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WVDP Workplace Radiological Surface Measurements Program And Technical Basis Document

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Table of Contents

1.0	PURPOSE	4
2.0	SCOPE AND APPLICATION	4
3.0	DEFINITIONS	5
4.0	RESPONSIBILITIES	10
5.0	CHARACTERIZATION GRID (REFERENCE COORDINATE) SYSTEMS	11
5.1	Indoor Characterization Grids	12
5.2	Outdoor Characterization Grids	12
5.3	Characterization Grid Marking and Grid Point Identification	13
6.0	INSTRUMENTATION	17
6.1	Radiation Detectors	17
6.1.1	Gas-Filled Detectors	17
6.1.2	Scintillation Detectors	17
6.1.3	Solid-State Detectors	17
6.1.4	Passive Integrating Detectors	18
6.2	Display and Recording Equipment	21
6.3	Detection Sensitivity (of Instrumentation)	22
6.4	Direct Measurement Sensitivity	22
6.5	Scanning Sensitivity	26
6.5.1	Scanning for Alpha Emitters	27
6.5.2	Scanning for Beta and Gamma Emitters	29
6.6	Applications - Instrument Selection	37
6.7	Instrument Calibration	38
7.0	MEASUREMENT METHODS	43
7.1	Direct Measurements	43
7.2	Scanning Surveys	44
7.2.1	Action Levels	45
7.3	Systematic Measurements and Sampling	45
7.4	Biased Measurements and Samples	46
7.5	Removable Surface Activity	46
7.6	Gamma Measurements	47
7.6.1	Direct Measurements for Photon Emitting Radionuclides	47
7.6.2	Scanning for Photon Emitting Radionuclides	48
7.7	Alpha Measurements	49
7.7.1	Direct Measurements for Alpha Emitting Radionuclides	49
7.7.2	Scanning for Alpha Emitting Radionuclides	49
7.8	Beta Measurements	50
7.8.1	Direct Measurements for Beta Emitting Radionuclides	50
7.8.2	Scanning for Beta Emitting Radionuclides	50

7.9	Soil Sampling	50
7.9.1	Surface Soil.....	50
7.9.2	Subsurface.....	52
7.10	Background Measurements.....	54
7.10.1	Select Background Reference Areas.....	54
7.10.2	Determining Numbers of Background (Reference) Data Points.....	56
8.0	ANALYSIS OF MEASUREMENTS.....	56
8.1	Analytical Methods.....	56
8.1.1	Analysis of Smears	56
8.1.2	Analysis of Soil and Sediment	57
8.2	Analytical Procedures	59
8.2.1	Photon Emitting Radionuclides.....	62
8.2.2	Beta Emitting Radionuclides.....	62
8.2.3	Alpha Emitting Radionuclides	63
9.0	CALCULATION.....	64
9.1	Data Conversion	64
9.1.1	Surface Activity	65
9.1.2	Surface Dose Rates.....	68
9.1.3	Soil Radionuclide Concentration and Exposure Rates.....	68
9.2	Measurement Uncertainty (Error)	70
9.3	Systematic and Random Uncertainties.....	70
9.4	Statistical Counting Uncertainty.....	72
9.5	Uncertainty Propagation	72
9.6	Reporting Confidence Intervals	73
9.7	Detection Sensitivities.....	74
10.0	RADIOACTIVE CONTAMINATION CONTROL - RELEASE CRITERIA.....	75
10.1	Title 10, CFR, Part 835 Requirements.....	75
10.2	DOE Radiological Control Standard Requirements.....	75
10.3	DOE O 458.1 Requirements.....	77
10.4	Application of Derived Concentration Guideline Levels for Soil Activity Correlations	79
11.0	REFERENCES	81

1.0 PURPOSE

This manual contains a discussion of guidelines, methods, and a technical basis for Radiological Controls Department personnel to use in planning, conducting, and/or evaluating radiological surveys in characterizing workplace surfaces and materials and for performing radiological measurements for releasing structures, equipment, and materials. This manual discusses measurements of both surficial soils and surfaces of structures, equipment, and materials. This manual also presents the survey requirements for releasing equipment and materials for unrestricted use. Radiological survey information obtained by the guidelines and methods presented in this manual may be used in deciding the fate of these materials or areas and their radiological contamination status. This manual does not address the requirements for routine radiological surveys of occupied work areas where biased measurements are made in determining the spread of contamination or for the conduct of specific work activities.

This manual provides a methodology that can be used to determine the radiological contamination status of surfaces through a series of measurements and/or samples collected. The data obtained may then be statistically tested to ensure that the area is adequately characterized with respect to average and ranges of radiological contamination levels. The technical basis for a characterization process is provided so that consistent evaluations and assessments are made with respect to each area surveyed. A technical basis is also provided for radiological instruments used and survey methodologies in performing surface radioactivity measurements.

The scope of radiological surveys in characterizing land, structures, and equipment, and materials should be commensurate with the potential for contamination of the area. Radiological surveys are performed to verify contamination levels for decontamination and decommissioning plans, to ensure or verify that an area will not expose individuals to unacceptable levels of radiation and radioactive materials, and when land, structures, or materials are being released from radiological control, to demonstrate that limits for residual radioactive material have not been exceeded.

2.0 SCOPE AND APPLICATION

This manual applies to workplace radiological measurements performed by Radiological Controls Department personnel on both surficial soils and surfaces of structures, equipment, and materials. This manual presents guidelines, methods, and a technical basis for Radiological Controls Department personnel to use in planning, conducting, and/or evaluating radiological surveys in characterizing workplace surfaces and materials and for performing radiological measurements for releasing structures, equipment, and materials. This manual supports Phase II and other similar activities of the WVDP that require formal radiological survey and measurement information.

Title 10, Code of Federal Regulations, Part 835; DOE O 458.1, NUREG-1575 (Multi-Agency Radiation Survey and Site Investigation Manual, MARSSIM), and DOE-STD-1098-2008 contain requirements for releasing soil and structures, equipment, and materials. Soil limits are established by pathway analysis per DOE O 458.1 requirements. Surface radioactivity limits are provided for structures, equipment, and materials that may be released to Controlled Areas in 10 CFR 835, and DOE-STD-1098-2008; however, for unrestricted releases to off-site areas, DOE O 458.1 requirements apply and MARSSIM discusses the free release of structures and surface soil in detail. If media other than structures and surface soil are potentially contaminated at the time of final survey, modifications to the MARSSIM survey design guidance and examples may be required.

WVDP-010, "WVDP Radiological Controls Manual," contains site requirements for conducting and documenting results of radiological surveys. Radiological Controls Departmental procedures (i.e., Radiological Control Procedures or RC's) provide implementing details, instructions, and forms for radiological control technicians when performing radiological surveys. This manual discusses programmatic requirements and provides a technical basis for methods used in the conduct of workplace radiological surveys for characterization of land or soil, and structures, equipment, and materials. To release soil, structures, equipment, and materials for unrestricted release, the requirements established in WVDP-010 and radiological control procedures must be satisfied.

3.0 DEFINITIONS

Action Level - the numerical value that will cause the decision maker to choose one of the alternative actions. It may be a regulatory threshold standard (e.g., Maximum Contaminant Level for drinking water), a dose- or risk-based concentration level (e.g., DCGL), or a reference-based standard.

Alpha (α) - the specified maximum probability of a Type I error. In other words, the maximum probability of rejecting the null hypothesis when it is true. Alpha is also referred to as the size of the test. Alpha reflects the amount of evidence the decision maker would like to see before abandoning the null hypothesis.

Area of Elevated Activity - an area over which residual radioactivity exceeds a specified value $DCGL_{EMC}$.

Area Factor (A_m) - a factor used to adjust $DCGL_W$ to estimate $DCGL_{EMC}$ and the minimum detectable concentration for scanning surveys in Class 1 survey units - $DCGL_{EMC} = DCGL_W \cdot A_m$. A_m is the magnitude by which the residual radioactivity in a small area of elevated activity can exceed the $DCGL_W$ while maintaining compliance with the release criterion. Examples of area factors are provided in Chapter 5 of this manual.

Becquerel (Bq) - the International System (SI) unit of activity equal to one nuclear transformation (disintegration) per second. $1 \text{ Bq} = 2.7 \times 10^{-11} \text{ Curies (Ci)} = 27.03 \text{ picocuries (pCi)}$.

Beta (β) - the probability of a Type II error, i.e., the probability of accepting the null hypothesis when it is false. The complement of beta ($1-\beta$) is referred to as the power or the test.

Bias - the systematic or persistent distortion of a measurement process which causes errors in one direction (i.e., the expected sample measurement is different from the sample's true value).

Biased Sample or Measurement (Judgment Measurement) - measurements performed at locations selected using professional judgment based on unusual appearance, location relative to known contaminated areas, high potential for residual radioactivity, general supplemental information, etc. Judgment measurements are not included in the statistical evaluation of the survey unit data because they violate the assumption of randomly selected, independent measurements. Instead, judgment measurements are individually compared to the $DCGL_W$.

Class 1 Area - an area that is projected to require a Class 1 final status survey.

Class 1 Survey - a type of final status survey that applies to areas with the highest potential for contamination, and meet the following criteria: (1) impacted; (2) potential for delivering a dose above the release criterion; (3) potential for small areas of elevated activity; and (4) insufficient evidence to support reclassification as Class 2 or Class 3.

Class 2 Area - an area that is projected to require Class 2 final status survey.

Class 2 Survey - a type of final status survey that applies to areas that meet the following criteria: (1) impacted; (2) low potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

Class 3 Area - an area that is projected to require a Class 3 final status survey.

Class 3 Survey - a type of final status survey that applies to areas that meet the following criteria: (1) impacted; (2) little or no potential for delivering a dose above the release criterion and (3) little or no potential for small areas of elevated activity.

Cleanup (survey) Unit - a geographical area of specified size and shape defined for the purpose of survey design and compliance testing.

Confidence Interval - a range of values for which there is a specified probability (e.g., 80%, 90%, 95%) that this set contains the true value of an estimated parameter.

Critical Level (L_c) - a fixed value of the test statistic corresponding to a given probability level, as determined from the sampling distribution of the test statistic. L_c is the level at which there is a statistical probability with a predetermined confidence of correctly identifying a background value as "greater than background."

Conversion Factor - a mathematically derived factor experimentally determined that converts experimental system response to actual values.

DCGL (Derived Concentration Guideline Level) - a derived, radionuclide-specific activity concentration within a survey unit corresponding to the release criterion. The DCGL is based on the spatial distribution of the contaminant and hence is derived differently for the nonparametric statistical test ($DCGL_W$) and the Elevated Measurement Comparison ($DCGL_{EMC}$). DCGLs are derived from activity/dose relationships through various exposure pathway scenarios.

DQA (Data Quality Assessment) - the scientific and statistical evaluation of data to determine if the data are of the right type, quality, and quantity to support their intended use.

DQOs (Data Quality Objectives) - qualitative and quantitative statements derived from the DQO process that clarify technical and quality objectives, define the appropriate type of data, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions.

Decontamination - the removal of chemical, biological, or radiological contaminants from, or their neutralization on, a person, object, or area to within levels established by governing regulatory agencies.

Delta (δ) - the amount that the distribution of measurements for a survey unit is shifted to the right of the distribution of measurements of the reference area.

Delta (Δ) - the width of the gray region. Δ divided by σ , the arithmetic standard deviation of the measurements, is the relative shift expressed in multiple of standard deviations.

Detection Limit - the net response level that can be expected to be seen with a detector with a fixed level of certainty.

Detection Sensitivity - the minimum level of ability to identify the presence of radiation or radioactivity.

Direct Measurement - radioactivity measurement obtained by placing the detector near the surface or media being surveyed. An indication of the resulting radioactivity level is read out directly.

Disintegration (nuclear) - a spontaneous nuclear transformation (radioactivity) characterized by the emission of energy and/or mass from the nucleus of an atom. When large numbers of nuclei are involved, the process is characterized by a definite half-life.

Effective Probe Area - the physical probe area corrected for the amount of the probe area covered by a protective screen.

Efficiency - a measure of the probability that a nuclear disintegration will be detected when radiation is incident onto a detector.

Elevated Measurement - a measurement that exceeds a specified value $DCGL_{EMC}$.

Elevated Measurement Comparison (EMC) - this comparison is used in conjunction with the Wilcoxon test to determine if there are any measurements that exceed a specified value $DCGL_{EMC}$.

Exposure Rate - radiation exposure delivered per unit time, normally in roentgens per hour.

False Negative Decision Error - the error that occurs when the null hypothesis (H_0) is not rejected when it is false. A false negative error is often referred to as a Type II decision error. The measure of the size of this error is called beta, and is also known as the complement of the power of a hypothesis test.

False Positive Decision Error - a false positive decision error occurs when the null hypothesis (H_0) is rejected when it is true. A false positive error is often referred to as a Type I decision error. The measure of the size of this error is called alpha, the level of significance, or the size of the critical region.

Final Status Survey - measurements and sampling to describe the radiological conditions of a site, following completion of decontamination activities (if any) in preparation for release.

Fixed contamination - radioactive material that has been deposited onto a surface and cannot be readily removed by non-destructive means, such as casual contact, wiping, brushing, or laundering. Fixed contamination does not include radioactive material that is present in a matrix, such as soil or cement, or radioactive material that has been induced in a material through activation processes.

Gamma Scintillator - a crystal detector that emits visible light in proportion to the intensity of a gamma-ray field. The visible light is converted to an electric current by a photomultiplier tube.

Geiger-Mueller Counter - highly sensitive, gas-filled device for measuring radiation that operates at voltages sufficiently high to produce multiple ionizations from each interaction with radiation.

Gray Region - a range of values of the parameter of interest for a survey unit where the consequences of making a decision error are relatively minor. The upper bound of the gray region in MARSSIM is set equal to the $DCGL_w$, and the lower bound of the gray region (LBGR) is a site-specific variable.

Grid - a network of parallel horizontal and vertical lines forming squares on a map that may be overlaid on a property parcel for the purpose of identification of exact locations.

Grid Block - a square defined by two adjacent vertical and two adjacent horizontal reference grid lines.

Historical Site Assessment (HSA) - a detailed investigation to collect existing information, primarily historical, on a site and its surroundings.

Ionization Chamber - an instrument that detects and measures ionizing radiation by measuring the electrical current that flows when radiation ionizes gas in a chamber, making the gas a conductor of the electricity.

Lower Bound of the Gray Region (LBGR) - the minimum value of the gray region. The width of the gray region ($DCGL-LBGR$) is also referred to as the shift, Δ

Lower Limit of Detection (L_D) - the smallest amount of radiation or radioactivity that statistically yields a net result above the method background. The critical detection level, L_C , is the lower bound of the 95% detection interval defined for L_D and is the level at which there is a 5% chance of calling a background value "greater than background." This value should be used when actually counting samples or making direct radiation measurements. Any response above this level should be considered as above background; i.e., a net positive result. This will ensure 95% detection capability for L_D . A 95% confidence interval should be calculated for all responses greater than L_C .

Minimum Detectable Activity - the lowest level of radioactivity that can be measured precisely using a particular device.

Minimum Detectable Concentration (MDC) - the minimum detectable concentration (MDC) is the *a priori* activity level that a specific instrument and technique can be expected to detect 95% of the time. When stating the detection capability of an instrument, this value should be used. The MDC is the detection limit, L_D , multiplied by an appropriate conversion factor to give units of activity.

Minimum Detectable Count Rate (MDCR) - the minimum detectable count rate (MDCR) is the *a priori* count rate that a specific instrument and technique can be expected to detect.

Multichannel Analyzer - an electronic device for sorting successive signal pulses into parallel amplitude channels.

Nonparametric Test - a test based on relatively few assumptions about the exact form of the underlying probability distributions of the measurements. As a consequence, nonparametric tests are generally valid for a fairly broad class of distributions. The Wilcoxon Rank Sum test and the Sign test are examples of nonparametric tests.

Physical Probe Area - the physical surface area assessed by a detector. The physical probe area is used to make probe area corrections in the activity calculations.

Power (1- β) - the probability of rejecting the null hypothesis when it is false. The power is equal to one minus the Type II error rate, i.e., (1- β).

Preliminary Survey - a radiological survey conducted on a site to determine if a radiological hazard exists or the site warrants a more comprehensive radiological survey because of the presence of residual radioactive materials.

Pressurized Ion Chamber - a pressurized ionization chamber, or ion chamber, is a detector that collects ion pairs formed by the interaction of radiation with high-pressure gases within the chamber.

Proportional Counter - gas-filled radiation detection tube in which the electrical pulse produced is proportional to the number of ions formed in the gas by the incident radiation.

Quality Assurance (QA) - an integrated system of management activities involving planning, implementation, assessment, reporting, and quality improvement to ensure that a process, item, or service is of the type and quality needed and expected by the customer.

Quality Assurance Project Plan (QAPP) - a formal document describing in comprehensive detail the necessary QA, QC, and other technical activities that must be implemented to ensure that the results of the work performed will satisfy the stated performance criteria. As defined for Superfund in the Code of Federal Regulations 40 CFR 300.430, the Quality Assurance Project Plan describes policy, organization, and functional activities and the Data Quality Objectives and measures necessary to achieve adequate data for use in selecting the appropriate remedy. The QAPP is a plan that provides a process for obtaining data of sufficient quality and quantity to satisfy data needs. It is a part of the Sampling and Analysis Plan.

Quality Control (QC) - the overall system of technical activities that measure the attributes and performance of a process, item, or service against defined standards to verify that they meet the stated requirements established by the customer, operational techniques and activities that are used to fulfill requirements for quality.

Radiological Survey - measurements of radiation levels and radioactivity associated with a site together with appropriate documentation and data evaluation.

Reference Area - geographical area from which representative reference measurements are performed for comparison with measurements performed in specific survey units at remediation site. A site radiological reference area (background area) is defined as an area that has similar physical, chemical, radiological, and biological characteristics as the site area being remediated, but which has not been contaminated by site activities. The distribution and concentration of background radiation in the reference area should be same as that which would be expected on the site if that site had never been contaminated. More than one reference area may be necessary for valid comparisons if a site exhibits considerable physical, chemical, radiological, or biological variability.

Reference Coordinate System - a grid of intersecting lines referenced to a fixed site location or benchmark. Typically the lines are arranged in a perpendicular pattern dividing the survey location into squares or blocks of equal areas. Other patterns include three-dimensional and polar coordinate systems.

Remedial Action - those actions that are consistent with a permanent remedy taken instead of, or in addition to, removal action in the event of a release or threatened release of a hazardous substance into the environment, to prevent or minimize the release of hazardous substances so that they do not migrate to cause substantial danger to present or future public health or welfare or the environment.

Removable radioactivity - surface activity that is readily removable by wiping the surface with moderate pressure and can be assessed with standard radiation detectors. It is usually expressed in units of $\text{dpm}/100 \text{ cm}^2$.

Residual Radioactive Material - material (including but not limited to waste material, soils, rocks, plants, shrubs, personal property, and building materials) present at a site that results in radiation levels that exceed background levels.

Sample - (as used in MARSSIM) a part or selection from a medium located in a survey unit or reference area that represents the quality or quantity of a given parameter for nature of the whole area or unit; a portion serving as a specimen.

Sample - (as used in statistics) a set of individual samples or measurements drawn from a population whose properties are studied to gain information about the entire population.

Scanning - an evaluation technique performed by moving a detection device over a surface at a specified speed and distance above the surface to detect radiation.

Scintillation Counter - the combination of phosphors, photomultiplier tube, and associated circuits for counting light emissions produced in the phosphors by incident ionizing radiation.

Sievert (Sv) - The special name for the International System (SI) unit of dose equivalent $1 \text{ Sv} = 100 \text{ rem} = 1 \text{ Joule per kilogram}$.

Sign Test - a nonparametric statistical test used to demonstrate compliance with the release criterion when the radionuclide of interest is not present in background and the distribution of data is not symmetric. (See also Wilcoxon Rank Sum Test).

Smear Sample - a sample taken by "smearing" a piece of filter paper over suspected areas of surface contamination.

Sodium Iodide (NaI) Detector - a detector that uses a sodium iodide (thallium activated) crystal for detecting gamma rays.

Soil - the top layer of the earth's surface, consisting of rock and mineral particles mixed with organic matter. A particular kind of earth or ground - e.g., sandy soil.

Subsurface Soil Sample - a soil sample that reflects the modeling assumptions used to develop the DCGL for subsurface soil activity. An example would be soil taken deeper than 15 cm below the soil surface to support surveys performed to demonstrate compliance with 40 CFR 192.

Surface Barrier Detector - a type of semiconductor detector, such as silicon, having a depletion region in the crystal and a thin gold film electrode.

Surface Soil Sample - a soil sample that reflects the modeling assumptions used to develop the DCGL for surface soil activity. An example would be soil taken from the first 15 cm of surface soil to support surveys performed to demonstrate compliance with 40 CFR 192.

Survey Meter - any portable radiation detecting instrument especially adapted for surveying or inspecting an area to establish the existence of radioactive material.

Survey Plan - a radiological survey plan for determining the radiological characteristics of a specific site.

Survey Unit - a geographical area consisting of structures or land areas of specified size and shape at a remediated site for which a separate decision will be made whether the unit attains the site-specific reference-based cleanup standard for the designated pollution parameter. Survey units are generally formed by grouping contiguous site areas with a similar use history and the same classification of contamination potential. Survey units are established to facilitate the survey process and the statistical analysis of survey data.

Systematic Sample/Measurement - samples/measurements taken under a definite method or plan.

Type I Decision Error - a decision error that occurs when the null hypothesis is rejected when it is true. The probability of making a Type I decision error is called alpha (α).

Type II Decision Error - a decision error that occurs when the null hypothesis is accepted when it is false. The probability of making a Type II decision error is called beta (β).

Wilcoxon Rank Sum (WRS) Test - a nonparametric statistical test used to determine compliance with the release criterion when the radionuclide of concern is present in background. (See also Sign Test)

Zinc Sulfide (ZnS) Detector - a detector that uses ZnS powder as the detection medium that is used for detection of alpha particles or other heavy ions.

4.0 RESPONSIBILITIES

- 4.1 Radiological Controls (RC) Manager, or designee - responsible for conducting and maintaining a radiological workplace sampling and monitoring program to characterize surface radioactivity concentrations in land or soil and on structures, equipment, and materials. The RC Manager provides radiological engineering and technical support for radiological workplace sampling and monitoring activities and technical evaluations of radiological measurement data collected. Responsible for providing personnel and/or for arranging for the collection and counting of samples, radiological measurements, instrument operational checks, calculation of sample results, and for reviewing results of radiological surveys and measurements for quality and completeness.
- 4.2 Radiological Control Technicians (RCTs) - responsible for collecting and counting samples, performing radiological measurements, performing instrument operational checks, and calculating and recording sample results.

5.0 CHARACTERIZATION GRID (REFERENCE COORDINATE) SYSTEMS

Radiological measurements and samples should be collected relative to a grid system that has been prepared for the particular area. Reference coordinate systems are established at the site to:

- Facilitate selection of measurement and sampling locations
- Provide a mechanism for referencing a measurement to a specific location so that the same survey point can be relocated

A survey reference coordinate system consists of a grid of intersecting lines, referenced to a fixed site location or benchmark. Typically, the lines are arranged in a perpendicular pattern, dividing the survey location into squares or blocks of equal area; however, other types of patterns (e.g., three-dimensional, polar) have been used.

The reference coordinate system used for a particular survey should provide a level of reproducibility consistent with the objectives of the survey. For example, a commercially available global positioning system will locate a position within tens of meters, while a differential global positioning system (DGPS) provides precision on the order of a few centimeters. On the other hand, a metal bar can be driven into the ground to provide a long-term reference point for establishing a local reference coordinate system.

Reference coordinate system patterns on horizontal surfaces are usually identified numerically on one axis and alphabetically on the other axis or in distances in different compass directions from the grid origin. Examples of structure interior and land area grids are shown in Figures 1 through 3. Grids on vertical surfaces may include a third designator, indicating position relative to floor or ground level. Overhead measurement and sampling locations (e.g., ceiling and overhead beams) are referenced to corresponding floor grids.

For surveys of MARSSIM Class 1 and Class 2 areas, basic grid patterns at 1 to 2 meter intervals on structure surfaces and at 10 to 20 meter intervals of land areas may be sufficient to identify survey locations with a reasonable level of effort, while not being prohibitive in cost or difficulty of installation. Gridding of MARSSIM Class 3 areas may also be necessary to facilitate referencing of survey locations to a common system or origin but, for practical purposes, may typically be at larger intervals, e.g., 5 to 10 meters for large structural surfaces and 20 to 50 meters for land areas.

Three basic coordinate systems are used for identifying points on a reference coordinate system.

- The reference system shown in Figure 1 references grid locations using numbers on the vertical axis and letters on the horizontal axis.
- The reference system shown on Figure 2 references distances from the 0,0 point using the compass directions N (north), S (south), E (east), and W (west).
- The reference system shown in Figure 3 references distances along and to the R (right) or L (left) of the baseline.
- In addition, a less frequently used reference system is the polar coordinate system, which measures distances along transects from a central point. Polar coordinate systems are particularly useful for survey designs to evaluate effects of stack emissions, where it may be desirable to have a higher density of samples collected near the stack and fewer samples with increasing distance from the stack.

5.1 Indoor Characterization Grids

A common grid spacing for building interiors is 1 m (3.25 ft); however, spacing will vary depending upon the needs of the survey. Grid size may be increased for areas having a low potential for contamination or decreased in areas of heavier contamination. Adjustment of grid size may be particularly applicable to areas where there is no documented or verifiable evidence that any radioactive materials were ever stored or used there. Thus, grids of greater than 1 m (3.25 ft) may be used where there is knowledge that no radioactive material was ever used. Gridding may be limited to the floor and lower walls [up to 2 m (6.4 ft) height], unless there is also a potential for upper wall and ceiling area contamination.

Horizontal grid patterns are typically identified numerically on one axis and alphabetically on the other axis. The floor grid pattern is usually extended up vertical surfaces (walls). Overhead measurement/sampling locations (e.g., ceiling and overhead beams) are referenced to corresponding floor grids. An example of a typical building grid is shown in Figure 1. For some radionuclides, scanning surveys have adequate sensitivity to detect the approved authorized limit (see DOE O 458.1). Under these circumstances, grid sizes may be irrelevant.

5.2 Outdoor Characterization Grids

Scoping surveys of small area properties may be performed without the benefit of gridding. Large open land areas and all properties receiving a characterization survey should be gridded. Typical open land grids are illustrated in Figures 2 and 3.

The following equipment is necessary to perform the tasks outlined in this procedure:

- Transit/tripod
- Level rod
- Measuring tape
- Grid markers (e.g., stakes, flags, flagging tape)
- Chaining pins
- Indelible marker
- Fluorescent paint
- Mallet
- Machete/axe

The grid area considered appropriate for outdoor surveys under the current guideline structure is 100 m² (1076 ft²), the area over which data must be averaged in order to compare findings with guidelines. The grid size may be increased or decreased depending on the potential for contamination and the type of survey being performed. This may include areas having a low probability for contamination for a variety of possible reasons, e.g., areas subject to contamination by windblown residues originating from nearby contaminated sites or properties. On the other hand, when performing a confirmatory/verification survey to assess the adequacy of remedial action, a 2.5 m (8.2 ft) grid system may be appropriate for decontaminated areas of 100 m² (1076 ft²) or larger.

For areas less than 25 m² (269 ft²), a 1 m (3.25 ft) grid system may be used. Figure 3 shows an example grid system for an outdoor land area. The first digit or set of digits includes an L or R (separated from the first set by a comma) to indicate the distance from the baseline in units (meters) and the direction (left or right) from the baseline. The second digit or set of digits refers to the perpendicular distance from the 0.0 point on the baseline and is measured in hundreds of units. Point A in the example of a reference coordinate system for survey of site grounds, Figure 3, is identified 100R, 2+00 (i.e., 200 m from the baseline and 100 m to the right of the baseline). Fractional distances between reference points are identified by adding the distance beyond the reference point and are expressed in the same units used for the reference coordinate system dimensions. Point B on Figure 3 is identified 25R, 1+30.

Open land reference coordinate systems should be referenced to a location on an existing State or local reference system or to a U.S. Geological Survey (USGS) bench mark. (This may require the services of a professional land surveyor.) Global positioning systems (GPS) are capable of locating reference points in terms of latitude and longitude (MARSSIM Section 6.10.1 provides descriptions of positioning systems).

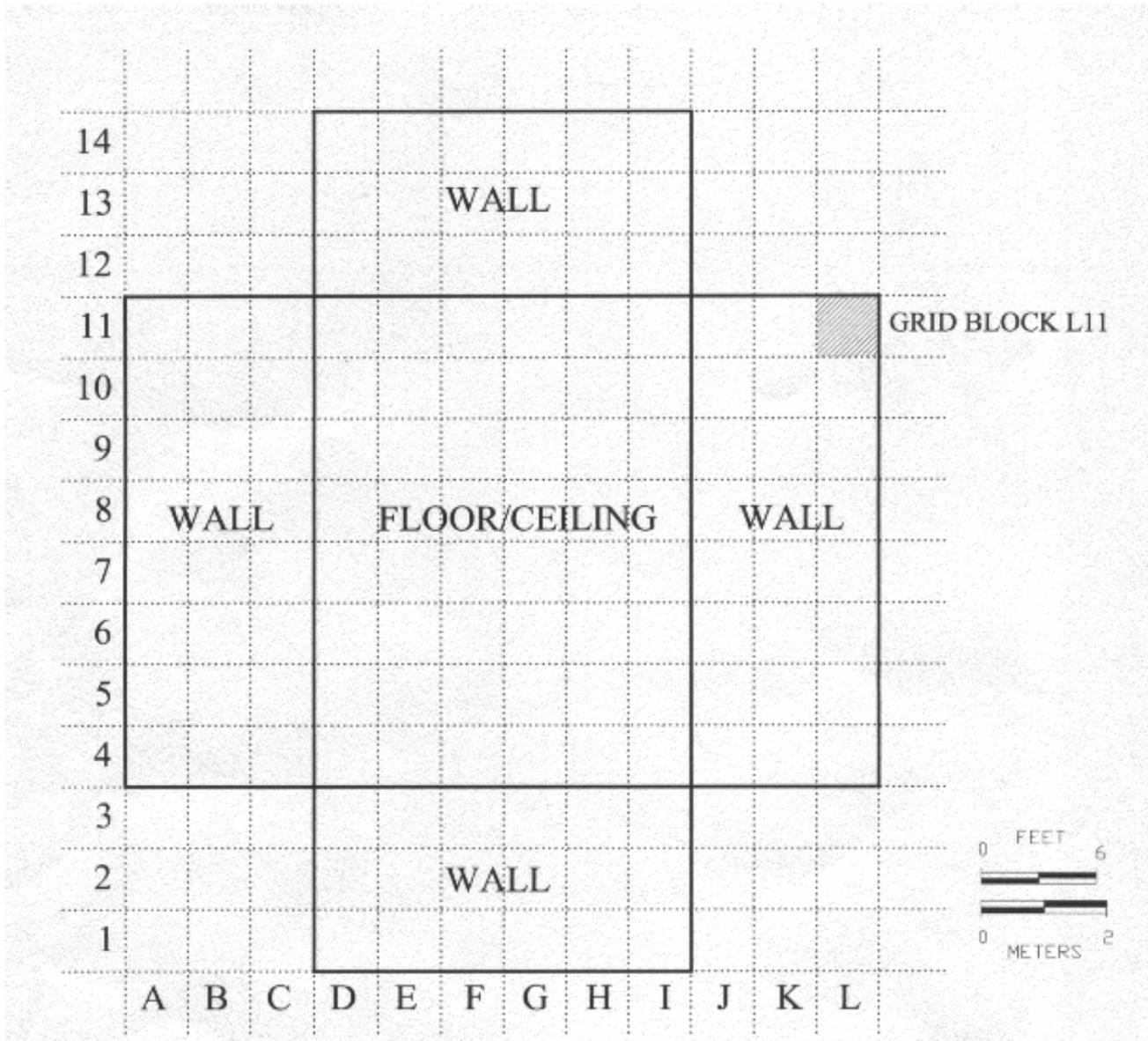
Following establishment of the reference coordinate system, a drawing is prepared by the survey team or the land surveyor. This drawing indicates the reference lines, site boundaries, and other pertinent site features and provides a legend showing the scale and a reference compass direction. The process used to develop the reference coordinate system should be recorded in the survey planning documentation (e.g., the Quality Assurance Project Plan or QAPP). Any deviations from the requirements developed during planning should be documented when the reference coordinate system is established. Characterization grid systems should be approved by the RC Manager or RC Supervisor.

It should be noted that the reference coordinate systems described in this section are intended primarily for reference purposes and do not necessarily dictate the spacing or location of survey measurements or samples. Establishment of a measurement grid to demonstrate compliance with the DCGL is discussed in MARSSIM, Section 5.5.2.5, and Chapter 8.

5.3 Characterization Grid Marking and Grid Point Identification

Reference coordinate systems on structure surfaces are usually marked by chalk line or paint along the entire grid line or at line intersections. Land area reference coordinate systems are usually marked by wooden or metal stakes, driven into the surface at reference line intersections. The selection of an appropriate marker depends on the characteristics and routine uses of the surface. Where surfaces prevent installation of stakes, the reference line intersection may be marked by painting. The selection of an appropriate marker depends on the characteristics and routine uses of the surface.

Figure 1



Indoor Grid Layout with Alphanumeric Grid Block Designation

*Walls and floor are diagramed as though they lay along the same horizontal plane.

Figure 2

Example of a Grid System for Survey of Site Grounds Using Compass Directions

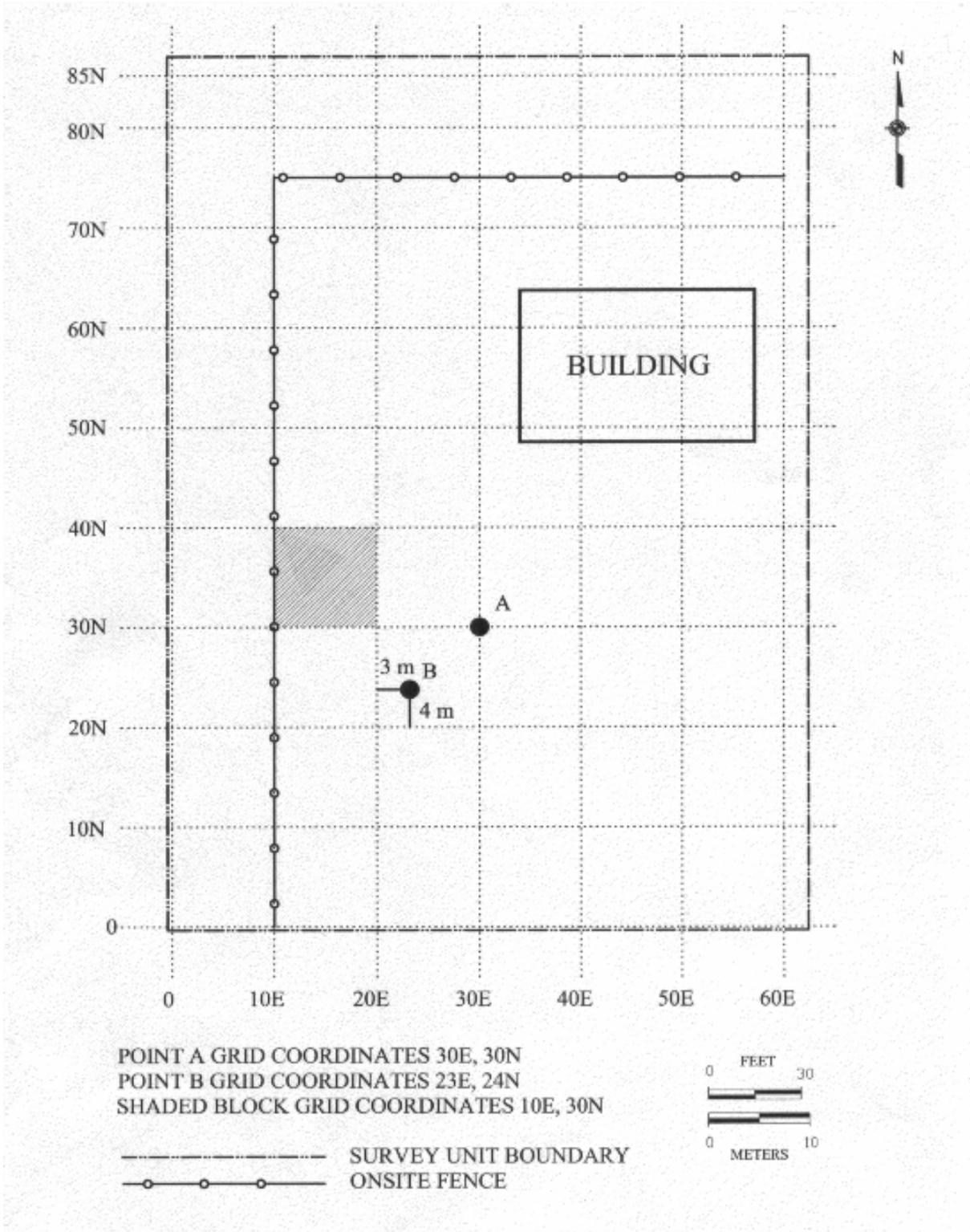
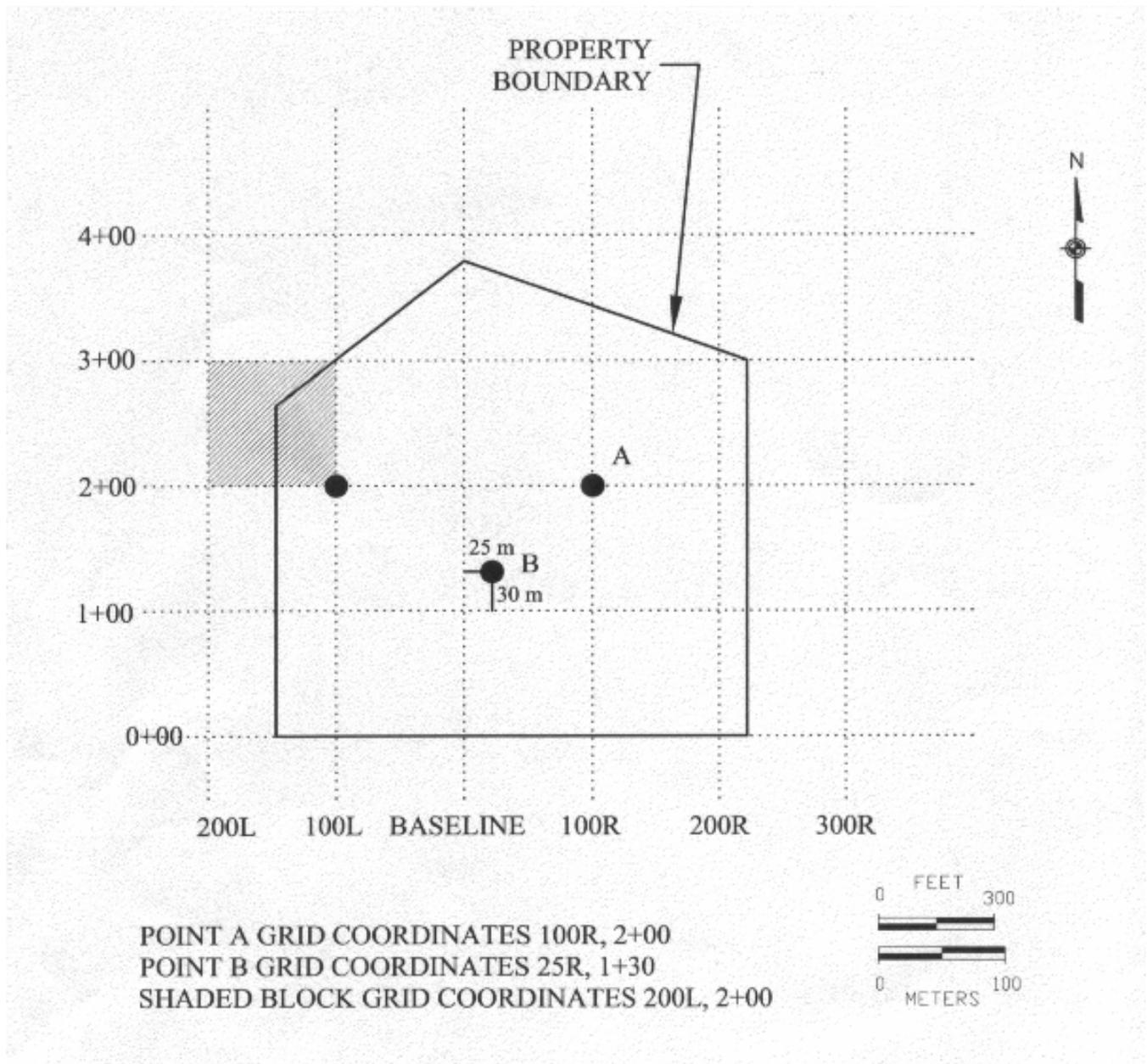


Figure 3

Example of a Grid System for Survey of Site Grounds Using Distances
Left or Right of the Baseline



6.0 INSTRUMENTATION

Traditional radiation instruments consist of two components: 1) a radiation detector, and 2) electronic equipment to provide power to the detector and to display or record radiation events. This section identifies and very briefly describes the types of radiation detectors and associated display or recording equipment that are applicable to survey activities in support of radiological assessment or remedial action.

Each survey typically requires performing direct field measurements using portable instrumentation and collection of samples for laboratory analysis. The selection and proper use of appropriate instruments for both direct measurements and laboratory analyses will likely be the most critical factors in assuring that the survey accurately determines the radiological status of a site and meets the survey objectives. MARSSIM, Chapter 7, provides specific information on laboratory analysis of collected samples. MARSSIM, Appendix H, contains instrument specific information for various types of field survey and laboratory analysis equipment currently in use.

6.1 Radiation Detectors

The particular capabilities of a radiation detector will establish its potential applications in conducting a specific type of survey. Radiation detectors can be divided into four general classes based on the detector material or the application. These categories are: 1) gas-filled detectors, 2) scintillation detectors, 3) solid-state detectors, and 4) passive integrating detectors. Lists of radiation detectors along with their usual applications to surveys are provided in Tables 1, 2, and 3.

6.1.1 Gas-Filled Detectors

Radiation interacts with the fill gas, producing ion-pairs that are collected by charged electrodes. Commonly used gas-filled detectors are categorized as ionization, proportional, or Geiger-Mueller (GM), referring to the region of gas amplification in which they are operated. The fill gas varies, but the most common are: 1) air, 2) argon with a small amount of organic methane (usually 10% methane by mass, referred to as P-10 gas), and 3) argon or helium with a small amount of a halogen such as chlorine or bromine added as a quenching agent.

6.1.2 Scintillation Detectors

Radiation interacts with a solid or liquid medium causing electronic transitions to excited states in a luminescent material. The excited states decay rapidly, emitting photons that in turn are captured by a photomultiplier tube. The ensuing electrical signal is proportional to the scintillator light output, which, under the right conditions, is proportional to the energy loss that produced the scintillation. The most common scintillate materials are NaI(Tl), ZnS(Ag), Cd(Te), and CsI(Tl) which are used in traditional radiation survey instruments such as the NaI(Tl) detector used for gamma surveys and the ZnS(Ag) detector for alpha surveys.

6.1.3 Solid-State Detectors

Radiation interacting with a semiconductor material creates electron-hole pairs that are collected by a charged electrode. The design and operating conditions of a specific solid-state detector determines the types of radiations (alpha, beta, and/or gamma) that can be measured, the detection level of the measurements, and the ability of the detector to resolve the energies of the interacting radiations. The semiconductor materials currently being used are germanium and silicon which are available in both n and p types in various configurations.

Spectrometric techniques using these detectors provide a marked increase in sensitivity in many situations. When a particular radionuclide contributes only a fraction of the total particle or photon fluence, or both, from all sources (natural or manmade background), gross measurements are inadequate and nuclide-specific measurements are necessary. Spectrometry provides the means to discriminate among various radionuclides on the basis of characteristic energies. *In situ* gamma spectrometry is particularly effective in field measurements since the penetrating nature of the radiation allows one to “see” beyond immediate surface contamination. The availability of large, high efficiency germanium detectors permits measurement of low abundance gamma emitters such as ^{238}U as well as low energy emitters such as ^{241}Am and ^{239}Pu .

6.1.4 Passive Integrating Detectors

There is an additional class of instruments that consists of passive, integrating detectors and associated reading/analyzing instruments. The integrated ionization is read using a laboratory or hand-held reader. This class includes thermoluminescent dosimeters (TLDs) and electret ion chambers (EICs). Because these detectors are passive and can be exposed for relatively long periods of time, they can provide better sensitivity for measuring low activity levels such as free release limits or for continuing surveillance. The ability to read and present data onsite is a useful feature and such systems are comparable to direct reading instruments.

- A. The scintillation materials in Section 6.1.2 are selected for their prompt fluorescence characteristics. In another class of inorganic crystals, called TLDs, the crystal material and impurities are chosen so that the free electrons and holes created following the absorption of energy from the radiation are trapped by impurities in the crystalline lattice thus locking the excitation energy in the crystal. Such materials are used as passive, integrating detectors.

After removal from the exposure area, the TLDs are heated in a reader which measures the total amount of light produced when the energy is released. The total amount of light is proportional to the number of trapped, excited electrons, which in turn is proportional to the amount of energy absorbed from the radiation. The intensity of the light emitted from the thermoluminescent crystals is thus directly proportional to the radiation dose. TLDs come in a large number of materials, the most common of which are LiF , $\text{CaF}_2:\text{Mn}$, $\text{CaF}_2:\text{Dy}$, $\text{CaSO}_4:\text{Mn}$, $\text{CaSO}_4:\text{Dy}$, $\text{Al}_2\text{O}_3:\text{C}$.

- B. The electret ion chamber consists of a very stable electret (a charged Teflon® disk) mounted inside a small chamber made of electrically charged plastic. The ions produced inside this air filled chamber are collected onto the electret, causing a reduction of its surface charge. The reduction in charge is a function of the total ionization during a specific monitoring period and the specific chamber volume. This change in voltage is measured with a surface potential voltmeter.

**Table 1 (MARSSIM Table 6.1)
 Radiation Detectors with Applications to Alpha Surveys**

Detector type	Detector description	Application	Remarks
Gas proportional	<1 mg/cm ² window; probe face area 50 to 1000 cm ²	Surface scanning; surface contamination measurement	Requires a supply of appropriate fill gas
	<0.1 mg/cm ² window; probe face area 10 to 20 cm ²	Laboratory measurement of water, air, and smear samples	
	No window (internal proportional)	Laboratory measurement of water, air, and smear samples	
Air Proportional	<1 mg/cm ² window; probe area ~ 50 cm ²	Useful in low humidity conditions	
Scintillation	ZnS(Ag) scintillator; probe face area 50 to 100 cm ²	Surface contamination measurements, smears	
	ZnS(Ag) scintillator; probe face area 10 to 20 cm ²	Laboratory measurement of water, air, and smear samples	
	Lucas scintillation flask	Laboratory measurement for low levels of radon	
	Liquid scintillation cocktail containing sample	Laboratory analysis, spectrometry capabilities	
Passive integrating electret ion chamber	<0.8 mg/cm ² window, also window-less, window area 50-180 cm ² , chamber volume 50-1,000 ml	Contamination on surfaces, in pipes and in soils	Useable in high humidity and temperature
Solid state	Silicon surface barrier detector	Laboratory analysis by alpha spectrometry	

**Table 2 (MARSSIM Table 6.2)
 Radiation Detectors with Applications to Beta Surveys**

Detector type	Detector description	Application	Remarks
Gas proportional	<1 mg/cm ² window; probe face area 50 to 1000 cm ²	Surface scanning; surface contamination measurement	Requires a supply of appropriate fill gas
	<0.1 mg/cm ² window; probe face area 10 to 20 cm ²	Laboratory measurement of water, air, and smear samples	
	No window (internal proportional)	Laboratory measurement of water, air, and smear samples	Can be used for measuring very low energy betas
Ionization (non-pressurized)	1-7 mg/cm ² window	Contamination measurements; skin dose rate estimates	
Geiger-Mueller	<2 mg/cm ² window; probe area 10 to 100 cm ²	Surface scanning; surface contamination measurement; laboratory analyses	
	Various window thickness; few cm ² probe face	Special scanning applications	
Passive integrating electret ion chamber	7 mg/cm ² window, also window-less, window area 50-180 cm ² , chamber volume 50-1000 ml	Low energy beta including H-3 contamination on surfaces and in pipes	Useable in high humidity and temperature
Scintillation	Liquid scintillation cocktail containing sample	Laboratory analysis; spectrometry capabilities	
	Plastic Scintillator	Contamination measurements	

**Table 3 (MARSSIM Table 6.3)
 Radiation Detectors with Applications to Gamma Surveys**

Detector type	Detector description	Application	Remarks
Gas ionization	Pressurized ionization chamber; Non-pressurized ionization chamber	Exposure rate measurements	
Geiger-Mueller	Pancake (<2 mg/cm ² window) or side window (~30 mg/cm ²)	Surface scanning; exposure rate correlation (side window in closed position)	Low relative sensitivity to gamma radiation
Scintillation	Nal(Tl) scintillator; up to 5 x 5 cm	Surface scanning; exposure rate correlation	High sensitivity; Cross calibrate with pressurized ionization chamber or for specific site gamma energy mixture for exposure rate measurements
	Nal(Tl) scintillator; large volume and "well" configurations	Laboratory gamma spectrometry	
	CsI or Nal(Tl) scintillator; thin crystal	Scanning; low energy gamma and x-rays	Detection of Low Energy Radiation
	Organic tissue equivalent (plastics)	Dose equivalent rate measurements	
Passive integrating electret ion chamber	7 mg/cm ² window, also window-less, window area 50-180 cm ² , chamber volume 50-1000 ml		Useable in high humidity and temperature
Solid state	Geranium semi-conductor	Laboratory and field gamma spectrometry and spectroscopy	

6.2 Display and Recording Equipment

Radiation detectors are connected to electronic devices to 1) provide a source of power for detector operation, and 2) enable measurement of the quantity and/or quality of the radiation interactions that are occurring in the detector. The quality of the radiation interaction refers to the amount of energy transferred to the detector. In many cases, radiation interacts with other material (e.g., air) prior to interacting with the detector, or only partially interacts with the detector (e.g., Compton scattering for photons). Because the energy recorded by the detector is affected, there is an increased probability of incorrectly identifying the radionuclide.

The most common recording or display device used for portable radiation measurement systems is a ratemeter. This device provides a display on an analog meter representing the number of events occurring over some time period (e.g., counts per minute). Digital ratemeters are also commercially available. The number of events can also be accumulated over a preset time period using a digital scaling device. The resulting information from a scaling device is the total number of events that occurred over a fixed period of time, where a ratemeter display varies with time and represents a short term average of the event rate. Determining the average level on a ratemeter will require judgment by the user, especially when a low frequency of events results in significant variations in the meter reading.

Pulse height analyzers are specialized electronic devices designed to measure and record the number of pulses or events that occur at different pulse height levels. These types of devices are used with detectors which produce output pulses that are proportional in height to the energy deposited within them by the interacting radiation. They can be used to record only those events occurring in a detector within a single band of energy or can simultaneously record the events in multiple energy ranges. In the former case, the equipment is known as a single-channel analyzer; the latter application is referred to as a multichannel analyzer.

6.3 Detection Sensitivity (of Instrumentation)

The detection sensitivity of a measurement system refers to a radiation level or quantity of radioactive material that can be measured or detected with some known or estimated level of confidence. This quantity is a factor of both the instrumentation and the technique or procedure being used.

The primary parameters that affect the detection capability of a radiation detector are the background count rate, the detection efficiency of the detector and the counting time interval. It is important to use actual background count rate values and detection efficiencies when determining counting and scanning parameters, particularly during final status and verification surveys. When making field measurements, the detection sensitivity will usually be less than what can be achieved in a laboratory due to increased background and, often times, a significantly lower detection efficiency. It is often impossible to guarantee that pure alpha emitters can be detected *in situ* since the weathering of aged surfaces will often completely absorb the alpha emissions. NUREG-1507 contains data on many of the parameters that affect detection efficiencies *in situ*, such as absorption, surface smoothness, and particulate radiation energy.

6.4 Direct Measurement Sensitivity

Prior to performing field measurements (for release of property), an investigator must evaluate the detection sensitivity of the equipment proposed for use to ensure that levels below the MARSSIM DCGL can be detected (see MARSSIM Section 4.3). After a direct measurement has been made, it is then necessary to determine whether or not the result can be distinguished from the instrument background response of the measurement system. The terms that are used in this manual to define detection sensitivity for fixed point counts and sample analyses are:

- Critical level (L_C)
- Detection limit (L_D)
- Minimum detectable concentration (MDC)

The critical level (L_C) is the level, in counts, at which there is a statistical probability (with a predetermined confidence) of incorrectly identifying a measurement system background value as greater than background. Any response above this level is considered to be greater than background.

The detection limit (L_D) is an *a priori* estimate of the detection capability of a measurement system, and is also reported in units of counts.

The minimum detectable concentration (MDC) is the detection limit (counts) multiplied by an appropriate conversion factor to give units consistent with a site guideline, such as $\mu\text{Ci/g}$.

6.4.1 All equations used in this document are based on information provided in MARSSIM.

The two parameters of interest for a detector system with a background response greater than zero are:

- A. L_C the net response level, in counts, at which the detector output can be considered above background
- B. L_D the net response level, in counts, that can be expected to be seen with a detector with a fixed level of certainty

Assuming that a system has a background response and that random uncertainties and systematic uncertainties are accounted for separately, these parameters can be calculated using Poisson statistics.

For these calculations, two types of decision errors should be considered. A Type I error (or “false positive”) occurs when a detector response is considered to be above background when, in fact, only background radiation is present. A Type II error (or “false negative”) occurs when a detector response is considered to be background when in fact radiation is present at levels above background. The probability of a Type I error is referred to as α (alpha) and is associated with L_C ; the probability of a Type II error is referred to as β (beta) and is associated with L_D . Figure 4 graphically illustrates the relationship of these terms with respect to each other and to a normal background distribution.

6.4.2 If α and β are assumed to be equal, the variance (σ^2) of all measurement values is assumed to be equal to the values themselves. If the background of the detection system is not well known, then the critical detection level and the detection limit can be calculated by using the following formulae:

$$L_C = k \sqrt{2B} \tag{Eq. (6.1)}$$

$$L_D = k^2 + 2k \sqrt{2B} \tag{Eq. (6.2)}$$

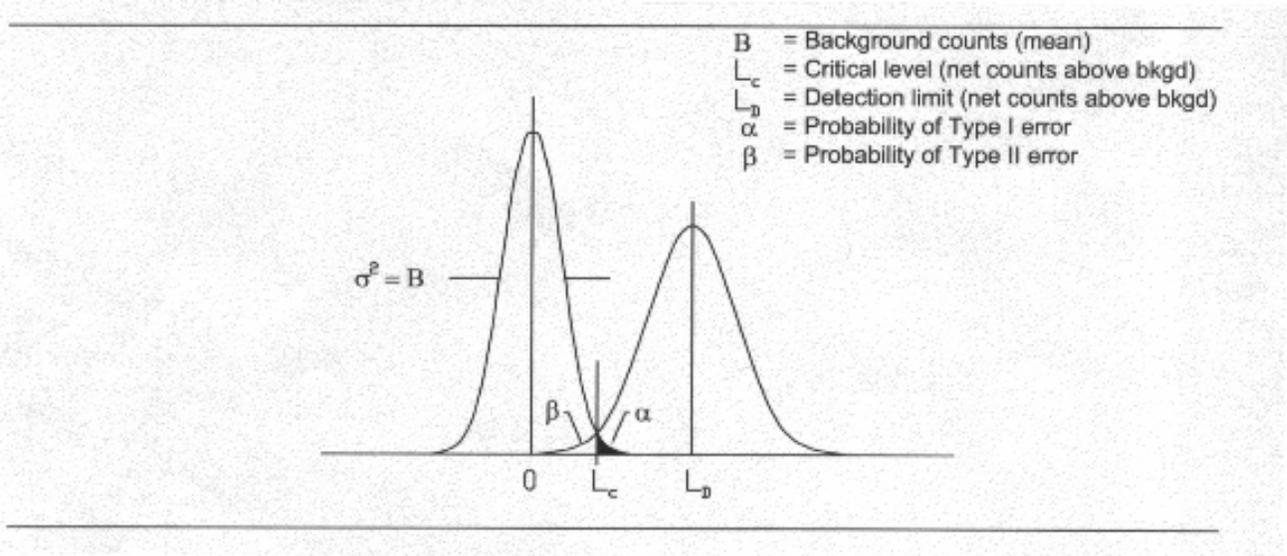
Where:

- L_C = critical level (counts)
- L_D = detection limit (counts)
- k = Poisson probability sum for α and β (assuming α and β are equal)
- B = number of background counts that are expected to occur while performing an actual measurement

The curve to the left in Figure 4 is the background distribution minus the mean of the background distribution. The result is a Poisson distribution with a mean equal to zero and a variance, σ^2 , equal to B. Note that the distribution accounts only for the expected statistical variation due to the stochastic nature of radioactive decay. “Paired blanks” are assumed when deriving the above stated relationships, which is interpreted to mean that the sample and background count times are the same.

Figure 4

Graphically Represented Probabilities for Type I and Type II Errors in Detection Sensitivity for Instrumentation with a Background Response



6.4.3 If values of 0.05 for both α and β are selected as acceptable, then $k = 1.645$ (from MARSSIM Appendix I, Table I.1) and Equations 6.1 and 6.2 can be written as:

$$L_C = 2.33 \sqrt{B} \tag{Eq. (6.3)}$$

$$L_D = 3 + 4.65 \sqrt{B} \tag{Eq. (6.4)}$$

NOTE: The constant factor of 3 in the L_D formula was stated as being 2.71, but since that time it has been shown and generally accepted that a constant factor of 3 is more appropriate. If the sample count times and background count times are different, a slightly different formulation is used.

6.4.4 For an integrated measurement over a preset time, the MDC can be obtained from Equation 6.4 by multiplying by the factor, C. This factor is used to convert from counts to concentration as shown below:

$$MDC = C \times (3 + 4.65 \sqrt{B}) \tag{Eq. (6.5)}$$

The total detection efficiency and other constants or factors represented by the variable C are usually not truly constants as shown in the equation above. It is likely that at least one of these factors will have a certain amount of variability associated with it which may or may not be significant. These varying factors are gathered together into the single constant, C, by which the net count result will be multiplied when converting the final data. If C varies significantly between measurements, then it might be best to select a value, C', from the observed distribution of C values that represents a conservative estimate. For example, a value of C might be selected to ensure that at least 95% of the possible values of C is less than the chosen value, C'.

The MDC calculated in this way helps assure that the survey results will meet the Data Quality Objectives. This approach for including uncertainties into the MDC calculation is recommended in both NUREG/CR-4007 and Appendix A to ANSI N13.30. Underestimating an MDC can have adverse consequences, especially if activity is later detected at a level below the stated MDC.

6.4.5 Summary of Direct Measurement Sensitivity Terms

The MDC is the *a priori* net activity level above the critical level that an instrument can be expected to detect 95% of the time. This value should be used when stating the detection capability of an instrument. The MDC is the detection limit, L_D , multiplied by an appropriate conversion factor to give units of activity. Again, this value is used before any measurements are made and is used to estimate the level of activity that can be detected using a given protocol.

The critical level, L_C , is the lower bound on the 95% detection interval defined for L_D and is the level at which there is a 5% chance of calling a background value "greater than background". This value should be used when actually counting samples or making direct radiation measurements. Any response above this level should be considered as above background (i.e., a net positive result). This will ensure 95% detection capability for L_D .

From a conservative point of view, it is better to overestimate the MDC for a measurement method. Therefore, when calculating MDC and L_C values, a measurement system background value should be selected that represents the high end of what is expected for a particular measurement method. For direct measurements, probes will be moved from point to point and, as a result, it is expected that the background will most likely vary significantly due to variations in background, source materials, and changes in geometry and shielding. Ideally, the MDC values should be calculated for each type of area, but it may be more economical to simply select a background value from the highest distribution expected and use this for all calculations. For the same reasons, realistic values of detection efficiencies and other process parameters should be used when possible and should be reflective of the actual conditions. To a great degree, the selection of these parameters will be based on judgment and will require evaluation of site-specific conditions.

MDC values for other counting conditions may be derived from Equation 6.5 depending on the detector and contaminants of concern. For example, it may be required to determine what level of contamination, distributed over 100 cm^2 , can be detected with a 500 cm^2 probe or what contamination level can be detected with any probe when the contamination area is smaller than the probe active area.

Table 4 lists several common field survey detectors with estimates of MDC values for ^{238}U on a smooth, flat plane. As such, these represent minimum MDC values and may not be applicable at all sites. Appropriate site-specific MDC values should be determined using the DQO Process.

Sample Calculations:

The following example illustrates the calculation of an MDC in Bq/m^2 for an instrument with a 15 cm^2 probe area when the measurement and background counting times are each one minute:

$$B = 40 \text{ counts}$$

$$C = (5 \text{ dpm/count})(\text{Bq}/60 \text{ dpm})(1/15 \text{ cm}^2 \text{ probe area}) \\ (10,000 \text{ cm}^2 / \text{m}^2) = 55.6 \text{ Bq} / \text{m}^2\text{-counts}$$

The MDC is calculated:

$$MDC = 55.6 \times (3 + 4.65 \sqrt{40}) = 1,800 \text{ Bq / m}^2 \text{ (1,080 dpm / 100 cm}^2\text{)}$$

The critical level, L_C , for this example is:

$$L_C = 2.33 \sqrt{B} = 15 \text{ counts}$$

Given the above scenario, if a person asked what level of contamination could be detected 95% of the time using this method, the answer would be 1,800 Bq / m² (1,080 dpm / 100 cm²). When actually performing measurements using this method, any count yielding greater than 55 total counts, or greater than 15 net counts (55-40 = 15) during a period of one minute, would be regarded as greater than background.

**Table 4 (MARSSIM Table 6.4)
 Examples of Estimated Detection Sensitivities
 for Alpha and Beta Survey Instrumentation
 (Static one minute counts for ²³⁸U calculated using
 Equations in MARSSIM Sections 6.4.7 and 6.4.8)**

Detector	Probe Area (cm ²)	Background (cpm)	Efficiency (cpm/dpm)	Approximately Sensitivity			
				L _C (counts)	L _D (counts)	MDC (dpm/100 cm ²) ^a	MDC (Bq/m ²) ^a
Alpha proportional	50	1	0.15	2	7	90	150
Alpha proportional	100	1	0.15	2	7	50	83
Alpha proportional	600	5	0.15	5	13	20	25
Alpha scintillation	50	1	0.15	2	7	90	150
Beta proportional	100	300	0.20	40	83	400	700
Beta proportional	600	1500	0.20	90	183	200	250
Beta GM pancake	15	40	0.20	15	32	1000	1800

^a Assumes that the size of the contamination area is at least as large as the probe area.

6.5 Scanning Sensitivity

The ability to identify a small area of elevated radioactivity during surface scanning is dependent upon the surveyor's skill in recognizing an increase in the audible or display output of an instrument. For notation purposes, the term scanning sensitivity is used throughout this section to describe the ability of a surveyor to detect a pre-determined level of contamination with a detector. The greater the sensitivity, the lower the level of contamination that can be detected.

Many of the radiological instruments and monitoring techniques typically used for occupational health physics activities may not provide the detection sensitivities necessary to demonstrate compliance with the DCGLs. The detection sensitivity for a given application can be improved (i.e., lower the MDC) by: 1) selecting an instrument with a higher detection efficiency or a lower background, 2) decreasing the scanning speed, or 3) increasing the size of the effective probe area without significantly increasing the background response.

Scanning is usually performed during radiological surveys in support of decommissioning to identify the presence of any areas of elevated activity. The probability of detecting residual contamination in the field depends not only on the sensitivity of the survey instrumentation when used in the scanning mode of operation, but is also affected by the surveyor's ability - i.e., human factors. The surveyor must make a decision whether the signals represent only the background activity, or residual contamination in excess of background. The greater the sensitivity, the lower the level of contamination that may be detected by scanning. Accounting for these human factors represents a significant change from the traditionally accepted methods of estimating scanning sensitivities.

An empirical method for evaluating the detection sensitivity for contamination surveys is by actual experimentation or, since it is certainly feasible, by simulating an experimental setup using computer software. The following steps provide a simple example of how one can perform this empirical evaluation:

- Step 1. A desired nuclide contamination level is selected.
- Step 2. The response of the detector to be used is determined for the selected nuclide contamination level.
- Step 3. A test source is constructed which will give a detector count rate equivalent to what was determined in Step 2. The count rate is equivalent to what would be expected from the detector when placed on an actual contamination area equal in value to that selected in Step 1.
- Step 4. The detector of choice is then moved over the source at different scan rates until an acceptable speed is determined.

The most useful aspect of this approach is that the source can then be used to show surveyors what level of contamination is expected to be targeted with the scan. They, in turn, can gain experience with what the expected response of the detector will be and how fast they can survey and still feel comfortable about detecting the target contamination level. The person responsible for the survey can then use this information when developing a fixed point measurement and sampling plan.

The remainder of this section provides information pertaining to the underlying processes involved when performing scanning surveys for alpha, beta, and gamma emitting radionuclides. The purpose is to provide relevant information that can be used for estimating realistic scanning sensitivities for survey activities.

6.5.1 Scanning for Alpha Emitters

Scanning for alpha emitters differs significantly from scanning for beta and gamma emitters in that the expected background response of most alpha detectors is very close to zero. The following discussion covers scanning for alpha emitters and assumes that the surface being surveyed is similar in nature to the material on which the detector was calibrated. In this respect, the approach is purely theoretical. Surveying surfaces that are dirty, non-planar, or weathered can significantly affect the detection efficiency and therefore bias the expected MDC for the scan. The use of reasonable detection efficiency values instead of optimistic values is highly recommended. MARSSIM Appendix J contains a complete derivation of the alpha scanning equations used here.

Since the time a contaminated area is under the probe varies and the background count rate of some alpha instruments is less than 1 cpm, it is not practical to determine a fixed MDC for scanning. Instead, it is more useful to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates.

For alpha survey instrumentation with backgrounds ranging from <1 to 3 cpm, a single count provides a surveyor sufficient cause to stop and investigate further. Assuming this to be true, the probability of detecting given levels of alpha surface contamination can be calculated by use of Poisson summation statistics.

Given a known scan rate and a surface contamination DCGL, the probability of detecting a single count while passing over the contaminated area is

$$P(n \geq 1) = 1 - e^{\left(-\frac{GE d}{60v}\right)} \quad \text{Eq. (6.6)}$$

where:

- P(n ≥ 1) = Probability of observing a single count
- G = Contamination activity (dpm)
- E = Detector efficiency (4π)
- d = Width of detector in direction of scan (cm)
- v = Scan speed (cm/s)

Once a count has been recorded and the guideline level of contamination is present, the surveyor should stop and wait until the probability of getting another count is at least 90%. This time interval can be calculated by:

$$t = 13, \frac{800}{CAE} \quad \text{Eq. (6.7)}$$

where:

- t = Time period for static count (s)
- C = Contamination guideline (dpm/100 cm²)
- A = Physical probe area (cm²)
- E = Detector efficiency (4π)

Many portable proportional counters have background count rates on the order of 5 to 10 cpm, and a single count should not cause a surveyor to investigate further. A counting period long enough to establish that a single count indicates an elevated contamination level would be prohibitively inefficient. For these types of instruments, the surveyor usually will need to get at least 2 counts while passing over the source area before stopping for further investigation.

Assuming this to be a valid assumption, the probability of getting two or more counts can be calculated by:

$$P(n \geq 2) = 1 - P(n = 0) - P(n = 1)$$

$$= 1 - \left[1 + \frac{(GE + B)t}{60} \right] \left[e^{-\frac{(GE + B)t}{60}} \right] \quad \text{Eq. (6.8)}$$

where:

- P(n ≥ 2) = probability of getting 2 or more counts during the time interval t
- P(n = 0) = probability of not getting any counts during the time interval t
- P(n = 1) = probability of getting 1 count during the time interval t
- B = background count rate (cpm)
- G = contamination activity (dpm)
- E = detector efficiency (4π)

MARSSIM Appendix J provides a complete derivation of the above equations and a detailed discussion of the probability of detecting alpha surface contamination for several different variables. Several probability charts are included at the end of MARSSIM Appendix J for common detector sizes. Table 5 provides estimates of the probability of detecting 300 dpm / 100 cm² for some commonly used alpha detectors.

**Table 5 (MARSSIM Table 6.8)
 Probability of Detecting 300 dpm/100 cm² of Alpha Activity While Scanning with
 Alpha Detectors Using an Audible Output (calculated using MARSSIM Equation 6.9)**

Detector Type	Detector Efficiency cpm/dpm	Probe Dimension in Direction of Scan (cm)	Scan Rate (cm/s)	Probability of detecting 300 dpm/100 cm ²
Proportional	0.20	5	3	80%
Proportional	0.15	15	5	90%
Scintillation	0.15	5	3	70%
Scintillation	0.15	10	3	90%

6.5.2 Scanning for Beta and Gamma Emitters

The minimum detectable concentration of a scan survey (scan MDC) depends on the intrinsic characteristics of the detector (efficiency, physical probe area, etc.), the nature (type and energy of emissions) and relative distribution of the potential contamination (point versus distributed source and depth of contamination), scan rate, and other characteristics of the surveyor.

Some factors that may affect the surveyor's performance include the costs associated with various outcomes, e.g., fatigue, noise, level of training, experience - and the surveyor's a priori expectation of the likelihood of contamination present.

For example, if the surveyor believes that the potential for contamination is very low, as in a Class 3 area, a relatively large signal may be required for the surveyor to conclude that contamination is present. NUREG/CR-6364 provides a complete discussion of the human factors as they relate to the performance of scan surveys.

A. Signal Detection Theory

Personnel conducting radiological surveys for residual contamination at decommissioning sites must interpret the audible output of a portable survey instrument to determine when the signal ("clicks") exceeds the background level by a margin sufficient to conclude that contamination is present. It is difficult to detect low levels of contamination because both the signal and the background vary widely. Signal detection theory provides a framework for the task of deciding whether the audible output of the survey meter during scanning is due to background or signal plus background levels. An index of sensitivity (d') that represents the distance between the means of the background and background plus signal (refer to Figure 4 for determining LD), in units of their common standard deviation, can be calculated for various decision errors (correct detection and false positive rate). As an example, for a correct detection rate of 95% (complement of a false negative rate of 5%) and a false positive rate of 5%, d' is 3.28 (similar to the static MDC for the same decision error rates). The index of sensitivity is independent of human factors, and therefore, the ability of an ideal observer

(theoretical construct), may be used to determine the minimum d' that can be achieved for particular decision errors. The ideal observer makes optimal use of the available information to maximize the percent correct responses, providing an effective upper bound against which to compare actual surveyors. Table 6 lists selected values of d' .

Table 6 (MARSSIM Table 6.5)
Values of d' for Selected True Positive and False Positive Proportions

False Positive Proportion	True Positive Proportion							
	0.60	0.65	0.70	0.75	0.80	0.85	0.90	0.95
0.05	1.90	2.02	2.16	2.32	2.48	2.68	2.92	3.28
0.10	1.54	1.66	1.80	1.96	2.12	2.32	2.56	2.92
0.15	1.30	1.42	1.56	1.72	1.88	2.08	2.32	2.68
0.20	1.10	1.22	1.36	1.52	1.68	1.88	2.12	2.48
0.25	0.93	1.06	1.20	1.35	1.52	1.72	1.96	2.32
0.30	0.78	0.91	1.05	1.20	1.36	1.56	1.80	2.16
0.35	0.64	0.77	0.91	1.06	1.22	1.42	1.66	2.02
0.40	0.51	0.64	0.78	0.93	1.10	1.30	1.54	1.90
0.45	0.38	0.54	0.66	0.80	0.97	1.17	1.41	1.77
0.50	0.26	0.38	0.52	0.68	0.84	1.04	1.28	1.64
0.55	0.12	0.26	0.40	0.54	0.71	0.91	1.15	1.51
0.60	0.00	0.13	0.27	0.42	0.58	0.82	1.02	1.38

B. Two Stages of Scanning

The framework for determining the scan MDC is based on the premise that there are two stages of scanning. That is, surveyors do not make decisions on the basis of a single indication, rather, upon noting an increased number of counts, they pause briefly and then decide whether to move on or take further measurements. Thus, scanning consists of two components: continuous monitoring and stationary sampling. In the first component, characterized by continuous movement of the probe, the surveyor has only a brief "look" at potential sources, determined by the scan speed. The surveyor's willingness to decide that a signal is present at this stage is likely to be liberal, in that the surveyor should respond positively on scant evidence, since the only "cost" of a false positive is a little time. The second component occurs only after a positive response was made at the first stage. This response is marked by the surveyor interrupting his scanning and holding the probe stationary for a period of time, while comparing the instrument output signal during that time to the background counting rate. Owing to the longer observation interval, sensitivity is relatively high. For this decision, the criterion should be stricter, since the cost of a "yes" decision is to spend considerably more time taking a static measurement or a sample.

Since scanning can be divided into two stages, it is necessary to consider the survey's scan sensitivity for each of the stages. Typically, the minimum detectable count rate (MDCR) associated with the first scanning stage will be greater due to the brief observation intervals of continuous monitoring provided that the length of the pause during the second stage is significantly longer. Typically, observation intervals during the first stage are on the order of 1 or 2 seconds, while the second stage pause may be several seconds long. The greater value of MDCR from each of the scan stages is used to determine the scan sensitivity for the surveyor.

C. Determination of MDCR and Use of Surveyor Efficiency

The minimum detectable number of net source counts in the interval is given by s_i . Therefore, for an ideal observer, the number of source counts required for a specified level of performance can be arrived at by multiplying the square root of the number of background counts by the detectability value associated with the desired performance (as reflected in d') as shown in the following equation:

$$s_i = d' \sqrt{b_i} \tag{Eq. (6.9)}$$

where the value of d' is selected from Table 6 based on the required true positive and false positive rates and b_i is the number of background counts in the interval.

For example, suppose that one chooses to estimate the minimum count rate that is detectable by scanning in an area with a background of 1,500 cpm. Note that the minimum detectable count rate must be considered for both scan stages - and the more conservative value is selected as the minimum count rate that is detectable. It will be assumed that a typical source remains under the probe for 1 second during the first stage, therefore, the average number of background counts in the observation interval is 25 ($b_i = 1500 \times (1/60)$).

Furthermore, it can be assumed that at the first scanning stage a high rate (e.g., 95%) of correct detections is required, and that a correspondingly high rate of false positives (e.g., 60%) will be tolerated. From Table 6, the value of d' , representing this performance goal, is 1.38. The net source counts needed to support the specified level of performance (assuming an ideal observer) will be estimated by multiplying 5 (the square root of 25) by 1.38. Thus, the net source counts per interval, s_i , needed to yield better than 95% detections with about 60% false positives is 6.9.

The minimum detectable source count rate, in cpm, may be calculated by:

$$\text{MDCR} = s_i \times (60 / i) \tag{Eq. (6.10)}$$

For this example, MDCR is equivalent to 414 cpm (1,914 cpm gross).

Table 7 provides the scan sensitivity for the ideal observer (MDCR) at the first scanning stage for various background levels, based on an index of sensitivity (d') of 1.38 and a 2-second observation interval.

**Table 7 (MARSSIM Table 6.6)
 Scanning Sensitivity (MDCR) of the Ideal Observer for Various Background Levels***

Background (cpm)	MDCR (net cpm)	Scan Sensitivity (gross cpm)
45	50	95
60	60	120
260	120	380
300	130	430
350	140	490
400	150	550
1000	240	1240
3000	410	3410
4000	480	4480

* The sensitivity of the ideal observer during the first scanning stage is based on an index of sensitivity (d') of 1.38 and a 2-second observation interval.

The minimum number of source counts required to support a given level of performance for the final detection decision (second scan stage) can be estimated using the same method. The performance goal at this stage will be more demanding. The required rate of true positives remains high (e.g., 95%), but fewer false positives (e.g., 20%) can be tolerated, such that d' (from Table 6) is now 2.48. One will assume that the surveyor typically stops the probe over a suspect location for about 4 seconds before making a decision, so that the average number of background counts in an observation interval is 100 ($b_i = 1,500 \times (4/60)$). Therefore, if the minimum detectable number of net source counts, s_i , needed will be estimated by multiplying 10 (the square root of 100) by 2.48 (the d' value); so s_i equals 24.8. The MDCR is calculated by $2.48 \times (60/4)$ and equals 37.2 cpm. The value associated with the first scanning stage (this example, 414 cpm) will typically be greater, owing to the relatively brief intervals assumed.

Laboratory studies using simulated sources and backgrounds were performed to assess the abilities of surveyors under controlled conditions. The methodology and analysis of results for these studies are described in NUREG/CR-6364 and NUREG-1507. The surveyor's actual performance as compared with that which is ideally possible (using the ideal observer construct) provided an indication of the efficiency of the surveyors. Based on the results of the confidence rating experiment, this surveyor efficiency (p) was estimated to be between 0.5 and 0.75. MARSSIM recommends assuming an efficiency value at the lower end of the observed range (i.e., 0.5) when making MDC estimates. Thus, the required number of net source counts for the surveyor, $MDCR_{\text{surveyor}}$, is determined by dividing the MDCR by the square root of p . Continuing with this example, the surveyor MDCR is calculated by $414 \text{ cpm}/0.707$, or 585 cpm (2,085 cpm gross).

D. Scan MDCs for Structure Surfaces and Land Areas

The survey design for determining the number of data points for areas of elevated activity (see MARSSIM Section 5.5.2.4) depends on the scan MDC for the selected instrumentation.

In general, alpha or beta scans are performed on structure surfaces to satisfy the elevated activity measurements survey design, while gamma scans are performed for land areas. Because of low background levels for alpha emitters, the approach described here is not generally applied to determining scan MDCs for alpha contaminants, refer to Section 6.5.1 for an appropriate method for determining alpha scan MDCs for building surfaces. In any case, the data requirements for assessing potential elevated areas of direct radiation depend on the scan MDC of the survey instrument (e.g., floor monitor, GM detector, NaI scintillation detector).

1. Scan MDCs for Building/Structure Surfaces

The scan MDC is determined from the minimum detectable count rate (MDCR) by applying conversion factors that account for detector and surface characteristics and surveyor efficiency. As discussed above, the MDCR accounts for the background level, performance criteria (d'), and observation interval. The observation interval during scanning is the actual time that the detector can respond to the contamination source—this interval depends on the scan speed, detector size in the direction of the scan, and area of elevated activity. Because the actual dimensions of potential areas of elevated activity in the field cannot be known a priori, MARSSIM recommends postulating a certain area (e.g., perhaps 50 to 200 cm²), and then selecting a scan rate that provides a reasonable observation interval.

Finally, the scan MDC for structure surfaces may be calculated:

$$\text{Scan MDC} = \frac{\text{MDCR}}{\sqrt{p} \epsilon_i \epsilon_s \frac{\text{probe area}}{100 \text{ cm}^2}} \quad \text{Eq. (6.11)}$$

where:

- MDCR = minimum detectable count rate
- ϵ_i = instrument efficiency
- ϵ_s = surface efficiency
- p = surveyor efficiency

As an example, the scan MDC (in dpm/100 cm²) for ⁹⁹Tc on a concrete surface may be determined for a background level of 300 cpm and a 2-second observation interval using a hand-held gas proportional detector (126 cm² probe area). For a specified level of performance at the first scanning stage of 95% true positive rate and 60% false positive rate (and assuming the second stage pause is sufficiently long to ensure that the first stage is more limiting), d' equals 1.38 (Table 6) and the MDCR is 130 cpm (Table 7). Using a surveyor efficiency of 0.5, and assuming instrument and surface efficiencies of 0.36 and 0.54, respectively, the scan MDC is calculated using the last equation:

$$\text{Scan MDC} = \frac{130}{\sqrt{0.5 (0.36)(0.54)(1.26)}} = 750 \frac{\text{dpm}}{100 \text{ cm}^2}$$

Eq. (6.12)

Additional examples for calculating the scan MDC may be found in NUREG-1507.

2. Scan MDCs for Land Areas

In addition to the MDCR and detector characteristics, the scan MDC (in pCi/g) for land areas is based on the area of elevated activity, depth of contamination, and the radionuclide (i.e., energy and yield of gamma emissions). If one assumes constant parameters for each of the above variables, with the exception of the specific radionuclide in question, the scan MDC may be reduced to a function of the radionuclide alone. NaI scintillation detectors are generally used for scanning land areas.

An overview of the approach used to determine scan MDCs for land areas follows. The NaI(Tl) scintillation detector background level and scan rate (observation interval) are postulated, and the MDCR for the ideal observer, for a given level of performance, is obtained. After a surveyor efficiency is selected, the relationship between the surveyor MDCR ($\text{MDCR}_{\text{surveyor}}$) and the radionuclide concentration in soil (in Bq / kg or pCi / g) is determined. This correlation requires two steps -first, the relationship between the detector's net count rate to net exposure rate (cpm per $\mu\text{R} / \text{h}$) is established, and second, the relationship between the radionuclide contamination and exposure rate is determined.

For a particular gamma energy, the relationship of NaI(Tl) scintillation detector count rate and exposure rate may be determined analytically (in cpm per $\mu\text{R}/\text{h}$). The approach used to determine the gamma fluence rate necessary to yield a fixed exposure rate (1 $\mu\text{R}/\text{h}$) as a function of gamma energy is provided in NUREG-1507. The NaI(Tl) scintillation detector response (cpm) is related to the fluence rate at specific energies, considering the detector's efficiency (probability of interaction) at each energy. From this, the NaI(Tl)scintillation detector versus exposure rates for varying gamma energies are determined. Once the relationship between the NaI(Tl) scintillation detector response (cpm) and the exposure rate is established, the $\text{MDCR}_{\text{surveyor}}$ (in cpm) of the NaI(Tl) scintillation detector can be related to the minimum detectable net exposure rate. The minimum detectable exposure rate is used to determine the minimum detectable radionuclide concentration (i.e., the scan MDC) by modeling a specified small area of elevated activity.

In MARSSIM, modeling (using Microshield™) of the small area of elevated activity (soil concentration) is used to determine the net exposure rate produced by a radionuclide concentration at a distance 10 cm above the source. This position is selected because it relates to the average height of the NaI(Tl)scintillation detector above the ground during scanning.

The factors considered in the modeling include:

- radionuclide of interest (considering all gamma emitters for decay chains)
- expected concentration of the radionuclide of interest
- areal dimensions of the area of elevated activity
- depth of the area of elevated activity
- location of dose point (NaI(Tl) scintillation detector height above the surface)
- density of soil

Modeling analyses are conducted by selecting a radionuclide (or radioactive material decay series) and then varying the concentration of the contamination. The other factors are held constant - the areal dimension of a cylindrical area of elevated activity is 0.25 m² (radius of 282cm), the depth of the area of elevated activity is 15 cm, the dose point is 10 cm above the surface, and the density of soil is 1.6 g/cm³. The objective is to determine the radionuclide concentration that is correlated to the minimum detectable net exposure rate.

As an example, the scan MDC for ¹³⁷Cs using a 1.5 in. by 1.25 in. NaI(Tl) scintillation detector is considered in detail. Assume that the background level is 4,000 cpm and that the desired level of performance, 95% correct detections and 60% false positive rate, results in a d' of 1.38. The scan rate of 0.5 m/s provides an observation interval of 1-second (based on a diameter of about 56 cm for the area of elevated activity). The MDCR_{surveyor} may be calculated assuming a surveyor efficiency (p) of 0.5 as follows:

$$b_i (4,000 \text{ cpm}) \times (1 \text{ sec}) \times (1 \text{ min}/60 \text{ sec}) = 66.7 \text{ counts}$$

$$\text{MDCR} (1.38) \times (\sqrt{66.7}) \times (60 \text{ sec}/1 \text{ min}) = 680 \text{ cpm}$$

$$\text{MDCR}_{\text{surveyor}} = 680 / \sqrt{0.5} = 960 \text{ cpm}$$

The corresponding minimum detectable exposure rate is determined for this detector and radionuclide. The manufacturer of this particular 1.5 in. by 1.25 in. NaI(Tl) scintillation detector quotes a count rate to exposure rate ratio for ¹³⁷Cs of 350 cpm per μR/h. The minimum detectable exposure rate is calculated by dividing the count rate (960 cpm) by the count rate to exposure rate ratio for the radionuclide of interest (350 cpm per μR/h). The minimum detectable exposure rate for this example is 2.73 μR/h.

Both ¹³⁷Cs and its short-lived progeny, ^{137m}Ba, were chosen from the Microshield™ library, as presented in MARSSIM. The source activity and other modeling parameters were entered into the modeling code. The source activity was selected based on an arbitrary concentration of 5 pCi/g. The modeling code performed the appropriate calculations and determined an exposure rate of 1.307 μR/h (which accounts for buildup). Finally, the radionuclide concentrations of ¹³⁷Cs and ^{137m}Ba (scan MDC) necessary to yield the minimum detectable exposure rate (2.73 μR/h) may be calculated using the following formula.

$$\text{Scan MDC} = \frac{\left(\frac{5 \text{ pCi}}{\text{g}}\right) \left(\frac{2.73 \text{ } \mu\text{R}}{\text{h}}\right)}{\frac{1.307 \text{ } \mu\text{R}}{\text{h}}}$$

$$= 10.4 \text{ pCi-g}$$

It must be emphasized that while a single scan MDC value can be calculated for a given radionuclide, other scan MDC values may be equally justifiable depending on the values chosen for the various factors, including the MDCR (background level, acceptable performance criteria, observation interval), surveyor efficiency, detector parameters and the modeling conditions of the contamination. It should also be noted that determination of the scan MDC for radioactive materials, like uranium and thorium, must consider the gamma radiation emitted from the entire decay series.

Table 8 provides scan MDCs for common radionuclides and radioactive materials in soil. It is important to note that the variables used in the above examples to determine the scan MDCs for the 1.25 in. by 1.5 in. NaI(Tl) scintillation detector, i.e., the $\text{MDCR}_{\text{surveyor}}$ detector parameters (e.g., cpm per $\mu\text{R/h}$), and the characteristics of the area of elevated activity, have all been held constant to facilitate the calculation of scan MDCs provided in Table 8. The benefit of this approach is that generally applicable scan MDCs are provided for different radioactive contaminants. Additionally, the relative detectability of different contaminants is evident because the only variable in Table 8 is the nature of the contaminant.

As noted above, the scan MDCs calculated using the approach in this section are dependent on several factors. One way to validate the appropriateness of the scan MDC is by tracking the residual radioactivity (both surface activity and soil concentrations) levels identified during investigations performed as a result of scanning surveys. The measurements performed during these investigations may provide an a posteriori estimate of the scan MDC that can be used to validate the a priori scan MDC used to design the survey.

**Table 8 (MARSSIM Table 6.7)
 NaI(Tl) Scintillation Detector Scan MDCs for Common Radiological Contaminants***

Radionuclide/ Radioactive Material	1.25 in. by 1.5 in. NaI Detector		2 in. by 2 in. NaI Detector	
	Scan MDC (Bq/kg)	Weighted cpm/ $\mu\text{R/h}$	Scan MDC (Bq/kg)	Weighted cpm/ $\mu\text{R/h}$
Am-241	1,650	5,830	1,170	13,000
Co-60	215	160	126	430
Cs-137	385	350	237	900
Th-230	111,000	4,300	78,400	9,580
Ra-226 (in equilibrium with progeny)	167	300	104	760

Radionuclide/ Radioactive Material	1.25 in. by 1.5 in. NaI Detector		2 in. by 2 in. NaI Detector	
	Scan MDC (Bq/kg)	Weighted cpm/μR/h	Scan MDC (Bq/kg)	Weighted cpm/μR/h
Th-232 decay series (Sum of all radionuclides in the thorium decay series)	1,050	340	677	830
Th-232 (In equilibrium with progeny in decay series)	104	340	66.6	830
Depleted Uranium** (0.34% U-235)	2,980	1,680	2,070	3,790
Natural Uranium**	4,260	1,770	2,960	3,990
3% Enriched Uranium**	5,070	2,010	3,540	4,520
20% Enriched Uranium**	5,620	2,210	3,960	4,940
50% Enriched Uranium**	6,220	2,240	4,370	5,010
75% Enriched Uranium**	6,960	2,250	4,880	5,030

* Refer to text for complete explanation of factors used to calculate scan MDCs. For example, the background level for the 1.25 in. by 1.5 in. NaI detector was assumed to be 4,000 cpm, and 10,000 cpm for the 2 in. by 2 in. NaI detector. The observation interval was 1-sec and the level of performance was selected to yield d' of 1.38.

** Scan MDC for uranium includes sum of 238U, 235U, and 234U.

6.6 Applications - Instrument Selection

This section describes the primary applications of instrumentation to field measurements and laboratory analyses for radiological surveys.

Radiation survey parameters that might be needed for site release purposes include surface activities, exposure rates, and radionuclide concentrations in soil. To determine these parameters, field measurements and laboratory analyses may be necessary. For certain radionuclides or radionuclide mixtures, both alpha and beta radiations may have to be measured. In addition to assessing average radiological conditions, the survey objectives should address identifying small areas of elevated activity and determining the extent and level of residual radioactivity.

Additionally, the potential uses of radiation instruments can vary significantly depending on the specific design and operating criteria of a given detector type. For example, a NaI(Tl) scintillator can be designed to be very thin with a low atomic number entrance window (e.g., beryllium) such that the effective detection capability for low energy photons is optimized. Conversely, the same scintillant material can be fabricated as a thick cylinder in order to optimize the detection probability for higher energy photons. On the recording end of a detection system, the output could be a ratemeter, scaler, or multichannel analyzer as described in Section 6.2 Operator variables such as training and level of experience with specific instruments should also be considered.

With so many variables, it is highly unlikely that any single instrument (detector and readout combination) will be capable of adequately measuring all of the radiological parameters necessary to demonstrate that criteria for release have been satisfied. It is usually necessary to select multiple instruments to perform the variety of measurements required.

Selection of instruments will require an evaluation of a number of situations and conditions. Instruments must be stable and reliable under the environmental and physical conditions where they will be used, and their physical characteristics (size and weight) should be compatible with the intended application. The instrument must be able to detect the type of radiation of interest, and the measurement system should be capable of measuring levels that are less than the DCGL for free release of structures and soil (see MARSSIM Section 6.7) and other release guideline values.

For gamma radiation scanning, a scintillation detector/ratemeter combination is the usual instrument of choice. A large-area proportional detector with a ratemeter is recommended for scanning for alpha and beta radiations where surface conditions and locations permit; otherwise, an alpha scintillation or thin-window GM detector (for beta surveys) may be used. For direct gamma measurements, a pressurized ionization chamber or in situ gamma spectroscopy system is recommended. As an option, a NaI(Tl) scintillation detector may be used if cross-calibrated to a pressurized ion chamber or calibrated for the specific energy of interest. The same alpha and beta detectors identified above for scanning surveys are also recommended for use in direct measurements.

There are certain radionuclides that, because of the types, energies, and abundances of their radiations, will be essentially impossible to measure at the guideline levels, under field conditions, using state-of-the-art instrumentation and techniques. Examples of such radionuclides include very low energy pure beta emitters, such as ^3H and ^{63}Ni , and low energy photon emitters, such as ^{55}Fe and ^{125}I . Pure alpha emitters dispersed in soil or covered with some absorbing layer will not be detectable because the alpha radiation will not penetrate through the media or covering to reach the detector. A common example of such a condition would be ^{230}Th surface contamination covered by paint, dust, oil, or moisture. In such circumstances, sampling and laboratory analysis would be required to measure the residual activity levels unless surrogate radionuclides are present as discussed in MARSSIM Section 4.3.2.

The number of possible design and operating schemes for each of the different types of detectors is too large to discuss in detail within the context of this document. For a general overview, lists of common radiation detectors along with their usual applications during surveys are provided in Tables 1 through 3. MARSSIM Appendix H contains specific information for various types of field survey and laboratory analysis equipment currently in use. Continual development of new technologies will result in changes to these listings.

6.7 Instrument Calibration

Calibration refers to the determination and adjustment of the instrument response in a particular radiation field of known intensity. Proper calibration procedures are an essential requisite toward providing confidence in measurements made to demonstrate compliance with cleanup criteria. Certain factors, such as energy dependence and environmental conditions, require consideration in the calibration process, depending on the conditions of use of the instrument in the field. Routine calibration of radiation detection instruments refers to calibration for normal use under typical field conditions. Considerations for the use and calibration of instruments include:

- use of the instrument for radiation of the type for which the instrument is designed
- use of the instrument for radiation energies within the range of energies for which the instrument is designed
- use under environmental conditions for which the instrument is designed
- use under influencing factors, such as magnetic and electrostatic fields, for which the instrument is designed
- use of the instrument in an orientation such that geotropic effects are not a concern
- use of the instrument in a manner that will not subject the instrument to mechanical or thermal stress beyond that for which it is designed.

Routine calibration commonly involves the use of one or more sources of a specific radiation type and energy, and of sufficient activity to provide adequate field intensities for calibration on all ranges of concern.

Actual field conditions under which the radiation detection instrument will be used may differ significantly from those present during routine calibration. Factors which may affect calibration validity include:

- the energies of radioactive sources used for routine calibration may differ significantly from those of radionuclides in the field
- the source-detector geometry (e.g., point source or large area distributed source) used for routine calibration may be different than that found in the field
- the source-to-detector distance typically used for routine calibration may not always be achievable in the field
- the condition and composition of the surface being monitored (e.g., sealed concrete, scabbled concrete, carbon steel, stainless steel, and wood) and the presence of overlaying material (e.g., water, dust, oil, paint) may result in a decreased instrument response relative to that observed during routine calibration

If the actual field conditions differ significantly from the calibration assumptions, a special calibration for specific field conditions may be required. Such an extensive calibration need only be done once to determine the effects of the range of field conditions that may be encountered at the site. If responses under routine calibration conditions and proposed use conditions are significantly different, a correction factor or chart should be supplied with the instrument for use under the proposed conditions.

As a minimum, each measurement system (detector/readout combination) should be calibrated annually and response checked with a source following calibration. Instruments may require more frequent calibration if recommended by the manufacturer. Re-calibration of field instruments is also required if an instrument fails a performance check or if it has undergone repair or any modification that could affect its response.

The user may decide to perform calibrations following industry recognized procedures, or the user can choose to obtain calibration by an outside service, such as a major instrument manufacturer or a health physics services organization. According to the DOE EIGRSP, each instrument should be calibrated annually and response-checked with a source following calibration. Recalibration of field instruments is also required following maintenance that could affect the validity of the a priori calibration. The calibration interval may be longer if the manufacturer can document that the extended frequency adequately ensures the validity of data obtained with the equipment.

Calibration sources should be traceable to the National Institute of Standards and Technology (NIST). Where NIST traceable standards are not available, standards obtained from an industry recognized organization (e.g., the New Brunswick Laboratory for various uranium standards) may be used. The Radiological Controls Department may decide to perform calibrations, following DOE-mandated standards (e.g., ANSI N323A, etc.), or may choose to obtain calibration by an outside service, such as a major instrument manufacturer or one of the health physics services organizations, depending on the instrument.

Calibration of instruments for measurement of surface contamination should be performed such that a direct instrument response can be accurately converted to the 4π (total) emission rate from the source. An accurate determination of activity from a measurement of count rate above a surface in most cases is an extremely complex task because of the need to determine appropriate characteristics of the source including decay scheme, geometry, energy, scatter, and self-absorption. For the purpose of release of contaminated areas from radiological control, measurements must provide sufficient accuracy to ensure that cleanup standards have been achieved. Inaccuracies in measurements should be controlled in a manner that minimizes the consequences of decision errors. The variables that affect instrument response should be

understood well enough to ensure that the consequences of decision errors are minimized. Therefore, the calibration should account for the following factors (where necessary):

- Calibrations for point and large area source geometries may differ, and both may be necessary if areas of activity smaller than the probe area and regions of activity larger than the probe area are present.
- Calibration should either be performed with the radionuclide of concern, or with appropriate correction factors developed for the radionuclide(s) present based on calibrations with nuclides emitting radiations similar to the radionuclide of concern.
- For portable instrumentation, calibrations should account for the substrate of concern (i.e., concrete, steel) or appropriate correction factors developed for the substrates relative to the actual calibration standard substrate. This is especially important for beta emitters because backscatter is significant and varies with the composition of the substrate. Conversion factors developed during the calibration process should be for the same counting geometry to be used during the actual use of the detector.

For cleanup standards for building surfaces, the contamination level is typically expressed in terms of the particle emission rate per unit time per unit area, normally Bq/m² or disintegrations per minute (dpm) per 100 cm². In many facilities, surface contamination is assessed by converting the instrument response (in counts per minute) to surface activity using one overall total efficiency. The total efficiency may be considered to represent the product of two factors, the instrument (detector) efficiency, and the source efficiency. Use of the total efficiency is not a problem provided that the calibration source exhibits characteristics similar to the surface contamination (i.e., radiation energy, backscatter effects, source geometry, self-absorption). In practice, this is hardly the case; more likely, instrument efficiencies are determined with a clean, stainless steel source, and then those efficiencies are used to determine the level of contamination on a dust-covered concrete surface. By separating the efficiency into two components, the surveyor has a greater ability to consider the actual characteristics of the surface contamination.

The instrument efficiency is defined as the ratio of the net count rate of the instrument and the surface emission rate of a source for a specified geometry. The surface emission rate is defined as the number of particles of a given type above a given energy emerging from the front face of the source per unit time. The surface emission rate is the 2π particle fluence that embodies both the absorption and scattering processes that effect the radiation emitted from the source. Thus, the instrument efficiency is determined by the ratio of the net count rate and the surface emission rate.

The instrument efficiency is determined during calibration by obtaining a static count with the detector over a calibration source that has a traceable activity or surface emission rate. In many cases, a source emission rate is measured by the manufacturer and certified as NIST traceable. The source activity is then calculated from the surface emission rate based on assumed backscatter and self-absorption properties of the source. The maximum value of instrument efficiency is 1.

The source efficiency is defined as the ratio of the number of particles of a given type emerging from the front face of a source and the number of particles of the same type created or released within the source per unit time. The source efficiency takes into account the increased particle emission due to backscatter effects, as well as the decreased particle emission due to self-absorption losses. For an ideal source (i.e., no backscatter or self-absorption), the value of the source efficiency is 0.5. Many real sources will exhibit values less than 0.5, although values greater than 0.5 are possible, depending on the relative importance of the absorption and backscatter processes.

Source efficiencies may be determined experimentally. Alternatively, ISO-7503-1 makes recommendations for default source efficiencies. A source efficiency of 0.5 is recommended for beta emitters with maximum energies above 0.4 MeV. Alpha emitters and beta emitters with maximum beta energies between 0.15 and 0.4 MeV have a recommended source efficiency of 0.25. Source efficiencies for some common materials are provided in NUREG-1507.

Instrument efficiency may be affected by detector-related factors such as detector size (probe surface area), window density thickness, geotropism, instrument response time, counting time (in static mode), scan rate (in scan mode), and ambient conditions such as temperature, pressure, and humidity. Instrument efficiency also depends on solid angle effects, which include source-to-detector distance and source geometry.

Source efficiency may be affected by source-related factors such as the type of radiation and its energy, source uniformity, surface roughness and coverings, and surface composition (e.g., wood, metal, concrete).

The calibration of gamma detectors for the measurement of photon radiation fields should also provide reasonable assurance of acceptable accuracy in field measurements. Use of these instruments for demonstration of compliance with cleanup standards is complicated by the fact that most cleanup levels produce exposure rates of at most a few $\mu\text{R/h}$.

Several of the portable survey instruments currently available for exposure rate measurements of $\sim 1 \mu\text{R/h}$ (often referred to as micro-R meters) have full scale intensities of ~ 3 to $5 \mu\text{R/h}$ on the first range. This is below the ambient background for most low radiation areas and most calibration laboratories. (A typical background dose equivalent rate of 100 mrem/y gives a background exposure rate of about $10 \mu\text{R/h}$.) Even on the second range, the ambient background in the calibration laboratory is normally a significant part of the range and must be taken into consideration during calibration. The instruments commonly are not energy-compensated and are very sensitive to the scattered radiation that may be produced by the walls and floor of the room or additional shielding required to lower the ambient background.

Low intensity sources and large distances between the source and detector can be used for low-level calibrations if the appropriate precautions are taken. Field characterization of low-level sources with traceable transfer standards is difficult because of the poor signal-to-noise ratio in the standard chamber. In order to achieve adequate ionization current, the distance between the standard chamber and the source generally will be as small as possible while still maintaining good geometry (5 to 7 detector diameters). Generally it is not possible to use a standard ionization chamber to characterize the field at the distance necessary to reduce the field to the level required for calibration. A high quality GM detector, calibrated as a transfer standard, may be useful at low levels.

Corrections for scatter can be made using a shadow-shield technique in which a shield of sufficient density and thickness to eliminate virtually all the primary radiation is placed about midway between the source and the detector. The dimensions of the shield should be the minimum required to reduce the primary radiation intensity at the detector location to less than 2% of its unshielded value. The change in reading caused by the shield being removed is attributed to the primary field from the source at the detector position.

In some instruments that produce pulses (GM counters or scintillation counters), the detector can be separated electronically from the readout electronics and the detector output can be simulated with a suitable pulser. Caution must be exercised to ensure that either the high voltage is properly blocked or that the pulser is designed for this application. If this can be accomplished, the instrument can first be calibrated on a higher range that is not affected by the ambient background and in a geometry where scatter is not a problem and, after disconnecting the detector, to provide the pulse-rate from the pulser which will give the same instrument response. The pulse rate can then be related to field strength and reduced to give readings on lower ranges (with the detector disconnected) even below the ambient background. This technique does not take account of any inherent detector background independent of the external background.

Ionization chambers are commonly used to measure radiation fields at very low levels. In order to obtain the sensitivity necessary to measure these radiation levels, the instruments are frequently very large and often pressurized. These instruments have the same calibration problems as the more portable micro-R meters described above. The same precautions (shadow shield) must be taken to separate the response of the instrument to the source and to scattered radiation. Generally, it is not possible to substitute an electronic pulser for the radiation field in these instruments.

For energy-dependent gamma scintillation instruments, such as NaI(Tl) detectors, calibration for the gamma energy spectrum at a specific site may be accomplished by comparing the instrument response to that of a pressurized ionization chamber, or equivalent detector, at different locations on the site. Multiple radionuclides with various photon energies may also be used to calibrate the system for the specific energy of interest.

In the interval between calibrations, the instrument should receive a performance check prior to use. In some cases, a performance check following use may also provide valuable information. This calibration check is merely intended to establish whether or not the instrument is operating within certain specified, rather large, uncertainty limits. The initial performance check should be conducted following the calibration by placing the source in a fixed, reproducible location and recording the instrument reading. The source should be identified along with the instrument, and the same check source should be used in the same fashion to demonstrate the instrument's operability on a daily basis when the instrument is in use. For analog readout (count rate) instruments, a variation of $\pm 20\%$ is usually considered acceptable. Optionally, instruments that integrate events and display the total on a digital readout typically provide an acceptable average response range of 2 or 3 standard deviations. This is achieved by performing a series of repetitive measurements (10 or more is suggested) of background and check source response and determining the average and standard deviation of those measurements. From a practical standpoint, a maximum deviation of $\pm 20\%$ is usually adequate when compared with other uncertainties associated with the use of the equipment. The amount of uncertainty allowed in the response checks should be consistent with the level of uncertainty allowed in the final data. Ultimately the decision maker determines what level of uncertainty is acceptable.

Instrument response, including both the background and check source response of the instrument, should be tested and recorded at a frequency that ensures the data collected with the equipment is reliable. For most portable radiation survey equipment, MARSSIM recommends that a response check be performed twice daily when in use - typically prior to beginning the day's measurements and again following the conclusion of measurements on that same day. Additional checks can be performed if warranted by the instrument and the conditions under which it is used. If the instrument response does not fall within the established range, the instrument is removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails the post-survey source check, all data collected during that time period with the instrument must be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. Ultimately, the frequency of response checks must be balanced with the stability of the equipment being used under field conditions and the quantity of data being collected. For example, if the instrument experiences a sudden failure during the course of the day's work due to physical harm, such as a punctured probe, then the data collected up until that point is probably acceptable even though a post-use performance check cannot be performed. Likewise, if no obvious failure occurred but the instrument failed the post-use response check, then the data collected with that instrument should be viewed with great skepticism and possibly re-collected or randomly checked with a different instrument. Additional corrective action alternatives are presented in MARSSIM Section 9.3. If re-calibration is necessary, acceptable response ranges must be reestablished and documented.

Record requirements vary considerably and depend heavily on the needs of the user. While Federal and State regulatory agencies all specify requirements, the following records should be considered a minimum.

Laboratory Quality Control

- records documenting the traceability of radiological standards
- records documenting the traceability of electronic test equipment

Records for Instruments to be Calibrated

- date received in the calibration laboratory
- initial condition of the instrument, including mechanical condition (e.g., loose or broken parts, dents, punctures), electrical condition (e.g., switches, meter movement, batteries), and radiological condition (presence or absence of contamination)
- calibrator's records including training records and signature on calibration records
- calibration data including model and serial number of instrument, date of calibration, recommended recalibration date, identification of source(s) used, as found calibration results, and final calibration results as returned for use.

In addition, records of instrument problems, failures, and maintenance can be included and are useful in assessing performance and identifying possible needs for altered calibration frequencies for some instruments. Calibration records should be maintained at the facility where the instruments are used as permanent records, and should be available either as hard copies or in safe computer storage.

7.0 MEASUREMENT METHODS

This section provides a technical basis for radiological measurement techniques that are used to determine the radiological contamination status of surfaces through a series of measurements and/or samples collected. Data obtained may be statistically tested to ensure that the area or material is adequately characterized with respect to average and ranges of radiological contamination levels. The technical basis is provided for radiological measurements used and survey methodology. Changes in equipment, techniques, and procedures used for monitoring shall be documented (**10 CFR 835.704(e)**).

Measurement methods used to generate field data can be classified into two categories commonly known as scanning surveys and direct measurements. The decision to use a measurement method as part of the survey design is determined by the survey objectives. Scanning is performed to identify areas of elevated activity that may not be detected by other measurement methods. Direct measurements are analogous to collecting and analyzing samples to determine the average activity in a survey unit.

7.1 Direct Measurements

To conduct direct measurements of alpha, beta, and photon surface activity, instruments and techniques providing the required detection sensitivity are selected. The type of instrument and method of performing the direct measurement are selected as dictated by the type of potential contamination present, the measurement sensitivity requirements, and the objectives of the radiological survey.

Direct measurements are taken by placing the instrument at the appropriate distance (Measurements at several distances may be needed. Near-surface or surface measurements provide the best indication of the size of the contaminated region and are useful for model implementation. Gamma measurements at 1 m provide a good estimate of potential direct external exposure.) above the surface, taking a discrete measurement for a pre-determined time interval (e.g., 10 s, 60 s, etc.), and recording the reading. A one minute integrated count technique is a practical field survey procedure for most equipment and provides detection sensitivities that are below most DCGLs. However, longer or shorter integrating times may be warranted (see

MARSSIM Section 6.4.1 for information dealing with the calculation of direct measurement detection sensitivities).

Direct measurements may be collected at random locations in the survey unit. Alternatively, direct measurements may be collected at systematic locations and supplement scanning surveys for the identification of small areas of elevated activity (see MARSSIM Section 5.5.2.5). Direct measurements may also be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify location for direct measurements to further define the areal extent of contamination and to determine maximum radiation levels within an area, although these types of direct measurements are usually associated with preliminary surveys (i.e., scoping, characterization, remedial action support). All direct measurement locations and results should be documented.

If the equipment and methodology used for scanning is capable of providing data of the same quality required for direct measurement (e.g., detection limit, location of measurements, ability to record and document results), then scanning may be used in place of direct measurements. Results should be documented for at least the number of locations required for the statistical tests. In addition, some direct measurement systems may be able to provide scanning data, provided they meet the objectives of the scanning survey.

If a direct radiation measurement exceeds an action level determined on the basis of the potential contaminant and the detector and survey parameters, the location should be noted for further investigation (measurement or sampling).

Sections 7.6.1, 7.7.1, and 7.8.1, briefly describe methods used to perform direct measurements in the field. The instruments used to perform these measurements are described in more detail in Section 6.6 and MARSSIM Appendix H.

7.2 Scanning Surveys

Scanning is the process by which the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific surface (i.e., ground, wall, floor, equipment). The term scanning survey is used to describe the process of moving portable radiation detectors across a suspect surface with the intent of locating radionuclide contamination. Investigation levels for scanning surveys are determined during survey planning to identify areas of elevated activity. Scanning surveys are performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action. These investigation levels may be based on the DCGLW, the DCGLEMC, or some other level as discussed in MARSSIM Section 5.5.2.6.

Small areas of elevated activity typically represent a small portion of the site or survey unit. Thus, random or systematic direct measurements or sampling on the commonly used grid spacing may have a low probability of identifying such small areas. Scanning surveys are often relatively quick and inexpensive to perform. For these reasons, scanning surveys are typically performed before direct measurements or sampling. This way time is not spent fully evaluating an area that may quickly prove to be contaminated above the investigation level during the scanning process.

Scans are conducted which would be indicative of all radionuclides potentially present, based on the Historical Site Assessment, surfaces to be surveyed, and survey design objectives. Surrogate measurements may be utilized where appropriate (see MARSSIM Section 4.3.2). Documenting scanning results and observations from the field is very important. For example, a scan that identified relatively sharp increases in instrument response or identified the boundary of an area of increased instrument response should be documented. This information is useful when interpreting survey results.

The scanning technique should employ the most sensitive instrumentation that is suitable for field use. The type of measurement, suitable portable instrumentation, and specific methods to perform the measurements are selected by the Radiological Controls Department and designated in the survey plan as dictated by the type of radioactive contamination present, the instrumentation sensitivity requirements, and the degree of surface coverage needed to meet the survey objectives. Scans are conducted for all radiations potentially present (alpha, beta, and gamma radiations) based on the operational history and surfaces to be surveyed. Monitoring for the unexpected is recommended. For instance, the presence of radionuclides in concentrations well above guidelines in subsurface soil may be indicated during a general scan showing only a small, localized increase in elevated radiation levels. The instruments used to perform scanning surveys measurements are described in more detail in Section 6.6 and MARSSIM Appendix H.

7.2.1 Action Levels

Usually, a surveyor will investigate any anomalous reading that is recognized as being greater than the background response of the detection system. As such, the sensitivity of the scanning method will determine what level of activity can be detected. Action levels are typically used only in cases where one wants to stop and investigate count rate levels that are significantly above the background detector response. Action levels are determined prior to performing a scan survey on the basis of the potential contaminant and the detector and survey parameters. The action level is the count rate at which the surveyor should flag a localized area during a scan survey. The action level, in units of cpm, is estimated by use of the following calculation:

$$\text{Action Level} = G \times c \times E \quad \text{Eq.(7.1)}$$

where:

- G = cleanup guideline (derived concentration guideline [DCG])
- c = user selectable multiplier.
- E = detection efficiency in units of cpm per "DCG unit."

Example, if the DCC is 5000 dpm per 100 cm², then the "DCG unit" would be dpm per 100 cm² and the detection efficiency used in the equation would need to be stated in "cpm per dpm per 100 cm²."

As mentioned above, action levels as defined in the above equation are usually not used when surveying for small amounts of activity where the expected detector response is near background. Depending on the parameters discussed in Section 6.5, a small increase in the detector response above background will usually be the trigger that causes an investigator to stop and investigate further. Therefore, for most cases, the action level will be equal to the detection limit of the scanning technique.

7.3 Systematic Measurements and Sampling

Systematic samples are collected according to a predetermined pattern based on such factors as accessibility and the features of the site without regard to external radiation levels. The purpose of these measurements or samples is to provide definitive radiation levels and/or radionuclide concentrations at precisely defined locations. Furthermore, these measurements permit the calculation of average radiation levels and/or radionuclide concentrations within a given area (by averaging individual measurements or sample analytical results) for purposes of comparison with other areas or background samples, or to estimate potential health effects to people occupying that area. Systematic measurements may be performed for alpha, beta, or gamma radiation. Samples typically include soil and routine surface smears. The type of measurement or sample, suitable portable instrumentation, and specific method to perform the measurement or collect the sample are again selected by the individual investigator as dictated by the type of contamination present, the instrumentation sensitivity requirements, and the objectives of the radiological survey. Measurements are taken by placing the instrument at the appropriate distance above the surface, taking a discrete measurement for some time interval (i.e., instantaneous, 10 s, 60 s, etc.), and

recording the measurement. Collected samples are packaged, labeled, and taken to an appropriate facility for analysis.

Compositing certain groups of samples may be desirable. A composite sample is a sample formed by combining several individual field samples (or portions of them) into a new sample, which is thoroughly mixed before being measured (in part or as a whole). Composite samples may be used to estimate average environmental concentrations with less cost than is possible using the original individual field samples. In no case can samples be composited over an area greater than that given in the relevant guideline. Hot spots may never be included in compositing samples for comparison to guidelines. Measurements of composite samples may also be more comparable to survey measurements obtained using *in situ* radiation detectors. Compositing must be used with care as compositing may average out (mask) small areas that have high concentrations. Also, measurements of composite samples may not be comparable to measurements of individual (uncomposited) samples or of composite samples of different sizes. The numbers and sizes of individual samples may be determined on the basis of cost and the precision desired in the estimated average.

7.4 Biased Measurements and Samples

At locations where anomalous radiation levels are observed or suspected, biased radiological measurements and samples may be taken (“biased” indicates that the locations are not chosen on a random or systematic basis). The purposes of these measurements and samples are to further define the areal extent of potential contamination and to determine maximum radiation levels within an area. Biased measurements may include alpha, beta, or gamma radiation; however, at these locations measurements may also be supplemented with other types of atypical measurements such as radon flux or gamma spectroscopic measurements. Air, water, soil, and smear samples may typically be taken at these locations; samples of vegetation or sediment samples may be appropriate. All sample and measurement locations and results are recorded.

7.5 Removable Surface Activity

Smears, also known as swipes, provide a semi-quantitative measure of removable activity and are obtained by wiping an area using a dry filter paper while applying moderate pressure. The area of concern for smear surveys will usually be 100 cm² (15.5 in²) since current surface contamination guideline values are specified in terms of this areal size. If a different area is swiped, as for objects with a smaller surface area, the results should be corrected back to the same area as specified in the surface contamination guideline. If the surface is thickly coated with particulate material, such as rust or dirt, a sample of the particulate material should be collected as a separate sample instead of attempting to use a smear.

Dry paper filters with diameters ranging from around 30 mm up to 50 mm are typically used for smears, although fabric materials have been growing in popularity as the material of choice. Moistened paper swipes may be used to collect tritium from dry surfaces, but dry swipes should be used when the surface is damp. Materials that dissolve well in solvent-based scintillation cocktails, such as styrofoam, are also used by some for collecting tritium swipes. “Sticky” smears may be necessary under certain conditions such as a surface consisting of dry particles. Smears are placed into envelopes or other individual containers to prevent cross-contamination while awaiting analysis.

It is unlikely that outside surfaces, exposed to wind and rain, will have significant levels of removable surface activity. If removable activity is suspected, smears or swabs may be obtained and analyzed. Smears for removable surface activity are not appropriate for use on soil.

7.6 Gamma Measurements

External gamma radiation measurements are sometimes made to evaluate potential personnel exposures, and to provide a radiation "map" to assist in planning and implementation of subsequent remedial action. These radiation measurements should include the following:

- Gamma radiation measurements at near contact with the ground surface and 1 m (3.25 ft) above ground; average and maximum measurements for both indoors and outdoors can then be determined.
- Surface gamma-ray scanning to identify radiation anomalies and to define the areal extent of above-background radiation exposures.
- Surface gamma-ray scanning of equipment and other materials at the site where appropriate.
- Pressurized ionization chamber (PIC) measurements at locations of differing gamma radiation spectra. Since NaI detectors are very energy dependent, exposure rate measurements with both a PIC, or equivalent, and a NaI detector can be used to correlate the NaI response to the actual exposure rate. Essentially, a site-specific correction factor is determined by collecting paired measurements at points of different exposure rates.

7.6.1 Direct Measurements for Photon Emitting Radionuclides

Gamma radiation measurements are made at near contact with the ground surface and at 1 m (3.25 ft) above the ground at systematic locations, and at locations of elevated radiation identified by area gamma scans. Some limited sampling or the use of gamma spectroscopy will be required to identify the radionuclide and to determine if the residual activity is distributed in the surface layer of soil or subsurface layers. Portable gamma spectrometers allowing on-site radionuclide identification may be useful. Measurements at 1 m will not be necessary if external exposure rates do not need to be measured.

There are a wide variety of instruments available for measuring photons in the field (see MARSSIM Appendix H) but all of them are used in essentially the same way. The detector is set up at a specified distance from the surface being measured and data are collected for a specified period of time. The distance from the surface to the detector is generally determined by the calibration of the instrument because photons do not interact appreciably with air. When measuring x-rays or low-energy gamma rays, the detector is often placed closer to the surface to increase the counting efficiency. The time required to perform a direct measurement may vary from very short (e.g., 10 seconds) to very long (e.g., several days or weeks) depending on the type of detector and the required detection limit. In general, the lower the required detection limit the longer the time required to perform the measurement. A collimator may be used in areas where activity from adjacent or nearby areas might interfere with the direct measurement. The collimator (usually lead, tungsten, or steel) shields the detector from extraneous photons but allows activity from a specified area of the surface to reach the detector.

Example: The portable germanium detector, or in situ gamma spectrometer, can be used to estimate gamma-emitting radionuclide concentrations in the field. As with the laboratory-based germanium detector with multichannel analyzer, in situ gamma spectrometry can discriminate among various radionuclides on the basis of characteristic gamma and x-ray energies to provide a nuclide-specific measurement. A calibrated detector measures the fluence rate of primary photons at specific energies that are characteristic of a particular radionuclide. This fluence rate can then be converted to units of concentration. Under certain conditions the fluence rate may be converted directly to dose or risk for a direct comparison to the release criterion rather than to the DCGLW. Although this conversion is generally made, the fluence rate should be considered the fundamental parameter for assessing the level of radiation at a specific location because it is a directly measurable physical quantity.

For outdoor measurements, where the contaminant is believed to be distributed within the surface soil, it may be appropriate to assume a uniform depth profile when converting the fluence rate to a concentration. At sites where the soil is plowed or overturned regularly, this assumption is quite realistic because of the effects of homogenization. At sites where the activity was initially deposited on the surface and has gradually penetrated deeper over time, the actual depth profile will have a higher activity at the surface and gradually diminish with depth. In this case, the assumption of a uniform depth profile will estimate a higher radionuclide concentration relative to the average concentration over that depth. In cases where there is an inverted depth profile (i.e., low concentration at the surface that increase with depth), the assumption of a uniform depth profile will underestimate the average radionuclide concentration over that depth. For this reason, MARSSIM recommends that soil cores be collected to determine the actual depth profile for the site. These soil cores may be collected during the characterization or remedial action support survey to establish a depth profile for planning a final status survey. The cores may also be collected during the final status survey to verify the assumptions used to develop the fluence-to-concentration correction.

For indoor measurements, uncollimated in situ measurements can provide useful information on the low-level average activity across an entire room. The position of the measurement within the room is not critical if the radionuclide of interest is not present in the building materials. A measurement of peak count rate can be converted to fluence rate, which can in turn be related to the average surface activity. The absence of a discernible peak would mean that residual activity could not exceed a certain average level. However, this method will not easily locate small areas of elevated activity. For situations where the activity is not uniformly distributed on the surface, a series of collimated measurements using a systematic grid allows the operator to identify general areas of elevated contamination.

NUREG-1506, "Measurement Methods for Radiological Surveys in Support of New Decommissioning Criteria," provides a detailed description of the theory and implementation of in situ gamma spectrometry. In situ spectrometry is provided as one example of a useful tool for performing direct measurements for particular scenarios, but interpretation of the instrument output in terms of radionuclide distributions is dependent on the assumptions used to calibrate the method site-specifically. The depth of treatment of this technique in this example is not meant to imply that in situ gamma spectrometry is preferred a priori over other appropriate measurement techniques described in this document.

7.6.2 Scanning for Photon Emitting Radionuclides

Sodium iodide survey meters (NaI(Tl) detectors) are normally used for scanning areas for gamma emitters because they are very sensitive to gamma radiation, easily portable and relatively inexpensive. The detector is held close to the ground surface (~6 cm or 2.5 in.) and moved in a serpentine (i.e., snake like, S shaped) pattern while walking at a speed that allows the investigator to detect the desired investigation level. A scan rate of approximately 0.5 m/s is typically used for distributed gamma emitting contaminants in soil; however, this rate must be adjusted depending on the expected detector response and the desired investigation level.

For ease of detection, changes in the instrument response are monitored via the audible output using headphones, rather than by noting fluctuations in the analog meter reading. Actual measurements for all areas, including background as well as anomalous readings, should be recorded. Locations of direct radiation exceeding the action level are marked on facility maps and identified for further measurements and/or sampling. Discussion of scanning rates versus detection sensitivity for gamma emitters is provided in Section 6.5.1.B.

Sodium iodide survey meters are also used for scanning to detect areas with elevated areas of low-energy gamma and x-ray emitting radionuclides such as ^{241}Am and ^{239}Pu . Specially designed detectors, such as the FIDLER (field instrument for the detection of low energy radiation) probe with survey meter, are typically used to detect these types of radionuclides.

7.7 Alpha Measurements

Indoor alpha measurements should include the following when applicable: systematic measurement of surface alpha activity on walls and floor surfaces, measurement of alpha activity at locations of elevated gamma or beta radiation levels (when the contaminant is both an alpha- and beta-emitter), and measurement of alpha activity on potentially contaminated equipment surfaces.

7.7.1 Direct Measurements for Alpha Emitting Radionuclides

Direct measurements for alpha emitting radionuclides are generally performed by placing the detector on or near the surface to be measured. The limited range of alpha particles (e.g., about 1 cm or 0.4 in. in air, less in denser material) means that these measurements are generally restricted to relatively smooth, impermeable surfaces such as concrete, metal, or drywall where the activity is present as surface contamination.

In most cases, direct measurements of porous (e.g., wood) and volumetric (e.g., soil, water) material cannot meet the objectives of the survey. However, special instruments such as the long range alpha detector (see MARSSIM Appendix H) have been developed to measure the concentration of alpha emitting radionuclides in soil under certain conditions. Because the detector is used in close proximity to the potentially contaminated surface, contamination of the detector or damage to the detector caused by irregular surfaces need to be considered before performing direct measurements for alpha emitters.

To conduct direct (static) measurements of surface alpha activity, instruments and techniques providing the required detection sensitivity are selected. Experience has shown that a 1 minute integrated count technique, using a large area [$>50\text{ cm}^2$ (8 in.²)] detector, is a practical field survey procedure and will provide detection sensitivities that are below most guideline levels. However, under certain circumstances, longer or shorter integrating times may be warranted.

7.7.2 Scanning for Alpha Emitting Radionuclides

Alpha scintillation survey meters and thin window gas-flow proportional counters are typically used for performing alpha surveys. Alpha radiation has a very limited range and, therefore, instrumentation must be kept close to the surface, usually less than 1 cm (0.4 in.). For this reason, alpha scans are generally performed on relatively smooth, impermeable surfaces (e.g., concrete, metal, drywall) and not on porous material (e.g., wood) or for volumetric contamination (e.g., soil, water). In most cases, porous and volumetric contamination cannot be detected by scanning for alpha activity and meet the objectives of the survey because of high detection sensitivities. Under these circumstances, samples of the material are usually collected and analyzed as discussed in MARSSIM Chapter 7. Determining scan rates when surveying for alpha emitters is discussed in MARSSIM Section 6.7.2.2 and Appendix J.

7.8 Beta Measurements

Indoor beta measurements should include the following when applicable: anomalies, systematic locations [a minimum of one measurement for each 1 m² (10.8 ft²) for current release guidelines], specific locations where contamination by beta-emitting radionuclides is suspected, locations where gamma-ray exposure rates are significantly elevated, and measurement of beta activity on selected equipment surfaces.

Instruments and techniques providing the required detection sensitivity are selected to conduct direct measurements of surface beta activity.

7.8.1 Direct Measurements for Beta Emitting Radionuclides

Direct measurements for beta emitting radionuclides are generally performed by placing the detector on or near the surface to be measured, similar to measurements for alpha emitting radionuclides. These measurements are typically restricted to relatively smooth, impermeable surfaces where the activity is present as surface contamination. In most cases, direct measurements of porous (e.g., wood) and volumetric (e.g., soil, water) material cannot meet the objectives of the survey. However, special instruments such as large area gas-flow proportional counters (see MARSSIM Appendix H) and arrays of beta scintillators have been developed to measure the concentration of beta emitting radionuclides in soil under certain conditions. Similar to direct measurements for alpha emitting radionuclides, contamination of the detector and damage to the detector need to be considered before performing direct measurements for beta emitters.

Surface activity measurements may be performed at systematically and randomly selected locations and at locations of elevated radiation identified by surface scans. A 1 minute integrated count technique using a large area [>50 cm² (8 in.²)] detector should be used and should provide detection sensitivities that are below guideline levels.

7.8.2 Scanning for Beta Emitting Radionuclides

Thin window gas-flow proportional counters are normally used when surveying for beta emitters, although solid scintillators designed for this purpose are also available. Typically, the beta detector is held less than 2 cm from the surface and moved at a rate such that the desired investigation level can be detected. Low-energy (<100 keV) beta emitters are subject to the same interferences and self-absorption problems found with alpha emitting radionuclides, and scans for these radionuclides are performed under similar circumstances. Determination of scan rates when surveying for beta emitters is discussed in Section 6.5.2.

7.9 Soil Sampling

7.9.1 Surface Soil

The purpose of surface soil sampling is to collect samples that accurately and precisely represent the radionuclides and their concentrations at the location being sampled. In order to do this and plan for sampling, a decision must be made as to the survey design.

The selection of a survey design is based on the Historical Site Assessment, results from preliminary surveys (i.e., scoping characterization, remedial action support), and the objectives of the survey developed using the Data Quality Objectives (DQO) Process. (Note: See Sections 7.3 and 7.4 for “biased” and “systematic” samples. The selection between judgmental, random, and systematic survey designs is discussed further in MARSSIM Section 5.5.3.)

A. Sample Volume

The volume of soil collected should be specified in the sample collection procedure. In general, large volumes of soil are more representative than small volumes of soil. In addition, large samples provide sufficient sample to ensure that required detection limits can be achieved and that sample re-analysis can be done if there is a problem. However, large samples may cause problems with shipping, storage, and disposal. All of these issues should be discussed with the sample collection team and the analytical laboratory during development of sample collection procedures. In general, surface soil samples range in size from 100 g up to several kilograms.

The sample collection procedure should also make clear if it is more important to meet the volume requirement of the survey design or the surface area the sample represents. Constant volume is related to comparability of the results while surface area is more closely related to the representativeness of the results. Maintaining a constant surface area and depth for samples collected for a particular survey can eliminate problems associated with different depth profiles. The actual surface area included as part of the sample may be important for estimating the probability of locating areas of elevated concentration.

B. Sample Content

The material present in the field at the sample location may or may not provide a representative sample. Vegetative cover, soil particle size distribution, inaccessibility, or lack of sample material are examples of problems that may be identified during sample collection. All deviations from the survey design as documented in the procedures should be recorded as part of the field sample documentation.

Sample content is generally defined by the assumptions used to develop the conceptual site model and the DCGLs. A typical agricultural scenario assumes that the top few centimeters of soil are available for resuspension in air, that the top 15 cm (6 in.) are homogenized by agricultural activities (e.g., plowing), that roots can extend down several meters to obtain water and nutrients depending on the plant, and that external exposure is based on an assumed thickness of contaminated soil (usually at the surface). Depending on the dominant exposure pathways for each radionuclide, this can result in a complicated set of instructions for collecting representative samples. This situation can be further complicated by the fact that the site is not currently being used for agricultural purposes. For this situation it is necessary to look at the analytical results from the preliminary surveys (i.e., scoping, characterization, remedial action support) to determine the expected depth of contamination.

In most situations the vegetative cover is not considered part of the surface soil sample and is removed in the field. For agricultural scenarios where external exposure is not the primary concern, soil particles greater than 2 mm (0.08 in.) are generally not considered as part of the sample. Foreign material (e.g., plant roots, glass, metal, or concrete) is also generally not considered part of the sample, but should be reviewed on a site-specific basis. It is important that the sample collection procedure clearly indicate what is and what is not considered part of the sample.

C. Sampling Equipment

The selection of proper sampling equipment is important to ensure that samples are collected effectively and efficiently. Sampling equipment generally consists of a tool to collect the sample and a container to place the collected sample in. Sample tracking begins as soon as the sample is collected, so it may be necessary to consider security of collected samples required by the objectives of the survey.

Sampling may be conducted using a variety of simple hand tools, such as a shovel, trowel, or "cookie-cutter" tool. Samples should be representative of a known surface area. Sampling tools are selected based on the type of soil, sample depth, number of samples required, and training of available personnel. The selection of a sampling tool may also be based on the expected use of the results. For example, if a soil sample is collected to verify the depth profile used to develop the calibration for *in situ* gamma spectrometry, it is important to preserve the soil core. Table 9 lists several examples of tools used for collecting soil samples, situations where they are applicable, and some advantages and disadvantages involved in their use. If there is a potential for soil activity beneath paved surfaces, the surface can be removed by coring and the underlying soil sampled, as surface soil. Sampling tools should be cleaned and monitored, as appropriate, after each use.

Sample containers are generally not a major concern for collecting surface soil samples. Polyethylene bottles with screw caps and wide mouths are recommended. These containers are fairly economical, provide easy access for adding and removing samples, and resist chemicals, breaking, and temperature extremes. Glass containers are also acceptable, but they are fragile and tend to break during shipment. Metal containers are sometimes used, but sealing the container can present a problem and corrosion can be an issue if the samples are stored for a significant length of time.

7.9.2 Subsurface

Subsurface investigations consist of measurements and samples taken beneath the ground or floor surface. The purpose of these investigations is to locate and define vertical extent of the contamination. These investigations are conducted by excavating the floor or ground surface (by trenching, augering, coring, shoveling, or other means) to depths that are below contaminated soil. These depths are controlled by several factors and must be determined during the logging/sampling procedure. It may be possible to determine the maximum drilling depth from field measurements or by excavating to undisturbed soil. The environmental conditions at some depth may appear to prevent further downward migration of contaminants; thus, no further drilling may be required; in other instances, it may be necessary to rely on the results of laboratory analyses of samples because some radionuclides are not detectable with field instrumentation.

Filled areas, buried piping and underground tanks, spills, and septic leach fields that may have received contaminated materials are locations that may require sampling of subsurface soil. The need for special sampling by coring or split-spoon equipment, usually by a commercial firm, should be anticipated.

(A split-spoon (or split-barrel") sampler is constructed in such a way as to allow the collection of samples from relatively precise and determinable locations within a hole with little possibility of contamination by soil from other depths of the hole. The split-spoon tool is available in various sizes and lengths, and is pipe-shaped in appearance. Soil fills the "pipe" as it is driven into the ground and is prevented from loss by a flanged basket device as the tool is withdrawn. The sampler "splits" vertically in half for sample removal. Samples collected in such a manner may also be called "core" samples.)

Excavated material or material from the sides of the vertical walls, and water or air in the excavated hole may be sampled for radionuclide analyses. The number of excavations and the type of measurements or samples to be obtained and appropriate procedures to be used will be determined by the type of contamination present, limitations in field conditions, and objectives of the survey plan.

Subsurface soil may be sampled using portable manual equipment or, if the sampling depth is greater than several meters, heavier truck-mounted sampling rigs. For shallow subsurface sampling, the hole is advanced to the desired starting depth, using a post-hole digger, shovel, twist auger, motorized auger, or punch-type Shelby tube sampler. Loose material is removed from the hole and the sample collected over the next 15- or 30-cm (6- or 12-in.) depth. Continuous coring samplers or barrel samplers, advanced through hollow stem augers, are usually used for obtaining deeper subsurface samples. The entire core can be retained and monitored intact to determine if layers of activity are present, or sections of the core can be removed for analysis. Unless there is prior information regarding the depth and distribution of subsurface activity, samples should be obtained at approximately 1-m (3.25-ft) intervals (or smaller if necessary for guideline compliance) from the surface to below the suspected depth of the residual activity. Many States and local governments have regulations restricting the drilling of boreholes and requiring special handling of drilling spoils and back-filling of holes. Investigators should consult these agencies before initiating subsurface investigations.

**Table 9 (MARSSIM Table 7.1)
 Soil Sampling Equipment***

Equipment	Application	Advantages/Disadvantages
Tier	Soft surface soil	Inexpensive; easy to use and decontaminate; difficult to use in stone or dry soil.
Scoop or trowel	Soft surface soil	Inexpensive; easy to use and decontaminate; trowels with painted surfaces should be avoided
Bulb Planter	Soft Soil, 0-15 cm (0-6 in.)	Easy to use and decontaminate; uniform diameter and sample volume; preserves soil core; limited depth capability; can be difficult to decontaminate
Soil Coring Device	Soft soil, 0-60 cm (0-24 in.)	Relatively easy to use; preserves soil core; limited depth capability; can be difficult to decontaminate
Thin-wall tube sampler	Soft soil, 0-3 m (0-10 ft)	Easy to use; preserves soil core; easy to decontaminate; can be difficult to remove cores
Split spoon sampler	Soil, to bedrock	Excellent depth range; preserves soil core; useful for hard soils; often used in conjunction with drill rig for obtaining deep cores
Shelby tube sampler	Soft soil, to bedrock	Excellent depth range; preserves soil core; tube may be used for shipping core to lab.; may be used in conjunction with drill rig for obtaining deep cores
Bucket auger	Soft soil, 7.5 cm - 3 m (3 in. - 10 ft)	Easy to use; good depth range; uniform diameter and sample volume; may disrupt and mix soil horizons greater than 15 cm
Hand-operated power auger	Soil, 15 cm - 4.5 m (6 in. - 15 ft)	Good depth range; generally used in conjunction with bucket auger; destroys soil core; requires two or more operators; can be difficult to decontaminate

7.10 Background Measurements

Many release criteria for residual radioactive material are presented in terms of radiation levels or activity levels above normal background for an area or facility. As a result, background measurements are collected in reference areas to provide baseline data for comparison with measurements and data collected at a site. Background measurements and samples should be site- or area-specific - or, when surveying special material, be material-specific - and for each type of measurement a comparable reference background radiation level should be known.

These background radiation levels may be determined by consulting radiological control procedures or the radiological control technician may take readings either before or during the radiological survey of the site. In some instances, background radiation levels may be determined by consulting documented values in published reports. Environmental baseline surveys may also be useful. NUREG-1501, "Background as a Residual Radioactivity Criterion for Decommissioning," provides a considerable amount of information pertaining to the sources of and evaluation of radiation and radionuclide backgrounds.

Certain radionuclides may also occur at significant levels as part of background in the media of interest (soil, building material, etc.). Examples include members of the naturally-occurring uranium, thorium, and actinium series; ^{40}K ; ^{14}C ; and tritium. ^{137}Cs and other radionuclides are also present in background as a result of nuclear weapons fallout. Establishing background concentrations that describe a distribution of measurement data is necessary to identify and evaluate contributions attributable to site operations. Determining background levels for comparison with the conditions determined in specific survey units entails conducting surveys in one or more reference areas to define the radiological conditions of the site. NUREG-1505 provides additional information on background reference areas.

7.10.1 Select Background Reference Areas

Reference areas provide a location for background measurements which are used for comparisons with survey unit data. The radioactivity present in a reference area would be ideally the same as the survey unit had it never been contaminated. If a site includes physical, chemical, geological, radiological, or biological variability that is not represented by a single reference background area, selecting more than one reference area may be necessary.

A site background reference area should have similar physical, chemical, geological, radiological, and biological characteristics as the survey unit being evaluated. Background reference areas are normally selected from non-impacted areas, but are not limited to natural areas undisturbed by human activities. In some situations, a reference area may be associated with the survey unit being evaluated, but cannot be potentially contaminated by site activities. For example, background measurements may be taken from core samples of a building or structure surface, pavement, or asphalt. This option should be discussed with the responsible regulatory agency during survey planning. Generally, reference areas should not be part of the survey unit being evaluated.

Background reference locations to be avoided when possible include those that may have been affected or disturbed by non-site commercial activities particularly those that may have dealt with the same contaminant.

It may be difficult to find a reference area within an industrial complex for comparison to a survey unit if the radionuclides of potential concern are naturally occurring. Background may vary greatly due to different construction activities that have occurred at the site. Examples of construction activities that change background include: leveling; excavating; adding fill dirt; importing rocks or gravel to stabilize soil or underlay asphalt; manufacturing asphalt with different matrix rock; using different pours of asphalt or concrete in a single

survey unit; layering asphalt over concrete; layering different thicknesses of asphalt, concrete, rock, or gravel; and covering or burying old features such as railroad beds or building footings. Background variability may also increase due to the concentration of fallout in low areas of parking lots where runoff water collects and evaporates. Variations in background of a factor of five or more can occur in the space of a few hectares.

There are a number of possible actions to address these concerns. Reviewing and reassessing the selection of reference areas may be necessary. Selecting different reference areas to represent individual survey units is another possibility. More attention may also be needed in selecting survey units and their boundaries with respect to different areas of potential or actual background variability. More detailed scoping or characterization surveys may be needed to better understand background variability. Using radionuclide-specific measurement techniques instead of gross radioactivity measurement techniques may also be necessary. If a background reference area that satisfies the above recommendations is not available, consultation and negotiation with the responsible regulatory agency is recommended. Alternate approaches may include using published studies of radionuclide distributions.

Verifying that a particular background reference area is appropriate for a survey can be accomplished using the techniques described or referenced in MARSSIM Chapter 8. Verification provides assurance that assumptions used to design the survey are appropriate and defensible. This approach can also prevent decision errors that may result from selecting an inappropriate background reference area.

If the radionuclide contaminants of interest do not occur in background, or the background levels are known to be a small fraction of the DCGLW (e.g., <10%), the survey unit radiological conditions may be compared directly to the specified DCGL and reference area background surveys are not necessary. If the background is not well defined at a site, and the decision maker is willing to accept the increased probability of incorrectly failing to release a survey unit (Type II error), the reference area measurements can be eliminated and a one-sample statistical test performed as described in Section 9.10.

The levels of many radionuclides occurring naturally in the environment are insufficient to be quantifiable using standard measurement techniques. Those naturally occurring concentrations may also be insignificant relative to the DCGs. On the other hand, levels of direct radiation (exposure rates) and some naturally occurring (uranium and thorium decay series, and ^{40}K) or man-made (^{137}Cs) radionuclides are typically present in the environment at levels that are easily quantifiable and may have background levels that are significant, relative to the DCG.

Localized geologic formations, different types of soil, and construction materials at the background measurement locations may result in background values that have greater variability. Consequently, the number of measurements required to ensure a representative average value is dependent on specific site conditions. Large sites with complex geology may require separate background determinations for selected areas of like geology and soil type. Soil moisture, for example, can account for 30% of the soil mass during wet periods and can significantly affect results when gamma-fluence rate measurements. An underlying layer of "Chattanooga shale" containing elevated concentrations of natural uranium may enhance both the soil concentrations and the surface exposure rate. Igneous rock contributes less radionuclide content to soils than does sedimentary rock because, although it is high in radioactive content, it weathers more slowly than the softer sedimentary rock.

Background levels for indoor measurements may differ from those in open land areas because construction materials often contain naturally occurring radioactive substances that can provide a shielding effect. Preferable locations for interior background determinations are within buildings of similar construction, but having no history of

involvement with radioactive materials. Since the amount of naturally occurring radioactivity varies with material type, the background levels for specific materials being surveyed should be evaluated when necessary. Masonry brick, for example, often contains elevated levels of naturally occurring ^{232}Th , ^{238}U and ^{40}K . The presence of naturally occurring radioactive materials will cause an increase in the count rate from most beta and gamma detectors. As a result, slower scanning rate will be required, and the possibility of detecting a contaminant at the DCG may even be prevented.

Total radiation or radioactivity levels measured in each survey unit will be compared to background levels obtained. Therefore, the background levels should be determined with an accuracy at least equivalent of the data to which it will be compared. This can be achieved by using the same instruments and techniques for background surveys as are used in assessing site conditions. The background radiation measurements should be presented in the survey report and should be discussed in the results. The procedure for testing the data and determining the number of additional samples and/or measurements needed is described as follows.

7.10.2 Determining Numbers of Background (Reference) Data Points

The number of data points needed in a reference area(s) is based upon the sampling statistics used and is discussed in MARSSIM Section 5.5.2.

8.0 ANALYSIS OF MEASUREMENTS

8.1 Analytical Methods

Analytical methods should be capable of measuring levels below the established release guidelines: Detection sensitivities of better than 10% of the guideline should be the target. However, unless technically impractical, methods selected should be capable of detecting at least 50% of the guideline value. Detection limits are subject to variation from sample to sample, instrument to instrument, and procedure to procedure, depending on sample size, geometry, background, instrument efficiency, abundance of the radiations being measured, counting time, self-absorption in the prepared sample, and interferences from other radionuclides present.

8.1.1 Analysis of Smears

A. Gross Alpha/Gross Beta

The most popular method for laboratory smear and air filter analysis shall be to count both gross alpha and beta levels in a low-background proportional system; both automatic sample changer and manual multidetector instruments are used. Such systems have low backgrounds, relatively good detection sensitivity, and the capability of processing large quantities of samples in a short time. Using counting times of several minutes, measurement sensitivities of less than 10 dpm alpha and 20 dpm beta can be achieved. Filter papers can also be measured using standard field instruments, such as alpha scintillation and thin-window G-M detectors with integrating scalers. The measurement sensitivities of such techniques are not nearly as low as the low-background proportional system; however, for 5-min counting times, alpha and beta levels below 20 dpm and 100 dpm, respectively, can be measured. One of the major drawbacks to such a procedure is that it is very labor-intensive.

Filter papers can also be covered with a thin disk of zinc sulfide scintillator and counted for gross alpha using a photomultiplier tube attached to a scaler. While such a system provides a sensitivity comparable to that of the low-background proportional counter, it is also not usually automated and it is, therefore, a labor-intensive method.

B. Liquid Scintillation

Liquid scintillation is the preferred method for counting low-energy beta-emitters (for example ^3H , ^{14}C , and ^{63}Ni) and is excellent for counting high energy beta (^{32}P) and low-energy gamma emitters (^{125}I). Smears can be placed directly in a scintillation cocktail and counted on a liquid scintillation spectrometer. The counting efficiency may be reduced, but for the screening method, this process will yield reasonable results. With the spectrum capability of the newer instruments, the analyst can (in most cases) identify the specific beta emitter(s) present. The introduction of the sample into the liquid scintillation medium produces quenching, a reduction in the efficiency of the scintillator as a result of the introduction of the sample. To evaluate the effect of quenching, a known amount of the identified radionuclide (referred to as an internal standard or spike) may be added to the sample after initial measurement and a recount performed to enable determination of the detection efficiency of the specific sample. It should be noted that even with identification of the nuclide(s) on the smears, this is still a gross analysis and caution is advised in trying to infer too much from this information.

As a precaution against accidental contamination of the laboratory facility, it is prudent to first screen smears by gross GM or gamma counting. If little contamination is expected, all smears collected at the facility (or in a particular survey area) maybe assayed at once by placing all the smears on the detector. This will provide a broad screen for expected and unexpected contaminants. If contamination is detected, the smears should be recounted in smaller groups until the contaminated smears are isolated. Since the procedure is non-destructive, it will not interfere with subsequent analysis of the smears. When performing such screening, the smears should be left in their protective "envelopes" to avoid cross contamination.

8.1.2 Analysis of Soil and Sediment

After the soil has been placed in an appropriate container, the sample is transferred to the Radiological Controls Laboratory for analysis, or to the Environmental Laboratory for shipping preparations and then transported to an offsite laboratory facility for analysis. Sample aliquots may be taken prior to shipment by WVES for field screening methods for both gross alpha and gross beta-gamma for transportation classification and to verify to the offsite laboratory, the concentration range of contaminants present. Field screening methods should be compared or correlated to laboratory analytical results to verify that screening methods are adequate.

The analysis of soil is dependent on the radionuclides of interest. If the contaminants include gamma emitters, the sample will be analyzed using gamma spectroscopy (a non-destructive analysis that can identify and quantify multiple gamma-emitting nuclides). It is prudent to subject at least a representative number of soil or sediment samples to gamma spectral analysis, even if no gamma emitters are expected, as a check on the reliability of the identification of potential contaminants.

Either solid-state germanium detectors or sodium iodide scintillation detectors may be used; however, the solid-state detector has an advantage in its ability to resolve multiple gamma photopeaks that differ by as little as 0.5 to 1 keV of each other.

Although state-of-the-art systems include inherent computer-based spectrum analysis capabilities, it is important that an experienced analyst carefully review each spectrum because at the low concentrations typically encountered in radiological surveys, problems with resolution, interferences, peak shifts and linearity may not be readily apparent. Spectra should also be reviewed for gamma photopeaks not previously identified as

principal facility contaminants of concern. Special attention must be given to those radionuclides that may have difficult to resolve photopeaks, for example ^{226}Ra (186.2 keV) and ^{235}U (185.7 keV), and select, secondary photopeaks or daughter photopeaks for calculations. An example would be the use of a daughter in the ^{226}Ra decay series, ^{214}Bi (609 keV peak), as an alternate for determining the quantity of ^{226}Ra present. When using such an approach, it is also necessary that the equilibrium status between the parent and daughters be known.

Soil/sediment analysis by gamma spectrometry can be performed with varying sample sizes, using geometries such as a 0.5 L Marinelli beaker, 100- to 400-mL cans or jars, various sizes of petri dishes, or standard 30-mL scintillation vials. Counting times ranging from one-half hour to 4 hours are usually adequate to detect most radionuclides at concentrations currently being used as cleanup guidelines. Longer counting times may be necessary for radionuclides with low gamma-emission rates (abundances) and/or low guideline concentration values.

Analysis of soil/sediment samples for most pure beta radionuclides, such as ^{90}Sr , ^{63}Ni and ^{99}Tc requires wet chemistry separation, followed by counting using liquid scintillation or beta proportional instruments. Each radionuclide (element) requires a specific procedure for the chemical separation. As with alpha spectrometry techniques a known amount of tracer is added to the sample to determine recovery. Lower limits of detection of less than 1 pCi/g are achievable using standard methods.

A recently introduced analytical technique uses liquid scintillation counting to measure alpha-emitting contaminant concentrations. This system is known as PERALS (photon electron rejecting alpha liquid scintillator). While this technique does not provide quite the resolution as conventional alpha spectrometry (solid-state detectors), it provides greater sensitivity, the chemical procedures are less rigorous, and the results are obtainable in a much shorter time.

Radionuclides emitting primarily alpha particles are best analyzed by wet chemistry separation, followed by counting to determine amount of specific alpha energies present. Elements of concern can be removed from a solid sample by acid leaching, or samples can be fused at high temperatures into fluoride and pyrosulfate fluxes. This latter process ensures that all chemical species are in an ionic state that is more readily dissolved. (The process of leaching certain chemical forms of radionuclides from the soil matrix has been found to be less consistent than total dissolution of the sample matrix.) After dissolution, barium sulfate is precipitated to carry the alpha emitters out of solution. The precipitate is dissolved and the various nuclides are separated by oxidation-reduction reactions. After final separation and cleanup, the nuclides of interest are electroplated onto a metal disc or co-precipitated (with either neodymium or cerium fluoride) and collected on a filter paper. The metal disc or filter paper is then counted using a solid-state surface barrier detector and alpha spectrometer.

A known amount of tracer is added to the sample before the chemical separation to determine the fraction of the radionuclide recovered in the procedure. This also provides a "calibration" of the analytical system for each sample processed. Lower limits of detection are less than 1 pCi/g using standard alpha spectrometry methods. Sample quantities for such procedures are typically a few grams or less.

8.2 Analytical Procedures

The selection of the appropriate radio-analytical methods is normally made prior to the procurement of analytical services and is included in the statement-of-work of the request for proposal. The statement-of-work may dictate the use of specific methods or be performance based. Unless there is a regulatory requirement, such as conformance to the EPA drinking water methods, the specification of performance based methodology is encouraged. One reason for this is that a laboratory will usually perform better using the methods routinely employed in its laboratory as contrasted to using other methods with which it has less experience. The laboratory is also likely to have historical data on performance for methods routinely used by that laboratory. However, the methods employed, i.e., references for routine analytical methods, in a laboratory should be derived from a reliable source.

This section briefly describes specific equipment and procedures to be used once the sample is prepared for analysis. The results of these analyses (i.e., the levels of radioactivity found in these samples) are the values used to determine the level of residual activity at a site. In a decommissioning effort, the DCGLs are expressed in terms of the concentrations of certain radionuclides. It is of vital importance, therefore, that the analyses be accurate and of adequate sensitivity for the radionuclides of concern. The selection of analytical procedures should be coordinated with the laboratory.

Analytical methods should be adequate to meet the data needs identified in the DQO process. Consultation with the laboratory performing the analysis is recommended before selecting a course of action. MARSSIM is not intended to limit the selection of analytical procedures, rather all applicable methods should be reviewed to provide results that meet the objectives of the survey. The decision maker and survey planning team should decide whether routine methods will be used at the site or if non-routine methods may be acceptable.

- Routine analytical methods are documented with information on minimum performance characteristics, such as detection limit, precision and accuracy, and useful range of radionuclide concentrations and sample sizes. Routine methods may be issued by a recognized organization (e.g., Federal or State agency, professional organization), published in a refereed journal, or developed by an individual laboratory. Section 8.1.3 lists examples of sources for routine methods.
- Non-routine methods address situations with unusual or problematic matrices, low detection limits, or new parameters, procedures or techniques. Non-routine methods include adjustments to routine methods, new techniques published in reference literature, and development of new methods.

References that provide information on radiochemical methodology and should be considered in the methods review and selection process are available from such organizations as:

- National Council on Radiation Safety and Measurements (NCRP)
- American Society of Testing and Materials (ASTM)
- Radiological and Environmental Sciences Laboratory (RESL), Idaho Falls, Idaho (Operated by the DOE)
- DOE Technical Measurements Center, Grand Junction, CO
- Environmental Measurements Laboratory (EML); formerly the Health and Safety Laboratory of the DOE

Equipment vendor literature, catalogs, and instrument manuals are often a source of useful information on the characteristics of radiation detection equipment. Table 10, "Typical Measurement Sensitivities for Laboratory Radiometric Procedures," provides a summary of common laboratory methods with estimated detection limits.

Analytical procedures in the laboratory consist of several parts that are assembled to produce an SOP for a specific project or sample type. These parts include:

- laboratory sample preparation
- sample dissolution
- sample purification
- preparation for counting
- counting
- data reduction

**Table 10 (DOE EIGRSP Table 6.1)
 Typical Measurement Sensitivities for Laboratory Radiometric Procedures**

Sample Type	Radionuclides or Radiation Measured	Procedure	Approximate Measurement Sensitivity
Smears (filter paper)	Gross alpha	Gas-flow proportional counter; 5-min count	5 dpm
		Alpha scintillation detector with scaler; 5-min count	20 dpm
	Gross beta	Gas-flow proportional counter; 5-min count	10 dpm
	Low energy beta (^3H , ^{14}C , ^{63}Ni)	End window GM with scaler; 5-min count (unshielded detector)	80 dpm
Soil Sediment	^{137}Cs , ^{60}Co , ^{226}Ra (^{214}Bi)*, ^{232}Th (^{228}Ac), ^{235}U 234 , 235 , ^{238}U ; 238 , 239 , ^{240}Pu ; 227 , 228 , 230 , ^{232}Th ; other alpha emitters	Germanium detector (25% relative efficiency) with multichannel analyzer; pulse height analyzer; 500-g sample; 15-min analysis	0.04-0.1 Bq/g (1-3 pCi/g)
		Alpha spectroscopy with multichannel analyzer - pyrosulfate fusion and solvent extraction; surface barrier detector; pulse height analyzer; 1-g sample; 16-hr count	0.004-0.02 Bq/g (0.1-0.5 pCi/g)
Water	Gross alpha	Gas-flow proportional counter; 100-ml sample, 200-min count	0.04 Bq/L (1 pCi/l)
		Gas-flow proportional counter; 100-ml sample, 200-min count	0.04 Bq/L (1 pCi/L)
	^{137}Cs , ^{60}Co , ^{226}Ra (^{214}Bi), ^{232}Th (^{228}Ac), ^{235}U 234 , 235 , ^{238}U ; 238 , 239 , ^{240}Pu ; 227 , 228 , 230 , ^{232}Th ; other alpha emitters	Germanium detector (25% relative efficiency) with multichannel analyzer; pulse height analyzer; 3.5L sample, 16-hr count	0.4 Bq/L (10 pCi/L)
		Alpha spectroscopy with multichannel analyzer - solvent extraction; surface barrier detector; pulse height analyzer; 100 ml sample, 30 min count	0.004-0.02 Bq/L (0.1-0.5 pCi/L)
^3H	Liquid scintillation spectrometry; 5-ml sample, 30-min count	100 Bq/L (300 pCi/L)	

* Indicates that a member of the decay series is measured to determine activity level of the parent radionuclide of primary interest.

8.2.1 Photon Emitting Radionuclides

There is no special sample preparation required for counting samples using a germanium detector or a sodium iodide detector beyond placing the sample in a known geometry for which the detector has been calibrated. The samples can be measured as they arrive at the laboratory, or the sample can be dried, ground to a uniform particle size, and mixed to provide a more homogeneous sample if required by procedures.

The samples are typically counted using a germanium detector with a multichannel analyzer or a sodium iodide detector with a multichannel analyzer. Germanium detectors have better resolution and can identify peaks (and the associated radionuclides) at lower concentrations. Sodium iodide detectors often have a higher efficiency and are significantly less expensive than germanium detectors. Low-energy photons (i.e., x-rays and gamma rays below 50 keV) can be measured using specially designed detectors with an entrance window made from a very light metal, typically beryllium. Descriptions of germanium and sodium iodide detectors are provided in MARSSIM Appendix H.

Data reduction is usually the critical step in measuring photon emitting radionuclides. There are often several hundred individual gamma ray energies detected within a single sample. Computer software is usually used to identify the peaks, associate them with the proper energy, associate the energy with one or more radionuclides, correct for the efficiency of the detector and the geometry of the sample, and provide results in terms of concentrations with the associated uncertainty. It is important that the software be either a well-documented commercial package or thoroughly evaluated and documented before use.

8.2.2 Beta Emitting Radionuclides

Laboratory sample preparation is an important step in the analysis of surface soil and other solid samples for beta emitting radionuclides. The laboratory will typically have a sample preparation procedure that involves drying the sample and grinding the soil so that all of the particles are less than a specified size to provide a homogeneous sample. A small portion of the homogenized sample is usually all that is required for the individual analysis.

Once the sample has been prepared, a small portion is dissolved, fused, or leached to provide a clear solution containing the radionuclide of interest. The only way to ensure that the sample is solubilized is to completely dissolve the sample. However, this can be an expensive and time-consuming step in the analysis. In some cases, leaching with strong acids can consistently provide greater than 80% recovery of the radionuclide of interest and may be acceptable for certain applications. Gross beta measurements may be performed on material that has not been dissolved.

After dissolution, the sample is purified using a variety of chemical reactions to remove bulk chemical and radionuclide impurities. The objective is to provide a chemically and radiologically pure sample for measurement. Examples of purification techniques include precipitation, liquid-liquid extraction, ion-exchange chromatography, distillation, and electrodeposition. Gross beta measurements may be performed on material that has not been purified.

After the sample is purified, it is prepared for counting. Beta emitting radionuclides are usually prepared for a specific type of counter in a specified geometry. Solid material is usually precipitated and collected on a filter in a circular geometry to provide a homogeneous sample. Liquid samples are typically converted to the appropriate chemical form and diluted to a specified volume in preparation for counting.

Measurements of solid samples are typically performed using a gas-flow proportional counter. Because total beta activity is measured, it is important that the purification step be performed to remove any interfering radionuclides. Liquid samples are usually diluted using a liquid scintillation cocktail and counted using a liquid scintillation spectrometer. Liquid scintillation spectrometers can be used for low-energy beta emitting radionuclides, such as ^3H and ^{63}Ni . They also have high counting efficiencies, but often have a high instrument background as well. Gas-flow proportional counters have a very low background. MARSSIM Appendix H provides a description of both the gas-flow proportional counter and the liquid scintillation spectrometer.

Data reduction for beta emitting radionuclides is less complicated than that for photon emitting radionuclides. Since the beta detectors report total beta activity, the calculation to determine the concentration for the radionuclide of interest is straightforward.

8.2.3 Alpha Emitting Radionuclides

Laboratory sample preparation for alpha emitting radionuclides is similar to that for beta emitting radionuclides. Sample dissolution and purification tasks are also similar to those performed for beta emitting radionuclides.

Because of the limited penetrating power of alpha particles, the preparation for counting is often a critical step. Gross alpha measurements can be made using small sample sizes with a gas-flow proportional counter, but self-absorption of the alpha particles results in a relatively high detection limit for this technique. Liquid scintillation spectrometers can also be used to measure alpha emitting radionuclides but the resolution limits the usefulness of this technique. Most alpha emitting radionuclides are measured in a vacuum (to limit absorption by air) using alpha spectroscopy. This method requires that the sample be prepared as a virtually weightless mount in a specific geometry. Electro-deposition is the traditional method for preparing samples for counting. This technique provides the highest resolution, but it requires a significant amount of training and expertise on the part of the analyst to produce a high quality sample. Precipitation of the radionuclide of interest on the surface of a substrate is often used to prepare samples for alpha spectroscopy. While this technique generally produces a spectrum with lower resolution, the preparation time is relatively short compared to electro-deposition, and personnel can be trained to prepare acceptable samples relatively quickly.

Alpha emitting radionuclides are typically measured using alpha spectroscopy. The data reduction requirements for alpha spectroscopy are greater than those for beta emitting radionuclides, and similar to those for photon emitting radionuclides. Alpha spectroscopy produces a spectrum of alpha particles detected at different energies, but because the sample is purified prior to counting, all of the alpha particles come from radionuclides of a single element. This simplifies the process of associating each peak with a specific radionuclide, but the lower resolution associated with alpha spectroscopy increases the difficulty of identifying the peaks. Although commercial software packages are available for interpreting alpha spectroscopy results, an experienced operator is required to ensure that the software is working properly.

9.0 CALCULATION

9.1 Data Conversion

This section describes methods for converting survey data to appropriate units for comparison to radiological criteria. Conditions applicable to satisfying decommissioning requirements include determining that any residual contamination will not result in individuals being exposed to unacceptable levels of radiation and/or radioactive materials.

Radiation survey data are usually obtained in units, such as the number of counts per unit time, that have no intrinsic meaning relative to DCGLs and guideline values. For comparison of survey data to DCGLs, the survey data from field and laboratory measurements should be converted to DCGL units.

Standard units used for expressing survey findings are:

- Surface contamination dpm/100 cm² (disintegrations per (alpha, beta-gamma) minute per 100 cm²)
- Soil radionuclide pCi/g (picocuries per gram) Concentration
- External exposure rate μ R/h (microrentgens per hour) for gamma only
- Shallow dose rate mrad/h (millirads per hour) or mGy/h (milligrays per hour)
- Dose equivalent rate μ rem/h (microrems per hour) or μ Sv/h (microsieverts per hour)

In performing the conversions for sections in this chapter in which factors are not specified, such as Section 9.1.1 D, it is necessary to know several factors; these are:

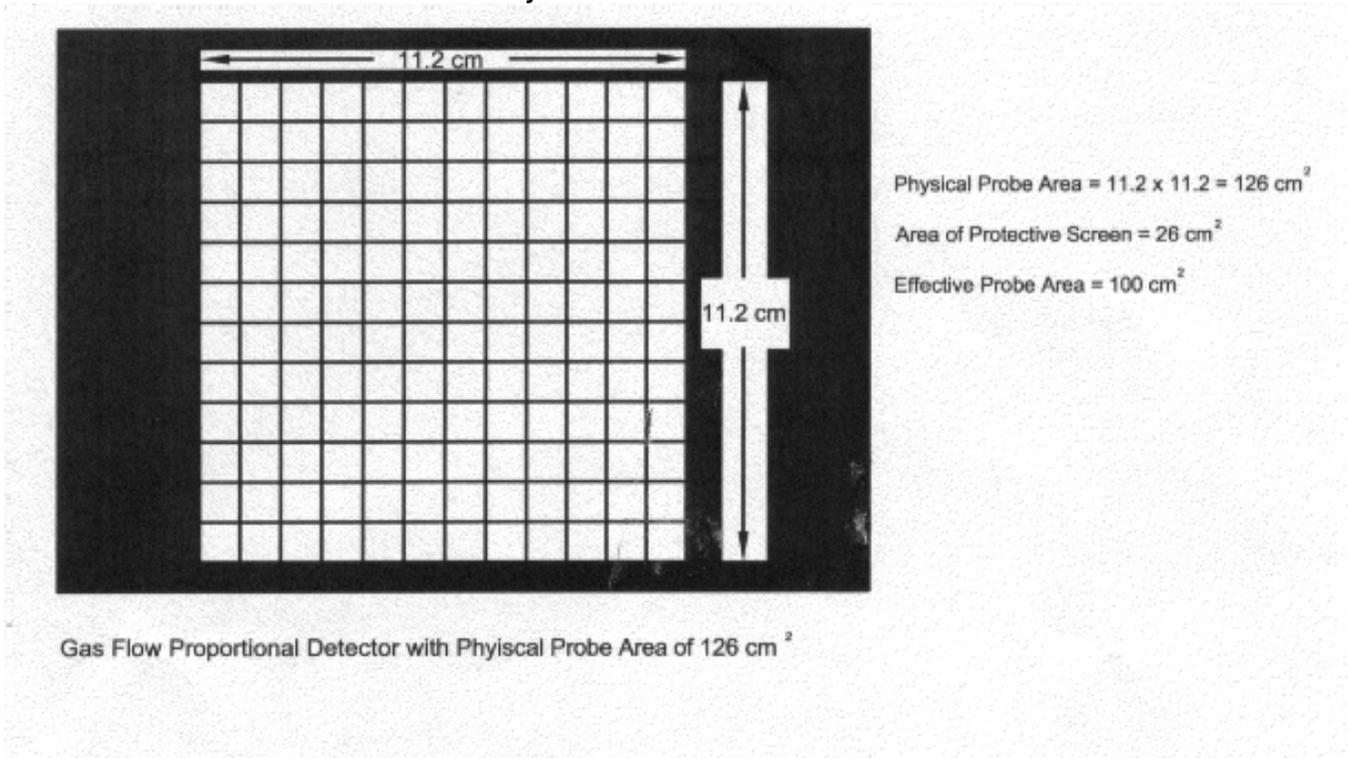
c	total integrated counts recorded by the measurement
c/m	total count rate from an analog (rate) instrument
t _g	time period (minutes) over which the gross count was recorded
t _b	time period (minutes) over which the background count was recorded
B	count during recording period, due only to background levels of radiation
B/m	background count rate
E	detection efficiency of instrument in counts per disintegration
A	active surface area of the detector in cm ²
M	mass of sample analyzed in grams (dry weight)
2.22	factor to convert a disintegration rate to activity units of picocuries, i.e. dpm/pCi.
0.0167	factor to convert dpm to Bq
0.037	factor to convert pCi to Bq
0.001	factor to convert g (grams) to kg (kilograms)
0.01	factor to convert mrad to mGy/h and μ rem/h to μ Sv/h
cf	combination of all other factors needed for converting measurements in c/m to standard reporting units

9.1.1 Surface Activity

- A. When measuring surface activity, it is important to account for the physical surface area assessed by the detector in order to make probe area corrections and report data in the proper units (i.e., Bq/m², dpm/100 cm²). This is termed the physical probe area. A common misuse is to make probe area corrections using the effective probe area which accounts for the amount of the physical probe area covered by a protective screen. Figure 5 illustrates the difference between the physical probe area and the effective probe area. The physical probe area is used because the reduced detector response due to the screen is accounted for during instrument calibration. For example, a Gas Flow Proportional Detector with physical probe area of 126 cm² (11.2 cm x 11.2 cm) has a protective screen area = 26 cm² and an effective probe area = 100 cm².

Care must be taken when calculating surface activity levels using a detector with a surface area differing from 100 cm². Generally, when the size of the contaminated region being measured is smaller than the probe area, the 100/A factors in the following equations should not be used. The 100/A correction factors are specifically included in these equations for measurements of areas larger than the probe size when the probe size is less than 100 cm². Probes with entrance window sizes greater than the maximum allowable averaging area used in the release criteria should not be used when making fixed point measurements. (Probe sizes larger than 100 cm² are recommended for use during scan surveys when the detection sensitivity is adequate.) For example, when evaluating the current 100 cm² surface contamination guideline, a probe area of around 100 cm² or less should be used. The use of a much larger probe size could result in an underestimate of activity within a small 100 cm² area beneath the probe since the remaining, less contaminated region surrounding the high activity spot will reduce the response of the detector. The response will accurately reflect the average contamination beneath the probe, but the smaller activity area, which may exceed the 100 cm² release limit, may go unnoticed.

Figure 5
The Physical Probe Area of a Detector



- B. The conversion of instrument display in counts to surface activity units is obtained using the following equation.

$$\frac{Bq}{m^2} = \frac{C_s}{T_s (\epsilon_T \times A)}$$

Eq. (9.1)

where:

- C_s = integrated counts recorded by the instrument
- T_s = time period over which the counts were recorded in seconds
- ε_T = total efficiency of the instrument in counts per disintegration, effectively the product of the instrument efficiency (ε_i) and the source efficiency (ε_s)
- A = physical probe area in m²

and to convert instrument counts to conventional surface activity units, the above equation can be modified to:

$$\frac{dpm}{100 \text{ cm}^2} = \frac{C_s}{T_s (\epsilon_T \times \frac{A}{100})}$$

Eq. (9.2)

where:

- T is recorded in minutes instead of seconds, and
- A is recorded in cm² instead of m².

- C. Some instruments have background counts associated with the operation of the instrument. A correction for instrument background can be included in the data conversion calculation as shown below. Note that the instrument background is not the same as the measurements in the background reference area used to perform the statistical tests described in MARSSIM Chapter 8.

$$\frac{B_q}{m^2} = \frac{\frac{C_s}{T_s} - \frac{C_b}{T_b}}{(\epsilon_T \times A)}$$

Eq. (9.3)

where:

- C_b = background counts recorded by the instrument
 T_b = time period over which the background counts were recorded in seconds

The above equation can be modified to provide conventional surface activity units as shown below:

$$\frac{\text{dpm}}{100 \text{ cm}^2} = \frac{\frac{C_s}{T_s} - \frac{C_b}{T_b}}{(\epsilon_T \times \frac{A}{100})}$$

Eq. (9.4)

where:

- T_s and T_b are recorded in minutes instead of seconds, and
 A is recorded in cm^2 instead of m^2 .

- D. If the measurement result is a count rate, i.e., in counts per minute, the conversion to $\text{dpm}/100 \text{ cm}^2$ is performed by:

$$\frac{\text{dpm}}{100 \text{ cm}^2} = \frac{\left(\frac{c}{m} - \frac{B}{m}\right) 100}{E \cdot A}$$

Eq. (9.5)

$$\frac{B_q}{\text{cm}^2} = \frac{\left(\frac{c}{m} - \frac{B}{m}\right) \left[\frac{100}{A}\right] \cdot [0.0167]}{E \cdot A}$$

Eq. (9.6)

- E. The level of removable activity collected by a smear is calculated generally in the same manner as for direct measurements, except, because the smear itself is performed over a 100 cm^2 area and the detector geometry correction is not considered when determining the efficiency, the detector area correction factor, $[100/A]$, is not necessary. (See Section 7.5 for detector area correction.) The equations for calculating removable activity are:

$$\frac{\text{dpm}}{100 \text{ cm}^2} = \frac{\left(\frac{c}{t_s} - \frac{B}{t_b}\right)}{E}$$

Eq. (9.7)

$$\frac{B_q}{\text{cm}^2} = \frac{\left(\frac{c}{t_s} - \frac{B}{t_b}\right)}{E \cdot 100} \cdot [0.0167] \quad \text{Eq. (9.8)}$$

F. The presence of multiple radionuclides at a site requires additional considerations for demonstrating compliance with a dose- or risk-based regulation. As demonstrated in MARSSIM Section 4.3.2, a gross activity DCGL should be determined.

For example, consider a site contaminated with ⁶⁰Co and ⁶³Ni, with ⁶⁰Co representing 60% of the total activity. The relative fractions are 0.6 for ⁶⁰Co and 0.4 for ⁶³Ni. If the DCGL for ⁶⁰Co is 8,300 Bq/m² (5,000 dpm/100 cm²) and the DCGL for ⁶³Ni is 12,000 Bq/m² (7,200 dpm/100 cm²), the gross activity DCGL is 9,500 Bq/m² (5,700 dpm/100 cm²) calculated using MARSSIM Equation 4-4.

When using the gross activity DCGL, it is important to use an appropriately weighted total efficiency to convert from instrument counts to surface activity units using MARSSIM Equations 6-1 through 6-4. In this example, the individual efficiencies for ⁶⁰Co and ⁶³Ni should be independently evaluated. The overall efficiency is then determined by weighting each individual efficiency by the relative fraction of each radionuclide.

9.1.2 Surface Dose Rates

A beta surface dose rate can be determined by taking a measurement with a beta-sensitive detector as long as the appropriate conversion factor is used. If the instrument display is in count rate (c/m), the conversion to mrad/h is given by:

$$\frac{\text{mrad}}{\text{h}} = \left(\frac{c}{m} - \frac{B}{m}\right)(cf) \quad \text{Eq. (9.9)}$$

$$\frac{\text{mGy}}{\text{h}} = \left(\frac{c}{m} - \frac{B}{m}\right)(cf)(0.01) \quad \text{Eq. (9.10)}$$

The conversion factor (cf) will vary for the different beta energies given by different isotopes. To determine a surface dose rate, the specific contaminant(s) and response of the detector being used must be known.

9.1.3 Soil Radionuclide Concentration and Exposure Rates

Analytical procedures, such as alpha and gamma spectrometry, are typically used to determine the radionuclide concentration in soil in units of Bq/kg. Net counts are converted to soil DCGL units by dividing by the time, detector or counter efficiency, mass or volume of the sample, and by the fractional recovery or yield of the chemistry procedure (if applicable). Refer to MARSSIM Chapter 7 for examples of analytical procedures.

Instruments, such as a PIC or micro-R meter, used to measure exposure rate typically read directly in μR/h. A gamma scintillation detector (e.g., NaI(Tl)) provides data in counts per minute and conversion to μR/h is accomplished by using site-specific calibration factors developed for the specific instrument (Section 6.7).

In situ gamma spectrometry data may require special analysis routines before the spectral data can be converted to soil concentration units or exposure rates.

A. Soil Radionuclide Concentration

To determine the radionuclide concentration in soil in units of pCi/g (dry weight) the calculation performed is:

$$\frac{\text{pCi}}{\text{g}} = \frac{\left(\frac{c}{t_g} - \frac{B}{t_b} \right)}{E \cdot 2.22 \cdot M} \quad \text{Eq. (9.11)}$$

$$\frac{B_{\text{d}}}{\text{kg}} = \frac{\left(\frac{c}{t_g} - \frac{B}{t_b} \right)}{t \cdot E \cdot M} \cdot [0.0167] \cdot \frac{1}{0.001} \quad \text{Eq. (9.12)}$$

It should be noted that concentrations in soil are to be expressed in terms of dry weight, and the soil will either require drying before analyses or a correction factor for moisture content applied in the calculation. If the analytical procedure includes a wet chemistry separation, it will also be necessary to correct for the fractional recovery (R), determined by a spike or tracer added to the sample.

$$\frac{\text{pCi}}{\text{g}} = \frac{\frac{C}{t_g} - \frac{B}{t_b}}{E \cdot 2.22 \cdot M \cdot R} \quad \text{Eq. (9.13)}$$

B. External Exposure Rate and Dose Equivalent Rate

If an instrument, such as a pressurized ionization chamber is used for measuring exposure rate, the instrument reading will be in the desired exposure rate units. Data in units of counts per minute or some preset time can be obtained by combining either a gamma scintillation or GM detector with one of two instruments; a count rate instrument or a digital scaling instrument. Conversion to exposure rate is accomplished using calibration factors developed for the specific instrument and survey site. It is possible that release criteria will be expressed in terms of dose equivalent rate ($\mu\text{rem/h}$ or $\mu\text{Sv/h}$). Tissue equivalent detectors are available which will allow direct measurement of dose equivalent rates, but at the time of this writing these instruments are generally not very stable in the 10 to 20 $\mu\text{rem/h}$ range. For purposes of measuring most environmental levels, one can assume that a direct gamma exposure of 1 μR is equivalent to 1 μrem . Given this assumption, pressurized ionization chamber measurements can be directly applied to dose equivalent rate comparisons. This net level is compared with the guideline value (cf is the site-specific calibration factor for the detector).

$$\mu \frac{R}{h} = \left(\frac{c}{m} \right) - \frac{B}{m} \cdot cf \quad \text{Eq. (9.14)}$$

$$\mu \frac{Sv}{h} = \frac{c}{m} \cdot cf \cdot 0.01 \quad \text{Eq. (9.15)}$$

$$\mu \frac{Rem}{h} = \frac{c}{m} \cdot cf$$

Eq. (9.16)

9.2 Measurement Uncertainty (Error)

The quality of measurement data will be directly impacted by the magnitude of the measurement uncertainty associated with it. Some uncertainties, such as statistical counting uncertainties, can be easily calculated from the count results using mathematical procedures. Evaluation of other sources of uncertainty require more effort and in some cases is not possible. For example, if an alpha measurement is made on a porous concrete surface, the observed instrument response when converted to units of activity will probably not exactly equal the true activity under the probe. Variations in the absorption properties of the surface for particulate radiation will vary from point to point and therefore will create some level of variation in the expected detection efficiency. This variability in the expected detector efficiency results in uncertainty in the final reported result. In addition, QC measurement results provide an estimate of random and systematic uncertainties associated with the measurement process.

The measurement uncertainty for every analytical result or series of results, such as for a measurement system, should be reported. This uncertainty, while not directly used for demonstrating compliance with the release criterion, is used for survey planning and data assessment throughout the Radiation Survey and Site Investigation (RSSI) process. In addition, the uncertainty is used for evaluating the performance of measurement systems using QC measurement results. Uncertainty can also be used for comparing individual measurements to the DCGL. This is especially important in the early stages of decommissioning (i.e., scoping, characterization, remedial action support) when decisions are made based on a limited number of measurements.

For most sites, evaluations of uncertainty associated with field measurements is important only for data being used as part of the final status survey documentation. The final status survey data, which is used to document the final radiological status of a site, should state the uncertainties associated with the measurements. Conversely, detailing the uncertainties associated with measurements made during scoping or characterization surveys may or may not be of value depending on what the data will be used for, i.e., the data quality objectives (DQOs). From a practical standpoint, if the observed data are obviously greater than the DCGL and will be eventually cleaned up, then the uncertainty may be relatively unimportant. Conversely, data collected during early phases of a site investigation that may eventually be used to show that the area is below the DCGL - and therefore does not require any clean-up action - will need the same uncertainty evaluation as the final status survey data. In summary, the level of effort needs to match the intended use of the data.

9.3 Systematic and Random Uncertainties

Measurement uncertainties are often broken into two sub-classes of uncertainty termed systematic (e.g., methodical) uncertainty and random (e.g., stochastic) uncertainty.

Systematic uncertainties derive from a lack of knowledge about the true distribution of values associated with a numerical parameter and result in data that is consistently higher (or lower) than the true value. An example of a systematic uncertainty would be the use of a fixed counting efficiency value even though it is known that the efficiency varies from measurement to measurement but without knowledge of the frequency. If the fixed counting efficiency value is higher than the true but unknown efficiency - as would be the case for an unrealistically optimistic value then every measurement result calculated using that efficiency would be biased low.

Random uncertainties refer to fluctuations associated with a known distribution of values. An example of a random uncertainty would be a well-documented chemical separation efficiency that is known to fluctuate with a regular pattern about a mean. A constant recovery value is used during calculations, but the true value is known to fluctuate from sample to sample with a fixed and known degree of variation.

To minimize the need for estimating potential sources of uncertainty, the sources of uncertainty themselves should be reduced to a minimal level by using practices such as:

- The detector used should minimize the potential uncertainty. For example, when making field surface activity measurements for ^{238}U on concrete, a beta detector such as a thin window Geiger-Mueller “pancake” probe may provide better quality data than an alpha detector depending on the circumstances. Less random uncertainty would be expected between measurements with a beta detector such as a pancake detector since beta emissions from the uranium will be affected much less by thin absorbent layers than will the alpha emissions.
- Calibration factors should accurately reflect the efficiency of a detector being used on the surface material being measured for the contaminant radionuclide or mixture of radionuclides (see Section 6.7). For most field measurements, variations in the counting efficiency on different types of materials will introduce the largest amount of uncertainty in the final result.
- Uncertainties should be reduced or eliminated by use of standardized measurement protocols (e.g., SOPs) when possible. Special effort should be made to reduce or eliminate systematic uncertainties, or uncertainties that are the same for every measurement simply due to an error in the process. If the systematic uncertainties are reduced to a negligible level, then the random uncertainties, or those uncertainties that occur on a somewhat statistical basis, can be dealt with more easily.
- Instrument operators should be trained and experienced with the instruments used to perform the measurements.
- QA/QC should be conducted as described in MARSSIM Chapter 9.

Uncertainties that cannot be eliminated need to be evaluated such that the effect can be understood and properly propagated into the final data and uncertainty estimates. As previously stated, non-statistical uncertainties should be minimized as much as possible through the use of good work practices.

Overall random uncertainty can be evaluated using the methods described in MARSSIM. Section 9.4 describes a method for calculating random counting uncertainty. Section 9.5 discusses how to combine this counting uncertainty with other uncertainties from the measurement process using uncertainty propagation.

Systematic uncertainty is derived from calibration errors, incorrect yields and efficiencies, non-representative survey designs, and blunders. It is difficult and sometimes impossible to evaluate the systematic uncertainty for a measurement process, but bounds should always be estimated and made small compared to the random uncertainty, if possible. If no other information on systematic uncertainty is available, use 16% as an estimate for systematic uncertainties (1% for blanks, 5% for baseline, and 10% for calibration factors).

9.4 Statistical Counting Uncertainty

When performing an analysis with a radiation detector, the result will have an uncertainty associated with it due to the statistical nature of radioactive decay. To calculate the total uncertainty associated with the counting process, both the background measurement uncertainty and the sample measurement uncertainty must be accounted for. The standard deviation of the net count rate, or the statistical counting uncertainty, can be calculated by

$$\sigma_n = \sqrt{\frac{C_{s+b}}{T_{s+b}^2} + \frac{C_b}{T_b^2}}$$

Eq. (9.17)

where:

- σ_n = standard deviation of the net count rate result
- C_{s+b} = number of gross counts (sample)
- T_{s+b} = gross count time
- C_b = number of background counts
- T_b = background count time

9.5 Uncertainty Propagation

Most measurement data will be converted to different units or otherwise included in a calculation to determine a final result. The standard deviation associated with the final result, or the total uncertainty, can then be calculated. Assuming that the individual uncertainties are relatively small, symmetric about zero, and independent of one another, then the total uncertainty for the final calculated result can be determined by solving the following partial differential equation (see MARSSIM, Section 6.8.3):

$$\sigma_u = \sqrt{\left(\frac{\partial u}{\partial x}\right)^2 \sigma_x^2 + \left(\frac{\partial u}{\partial y}\right)^2 \sigma_y^2 + \left(\frac{\partial u}{\partial z}\right)^2 \sigma_z^2 + \dots}$$

Eq. (9.18)

where:

- u = function, or formula, that defines the calculation of a final result as a function of the collected data. All variables in this equation, i.e., x, y, z..., are assumed to have a measurement uncertainty associated with them and do not include numerical constants
- σ_u = standard deviation, or uncertainty, associated with the final result
- $\sigma_x, \sigma_y, \dots$ = standard deviation, or uncertainty, associated with the parameters x, y, z, ...

The above equation, generally known as the error propagation formula, can be solved to determine the standard deviation of a final result from calculations involving measurement data and their associated uncertainties. The solutions for common calculations along with their uncertainty propagation formulas are included below.

Data Calculation

Uncertainty Propagation

u = x + y , or u = x - y :

$$\sigma_u = \sqrt{\sigma_x^2 + \sigma_y^2}$$

Eq. (9.19)

$$u = x \pm y, \text{ or } u = x \cdot y : \quad \sigma_u = u \sqrt{\left(\frac{\sigma_x}{x}\right)^2 + \left(\frac{\sigma_y}{y}\right)^2}$$

Eq. (9.20)

$$u = c + x, \text{ where } c \text{ is a positive constant: } \sigma_u = c\sigma_x$$

Eq. (9.21)

$$u = x \pm c, \text{ where } c \text{ is a positive constant: } \sigma_u = \frac{\sigma_x}{c}$$

Eq. (9.22)

NOTE: In the above examples, x and y are measurement values with associated standard deviations, or uncertainties, equal to σ_x and σ_y respectively. The symbol “ c ” is used to represent a numerical constant which has no associated uncertainty. The symbol σ_u is used to denote the standard deviation, or uncertainty, of the final calculated value u .

9.6 Reporting Confidence Intervals

Throughout Section 9.2, the term “measurement uncertainty” is used interchangeably with the term “standard deviation”. In this respect, the uncertainty is qualified as numerically identical to the standard deviation associated with a normally distributed range of values. When reporting a confidence interval for a value, one provides the range of values that represent a pre-determined level of confidence (i.e., 95%). To make this calculation, the final standard deviation, or total uncertainty σ_u as shown in the error propagation formula in section 9.5, is multiplied by a constant factor k representing the area under a normal curve as a function of the standard deviation. The values of k representing various intervals about a mean of normal distributions as a function of the standard deviation is given in Table 11. The following example illustrates the use of this factor in context with the propagation and reporting of uncertainty values.

**Table 11 (MARSSIM Table 6.9)
 Areas Under Various Intervals About the Mean of a Normal Distribution**

Interval ($\mu \pm k\sigma$)	Area
$\mu \pm 0.674\sigma$	0.500
$\mu \pm 1.00\sigma$	0.683
$\mu \pm 1.65\sigma$	0.900
$\mu \pm 1.96\sigma$	0.950
$\mu \pm 2.00\sigma$	0.954
$\mu \pm 2.58\sigma$	0.990
$\mu \pm 3.00\sigma$	0.997

Example:

Uncertainty Propagation and Confidence Interval: A measurement process with a zero background yields a count result of 28 ± 5 counts in 5 minutes, where the ± 5 counts represents one standard deviation about a mean value of 28 counts. The detection efficiency is 0.1 counts per disintegration ± 0.01 counts per disintegration, again representing one standard deviation about the mean.

Calculate the activity of the sample, in dpm, total measurement uncertainty, and the 95% confidence interval for the result.

9.6.1 The total number of disintegrations is:

$$\frac{28 \text{ counts}}{0.1 \frac{\text{c}}{\text{d}}} = 280$$

9.6.2 Using the equation for error propagation for division, total uncertainty is:

$$280 \sqrt{\left(\frac{5}{28}\right)^2 + \left(0. \frac{01}{0} .1\right)^2} = 57 \text{ disintegrations}$$

9.6.3 The activity will then be $280 \div 5$ minutes = 56 dpm and the total uncertainty will be $57 \div 5$ minutes = 11 dpm. (Since the count time is considered to have trivial variance, this is assumed to be a constant.)

Referring to Table 11, a k value of ± 1.96 represents a confidence interval equal to 95% about the mean of a normal distribution. Therefore, the 95% confidence interval would be 1.96×11 dpm = 22 dpm. The final result would be 56 ± 22 dpm.

9.7 Detection Sensitivities

For the purposes of thorough data presentation, the detection sensitivity for each measurement procedure (and each instrument if more than one instrument is used for a given procedure) is calculated. Data from surveys will often be near background levels and/or may not be discernable from background. Many measurements near background levels may be at or below the critical detection level of the measurement equipment being used.

If any measurement below the mean background is to be included in an averaging process to determine a mean value for an area or unit in a final status survey, the value should be used as measured, i.e., both positive and negative values should be used. Negative data will be a frequent result of calculations. Use of the MDA or critical level for data that have a value less than the detection capability is a common practice accepted by the EPA. This approach enables the surveyor to significantly reduce the number of calculations; however, use of a detection capability in place of actual data when calculating averages will bias the results on the high side; and the true conditions of the site will not be described. Substituting detection sensitivities for actual data will also result in overestimates of source inventory and dose assessments, possibly leading to decisions for further actions that may not be justified. Finally, when evaluating data distributions (e.g., in a normalcy test), use of detection sensitivity values will result in a skewed distribution and may lead to incorrect conclusions. To avoid the pitfalls associated with use of detection sensitivity values, it is recommended that actual data be presented and used for calculational purposes. One exception to this approach might be the use of such values for averaging site activity levels when the detection sensitivity is small in comparison to the applicable guideline. For the purposes of the DOE EIGRSP small may be considered as less than 10% of the guideline value.

10.0 RADIOACTIVE CONTAMINATION CONTROL - RELEASE CRITERIA

10.1 Title 10, CFR, Part 835 Requirements

The following requirements apply to the control of material and equipment:

- 10.1.1 Except as provided in 10.1.1.A and 10.1.1.B (**10 CFR 835.1101(b) and (c)**) material and equipment in contamination areas, high contamination areas, and airborne radioactivity areas shall not be released to a controlled area if (**10 CFR 835.1101(a)**):
- A. Removable surface contamination levels on accessible surfaces exceed the removable surface contamination values specified in Title 10, CFR, Part 835, Appendix D (**10 CFR 835.1101(a)(1)**); or
 - B. **Prior** use suggests that the removable surface contamination levels on inaccessible surfaces are likely to exceed the removable surface contamination values specified in Title 10, CFR, Part 835, Appendix D (**10 CFR 835.1101(a)(2)**).
- 10.1.2 Material and equipment exceeding the removable surface contamination values specified in Title 10, CFR, Part 835, Appendix D may be conditionally released for movement on-site from one radiological area for immediate placement in another radiological area only if appropriate monitoring is performed and appropriate controls for the movement are established and exercised (**10 CFR 835.1101(b)**).
- 10.1.3 Material and equipment with fixed contamination levels that exceed the total surface contamination values specified in Title 10, CFR, Part 835, Appendix D may be released for use in controlled areas outside of radiological areas only under the following conditions:
- A. Removable surface contamination levels are below the removable surface contamination values specified in Title 10, CFR, Part 835, Appendix D (**10 CFR 835.1101(c)(1)**); and
 - B. The material or equipment is routinely monitored and clearly marked or labeled to alert personnel of the contaminated status (**10 CFR 835.1101(c)(2)**).
- 10.1.4 The following information shall be documented and maintained (**10 CFR 835.703(a) and (c)**):
- A. Results of monitoring for radiation and radioactive material as required by 10 CFR 835 Subpart E "Monitoring of Individuals and Areas," and 10 CFR 835 Subpart L Radioactive Contamination Control," except for monitoring of individuals exiting contamination, high contamination, or airborne radioactivity areas, as required by **10 CFR 835.1102(d) (10 CFR 835.1101(c) and 10 CFR 835.703(c))**.
 - B. Results of monitoring for the release and control of material and equipment, as required by 10 CFR 835.1101, "Control of material and equipment" (**10 CFR 835.1101(c) and 10 CFR 835.703(c)**).

10.2 DOE Radiological Control Standard Requirements

Once materials and equipment have entered radiological areas controlled for surface contamination or airborne radioactivity, comprehensive and time-consuming evaluations of the potential for contamination are required prior to releasing the material or equipment to controlled areas. Likewise, exposure of certain materials and equipment to a beam of neutrons or other particles produced in a nuclear reactor or particle accelerator may result in activation of that material or equipment, resulting in the creation of radioactive material requiring controlled use, storage, and disposal. The need for evaluation of the radiological characteristics of these materials

and equipment and implementation of appropriate controls provides substantial impetus for implementation of measures to limit the amount of material and equipment that enters radiological areas and to prevent contamination or activation of materials and equipment that do enter these areas.

- 10.2.1 Accessible surfaces of material or equipment that has entered contamination, high contamination, or airborne radioactivity areas shall be surveyed prior to release from these areas to controlled areas. Guidance for conducting these surveys is provided in the footnotes to DOE-STD-1098-2008, Table 2-2 (**DOE-STD-1098-2008, Article 421.1**).
- 10.2.2 If an assessment of the prior use of the material or equipment indicates that inaccessible surfaces are not likely to be contaminated in excess of applicable limits, a complete survey of accessible surfaces and documentation of the assessment may be an appropriate basis to release materials to the controlled area (**DOE-STD-1098-2008, Article 421.2**).
- 10.2.3 If an assessment of the prior use of the material or equipment indicates that inaccessible surfaces are likely to be contaminated to levels in excess of the DOE-STD-1098-2008, Table 2-2 values, then the material shall not be released from the radiological area, except as permitted under DOE-STD-1098-2008, Article 421.5 or 421.6. If it is necessary to release the material or equipment from the radiological area, the material or equipment should be disassembled to the extent necessary to perform adequate surveys (**DOE-STD-1098-2008, Article 421.3**).
- 10.2.4 Removable contamination levels shall be less than DOE-STD-1098-2008, Table 2-2 values prior to releasing material and equipment for unrestricted use in controlled areas (**DOE-STD-1098-2008, Article 421.4**).
- 10.2.5 Material and equipment with fixed contamination levels that exceed the total contamination values specified in DOE-STD-1098-2008, Table 2-2, and removable contamination levels less than Table 2-2 values, may be released for restricted use in controlled areas outside of radiological areas. The material or equipment shall be routinely monitored and clearly marked or labeled to alert individuals to the contaminated status. Written procedures should be developed to establish requirements for monitoring of the material or equipment and surrounding areas, control of access to these areas, authorized uses of the material or equipment, and contingency plans for spread of radioactive contamination (**DOE-STD-1098-2008, Article 421.5**).
- 10.2.6 Material and equipment with total or removable contamination levels exceeding DOE-STD-1098-2008, Table 2-2 values may be moved on site from one radiological area to another if appropriate monitoring is performed and appropriate controls are established and implemented. These controls should include provisions for containment to the extent practicable, labeling in accordance with DOE-STD-1098-2008, Article 412, monitoring and control of the transfer route and participating individuals, and control of spills (**DOE-STD-1098-2008, Article 421.6**).
- 10.2.7 The requirements of 10 CFR 835.1101 apply only to material and equipment that is radioactive due to the deposition of radioactive contamination. DOE has not established any specific controls over the release of other radioactive materials (e.g., materials that are naturally-radioactive) to controlled areas, the release of these materials is subject to other requirements of 10 CFR 835. The following regulatory requirements are applicable to the release of this type of material. (**DOE-STD-1098-2008, Article 421.7**).
 - A. Controls shall be adequate to ensure compliance with the Radiological Controls training requirements of 10 CFR 835.901. Release of material and equipment to controlled areas may result in occupational or non-occupational exposure of individuals to radiation. DOE-STD-1098-2008, Chapter 6 provides guidance for implementing an appropriate training program.

- B. Controls shall be adequate to ensure compliance with the 100 millirem in a year controlled area maximum total effective dose equivalent expectation. DOE sites should adopt site- or facility-specific criteria that will ensure that intrinsically-radioactive material and equipment that is released to the controlled area, in combination with other sources of radiation in the controlled area, will not result in any individual exceeding this dose expectation.
 - C. Controls shall be adequate to ensure the ALARA process is properly implemented. Given the low levels of radioactivity that are likely to be present in material and equipment being considered for release to controlled areas, the controls should not be burdensome. Options that should be considered include retention in radiological areas, placement in specified areas with appropriate access restrictions and usage controls, posting, labeling or color-coding, storage for decay, removal of radioactive components, and disposal as radioactive waste.
- 10.2.8 When radioactive materials are moved outside of radiological areas, controls should be established to ensure no unmonitored individual is likely to exceed a dose equivalent that would require monitoring in accordance with DOE-STD-1098-2008, Article 511 or 521 (**DOE-STD-1098-2008, Article 421.8**).
- 10.2.9 Records for release of materials should describe the property, date of last survey, identity of the individual who performed the survey, type and identification number of the survey instruments used, and survey results. For small items and packages of similar items (such as boxes of tools or boxes of fasteners), it is not necessary to create a separate survey record for each item. However, the survey record should provide traceability to the individual removing the item from the radiological area (**DOE-STD-1098-2008, Article 421.9**).
- 10.2.10 Material not immediately released after survey shall be controlled to prevent contamination while awaiting release (**DOE-STD-1098-2008, Article 422.4**).
- 10.3 DOE O 458.1 Requirements
- 10.3.1 Release of real property (land and structures) shall be in accordance with the guidelines and requirements for residual radioactive material presented in DOE O 458.1, Attachment 1. These guidelines and requirements apply to both DOE-owned facilities and to private properties that are being prepared by DOE for release. Real properties owned by DOE that are being sold to the public are subject to the requirements of Section 120(h) of the Comprehensive Environmental Response Compensation and Liability Act (CERCLA), as amended, concerning hazardous substances, and to any other applicable Federal, State, and local requirements (**DOE O 458.1, Section 2.k.7(a)**).
 - 10.3.2 Personal property, which potentially could be contaminated, may be released for unrestricted use if the results of a survey with appropriate instruments indicate that the property is within the contamination limits presented in (**DOE O 458.1, Section 2.k.5.(d)**).
 - 10.3.3 For the release of materials and equipment, the following requirements shall be applied (**DOE O 458.1, Section 2.k.**):
 - A. Prior to materials and equipment being released, property shall be surveyed to determine whether both removable and total surface contamination (including contamination present on and under any coating) is greater than the levels given in DOE O 458.1, Section 2.k.5.(d), and that the contamination has been subjected to the ALARA process.

- B. Property shall be considered to be potentially contaminated if it has been used or stored in radiological areas that could contain unconfined radioactive material or that are exposed to beams of particles capable of causing activation (neutrons, protons, etc.).
- C. Surfaces of potentially contaminated property shall be surveyed using instruments and techniques appropriate for detecting the limits stated in DOE O 458.1, Section 2.k.3.(b)(4).
- D. Where potentially contaminated surfaces are not accessible for measurement (as in some pipes, drains, and ductwork), such property may be released after case-by-case evaluation and documentation based on both the history of its use and available measurements demonstrate that the nonsurveyable surfaces are likely to be within the limits given in DOE O 458.1, Section 2.k.
- E. The records of released property shall include:
 - 1. A description or identification of the property;
 - 2. The date of the last radiation survey;
 - 3. The identity of the organization and the individual who performed the monitoring operation;
 - 4. The type and identification number of monitoring instruments;
 - 5. The results of the monitoring operation; and
 - 6. The identity of the recipient of the released material.
- F. No guidance is currently available for release of material that has been contaminated in depth, such as activated material or smelted contaminated metals (e.g., radioactivity per unit volume or per unit mass). Such materials may be released if criteria and survey techniques are approved by DOE EH-1.

10.3.4 Guidelines for residual concentrations of radionuclides in soil for unrestricted release, other than radium and thorium, shall be derived from the basic dose limits by means of an environmental pathway analysis using specific property data where available. Procedures for these derivations are given in DOE/CH-8901, "A Manual for Implementing Residual Radioactive Material Guidelines, A Supplement to the U.S. Department of Energy Guidelines for Residual Radioactive Material at FUSRAP and SFMP Sites," June 1989 (**DOE O 458.1, Section 2.k.3**).

- A. Residual concentrations of radioactive material in soil are defined as those in excess of background concentrations averaged over an area of 100 square meters and in 15 centimeter thick layers or increments.
- B. For Hot Spots, if the average concentration in any surface or below-surface area less than or equal to 25 m², exceeds the limit or guideline by a factor of $(100/A)^{1/2}$, [where A is the area (in square meters) of the region in which concentrations are elevated], limits for "hot-spots" shall also be developed and applied. Procedures for calculating these hot-spot limits, which depend on the extent of the elevated local concentrations, are given in DOE/CH-8901. In addition, reasonable efforts shall be made to remove any source of radionuclide that exceeds 30 times the appropriate limit for soil, irrespective of the average concentration in the soil (**DOE O 458.1, Section 2.k.6.(f)**).
- C. Information on applications of the guidelines and requirements, including procedures for deriving specific property guidelines for allowable levels of residual radioactive material from basic dose limits, is contained in DOE/CH-8901 (**DOE O 458.1, Section 2.k.6.(a)**).

- D. Residual concentrations of radionuclides in soil, for these purposes, is defined as unconsolidated earth material, including rubble and debris that might be present in earth material.
- E. Specific property guidelines are derived from basic dose limits using specific property models and data. Procedures and data for deriving specific property guideline values are given by DOE/CH-8901. The basis for the guidelines is generally a presumed worst-case plausible-use scenario for the property (**DOE O 458.1, Sections 2.k and 2.k.6.(c).(3)**).

10.4 Application of Derived Concentration Guideline Levels for Soil Activity Correlations

An investigation was conducted to determine if Cs-137 could be used as a target radionuclide when performing in-situ measurements of contaminated soil with a hand-held NaI scintillation detector operated as a count rate instrument. Derived concentration guideline levels (DCGLs), see WVDP-304, and location-specific Cs-137 surrogate values were developed based on site characterization data. Cs-137 was chosen as the target radionuclide because of the relative ease of detecting the associated 662 keV gamma with field instrumentation and the relatively high abundance of Cs-137 as compared to other gamma-emitting radionuclides at the WVDP.

As part of the WVDP's Contaminated Soil Management Program, contaminated soil samples were obtained and analyzed in early 1998. Results of the soil analysis produced characterization information including Cs-137 content ($\mu\text{Ci/g}$), percent moisture content, etc. These soil samples represented 14 distinct areas of the site. Residual soil from the samples was labeled, containerized, and stored. Fifty-seven direct radiation measurements were taken on samples of this previously characterized soil using a shielded NaI detector. The results were graphed to display net counts per minute (ncpm) versus volumetric activity of Cs-137 in the first 15 centimeters (six inches) depth of soil. Calculations produced the linear regression line describing the relationship between ncpm (the x variable) and the concentration of Cs-137 (the y variable) in pico curies per gram (pCi/g). Biased in-situ gamma measurements and soil sampling were performed to validate the correlation line. The validation regression line produced values that remained within $\pm 10\%$ of the correlation line. Statistics were applied to the correlation line producing a final correlation equation of:

$$\text{Cs-137(pCi/g)} = (0.0215 \text{ ncpm}) + 19.6, \text{ representing the } + 95\% \text{ confidence level}$$

Location specific surrogate values were developed for Cs-137 using site characterization information and defined DCGLs.

If Cs-137 was the only radioisotope in WVDP soil, the correlation equation developed could be directly applied to establish an instrument reading where soil containerization would be required. Because soil at the WVDP contains a host of radioisotopes that do not emit gamma radiation, a Cs-137 surrogate value was developed to account for the other isotopes. The Cs-137 surrogate DCGLs (D_{total}) were calculated using the site-specific DCGLs contained in Table 12. Cs-137 surrogates including correlated ncpm values are listed in Table 13.

With the development of NaI instrument correlation and surrogate values, Radiological Controls personnel can use a NaI detector in the field to determine if soil meets or exceeds the containerization criteria. The ability to make these field decisions will improve efficiency and provide WVNSCO with more options for managing contaminated soil.

Table 12
Site-Specific Soil Containerization DCGLs

Nuclide	DCGL (pCi/g)						
H-3	4500	Sb-125	1600	Ra-228	210	Pu-241	4500
C-14	4500	I-129	4500	U-232	210	Pu-242	900
Fe-14	4500	Cs-134	350	U-233	900	Np-237	900
Ni-59	4500	Cs-135	4500	U-234	900	Am-241	900
Ni-63	4500	Cs-137	710	U-235	900	Am-243	900
Co-60	170	Pm-147	4500	U-236	900	Cm-243	900
Nb-94	350	Eu-152	350	U-238	900	Cm-244	900
Sr-90	4500	Eu-154	330	Pu-238	900		
Tc-99	4500	Eu-155	4500	Pu-239	900		
Ru-106	3800	Ra-226	210	Pu-240	900		

Table 13
Location Specific Cs-137 Surrogate Values

Location	Cs-137 surrogate (pCi/g)	Cs-137 surrogate (ncpm)
LLWTF	224	9525
EDR	276	11915
NDA rolloffs	309	13451
VIT	361	15887
NDA S-70	382	16871
WTF	449	19952
SPR	466	20754
UR	490	21859
MNT	503	22469
LAG	557	24988
NW	595	26768
NDA B-25	616	27729
CSS	658	29699
VTF	661	29833

11.0 REFERENCES

- 11.1 American National Standard ANSI/HPS N13.30-1996, "Performance Criteria for Radiobioassay."
- 11.2 American National Standard ANSI/IEEE N323B-2003, "Radiation Safety Instrumentation Test and Calibration, Portable Survey Instruments."
- 11.3 U.S. Department of Energy, DOE O 458.1, "Radiation Safety for the General Public and the Environment," Change 1, March 2011.
- 11.4 U.S. Department of Energy, Standard, DOE-STD-1098-2008, "Radiological Control."
- 11.5 U.S. Department of Energy, DOE/CH-8901, "A Manual for Implementing Residual Radioactive Material Guidelines, A Supplement to the U.S. Department of Energy Guidelines for Residual Radioactive Material at FUSRAP and SFMP Sites," June 1989.
- 11.6 International Standard for Standardization ISO-7503-1, "Evaluation of Surface Contamination - Part 1: Beta Emitters and Alpha Emitters," 1988.
- 11.7 U.S. Nuclear Regulatory Commission, NUREG-1501, "Background as a Residual Radioactivity Criterion for Decommissioning," 1994.
- 11.8 U.S. Nuclear Regulatory Commission, NUREG-1505, "A Proposed Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," 1997.
- 11.9 U.S. Nuclear Regulatory Commission, NUREG-1506, "Proposed Methodologies for Measuring Low Levels of Residual Radioactivity for Decommissioning," 1995.
- 11.10 U.S. Nuclear Regulatory Commission, NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," 1997.
- 11.11 U.S. Nuclear Regulatory Commission, NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual," December 1997.
- 11.12 U.S. Nuclear Regulatory Commission, NUREG/CR-4007, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," 1984.
- 11.13 U.S. Nuclear Regulatory Commission, NUREG/CR-6364, "Human Performance in Radiological Survey Scanning," 1997.
- 11.14 U.S. Department of Energy, Title 10, Code of Federal Regulations, Part 835, "Occupational Radiation Protection.
- 11.15 U.S. Environmental Protection Agency, Title 40, Code of Federal Regulations, Part 192, "Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings," 1994.
- 11.16 U.S. Environmental Protection Agency, Title 40, Code of Federal Regulations, Part 300, "Hazard Ranking System," 1990.
- 11.17 WVDP-010, "WVDP Radiological Controls Manual," (latest revision).
- 11.18 WVDP-302, "Technical Basis for Contaminated Soil Management"

WVDP RECORD OF REVISION

Rev. No.	Description of Changes	Revision On Page(s)	Dated
0	Original Issue	All	08/24/95
FC1	2.1 - Deleted DOE Order 5480.11 11.0 - Deleted paragraph	2 56	12/04/98
1	General Revision Changes from 10 CFR 835 Amendment were incorporated	All	11/04/99
FC1	Section 6.3.2, revised step for reporting detection results Section 10.2, revised to implement requirements from DOE-STD-1098-99 Section 11.0, revised reference to DOE RCS Document repaginated and formatted for EDOCs	19 56-59 61 All	12/19/00
2	General Revision Changes from NUREG-1575, "MARSSIM," dated December 1997 and DOE "EIGRSP," dated February 1997 were incorporated.	All	02/21/01
FC1	Table of Contents, Add line for section 10.4 Table of Contents, Add lines 12 & 13 in References Section 4.1, Revised Title Section 4.2, Revised Title Section 10.4, Add new text and table Section 11.0, Add two references Corrected cover page to update form The Radiation Protection department is affected by these changes	3 3 12 13 101-103 105	05/20/02
3	Minor Revision Updated manager and organization titles, acronyms, and references throughout. Radiation Protection organization is affected	All	10/04/05
4	Minor Revision Updated cover page, manager and organization titles, acronyms, and references throughout. Radiological Controls organization is affected	All	12/28/07
5	Converted WordPerfect to Word Revised reference to Standard, DOE-STD-1098-2008 Revised reference to 10, CFR, Part 835 Radiological Controls organization is affected	All	03/03/11
6	General Revision Updated cover page, organization titles, acronyms, and references throughout. Radiological Controls organization is affected	All	03/28/12