ba-137m	32.19	3.82E-02	1.77E-03
Ba-140	537.38	1.99E-01	1.28E+01
Ba-140	29.96	1.43E-01	1.28E+01
Bi-207	569.67	9.80E-01	1.39E+04
Bi-207	1063.62	7.70E-01	1.39E+04
Cd-109	24.95	1.43E-01	4.53E+02
Cd-113m	263.7	6.00E-05	5.33E+03
Cd-113m	23.17	6.00E-05	5.33E+03
Ce-141	145.45	4.80E-01	3.24E+01
Ce-141	36.03	8.88E-02	3.24E+01
Ce-143	293.3	4.34E-01	1.40E+00
Ce-143	36.03	3.23E-01	1.40E+00
Ce-144	133.53	1.08E-01	2.84E+02
Ce-144	36.03	4.80E-02	2.84E+02
Cf-252	43.4	1.30E-04	8.99E+02
Cm-242	44.03	3.25E-04	1.63E+02
Cm-243	103.75	2.08E-01	1.04E+04
Cm-244	42.82	2.55E-04	6.61E+03
Cm-245	103.76	2.30E-01	3.11E+06
Co-58	810.75	9.95E-01	7.08E+01
Co-58	511	3.00E-01	7.08E+01
Co-60	1332.51	1.00E+00	1.93E+03
Co-60	1173.23	9.99E-01	1.93E+03
Co-60	2158.7	8.00E-06	1.93E+03
Cr-51	320.07	9.83E-02	2.77E+01
Cs-134	604.66	9.76E-01	7.53E+02
Cs-134	795.76	8.54E-01	7.53E+02
Cs-136	818.5	1.00E+00	1.30E+01
Cs-136	1048.07	8.00E-01	1.30E+01
Cs-137	661.62	8.46E-01	1.10E+04
Cs-137	32.19	3.70E-02	1.10E+04
Eu-152	40.12	3.00E-01	4.64E+03
Eu-152	121.78	2.92E-01	4.64E+03
Eu-154	123.1	4.05E-01	3.11E+03
Eu-154	1274.8	3.55E-01	3.11E+03
Eu-155	86.45	3.27E-01	1.81E+03
Eu-155	105.31	2.18E-01	1.81E+03
Fe-59	1099.22	5.65E-01	4.51E+01
Fe-59	1291.56	4.32E-01	4.51E+01
Gd-153	41.54	6.00E-01	2.42E+02
Gd-153	40.9	3.20E-01	2.42E+02
Hf-181	482.16	8.60E-01	4.25E+01
Hf-181	133.05	4.30E-01	4.25E+01
Hg-203	279.17	8.15E-01	4.66E+01
Hg-203	72.87	6.40E-02	4.66E+01
Ho-166m	184.41	7.39E-01	4.38E+05
Ho-166m	810.31	5.97E-01	4.38E+05
I-125	27.47	7.30E-01	6.01E+01
I-125	27.2	3.92E-01	6.01E+01
I-129	29.78	3.60E-01	5.73E+09
I-129	29.46	1.90E-01	5.73E+09
I-131	364.48	8.12E-01	8.04E+00
I-131	636.97	7.27E-02	8.04E+00
I-131	284.29	6.06E-02	8.04E+00
I-131	80.18	2.62E-02	8.04E+00
I-131	29.78	2.59E-02	8.04E+00
I-132	667.69	9.87E-01	9.92E-02
I-132	772.61	7.62E-01	9.92E-02
In-114m	24.21	2.00E-01	4.95E+01
In-114m	189.9	1.77E-01	4.95E+01
Ir-192	316.49	8.70E-01	7.40E+01

TABLE 1 *Continued*

Nuclide	Gamma Energy (keV)	Gamma Fraction	Half-Life (d)
Ir-192	468.06	5.18E-01	7.40E+01
K-40	1460.75	1.07E-01	4.68E+11
La-140	1596.2	9.55E-01	1.68E+00
La-140	487.03	4.30E-01	1.68E+00
Mn-54	834.81	1.00E+00	3.12E+02
Mo-99	140.51	9.09E-01	2.76E+00
Mo-99	739.47	1.30E-01	2.76E+00
Na-22	511	1.80E+00	9.50E+02
Na-22	1274.54	9.99E-01	9.50E+02
Nb-94	871.1	1.00E+00	7.42E+06
Nb-94	702.5	1.00E+00	7.42E+06
Nb-95	765.82	9.90E-01	3.52E+01
Nd-147	91.1	2.83E-01	1.11E+01
Nd-147	38.72	2.30E-01	1.11E+01
Nd-147	531	1.35E-01	1.11E+01
Np-237	86.49	1.31E-01	7.82E+08
Np-237	29.38	9.80E-02	7.82E+08
Np-237	95.87	2.96E-02	7.82E+08
Np-239	103.7	2.40E-01	2.36E+00
Np-239	106.13	2.27E-01	2.36E+00
Pa–234m	1001.03	5.90E-03	8.13E-04
Pa-234m	766.6	2.07E-03	8.13E-04
Pb-210	46.52	4.00E-02	7.45E+03
Pm-145	37.36	3.86E-01	6.47E+03
Pm-145	36.85	2.11E-01	6.47E+03
Pm-147	121.2	4.00E-05	9.58E+02
Pm-149	285.9	3.10E-02	2.21E+00
Pm-149	859.4	1.00E-03	2.21E+00
Pm-151	340.08	2.24E-01	1.18E+00
Pm-151	40.12	1.66E-01	1.18E+00
Po-210	803	1.10E-05	1.38E+02
Pr-144	696.49	1.49E-02	1.20E-02
Pr-144	2185.61	7.70E-03	1.20E-02
Pu-236	47.6	6.90E-04	1.04E+03
Pu-236	109	1.20E-04	1.04E+03
Pu-238	43.45	3.80E-04	3.21E+04
Pu-238	99.86	7.24E-05	3.21E+04
Pu-239	51.62	2.08E-04	8.81E+06
Pu-239	129.28	6.20E-05	8.81E+06
Pu-240	45.24	4.50E-04	2.39E+06
Pu-240	104.23	7.00E-05	2.39E+06
Pu-241	98.44	2.20E-05	5.54E+03
Pu-241	94.66	1.20E-05	5.54E+03
Pu-241	111	8.40E-06	5.54E+03
Pu-242	44.7	3.60E-02	1.41E+08
Pu-242	103.5	7.80E-03	1.41E+08
Ra-226	185.99	3.28E-02	5.84E+05
Ra-226	83.78	3.10E-03	5.84E+05
Rb-86	1076.63	8.76E-02	1.86E+01
Rh-106	511.8	2.06E-01	3.46E-04
Rh-106	621.8	9.81E-02	3.46E-04
Ru-103	497.08	8.64E-01	3.94E+01
Ru-103	610.33	5.30E-02	3.94E+01
Sb-124	602.71	9.81E-01	6.02E+01
Sb-124	1691.04	5.00E-01	6.02E+01
Sb-126	695.1	9.97E-01	1.25E+01
Sb-126	666.2	9.97E-01	1.25E+01
Sb-127	685.5	3.57E-01	3.85E+00
Sb-127	473	2.50E-01	3.85E+00
Sc-46	1120.52	1.00E+00	8.39E+01
Sc-46	889.26	1.00E+00	8.39E+01
Se-75	264.65	5.86E-01	1.20E+02
Se-75	136	5.60E-01	1.20E+02
Sn-113	391.71	6.42E-01	1.15E+02
Sn-113	24.21	3.90E-01	1.15E+02
Sn-123	1089	6.00E-03	1.29E+02
Sn-123	1032	4.00E-04	1.29E+02
Sn-125	1066.6	9.00E-02	9.62E+00
Sn-125	915.5	4.25E-02	9.62E+00
Sn-126	87.57	3.75E-01	3.65E+07
Sn-126	26.11	1.89E-01	3.65E+07
Sr-89	909.2	9.50E-04	5.05E+01
Ta-182	67.75	4.13E-01	1.15E+02
Ta-182	1121.28	3.50E-01	1.15E+02
Tb-160	876.37	3.00E-01	7.21E+01
Tb-160	298.57	2.74E-01	7.21E+01
Tc-99	89.6	6.50E-06	7.82E+07
Te-127	417.9	9.93E-03	3.90E-01
Te-127	360.3	1.35E-03	3.90E-01
Te-129	27.77	1.64E-01	4.83E-02
Te-129	459.5	7.14E-02	4.83E-02
Te-129m	27.47	1.53E-01	3.36E+01
Te-129m	27.2	7.80E-02	3.36E+01
Te-131m	773.67	3.81E-01	1.25E+00
Te-131m	852.21	2.06E-01	1.25E+00
Te-132	228.16	8.85E-01	3.25E+00
Te-132	28.5	5.40E-01	3.25E+00
Th-227	236	1.12E-01	1.85E+01
Th-227	50.2	8.50E-02	1.85E+01
Th-227	256.25	6.80E-02	1.85E+01
Ti-44	78.4	9.47E-01	1.73E+04
Ti-44	67.8	8.77E-01	1.73E+04
Tl-204	70.82	7.40E-03	1.38E+03
Tl-204	68.89	4.00E-03	1.38E+03
Tm-170	84.26	1.00E-01	1.29E+02
Tm-170	52.39	6.80E-02	1.29E+02
Tm-170	51.35	3.60E-02	1.29E+02
U-235	185.72	5.40E-01	2.57E+11
U-235	143.76	1.05E-01	2.57E+11
U-235	163.35	4.70E-02	2.57E+11
U-238	48	7.50E-04	1.72E+12
V-48	983.5	1.00E+00	1.61E+01
V-48	1311.6	9.80E-01	1.61E+01
V-48	511	9.80E-01	1.61E+01
W-187	685.74	2.92E-01	9.96E-01
W-187	479.57	2.34E-01	9.96E-01
Y-90	1760.7	1.15E-04	2.67E+00
Y-91	1204.9	3.00E-03	5.85E+01
Yb-169	50.74	7.81E-01	3.07E+01
Yb-169	63.12	4.50E-01	3.07E+01
Yb-169	49.77	4.22E-01	3.07E+01
Zn-65	1115.52	5.08E-01	2.44E+02
Zn-65	511	2.83E-02	2.44E+02
Zr-95	756.72	5.48E-01	6.44E+01
Zr-95	724.18	4.42E-01	6.44E+01

TABLE 2 Example of most likely radionuclides for emergency response subsequent to an incident involving a radiological dispersal device

Alpha Emitters		Beta/Gamma Emitters	
Am-241	Ra-226	Ac-227	P-32
Cm-242	Th-228	Bi-210	Pd-103
Cm-243	Th-230	Bi-212	Pb-210
Cm-244	Th-232	Bi-214	Pb-212
Np-237	U-234	Co-57	Pb-214
Po-210	U-235	Co-60	Pu-241
Pu-238	U-238	I-125	Ra-228
Pu-239	U-Nat	I-129	Se-75
Pu-240		Ir-192	

7. Apparatus

7.1 *Analytical Balance,* readable to 0.1 g.

7.2 *Sample Container*—a container suitable for holding the sample material to be analyzed. The container may be of any suitable configuration, but should be reproducible in its dimensions and capacity. This should be the same container design for which the counting system is calibrated. An ideal container is smaller in diameter than the detector to be used for analysis (7.3) and should be as short in the vertical dimension as is practical. A re-entrant beaker (e.g., Marinelli-style) may be

TABLE 3 Example of most likely radionuclides for emergency response subsequent to an incident involving an improvised nuclear device

Alpha Emitters	Beta/Gamma Emitters		
Am-241	Ba-140/ La-140	Nd-147/Pr-147	Sb-125
U-234	Ce-141	Eu-155	Sr-89
U-235	Ce-143/Pr-143	H-3	Sr-90/Y-90
U-238	Ce-144/Pr-144	I-131/Xe-131	Tc-99
Pu-238	Cs-134	I-133	Te-132/I-32
Pu-239	Cs-137	Np-239	Zr-95/Nb-95
Pu-240	Eu-154	Ru-103/Rh-103	Zr-97/Nb-97
	Mo-99/ Tc-99m	Ru-106/Rh-106	
	Activation Products		
	Ag-110m	Cr-51	Mn-54
	Co-60	Fe-59	Na-24

used to improve the counting efficiency for low Z-value materials. The container should be durable and sealable to prevent content loss during handling.

7.3 *Gamma-Ray Spectrometry System*—high resolution high purity germanium gamma spectrometer with an energy range of approximately 20 keV to 2200 keV (see Practice D3649). Note: The useable energy range of the gamma spectrometer will be determined by the efficiency calibration. Further guidance on the use of high purity germanium systems may be found in Practice D3649.

8. Reagents and Materials

8.1 *Radioactive Purity*—Radioactive purity should be such that the measured radioactivity of blank samples does not compromise the applicable measurement quality objectives.

8.2 *Calibration standard*—Known amounts of specific radionuclides whose gamma-ray emission energies cover a wide energy range should be used for calibration, provided that they have gamma ray energies covering the energy range of the radionuclides of interest. The known activity of the radionuclides should be traceable to a national standardizing body such as NIST in the USA.

9. Calibration of High-resolution Gamma-ray Spectroscopy System

9.1 Accumulate an energy spectrum using a calibration standard (8.2) traceable to a national standards body, in the geometrical position representing that of the samples to be analyzed. Accumulate sufficient net counts (total counts minus the Compton) in each full-energy gamma-ray peak of interest to obtain a one-sigma relative counting uncertainty of ≤1%.

9.2 Using the gamma emission data from the calibration standard and the peak location data from the calibration spectrum establish the energy per channel relationship (energy calibration) as:

$$En = Offset + (Ch \times Slope) \tag{1}$$

where:

En = peak energy (keV),
$Offset$ = energy offset for the energy calibration equation (keV),
Ch = peak location channel number, and
$Slope$ = energy calibration equation slope (keV/channel).

NOTE 1—Most modern spectroscopy software packages perform this calculation, and may include higher-order polynomial terms to account for minor non-linearity in the energy calibration.

9.3 Using the gamma emission data from the calibration standard and the peak resolution data from the calibration spectrum establish the resolution versus energy relationship (resolution calibration) as:

$$FWHM = Offset + (En \times Slope) \tag{2}$$

where:

$FWHM$ = Full Width of the peak at one-Half the Maximum counts in the centroid channel (keV),
$Offset$ = width offset for the resolution calibration equation (keV),
En = peak energy (keV), and
$Slope$ = resolution calibration equation slope (keV/keV).

NOTE 2—Most modern spectroscopy software packages perform this calculation, and may include higher-order polynomial terms to account for non-linearity in the resolution calibration.

9.4 Calculate the full-energy peak efficiency, εf, using the equation defined in the laboratory quality manual or with example equation (3):

$$\epsilon_f = \frac{R_n}{A_g \times DF} \tag{3}$$

where:

ϵ_f = full-energy peak efficiency (counts per gamma ray emitted),
R_n = net gamma-ray count in the full-energy peak of interest counts per second (s^{-1}),
A_g = gamma-ray emission rate (gamma rays per second),
DF = decay factor for the calibrating radionuclide,

$$DF = e^{-\lambda \times (t_1 - t_0)} \tag{4}$$

where:

λ = 0.693/ $t_{1/2}$,
$t_{1/2}$ = half-life of calibrating radionuclide (half-life units must match those used for the difference $t_1 - t_0$),
t_0 = reference date and time of the calibrating radionuclide radioactivity value, and
t_1 = midpoint of sample count (date and time).

NOTE 3—Most modern spectrometry systems are computerized and the determination of the gamma-ray efficiencies is performed automatically at the end of an appropriate counting interval. Refer to the manufacturer instructions for specific requirements.

9.5 Plot the values for the full-energy peak efficiency (as determined in 9.4) versus gamma-ray energy. Compare the efficiency curve to the typical efficiency curve for the detector type. The curve should be smooth, continuous and have a shape similar to that expected for the detector being used. The plot will allow the determination of efficiencies at energies throughout the range of the calibration energies and to show that the algorithms used by the computerized system are providing valid efficiency calibrations. Select the fit that has the best 95% confidence limit around the fitted curve and/or has all data points within ± 8% of the value of the fitted curve. This is accomplished by calculating the bias between the actual efficiency and the efficiency calculated with the fitted curve.

9.6 Save or store the values of energy versus efficiency for future reference, to be used in the calculation of radioactivity for each nuclide of interest.

10. Sampling

10.1 Collect a sample in accordance with procedures applicable to the media; for example, see Practice D3370 for water sampling or C998 for soil sampling.

11. Procedure

11.1 Measure or weigh a suitable aliquant of sample material into an appropriate container for which an empty (tare) weight has been established. Note: this step may be performed in the field, in which case the mass information should be transferred with the sample.

11.2 Position the sample for counting in a reproducible geometrical arrangement for which the gamma-ray spectrometry system has been or will be calibrated for counting efficiency.

11.3 Using the high resolution gamma-ray spectrometry system, determine the net counting rate for the gamma-ray energy lines of each peak present in the spectrum. Count the sample for a sufficient amount of time to reach the specified measurement quality objectives. A list of recommended gamma energy lines and gamma emission fractions is provided in Table 1. A list of radionuclides which may be present following some radiological incidents is provided in Table 2 and Table 3.

12. Calculation

NOTE 4—NOTE—Most modern spectrometry systems are computerized and the determination of the activity concentration may be performed automatically with appropriate inputs from the user. Refer to the manufacturer instructions for specific software operating procedures and setup of the software to match the calculation methodology described below.

NOTE 5—The following calculations assume that there are no interfering photopeaks in the background or sample spectrum, and none of the photopeaks is found in a multiplet.

12.1 For each photopeak the net count rate, R_{net}, and its standard uncertainty, $u(R_{net})$, are given by the following equations:

$$R_{net} = R_p - \left(R_b \times \frac{n_p}{n_b}\right) = \frac{C_p - \left(C_b \times \frac{n_p}{n_b}\right)}{t_s} \tag{5}$$

$$u(R_{net}) = \frac{\sqrt{C_p + \left(C_b \times \frac{n_p^2}{n_b^2}\right)}}{t_s} \tag{6}$$

where:

R_p = photopeak count rate (s^{-1}); the total counts in the peak area divided by the counting time,
R_b = Compton baseline count rate (s^{-1}); the counts in the baseline of the peak, taken as the area under a line drawn from the peak start channel to the peak end channel divided by the counting time,
C_p = total photopeak counts,
C_b = total Compton baseline counts,
t_s = sample count time (s),
n_p = number of channels in the photopeak, and
n_b = number of channels used in the baseline measurement.

12.2 When only one photopeak is used in the measurement, the activity concentration, *AC*, is given by:

$$AC(Bq/unit) = \frac{R_{net}}{\epsilon \times I \times V_a \times e^{-\lambda t}} \tag{7}$$

where:

ϵ = absolute detection efficiency of the gamma-ray spectrometer, counts per disintegration (Eq. 4), at the energy of interest
λ = decay constant for the isotope of interest , d^{-1} (Reference 6)
R_{net} = net count rate of the photopeak of interest (Eq. 5)
I = gamma fraction, gammas per disintegration, for that gamma-ray energy
t = elapsed time from sample collection to midpoint of sample counting time, days,
V_a = volume or mass of sample aliquant, L, kg, sample, or other unit as requested.

NOTE 6—If a photopeak representing an analyte, or exactly matching the energy of an analyte's gamma-ray, is known to exist in the background spectrum of the spectrometer, the known background photopeak count rate should be added to the Compton baseline count rate above (R_b) prior to the determination of the net count rate R_{net}, and the uncertainty of R_{net} should be increased accordingly.

12.3 Using a single peak, the standard counting uncertainty of the sample activity concentration, $u_{cC}(AC)$, is given by:

$$u_{cC}(AC) = \frac{u(R_{net})}{\epsilon \times I \times V_a \times e^{-\lambda t}} \tag{8}$$

In this case the combined standard uncertainty $u_c(AC)$ is given by:

$$u_c(AC) = \sqrt{u_{cC}^2(AC) + AC^2 \times \left(u_r^2(\epsilon) + u_r^2(I) + u_r^2(V_a) + u_r(...)\right)} \tag{9}$$

where:

$u_r(\epsilon)$ = relative standard uncertainty of the detection efficiency, ϵ,
$u_r(V_a)$ = relative standard uncertainty of the sample volume or mass, V_a,
$u_r(I)$ = relative standard uncertainty of the gamma fraction, I, and
$u_r^2(...)$ = any additional relative uncertainty that has been determined or estimated.

Other symbols are as defined in 12.1 and 12.2.

12.4 To calculate activity based on multiple peaks from the same nuclide, use the following equation:

$$AC_{avg}(Bq/unit) = \frac{\sum_{i=1}^{n}(AC_i \times I_i)}{\sum_{i=1}^{n} I_i} \tag{10}$$

where:

AC_{avg} = average activity concentration determined from *n* peaks,

n = number of peaks used for the calculation,
i = peak number (i = 1,2, …, n),
AC_i = activity concentration determined from the i^{th} peak (see Equation 6), and
I_i = gamma fraction for the i^{th} peak (gamma-rays per disintegration).

12.5 Using the average activity from equation (10), the standard counting uncertainty of the average activity concentration, $u_{cC}(AC_{avg})$, is given by:

$$u_{cC}(AC_{avg}) = \frac{\sqrt{\sum_{i=1}^{n}\left(u_{cC}^2\,(AC_i)\,\times\,I_i^2\right)}}{\sum_{i=1}^{n} I_i} \tag{11}$$

where:

$u_c(AC_i)$ = standard counting uncertainty of AC_i (see Equation 8) and where the other symbols are as defined in 12.4.

In this case the combined standard uncertainty of AC_{avg} is given by: $u_{cC}(AC_{avg})$=

$$\sqrt{u_{cC}^2(AC_{avg}) + \frac{\sum_{i=1}^{n}\left(AC_i^2\,\times\,I_i^2\,\times\,u_r^2\,(\epsilon_i)\right)}{\left(\sum_{i=1}^{n} I_i\right)^2} + AC_{avg}^2 \times \left(\frac{\sum_{i=1}^{n} u^2(I_i)}{\left(\sum_{i=1}^{n} I_i\right)^2} + u_r^2\,(V_a) + u_r^2\,(\ldots)\right)} \tag{12}$$

where:

$u_r(\epsilon_i)$ = relative standard uncertainty of the detection efficiency for the i^{th} peak,
$u(I_i)$ = standard uncertainty of the gamma fraction I_i,
$u_r(V_a)$ = relative standard uncertainty of the sample volume or mass measurement,

Other symbols are as defined in 12.4.

NOTE 7—There may be additional hard-to-quantify uncertainty due to correlations among the peak efficiencies ϵ_i and gamma fractions I_i. The correlation coefficients may be unknown.

12.6 Expanded Uncertainty *(U)*:

$$U(AC) = ku_c(AC) \qquad or \qquad U(AC_{avg}) = ku_c(AC_{avg}) \tag{13}$$

where:

k = coverage factor for the desired level of confidence (typically k = 2 for approximately 95 % confidence).

12.7 Calculate the critical activity concentration (L_C) for each sample measurement as follows:

$$L_C(Bq/unit) = \frac{1.645\sqrt{\frac{R_b}{t_s}\times(F + F^2)}}{\epsilon \times I \times V_a \times e^{-\lambda t}} \tag{14}$$

where:

F = n_p/n_b, as described above

To decide whether the radionuclide of interest is clearly present in the sample, compare the measured activity concentration, AC, to L_C.

NOTE 8—Generally the critical activity concentration is calculated from data associated with the most abundant gamma-ray energy (largest value of I). However, if interferences, a large Compton continuum, or other factors prevail, use of data associated with gamma-rays of lesser abundance may be necessary. In any case, it is necessary to select a single gamma-ray full energy peak to represent the nuclide in question and use this peak for the comparison of measured activity and critical activity.

12.8 Calculate the *a priori* minimum detectable activity concentration (MDC) as follows:

$$MDC(Bq/unit) = \frac{\frac{2.71}{t_s} + 3.29\sqrt{\frac{R_b}{t_s}\times(F + F^2)}}{\epsilon \times I \times V_a \times e^{-\lambda t}} \tag{15}$$

13. Quality Control

13.1 In order to provide reasonable assurance that the analytical results obtained using this practice are valid and accurate within the confidence limits of the practice, Quality Control (QC) samples are analyzed with each batch of samples undergoing analysis. Each batch should include not more than 20 samples excluding those used for QC purposes. Laboratory or project quality assurance plans may contain more restrictive process QC requirements. The following minimum QC procedures should be followed when running the practice unless project measurement quality objectives provide different guidance.

13.2 *Calibration and System Quality Control Testing:*

13.2.1 Standards used in the practice should be traceable to a national standards laboratory (such as NIST or NPL).

13.2.2 The detector system performance should be verified on a daily basis or prior to use. See Practice D7282 for additional guidance. This recommendation may be relaxed if time does not permit, for example when screening samples following an emergency.

13.3 *Laboratory Control Sample (LCS):*

13.3.1 To ensure that the practice is within control limits, analyze an LCS with each batch of no more than 20 samples. The activity of the LCS should be appropriate for the type of samples analyzed and allow sufficient precision to ensure a meaningful assessment of accuracy. The result obtained for the LCS should fall within the limit of ±25 % of the expected value. The LCS may be a calibration verification standard that is reused with each batch of samples counted.

13.4 *Method Blank:*

13.4.1 Prepare and analyze a test blank with each batch of no more than 20 samples. The test blank should be expected to contain no significant concentration of radionuclides and if possible should be of similar media and geometry of the test samples. The activity concentration of radionuclides found in the blank should not compromise the applicable measurement quality objectives. The blank should be prepared in the same location as the samples to monitor for contamination in the sample handling and preparation facilities.

13.5 *Duplicate:*

13.5.1 To check the precision of sample analyses, analyze a sample in duplicate with each batch of no more than 20 samples. Calculate the statistical agreement (duplicate error ratio (DER)) between the two results. This calculation is performed using the combined standard uncertainty of each result as shown below.

$$DER = \frac{|AC_{original} - AC_{dup}|}{\sqrt{u_c^2(AC_{original}) + u_c^2(AC_{dup})}} \quad (16)$$

where:

$AC_{original}$ = original sample activity concentration,

AC_{dup} = duplicate sample activity concentration,

$u_c(AC_{original})$ = combined standard uncertainty of the original sample, and

$u_c(AC_{dup})$ = combined standard uncertainty of the duplicate sample.

13.5.2 In cases where there is insufficient sample volume to prepare two unique aliquots for counting, recount a sample and compare the results using the same DER criteria.

13.5.3 The value of DER should be less than or equal to 3.0. If the sample duplicate or recount result is not within these limits all samples in the batch should be recounted or an explanation provided in a case narrative.

13.6 *Independent Reference Material (IRM):*

13.6.1 In order to verify the quantitative value produced by the practice, analyze an IRM sample, which was submitted on at least a single-blind basis (if practical) to the laboratory at least once per quarter. The concentration of analyte in the traceable reference material should be appropriate to the typical purpose for which the practice is used. The value obtained should demonstrate acceptable performance as defined by the program or the outside source.

13.6.2 In the absence of other acceptance criteria for the IRM sample, compare the IRM sample result to the IRM known value as follows:

$$R = \frac{|IRM_{found} - IRM_{known}|}{\sqrt{u_c^2(IRM_{found}) + u_c^2(IRM_{known})}} \quad (17)$$

where:

R = relative difference,

IRM_{found} = found concentration of the IRM,

IRM_{known} = known concentration of the IRM,

$u_c(IRM_{found})$ = combined standard uncertainty of the IRM found concentration, and

$u_c(IRM_{known})$ = combined standard uncertainty of the IRM known concentration.

13.6.3 The value of R should be less than or equal to 3.0. If the value of R is greater than 3.0, the practice should be investigated to determine the cause.

13.7 The user should consult practice D7282 for additional quality control guidance.

14. Keywords

14.1 debris; emergency response; food; gamma spectrometry; germanium; HPGe; low-level activity; radioactivity; radionuclide ; soil; vegetation; water