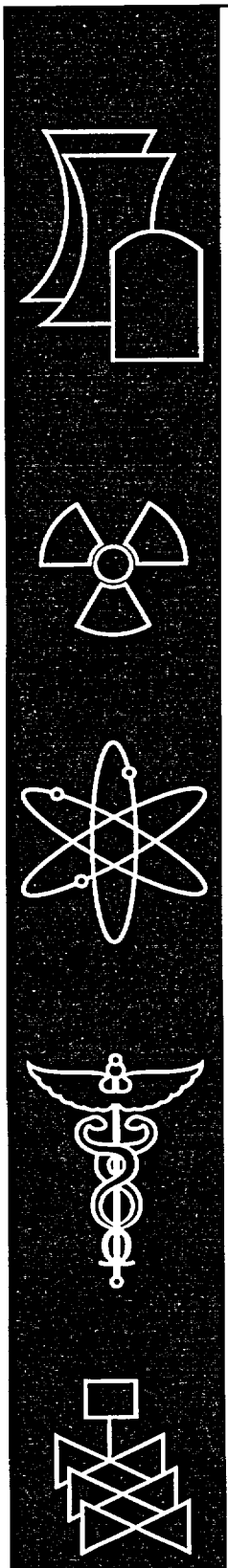
NUREG-1761



Radiological Surveys for Controlling Release of Solid Materials

Draft Report for Comment

Oak Ridge Institute for Science and Education

Oak Ridge National Laboratory

U.S. Department of Energy

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

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NUREG-1761

Radiological Surveys for Controlling Release of Solid Materials

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is developing a basis to support decisions on whether to undertake a rulemaking that would set specific requirements on controlling licensees' releases of solid materials. Specifically, the solid materials being evaluated include metals, building concrete, onsite soils, equipment, furniture, etc., which are present at, and/or used in, licensed nuclear facilities during routine operations. Historically, licensees have released solid materials on a case-by-case basis, without a consistent approach to designing and conducting clearance surveys. This draft report provides information about measuring residual radioactivity in materials that are to be cleared from nuclear facilities, including guidance about designing, performing, and documenting radiological surveys of solid materials to address the need for consistency in the surveys.

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) is developing a basis to support decisions on whether to undertake a rulemaking that would set specific requirements on controlling licensees' releases of solid materials. Specifically, the solid materials being evaluated include metals, building concrete, onsite soils, equipment, furniture, etc., which are present at, and/or used in, licensed nuclear facilities during routine operations. Historically, licensees have released solid materials on a case-by-case basis, without a consistent approach to designing and conducting clearance surveys. This document provides guidance on designing, performing, and documenting surveys of solid materials to address the need for consistency in the surveys. For convenience, Section 2 provides a roadmap, or flow diagram, of the survey process described in this report.

The Data Quality Objectives (DQO) Process (discussed in Section 3) is the foundation for designing and implementing surveys of solid materials. However, before beginning to plan for the survey, the licensee must decide whether to dispose of the solid material as radioactive waste or perform surveys to determine whether the material can be released. That is, it may be more cost-effective to simply dispose of the material as radioactive waste, rather than performing clearance surveys. In general, solid materials that have a limited potential to be contaminated would likely be surveyed for clearance, while those materials that are known (or likely) to have contamination in excess of the release criteria, which would therefore require cleaning and reevaluation prior to release, would probably be disposed of as radioactive waste.

After determining that clearance is the preferred option, the licensee would use the DQO Process to determine the most advantageous survey protocol based on the solid material being released (Section 4.2), the available survey instrumentation, the need for laboratory analyses, and the applicable release criteria. Effective survey design should consider the available process knowledge of the solid materials and the need for additional characterization of the material (Section 4.3). Characteristics that impact the release of solid materials include their physical description, potential for contamination (Section 4.4), nature of the contamination, and degree of inaccessible areas (Section 4.7).

It should be noted that this report does not provide release criteria, but does presume that criteria have been obtained prior to survey design (Section 4.1). Specifically, this report assumes that derived concentration guideline levels for clearance ($DCGL_C$) are available for use, and focuses on how those release criteria can be applied when multiple radionuclides may be present (Section 4.5).

This report describes a number of different survey approaches, including conventional scanning, automated scanning using a conveyorized survey monitor, and *in toto* techniques, such as *in situ* gamma spectrometry and tool monitors. In addition, because detection limits for survey instrumentation are an important criterion for selecting a particular approach, this report addresses the measurement of contamination (Section 4.6) for each survey approach considered. This report also stresses the use of situation-specific measurement sensitivity of scanning to release solid materials whenever the minimum detectable concentration (MDC) of the scan is less than the $DCGL_C$. Statistical survey designs, such as those discussed in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM), Rev. 1, are recommended in cases where the scan MDC is greater than the $DCGL_C$. [Note: Appendix A provides a primer on the basic radiation properties, which are relevant to the measurement of radioactivity in and on solid materials. It also addresses some of the fundamental principles of radiation detection and measurements.]

Survey approaches (discussed in Section 5) were determined using the DQO Process, giving due consideration to two major requirements. Specifically, (1) the survey result must be able to demonstrate that clearance criteria have been met within predetermined confidence levels, and (2) the survey unit size must be sufficiently evaluated to develop a technically defensible approach for area or volume averaging.

The general release survey approaches identified in Section 5 include (1) surveys using conventional instruments that incorporate both scanning and statistical designs for determining sample sizes; (2) automated scanning surveys (conveyorized survey monitors); (3) *in toto* surveys performed using gamma spectrometers, bag monitors, tool monitors, and portal monitors; and (4) analytical methods and laboratory analyses on representative samples based on statistical sampling designs. Section 6 provides guidance on reducing survey data, demonstrating compliance with clearance release criteria, and documenting results. Appendix B provides additional information on advancements in general radiation detectors and survey instruments that utilize new detection materials and software.

FOREWORD

This report provides technical information on conducting radiation surveys of solid materials at nuclear facilities.

NRC Examination of its Approach for Controlling the Release of Solid Material

On June 30, 1999, the NRC published, for public comment, an issues paper indicating that the agency was examining its approach for control of solid material. The issues paper presented alternative courses of action for controlling the release of solid materials that have very low amounts of, or no, radioactivity.

In August 2000, the Commission decided to defer its final decision on whether to proceed with rulemaking on controlling the release of solid materials while it requested a study by the National Academies on possible alternatives for controlling the release of slightly contaminated materials. While the National Academies' study was ongoing, the Commission directed its staff to continue developing the technical information base that the Commission needed to support a policy decision in this area.

As part of this decisionmaking, it is useful to have information on methods that could be used to perform radiation surveys to control the release of solid material. The alternatives described in the June 1999 issues paper were to (1) continue current practice (without a rulemaking) and (2) issue a proposed rule to establish a standard. If the Commission were to develop a rule, rulemaking alternatives in the issues paper were to (1) permit release of material for unrestricted use if it meets certain dose levels, (2) prohibit release of material that had been in an area in a licensed facility where radioactive material was used or stored, and (3) restrict release to only certain authorized uses. For any of the alternatives, a radiological survey is necessary in order to ensure that the criteria are implemented appropriately. The extent of the survey needed depends on the alternative chosen by the Commission to ensure protection of public health and safety.

This report evaluates methods available at the time of its creation for conducting radiological surveys of material at NRC-licensed facilities for the various alternatives.

Further Development of Use of the Data Quality Objectives Process

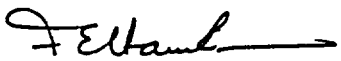
During the 1990s, the NRC and the industry made a concerted effort to improve the planning, conduct, evaluation, and documentation of final radiological surveys of building surfaces and surface soil to demonstrate compliance with established standards. This effort included preparing NUREGs-1505 and 1507 and culminated in 1997 with the issuance of NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM), as a result of a joint effort by the NRC, U.S. Environmental Protection Agency (EPA), U.S. Department of Defense (DOD), and U.S. Department of Energy (DOE) to develop a consistent approach for planning, performing, and assessing the ability of surveys to meet standards, while encouraging effective use of resources. The MARSSIM provides guidance on developing appropriate survey designs using the Data Quality Objectives (DQO) Process to ensure that survey results are of sufficient quality and quantity to support a final decision. The MARSSIM and NUREG reports replaced the previous approach for such surveys contained in NUREG/CR-5849.

This report provides technical information with regard to extending the DQO Process to issues concerning controlling the release of solid materials, and specifically to the design and implementation of surveys for these materials. This information is important to ensure protection of public health and safety. In particular, this information is important to ensure that materials being released meet the established standard.

Scope and Approach of this Report

This report provides technical information on survey approaches for a range of possible alternatives for controlling the release of solid material. It provides information on surveys associated with options where material would not be released, as well as surveys for a range of nuclide concentrations for options where material would be released. In so doing, it discusses the need for increased survey complexity as allowable material levels decrease to allow for the ability to distinguish actual residual radioactivity levels in solids against background.

The alternative of not permitting material to be released if it is located in an area where radioactive materials are used or stored, referred to in the issues paper as "prohibition," would rely principally on process knowledge of where the material originated because it would use that information as a basis for determining disposition of the material. Information on process knowledge is presented in Section 4.3 of this report. This alternative would not be as dependent upon detailed methods for radiological surveys and, thus, much of the information in later sections of this report would not apply to this alternative. The alternatives of continuing current practice or permitting release using dose-based criteria rely upon process knowledge of where the solid materials originated in the facility, as well as comprehensive radiological surveys to demonstrate that the level of radioactivity on the material would meet the required criteria. Information on various survey methodologies is presented in Section 5. The alternative of restricted use may use process knowledge to determine those materials that would be limited to authorized uses, but may be similar to unrestricted use in the need for comprehensive surveys.



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ABBREVIATIONS AND SYMBOLS

ADP	automated data processing
AEC	Atomic Energy Commission
ALARA	as low as is reasonably achievable
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CSM	conveyorized survey monitor
CSS	Compton suppression spectrometer
d'	detectability index
D&D	decontamination and decommissioning
$DCGL_C$	derived concentration guideline level for clearance
dpm	disintegrations per minute
DQA	data quality assessment
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DQI	data quality indicator
DQO	data quality objective(s)
ε_i	instrument efficiency
ε_s	surface efficiency
EIC	electret ion chamber
EML	Environmental Measurements Laboratory
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FIDLER	Field Instrument for the Detection of Low-Energy Radiation
FPXRF	field-portable x-ray flourescence
FWHM	full width at half maximum
GDP	gaseous diffusion plant
GeLi	germanium-lithium
GM	Geiger-Mueller
HPGe	high-purity germanium (detector)
HVT	half-value thickness
i	observation interval
ICP-MS	inductively coupled plasma mass spectrometer
ISGS	*in situ* gamma spectrometry
ISO	International Organization for Standardization
IUPAC	International Union of Pure and Applied Chemistry
LBGR	lower bound of the gray region
LN	liquid nitrogen
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCNP	Monte Carlo N-Particle
MDA	minimum detectable activity
MDC	minimum detectable concentration
MDCR	minimum detectable count rate
MFP	mean-free-path
MQC	minimum quantifiable concentration

392	NaI	sodium iodide
393	NCRP	National Council on Radiation Protection and Measurements
394	NDA	nondestructive assay
395	NIST	National Institute of Standards and Technology
396	NRC	U.S. Nuclear Regulatory Commission
397	ORISE	Oak Ridge Institute for Science and Education
398	ORNL	Oak Ridge National Laboratory
399	p	surveyor efficiency
400	PARCC	precision, accuracy (bias), representativeness, comparability, and completeness
401	PE	performance evaluation
402	PGT	Princeton Gamma Tech
403	PMT	photomultiplier tube
404	ppq	part per quintillion (one part per 10^{18})
405	QA	quality assurance
406	QAPP	quality assurance project plan
407	QC	quality control
408	R&D	research and development
409	RG	regulatory guide
410	ROI	region of interest
411	SGS	segmented gate system
412	SNR	signal-to-noise ratio
413	SOP	standard operating procedure
414	TAP	total absorption peak
415	TLD	thermoluminescent dosimeter
416	UBGR	upper bound of the gray region
417	WRS	Wilcoxon Rank Sum test
418	XRF	x-ray flourescence
419	ZnS	zinc sulfide

1 INTRODUCTION

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) is developing a basis to support decisions on whether to undertake a rulemaking that would set specific requirements on controlling licensees' releases of solid materials, which are potentially available for release of NRC-licensed sites during operations as well as during decommissioning. Specifically, the solid materials being evaluated include metals, building concrete, onsite soils, equipment, piping, conduit, furniture, etc., which are present at, and/or used in, licensed nuclear facilities during routine operations. Historically, licensees have released solid materials on a case-by-case basis, using release criteria that varied from "no detectable activity greater than background" to the surface activity guidelines found in, or adapted from, Regulatory Guide (RG) 1.86 (AEC, 1974).

1.2 Need for This Report

This report provides technical information, based on the Data Quality Objectives (DQO) Process, designing, performing, and documenting clearance surveys for solid materials. Toward that end, this report discusses a number of clearance survey approaches, which use a variety of survey technologies and instrumentation. This report also provides guidance for using the DQO Process to determine the most advantageous clearance survey protocol based on the solid material being released, available survey instrumentation, required laboratory analyses, and applicable release criteria. The various survey protocols discuss analytical and field survey instrumentation criteria, material parameters (e.g., physical nature of material, survey unit sizes), and techniques that can be applied to clearance surveys of materials. The DQO Process also helps to address clearance survey approaches for radioactive materials that may have inaccessible surfaces or may not be in directly accessible areas. The overall objective is to provide guidance for selecting and properly applying clearance survey strategies.

1.3 Scope

The major emphasis of this report is to provide technical information on designing, performing, and documenting clearance surveys for solid materials. Specifically, the solid materials covered include scrap metals, building concrete rubble, onsite soils, equipment, and building debris[1]. This report describes a number of different clearance survey approaches, including conventional scanning, automated scanning using a conveyorized survey monitor, and *in toto* techniques, such as *in situ* gamma spectrometry and tool monitors.

Importantly, this report stresses the use of situation-specific measurement of scanning to release solid materials whenever the scan minimum detectable concentration (MDC) is less than the derived concentration guideline level for clearance ($DCGL_C$). Statistical survey designs, such as those discussed in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), NUREG-1575, Rev. 1, are recommended for direct measurements of surface activity and media samples in cases where the scan MDC is greater than the $DCGL_C$.

[1]Note that the U.S. Department of Energy uses the term "non-real property" to refer to solid materials such as tools, equipment, office items (furniture), consumable items and debris, while "real property" refers to land and building structures.

Appendix A provides a primer on the basic radiation properties, which are relevant to the measurement of radioactivity in and on solid materials. It also addresses some of the fundamental principles of radiation detection and measurements.

In preparing this report, the NRC staff considered various types of instruments that are used to perform clearance surveys, including gas proportional, Geiger-Mueller (GM), zinc sulfide (ZnS) scintillation, sodium iodide (NaI) scintillation, and high-purity germanium (HPGe) detectors. It was not the intent of this study to compare different manufacturers' field survey instruments. Rather, the various instruments that were used in this study are generally representative, with the notable exception of the conveyorized survey monitor (CSM). Moreover, the reader should note that the use of these survey instruments in conducting this study does not, in any way, constitute endorsement of a particular product or manufacturer by the NRC or its contractors.

This report assumes that the user has some knowledge of the solid materials to be cleared. The role of process knowledge (covered in Section 4.3) is important both in deciding whether to pursue clearance of the solid material, and in providing information on the nature and degree of contamination that the solid material might be expected to have. Specifically, characteristics of the solid material that impact its clearance include the material's physical description, contamination potential, nature of the contamination, and degree of inaccessible areas.

1.4 Methodology

Clearance survey approaches were determined using the DQO Process, giving due consideration to two major requirements. Specifically, (1) the survey result must be able to demonstrate that the clearance criterion has been met within predetermined confidence levels, and (2) the survey unit size must be sufficiently evaluated to develop a technically defensible approach for area or volume averaging. The clearance survey should also follow the DQO Process to address the potential presence of elevated contamination. That is, the solid material should meet any established release criterion limiting contamination over specified smaller portions of the surveyed material be met, ***and*** the average radioactive concentration over the material survey unit, as determined by a sufficient number of measurements, should satisfy the average clearance concentration limit ($DCGL_C$) that has been established. Additionally, the clearance survey approaches discussed herein recognize the importance of process knowledge in survey design, as well as the usefulness of scanning, particularly when the survey instrument has sufficient scan sensitivity and lends itself to the automatic documentation of scan results.

The general clearance survey approaches identified include (1) material release surveys using conventional instruments that incorporate both scanning and statistical designs for determining sample sizes; (2) automated scanning surveys that use data acquisition systems (conveyorized survey monitors) to automatically document scan results; (3) *in toto* surveys (i.e., survey techniques that measure the entire material at once) performed using gamma spectrometers, bag monitors, tool monitors, and portal monitors; and (4) analytical methods and laboratory analyses on representative samples based on statistical sampling designs. The clearance survey approach should also consider whether the solid material has potential surficial or volumetric contamination, or both. A working definition of volumetric contamination is contamination that is present beneath the surface of the material. One might, in turn, define surficial contamination as the activity contained within a surface layer with a thickness equal to that of the saturation layer, which ISO (1988) defines as the thickness of the medium (surface material) equal to the maximum range of the specified particulate radiation.

Appendix B provides additional information on advancements in general radiation detectors and survey instruments that utilize new detection materials and software. These clearance survey approaches are sufficiently comprehensive to include and account for physical measurement parameters, including radionuclide(s); concentrations; difficulty and expense of detection; and complexity, size, or configuration of clearance item(s).

This report considers both the material matrices being cleared, as well as the facility types releasing these materials. For example, this study considered the following facility types:

- nuclear power reactor
- sealed source facility
- transuranic facility
- fuel fabrication facility
- broad research and development (R&D) facility
- gaseous diffusion plant
- uranium mill facility
- rare earth facility

In addition, the clearance survey approach should consider the typical radioactivity mixtures associated with the given facility type. Knowledge of the radionuclide mixture is necessary to develop appropriate derived concentration guideline levels for clearance and, therefore, is essential for proper survey design.

2 ROADMAP

The flow diagram (Figure 2.1) for the clearance of solid materials serves as an overview of the clearance process described in this report. Section references in the flow diagram boxes direct the reader to the section of this report that discusses the particular guidance.

As illustrated in the flow diagram, the clearance process consists of a series of steps that provide sufficient confidence that the established clearance criterion has been met. With the DQO Process as the underlying foundation, the steps of the process are summarized as follows:

a. Evaluate and sort solid materials in terms of handling issues, such as the size and physical nature of the material (e.g., many small regular pieces or a few large, irregularly shaped pieces).

b. Research and document the process knowledge for the solid material, and characterize the material as necessary.

c. Based on the process knowledge of the material, determine whether the solid material is impacted. If not, the solid material can be considered for release.

d. Specify the release criterion, including conditions for applying the criterion, for the given solid material.

e. Classify the impacted solid materials according to their potential for containing radioactivity into Class 1, 2, or 3 material survey units (also termed lots or batches).

f. Depending on a number of cost considerations (e.g., cost of radioactive waste disposal, value of the cleared material, cost of cleaning and dismantlement, and cost of the clearance survey), determine whether clearance is the best material disposition option.

g. Use the DQO Process to select clearance survey approaches and instrumentation based on the nature of the solid material and contamination type and potential.

h. Decide whether the solid material can be released via scanning (considering the material and contamination type and scan MDC). Solid materials are either released via scanning (e.g., using conventional hand-held instruments or conveyorized survey monitors) or via static direct measurements using conventional instruments, *in toto* measurement techniques, or media samples.

i. Based on the selected clearance survey approach(es), assess the survey design issues related to the radiation type and presence of multiple radionuclides (i.e., application of derived concentration guideline levels, such as the use of surrogates and unity rule) and address inaccessible areas.

j. Determine the background distribution for the solid materials of concern for each instrument and detector type. The distribution should consider the variability caused by spatial and temporal background variances in the area where surveys will actually be performed, as well as variations associated with the various material types.

k. Determine the static MDCs and scan MDCs for the selected clearance survey approach(es).

l. Compare the static MDC and scan MDC to the $DCGL_C$. If the static MDC is less than the $DCGL_C$, perform survey (step *p*); but if the scan MDC is less than the $DCGL_C$, evaluate whether a scanning instrument can document the survey results (step *o*). If the MDC and scan MDC are greater than the $DCGL_C$, determine whether the measurement parameters can be changed to reduce the MDCs (step *m*).

m. Determine whether the measurement parameters can be changed to reduce the static MDC. If so, calculate a new static MDC and compare it to the $DCGL_C$. If the new static MDC is less than the $DCGL_C$, perform survey (step *p*). If the static MDC cannot be reduced to a level below the $DCGL_C$, reevaluate disposition options (step *r*).

Determine whether the measurement parameters be changed to reduce the scan MDC. If so, calculate a new scan MDC and compare it to the $DCGL_C$. If the new scan MDC is less than the $DCGL_C$, evaluate whether a scanning instrument can document the survey results (step *o*). If the scan MDC cannot be reduced to a level below the $DCGL_C$, consider using static direct measurements (step *n*).

n. Since the scan MDC cannot be reduced to a level below the $DCGL_C$, determine whether another clearance survey approach is feasible. If so, proceed with the alternative clearance survey approach based on static direct measurements using conventional instruments, *in toto* measurement techniques, or media samples. If another approach is not feasible, reevaluate the disposition options (step *r*).

o. Determine whether the scanning instrumentation has the ability to automatically document scan results. If so, perform a scanning-only survey; otherwise, perform a scanning survey using direct measurements or media samples for documentation purposes. The number of these measurements should be determined using the DQO Process, and may be determined using a statistically based sampling design.

p. For scanning release surveys, perform surface scans using hand-held survey equipment or conveyorized survey monitors. If automatic logging capability exists, perform a scanning-only survey; otherwise, use direct measurements or media samples for documentation purposes. Scan survey coverage is governed by the material classification.

For static direct measurement surveys, use a statistically based sampling design for conventional static measurements with hand-held instrumentation or perform *in toto* measurements using in situ gamma spectrometry, tool monitors, bag monitors, etc. Collect and analyze media samples, such as smears, in lieu of direct measurements when difficult-to-measure radionuclides may be present. Survey coverage is governed by the material classification.

q. Evaluate survey results and appropriately dispose of any solid materials that fail to meet the release criterion. If appropriate, remaining materials from a lot where a failed item was found may be reclassified and resurveyed with a higher degree of rigor if the survey results suggest an original misclassification based on established investigation levels. Clearance survey results are documented.

r. Reevaluate solid material disposition options.

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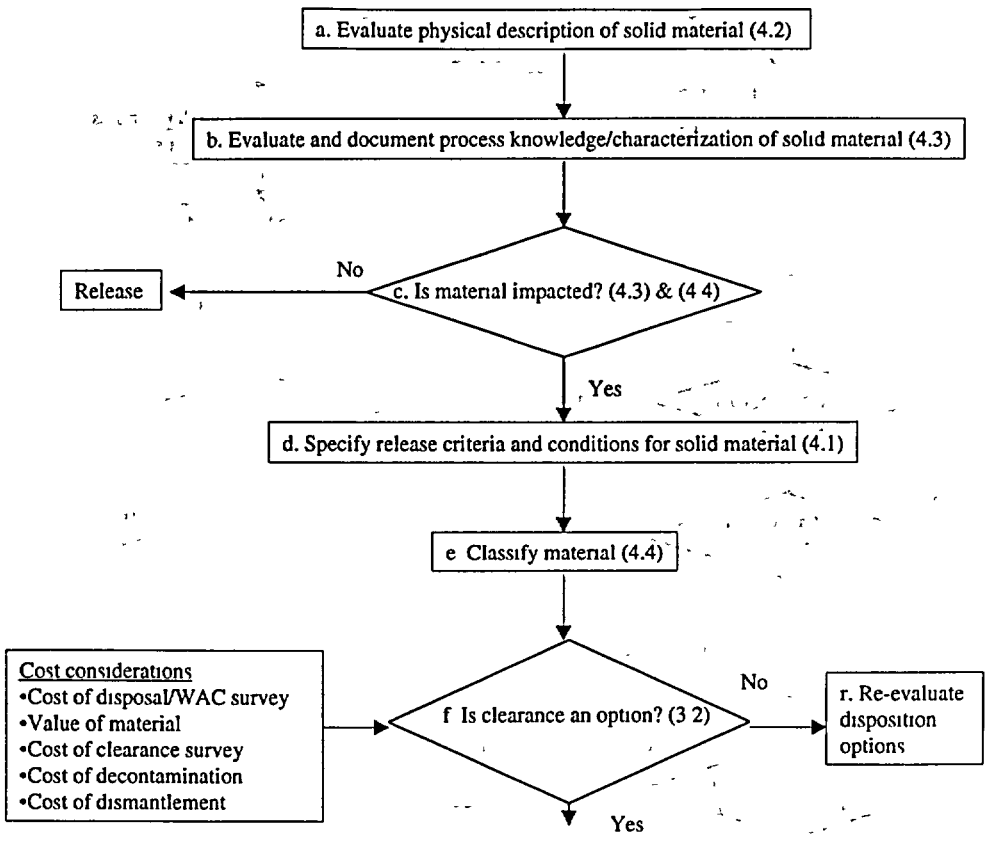
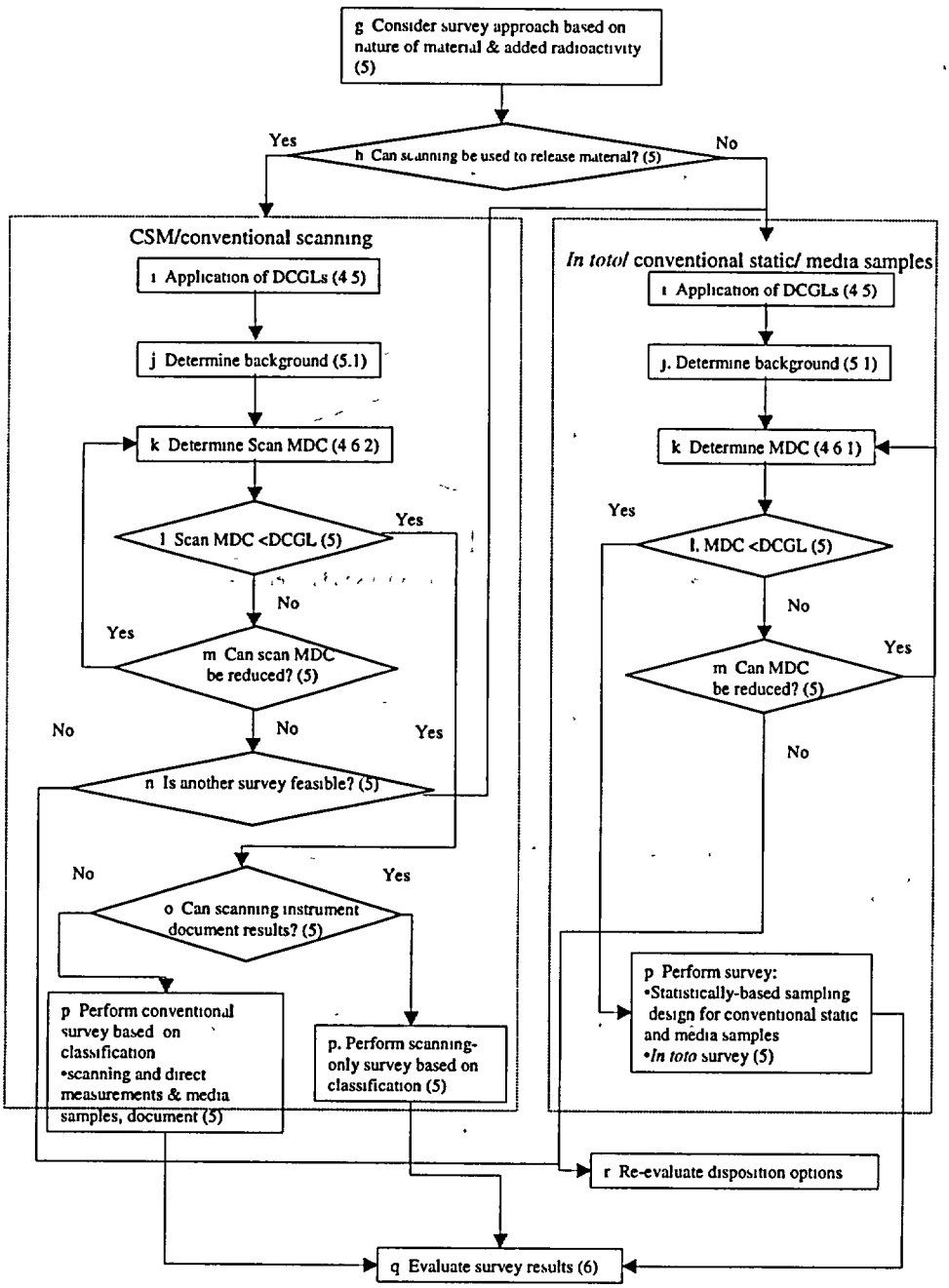


587 **Figure 2.1: Flow diagram for clearance of solid materials**



588 **Figure 2.1: Flow diagram for clearance of solid materials (continued)**

3 DATA QUALITY OBJECTIVES

The approach used in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM, 1997) has proven to be very useful for designing efficient, objective, and defensible final status surveys to collect data to support decisions concerning the release of lands and structures for unrestricted use according to the criteria established by the Commission's final rule (NRC, 1997). Many of the improvements in the design of final status surveys using the MARSSIM were achieved through the extensive use in that document of the Data Quality Objectives (DQO) Process.

The DQO Process is a systematic planning tool based on the scientific method using a graded approach to ensure that the level of detail in planning a survey and the level of effort applied in conducting a survey are commensurate with the intended use of the resulting data and the degree of confidence needed in the results. This process focuses the need for data collection on the decisions that will be made using the data. Data that do not contribute to better decisionmaking are superfluous. By focusing the surveys on the data needed for a *decision* resulting in a specific *action* or its alternative being chosen leads naturally to an efficient design.

The DQO Process is quite general and certainly can be applied to solid material surveys. Some of the specific concepts developed for the MARSSIM, such as survey unit classification (Section 4.3), will continue to be useful in controlling the release of solid materials. However, surveys of solid materials and final status surveys of lands and structures differ in some fundamental ways. The remainder of this section discusses the DQO Process specifically to examine the quality and quantity of survey data that may be needed in order to make decisions about releasing solid materials from radiological controls.

3.1 State the Problem

The basic issue is whether solid materials that may contain contamination from a licensed facility can be released from radiological controls. To state the problem clearly, the process begins with developing a conceptual model of any potential radiological exposure, which identifies (1) any known or expected locations of radioactivity, (2) potential sources of radioactivity, (3) the nature of the solid material that may contain contamination, (4) whether such radioactivity is likely to be on the surface of the material or distributed through a portion of its volume, and (5) potential exposure scenarios for the material. Process knowledge is very important in completing this step.

If solid material has the potential for containing contamination from facility operations, a survey is generally required before the material may be released from controls. The types and sensitivity of equipment, procedures, and resources available for measuring any contamination in or on the material should be also be addressed. The regulatory criteria for preventing the release from control of materials with unacceptable levels of contamination must also be established. These may be either activity-based or dose-based. If the criteria are dose-based, the equivalent criteria in terms of an activity concentration must be obtained from an approved dose modeling procedure; NUREG-1640 provides an example of a methodology for converting activity concentration to potential dose.

3.2 Identify the Decision

Following the collection of survey data, a decision is made as to whether the material can be released from radiological controls. That decision is based on whether the survey data indicate that the criteria established for the prevention of release of materials with unacceptable levels of contamination have been exceeded. If not, the material is allowed to be released from radiological controls.

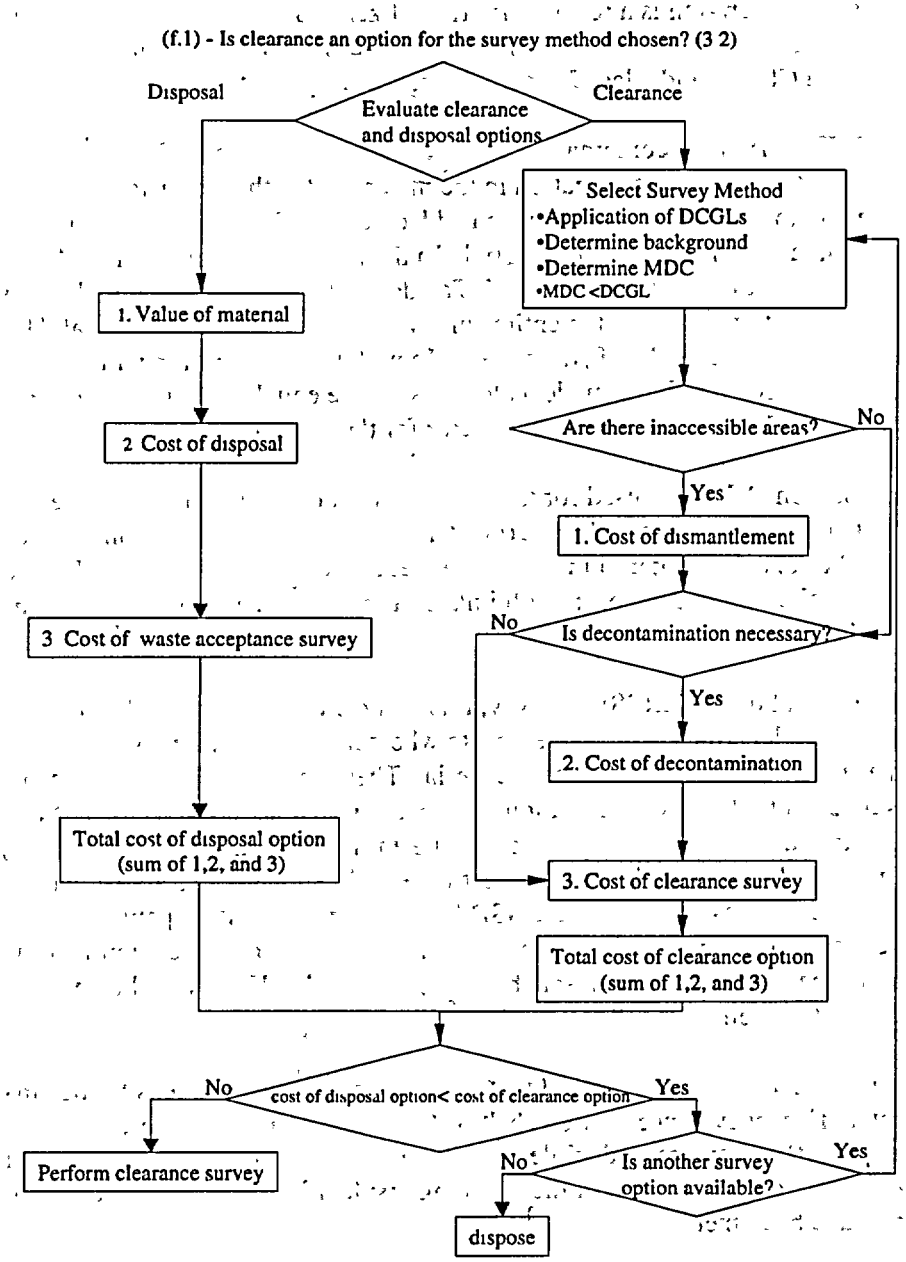
By contrast, if the level of contamination in or on the material exceeds the release criteria, the material may not be released from control. However, further actions may be possible. One course of action may be to remove radioactivity from the material until the release criteria are met. Another possibility is to abandon release as an option, and dispose of the material as radioactive waste. Figure 3.1 expands step f in the flow diagram for clearance of solid materials (Figure 2.1) to illustrate how the DQO Process might be applied to the decision of whether to attempt to clear the material, rather than disposing of it as radioactive waste. The cost of a survey may exceed the cost of disposal, even taking into account the value of the recycled material. For release of materials, it may be important to decide first whether it is *practical* to perform a survey. In some cases, this may be a close decision that may require actually designing the survey. In others, there may be considerations that make it easier to decide one way or the other. Among these considerations are the radionuclides of concern and how readily they are detected (Section 4.6), and the accessibility of measurement surfaces (Section 4.7). In making these decisions, the cost of the alternative action should include the cost of measurements necessary for waste characterization and disposal costs. A detailed discussion of these alternatives is beyond the scope of this report.

3.3 Identify Inputs to the Decision

Other than the data to be collected, the decision regarding material release is based on certain information, including (1) the actual release criterion (Section 4.1), (2) the material in question (Section 4.2), (3) the radionuclides involved (Section 4.3) and (4) their detectability (Section 4.6).

In the MARSSIM, survey unit classification is used to determine the appropriate type of final status survey to perform, based on all of the information on hand about the survey unit. For surveys of solid materials, process knowledge (Section 4.3) is used much as an historical site assessment would be to assist in the classification (Section 4.4). There is a great advantage to applying this system to surveys of solid materials, in that it allows the survey to focus where it is most needed. In essence, professional judgment is incorporated wherever possible to eliminate the necessity for overly burdensome or prescriptive data collection. This is a key element in using a graded approach to survey design.

Material that has not been exposed to radioactivity can be classified as "non-impacted." Class 3 materials are not expected to contain *any* contamination. Class 2 materials are not expected to contain contamination concentrations in excess of the release criteria over any portion. Class 1 material may contain contamination in excess of the release criteria over some portions.



660 **Figure 3.1: Example of DQO Process applied to clearance vs. disposal**

An alternative under consideration is a release criterion of zero contamination; that is, any detectable radioactivity over background would be unacceptable for release from radiological controls. In this case, the distinction between Class 1 and Class 2 material largely disappears.

As with the MARSSIM surveys, a combination of direct measurements and scanning is used to ensure that the average concentration of contamination in the material is within the established criteria and also to ensure that there are no smaller areas of elevated added activity that may exceed criteria specifically established for such areas on or in the solid material. In the MARSSIM, a dose model is used to establish two sets of criteria through the use of area factors. The derived concentration guideline level ($DCGL_W$) is the radionuclide concentration across the entire survey unit for which the model calculates a dose equal to the release criterion. The $DCGL_{EMC}$ is the radionuclide concentration within a specified smaller portion of the survey unit for which the model calculates a dose equal to the release criterion. The ratio of the $DCGL_{EMC}$ to the $DCGL_W$ is called the area factor for the specified area.

In this report, the notation $DCGL_C$ is used for the average concentration throughout the solid material being surveyed that corresponds to the release criterion. Criteria limiting contamination over specified smaller portions of the surveyed material must also be met if such are established. Note however, that the size and geometrical configuration of the solid material may change significantly from that surveyed to that of a modeled exposure scenario.

In the typical development of a MARSSIM survey, it is assumed that a statistical sample of measurements at discrete locations is used to estimate whether the population average concentration of contamination in a survey unit meets the release criteria. There are cases, however, when scanning sensitivities are sufficient to detect concentrations below the $DCGL_W$. In such cases, if the data are logged so that they are quantitative and reproducible, the entire material survey unit (batch) has essentially been measured and there is no need to estimate the average with a statistical sample. This case was not specifically discussed in the MARSSIM because instruments capable of such sensitivity with logging were just becoming available. When essentially the entire survey unit is measured, the spatial component of the measurement variable becomes negligible. However, the uncertainty of the measurement process itself remains.

For surveys of solid materials, it is anticipated that in many cases, scanning sensitivities may be sufficient to detect and quantify concentrations below the $DCGL_C$. In such cases, provided that the scanning data are quantitative and reproducible, measurements at discrete locations on the material may not be needed. Adequate documentation of the scanning results may be sufficient to establish whether the release criteria have been met.

Conveyorized scanning systems can perform much the same function as scanning with a data logger for the survey of solid materials. In this case, the survey unit is moved under the instrument rather than moving the instrument over the survey unit. By contrast, a box or drum counter can measure the entire "survey unit" or "batch" at once.

In designing surveys of solid materials, a crucial issue is whether measurements and/or samples taken at discrete locations are necessary. This is emphasized in Figure 2.1 (step h), where different paths are taken depending on whether the scanning sensitivity is sufficient to detect the $DCGL_C$. It is also important to determine whether there is a method by which the entire solid survey unit may be measured at once, *in toto*. Box, drum, and tool counters have been mentioned as one possibility. *In situ* gamma spectrometry is another. These approaches and options are discussed in detail in Section 5 of this report.

3.4 Define the Study Boundaries

In the MARSSIM, the size of a survey unit is established to be consistent with the size of the area assumed in the dose modeling. The same criteria should be used to establish survey unit sizes for solid materials, if possible, using exposure scenarios such as those described in NUREG-1640. The potential exposure scenarios can be examined to determine how material is transported through the environment, industry, and commerce to the point of exposure. This could identify whether certain critical areas or volumes require special consideration, or whether homogenization of the material during processing reduces the importance of such areas or volumes.

In some cases, there may be a more natural connection between the "batch size" of a lot of material and the type of survey that should be performed. This is discussed at length in Sections4.1, 4.2, and 5. Here, the reader should simply note that for material that consists of many small regular pieces, a conveyorized scanning system may be used. In this case, a batch might be the amount of material within the instrument field of view. If the material consists of a few large irregularly shaped pieces, a batch might be a single piece that is hand-scanned, or perhaps a few pieces scanned *in toto* using a box or drum counter, or measured using an *in situ* gamma spectrometer.

3.5 Develop a Decision Rule

Section 3.3 discussed three types of survey design, including (1) those in which measurements are made at discrete points together with scans, (2) those in which scanning alone is sufficiently sensitive, and (3) those in which the material is measured *in toto*. The decision rules are slightly different for each type of survey. One decision rule (discussed first) compares the measurement(s) to the $DCGL_C$, while another possible decision rule (discussed subsequently) concerns higher concentrations over smaller areas.

When scanning alone is sufficient, the result of the survey is the average of a great many measurements over the material, far in excess of the number that would be needed to satisfy the requirements of a statistical design. The decision rule is to prevent the release of the solid material from control if the average concentration exceeds the established criteria.

By contrast, when scanning alone is not sufficiently sensitive, it is necessary to obtain a statistical sample consisting of direct measurements or laboratory analyses of the material. The decision rule can be formulated using the same type of hypothesis tests that are used in the MARSSIM, to prevent the release of the solid material from control if the average concentration exceeds the established criteria. The parameter of interest is the average of the measurements.

In the third case, when a single measurement is made of the material *in toto*, the decision is based on this single result rather than the average of several measurements. Decisions of this type, which involve comparing a single measurement to a limit, are essentially based on detector sensitivity. The hypothesis testing framework becomes one of determining the minimum detectable concentration (MDC) of the method. If the MDC is less than the $DCGL_C$, the decision rule is to prevent the release of the solid material from control if the concentration detected exceeds the established criteria.

For the release of materials, then, the fundamental issue is whether the decision rule is to be based on a single measurement or an average. When the decision rule is based on a single measurement, it is essentially a detection decision, and the appropriate framework for considering such decision rules is in the MDC calculations.

A decision rule concerning smaller areas of elevated contamination requires a natural equivalent to the $DCGL_{EMC}$. At minimum, a specific area and area factor must be identified (Section 3.3). For survey design, a conservative choice would be to assume an area factor of 1, making the $DCGL_{EMC}$ equal the $DCGL_C$. This causes no difficulty in the case where the scanning MDC is sufficiently sensitive to detect the $DCGL_C$, but could essentially preclude the release of Class 1 material in other cases. Scanning might still be performed, recognizing that there is a risk of missing an area with a concentration between the $DCGL_C$ and the actual scan MDC. How serious a risk this poses depends on the radionuclide, the material, its potential uses, and, of course, the magnitude of the scan MDC. This would have to be evaluated during the DQO Process (refer to examples in Section 5). For Class 2 material, the scan sensitivity does not drive the survey design since concentrations in excess of the release criterion are not expected over any portion of the material. It does, however, underline the importance of correct material survey unit classification. Judgmental scans (i.e., scans at locations that the surveyor deems to be potentially contaminated) should be performed over a portion of the batch, regardless of the classification. Investigation levels are defined as in the MARSSIM; for Class 3, any positive identification of contamination, and for Class 2 or Class 1, any positive indication of activity above the release criteria.

It may seem, at first, too restrictive to flag any positive indication of activity above the release criterion in Class 1 areas. However, this practice can identify any portion of the material that might cause the overall average to exceed the limit despite the result of the statistical tests. There are also "as low as is reasonably achievable (ALARA) considerations, which would dictate that the contamination in such areas must be removed if it is reasonable to do so. Alternatively, that portion of the material could be segregated and disposed of as waste. This is another fundamental difference between material clearance surveys and lands and structures surveys, in that such segregation is much more easily done "on the fly." Removal of a portion of material is not likely to be disruptive of a "survey unit," as it would be for lands and structures, where it may involve earth moving equipment. Of course, for very large pieces of material or equipment, these advantages will diminish.

An alternative approach is to base the release decision solely on an estimate of the average concentration or the estimated total activity (inventory) of the material to be released. This is equivalent to the assumption that the dose or risk does not depend on the distribution of activity in the material, but only its total amount. This may be a reasonable assumption when the materials from many batches are likely to be mixed during processing. It is less justifiable for equipment that is released for reuse.

When a single measurement is made of the material *in toto,* it is not possible to detect and distinguish small areas of elevated activity. That is, the radiation from such areas may be detected, but will be attributed to the overall concentration. However, the calibration of such detectors usually includes some assumptions about the distribution of activity over the material. The uncertainty analysis of this calibration should include a discussion of the effect of inhomogeneities in the source distribution on the data interpretation. This might be used to estimate bounds on the added activity that might exist over only a portion of the material.

3.6 Specify Limits on Decision Errors

For surveys that involve measurements at discrete locations on the material, several considerations apply in specifying the limits on decision errors. First, is the form of the null hypothesis.

Null Hypothesis: The contamination in the solid material surveyed exceeds the release criterion.

If an activity limit is specified, the Scenario A hypothesis used in MARSSIM would be appropriate. The material is assumed to contain an average concentration above the limit. Unless the data cause this hypothesis to be rejected, the material would not be released. A Type I error involves deciding that the solid material meets the release criterion when it actually does not. The survey would be designed so that the probability of a Type I error occurring is limited to an agreed value alpha when the material contains added activity just at the limit imposed by the release criterion. The probability of a Type I error decreases as the concentration of added activity increases. A Type II error involves deciding that the solid material does not meet the release criterion when it actually does. The probability of a Type II error rate occurring is limited to an agreed value beta when the material contains added activity at a specified concentration lower than the release criterion, as defined by process knowledge or preliminary surveys indicating how much activity is likely to be present. The probability of a Type II error decreases as the concentration of added activity decreases. The concentration range between where the Type I error rate is set (the $DCGL_C$) and where the Type II error rate is set is called the "gray region" because the decision error rates in that range may be higher. The concentration where the Type II error rate is set is, therefore, called the "lower bound of the gray region" (LBGR). The difference ($DCGL_C$ -LBGR) is denoted Δ. In this scenario, the burden of proof is on the surveyor to establish that the release criterion is met.

Null Hypothesis: The solid material surveyed contains no contamination.

It may be that the criterion established for the release of solid material from controls is that there must be no added activity above background. In this case, a form of the Scenario B hypothesis, as developed in NUREG-1505 (NRC, 1998b), would be used. The material is assumed to contain no added activity. Unless the data cause this hypothesis to be rejected, the material would be released. The roles of Type I and Type II errors are reversed from those in Scenario A. A Type I error involves deciding that the solid material contains contamination when it actually does not. The survey would be designed so that the probability of a Type I error occurring is limited to an agreed value alpha when the material contains only background radioactivity. A Type II error involves deciding that the solid material does not contain contamination when it actually does. The probability of a Type II error rate occurring is limited to an agreed value beta when the material contains added activity at a specified concentration. The probability of a Type II error decreases as the concentration of added activity increases. The specification of the Type II error rate at a given concentration is crucial because it dictates how rigorous the survey must be. It specifies the smallest amount of added activity that would be reliably detected in the survey. It is not sufficient to declare that there is no added activity detected without specifying precisely the amount that would have been detected had it been there. The gray region is that between zero added activity (the LBGR) and the specified minimum detectable contamination concentration, which marks the "upper bound of the gray region" (UBGR). Note that if the radionuclide in question does not appear in background and radionuclide-specific measurements are made, any positive measurement would cause the null hypothesis to be rejected. This is based not on the hypothesis test, but on the fact that added activity has unambiguously been identified in the material.

As in the MARSSIM, these hypotheses are tested using a Sign test when the contamination does not appear in background and radionuclide-specific measurements are made. Otherwise, the Wilcoxon Rank Sum (WRS) test is used. For both tests and in both of these scenarios, specifying α, β, and Δ, together with an estimate of the anticipated variability of the measured concentrations over the material, σ, provides sufficient information to calculate the number of measurements that should be made during the survey.

Material survey approaches based on scanning alone with data logging generally require many more measurements than would be required based on hypothesis testing and the determination of statistically based sample sizes using specified Type I and II decision errors rates. An alternative way of viewing this situation is that the number of measurements is so large that the decision error rates are very small and the gray region is very narrow. If there is 100-percent coverage of the material, the entire population of concentrations has been measured. In these cases, a formal statistical test is unnecessary and it is appropriate to simply compare the measured average concentration to the release limit to determine whether it has been met. This is true, provided that there is no bias in the calibration of the instrument or method. Specifically, it is important that the calibrations be determined realistically. For example, the efficiency of the particular clearance measurement depends on the distribution of the contamination. Given that the radionuclide distribution is often non-uniform, it is important to ensure that the uncertainty in the efficiency fully considers the contamination variability, and that a conservative estimate of efficiency is used in the calibration.

The above discussion assumes that a set of sample data is being taken in a survey unit in order to base the release decision on a rule concerning the average concentration. However, as discussed in Section 3.5, the decision rule for surveys conducted with conveyorized scanners or *in toto* detectors may be of a somewhat different form, involving whether or not the concentration estimated for a single batch of material exceeds a specified limit. In this case, the decision rule is essentially a detection decision. Thus, the development of the decision rule and the specification of limits on decision errors are the same as those entering the MDC calculations. NUREG-1505, Rev. 1, Section 2.4, discusses the similarities and differences between MARSSIM-like decision rules and MDC calculations. Both involve specifying a gray region and limiting Type I and Type II decision errors. Both can be framed in the context of a Scenario A null hypothesis (the material surveyed exceeds the release criterion) or a Scenario B null hypothesis (the material surveyed unit does not contain contamination). MDC calculations are usually done for a Scenario B null hypothesis, and the Type I and Type II error rates are set at 0.05. Incorporating the estimated uncertainty for the measurement process, usually denoted σ, the MDC calculation provides the value of the concentration to which the specified Type II error rate applies. Alternatively, starting with a $DCGL_C$ as the concentration at which the Type II rate is set, the MDC calculational framework can be used to design the measurement process in the same way that MARSSIM surveys are designed. All sources of measurement uncertainty must be carefully considered, including possible inhomogeneities in the distribution of activity over the material. The entire decision rule and DQO Process depend on the estimated measurement uncertainty, σ, near the detection limit since the resulting MDC is typically about 3 or 4 times σ. Further guidance on evaluating and expressing uncertainty may be found in Taylor and Kuyatt, 1994.

3.7 Optimize the Design for Obtaining Data

The DQO Process emphasizes a graded approach so that the survey effort is commensurate with the likelihood that the material contains sufficient contamination that it should remain under radiological control. The extent of the survey depends on the classification of the material. Process knowledge plays a crucial role in this classification, and the better documented the use of the material, the more accurate the classification will be.

The details of material survey designs are discussed in Section 10. Non-impacted material is clean and requires no survey. Class 3 material is very likely to be clean and usually requires only judgmental scans over a small portion of the material, in addition to direct measurements. Class 2 material is nearly clean, but may require more systematic scanning of 50 percent or more. Class 1 material will require systematic scanning of 100 percent of the material.

With sufficient scanning sensitivity, direct measurements are not required. Conveyorized survey monitors may be able to efficiently scan 100 percent of the material, again without the need for direct measurements. Measurements of an entire batch of material using *in toto* techniques in essence combine the attributes of a direct measurement with a measurement that has some of the attributes of a 100-percent scan.

For cases in which only one *in toto* measurement is made, the significant source of variability is measurement error, and the hypothesis test is a detection decision similar to that used in calculating an MDC, with the exception of the possible reversal of the usual null and alternative hypotheses. However, the survey should consider the possible effect of source inhomogeneity on the calibration, which will play the role of spatial variability in this case. Similar considerations will apply for conveyorized scanning.

For batches of material that require statistical sampling, the variability of concentrations across the batch may have a significant impact on the number of samples required. Pre-screening and careful documentation of the prior use of the material can improve the classification, and will also allow construction of more homogeneous batches. As with the MARSSIM, the number of samples depends on the variability of activity within a survey unit, not the size of the survey unit. A few large items with similar activity could make a Class 2 batch, while one large item with spotty contamination might have to be treated separately as a Class 1 batch requiring more samples.

When realistically calculated scanning MDCs are below the $DCGL_C$, clearance surveys based on simple detection decisions are usually most efficient to segregate any material above the $DCGL_C$ for either cleaning or disposal. Issues of survey unit size and elevated measurements become largely irrelevant. However, the defensibility of such surveys rests entirely on how carefully the MDCs are calculated.

The relationship between MDCs, minimum quantifiable concentrations (MQCs), and the calculation of combined standard measurement uncertainties is being actively investigated by international standards groups. See for example, ISO, 1995, 1997, 2000a, and 2000b, as well as IUPAC, 1995.

4 SURVEY DESIGN CONSIDERATIONS

This section addresses specific areas of consideration common to radiological surveys for controlling release of solid materials. The topics discussed include release guidelines and their application, the nature of solid materials being considered for release, process knowledge used to classify materials based on their potential for contamination, the measurability of contamination, and inaccessible areas. These topics should be addressed during the planning stages of radiological surveys for solid materials.

4.1 Release Guidelines

Sections 4.1.1 and 4.1.2 introduce the various forms of release guidelines, and then discuss the related averaging conditions and survey unit considerations.

4.1.1 Forms of Release Guidelines

Release guidelines can either take the form of activity concentrations or be based on the potential dose to an individual. Regulatory Guide (RG) 1.86 (AEC, 1974) provides an example of surface-based guidelines, which are generally based on the detection capabilities of commercially available survey instruments. Table 4.1 provides the RG 1.86 surface activity guidelines and conditions for implementation, and is reproduced here to provide historical perspective on clearance criteria. Removable surface activity guidelines are 20 percent of the average surface activity guidelines for each grouping.

Table 4.1: Regulatory Guide 1.86 surface activity guidelines

Radionuclide	Average Total Surface Activity in 1 m^2 ($dpm/100\ cm^2$)	Maximum Surface Activity in 100 cm^2 ($dpm/100\ cm^2$)[2]
U-nat, ^{235}U, ^{238}U and associated decay products	5,000 α	15,000 α
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	100	300
Th-nat, ^{232}Th, ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	1,000	3,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr and others noted above	5,000	15,000

[2]The maximum surface activity guidelines (which are three times the average guidelines) in RG 1.86 effectively provide for an area factor of 3 for 100-cm^2 areas.

The application of the surface activity guidelines shown Table 4.1 requires some explanation. First, it is important to understand that surface activity levels may be averaged over 1 m^2, but no surface activity levels can exceed the maximum surface activity specified for a 100-cm^2 area. Hence, RG 1.86 provides release criteria for surface activity, as well as averaging conditions for the application of those criteria. Also note that RG 1.86 does not include volumetric release criteria. The standards were to be dose-based; hence, the release criteria should include the dose criterion upon which to base the $DCGL_C$ (clearance DCGL), as well as any necessary conditions for the implementation of the $DCGL_C$. For example, any limits on the area or volume averaging of solid materials should be clearly expressed. Restrictions on the averaging area or volume of solid materials will necessarily impact the material survey unit or batch size.

Draft NUREG-1640 (NRC; 1999), "Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities," considers both reuse and recycle scenarios, and was written to provide a method for converting a dose criterion to a concentration that can be measured on equipment and materials. NUREG-1640 contains dose factors for a number of different metals and concrete for many radionuclides, and these dose factors address contamination both surficially on equipment and volumetrically in scrap materials. The dose factors are normalized and are expressed in units of annual dose per unit of radioactivity (e.g., in μSv/y per Bq/g or mrem/y per pCi/g).

4.1.2 Release Guidelines — Averaging Conditions and Survey Unit Considerations

As mentioned in Section 4.1.1, the regulatory criteria for preventing the release from control of materials with unacceptable levels of contamination may be either activity- or dose-based. Regulatory Guide 1.86 is an example of the former, while draft NUREG-1640 provides an example of a dose-based approach for calculating activity concentrations that equate to the release criterion. Furthermore, in the case of dose-based criteria, it is possible that area or volume factors will be determined. Area and volume factors, as derived from dose modeling, can be used to determine maximum limits on activity concentrations greater than the $DCGL_C$ that could exist in smaller surface areas (or volumes) than those modeled to derive the $DCGL_C$, and still demonstrate compliance with the dose criteria. Therefore, the radiological survey approaches discussed herein should address both the average contamination in the survey unit, as well as the contamination that may be present in smaller areas and volumes within the survey unit.

One of the technical challenges is defining a "survey unit" for clearance surveys of materials. The material survey unit (or batch) concept is at the core of statistical designs for release surveys. In the MARSSIM, the survey unit represents a specific land area or building surface area. For clearance of solid materials, the survey unit may consist of equipment surface area, volume of bulk material (soil or rubblized concrete), number of small items, lengths of pipe, etc. Like the survey unit concept in the MARSSIM, any relationship between the survey unit size (i.e., batch size) and the modeling input used to establish the $DCGL_C$ should be adhered to. Thus, the definition of a material survey unit (or batch) for solid materials released using a conveyorized survey monitor (CSM) may relate to the amount of material scanned as it passes under the detector(s) for a specified observation interval and given belt speed. Based on the material's classification, 10 to 100 percent of the material might be selected for analysis on the CSM. Another example might include a few large pieces of equipment. In this case, the survey unit might consist of the entire piece itself, such as a large electrical panel. Therefore, material survey unit selection is ultimately based on the DQO Process, consistent with the nature of the material, the clearance survey technique selected, and the material's potential for contamination.

4.2 Solid Materials

This section discusses the physical nature of the solid materials being cleared. The physical nature of the material refers to attributes such as the size of the material and composition (or homogeneity) of the material, and it directly impacts the handling issues, as well as the selection of the clearance survey approach. For example, large, discrete pieces of metal can be surveyed using conventional hand-held survey instruments, while peanut-sized pieces of copper chop are perhaps best surveyed using a conveyorized survey monitor or via laboratory analyses. These smaller solid materials consisting of many small regular pieces are best handled and released as bulk material, perhaps using a conveyorized survey monitor or an *in toto* clearance technique. By contrast, a concrete slab may be released on the basis of a surface scan using a large-area gas proportional detector, as compared to rubblized concrete which is cleared on the basis of a number of representative samples analyzed in a laboratory.

Therefore, it may be appropriate to consider solid materials as being comprised of (1) many small regular pieces, (2) individual, large pieces of equipment and metal, or (3) medium-sized items and materials that fit on a pallet (e.g., perhaps 10 to 100 pieces of cut pipe, fan blades, etc.). Figures 4.1 through 4.6 provide photographic examples of typical solid materials being offered for release.

It may be advantageous for the material to be processed before being surveyed. Solid materials that can be made homogenous via melting, chopping, cutting, etc. are more easily surveyed. For example, copper wire can be surveyed with hand-held survey instruments, but it can be more effectively surveyed using a CSM if the wire is chopped into small pieces. Similarly, material processing might include cleaning techniques (e.g., grit blasting, melting), which can homogenize and reduce the material's contamination potential.

Addressing inaccessible areas (Section 4.7) is another important issue that impacts the decision of whether to clear the material. If material preparation activities include dismantling (i.e., cutting, disassembly) or use of specialized survey instruments to gain access to inaccessible areas, it may be deemed too expensive to survey and release the material. In such situations, disposal may be a more appropriate option.

This section provides a number of material examples that address the design of clearance surveys for solid materials. Each of the following solid materials is described in terms of its composition, weight, material survey unit dimensions, and estimated percent of inaccessible areas.

Concrete rubble consists of crushed concrete of a soil-like consistency from the demolition of buildings and structures. The reinforcing steel rebar has been removed from the concrete rubble. The primary assessment techniques include laboratory analysis of a statistically determined number of representative samples and surface scans, or use of a CSM. The total surface area of the crushed concrete when spread out to a height of 15 cm (to facilitate scanning) is about 50 m^2. This survey unit is assumed to have no inaccessible areas.

A **concrete slab** consists of 30-cm thick medium density concrete (2.4 g/cm^3), with surface dimensions of 1.2 m by 1.8 m. The primary assessment technique is surface activity measurements, perhaps with the number of measurements statistically determined, and surface scans. This survey unit is assumed to have no inaccessible areas and only to have contamination surficially. If volumetric contamination is expected, alternative clearance survey techniques, such as concrete core samples, are warranted.

Small-bore pipe (<6 cm diameter) from piping systems and electrical conduit is assumed to be sectioned into 1.2-m to 1.8-m lengths. It is assumed that conventional survey instrumentation cannot access the pipe interiors. For Class 2 and 3 survey units—so classified because the pipe interiors are very unlikely to have contamination—the primary assessment technique is surface activity measurements of pipe exteriors, with a number of smears from the pipe interiors, and surface scans. Class 1 survey units should be fully surveyed inside—so either the pipe must be cut open or specialty survey equipment employed. It may also be possible to evaluate the cut pipe using a CSM or *in situ* gamma spectrometer. The surface area for pipe section exteriors per survey unit is 17 m^2 (based on a pipe diameter of 6 cm and 1.5-m lengths).

Large-bore pipe (>6 cm diameter) from piping systems is assumed to be sectioned into 1.2-m to 1.8-m lengths. It is assumed that conventional survey instrumentation can access the pipe interiors. Therefore, this survey unit is assumed to have no inaccessible areas. The primary assessment technique is surface activity measurements of pipe interiors and exteriors, and surface scans. The surface area for pipe section interiors and exteriors per survey unit is 72 m^2 (based on a pipe diameter of 30 cm and 1.5-m lengths).

Structural steel consists of light and heavy gauge steel that may require sizing to fit on a pallet (1.2-m to 1.8-m lengths). The structural steel may consist of I-beams, structural members, decking, ductwork, tanks, and other containers. This survey unit is assumed to have no inaccessible areas. The primary assessment technique is surface activity measurements, with the number of measurements statistically determined, and surface scans. *In toto* clearance techniques may also be useful to assess structural steel.

Copper wire consists of insulated and non-insulated wire (0.6 cm or larger), copper windings, and bus-bars. It is assumed that this amount of copper weighs 0.75 tons. The primary assessment technique is surface activity measurements, with the number of measurements statistically determined, and surface scans. *In toto* clearance techniques may also be useful to assess copper wire. This survey unit is assumed to have no inaccessible areas.

Copper ingots (bulk) consist of size-reduced pieces of copper and ingots. The primary assessment technique is laboratory analysis of a statistically determined number of representative samples and surface scans, or use of a CSM. The total surface area of the bulk copper when spread out to a height of 5 cm is about 15 m^2. This survey unit is assumed to have no inaccessible areas.

Soil includes materials that are soil-like, consisting of a finely divided mesh. The primary assessment technique is laboratory analysis of a statistically determined number of representative samples and surface scans. Other clearance survey techniques that might be employed include use of a CSM or *in toto* techniques. The total surface area of the soil when spread out to a height of 15 cm (to facilitate scanning) is about 50 m^2. This survey unit is assumed to have no inaccessible areas.

Large items for reuse include transformers, specialty equipment (e.g., lathes), electrical panels, and other complete systems. These materials are assumed to require some amount of disassembly to allow access to interior surfaces, but consideration must be given to the fact that these items are valued for their function, so cutting is usually not an option. The nominal weight of a large item is 1.5 tons. The primary assessment technique is surface activity measurements, with the number of measurements statistically determined, and surface scans. *In toto* clearance techniques may also be useful to assess large items for reuse.

Scrap metal pile consists of miscellaneous mixed metals with no common configuration. The scrap may require sizing to fit on a pallet. The nominal weight of the material on a pallet is assumed to be 1 ton. The primary assessment technique is surface activity measurements, with the number of measurements statistically determined, and surface scans. *In toto* clearance survey techniques might also prove useful. The total surface area of the scrap metal pile is assumed to be about 10 m^2. This survey unit is assumed to have no inaccessible areas.

Scrap equipment and small items for reuse include small pumps, motors, hand tools, power tools, scaffolding, and the like. These materials are often associated with operational releases and are assumed to require some amount of disassembly to allow access to interior surfaces. The nominal weight of the material on a pallet is assumed to be 1.5 tons. The primary assessment technique is surface activity measurements, with the number of measurements statistically determined, and surface scans. Both *in toto* and CSM clearance survey techniques might be used to release scrap equipment.

As mentioned in Section 4.1.2, survey units should be selected based on the DQO Process, consistent with the nature of the material, the clearance survey technique selected, the material's potential for contamination, and considering any relationship between the survey unit size (i.e., batch size) and the modeling input used to establish the $DCGL_C$. Table 4.2 provides typical survey unit sizes.

Table 4.2: Typical material survey unit sizes

Solid Materials	Examples	Survey Unit Sizes
Bulk materials	soil, concrete rubble, copper ingots	1 to 7.5 m^3 (smaller for CSMs)
Few, large pieces of equipment and material	concrete slabs, large items	item itself
Small items on a pallet	small- and large-bore pipe sections, structural steel, equipment, scrap metal, copper wire	10 to 100 m^2

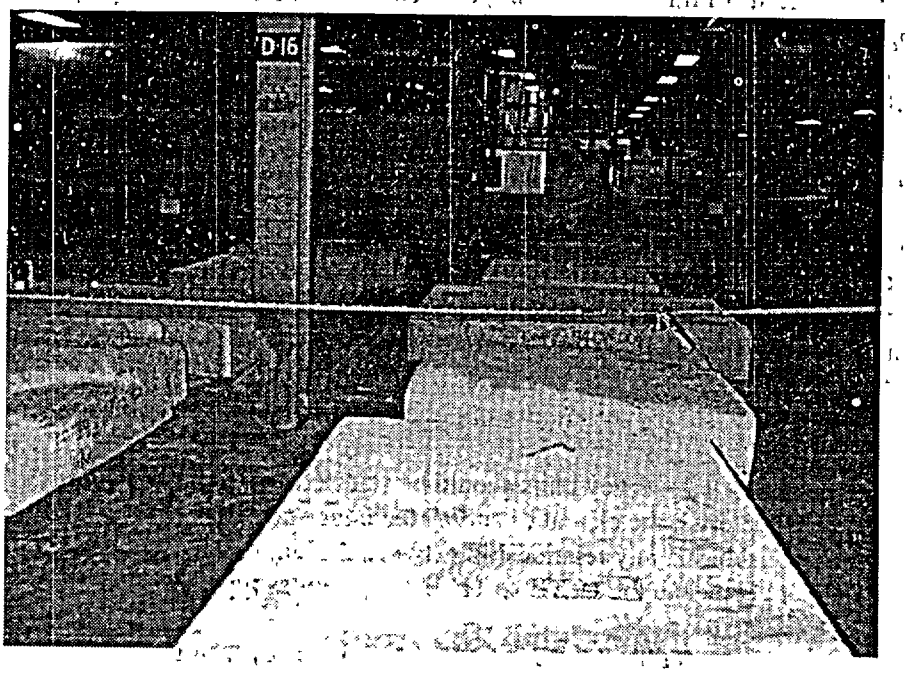


Figure 4.1: Concrete slabs staged for clearance surveys

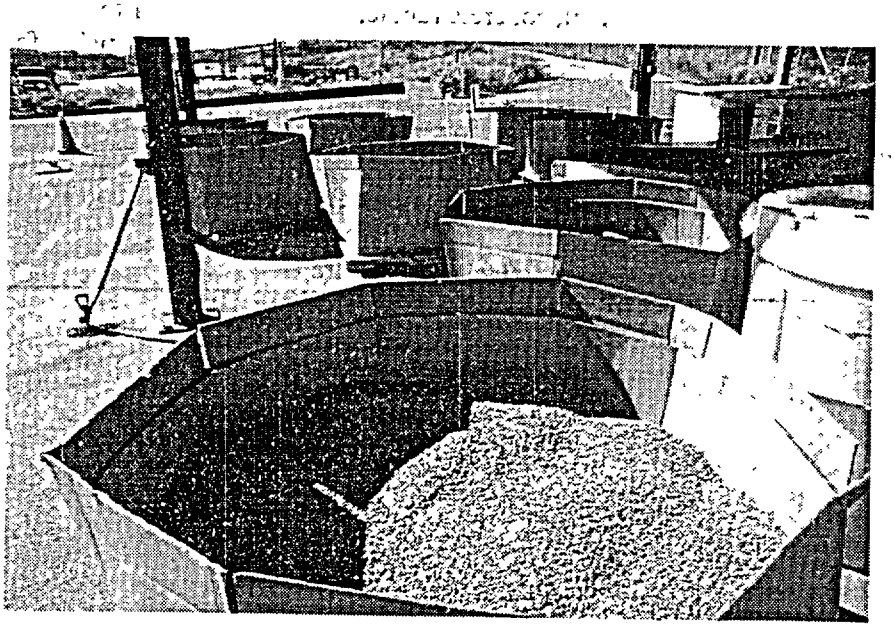


Figure 4.2: Containers of copper chop (recently surveyed using the conveyorized survey monitor)

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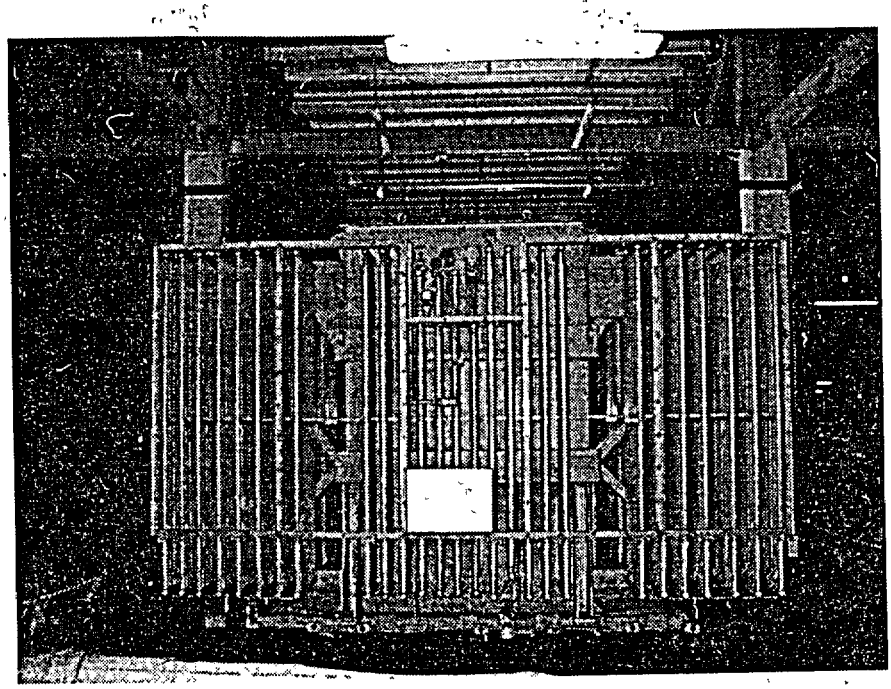

1079 **Figure 4.3: Transformer being surveyed for reuse**

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1081 **Figure 4.4: Scrap equipment (rotors) that may need disassembly prior to release**

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1083 **Figure 4.5: Scrap metal piles being prepared for survey**

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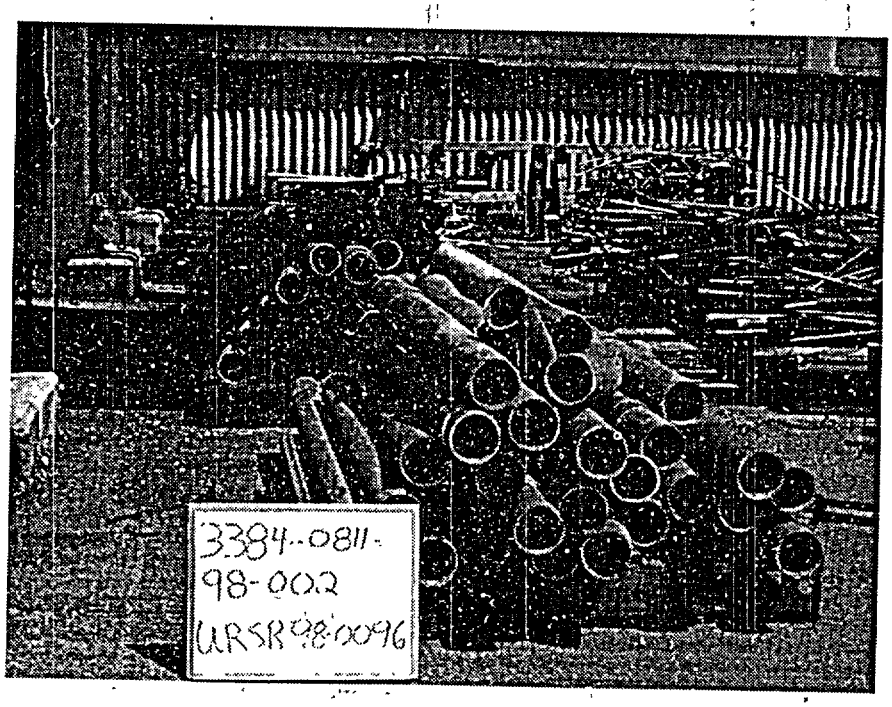

1085 **Figure 4.6: Large-bore piping that has been sectioned to permit release surveys**

4.3 Process Knowledge and Characterization

The release of solid materials can occur during both normal operations and decommissioning of a facility. Releases that occur during operations typically involve smaller quantities of materials than those that occur during facility decommissioning, and the materials' potential for having contamination is usually better known for operational releases than for decommissioning releases since the materials' origin is more certain. Regardless of when the materials are offered for release, process knowledge concerning the solid material is critical. In fact, it may be worthwhile to use the DQO Process to develop the materials' process knowledge. The following section identifies inputs that are relevant to any material release decisions involving process knowledge.

4.3.1 Evaluating a Solid Material's Contamination Potential

One of the first steps in the clearance process is to use process knowledge to determine whether licensed operations impacted (contaminated) the solid material. Operational surveys are expected to provide information supporting the classification decisions discussed in the next section. Process knowledge is obtained through a review of the operations conducted in facilities where materials may have been located and the processes in which the materials may have been involved. This information is used to evaluate whether the solid material (such as structural steel, ventilation ductwork, or process piping) may have been in direct contact with radioactive materials by design. Reviews should also include operational records to evaluate whether spills, fires, and/or airborne or similar releases occurred that may have resulted in material contamination. The records review should also include survey data that may indicate the presence of contamination.

In some instances, process knowledge may not be available for the solid material being considered for clearance. For example, consider an outdoor material staging area, where various pieces of rusty equipment and metal have accumulated over the years. The origin of these solid materials is unknown. In this case, it is particularly important to perform characterization surveys of the materials to establish their contamination potential and the radionuclide identity of the contamination on these solid materials. Furthermore, surveys are useful to validate the material's process knowledge, even when the solid material has a well-documented history.

After reviewing the material's process knowledge and completing the characterization, an initial classification is performed. The selection of material classification should be based on the process knowledge, as well as previous operational records and survey data, to establish the potential for solid materials to have contamination. This may include considering the function and use of the material, location(s) where the material was used, determinations as to whether previous surveys were performed to supplement the process knowledge, and whether there is a potential for internal contamination and how it affects the classification. Additionally, the potential for the materials to have been exposed to a neutron fluence resulting in the formation of long-lived activation products should be evaluated.

Materials that have never been in a radiological area are typically classified as non-impacted. For example, virgin steel I-beams that resulted from the demolition of an office building that was located outside of control areas and had never housed radiological activities of any type would be classified as non-impacted. Impacted solid materials are those items that were, at any period in time, stored or used within a radiological area. These items could have contamination and, therefore, require further evaluation before they may be considered for release.

The contamination potential of the solid material is used to further classify the material as either Class 1, 2, or 3 (Section 4.4). The specific classification will assist in defining the survey approach prior to release. Those materials having the highest potential for contamination would receive the greatest clearance survey effort.

> Solid materials are classified as Class 1, 2, or 3 based on the contamination potential of the material. The specific classification dictates the required rigor of the clearance survey.

4.3.2 Evaluating the Nature of Contamination

Process knowledge can also be used to determine the nature of contamination (i.e., the identity, extent, and location of the radionuclide contamination on the solid material). The type of facility from which the materials originated is an important factor. For example, if the solid materials came from a nuclear power reactor, the likely radioactivity includes fission and activation products; if the materials were from a gaseous diffusion plant, the radioactivity may include enriched uranium and ^{99}Tc. A number of studies have investigated screening (release/clearance) levels for key radionuclides associated with clearance (IAEA 1996, Hill 1995, NRC 1999, ANSI 1999). Rather than develop a new list or augment existing lists, this section focuses on a few important radionuclides to explore specific issues related to their presence and detection in solid materials.

The radionuclide mixtures for each facility type (or industry category) should be known in order to effectively design the clearance survey. The specific facility type provides a general indication of the expected radionuclides. Short-lived radionuclides (i.e., half-lives from less than a day to several months) that may be associated with a particular facility are not shown. It is necessary to account for the potential presence of short-lived radionuclides, which may include justification that the radionuclides are not a concern because of their expected contamination levels considering radioactive decay. Common radionuclides at various types of facilities are as follows:

Nuclear Power Reactor	^{60}Co
	^{137}Cs
	^{63}Ni
	^{55}Fe
	fission and activation products
	transuranics
Fuel Fabrication Facility	enriched uranium
Sealed Source Facility	^{241}Am
	^{60}Co
	^{137}Cs
	^{90}Sr
Broad R&D Facility	^{3}H
	^{14}C

Transuranic Facility	^{241}Am
	^{239}Pu
	^{238}Pu
Gaseous Diffusion Plant	^{99}Tc
	enriched uranium
	transuranics
Uranium Mill Facility	^{238}U
	^{230}Th
	^{226}Ra
	progeny
Rare Earth Facility	Thorium

Scoping and characterization surveys would likely be performed, and may include field measurements and sample collection with laboratory analysis, to identify the specific radionuclides that are present and their radiation characteristics. Identification of radionuclides is generally performed through laboratory analyses, such as alpha and gamma spectrometry, and other radionuclide-specific analyses. For instance, the radionuclide mixture of contamination on solid materials that originate from a power reactor facility may be assessed by collecting representative samples, and performing gamma spectrometry analyses to determine the relative fractions of activation and fission products present. Radionuclide analyses are also used to determine the relative ratios among the identified radionuclides, as well as to provide information on the isotopic ratios and percent equilibrium status for common radionuclides like uranium and thorium decay series. This information is useful in establishing and applying the $DCGL_C$ for the material being released. Table A.4 in Appendix A provides information on radionuclide characteristics and lists some standard methods for detecting their radiations.

It is useful to consider the possible contamination scenarios associated with the radionuclide(s) of concern. Radionuclides that can be connected to a specific function in a power reactor or gaseous diffusion plant, for example, will have a very specific contamination pattern or scenario based on the materials and processes involved. For example, ^{55}Fe and ^{54}Mn are activation-corrosion products, which can be found in irradiated metals from reactors (e.g., core shrouds, support plates, and core barrels), but it is unlikely that facilities would be attempting to clean (if possible) and release these materials. The more likely scenario involves materials that are associated with items that are not typically linked with any process that would expose them to radiation (e.g., neutrons) or radionuclides. Such items include structural materials (e.g., wood and steel), tools, pipework, heating and ventilation ductwork, and office equipment. Contamination found on these materials is most likely a result of the inadvertent movement of radionuclides by personnel and circulating air. However, it is clear in the case of reactor facilities that the radionuclides ^{60}Co, ^{55}Fe, ^{63}Ni and ^{54}Mn are associated with steel. Tritium (^{3}H) is the most mobile and is usually in the form of tritiated water when released. This means it can penetrate porous materials (such as concrete and wood) and form oxide layers on metals. In general, soluble radionuclides can penetrate porous materials to create contamination at depth. They can also become airborne and be transported by air currents to remote and inaccessible areas. Fine particles created by machining operations can become airborne and be deposited in cracks and on horizontal surfaces. With the exception of the corrosion-activation products, most of the contamination will reside on surfaces of various materials.

To summarize, the nature of contamination on solid material can be described in terms of its distribution on the material. For example, the contamination distribution on most items and materials is generally spotty, although some materials (particularly those that were designed to have intimate contact with radioactivity) exhibit a more uniform contamination distribution. This is an important consideration when selecting the clearance survey approach. Scanning is the preferred clearance survey methodology, precisely for its ability to detect the predominantly spotty contamination on solid materials.

4.4 Classification

All materials can be divided into two types—non-impacted and impacted. Non-impacted solid materials have no contamination potential based on process history, while impacted solid materials have some contamination potential based on operations and process knowledge. Impacted materials are further subdivided into three classes based on the materials' known contamination levels or contamination potential, as outlined in the following subsections.

The classification of solid materials is used to determine the clearance survey coverage for that material. The basic philosophy is that the greater the potential for the material to have contamination, the greater the clearance survey effort. This is the philosophy in the MARSSIM, as well. The solid material classification will specify, for example, how much metal scrap on a pallet must be surveyed, or what fraction of soil must be processed through a conveyorized survey monitor.

Improper classification of materials has serious implications, particularly when it leads to the release of materials with contamination in excess of clearance criteria. For example, if materials are mistakenly thought to have a very low potential for having contamination, these materials will be subjected to a minimal survey rigor. This misclassification results in a higher potential for releasing materials in error. To minimize these potential errors, investigation levels should be established and implemented to indicate when additional investigations are necessary. For example, a measurement that exceeds an appropriately set investigation level may indicate that the material survey unit has been improperly classified.

4.4.1 Class 1 Solid Materials

Class 1 solid materials are those materials that have (or had) a potential for contamination (based on process knowledge) or known contamination (based on previous surveys) above the release criterion ($DCGL_C$). These solid materials include materials that comprise processing equipment or components that may have been affected by a spill or airborne release.

Basically, Class 1 solid materials are those materials that were in direct contact with radioactive materials during the operations of the facility or may have become activated. Additionally, solid materials that have been cleaned to remove contamination are generally considered to be Class 1. An exception may be considered if there are no inaccessible areas and any contamination is readily removable using cleaning techniques. Examples of such methods may include vacuuming, wipe downs, or chemical etching that confidently remove all contamination such that surface activity levels would be less than the release criteria. Documented process knowledge of these cleaning methods should be provided to justify this exception to the cognizant regulatory authorities.

4.4.2 Class 2 Solid Materials

Class 2 solid materials are those materials that have (or had) a potential for or known contamination, but are not expected to have concentrations above the release criteria. These materials include those items that are within radiologically posted areas, but are not expected to have contamination. This class of materials might consist of electrical panels, water pipe, conduit, ventilation ductwork, structural steel, and other materials that might have come in contact with radioactive materials.

Any Class 2 solid materials that exceed the release criteria, based on previous surveys, should be reclassified as Class 1 for clearance surveys. For items of unknown or questionable origin, scoping surveys should be performed to determine whether residual surface contamination is present. Provided that no activity is identified, the minimum classification for such materials should be Class 2.

4.4.3 Class 3 Solid Materials

Class 3 solid materials are those materials that either are not expected to contain any contamination, or are expected to contain contamination less than some small specified fraction of the release criteria based on process knowledge or previous surveys. Any solid materials that exceed the specified fraction of the release criteria, from previous surveys, should be reclassified as Class 2 for clearance surveys. Additionally, if the historical assessment data are insufficient to clearly document that an item or area is non-impacted, the minimum classification for such materials would be Class 3.

4.5 Application of Release Guidelines

Section 4.1 discussed release guidelines for clearance and the concept of the derived concentration guideline limit for clearance ($DCGL_C$) based on dose factors, such as from NUREG-1640. This section addresses how individual DCGLs for clearance can be combined and applied when more than one radionuclide is potentially present. Options may include the use of gross activity DCGLs for surface activity compliance and use of surrogate measurements or the unity rule for volume activity compliance.

Regardless of the option used to modify the DCGLs to account for multiple radionuclides, it is necessary to identify the potential radionuclides, as well as the relative ratios of these radionuclides, if a relative ratio indeed exists. Section 4.3.2 discusses the approach for determining the nature of the contamination, as well as calculating the relative ratios among the multiple radionuclides and state of equilibrium for decay series radionuclides.

4.5.1 Surface Activity Assessment when Multiple Radionuclides are Present

Surface activity DCGLs for clearance apply to the total surface activity level. For cases in which the surface contamination is entirely attributable to one radionuclide, the $DCGL_C$ for that radionuclide is used for comparison to clearance data. The clearance data may be obtained from direct measurements of surface activity, scanning with data logging, CSM surveys, etc.

For situations in which multiple radionuclides with their own DCGLs are present, a gross activity $DCGL_C$ can be developed. This approach enables field measurement of gross activity (using static direct measurements or scans), rather than determination of individual radionuclide activity, for comparison to the $DCGL_C$. The gross activity DCGL for surfaces with multiple radionuclides is calculated as follows:

(1) Determine the relative fraction (f) of the total activity contributed by the radionuclide.
(2) Obtain the $DCGL_C$ for each radionuclide present.
(3) Substitute the values of f and $DCGL_C$ in the following equation.

$$Gross\ Activity\ DCGL_C = \frac{1}{\left(\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \ldots \frac{f_n}{DCGL_n}\right)}$$

For example, assume that 40 percent of the total surface activity was contributed by a radionuclide with a $DCGL_C$ of 1.4 Bq/cm^2 (8,300 dpm/100 cm^2); 40 percent by a radionuclide with a $DCGL_C$ of 0.3 Bq/cm^2 (1,700 dpm/100 cm^2); and 20 percent by a radionuclide with a $DCGL_C$ of 0.1 Bq/cm^2 (830 dpm/100 cm^2). Using the above equation,

$$Gross\ Activity\ DCGL_C = \frac{1}{\frac{0.40}{1.4} + \frac{0.40}{0.3} + \frac{0.20}{0.1}}$$

$$= 0.3\ \text{Bq/cm}^2\ (1{,}900\ \text{dpm/100 cm}^2)$$

Note that the above equation may not work for sites that exhibit surface contamination from multiple radionuclides having unknown or highly variable concentrations of radionuclides throughout the site. In these situations, the best approach may be to select the most conservative surface activity DCGL from the mixture of radionuclides present. If the mixture contains radionuclides that cannot be measured using field survey equipment, such as ^{3}H or ^{55}Fe, laboratory analyses of solid materials may be necessary.

Meeting with surface activity DCGLs for radionuclides of a decay series (e.g., radium, thorium, and uranium) that emit both alpha and beta radiation may be demonstrated by assessing alpha, beta, or both radiations. However, relying on the use of alpha surface activity measurements often proves problematic because of the highly variable level of alpha attenuation by rough, porous, and dusty surfaces. Beta measurements typically provide a more accurate assessment of thorium and uranium contamination on most building surfaces because surface conditions cause significantly less attenuation of beta particles than alpha particles. Beta measurements, therefore, may provide a more accurate determination of surface activity than alpha measurements.

The relationship of beta and alpha emissions from decay chains or various enrichments of uranium should be considered when determining the surface activity for comparison with the $DCGL_C$ values. When the initial member of a decay series has a long half-life, the radioactivity associated with the subsequent members of the series will increase at a rate determined by the individual half-lives until all members of the decay chain are present at activity levels equal to the activity of the parent. This condition is known as secular equilibrium.

Consider an example in which the radionuclide of concern is ^{232}Th, and all of the progeny are in secular equilibrium. Assume that a gas proportional detector will be used for surface activity measurements. The detector's efficiency is dependent upon the radionuclide mixture measured and the calibration source area (greater than 100 cm^2 area calibration sources are recommended). The ^{232}Th efficiency is calculated by weighting the individual efficiencies from each of the radionuclides present (see Table 4.3). This value is greater than 100 percent because of all of the progeny that are assumed to be in equilibrium with the ^{232}Th. It is important to recognize that if the $DCGL_C$ for ^{232}Th includes the entire ^{232}Th decay series, the total efficiency for ^{232}Th must account for all of the radiations in the decay series.

Table 4.3: Detector efficiency for the rare earth facility (^{232}Th in complete equilibrium with its progeny) using a gas proportional detector

Radionuclide	Average Energy (keV)	Fraction	Instrument Efficiency	Surface Efficiency	Weighted Efficiency
^{232}Th	alpha	1	0.40	0.25	0.1
^{228}Ra	7.2 keV beta	1	0	0	0
^{228}Ac	377 keV beta	1	0.54	0.50	0.27
^{228}Th	alpha	1	0.40	0.25	0.1
^{224}Ra	alpha	1	0.40	0.25	0.1
^{220}RN	alpha	1	0.40	0.25	0.1
^{216}Po	alpha	1	0.40	0.25	0.1
^{212}Pb	102 keV beta	1	0.40	0.25	0.1
^{212}Bi	770 keV beta	0.64	0.66	0.50	0.211
^{212}Bi	alpha	0.36	0.40	0.25	0.036
^{212}Po	alpha	0.64	0.40	0.25	0.064
^{208}Tl	557 keV beta	0.36	0.58	0.50	0.104
					Total efficiency = 1.29

4.5.2 Volume Activity Assessment when Multiple Radionuclides are Present

Typically, DCGLs correspond to a release criterion (e.g., a regulatory limit) in terms of dose or risk. However, in the presence of multiple radionuclides, the total of the DCGLs for all radionuclides could exceed the release criterion. In this case, the individual DCGLs would need to be adjusted to account for the presence of multiple radionuclides contributing to the total dose. One method for adjusting the DCGLs is to modify the assumptions made during exposure pathway modeling to account for multiple radionuclides. The surrogate measurements discussed in this section describe another method for adjusting the DCGL to account for multiple radionuclides when radionuclide-specific laboratory analyses of media samples or *in toto* measurements are performed. Other methods include the use of the unity rule and development of a gross activity DCGL for surface activity to adjust the individual radionuclide DCGLs.

The unity rule, represented in the following expression, is satisfied when radionuclide mixtures yield a combined fractional concentration limit that is less than or equal to one:

$$\frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \ldots \frac{C_n}{DCGL_n} \leq 1$$

where

C = concentration

DCGL = clearance guideline value for each individual radionuclide (1, 2, ... n)

For the clearance of solid materials that have potential contamination with multiple radionuclides, it may be possible to measure just one of the radionuclides and still demonstrate compliance for all of the other radionuclides present through the use of surrogate measurements. In the use of surrogates, it is often difficult to establish a "consistent" ratio between two or more radionuclides. Rather than follow prescriptive guidance on acceptable levels of variability for the surrogate ratio, a more reasonable approach may be to review the data collected to establish the ratio and to use the DQO Process to select an appropriate ratio from that data. The $DCGL_C$ must be modified to account for the fact that one radionuclide is being used to account for one or more other radionuclides.

The following equation illustrates how the DCGL for the measured radionuclide is modified ($DCGL_{meas,mod}$) to account for the inferred radionuclide:

where

$$DCGL_{meas,\mathrm{mod}} = (DCGL_{meas}) \left(\frac{(DCGL_{infer})}{\left(\frac{C_{infer}}{C_{meas}} \right) DCGL_{meas} + DCGL_{infer}} \right)$$

C_{infer}/C_{meas} = surrogate ratio for the inferred to the measured radionuclide

When it is necessary for the measured radionuclide to be used as a surrogate for more than one radionuclide, Equation I-14 on MARSSIM page I-32 can be used to calculate the modified DCGL for the measured radionuclide:

$$DCGL_{meas,\mathrm{mod}} = \frac{1}{\left(\frac{1}{D_1} + \frac{R_2}{D_2} + \frac{R_3}{D_3} + \ldots \frac{R_n}{D_n} \right)}$$

where D_1 is the $DCGL_C$ for the measured radionuclide by itself, D_2 is the $DCGL_C$ for the second radionuclide (or first radionuclide being inferred) that is being inferred by the measured radionuclide. R_2 is the ratio of concentration of the second radionuclide to that of the measured radionuclide. Similarly, D_3 is the $DCGL_C$ for the third radionuclide (or second radionuclide being inferred) that is being inferred by the measured radionuclide, and R_3 is the ratio of concentration of the third radionuclide to that of the measured radionuclide.

Recall that the benefit of using surrogates is the avoidance of costly laboratory-based analytical methods to detect radionuclides with weakly penetrating radiation. Surrogates usually emit γ rays, which enable the use of noninvasive and nondestructive methods. The surrogates come in two forms: (1) surrogates by virtue of a decay series, and (2) surrogates by virtue of association. The difficulty with surrogates that are part of a series is that a time for sufficient number of half-lives of the longest lived progeny that intervenes between and including itself and its parent must pass in order to establish secular equilibrium. In the case of ^{232}Th, this is almost 40 years. This is because ^{232}Th decays into ^{228}Ra, which has a half-life of 5.75 years. In the case ^{238}U and ^{226}Ra, the half-lives of the intervening progeny are relatively short. However, ^{226}Ra possesses a special problem because it decays into ^{222}Rn, which is a noble gas that can escape the matrix and disrupt equilibrium. Radionuclides that are not part of a decay series have the potential to be surrogates because they are produced by the same nuclear process (usually fission or activation) and have similar chemical properties and release mechanisms. However, this type of surrogate needs some special attention because there must be a consistent ratio between the measured radionuclide and surrogate, which is not always easy to demonstrate. For example, in the case of reactors, ^{60}Co can be used as a surrogate of ^{55}Fe and ^{63}Ni because both are activation-corrosion products with similar chemical properties. Similarly, ^{137}Cs can be used as a surrogate for the β-emitting ^{90}Sr because both are fission products and are generally found in soluble cationic forms. While ^{137}Cs has been suggested as a possible surrogate for ^{99}Tc, it must be noted that ^{99}Tc does not have different chemical properties and, in power reactors, it has different release mechanisms. For a further discussion of surrogates and establishing ratios, see MARSSIM (1997) and Best and Miller (1987).

4.6 Measurability of Contamination

Detection limits for field survey instrumentation are an important criterion in the selection of appropriate instrumentation and measurement procedures. For the most part, detection limits need to be determined in order to evaluate whether a particular instrument or measurement procedure is capable of detecting residual activity at the regulatory release criteria (DCGLs). For example, the MARSSIM recommends that the minimum detectable concentration (MDC) should be sufficiently less than the DCGL (e.g., no greater than 10 to 50 percent of the DCGL). This is a reflection of two concerns. First, when calculated a priori, the MDC frequently tends to be optimistic in that some factors that may adversely impact detection sensitivity are either unknown or not included (e.g., surface roughness, interfering radionuclides, or radiations). Second, the objective is not simply to detect whether radioactivity exists at levels approaching the DCGL, but to quantify the actual concentration level within a reasonable overall uncertainty.

Sections 4.6.1 and 4.6.2 address the measurability of contamination under the general survey approaches of (1) static measurements and (2) scanning, respectively. Static MDCs are calculated when the clearance survey approach includes conventional direct measurements of surface activity, *in toto* measurements, or laboratory analyses of media samples. Scan MDCs are calculated when the clearance survey approach includes scanning with conventional detectors, or when using automated scanning equipment such as the conveyorized survey monitor.

4.6.1 Static MDCs

The measurement of contamination during clearance surveys often involves measuring contamination at near-background levels. Thus, it is essential to determine the minimum amount of radioactivity that may be detected using a given survey instrument and measurement procedure. In general, the MDC is the minimum activity concentration on a surface, or within a material volume, that an instrument is expected to detect (e.g., activity expected to be detected with 95-percent confidence). It is important to note, however, that this activity concentration, or MDC, is determined a priori (that is, before survey measurements are conducted).

The MDC corresponds to the smallest activity concentration measurement that is practically achievable with a given instrument and type of measurement procedure. That is, the MDC depends on the particular instrument characteristics (efficiency, background, integration time, etc.), as well as the factors involved in the survey measurement process, which include surface type, source-to-detector distance, source geometry, and surface efficiency (backscatter and self-absorption). More information on detectability, detection limits, and formulas to compute MDCs is available in the literature (Currie 1968, NRC 1984, Brodsky 1992 and 1993, Chambless 1992, ANSI 1996, ISO 2000a and b).

The methodology to determine an MDC for a given instrument, radionuclide, matrix or surface, and measurement protocol is based on the specific formulation of the MDC for the application in question. For example, the formula for calculating the MDC for a technician scanning copper tubing for alpha contamination would be different than the formula for calculating the MDC for ^{137}Cs in soil using a shielded gamma-ray spectrometer. However, all forms of the MDC equation do have the following structure (NCRP 1985):

$$MDC = k \frac{detection\ limit}{efficiency\ x\ sample\ size} \qquad (4\text{-}1)$$

where k is a unit conversion (from instrument response to activity and the desired units).

The detection limit considers both the instrument background and backgrounds from other sources, such as interfering radiations from the environment (both natural and anthropogenic), in determining the response of the instrument that is statistically different from background. This detection limit is determined using a statistical hypothesis test with a specified gray region and Type I and Type II errors. The overall uncertainty of the measurement process when measuring a blank sample is a key parameter for determining realistic detection limits.

The efficiency term includes the efficiency associated with the detector (instrument or intrinsic efficiency), geometrical efficiency, surface or sample efficiency, absorption efficiency, and, in some applications, surveyor efficiency (see Section 4.6.2). The surface efficiency accounts for field conditions such as rusty metal, damp surfaces, or scabbled concrete.

The sample size term takes on different values depending on the type of measurement. For field survey instruments, this is usually well-defined as the physical probe area of the detector. For laboratory measurements, it is again a well-defined quantity defined as a measured amount of the sample. However, in the case of an *in situ* or *in toto* measurement, the sample size is a function of the detector's field-of-view, which is usually not well-defined (or difficult to define accurately). Section 5.4 further addresses MDC issues for the *in situ* gamma spectrometer used to release materials.

The following equation is used to calculate the MDC for surface activity assessments using conventional survey instrumentation (NRC 1998a):

$$MDC = \frac{3 + 4.65\sqrt{C_B}}{KT} \quad (4\text{-}2)$$

where C_B is the background count in time, T, for paired observations of the sample and blank. The quantities encompassed by the proportionality constant, K, include the instrument efficiency, surface efficiency, and probe geometry. Based on the radionuclides of concern, specific instrument and surface efficiencies are used to calculate the static MDC for surface activity measurements. The MDC is also a function of the surface material background level and, therefore, varies with the nature of the surfaces being surveyed.

The detection and detectability of contamination when using other than the conventional survey approach must also be considered. Tritium (^{3}H) and ^{14}C create a significant challenge for detection (because of the associated low instrument efficiency). They each emit a low-energy β radiation, and they are not amenable to the surrogate approach. Similarly, ^{63}Ni and ^{99}Tc are somewhat difficult to detect because they too have primary radiations of low-energy betas. Conversely, ^{60}Co, Cs-134, and ^{137}Cs (via Ba-137m) are easily detected because of their intense and rather energetic gamma-rays and readily-measured beta radiations. T he evaluation of detectability for these seven radionuclides is more or less independent of the matrix and nature of the contamination. In general, all of the radionuclides (with the exception of ^{3}H) can be detected with hand-held devices using standard survey methods. The issue is whether hand-held devices and standard survey methods can detect these radionuclides, separately or in combination, at the levels established for release.

Therefore, the recipe to calculate the MDC for any measurement method (such as for an *in toto* technique or laboratory analysis) is to determine the detection limit, relevant efficiencies, and sample size for the given instrument and measurement protocol. For some of the more common (conventional) techniques of measuring radionuclides and materials, these quantities have been either measured, calculated, or estimated and MDCs are available in the literature (ANSI 1999, MARSSIM 1997, NRC 1998a, EC 1998, and Goles *et al.* 1991). The reader should note, however, that the MDC provided in these references apply only to the situation described and must not be construed to be a universal MDC for a particular instrument or protocol. Rather, they should be viewed only as a general measure of the capability of the instruments for the application described.

4.6.2 Scanning-Based MDCs

Scanning-based MDCs must also be assessed in order to appropriately design the clearance survey approach. Relevant information on scanning-based MDCs for conventional survey approaches exists in the MARSSIM (Section 6), NUREG-1507, and Abelquist and Brown, 1999. In general, when planning surveys, one must often consider minimum detectable count rates (MDCRs) in order to evaluate the effectiveness of a given scan. An MDCR is an a priori estimate of the signal level that a real surveyor is expected to recognize as having a signal-to-noise ratio that is distinctly above the ambient detector background noise. In general, the MDCR is defined as the detector signal level, or count rate for most equipment, that a surveyor is likely to flag as being "greater than background." The MDCR will depends on a number of factors, including scan speed, detector type, detector background, and surveyor performance.

4.6.2.1 Hand-Held Detector Scan MDCs

To illustrate the calculation of scanning-based MDCs, the scanning sensitivity for conventional hand-held survey instruments is provided for materials being cleared from a gaseous diffusion facility. [Note: Example 2 in Section 5 of this report pertains to nuclear power plants.] Assuming that a gas proportional detector is used as the primary instrument used for surface scanning, the instrument efficiency for scanning is slightly less than that used for static measurements. This is because the detector is not directly on the surface of the material during scanning. [Note: The fact that the detector is being moved over the source is separately accounted for in the scan efficiency by determining the observation interval. The instrument efficiency for scanning is determined based on the detector-surface geometry for the observation interval, which is on the order of seconds.] Table 4.4 shows the determination of detection efficiency for a gas proportional detector used for scanning.

Table 4.4: Detector efficiency when scanning for GDP-enriched uranium (1.2%) and ^{99}Tc using a gas proportional detector (0.4 mg/cm^2 window)

Radionuclide	Radiation/Average Energy (MeV)	Activity Fraction	ε_i	ε_s	Weighted Efficiency
^{99}Tc	Beta/0.085	0.7082	0.30	0.25	5.3×10^{-2}
^{238}U	Alpha/4.2	0.1077	0.32	0.25	8.6×10^{-3}
^{234}Th	Beta/0.0435	0.1077	0.20	0.25	5.4×10^{-3}
^{234m}Pa	Beta/0.819	0.1077	0.58	0.50	3.1×10^{-2}
^{234}U	Alpha/4.7	0.1728	0.32	0.25	1.4×10^{-2}
^{235}U	Alpha/4.4	0.0084	0.32	0.25	6.7×10^{-4}
^{231}Th	Beta/0.0764	0.0084	0.29	0.25	6.1×10^{-4}
		Total Weighted Efficiency			**0.11**

The scan MDC for structure surfaces may be calculated as

$$scan\ MDC = \frac{MDCR}{\sqrt{p}\ \varepsilon_i\ \varepsilon_s} \quad (4\text{-}3)$$

where the minimum detectable count rate (MDCR), in counts per minute, can be written as

$$MDCR = d' * \sqrt{b_i} * (60/i) \quad (4\text{-}4)$$

where d' = detectability index (the value can be obtained from Table 6.5 in the MARSSIM),
b_i = background counts in the observation interval,
i = observational interval (in seconds), based on the scan speed and areal extent of the contamination (usually taken to be 100 cm^2),
ε_i is the instrument or detector efficiency (unitless),
ε_s is the surface efficiency (unitless), and
p is the surveyor efficiency (usually taken to be 0.5).

Consider an example that involves determining the gas proportional scan MDC for the GDP mixture on concrete slabs. The scan MDC will be determined for a background level of 400 cpm and a 1-second observation interval. For a specified level of performance at the first scanning stage of 95-percent "true positive" rate and 25-percent "false positive" rate, d' equals 2.32 (from Table 6.5 in the MARSSIM), and the MDCR is calculated as follows:

$$b_i = (400\ cpm)(1\ s)(1\ min/60\ s) = 6.67\ counts,$$

$$s_i = (2.32)(6.67)^{1/2} = 6.0\ counts, \text{ and}$$

$$MDCR = (6.0\ counts)[(60\ s/min)/(1\ s)] = 360\ cpm.$$

Using a surveyor efficiency of 0.5 and the total weighted efficiency determined in Table 9.1 (0.11), the scan MDC is calculated as

$$scan\ MDC = \frac{360}{\sqrt{0.5}\ (0.11)} = 4{,}600\ dpm/100\ cm^2\ \ (0.77\ Bq/cm^2)$$

A Geiger-Mueller (GM) detector is often used to scan material surfaces that are difficult (or impossible) to access using the larger gas proportional detector. The efficiency of a GM detector in scanning this radionuclide mixture can be determined in a manner similar to that used in Table 4.4. It is important to note, however, that the scan MDC calculations usually require the assumption that the instrument efficiencies are determined relative to a 100-cm^2 calibration source to yield the appropriate units (dpm/100 cm^2). This is in contrast to the static MDC equation, which uses a physical probe area correction in the calculation of surface activity.

Table 4.5 provides instrument efficiencies that correspond to a 100-cm^2 calibration source, without reducing the 2π emission rate for the smaller area subtended by the GM detector. [Note: This is precisely what would be performed for static measurements of surface activity.] In other words, as long as 100 cm^2 is used as the size of the postulated small, elevated area, and the instrument efficiency is calculated for the same area, there is no need for a probe area correction in the scan MDC equation.

Table 4.5: Detector efficiency when scanning for GDP-enriched uranium (1.2%) and ^{99}Tc using a GM detector

Radionuclide	Radiation/Average Energy (MeV)	Activity Fraction	ε_i	ε_s	Weighted Efficiency
^{99}Tc	Beta/0.085	0.7082	0.05	0.25	8.9×10^{-3}
^{238}U	Alpha/4.2	0.1077	0.02	0.25	5.4×10^{-4}
^{234}Th	Beta/0.0435	0.1077	0.025	0.25	6.7×10^{-4}
^{234m}Pa	Beta/0.819	0.1077	0.12	0.50	6.5×10^{-3}
^{234}U	Alpha/4.7	0.1728	0.02	0.25	8.6×10^{-4}
^{235}U	Alpha/4.4	0.0084	0.02	0.25	4.2×10^{-5}
^{231}Th	Beta/0.0764	0.0084	0.045	0.25	1.8×10^{-5}
		Total Weighted Efficiency			**0.018**

As an example, consider evaluating the scanning-based MDC for the gaseous diffusion plant (GDP) mixture on stainless-steel materials. The scanning-based MDC will be determined for a background level of 70 cpm and a 1-second interval using a GM detector. For a specified level of performance at the first scanning stage of 95-percent true positive rate and 25-percent false positive rate, d' equals 2.32 (from Table 6.5 in the MARSSIM), and the MDCR is calculated as follows:

$$b_i = (70\ cpm)(1\ s)(1\ min/60\ s) = 1.2\ counts,$$

$$s_i = (2.32)(1.2)^{1/2} = 2.5\ counts, \text{ and}$$

$$MDCR = (2.5\ counts)[(60\ s/min)/(1\ s)] = 150\ cpm.$$

Using a surveyor efficiency of 0.5 and the total weighted efficiency determined in Table 9.2 (0.018), the scan MDC is calculated as

$$scan\ MDC = \frac{150}{\sqrt{0.5}\ (0.018)} = 12{,}000\ dpm/100\ cm^2\ \ (2\ Bq/cm^2)$$

4.6.2.2 Conveyor Survey Monitor Scan MDCs

The scan MDC for a CSM can be estimated using Equation 4-1, with some modification to account for the automated nature of a CSM. That is, the parameters that impact the CSM scan MDC include the detection limit, efficiency, and sample size. The detection limit is based on the background counts obtained over the counting interval and the acceptable rate of true (correct detection) and false positives. The background level depends on the nature of the material, while the counting interval is a function of both the detector's field-of-view and the system belt speed (i.e., it establishes the length of time that the detector(s) can respond to a fixed length of material). Basically, the MDCR can be calculated for the CSM in much the same manner as it is for conventional scans, with the primary difference being that automated systems interpret the signal stream (data) using a computer-based analysis algorithm rather than by calculation (Equation 4-4).

Sample or survey unit size is a function of the belt geometry, speed (which establishes the observation interval), and the detector's field-of view and, therefore, has a fundamental impact on the scanning detection limit (cpm) and MDC (Bq/g) of a CSM. The detection efficiency for a CSM depends on the detector characteristics, nature of the contamination, the material being surveyed, and source-to-detector geometry. Modeling was performed to support the determination of beta detection efficiencies for automated scanning systems, as further discussed in Section 5.3.

4.6.2.3 Empirical Determinations of Scanning-Based MDCs

Empirical determination of scanning-based MDCs can serve as an alternative to calculation. That is, it is possible to design experiments to assess (and empirically determine) the scanning-based MDCs for particular survey instruments and scan procedures. A number of researchers, as well as R&D professionals, have developed mockups of surfaces with contamination to determine scanning-based MDCs. For instance, in a study by Goles *et al.* (1991), empirical results included MDCRs as a function of background levels: 305 net cpm detected in 50-cpm background level, 310 cpm in 250-cpm background, and 450 cpm in 500-cpm background. It is important to note that these MDCRs were quoted for detection frequencies of 67 percent (compared to the usual 95 percent). Empirical assessments of scanning-based MDC can also be valuable for determining the scanning capabilities of specific survey technicians.

The uncertainty in the scanning-based MDCs calculated using the approaches described in this section should be viewed in the context of their use. That is, scanning-based MDCs are used to help design the clearance survey approach, and should represent a "reasonable estimate" of the activity concentration that can be detected when scanning. In other words, while the scanning-based MDC should be carefully assessed, it is important to remember that such MDCs are inherently subject to uncertainties (e.g., human factors, unknown characteristics of contamination prior to survey, variable background levels, etc.). Recognizing this uncertainty in the scanning-based MDCs, it is worthwhile to consider additional means of evaluating these values.

Empirical evaluation of scanning-based MDCs can also be an important validation tool. This validation is performed by assessing the contamination levels that are flagged on solid materials during scanning. These radionuclide concentrations are evaluated by direct measurements or laboratory analyses, and the concentrations at the lower end of the range of results should provide a reasonable estimate of the scanning-based MDC achieved. That is, an empirical evaluation might indicate that the lower values in the range represent a ballpark estimate of the scanning-based MDC. Obviously, increasing the number of samples that are actually flagged during the scan, as well as the number of subsequently measured samples will improve the accuracy of this empirical assessment of scanning-based MDCs.

4.7 Inaccessible Areas

A question that often arises is how to handle the release of materials that have inaccessible areas that may have contamination. If the material surfaces are inaccessible, then by definition, it is not possible to demonstrate that release criteria have been satisfied using conventional survey activities. In such cases, a couple of options exist. First, the material might not be released for unrestricted use; that is, the surveyor might conclude that since surfaces are not accessible, they must be assumed to have contamination at levels greater than the release criteria. Thus, the materials might be disposed of as radioactive waste. In fact, this approach has been used to deal with materials that have inaccessible surfaces.

A second alternative might be to make the surfaces accessible, either by cutting or dismantling the material, or by using specialized survey equipment (e.g., small detectors). This option requires additional resources beyond those required for conventional clearance surveys. The discussion throughout this report suggests a number of research opportunities for handling materials that have inaccessible areas.

4.7.1 Inaccessible Material Scenarios

It is important to recognize the various inaccessible material scenarios that can occur during the clearance of materials. Perhaps the most common scenario is when contamination exists on the interior surfaces of scrap equipment, such pumps, motors, and other equipment. These items can become contaminated through a number of mechanisms, including their operation in airborne contamination areas where air is drawn into the equipment, thereby contaminating internal surfaces. Similarly, contaminated lubricating oil can spread contamination to a number of components within the scrap equipment. Thus, because of the small openings on these items, it is nearly impossible to use conventional survey activities to assess the potential for internal contamination.

Another inaccessible material scenario involves contamination on the interior surfaces of pipes that are difficult to access, such as buried or embedded pipes. Buried and embedded pipes may become contaminated as a result of their function of transporting radioactive liquids or gases. Buried pipes are usually at some depth beneath the soil surface and cannot be accessed unless they are excavated. Process piping, such as that associated with nuclear power reactor systems, can be embedded in concrete, which further complicates the assessment. In addition, the small diameter of embedded piping typically makes it extremely difficult to access the interior surfaces.

One final inaccessible material scenario includes some of the material surfaces in a scrap metal (or other material) pile. This complex geometry is somewhat different from the first two scenarios, in that these surfaces can be made accessible, but separating the materials for examination might be considered too labor-intensive to warrant conventional clearance surveys. Therefore, it might be worthwhile to consider releasing a pile of scrap metal by taking *in situ* gamma spectrometry measurements of the scrap metal pile. In this case, some of the scrap metal surfaces are considered to be inaccessible because they do not directly contribute to the detector's response. However, provided that a sufficient fraction of gamma radiation from the contamination is detected, *in situ* gamma spectrometry might provide a reasonable clearance technique for scrap metal piles. (Refer to Section 5.4 for a discussion of this survey approach.)

4.7.2 Making an Inaccessible Area Accessible

As previously indicated, one strategy that can be considered when dealing with materials that have inaccessible areas is to make the inaccessible areas accessible. For example, this can be accomplished by dismantling scrap equipment or by excavating buried or embedded pipes. Inaccessible areas that might require disassembly include small pumps, motors, hand tools, power tools, and electrical control panels. These materials are assumed to require some amount of disassembly to allow access to their interior surfaces. The dismantling might be deliberate to ensure that the item is still functional following the efforts to gain access to internal surfaces. Conversely, cutting techniques can be employed to expedite the process if reuse is not an option.

Another technique that may be considered is the use of thermoluminescent dosimeters (TLDs) or small detectors to measure surface activity levels within buried and embedded piping systems. TLDs can be deployed for some period of time within small bore piping or conduit to respond to the contamination levels on the interior surfaces. An important aspect of this application is the calibration of the TLDs to surface activity in the given pipe geometries. Small detectors, such as miniature GM detectors, and other "pipe-crawling" detector systems have been used to assess surface contamination in pipe systems.

Nondestructive assay (NDA) is any quantitative technique that does not require sampling or sample preparation, and will not alter the physical or chemical state of the object being measured. NDA techniques have been developed and used on nuclear fuel materials, transuranic waste, soils, and scrap metal. The two basic approaches to NDA involve passive and active techniques. A passive technique involves directly measuring the spontaneous decay of nuclear material, while an active technique attempts to excite atoms and molecules to emit characteristic radiation that can be measured and used for identification and quantification. With the exception of nuclear activation analysis, active techniques cannot distinguish between nuclear isotopes like some passive techniques. However, active techniques are potentially more sensitive than passive techniques associated with decay counting. In general, NDA techniques are less sensitive than laboratory techniques.

5 CLEARANCE SURVEY APPROACHES

As discussed in previous sections of this report, the predominant factor in determining how much effort should be expended in conducting a clearance survey to release the given solid material is the material's potential to have contamination in excess of the release criteria. That is, the closer the radionuclide concentration is to the release criteria, the greater the degree of survey effort that should be expended to release the material. Process knowledge and characterization activities are used to estimate the material's contamination potential. The MARSSIM survey approach can be applied to clearance of materials, by designating the materials as Class 1, 2, or 3 based on each material's contamination potential.

The decision to implement a particular clearance survey approach depends on the material characteristics, nature of the contamination, detectability of the emitted radiation, and availability of survey instrumentation. The reader is encouraged to revisit the DQO Process discussion in Section 3 before selecting a particular clearance survey approach.

5.1 Background Measurements

Release criteria for the clearance of solid materials may be expressed as the concentration of radioactivity that exceeds background levels. Consequently, an important aspect of clearance surveys is to adequately assess the background levels associated with specific solid materials. This can be achieved by selecting background reference materials that are non-impacted (i.e., materials that have no reasonable potential to be contaminated) and representative of the solid materials being considered for release. Background measurements are also necessary to calculate the MDC of the selected clearance survey approach.

The number and type of background measurements that are necessary to support the design of clearance surveys depends on the particular clearance survey approach, the survey instrument, and the nature of the solid material. The number of background measurements should be based on the requirements of the statistical test (if a statistical test is used) or on the DQO Process. [Note: If background levels are a small fraction of the release criteria, one might consider ignoring the background in demonstrating compliance. Refer to Section 6 for more information on this conservative practice.]

Background surface activity levels for instrumentation used to measure beta radiation can be expected to vary in response to a number of influences. The primary variance is attributable to survey conditions (such as gamma contributions from ambient environmental and building materials), while variations in the solid materials themselves and temporal fluctuations attributable to sources such as radon can add additional variance. Backgrounds for alpha-measuring instrumentation can be expected to vary primarily as a result of natural material contributions and temporal variations in radon, where radon concentrations tend to be elevated. In all cases, surveys should be performed in areas where instrument backgrounds from ambient radiation levels allow the detection sensitivity requirements to be met.

Appropriate background data sets should be collected for each detector type, such that all significant sources of variance are properly accounted for. Background measurements should be collected on material types representing items that will be surveyed and should also account for fluctuations within the area where surveys will be performed. Although not required, it is suggested that data sets be formed for beta-gamma detection equipment by collecting measurements on non-impacted solid materials at varying locations to establish a good representation of background variance. For those areas where radon progeny or other external influences on detector response may pose a significant problem, it is suggested that the materials be moved elsewhere before being surveyed.

Dependent upon site- and material-specific considerations, the background data sets may be pooled or analyzed individually according to material types. The mean and variance of the background measurements would then be calculated for the complete data set(s). At a minimum, materials with very dissimilar background radiological properties should not be grouped together. For example, the background means for various metal types generally should not differ by greater than 30 percent in order to be considered for grouping.

Background measurements for the conveyor survey monitor should be determined for each type of non-impacted solid material being considered for release. For example, non-impacted soil could be run through the CSM repeatedly to develop a background database for that material. (Refer to Section 5.3.)

At least one ambient background measurement for the *in situ* gamma spectrometer (ISGS) should be performed in the area where clearance surveys will be conducted. This background spectrum should be collected for a sufficient time to provide the necessary sensitivity for the radionuclide(s) and material being considered for release. (Refer to Section 5.4.) Provided that the radionuclide(s) being measured are not naturally present in the solid material being assessed, additional ISGS background measurements are unwarranted. By contrast, when the radionuclide(s) being measured are naturally present in the solid material (e.g., uranium, thorium), a number of background measurements should be performed on the same type of non-impacted solid materials to permit comparison to the materials being released. It is likely that the number of background measurements required in this case will be based on WRS test data needs.

5.2 Survey Approach Using Conventional Instrumentation

In general, survey methods that use conventional instrumentation can be classified into three survey categories, which are commonly known as (1) scanning, (2) direct measurements of surface activity, and (3) smear and miscellaneous sampling. These survey approaches are based on the use of hand-held, portable field survey instruments, which should have a minimum measurement detection ability, typically referred to as minimum detectable concentration (MDC), that is less than applicable release criterion ($DCGL_C$). For difficult-to-detect radionuclides, the survey should use surrogates, or collection methods and laboratory analysis techniques, that have minimum detection abilities that are less than applicable release limits for media samples.

5.2.1 Survey Instrumentation

To maintain sufficient survey instrument detection capabilities, release surveys should be conducted in areas with low background radiation levels. Survey instrument parameters to consider include count times (for direct measurements of surface activity), background levels, and detection efficiencies to determine if they yield MDCs that are sufficiently below the release criteria to allow unambiguous decisions regarding the acceptability for release. Section 4.6 provides detailed information on measurability issues.

All measurement instrumentation should be calibrated and monitored for performance in accordance with accepted standards applicable to performing surveys before releasing materials from radiological control. Survey instruments typically include gas proportional, GM, ZnS, and NaI scintillation detectors, coupled to ratemeters or ratemeter-scalers with audible indicators. Calibration and efficiency data are necessary to ensure that individual detectors are capable of meeting the minimum performance specifications, as previously discussed.

5.2.2 Survey Activities (Measurement Methods)

As previously mentioned, conventional clearance survey methods include scanning, direct measurements, and sampling surveys. Given these options, the measurement techniques for a given clearance survey should be selected on the basis of the radionuclides (radiations) of concern and appropriately sensitive instrumentation should be selected for field use. The types of measurements, specific portable instrumentation, and specific measurement methods should be consistent with the appropriate standard operating procedures (SOPs) and presented in clearance survey plans.

5.2.2.1 Scanning and Direct Measurements of Surface Activity

Surface activity surveys are performed using both scans and static, integrated direct measurements. Clearance materials should be assessed on the basis of process knowledge and other historical information, and should also be scanned for alpha, beta, or gamma radiation according to the nature of the potential radionuclides. When pausing during scans, a surveyor should compare the resulting signal to the expected background level to determine whether the observation indicates an elevated radiation level. Any locations of elevated direct radiation should be marked for further investigation, which should include judgmental measurements of surface activity. Scans should be performed using survey instruments that have been appropriately calibrated for the radiations present. Appropriate investigation levels should be established and implemented for evaluating elevated radiation.

Direct measurements of surface activity should be performed for materials being considered for release. The type of surface activity measurement (gross alpha or gross beta) should be selected on the basis of the potential radionuclides present. Direct measurements should be performed using appropriately calibrated survey instruments, including gas proportional, GM, and ZnS detectors coupled to ratemeter-scalars. Material-specific background measurements should also be obtained for each material type. (Refer to Section 5.1.) In addition, all measurement locations should be properly documented on detailed survey maps.

5.2.2.2 Smear and Miscellaneous Sampling

Materials considered for release may include miscellaneous samplings, such as smear, residue, and/or swab samples, with the methods chosen on the basis of the inaccessibility of some surfaces. [Note: Given the significant variations in smear collection efficiencies, smear results are usually considered to be semi-quantitative]. Smear samples for the determination of removable activity may be collected at direct measurement locations. Residue and/or swab samples may also be collected at specific locations where the surface area is inaccessible for direct measurements.

The selected frequency of sampling should be based on the appropriate classification (based on surface area, minimum number per item), and measurement locations should be properly documented on detailed survey maps. Procedures and equipment used for sampling (smears, Q-tips, swabs, etc.) should be appropriate for the assessment of the contamination. A comprehensive reference on the use and purpose of smears is Frame and Abelquist, 1999.

5.2.3 Clearance Survey Designs Using Conventional Instrumentation

The following sections discuss various applications of the conventional survey approach based primarily on the capability of the survey instrumentation. These conventional survey applications include (1) scanning-only, (2) scanning and direct measurements, and (3) statistically based sampling. [Note: In the following discussion, the statistical term "sample" refers to both direct measurements of surface activity and media samples (smears, soil, etc.)].

As mentioned in Section 1.3, this report stresses the use of scanning to release materials whenever the scan MDC is sufficiently sensitive. As such, the conventional survey approaches discussed in the following sections are ordered in terms of relative ease in performing survey activities. That is, scanning-only is the most direct survey approach, followed by scanning and direct measurements, and lastly statistically based sampling. The NRC staff recognizes that constraints in the availability of specific survey instrumentation, in terms of scan sensitivity or ability to automatically record scanning results, may limit the conventional survey options that are available to the licensee. However, the reader should note that each of the techniques discussed in Sections 5.2.3.1 – 5.2.3.3 is equally acceptable for demonstrating the acceptable release of materials.

5.2.3.1 Scanning-Only

This clearance survey approach can be used to release solid materials only when two conditions are met. First, the survey instrumentation must exhibit sufficient scan sensitivity. That is, the scan MDC must be less than the $DCGL_C$. (Refer to Section 4.6 for guidance on determining the scan MDC for comparison to the $DCGL_C$.) Second, the survey instrumentation must have the capability to automatically document the survey results, which may be accomplished using a data logger or similar device. This condition cannot be satisfied by the surveyor manually recording the scan results; automatic documentation is much more reliable. (Manually recorded scan results are a function of the surveyor's memory.)

The scan coverage should be graded based on the material's classification. That is, 100 percent of surfaces should be scanned for Class 1 materials, 50 to 100 percent for Class 2, and 10 to 50 percent for Class 3. The size of the material survey unit may also be a function of the material's classification. That is, the amount of material comprising Class 1 survey units may be smaller than either Class 2 or 3 survey units. The size of all survey units may have to be consistent with any dose modeling used to obtain the $DCGL_C$.

Whenever less than 100 percent of the survey unit is scanned, there is the potential to reintroduce uncertainty attributable to spatial variability, because the entire population of measurement locations is not being sampled and the scanning coverage is not random. These factors are expected to be of minimal consequence in Class 2 and Class 3 survey units because the level of contamination is expected to be fairly low and not as spotty as in Class 1 survey units. Nonetheless, with less than 100-percent scan coverage, these measurements should be considered a potentially biased sample, and the resulting average will be a somewhat biased estimate of the population average.

5.2.3.2 Scanning and Direct Measurements (and Media Samples)

This clearance survey approach is possible when the survey instrumentation exhibits sufficient scan sensitivity (i.e., the scan MDC is less than the $DCGL_C$), but the survey instrumentation does not have the capability to automatically document the survey results. In this situation, a number of direct measurements (or media samples) are performed, primarily to document the scan results. The number of these measurements should be determined using the DQO Process, and may be determined using the statistically based sampling design discussed in Section 5.2.3.3.

Again, the scan coverage should be graded based on the material's classification. That is, 100 percent of surfaces should be scanned for Class 1 materials, 50 to 100 percent for Class 2 and 10 to 50 percent for Class 3. The size of the material survey unit may also be a function of the material's classification. That is, the amount of material comprising Class 1 survey units may be smaller than either Class 2 or 3 survey units. Again, the size of all survey units may have to be consistent with any dose modeling used to obtain the $DCGL_C$.

5.2.3.3 Statistically Based Sampling

This clearance survey approach is necessary when the survey instrumentation does not exhibit a sufficient scan sensitivity (i.e., the scan MDC is greater than the $DCGL_C$). In this instance, scanning is not capable of demonstrating compliance with the release criteria. Therefore, it is necessary to design the conventional clearance survey based on a statistical sample size. Scans are still performed to identify contamination that may exceed the scan MDC, recognizing that areas of contamination falling between the $DCGL_C$ and the scan MDC in concentration may not always be detected. The scan coverage should be graded on the basis of the material's classification. That is, 100 percent of surfaces should be scanned for Class 1 materials, 50 to 100 percent for Class 2, and 10 to 50 percent for Class 3. The size of the material survey unit may also be a function of the material's classification. That is, the amount of material comprising Class 1 survey units should be smaller than either Class 2 or 3 survey units. The size of all survey units should be consistent with any dose modeling used to obtain the $DCGL_C$.

In most cases, the statistical tests used in the MARSSIM are recommended, and for the same reasons. The criteria for choosing between the Sign test and the Wilcoxon Rank Sum (WRS) test are also the same. In general, when the radionuclide is not in background (or its background concentration is negligible) and radionuclide-specific measurements are made, the Sign test is used; otherwise, the WRS test is used. These nonparametric statistical tests, described below, can be used for both surface activity assessments and volumetric concentrations in materials. As discussed in Section 3.6, there are two possible scenarios under which these tests may be conducted. In Scenario A, the survey data are tested against a specified activity, known as the $DCGL_C$, to determine whether the concentration in the material survey unit exceeds that value. In Scenario B, the criterion is that no contamination is allowed in materials that are to be released from radiological controls.

One-Sample Statistical Test (Sign Test)

The Sign test is designed to detect whether there is contamination in the material survey unit in excess of the $DCGL_C$. This test does not assume that the data follow any particular distribution, such as normal or log-normal. If any measurement exceeds this $DCGL_C$, additional investigation is recommended, at least locally, to determine the actual areal extent of the elevated concentration.

The following formal null and alternative hypotheses are tested by the Sign test under Scenario A:

Null Hypothesis

H_0: The median concentration of contamination in the material survey unit is greater than the $DCGL_C$

versus

Alternative Hypothesis

H_a: The median concentration of contamination in the material survey unit is less than the $DCGL_C$

The null hypothesis is assumed to be true unless the statistical test indicates that it should be rejected in favor of the alternative hypothesis. The null hypothesis states that the probability of a measurement less than the $DCGL_C$ is less than one-half (i.e., the 50^{th} percentile, or median, is greater than the $DCGL_C$). Note that some individual survey unit measurements may exceed the $DCGL_C$ even when the survey unit as a whole meets the release criteria. In fact, a survey unit average that is close to the $DCGL_C$ might have almost half of its individual measurements greater than the $DCGL_C$. Such a material survey unit may still not exceed the release criteria.

The assumption is that the survey unit measurements are independent random samples from a symmetric distribution. If the distribution of measurements is symmetric, the median and the mean are the same. To the extent that the mean may be larger than the median, there should be some areas of larger concentration that cause the distribution to be skew. When that is the case, they will be identified by scanning, and will trigger appropriate investigation levels as described in Section 6. This is the reason for combining direct measurements with scans in the survey design.

The hypothesis specifies a release criterion in terms of a $DCGL_C$. The test should have sufficient power ($1-\beta$, as specified in the DQO Process) to detect residual radioactivity concentrations at the lower bound of the gray region (LBGR). The LBGR should be set at the expected mean contamination level for the material survey unit. If σ is the standard deviation of the measurements in the material survey unit, then Δ/σ expresses the size of the shift (i.e., $\Delta = DCGL_C$ - LBGR) as the number of standard deviations that would be considered "large" for the distribution of measurements in the survey unit. Table 5.5 in the MARSSIM provides sample sizes for the Sign test as a function of relative shift and Type I and II decision errors.

If the criterion specified for controlling the release of material is that there must be no contamination, the clearance survey requires a different approach, similar to Scenario B described in NUREG-1505. The following formal null and alternative hypotheses are tested by the Sign test under Scenario B:

Null Hypothesis
H_0: The median concentration of contamination in the material survey unit is zero.

versus

Alternative Hypothesis
H_a: The median concentration of contamination in the material survey unit is greater than the upper bound of the gray region (UBGR).

As in Scenario A, in order to design a survey to test the null hypothesis for Scenario B, it is necessary to specify a gray region. Since no contamination is the criterion, the LBGR is zero, but it is still necessary to specify the UBGR. This is essential for determining an appropriate sample size, and for specifying the needed measurement sensitivity (i.e.,MDC, as discussed in Section 9.1). The width of the gray region, Δ = UBGR - LBGR = UBGR - 0 = UBGR. If σ is the standard deviation of repeated "blank" measurements (i.e., measurements on material that is known to contain no contamination), Δ/σ expresses the width of the gray region as a relative shift. Table 5.5 in the MARSSIM shows that when this relative shift falls below 1, the sample size required for the test increases dramatically. For example, if $\Delta/\sigma = 1$, and the DQOs for the Type I and Type II error rates, $\alpha = \beta = 0.05$, 29 measurements are required. If $\Delta/\sigma = 0.5$, 89 measurements are required. If Δ/σ falls as low as 0.1, more than 2,000 measurements are required. Thus, it is generally recommended that the relative shift Δ/σ be between 1 and 3. Increasing the relative shift much above 3 does not appreciably reduce the required number of measurements.

There is a direct connection between the UBGR and the MDC. For every instrument and procedure, there is an associated MDC, which is usually defined to be the concentration that will be detected with a 95-percent probability when it is present, while limiting to 5 percent the probability that a detection decision will be made when there is actually no contamination. (Refer to Section 4.6.) This decision is made separately for each measurement. It is a test of the hypothesis that there is no contamination at that single location on the material. The detection decision is based on whether the instrument signal is above a critical level corresponding to a concentration equal to about one-half the MDC. The MDC is usually 3 to 4 times the measurement uncertainty, σ. Since the MDC should not exceed the UBGR, the smallest practical value of the UBGR occurs when it equals the MDC. *Thus, an essential part of the DQO process for this case is setting the required MDC.* This ultimately defines the gray region, the sample size, and the effort that should be expended to find any contamination that might be present. When the UBGR = MDC, Δ/σ is about 3. Table 5.5 in the MARSSIM then indicates that between 8 and 20 samples must be taken, depending on the Type I and Type II error rates that are set.

In practice, the very use of the Sign test implies that radionuclide-specific measurements are being made to detect radionuclides that do not appear in background. Thus, any *unambiguously* detected positive concentration measured anywhere on the material obviously shows that it does not meet the criterion of no contamination, even though the *median* added concentration may be zero. This is analogous to the procedure used in the MARSSIM, namely, if the average concentration exceeds the release criterion, the survey unit may not be released regardless of the result of the statistical test.

Two-Sample Statistical Test (WRS Test)

Measurements from the reference material and material survey unit are compared using the Wilcoxon Rank Sum (WRS) test (also called the Mann-Whitney test). The WRS test should be conducted for each material survey unit. If any measurement in the material survey unit exceeds the average of the reference material by more than $DCGL_C$ additional investigation is recommended, at least locally, regardless of the outcome of the WRS test.

The WRS test is most effective when contamination is uniformly present throughout a survey unit. The test is designed to detect whether this activity exceeds the $DCGL_C$. The advantage of the nonparametric WRS test is that it does not assume that the data are normally or log-normally distributed. The WRS test also allows for "less than" measurements to be present in the reference material and the survey units. As a general rule, the WRS test can be used with up to 40 percent "less than" measurements in either the reference material or the survey unit. However, the use of "less than" values in data reporting is not recommended. When possible, report the actual result of a measurement together with its uncertainty.

The following formal null and alternative hypotheses are tested by the WRS test under Scenario A:

Null Hypothesis

H_0: The median concentration in the material survey unit exceeds that in the reference material by more than the $DCGL_C$

versus

Alternative Hypothesis

H_a: The median concentration in the material survey unit exceeds that in the reference material by less than the $DCGL_C$

The null hypothesis is assumed to be true unless the statistical test indicates that it should be rejected in favor of the alternative. One assumes that any difference between the distributions of the reference material and material survey unit concentrations is attributable to a shift in the survey unit concentrations to higher values (i.e., because of the presence of contamination in addition to background).

If the distribution of measurements is symmetric, the median and the mean are the same. To the extent that the mean may be larger than the median, there should be some areas of larger concentration that cause the distribution to be skew. When that is the case, they will be identified by scanning, and will trigger appropriate investigation levels as described in Section 6. This is the reason for combining direct measurements with scans in the survey design.

The assumptions underlying the WRS test are that (1) the samples from the reference material are independent random samples from the same reference concentration distribution, (2) samples from the material survey unit are independent random samples from the same material survey unit concentration distribution, and (3) each measurement is independent of every other measurement, regardless of which set of samples it came from.

Note that some or all of the material survey unit measurements may be larger than some reference material measurements, while still meeting the release criterion. Indeed, some survey unit measurements may exceed some reference material measurements by more than the $DCGL_C$. The result of the hypothesis test determines whether or not the material survey unit as a whole is deemed to meet the release criterion. Individual measurements exceeding the $DCGL_C$ are further investigated to the extent necessary to ensure that the overall average in the survey unit does not exceed the $DCGL_C$. Additionally, the test should consider whether any smaller areas with elevated levels of contamination may exceed a separate criterion set for such areas.

The test should have sufficient power ($1-\beta$, as specified in the DQO Process) to detect residual radioactivity concentrations at the lower bound of the gray region (LBGR). The LBGR should be set at the expected mean residual contamination level in the material survey unit. The larger of the two values of σ estimated from the reference material and material survey unit should be used for the WRS test sample determination. As described in the MARSSIM, the relative shift, Δ/σ, where $\Delta = DCGL_C - LBGR$, is calculated. Table 5.3 in the MARSSIM provides sample sizes for the WRS test as a function of relative shift and Type I and II decision errors.

If the criterion specified for controlling the release of material is that there must be no contamination, the clearance survey requires an approach similar to Scenario B described in. The following formal null and alternative hypotheses are tested by the WRS test under Scenario B:

Null Hypothesis

H_0: The median concentration in the material survey unit does not exceed that in the reference material (i.e., there is no contamination).

versus

Alternative Hypothesis

H_a: The median concentration in the material survey unit exceeds that in the reference material by more than the upper bound of the gray region (UBGR).

For this test, the lower bound of the gray region is set at zero contamination. As for the Sign test using Scenario B, it is again necessary to specify a UBGR. It is essential for determining an appropriate sample size and the needed measurement sensitivity. The width of the gray region, $\Delta = UBGR - LBGR = UBGR - 0 = UBGR$. If σ is the standard deviation of repeated "background" measurements (i.e., measurements on material known to contain no contamination), Δ/σ expresses the width of the gray region as a relative shift. Table 5.3 in the MARSSIM shows that when this relative shift falls below 1, the sample size required for the test increases dramatically. For example, if $\Delta/\sigma = 1$, and the DQOs for the Type I and Type II error rates, $\alpha = \beta = 0.05$, 32 measurements are required on both the survey material and on the background reference material. If $\Delta/\sigma = 0.5$, 114 measurements are required on each. If Δ/σ falls as low as 0.1, more than 2,700 measurements are required on each. Thus, it is generally recommended that the relative shift Δ/σ be between 1 and 3. Increasing the relative shift much above 3 does not appreciably reduce the required number of samples.

There is a direct connection between the UBGR and the required measurement sensitivity. To distinguish between a measurement of background on the reference material and a measurement equal to background plus the UBGR, the instrument or procedure must be able to reliably detect the difference (i.e., the UBGR). Unless the uncertainty of a typical background measurement, σ_M, is less than the UBGR, the relative shift $\Delta/\sigma = UBGR/\sigma$ will fall below 1, even if there is no spatial variability contributing to σ. Conversely, setting the UBGR to be less than σ_M will cause the number of measurements required to achieve the DQOs to rise dramatically. *Thus, an essential part of the DQO Process for this case is in setting the UBGR, recognizing the implicit demand on the required relative measurement uncertainty at near-background levels.*

Application to Surface Activity Measurements

Either the Sign test or WRS test can also be used for surface activity measurements. Given that many material survey units are composed of the same material types, using the WRS test should be relatively straightforward (i.e., same as described in the MARSSIM). In some cases however, the number of materials present in a batch may make it impractical to use the WRS test. In such cases, it is possible to perform the Sign test on the difference of paired measurements on similar materials, one from the survey unit and one from a reference material, as outlined in Section 12 of NUREG-1505 (NRC, 1998b).

When surface activity measurements are performed using non-radionuclide-specific (gross) survey instruments (e.g., GM and gas proportional detectors), a commonly used procedure is to subtract an "appropriate average background" from each gross measurement on the solid material, and then analyze the resulting data using a one-sample statistical test, such as the Sign test. Before doing so, however, the surveyor should recognize that the WRS test may be more advantageous for the following reasons:

(1) The number of samples taken to compute an appropriate background average is left purely to judgment. When the WRS test is used, the appropriate number of background measurements has a statistical basis.

(2) The Sign test will generally not be as powerful as the WRS test (more important as the expected contamination level approaches the $DCGL_C$).

(3) The same data that are used to calculate the average background can always be used in the WRS test as well.

The Sign test offers no real savings (compared to the WRS test), with the possible exception of the time needed to perform the calculations. However, when the material survey unit is very clean, the maximum survey unit measurement and minimum reference area measurement will likely not exceed the DCGL, and the survey unit will pass the WRS test without any need for calculations. When the material is contaminated above the DCGL, a simple comparison of the averages will likely show that the material cannot be released. It is only in cases where the contamination is near the DCGL that the extra computations involved in the WRS test will be necessary; however, it is precisely in those cases that the higher statistical power of the WRS test makes its use more desirable.

Statistical Sample Locations

While many sampling and analysis procedures for solid materials clearance surveys are the same as those recommended in the MARSSIM, the major exception is the selection of sampling points on a survey unit consisting of a few large, irregularly-shaped pieces. It is virtually impossible in most cases to identify random locations on material with odd shapes, simply because such materials are virtually impossible to grid. Materials consisting of many small regularly shaped pieces can be spread out evenly, as discussed in Section 4.2. A random start grid (rectangular or triangular) can be used to locate samples. It is important to emphasize that the objective in this case is to give every portion of the batch the same opportunity to be sampled. Thus, it is only necessary to locate and lay out the grid sufficiently to ensure that sampling locations are chosen objectively.

One way to approximate this procedure for a survey unit consisting of a few large, irregularly shaped pieces is to lay out a grid in the area where measurements are to be made. The batch of material should be laid out in a single layer on top of this grid. A randomly selected grid node is sampled by measuring whatever piece (or portion) is nearest that node. If no piece is near, select another point until the required number are obtained. If there is a well-defined inside and outside (as for a pipe), an additional random number can be used to determine whether the inside or outside is sampled. Even this procedure may not be workable for large pieces of equipment that cannot be placed on a grid so that every point has an equal choice of being sampled. In such cases, there may be no alternative other than to choose biased sampling locations, giving preference to samples that are more likely to contain radioactivity. This involves professional judgment, and often results in overestimating the average concentration. This is not a guarantee, of course, because such judgments are not perfect. It is important to document the criteria used for selecting sampling locations in a standard operating procedure (SOP), and to document that these criteria were followed. These criteria, and the associated logic, should be specified before the actual sampling.

Another possible method for sampling a lot of similarly sized small pieces of material is to systematically measure every m^{th} piece. This requires some estimate of the total number of pieces, N, so that N/m equals or exceeds the number, n, required for the statistical tests.

5.3 Automated Scanning Surveys (conveyorized survey monitors)

Systems that automate the collection of measurements can offer an appealing alternative to manual surveys. By design, automated systems require little in the way of human intervention during operation and analyze the data on-the-fly, while storing the information in digital form. These features can provide several advantages when compared to manual surveys by personnel using hand-held equipment; however, such automation typically requires equipment that is both expensive and bulky.

Conveyorized survey monitors (CSMs) offer a form of automation that may be particularly well-suited for use where significant quantities of bulk material are subject to clearance requirements. As the name implies, these systems operate by moving materials past radiation detectors using a conveyor system, while automatically storing and analyzing the resulting signals. The radiation detectors themselves can be of any type and are chosen to match the application. The most common detectors in use are NaI crystals for gamma-detection and thin-window proportional counters for beta-detection.

Sections 5.3.1 – 5.3.3 discuss CSM systems and their possible application as a measurement method when releasing solid materials during clearance surveys. Like all measurement methods, CSMs are viewed as tools that may be used alone or in tandem with other methods. Although specific manufacturers' systems are not discussed, Appendix B, "Advanced/Specialized Instrumentation," includes a sampling of platforms that are presently being marketed for this application, as well as supporting information about various types of detectors and materials.

5.3.1 Equipment

Conveyorized survey monitors typically include a motorized conveyor, a detector array, supporting measurement electronics, and an automated data acquisition subsystem. Monitors may also include segmented pathways along the conveyor so that suspect material may be transported to a destination other than that of the non-suspect (or releasable) material.

The conveyor portion of a system consists of a belt that is moved by a variable-speed motor from a loading area, past a detector assembly or set of assemblies, and onto the final destination, which may be either a disposal container or an intermediate pile. If a mechanical diverter is used, the system controls the final material destination based upon user-configured measurement parameters. Without automated segmentation of the material, a system would need to be used in a "shutdown" mode to allow manual separation of suspect material.

Since the conveyor operates in a continuous loop, it creates the possibility for cross-contamination on the belt. When processing materials with a low probability of contamination, as is usually the case during clearance surveys, this issue is of little concern. For applications where cross-contamination poses a real issue, however, it would seem reasonable to use a continuously replaced rolled sheeting material as a protective barrier.

Automated Data Processing (ADP) — Measurements collected using a CSM are usually digitized before being analyzed and digitized. The data are analyzed on-the-fly using a preset algorithm, and decisions concerning suspect materials are usually made in real-time. The resulting data, together with the analysis results, are then archived to a long-term digital storage medium.

The counting parameters associated with measuring a stream of material passing near a CSM detector are very similar to those encountered with other detection systems. Although each manufacturer's system employs a proprietary analysis mechanism, the fundamental physics and statistical parameters are independent of the software design. As such, one can estimate the detection sensitivity of a CSM detector system without detailed knowledge of the analysis methods that are actually used, provided that the type of detector and electronic configuration are known.

A very interesting capability that is unique to automated systems is the ability to perform multiple, parallel analyses. As a practical example, a CSM could be configured to monitor over multiple time intervals, in order to optimize the detection capability for both small and large regions at the same time. Additionally, the data collected from shorter time intervals could be used to augment the decision criterion applied to longer time intervals, so that small increases over the long interval may be corrected for anomalies (e.g., such as from potential hot spots) observed during short-interval measurements.

Detectors — The heart of any radiation measurement system is the detector(s). The selection and configuration of detectors and associated electronics is the single most important aspect of designing any radiation measurement device, since it defines the system's baseline capability. Auxiliary components, such as data analysis engines and hardware controls, certainly affect the overall performance of a CSM, but not to the same degree as the detector(s). The ability of any detector to measure radiation is defined by physical constraints that cannot be easily manipulated or changed by users, so the initial selection of this component more-or-less establishes the system's capability.

Gross screening of gamma-emitting radionuclides is usually best performed using scintillation detectors, such as NaI or plastic scintillators. While these detectors are not the best selection for quantitative measurement of complex spectra, their excellent detection efficiencies and relatively low cost make them top candidates for gross gamma measurement applications where CSMs may be desired. Solid-state gamma-ray detectors, such as high-purity germanium (HPGe) detectors, offer much better assay capability, but are fairly expensive to purchase and maintain, especially if one is interested in achieving the same level of detection efficiency offered by large-volume scintillation crystals.

The type, shape, encapsulation, and electronic configuration of a scintillation detector determine its overall detection efficiency and background response, thereby defining its signal-to-noise ratio. Consequently, it is important to select detectors that balance background response with detection efficiency for the suspected radionuclide(s). As an example, a 3" x 3" NaI detector yields a good signal-to-background ratio for a high-energy gamma-emitter such as ^{60}Co, but is a poor selection for a low-energy emitter such as ^{241}Am. Beyond the base selection of the detector material and physical design, one should consider the selection and placement of photodetectors and driving electronics when considering the optimization of a system. For example, simply reducing (or increasing) the detection input threshold at the amplifier stage can sometimes critically alter the overall system performance.

High-purity germanium detectors could play an important role in some CSM systems, even though they are more expensive and difficult to maintain. These detectors are excellent for gamma-ray spectrometry, as they facilitate an unparalleled capability for nondestructive identification and quantification of gamma-emitting radionuclides. With the exception of very expensive large-volume crystals, however, these detectors cannot compete with low-cost scintillation materials when gross sensitivity is desired. Their use in a CSM system could be warranted in some instances for nuclide identification following a positive detection during a gross scan. For example, a system could plausibly be configured to automatically stop a conveyor following a positive *detect,* and then attempt to identify the gamma-emitting radionuclides present before passing the material to its final destination.

Measurement of beta-emitting radionuclides in (or on) bulk materials may also be possible, depending on the radionuclide, material type, and release limit. Beta detection can be accomplished using thin-window gas-filled detectors, such gas proportional and Geiger-Mueller detectors, and thin-windowed scintillators. The most likely candidate for measuring beta-emitters is large-area gas flow through proportional detectors with thin Mylar entrance windows; however, large-area sealed proportional and GM detectors are also expected to perform well. Scintillation materials universally suffer from an inferior signal-to-background ratio when measuring beta-emitters, but may still be adequate for some applications.

The surface area and window thickness of beta detectors are the critical design parameters that affect detection efficiency. Ideally, one would desire a large array of small detectors, so that each segment monitors a small area while keeping its background to a low level. This would be an expensive option, so actual systems usually employ intermediate-sized detectors with thin windows, with each detector often occupying 100 cm^2 to 500 cm^2 of sensitive area. Smaller detectors are also often grouped together in parallel assemblies with common electronics to minimize the overall system cost. These detector sizes provide a good balance between cost and detection sensitivity for CSM applications.

As another, somewhat uncommon option for CSM systems, electronically segmented proportional counters overcome the size-versus-background design issue. Detector systems operating in this mode attempt to subdivide large-area proportional detectors into small, virtual regions by using advanced timing electronics to optimize the signal-to-background ratio for small areas, while keeping the number of detectors low. These designs require more advanced electronics and analysis algorithms, and are not typically used in CSM systems today.

5.3.2 Detection Sensitivity

The selection of detectors and supporting electronics is the key to optimizing overall system performance for specific applications. Other parameters that should be considered include the quantity and placement of detectors, as well as the speed of materials past the sensitive regions of the detector(s).

As a rule, the signal-to-background ratio of a radiation detector array is directly proportional to the square root of the number of detectors employed when measuring uniform radiation fields. To illustrate this principal, two identical detectors operated in tandem (parallel) yield a signal-to-background ratio that is about 40 percent higher than the ratio that a single detector would yield when measuring a material with homogeneously distributed contamination. Grouping the detectors together in parallel, with a single set of driving electronics, reduces the detection ability for small regions near a given detector. By contrast, if the two detectors are operated independently of each other, with separate driving electronics, the measurement sensitivity for homogenous media would also be 40 percent higher than the capability of a single detector, but without penalizing the ability to detect small, elevated regions.

Placement is also critical — particularly for the measurement of beta emitters — since the inverse square relationship and absorption within the intermediate air can greatly affect sensitivity. While this is less important for gamma-detection equipment, it is essential to place beta-measurement detectors as close as practical to the material being monitored. As with portable survey equipment, it is also advisable to establish a CSM detector configuration that offers an acceptable detection ability without placing the detector into harms way (as might occur when jagged materials pass too near a fragile detector face).

Belt speed significantly affects the measurement capability of a CSM. Detection sensitivity for small- to intermediate-sized regions varies (roughly) with the square root of the observation interval (time) for any segment of material being monitored. In other words, a slower-moving belt facilitates a more sensitive detection capability for smaller regions. Interestingly, belt speed has no impact on detection ability for a continuous stream of truly homogeneous materials since, by definition, the radioactivity is present at an equal concentration throughout all of the material. In practice, however, material with homogeneously distributed contamination is atypical, and the detection ability for smaller regions should be considered when designing a scan protocol.

To deal with this fact while using a CSM during clearance surveys, one can assume, for better or worse, that homogeneity exists within sub-regions of the suspect material and, to be consistent with traditional survey design, these regions should be labeled as survey units or batches. The desired belt speed should, therefore, be determined as a function of the release limit (DCGL), the allocated survey unit size, and the detection efficiency of the system for the target media and expected radionuclide(s).

Detection Efficiency for Gamma-Emitters using NaI Detectors — The detection ability of NaI detectors operating in a gross count rate mode[3] will be dependent on the design, quantity and electronic configuration of selected detectors. For purposes of providing an example of an expected detection capability, this section discusses a hypothetical system that has been configured with moderately sized 3" x 3" cylindrical crystals with supporting electronics. It is assumed that three such detectors will be operated in tandem in a detector *bank* and that the total detector volume per bank will therefore be about 1000 cm^3.

A common radionuclide that may be measured using such a system would be ^{137}Cs—with a primary gamma-ray emitted by its daughter (^{137m}Ba) at ~662 keV with an emission ratio of ~0.85. If one assumes that cesium is mixed relatively homogeneously within each region of a CSM conveyor stream, then a fairly accurate estimate of detection ability can be calculated by coupling empirical data with modeled exposure rates. The two empirical parameters that should be known are the total background count rate and the detection efficiency for ^{137}Cs. In general, although certainly depending on location and configuration, the background count rate for 3" x 3" cylindrical NaI crystals operating in full-open gross count rate mode will be in the range of about 8×10^3 to 1×10^4 counts per minute (cpm) and the detection efficiency will be approximately 4×10^6 cpm per mR/h when measuring ^{137}Cs. For three detectors ganged into a single electronic bank, these values correlate to a total system background of about 2.7×10^4 cpm and a total detection efficiency of about 1.2×10^7 cpm per mR/h.

These parameters can be coupled to calculated exposure rates in the vicinity of material passing along a conveyor system to evaluate detection sensitivity as a function of the material geometry and radionuclide. As an example application, consider a scenario where a CSM will be used to scan for ^{137}Cs in soil having a bulk density of 2 g/cm^3. The center-line of the three detectors is assumed to be placed approximately 15 cm above a 76-cm (30-in) wide conveyor belt such that they are evenly spaced across the breadth of the belt at 13, 38 and 64 centimeters (5, 15 and 25 inches) from one edge. If the soil is assumed to be 2.5-cm (1-inch) thick and to extend on the conveyor for 76-cm (30-inches) along the conveyor to either side of the detector bank then the expected exposure rate will be about 120 mR/h per μCi/g at the two outside detectors and approximately 140 mR/h per μCi/g for the center detector. Coupling these data with the expected detection efficiency previously given, the total efficiency for this geometry—using all three detectors in an electronically ganged configuration—is expected to be about 1.5×10^3 cpm per pCi/g of ^{137}Cs. If the soil thickness is increased to 10-cm (4-in) and the detectors are positioned 20-cm (8-in) from the belt, then the system detection efficiency will increase to about 4×10^3 cpm per pCi/g of ^{137}Cs. The latter case represents a count rate increase of 15% above background for each pCi/g of ^{137}Cs.

[3] *Gross count rate mode* refers to operating a detector such that all measured pulses within a pulse-height window, whether it be narrow or wide open, are summed together into a single value representing the gross count rate for the detector configuration being used.

An estimate of the minimum detectable concentration (MDC) can be estimated while operating such a detector configuration in a scan mode by assuming a false positive detection rate of 1% and a false negative detection rate of 5% (Currie 1968). These values mean that true contamination will be missed 5 percent of the time, and false alarms will occur 1 percent of the time. For an observation interval of 6 seconds, the MDC for a 2.5-cm (one-inch) thick layer of soil containing ^{137}Cs is expected to be about 2 pCi/g and will decrease to 0.7 pCi/g when the soil thickness is increased to 10 cm.

Detection Efficiency for Beta-Emitters Using Thin-Window Proportional Detectors — Beta particles originating within or on a target media usually undergo significant interaction before reaching the sensitive volume of a CSM detector. As such, the process for estimating detection ability is significantly more problematic than is necessary when evaluating detection capability for gamma-emitting radionuclides. As previously mentioned, the most common type of detector for this application is a thin-window gas-flow proportional detector. Such detectors have a thin Mylar entrance window with a density thickness ranging from less than 1 to a few mg/cm^2. Although the mixture may vary, the most commonly used gas is P-10, containing 90 percent argon and 10 percent methane.

This section provides an analysis of the beta detection ability for gas-flow proportional counters and, in particular, that which is applicable to a CSM. The first scenario considers surface contamination with ^{99}Tc and ^{90}Sr on flat surfaces, while the second looks at ^{99}Tc and ^{90}Sr in soil, and the third evaluates ^{137}Cs in soil. These evaluations are summarized in the following paragraphs:

Surface activity refers to contamination on the surface of solid materials. As simple as this sounds, it is difficult to define what constitutes a "surface," since real-world materials have a thickness when viewed from the perspective of a radioactive atom deposited *within* their surfaces. One might define surface contamination as the activity contained within a surface layer that has a thickness equal to that of the saturation layer (ISO 1988), where the thickness of the saturation layer is defined as the thickness of the medium (surface material) equal to the maximum range of the specified particulate radiation. While some materials are more porous than others, all have some level of absorptive capacity. The definition of "surface," therefore, becomes significant when evaluating the detection ability for charged particles emitted from the surface of materials, and is amplified significantly when constructing a model.

Consider an 80-cm (31-inch) wide conveyor using five proportional counters with open, or sensitive, areas of 500-cm^2 each, placed 5 cm above the belt surface. The detectors are rectangular in shape, with each window region measuring 50 cm x 10 cm (20 in by 4 in), with the long dimension placed parallel to the direction of belt travel in the CSM. If five such detectors are placed side-by-side across the breadth of the conveyor, the total sensitive area is 2,500 cm^2 (390 in^2). Each detector is assumed to be configured individually (not grouped), with 0.8 mg/cm^2 of window material without protective screens, and the detection capability is assumed to have been maximized for low- to intermediate-energy beta detection. The background response for such a detector is in the range of 2 to 3 cpm/cm^2 of window area, so each detector has a non-shielded typical background of about 1,300 cpm. Again, the reader should note that this configuration is defined for the purpose of estimating beta detection ability as an example; however, the detection abilities of actual systems will vary by manufacturer (although not very much).

First, the pure beta-emitting radionuclides ^{99}Tc and ^{90}Sr(^{90}Y), having maximum-energy beta emissions of 294 and 546(2280) keV, respectively, are assumed to be placed onto the surface of a thin, flat plane in contact with a CSM conveyor belt. Although unrealistic for most real-world measurement scenarios, this finite plane, zero-thickness geometry provides the highest possible beta-detection sensitivity for a system without improving the detector to belt distance. As an extension to this *pure* geometry, it is then assumed that the radionuclides are not restricted to the outermost surface, but instead that they have absorbed homogeneously within the top 50 µm of a masonry-type material (e.g., cement) having a bulk density of 2 g/cm^3. This scenario is much more plausible when evaluating real-world applications. Table 5.1 presents the results of these geometry calculations.

The second geometry places the same isotopes (i.e., ^{99}Tc and ^{90}Sr(^{90}Y)) into a soil matrix and varies the depth of the material from 0.1 to 1 cm, while keeping the belt to detector distance constant. The results of this analysis display, both qualitatively and quantitatively, the impact on detection capability that occurs when beta particles interact within the source-matrix material. Table 5.1 presents the results.

Finally, the isotope ^{137}Cs, which is both a beta- and a gamma-emitter, is modeled within a soil matrix. Cesium-137 decays with the emission of a 512-keV$_{max}$ beta 94.6 percent of the time, and decays with the emission of a 1,173-keV$_{max}$ beta for the remainder. As previously mentioned, ^{137m}Ba is produced by 94.6 percent of ^{137}Cs decays, and it, in turn, emits a 662-keV photon during 90 percent of its decays, yielding an overall γ-emission ratio of 0.85. Although not previously discussed within this section, gas-flow proportional counters also detect ionizing electromagnetic radiations (e.g., gamma and x-rays) by measuring secondary electrons produced both within and outside the gas volume. The probability of interaction varies; however, the sensitivity is roughly proportional to the mass of intervening material within the vicinity of the detector, times the probability of interaction within the mass, times the fraction of those particles carrying enough energy to travel into the detector. For ^{137}Cs, the intrinsic efficiency expected with a thin-window proportional detector is about 0.01 counts per photon. The photon detection capability for this scenario was estimated for each CSM detector by calculating the average solid-angle for the geometry and coupling the result with the activity, source-material absorption probability and finally the detector interaction probability. Table 5.1 presents the result for the summed beta and gamma detection capability.

Table 5.1: Model results for the detection capability of a CSM configured with a bank of 500-cm² gas proportional detectors[a]

Isotope	Material[b]	Single 500-cm² Detector[c]		Five Detectors Grouped as One 2,500-cm² Detector	
		Efficiency in cpm per [dpm/cm² or pCi/g*]	$MDC_{6\text{-sec},95\%}$[d] [dpm/cm² or pCi/g*]	Efficiency in cpm per [dpm/cm² or pCi/g*]	$MDC_{6\text{-sec},95\%}$ [dpm/cm² or pCi/g*]
^{99}Tc	Surface [0-µm]	60	10	300	5
	Surface [50-µm]	30	20	150	10
	Soil [0.5 cm thick]	1*	650*	5*	300*
	Soil [1.0 cm thick]	1*	650*	5*	300*
^{90}Sr	Surface [0-µm]	130	5	650	2
	Surface [50-µm]	95	7	480	3
	Soil [0.5 cm thick]	6*	110*	30*	50*
	Soil [1.0 cm thick]	6*	110*	30*	50*
^{90}Y	Surface [0-µm]	250	3	1300	1
	Surface [50-µm]	230	3	1200	1
	Soil [0.5 cm thick]	60*	10*	300*	5*
	Soil [1.0 cm thick]	60*	10*	300*	5*
^{137}Cs [e]	Soil [0.5 cm thick]	10*	65*	50*	30*
	Soil [0.8 cm thick]	12*	55*	60*	25*
	Soil [1.0 cm thick]	14*	45*	70*	20*

[a] Section 5.3 describes each geometry.

[b] A 0-µm surface is defined as a zero-thickness source, where all isotope material is present exactly at the surface. Such surfaces are similar to an electroplated laboratory standard, but would not be expected during typical CSM operation. A 50-µm surface assumes that the source material is homogeneously distributed within the top 50-µm layer of a low atomic number material (e.g., masonry) with a density of 2 g/cm³, and the material is present as a continuous plane beneath the detector. Soil describes a homogenous mixture with a bulk density of 2 g/cm³.

[c] All detection efficiencies are reported in cpm /dpm /cm² of source area for surface scenarios and cpm /pCi/g) for soil. Single-detector values represent the average response expected for five detectors spread across the breadth of a 80-cm wide CSM. All values have been rounded to no more than two significant digits.

[d] Minimum detectable concentration (MDC) calculated including the variability of background for each 500-cm² detector equal to 130 counts during 6-second count intervals (1,300 cpm), based on a given belt speed. The probability of false-detection is assumed to be set at 1 percent and the probability of missing existing (true) contamination is assumed to set at 5 percent. Results have been rounded to no more than two significant digits and are given in units of dpm/cm² for surfaces and pCi/g for soil.

[e] Detection ability calculated for beta-emissions from ^{137}Cs as well as gamma-emissions from ^{137m}Ba. The observed increase in detection efficiency with soil thickness is due to the increased number of 662-keV gamma rays produced with increased soil mass.

5.3.3 CSM Survey Design Considerations

Conveyorized survey monitors are expected to be used in conjunction with other survey methods during the release of materials for unrestricted use. These relatively massive devices are primarily designed for scanning applications; however, it is possible to construct control algorithms that combine a number of complementary survey stages. Examples include the combination of different detector types, scan and static measurement modes, and the ability to make parallel decisions based on various combinations of measurement results. Ultimately, it is expected that CSM machines could be applied as an advanced, automated scanning process in lieu of using hand-held equipment as discussed elsewhere in Section 5.

As an example, consider an application for the detector assemblies discussed above, which include a set of three grouped 3" x 3" NaI crystals placed in series, with a set of five 500-cm^2 gas-flow proportional counters. Fine concrete rubble is to be surveyed and is expected to contain ^{137}Cs and $^{90}Sr(^{90}Y)$ at varying ratios, which means that a simple correlation cannot be assumed for ^{90}Sr based solely on gamma measurements for ^{137}Cs. Furthermore, the radioactivity is primarily expected to be present throughout moderate-sized volumes of the material, and the hypothetical release limits (DCGLs), based on draft NUREG-1640 dose factors, are assumed to be set at 0.16 Bq/g (4.4 pCi/g) for ^{137}Cs and 4.4 Bq/g (120 pCi/g) for ^{90}Sr. The daughter, ^{90}Y, is assumed to be present at the same concentration as ^{90}Sr.

A number of design decisions can be made for such a CSM system to help automate the clearance of material. A configuration decision would be to use the NaI detectors to look for ^{137}Cs and to use the gas-proportional detectors to monitor gross beta emissions from ^{90}Y and, to a much lesser degree, ^{90}Sr and ^{137}Cs. Referencing the preceding analyses, the detection MDC for ^{137}Cs for the proposed bank of NaI detectors will be 2 pCi/g for a 2.5-cm (1-inch) thick layer of soil, and will decrease to about 0.7 pCi/g when the soil thickness increases to 10 cm. These values are fairly accurate for our concrete rubble scenario. Similarly, the detection sensitivities (MDCs) for ^{90}Sr and ^{90}Y in soil were given as 50 pCi/g and 5 pCi/g, respectively, and represent reasonably accurate estimates for the granulated concrete scenario. To reiterate, all of these detection sensitivity values were calculated for 6-second observation intervals, while assuming 5 percent false-negative and 1 percent false-positive detection probabilities.

As is readily seen, the detection capabilities for the target radionuclides for a 2.5-cm (1-inch) thick layer of material are less than the hypothetical release limits. Therefore, it is plausible that the CSM could be used for the majority of the release scan process without complicated detection schemes. It is important to recognize that the premise of homogeneously distributed contamination over the volume of the solid material is the basis for assuming that the beta-emitting radionuclides are on or near the material's surface. Otherwise, there is only a slim likelihood of detecting a discrete amount of $^{90}Sr(^{90}Y)$ activity present a few millimeters beneath the soil surface.

5.4 *In Toto* Surveys

In contrast to sampling and direct measurements, which use discrete samples and measurements to assay contamination, an *in toto* approach assays the solid material as a whole. Examples of instruments that use an *in toto* assay approach are *in situ* gamma spectrometry systems, drum and box counters, tool and bag monitors, and portal monitors.

In toto survey techniques can be used to demonstrate compliance with the average contamination level over the entire material survey unit, and can be used as a technique for measuring individual samples. When used to measure contamination over the entire material survey unit, this clearance survey approach is well-suited for solid materials that do not have a potential for small elevated areas of radioactivity (i.e., solid materials classified as Class 2 or 3).

When small elevated areas of radioactivity are potentially present (e.g., Class 1 materials), their impact on the average contamination level should be properly addressed during the calibration and efficiency determination for *in toto* survey techniques. Alternatively, when potential small elevated areas of radioactivity are a concern, it may be appropriate to consider combining the in toto techniques with conventional scanning for locations of elevated direct radiation.

When employing *in toto* clearance survey techniques, it is important to consider both the classification of solid materials and the difference between the material survey unit size and sample size. Consider a pallet of 1.5-m long steel pipes that is assayed using a calibrated *in situ* gamma spectrometer system. This pallet represents a material survey unit, which would likely be surveyed via *in situ* gamma spectrometry in the same manner regardless of its classification.

Consider a large container filled with hundreds of small pieces of equipment and tools that are proposed for clearance. Assume that a tool monitor will be used to demonstrate compliance with the release criteria. In this instance, the amount of material (perhaps no more than 10 items at a time) that can be analyzed by the *in toto* technique represents the sample size, rather than the survey unit size. When *in toto* survey techniques are used to measure samples, the statistical design methods discussed in Section 5.2.3.3 should be used to determine the sample size.

The DQO Process should be used to establish the appropriate survey coverage. The material's classification should be considered when setting the size of the material survey unit. For example, the amount of material comprising Class 1 survey units may be smaller than either Class 2 or 3 survey units. Alternatively, it may be reasonable to maintain consistent survey unit sizes for all material classes, while adjusting the survey coverage based on classification. In this situation, the tool monitor might be used to assay 100 percent of the materials in Class 1, while smaller fractions of the total material would be analyzed in Class 2 and 3 survey units. For example, it may not be necessary to survey each and every brick that comprises a lot of Class 2 bricks. Regardless of the selected approach, the solid materials having the greatest potential for contamination should receive the highest degree of survey coverage.

Sections 5.4.1 – 5.4.3 discuss *in situ* gamma spectrometry, volume counters (e.g., drum counters, tool and bag monitors), and portal monitors. Calibration and implementation considerations for using these systems are also discussed.

5.4.1 *In Situ* Gamma Spectrometry

In situ gamma spectrometry (ISGS) measurements for solid materials, particularly in a complex geometry that renders some of the surfaces inaccessible, may be a viable release survey option. This section discusses some of the considerations and the overall plan for implementing ISGS as a tool for surveying solid materials, including experimental results for applying ISGS to surveys of scrap metal. Appendix C provides a few examples of commercial applications of ISGS.

5.4.1.1 Equipment

An ISGS system typically consists of a semiconductor detector, electronics for pulse amplification and pulse height analysis, a computer system for data collection and analysis, and a portable cryostat. The most common detector is the high-purity germanium (HPGe) semiconductor, but other semiconductors such as developing room temperature variants can be deployed. The HPGe crystal should be cooled to liquid-nitrogen (LN) temperature for operation, but can be stored at room temperature without destroying its detection properties.

This is an important distinction between HPGe semiconductor detectors and germanium-lithium (GeLi) semiconductor detectors, which must be cooled to LN temperature at all times. Scintillating detectors, such as sodium iodide (NaI), have limited application (e.g., when energy resolution is not a primary concern). Additionally, depending on the application, lead shielding and collimation may be required.

5.4.1.2 Technological Advances

Many technological advances have allowed ISGS to become more of a mainstream survey methodology. As previously mentioned, one of the most important advancements was the HPGe detector, which only required cooling to LN temperature during operation. Also, these detectors have increased in volume, resulting in much higher efficiency, while maintaining excellent energy resolution. These systems can only be used if the detector is maintained at LN temperature, but the advancements of rugged, multi-attitude LN cryostats have permitted ISGS systems to be deployed in almost any environment. The electronics have also been improved by reducing their size, which increased their portability. Typically, these electronics have been analog in design, which means that they suffered from instability under certain conditions. Digital electronics packages have overcome the limitations of the analog designs. The portable computing systems used to collect and analyze the ISGS data have also increased in power while also decreasing in size.

5.4.1.3 Sensitivity

Unlike hand-held detectors used to scan and/or perform direct measurements to qualify or quantify primarily alpha and beta surface activity, ISGS can be used to quantify volumetric contamination of gamma-ray-emitting radionuclides. Many factors determine the overall efficiency and sensitivity of an ISGS system for quantifying volumetric contamination, as follows:

Intrinsic detector efficiency

The intrinsic efficiency of a detector is the measure of how efficient the detector medium absorbs gamma-ray energy, as a function of energy. At very low energies, gamma-rays are absorbed outside the detector, in the casing or faceplate. As the energy increases, the intrinsic efficiency increases until a maximum intrinsic efficiency is reached, typically at an energy of a few hundred keV. After the maximum is reached, the intrinsic efficiency decreases with increasing energy.

Radionuclide gamma-ray energy and abundance

As discussed above, the intrinsic efficiency of a detector depends on the gamma-ray energy. Also, attenuation from the material being surveyed increases as the gamma-ray energy decreases. Solid materials with potential contamination involving radionuclides of low gamma-ray decay abundance, or yield, require longer count times than radionuclides with high gamma-ray decay abundance.

Background, including shielding and collimation

High background, for the gamma-ray energies of concern, decreases the sensitivity of the ISGS. This effect is more pronounced at lower energies because of the Compton continuum contributions from ambient gamma-rays, which are higher in energy than the energy of concern. To reduce the effect of background, lead shielding and collimation can be used. While generally increasing the sensitivity of the ISGS measurement, collimation can actually lower the overall efficiency of the ISGS system by effectively shielding the contamination from the detector. This is a concern when using small-opening collimators.

Count time

Many factors influence the amount of time required to count the material. These include the overall efficiency, source and background count rates, and desired uncertainty. In general, as the background increases, the sensitivity decreases. To compensate, increasing count time increases sensitivity. In order to reduce the uncertainty of the measurement by half, the count time would need to be increased by a factor of four.

Geometry

Geometry refers to the orientation of the source material and the detector relative to the source material. For example, the overall efficiency and, therefore, the sensitivity of the ISGS measurement would be different if a lot of 25 pipes is stacked in a pyramid, rather than placed flat and unstacked. The overall efficiency of the ISGS measurement is also affected by the distance the detector is placed from the source material.

5.4.1.4 Experimentation to Determine Sensitivity

Oak Ridge Institute for Science and Education (ORISE) performed an experiment to determine the magnitude of the ISGS detection capabilities for a release of scrap metal from a nuclear facility. In this case, 1 metric ton of 12.7-cm (5-in.) diameter steel conduit was selected. To determine how much radioactivity was required for the experiment, the mass-based, critical-group dose factors reported in draft NUREG-1640 were used. For comparison with draft NUREG-1640, a normalized unit dose factor of 10 μSv/y (1 mrem/y) was assumed in the calculations. As the following equation shows, 38 kBq (1 μCi) of ^{137}Cs on steel would produce approximately 10 μSv/y (1 mrem/y) to the critical member of the group.

$$\frac{10\,\mu Sv\,y^{-1}}{260\,\mu Sv\,y^{-1}\,Bq^{-1}\,g}\cdot 1E6\,g\cdot\frac{kBq}{1000\,Bq}=38\,kBq$$

Therefore, if the ISGS system can demonstrate a sensitivity less than 38 kBq (1 μCi), this is a candidate technique. Table 5.2 summarizes the total activity calculations for steel.

Table 5.2: Calculated total activity for selected radionuclides using mass-based, critical-group dose factors for steel (1x10^6 g)

Radionuclide	Key Gamma(s) (keV)	Mean Dose Factor (μSv y^{-1} Bq^{-1} g)[a]	Total Activity for 10 μSv y^{-1} (kBq)[b]
^{137}Cs	662	260	38
^{60}Co	1173, 1332	250	40

[a] To convert to units of mrem y^{-1} pCi^{-1} g, multiply by 3.7x10^{-3}.
[b] To convert to units of μCi, divide by 37.

Twenty sources each for ^{137}Cs and ^{60}Co were fabricated; each source was approximately one-twentieth of 38 kBq (1 μCi). The ^{137}Cs sources were randomly placed inside the conduit interiors. A measurement was performed at the midpoint of each side of the pallet for 10 minutes, for a total of 40 minutes of count time. The process was repeated for nine additional measurement sets with the ^{137}Cs sources placed randomly each time. The ^{60}Co measurements were independently performed in the same manner. No shielding or collimation was used, and the detector was placed 1 meter (vertically) from the floor, and generally as close as possible to the pallet of steel conduit.

The efficiency, ϵ, for the region-of-interest (ROI) corresponding to the appropriate total absorption peak (TAP) for ^{60}Co or ^{137}Cs was calculated. First, the net counts in the TAP ROI were calculated by subtracting the Compton continuum counts in the ROI from the gross counts in the TAP ROI. Next, the net counts for the TAP ROI were divided by the total activity of the particular source, and the count time in minutes to determine efficiency in net counts per minute per kBq. The minimum detectable activity (MDA), in kBq, for the TAP ROI was calculated by the equation below, using the experimentally determined efficiency, where the BKG values, or continuum counts, were determined by the gross peak counts minus the net peak counts.

$$MDA\,[kBq] = \frac{3 + 4.65\sqrt{BKG\,[counts]}}{T\,[min]\;\varepsilon\;[net\ peak\ counts\ per\ min\ per\ kBq]}$$

Table 5.3 below summarizes the results of the ISGS measurements of the steel conduit pallet.

Table 5.3: Efficiency and MDA summary for ISGS measurements of scrap steel pallet (10-minute count time)

Radionuclide (keV)	Efficiency (Standard Deviation[a]) [net counts min^{-1} kBq^{-1}][b]	Efficiency 2-Sigma Range (net counts min^{-1} kBq^{-1})	MDA (kBq)[c]	MDA 2-Sigma Range (kBq)
^{137}Cs (662)	0.41 (0.09)	0.23 – 0.59	11	7 – 19
^{60}Co (1173)	0.33 (0.07)	0.19 – 0.47	11	7 – 22
^{60}Co (1332)	0.30 (0.06)	0.18 – 0.42	11	7 – 15

[a] Total propagated uncertainty.
[b] To convert to units of net counts min^{-1} μCi^{-1}, multiply by 37.
[c] To convert to units of μCi, divide by 37.

Multiple sets of measurements with randomly placed sources (in a non-uniform geometry) were performed to calculate an unbiased range of efficiencies for this particular geometry. Using the lower 5-percent confidence interval on the 2-sigma range of the efficiency from Table 5.3 allows the MDA to be conservatively reported for comparison to potential dose limits.

Table 5.3 shows that at an alternative dose criterion of 10 μSv/y (1 mrem/y), ISGS is a viable technology for 1 metric ton of 5-inch diameter steel conduit released from a nuclear facility. The upper range MDA for ^{137}Cs at 19 kBq (0.5 μCi) is below the total activity of 38 kBq (1.0 μCi) required to produce 10 μSv/y (1 mrem/y). The upper range MDA for ^{60}Co at 22 kBq (0.6 μCi) is below the total activity of 40 kBq (1.1 μCi) required to produce 10 μSv/y (1 mrem/y). However, if the more-restrictive dose limit of 1 μSv/y (0.1 mrem/y) is assumed, ISGS would lack the necessary sensitivity to detect 3.8 kBq (0.1 μCi) of either ^{60}Co or ^{137}Cs.

With the same ^{137}Cs and ^{60}Co sources used with the steel conduit experiment, a second experimental configuration consisting of a pallet of 148 insulated copper wires with a total weight of 490 kg (1,080 pounds) was set up. The only difference between the steel and copper experiment was that the count time was increased from 10 to 30 minutes per measurement to allow for the increased attenuation of the gamma-rays by the copper. Table 5.4 shows the dose calculation results.

Table 5.4: Calculated total activity for selected radionuclides using mass-based, critical-group dose factors for copper (4.9×10^5 g)

Radionuclide	Key Gamma(s) (keV)	Mean Dose Factor ($\mu Sv\ y^{-1}\ Bq^{-1}\ g$)[a]	Total Activity for 10 $\mu Sv\ y^{-1}$ (kBq)[b]
^{137}Cs	662	62	78
^{60}Co	1173, 1332	250	19

[a] To convert to units of mrem y^{-1} pCi^{-1} g, multiply by 3.7×10^{-1}.
[b] To convert to units of μCi, divide by 37.

Table 5.5 shows that for an alternative dose criterion of 10 μSv/y (1 mrem/y) and for the given experimental conditions, ISGS may not be a viable technology for a typical volume of copper released from a nuclear facility. The upper range MDA for ^{137}Cs at 89 kBq (2.4 μCi) is above the total activity of 78 kBq (2.1 μCi) required to produce 10 μSv/y (1 mrem/y). The upper range MDA for ^{60}Co at 59 kBq (1.6 μCi) is above the total activity of 19 kBq (0.5 μCi) required to produce 10 μSv/y (1 mrem/y). However, if the less-restrictive dose limit of 100 μSv/y (10 mrem/y) were adopted, ISGS would have the necessary sensitivity to detect 780 kBq (21 μCi) of ^{137}Cs or 190 kBq (5 μCi) of ^{60}Co in this copper matrix.

Table 5.5: Efficiency and MDA summary for ISGS measurements of scrap copper pallet (30-minute count time)

Radionuclide (keV)	Efficiency (Standard Deviation[a]) [net counts min^{-1} kBq^{-1}][b]	Efficiency 2-Sigma Range (net counts min^{-1} kBq^{-1})	MDA[c] (kBq)[d]	MDA 2-Sigma Range (kBq)
^{137}Cs (662)	0.13 (0.04)	0.05 – 0.21	33	22 – 89
^{60}Co (1173)	0.11 (0.03)	0.05 – 0.17	37	22 – 85
^{60}Co (1332)	0.09 (0.02)	0.05 – 0.13	30	22 – 59

[a] Total propagated uncertainty.
[b] To convert to units of net counts min^{-1} μCi^{-1}, multiply by 37.
[c] MDA values calculated for a 10 minute count.
[d] To convert to units of μCi, divide by 37.

5.4.1.5 ISGS Measurement Considerations

The average contamination in the material determined by the ISGS system should be representative of the true average for comparison to the volumetric guidelines. For materials with uniform or near-uniform contamination, only one measurement, from any orientation, may sufficiently determine the average contamination. For materials that do not have uniform contamination, different ISGS measurement approaches may be necessary to determine a more accurate average contamination level. For instance, for Class 1 materials that potentially contain small elevated areas of radioactivity, the ISGS calibration should address the impact that these small elevated areas of radioactivity have on the efficiency of this survey technique, so that an accurate average contamination level is determined.

One approach is to perform multiple measurements at different angles around the material, such as all four sides, and then average the measurement results. Another approach, which is commonly used in drum counters, is to rotate the material during the measurement time. However, rotating a pallet of pipes or wire can be unwieldy, if not impossible, so to effectively rotate the material, one might perform part of one measurement at each location around the material. For example, suppose a count time of 40 minutes was required to meet the required sensitivity and the material was to be measured from all four sides. The first 10 minutes of the single measurement would be performed, and then the acquisition would be paused while the detector was moved to the second measurement location, and then the acquisition would continue for another 10 minutes. This process would be repeated for the remaining two positions.

5.4.2 Volume Counters

Various designs of volume counters can be used to quantify surface activity or total activity. Volume counters, while generally designed for specific counting applications, have common characteristics. These include a counting chamber, array of detectors, and electronic package for analysis.

The counting chambers are designed specifically for the measurement application. The size determines what type of materials or containers the system is capable of measuring. Volumes range from small items to large shipping containers. A variety of detectors, including gas proportional, plastic and NaI scintillators, HPGe semiconductors, and long-range alpha detection configurations, are used in volume counters, depending on the application. Many designs focus on detecting specific waste streams (e.g., transuranic waste, with a high throughput). Systems designed to quantify alpha and/or beta surface activity use gas proportional and plastic scintillator detectors or long-range alpha detection. Plastic and NaI scintillators and HPGe semiconductor detectors are used for volumetric gamma radioactivity.

Calibrations are usually performed with standard packages or suitable geometries containing sources of known activity. Shielded configurations are frequently used to reduce the background, thereby increasing the signal-to-noise ratio. In many systems, the shielded configuration completely surrounds the material to be measured (i.e., 4π counting geometry). An example of this configuration is the drum counter, in which a conveyor belt typically moves the drum into the counting chamber, where the drum is usually rotated during the measurement to obtain a more representative average. After the count, the drum is then moved out and another drum counted.

Considerations for applying volume counters do not vary significantly from the individual application of each of the mentioned detectors. For example, gas proportional detectors need to be calibrated to a calibration source representative of the radioactivity, and the considerations listed for ISGS apply for systems using HPGe detectors for volume counting.

5.4.3 Portal Monitors

A common example of a portal monitor is a truck or rail car scrap metal radiation detection system. These use large-area plastic scintillation detectors to detect buried radioactive sources in scrap metal. The radioactive sources are identified by detecting small changes in the ambient gamma background. Entities in the United States have used portal monitors upon receipt of materials in incoming shipments. Advances in portal monitor technology may one day allow surveyors to use this technique as a primary material survey technique.

5.5 Laboratory Analytical Methods

Sections 5.5.1 – 5.5.3 discuss the laboratory analyses for hard-to-detect nuclides and various media matrices (i.e., bulk materials). This discussion ties in with the conventional survey approach, in the sense that some statistical samples (such as ^{3}H in concrete) are much more complex to analyze than others (such as simple direct measurement of surface activity).

5.5.1 Representative Sampling and Laboratory Analysis

Laboratory analysis provides the greatest level of accuracy and precision, with the lowest detection levels. Indeed, some techniques have remarkable detection limits. For example, an inductively coupled plasma mass spectrometer (ICP-MS) can have detection limits less than 1 part per quintillion (ppq). Furthermore, laboratory analyses usually do not suffer from the calibration issues that plague ISGS and *in toto* systems (namely, the expense associated with producing or obtaining reference materials needed to develop or validate a calibration).

Laboratory methods for measuring radioactivity cover a broad range of techniques. It is difficult to reduce all of the standard techniques to a single recipe. However, once the samples are collected, they are usually subject to a destructive process (gamma spectrometry is a notable exception), which changes the physical or chemical state of the sample. Next, the samples are usually purified or chemically separated into a solution to which a tracer is usually added. The sample is then put in a form that will allow it to be counted efficiently. This preparation can be time-consuming and costly. Table 5.6 provides cost information on routine radiochemical analysis. Ultimately, the decision to follow an approach that uses laboratory techniques will balance data quality objects against available resources.

Table 5.6: Cost information on routine radiochemical analysis

Radiation	Technique/Instrumentation	Estimated Cost per Measurement†	Relative degree of precision
	Energy Spectrometry		
α	Alpha spectroscopy using solid-state semiconductor detector, (surface barrier detector[1]).	$250 – $ 400	high
β	Gross activity measurements using gas-flow proportional counter (typically for swipe samples)	$50	low
	Beta spectroscopy using liquid scintillation counting	$100 – $200	high
	Gamma and X-ray spectroscopy using NaI scintillator	$100 – $ 200	medium
γ	Gamma and X-ray spectroscopy using germanium detector	$100 – $ 200	high
	Mass Spectrometry		
Inductively Coupled Plasma Mass Spectrometer (ICP-MS)		> $4000◊	
Chemical speciation laser ablation/mass spectrometer		> $4000	

† From Appendix H of the MARSSIM

◊Recent data from commercial laboratories suggest that this value should be closer to the value for alpha spectroscopy

5.5.2 Sample Collection

The assay process actually begins with the collection of samples. The critical issue regarding the use of laboratory methods is that the object that is sampled must be disturbed; that is, some amount of material must be removed from the object. The amount can range from a fraction of a gram in the case of a swipe or wipe sample for removable alpha contamination, to several kilograms in the case of soil sampling. While extracting samples from surface soil, for example, is relatively simple and involves the use of trowels and augers, the collection of samples from steel and concrete can be very difficult. Sampling these materials requires chisels, hammers, drills, and other more specialized equipment. The collection of samples, specifically the number and location of the samples, is fundamental to characterizing and quantifying the contamination. Morever, the number and location of the samples should follow the DQO Process (see Section 3).

5.5.3 Sample Preparation

Most samples that are collected cannot be assayed directly, but should be converted to a suitable form for assay. The type and energy of the radiation to be measured determine the ultimate form. For example, samples containing α or low-energy β activity have problems with self-absorption and, therefore, the form of the sample should be as thin as possible. More importantly, chemical purification may be required if interferences are anticipated. Table 5.7 provides a general indication of the sample preparation for α and β assay for low to medium activities in solid samples. The preparation of samples for gamma-ray analysis is usually less involved. For example, the preparation of soil involves nothing more than drying and homogenization. For a more complete listing of standard laboratory methods and instruments, see the MARSSIM; for specific radiochemical techniques, consult the Environmental Measurements Laboratory (EML) Procedures Manual (U.S. DOE, 1990) and Radiochemical Analytical Procedures for Analysis of Environmental Samples (EPA, 1979).

Table 5.7: Sample preparation for α and β assay for low to medium radioactivity levels

Detector	Sample preparation	Preparation time
Sample preparation for α assay (solid sample)		
Solid-State Semiconductor	If the sample is thin, count directly. If not, dissolve and redeposit as a thin source	a week or more
Liquid Scintillator	Dissolve in suitable solvent and heat as liquid, or count directly as a suspension in a gel	several days to a week
Sample preparation for β assay (solid sample)		
Proportional Counter	May be counted directly unless low energy β (< 50 keV) requires pretreatment	day
Solid-State Semiconductor	Same as proportional counter	day
Liquid Scintillator	Should be dissolved in a suitable solvent and treated as a liquid sample. Can be counted directly as a suspension in a suitable gel mixture.	a week or more

5.6 Assay Quality Assurance

Sections 5.6.1 – 5.6.3 address quality assurance (QA) issues involving the measurement systems associated with clearance surveys, including the calibration process, data quality indicators, and quality control (QC). In general, any assay or measurement strategy must develop and follow a quality assurance process, which should be part of an overall quality assurance program. For guidance in establishing quality assurance programs, see ASME NQA-1-1994, EPA Guidance Document QA/G-5, and Regulatory Guide 4.15 (NRC, 1979). At a minimum, the quality assurance program should address the quality following elements:

- organizational structure and responsibilities
- procedures and instruction
- records
- personnel qualifications
- quality control of measurement systems

5.6.1 The Calibration Process

An important consideration associated with the calibration of instrumentation for use in clearance surveys (see Appendix B) is the lack of appropriate reference materials and guidance on methods to calibrate these systems. Therefore, a calibration process should be developed and documented in a standard operating procedure (SOP). For general requirements that apply to calibrations see ANSI/ASQC M1-1987 and ANSI/ISO/IEC 17025:2000.

The following items should be part of the calibration process and included in a QA document:

- Describe the type of instrument to be calibrated.
- Describe the calibration method in sufficient detail so that others can duplicate the method.
- Justify and document the calibration methods.
- Describe how calibration data will be analyzed.
- List the parameters, quantities, and ranges to be determined.
- Describe any corrective action, including recalibration, that will be taken if calibration data fail to meet the acceptance criteria.
- Describe the calibration standards. If the standards are not traceable (to NIST or some other national certifying organization), describe how the standards will be prepared. Any method used to verify the certified value of the standard independently should also be described.
- Describe the frequency of the calibration and whether the frequency is related to any temporal variation of the system.

5.6.2 Data Quality Indicators

Data quality indicators (DQIs) are qualitative and quantitative descriptors used in interpreting the degree of acceptability or utility of data. The principal DQIs are *precision, bias, representativeness, comparability, and completeness*. These are referred to as the "PARCC" parameters, where the "A" refers to accuracy rather than bias, but the two are generally regarded as synonymous. Of the five DQIs, precision and bias are crucial when evaluating the performance of an instrument or measurement method. Establishing acceptance criteria for precision and bias sets quantitative goals for the quality of the data generated by measurement instrument. DQIs are established during the planning phase of the DQO Process. More information on DQIs may be found in the MARSSIM.

Comparability is also important, in that it can establish the validity of a measurement technique, calibration method, or instrument. For example, calibrations of CSM, ISGS, and *in toto* systems may need to establish comparability with representative sampling and laboratory techniques. There are several examples of this approach involving ISGS (DOE 1999a, DOE 1999b, Kalb *et al.* 2000). Two of the studies (DOE 1999a and Kalb *et al.* 2000) utilize the DQO Process. The intent of these studies was not to show that ISGS produces data that is indistinguishable from the baseline approach (sampling and laboratory analysis) on a sample-to-sample basis, but that the decision drawn from the data is the same.

An effective tool for evaluating sources of bias, providing a mechanism for standardization and establishing traceability are intercomparison or intercalibration exercises. Such exercises have long been a key element in quality assurance programs for field measurement techniques.

5.6.3 Quality Control

Quality control (QC) is an important element of the quality assurance process. The purpose of QC is to ensure that the measurements and other data-producing systems operate within defined performance limits as specified in planning (EPA 1998a). QC activities help to identify sources of error and uncertainty, as well as the impact these quantities will have on the decisionmaking process. QC activities involve the use of QC samples to detect when attributes of the measurement process are exceeding their performance limits so that corrective actions can be initiated. The measurement attributes that QC samples monitor include contamination, calibration drift, bias, and precision. The following is a brief description of standard QC samples.

Blanks are samples that contain little or no radioactivity, and none of the radionuclide of interest.

Performance Evaluation (PE) Matrices are samples with enhanced levels of radioactivity (compared to a surrogate material) at a known concentration of the radionuclide(s) of interest.

Calibration Checks are samples containing a source or radioactive material, which is independent of a calibration standard, and can ensure that the calibration remains in a state of statistical control.

Replicates are samples that are measured repeatedly to check the precision of the system.

The quality assurance document should describe the QC procedure, which should identify the QC checks that are to be performed, the frequency with which they will be performed, their acceptance criteria, and a correction action plan to be followed if the acceptance criteria are not met. Table 5.8 provides additional information on QC samples.

Table 5.8: Suggested QC checks for measurement systems used in clearance surveys

QC Check	Measurement Attribute	Frequency	Corrective Action	Comments
Calibration check	Calibration drift	beginning and end of every shift	recalibrate instrument	control charts are a useful method of documenting drift
PE Spike	Bias	on a change of material, matrix, radionuclide mix, and/or environmental/ operating conditions (if it can be shown that these properties affect the measurement result)	adjust measurement parameters (e.g., count time, belt speed, standoff distance) reevaluate measurement method and/or instrumentation	not readily available for all types of clearance materials; user may have to prepare their own
Blank	Contamination	on a change of material classification (e.g., measuring Class 2 or 3, or non-impacted material after measuring a Class 1 or impacted material) whenever a measurement has a reasonable chance of contaminating the instrument	decontaminate instrument adjust background or baseline	used to establish a baseline or background value used to adjust or correct measurement results
Replicate	Precision	once/day or once/shift	check environmental or operating parameters system might be unstable and need repair	

5.7 Clearance Survey Examples

The clearance survey examples presented on the following pages illustrate possible clearance survey approaches for pipe sections being released from a power reactor facility. The flow diagram for clearance of solid materials (Section 2) served as a guide for developing these examples; the letters in the examples correspond to the steps in Figure 2.1.

Example 1 Clearance of small-bore pipes from nuclear power reactor

a. Evaluate the physical description of the solid material.

The solid material being considered for release is small-bore pipe (steel). The material survey unit consists of approximately 60 sections of pipe and conduit, each of which is 1.2 to 1.8 m in length. The diameter of each pipe section is less than 6 cm, with a total interior surface area of 17 m^2 and a weight of 2 tons. The pipe interiors are considered to be inaccessible with conventional hand-held detectors.

b. Evaluate and document process knowledge and characterization of the solid material.

The small-bore pipes are from a nuclear power plant. Process knowledge indicates that the pipes were used to transport radioactive liquids from the nuclear laundry. The radionuclide mixture for the nuclear power reactor consists of a number of radionuclides, including fission products, activation products, and even trace quantities of transuranics.

During characterization, three samples of pipe residue were collected and analyzed from the total pipe population. The radionuclide mixture was as follows:

^{60}Co	15%
^{137}Cs	27%
$^{90}Sr(^{90}Y)$	8%
^{14}C	13%
^{55}Fe	11%
^{63}Ni	6%
^{3}H	20%

Therefore, the radionuclide mixture from characterization confirms the process knowledge that fission and activation products comprise the contamination. The mixture includes radionuclides that are readily-detected (^{60}Co, ^{137}Cs, $^{90}Sr(^{90}Y)$), as well as those that are hard-to-detect (^{3}H, ^{63}Ni, and ^{55}Fe).

c. Is the material impacted?

Yes, these small-bore pipe sections are certainly impacted, given that they were used to transport radioactive liquids.

d. Specify release criteria and conditions for the solid material.

For this example, Regulatory Guide 1.86 will be used. The surface activity guideline for all radionuclides (except $^{90}Sr(^{90}Y)$) is 5,000 dpm/100 cm^2 averaged over 1 m^2. The guideline for $^{90}Sr(^{90}Y)$ is 1,000 dpm/100 cm^2.

e. Classify the material.

The small-bore pipe sections are Class 1. This classification is based on the fact that the material was designed to be in contact with radioactivity, as further supported by the characterization results.

f. Is clearance an option?

Yes, the licensee in this example has decided to perform a clearance survey.

g. Consider the survey approach based on the nature of the material and contamination.

Given that the interior of the pipe sections is potentially contaminated, it will be necessary to cut the pipes along their lengths (resulting in semi-cylindrical sections). The nature of the radioactivity suggests that beta-sensitive detectors would work well.

h. Can scanning be used to release the material?

Yes, the proposed clearance survey approach is to scan the interior of the semi-cylindrical pipe sections using GM detectors. Before this approach can be implemented, it is necessary to demonstrate that the scan MDC is less than the $DCGL_C$.

i. Application of DCGLs.

To demonstrate compliance with the clearance release criteria, the clearance survey will consist of surface scans with a GM detector. Given the radioactive decay emissions from these radionuclides, the GM will respond to gross beta radiation. Therefore, it is necessary to calculate the gross activity $DCGL_C$ for surface activity using the following equation:

$$Gross\ Activity\ DCGL_C = \frac{1}{\left(\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \ldots \frac{f_n}{DCGL_n}\right)} \quad (14)$$

where f_1, f_2, etc. are the fractional amounts of each radionuclide present.

A simplifying observation is that 92 percent of the radionuclide mixture consists of radionuclides for which the surface activity guideline is 5,000 dpm/100 cm^2, while $^{90}Sr(^{90}Y)$ makes up 8 percent with a guideline of 1000 dpm/100 cm^2. Substituting into the above equation, the gross activity DCGL is 3,800 dpm/100 cm^2.

j. Determine background.

Measurements were performed on similar, non-impacted pipe sections to determine the GM background; this resulted in a background level of approximately 60 cpm.

2806 k. Determine scan MDC.

2807 Scan MDCs are determined from the MDCR by applying conversion factors to obtain results in terms of
2808 measurable surface activities. The scan MDC for a material surface can be expressed as

$$scan\ MDC = \frac{MDCR}{\sqrt{p}\ \varepsilon_i\ \varepsilon_s}$$

2809 where the minimum detectable count rate (MDCR), in counts per minute, can be written

$$MDCR = d' * \sqrt{b_i} * (60/i)$$

2810 d' = detectability index (the value can be obtained from MARSSIM Table 6.5),
2811 b_i = background counts in the observation interval,
2812 i = observational interval (in seconds), based on the scan speed and areal extent of the contamination
2813 (usually taken to be 100 cm^2),
2814 ε_i is the instrument or detector efficiency (unitless),
2815 ε_s is the surface efficiency (unitless), and
2816 p is the surveyor efficiency (usually taken to be 0.5).

2817 The scan MDC is determined for a background level of 60 cpm and a 2-second observation interval using
2818 a GM detector (b_i = 2 counts). For a specified level of performance at the first scanning stage of 95-
2819 percent true positive rate and 25-percent false positive rate, d'equals 2.32 and the MDCR is 98 cpm.

2820 Before the scan MDC can be calculated, it is necessary to determine the total efficiency for the
2821 radionuclide mixture.

	ε_i	ε_s	Radionuclide Fraction	Weighted Efficiency
2822 ^{60}Co	0.05	0.25	0.15	1.88×10^{-3}
2823 ^{137}Cs	0.08	0.5	0.27	1.08×10^{-2}
2824 ^{90}Sr	0.12	0.5	0.08	4.80×10^{-3}
2825 ^{14}C	0.03	0.25	0.13	9.75×10^{-4}
2826 ^{55}Fe	0	0.25	0.11	0
2827 ^{63}Ni	0.01	0.25	0.06	1.50×10^{-4}
2828 ^{3}H	0	0	0.2	0
2829			**Total Weighted Efficiency**	**1.9×10^{-2}**

Using a surveyor efficiency of 0.5 and the total weighted efficiency of 1.9×10^{-2}, the scan MDC is calculated as

$$Scan\ MDC = \frac{98}{\sqrt{0.5}\ (1.9E\text{-}2)} = 7{,}400\ dpm/100\ cm^2\ \ (1.2\ Bq/cm^2)$$

l. Is the scan MDC less than the $DCGL_C$?

No, the scan MDC of 7,400 dpm/100 cm^2 (1.2 Bq/cm^2) is not less than 3,800 dpm/100 cm^2 (0.6 Bq/cm^2).

m. Can the scan MDC be reduced?

It is not likely that modifying the scanning parameters will lower the scan MDC to a value less than the $DCGL_C$. (Note: If the scan MDC could be sufficiently reduced below the $DCGL_C$, the next step is to evaluate the instrument's ability to automatically document scan results (step *o*).)[4]

n. Is another clearance survey design feasible?

Since the scan MDC is not sufficiently sensitive, the next step is to determine whether conventional static measurements are feasible. Example 2 provides the details of the design.

Example 2 Clearance of small-bore pipes from nuclear power reactor (using statistical design for static direct measurements)

Based on the information obtained in Example 1, step *h* in the flow diagram of Figure 2.1 results in the decision that scanning with a GM detector cannot be used to release the pipe sections. This example continues from step *n* in Example 1 (now at the right side of Figure 2.1).

i. Application of DCGLs.

To demonstrate compliance with the clearance release criteria, the clearance survey will consist of static direct measurements of surface activity using a GM detector. The gross activity $DCGL_C$ for surface activity determined in Example 1 is the same for this example (i.e., the gross activity $DCGL_C$ is 3,800 dpm/100 cm^2).

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[4] o. Can scanning instrument automatically document results? (Note: This step, as well as step *p*, is not possible in this example because the scan MDC is not less than the $DCGL_C$; it is covered in this footnote for illustration only).

p. If the scanning instrument can automatically document results, the material survey unit is scanned and the results are automatically logged. Since it is a Class 1 survey unit, 100 percent of the pipe sections are scanned. However, if the scanning instrument cannot automatically document results, it is necessary to collect a number of static direct measurements to serve as scan documentation, in addition to scanning 100 percent of the Class 1 material survey unit. The number of these measurements should be determined using the DQO Process, and may be determined using a statistically based sampling design.

j. Determine background.

Fifteen measurements, as determined based on the WRS test (step *p*), were performed on non-impacted pipe sections to determine the GM background. The mean background was 60 cpm, with a standard deviation of 8 cpm.

k. Determine the static MDC.

The static MDC for the GM detector can be calculated as

$$MDC = \frac{3 + 4.65\sqrt{C_B}}{\epsilon_i \, \epsilon_s \, T \, \frac{probe\ area}{100\ cm^2}}$$

where C_B is the background count in time, T, for paired observations of the sample and blank, ϵ_i is the instrument efficiency, and ϵ_s is the surface efficiency. However, before the static MDC can be calculated, it is necessary to determine the total efficiency for the radionuclide mixture. [Note: The instrument efficiencies for the GM detector used for static measurements (based on the detector's response to a source area equal to its physical probe area of 20 cm²) are higher than instrument efficiencies for the GM detector used for scanning (based on the detector's response to a source area of 100 cm²), by a factor of 5.]

	ϵ_i	ϵ_s	Radionuclide Fraction	Weighted Efficiency
^{60}Co	0.25	0.25	0.15	9.40×10^{-3}
^{137}Cs	0.40	0.5	0.27	5.40×10^{-2}
^{90}Sr	0.60	0.5	0.08	2.40×10^{-2}
^{14}C	0.15	0.25	0.13	4.88×10^{-3}
^{55}Fe	0	0.25	0.11	0
^{63}Ni	0.05	0.25	0.06	7.50×10^{-4}
^{3}H	0	0	0.2	0
		Total Weighted Efficiency		$\mathbf{9.3 \times 10^{-2}}$

Therefore, the static MDC for the GM for 1-minute counts is

$$MDC = \frac{3 + 4.65\sqrt{60}}{9.3E\text{-}2\ (1\ \text{min})\ \frac{20\ cm^2}{100\ cm^2}} = 2{,}100\ dpm/100\ cm^2\ \ (0.4\ Bq/cm^2)$$

l. Is the static MDC less than the $DCGL_C$?

Yes, the static MDC of 2,100 dpm/100 cm^2 is less than the $DCGL_C$ of 3,800 dpm/100 cm^2.

p. Perform clearance survey based on statistical sampling design for the number of direct measurements of surface activity.

The WRS test can be used to determine the number of surface activity measurements needed for the clearance survey. The number of data points necessary for this material survey unit is determined through the DQO Process. Specifically, the sample size is based on the $DCGL_C$, the expected standard deviation of the radionuclides in the pipe sections, and the acceptable probability of making Type I and Type II decision errors.

- The gross activity $DCGL_C$ is 3,800 dpm/100 cm^2.

- Process knowledge, coupled with results from characterization surveys, was used to estimate the contamination on the pipe sections. The contamination, as measured in gross cpm with a GM detector, averaged 82 cpm, with a standard deviation of 18 cpm.

- Other DQO inputs include the LBGR set at the expected contamination level on the pipe sections (82 - 60 cpm, or 22 cpm), and Type I and II errors of 0.05 and 0.01 respectively.

The $DCGL_C$, and the expected standard deviation of the material survey unit and background measurements are used to estimate the relative shift, Δ/σ.

First, it is necessary to convert the $DCGL_C$ into the same units as the standard deviation:

$$gross\ activity\ DCGL_C = (3{,}800\ dpm/100\ cm^2) \cdot (9.3E\text{-}2)\ 20/100 = 70.7\ cpm$$

The larger of the values of the estimated measurement standard deviations from the survey unit and the reference area should be used. Since the estimated standard deviation in the survey unit is 18 and that for the reference area is 8, the survey unit value of $\sigma = 18$ will be used to calculate the relative shift. The relative shift can now be calculated: $(70.7 - 22)/18 = 2.7$.

Table 5.3 in MARSSIM (1997) provides a list of the number of data points to demonstrate compliance using the WRS test for various values of Type I and II errors and Δ/σ. For $\alpha = 0.05$ and $\beta = 0.01$, the required sample size is about 15 direct measurements for this material survey unit and 15 measurements on non-impacted pipe sections (background).

The scan coverage for these pipe sections is 100 percent because of their classification (i.e., Class 1). Note, however, that the scan MDC is 7,400 dpm/100 cm^2; therefore surface activity levels between the $DCGL_C$ (3,800 dpm/100 cm^2) and the scan MDC will likely be missed during scanning. At a minimum, however, scanning can detect surface activity at a level of 7,400/3,800, or about two times the $DCGL_C$[5]. A provision for area factors as a function of specific areas of materials may be appropriate to serve as a possible driver for collecting additional direct measurements. If not, the DQO Process should be used to assess the risk of missing an area with concentration between the $DCGL_C$ and the scan MDC, and whether the material is candidate for release.

Direct measurement locations are determined by random number generation. Fifteen pairs of random numbers are generated, with the first number specifying the particular pipe section to be measured, and the second number determining the distance from the end of the pipe section for the direct measurement.

Example 3 Clearance of small-bore pipes from nuclear power reactor (using *in situ* gamma spectrometry)

This clearance survey approach is similar to the approach illustrated in Example 2, with two major exceptions. First, this approach does not require the pipes to be cut in half; in fact, the entire material survey unit is measured and results in minimal handling of the material. Second, the clearance survey is based on one "total" measurement, rather than a statistically based sampling design. Steps *a* through *f* are the same in Example 3 as they were for the first two examples.

g. Consider survey approach based on nature of material and contamination.

Given that the interior of the pipe sections is potentially contaminated with some gamma-emitting radionuclides among the mix, the use of *in situ* gamma spectrometry (ISGS) is considered as a clearance survey approach.

h. Can scanning be used to release material?

The proposed clearance survey approach is to use ISGS measurements; therefore, scanning is not used to release the pipe sections.

i. Application of DCGLs.

Considering the radionuclide mixture provided in step *b* (shown in Example 1), ^{60}Co and ^{137}Cs comprise 42 percent of the radioactivity. Therefore, these two radionuclides are measured using ISGS, and are used as surrogates for the entire mix of radionuclides. In order to use this approach, it is necessary to assume that this mixture is representative of the potential contamination on the pipe sections (refer to step *b*).

It is necessary to convert the surface activity guidelines (from RG 1.86) to total activity limits. This is performed for each radionuclide by multiplying the surface activity guideline by the total surface area of the pipes in the material survey unit (17 m^2). For example, the total dpm that corresponds to 5,000 dpm/100 cm^2 can be calculated as

$$(5{,}000 \text{ dpm}/100 \text{ cm}^2) \times (17 \text{ m}^2) \times (10{,}000 \text{ cm}^2/1 \text{ m}^2) = 8.5\text{E}6 \text{ dpm}$$

[5]For comparison, Regulatory Guide 1.86 provides for an effective area factor of 3.

Each of the radionuclides, with the exception of $^{90}Sr(^{90}Y)$, has a surface activity guideline of 5,000 dpm/100 cm^2. The total activity limit for $^{90}Sr(^{90}Y)$, based on its 1,000 dpm/100 cm^2 guideline, is $1.7x10^6$ dpm.

Returning to the use of ^{60}Co and ^{137}Cs as surrogates, it is necessary to modify the $DCGL_C$ for these two radionuclides to account for all of the other radionuclides. First, note that the limit for both ^{60}Co and ^{137}Cs is $8.5x10^6$ dpm; therefore, when both are measured, the sum of both radionuclides should not exceed $8.5x10^6$ dpm (when they are the only radionuclides present). Equation I-14 on page I-32 o f the MARSSIM can be used to calculate the modified $DCGL_C$ for Co+Cs:

$$DCGL_{Co+Cs,mod} = \frac{1}{\left(\frac{1}{D_1} + \frac{R_2}{D_2} + \ldots \frac{R_n}{D_n}\right)}$$

where D_1 is the $DCGL_C$ for the sum of ^{60}Co and ^{137}Cs ($8.5x10^6$ dpm), D_2 is the $DCGL_C$ for the first radionuclide ($^{90}Sr(^{90}Y)$) that is being inferred by ^{60}Co and ^{137}Cs. R_2 is the ratio of concentration of the $^{90}Sr(^{90}Y)$ to that of the sum of ^{60}Co and ^{137}Cs (8% divided by 42%, or 0.19), and R_3 is the ratio of the concentration of ^{14}C to that of the sum of ^{60}Co and ^{137}Cs (or 0.31). Therefore, $DCGL_{Co+Cs,\,mod}$ can be calculated for the mixture as follows:

$$DCGL_{Co+Cs,mod} = \frac{1}{\left(\frac{1}{8.5E6} + \frac{0.19}{1.7E6} + \frac{0.31}{8.5E6} + \frac{0.26}{8.5E6} + \frac{0.14}{8.5E6} + \frac{0.476}{8.5E6}\right)} = 2.7E6\ dpm$$

Therefore, to demonstrate compliance, the ISGS result should be less than $2.7x10^6$ dpm (1.22 µCi) for the sum of ^{60}Co and ^{137}Cs.

j. Determine background.

Since neither ^{60}Co nor ^{137}Cs is present naturally in the material (pipe sections), the background value (i.e., Compton continuum) for each radionuclide's region of interest (ROI) was determined from an ambient count at the location where the pipe section clearance measurements will be performed. The count time should be long enough to result in sufficiently sensitive MDC.

k. Determine static MDC.

The static MDC for the *in situ* gamma spectrometer can be calculated as

$$MDC = \frac{3 + 4.65\sqrt{BKG}}{\varepsilon T}$$

where BKG is the background continuum counts determined in time T, and ε is the efficiency in net peak counts per minute per activity (μCi or Bq). This MDC is the general MDC for the measurement process, rather than an individual MDC for each measurement.

The measurement protocol consisted of four 10-minute measurements at the midpoint of each side of the material survey unit. The efficiency for a particular distribution of radioactivity within the pipe sections was determined by randomly positioning a known quantity of ^{60}Co and ^{137}Cs radionuclide sources within a non-impacted geometry of pipe sections. The efficiencies for the ^{60}Co (1,173 keV) ranged from 7.2 to 17.3 net counts per minute per μCi, while the efficiencies for the ^{137}Cs ranged from 8.8 to 21.8 net counts per minute per μCi. To be conservative, the MDCs for both ^{60}Co and ^{137}Cs were calculated for the lowest efficiencies observed. The MDCs for ^{60}Co and ^{137}Cs were 0.6 and 0.5 μCi, respectively.

l. Is the static MDC less than the $DCGL_C$?

Yes, the static MDCs for ^{60}Co and ^{137}Cs are less than the $DCGL_C$ of 1.22 μCi. If either of the MDCs were greater than the $DCGL_C$ of 1.22 μCi, step *m* would be performed to determine whether the MDCs could be reduced (e.g., by using longer count times).

p. Perform *in toto* survey.

Perform clearance survey based on ISGS measurements for ^{60}Co and ^{137}Cs. Each measurement consists of four 10-minute measurements at the midpoint of each side of the material survey unit. The total activity for both ^{60}Co and ^{137}Cs is summed, and then compared to the $DCGL_C$ of 1.22 μCi. Survey results are documented.

6 DATA QUALITY ASSESSMENT

6.1 Overview

This section discusses the interpretation of survey results, focusing primarily on those of the clearance survey. Interpreting a survey's results is most straightforward when measurement data are entirely higher or lower than the $DCGL_W$. In such cases, the decision that a survey unit meets or exceeds the release criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit's measurements are neither clearly above nor entirely below the $DCGL_C$. Nevertheless, the survey design *always* makes use of the statistical tests in helping to ensure that the number of sampling points and the measurement sensitivity are adequate, but not excessive, for the decision to be made.

Section 6.2 discusses the assessment of data quality, while Sections 6.3 and 6.4 deal with the application of the statistical tests used in the decisionmaking process, and Section 6.5 focuses on the evaluation of the test results.

6.2 Data Quality Assessment

Data quality assessment (DQA) is a scientific and statistical evaluation that determines whether the data are of the right type, quality, and quantity to support their intended use. There are five steps in the DQA Process:

- Review the data quality objectives (DQOs) and survey design.
- Conduct a preliminary data review.
- Select the statistical test.
- Verify the assumptions of the statistical test.
- Draw conclusions from the data.

The effort expended during the DQA evaluation should be consistent with the graded approach used in developing the survey design. The EPA guidance document QA/G-9 QA00 Update (EPA 2000) provides more information on the DQA Process. Data should be verified and validated as described in the site quality assurance project plan (QAPP) for clearance surveys. Information on developing QAPPs is contained in EPA guidance document QA/G-5 (EPA 1998a).

6.2.1 Review the Data Quality Objectives (DQOs) and Sampling Design

The first step in the DQA evaluation is a review of the DQO outputs to ensure that they are still applicable. For example, if the data suggest that the survey unit was misclassified as Class 3 instead of Class 1, the DQOs should be redeveloped for the correct classification.

The sampling design and data collection should be reviewed for consistency with the DQOs. For example, the review should verify that the appropriate number of samples were taken in the correct locations and that they were analyzed with measurement systems with appropriate sensitivity.

In cases where the survey does not involve taking discrete measurements or samples (i.e., scanning only, CSM, or *in toto* surveys), it is imperative that the MDCs be calculated realistically, and that they truly reflect at least a 95-percent chance that concentrations at or above that level will be detected. Periodic QA measurements must be made to ensure that the measurement systems remain within acceptable calibration and control limits.

When discrete sampling is involved, determining that the sampling design provides adequate power is important to decisionmaking, particularly in cases where the levels of contamination are near the $DCGL_C$. This can be done both prospectively, during survey design to test the efficacy of a proposed design, and retrospectively, during interpretation of survey results to determine that the objectives of the design are met. The procedure for generating power curves for specific tests is discussed in Appendix I to the MARSSIM. Note that the accuracy of a prospective power curve depends on estimates of the data variability, σ, and the number of measurements. After the data are analyzed, a sample estimate of the data variability, namely the sample standard deviation (s) and the actual number of valid measurements will be known. The consequence of inadequate power is that a survey unit that actually meets the release criterion has a higher probability of being incorrectly deemed *not* to meet the release criterion.

6.2.2 Conduct a Preliminary Data Review

To learn about the structure of the data — identifying patterns, relationships, or potential anomalies — one can review quality assurance (QA) and quality control (QC) reports, prepare graphs of the data, and calculate basic statistical quantities.

6.2.2.1 Data Evaluation and Conversion

Quality assurance reports that describe the data collection and reporting processes can provide valuable information about potential problems or anomalies in the data. EPA Report QA/G-9 (EPA 2000) recommends a review of (1) data validation reports that document the sample collection, handling, analysis, data reduction, and reporting procedures used; (2) quality control reports from laboratories or field stations that document measurement system performance, including data from check samples, split samples, spiked samples, or any other internal QC measures; and (3) technical systems reviews, performance evaluation audits, and audits of data quality, including data from performance evaluation samples. This report also suggests that when reviewing QA reports, particular attention should be paid to information that can be used to check assumptions made in the DQO Process, especially any anomalies in recorded data, missing values, deviations from standard operating procedures, or the use of nonstandard data collection methodologies.

Verification of instrument calibrations and calculations of minimum detectable concentrations (MDCs) are particularly important to surveys of solid materials. Clearly, MDCs must be capable of detecting contamination at the $DCGL_C$. When making quantitative comparisons of the average of survey data to a limit, the MARSSIM recommends that the MDC target should be 10–50 percent of the $DCGL_C$. This is an expression of the fact that a simple detection decision does not address the relative uncertainty of the data value obtained. The minimum quantifiable concentration (MQC) is often defined as the smallest concentration that can be measured with a relative standard uncertainty of 10 percent. As a rule of thumb mentioned previously, the MDC is generally about 3 to 4 times the standard uncertainty of repeated background or blank measurements. An extension of this rule of thumb is that the MQC is about 10 times the standard uncertainty. Hence, if one wishes to not merely detect but also quantify concentrations near the $DCGL_C$, the MQC should be no larger than the $DCGL_C$. Combining the approximations for the MQC as 10 times the uncertainty and the MDC as about 3 or 4 times the uncertainty, the MDC should be about one-third of the MQC. Thus, the recommendation that the MDC should be 10–50 percent of the $DCGL_C$ is really an expression of the fact that the MQC should be no larger than the $DCGL_C$.

These rough guides can sometimes point out inconsistencies or shortcomings in the data analysis. For example, suppose that the $DCGL_C$ is 200, and the claimed MDC is 100. Data are then reported as 100±75, 50±75, -25±50, and 75±75. The relative uncertainties are rather high. Are they consistent with the quoted MDC? If the MDC is estimated as 3 to 4 times these uncertainties, we get values of 150 to 300, much higher than the quoted 100. This is an indication that the data quality targets are not being met.

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

Basic statistical quantities that should be calculated for the sample data set are as follows:

- mean
- standard deviation
- median

Example:

Suppose the following 10 measurement values are from a survey unit composed of materials:

9.1, 10.7, 13.6, 3.4, 13.3, 7.9, 4.5, 7.7, 8.3, 10.4

First, the average of the data (8.88) and the sample standard deviation (3.3) should be calculated.

These next 10 measurements are from an appropriate matching reference material:

6.2, 13.8, 15.2, 9.3, 6.7, 4.9, 7.1, 3.6, 8.8, 8.9.

The average of these data is 8.45 and the standard deviation is 3.7.

The average of the data can be compared to the reference material average and the $DCGL_C$ to get a preliminary indication of the survey unit status. The difference in this case is 0.43.

Where there is much added activity, this comparison may readily reveal that the material survey unit should not be released — even before applying statistical tests. For example, if the difference between the survey unit and reference material averages of the data exceeds the $DCGL_C$, the survey unit clearly does not meet the release criterion. On the other hand, if the difference between the largest survey unit measurement (13.6) and the smallest reference material measurement (3.6) is below the $DCGL_C$, the survey unit clearly meets the release criterion.[6]

[6] It can be verified that if the largest difference between survey unit and reference material measurements is below the $DCGL_C$, the conclusion from the WRS test will always be that the survey unit does not exceed the release criterion, provided that an adequate number of measurements were made to meet the DQOs.

The value of the sample standard deviation is especially important. If it is too large (compared to that assumed during the survey design), this may indicate that an insufficient number of samples were collected to achieve the desired power of the statistical test. Again, inadequate power can lead to an increased probability of incorrectly failing a material survey unit.

The median is the middle value of the data set when the number of data points is odd, and is the average of the two middle values when the number of data points is even. Thus 50 percent of the data points are above the median, and 50 percent are below the median. Large differences between the mean and median would be an early indication of a skew in the data. This would also be evident in a histogram of the data. For the example data above, the median is 8.7 (i.e., (8.3 + 9.1)/2). The difference between the median and the mean (i.e., 8.45 - 8.7 = -0.25) is a small fraction of the sample standard deviation (*i.e.*, 3.3). Thus, in this instance, the mean and median would not be considered significantly different.

Examining the minimum, maximum, and range of the data may provide additional useful information. The minimum in this example is 3.4 and the maximum is 13.6, so the range is 13.6 - 3.4 = 10.2. This is only 3.1 standard deviations. Thus, the range is not unusually large. When there are 30 or fewer data points, values of the range much larger than about 4 to 5 standard deviations would be unusual. For larger data sets, the range might be wider.

6.2.2.2 Graphical Data Review

Graphical data review may consist of a posting plot and a histogram or quantile plots. A *posting plot* is simply a map of the survey unit with the data values entered at the measurement locations. This potentially reveals heterogeneities in the data, especially possible patches of elevated contamination. Even in a reference material survey, a posting plot can reveal spatial trends in background data, which might affect the results of the two-sample statistical tests. Posting plots are most useful when the data are obtained by discrete measurements.

If the posting plot reveals systematic spatial trends in the survey unit, the cause of the trends would need to be investigated. In some cases, such trends could be attributable to contamination, but they may also be caused by inhomogeneities in the survey unit background. Other diagnostic tools for examining spatial data trends may be found in EPA Guidance Document QA/G-9.

The role of a posting plot for a CSM would be a time series display of the data, showing any trends between adjacent batches of material being conveyed beneath the detector.

However, the geometric configuration of most survey units composed of a few large irregularly shaped pieces of material is transitory. The arrangement of tools, piles of scrap, and the like will change as pallets of material are moved around and even while pieces are lifted to be surveyed. In these cases, some identifying marks, numbers, or bar-code labels should be used to identify and track where measurements were made, at least until it is determined that the material can be released. Such marking or labeling need not be permanent, but may be made with chalk and removable labels.

A *frequency plot* (or histogram) is a useful tool for examining the general shape of a data distribution. This plot is a bar chart of the number of data points within a certain range of values. A frequency plot reveals any obvious departures from symmetry, such as skewing or bimodality (two peaks), in the data distributions for the survey unit or reference material. The presence of two peaks in the survey unit frequency plot may indicate the existence of isolated areas of contamination. In some cases, it may be possible to determine an appropriate background for the survey unit using this information. The interpretation of the data for this purpose is generally highly dependent on site-specific considerations and should only be pursued after a consultation with the responsible regulatory agency.

The presence of two peaks in the background reference material or survey unit frequency plot may indicate a mixture of background concentration distributions as a result of different soil types, construction materials, etc. The greater variability in the data caused by the presence of such a mixture reduces the power of the statistical tests to detect an adequately decontaminated survey unit. These situations should be avoided whenever possible by carefully matching the background reference materials to the survey units, and choosing material survey units with homogeneous backgrounds.

Skewness or other asymmetry can impact the accuracy of the statistical tests. A data transformation (e.g., taking the logarithms of the data) can sometimes be used to make the distribution more symmetric. The statistical tests would then be performed on the transformed data. When the underlying data distribution is highly skewed, it is often because there are a few high activity concentration areas. Since scanning is used to detect such areas, the difference between using the median and the mean as a measure for the degree to which uniform contamination remains in a survey unit tends to diminish in importance.

When data are obtained from scanning surveys alone using data loggers, a large number of data points is usually logged. In essence, the entire Class 1 material survey unit is measured and, while the survey coverage is less for Class 2 and 3 materials, there will still likely be a large number of data points. In this case, the frequency plot will be close to the population distribution of concentrations in the survey unit. The mean and standard deviation calculated from these logged values should be very close to their population values. In other words, when nearly the entire material survey unit has been measured, statistical sampling is unnecessary.

Similarly, when an *in toto* measurement has been performed, the entire survey unit has been measured. Again, statistical sampling is not necessary.

For conveyorized survey monitors, the data may be interpreted batch by batch as it is scanned, in which case, the data treatment would be most similar to an *in toto* measurement. If the data were logged continuously, the data treatment would be similar to that for a scanning survey using data loggers.

6.2.3 Select the Tests

As mentioned above, when data are obtained from scanning surveys alone using data loggers, a large number of data points is usually logged. In essence, the entire survey unit is measured. The mean and the standard deviation calculated from these logged values should be very close to their population values. In other words, when the entire survey unit has been measured, statistical sampling is unnecessary, as are statistical tests. There is no uncertainty contribution from spatial variability in survey unit concentrations because the entire survey unit has been measured. The average of the logged values may simply be compared to the $DCGL_C$. However, there remains an uncertainty component as a result of the variability in the measurement process. Measurement variability, unlike spatial variability, can often be modeled realistically using a normal distribution. In that case, parametric statistical tests may be more appropriate; however, because removing spatial variability is often the major concern in these surveys, it is suggested that a simple comparison of the mean to the $DCGL_C$ is sufficient. As long as the measurement uncertainty is a small fraction of the $DCGL_C$, the gray region should be very narrow.

When an *in toto* measurement has been performed, the entire survey unit has been measured. Only a single measurement is made, and so the decision is really a detection decision. The statistical test is that used to calculate the MDC. However, assumptions are made about the distribution of activity inherent in the calibration of such detectors, and the validity of those assumptions determines the appropriateness of the measurement.

Again, data from conveyorized survey monitors may be treated as a series of detection decisions on a batch-by-batch basis, or may be analyzed by aggregating the data, much as with a logging scanner.

When conventional surveys are used, they should address the statistical considerations important for clearance surveys, as presented in Section 5.2.3.3. The statistical tests recommended for conventional clearance surveys are the same as those recommended by the MARSSIM for final status surveys of lands and structures.

The most appropriate procedure for summarizing and analyzing the data is chosen based on the preliminary data review. The parameter of interest is the mean concentration in the material survey unit. The nonparametric tests recommended in this report, in their most general form, are tests of the median. If one assumes that the data are from a symmetric distribution — where the median and the mean are effectively equal — these are also tests of the mean. If the assumption of symmetry is violated, nonparametric tests of the median only approximately test the mean. Note that the mean and median only differ greatly when large concentration values skew the distribution. Such areas can be identified while scanning. This is precisely why the survey strategies in this report emphasize using *both* direct measurements and scans. In addition, computer simulations (e.g., Hardin and Gilbert, 1993) have shown that the approximation of the mean by the median implicit in using the nonparametric tests is a fairly good technique as far as decisionmaking is concerned. That is, the correct decision will be made about whether the mean concentration exceeds the DCGL, even when the data come from a skewed distribution. In this regard, the nonparametric tests are found to be correct more often than the commonly used Student's t test. The robust performance of the Sign and WRS tests over a wide range of conditions is the reason that they are recommended in this report.

When a given set of assumptions is true, a parametric test designed for exactly that set of conditions will have the highest power. For example, if the data are from a normal distribution, the Student's t test will have higher power than the nonparametric tests. It should be noted that for large enough sample sizes (e.g., large number of measurements), the Student's t test is not a great deal more powerful than the nonparametric tests. On the other hand, when the assumption of normality is violated, the nonparametric tests can be very much more powerful than the t test. Therefore, any statistical test may be used, provided that the data are consistent with the assumptions underlying their use. When these assumptions are violated, the prudent approach is to use the nonparametric tests, which generally involve fewer assumptions than their parametric equivalents.

The one-sample statistical test (Sign test) described in Section 5.5.2.3 of the MARSSIM should only be used if the radionuclide being measured is not present in background and radionuclide-specific measurements are made. The one-sample test may also be used if the radionuclide is present at such a small fraction of the $DCGL_C$ value as to be considered insignificant. In this case, background concentrations of the radionuclide are included with any contamination that may be present (i.e., the entire amount is attributed to facility operations). Thus, the total concentration of the radionuclide is compared to the release criterion. This option should only be used if one expects that ignoring the background concentration will not affect the outcome of the statistical tests. The advantage of ignoring a small background contribution is that no reference material is needed. This can simplify the survey considerably.

The one-sample Sign test (Section 6.3.1) evaluates whether the median of the data is above or below the $DCGL_C$. If the data distribution is symmetric, the median is equal to the mean. In cases where the data are severely skewed, the mean may be above the $DCGL_C$, while the median is below the $DCGL_C$. In such cases, the survey unit does *not* meet the release criterion regardless of the result of the statistical tests. On the other hand, if the largest measurement is below the $DCGL_C$, the Sign test will *always* show that the survey unit meets the release criterion, provided that enough samples were taken to meet the DQOs.

For clearance surveys, the two-sample statistical test (WRS test, discussed in Section 5.5.2.2 of the MARSSIM) should be used when the radionuclide of concern appears in background or if measurements are used that are not radionuclide-specific. The two-sample WRS test (Section 6.4.1) assumes the reference material and survey unit data distributions are similar except for a possible shift in the medians. When the data are severely skewed, the value for the mean difference may be above the $DCGL_C$, while the median difference is below the $DCGL_C$. In such cases, the survey unit does *not* meet the release criterion regardless of the result of the statistical test. On the other hand, if the difference between the largest survey unit measurement and the smallest reference material measurement is less than the $DCGL_C$, the WRS test will always show that the survey unit meets the release criterion, provided that enough samples were taken to meet the DQOs.

6.2.4 Verify the Assumptions of the Tests

An evaluation to determine that the data are consistent with the underlying assumptions made for the statistical procedures helps to validate the use of a test. One may also determine that certain departures from these assumptions are acceptable when given the actual data and other information about the study. The nonparametric tests described in this chapter assume that the data from the reference material or survey unit consist of independent samples from each distribution.

Asymmetry in the data can be diagnosed with a stem and leaf display, a histogram, or a Quantile plot. As discussed in the previous section, data transformations can sometimes be used to minimize the effects of asymmetry.

One of the primary advantages of the nonparametric tests used in this report is that they involve fewer assumptions about the data than their parametric counterparts. If parametric tests are used, (e.g., Student's t test), any additional assumptions made in using them should be verified (e.g., testing for normality). These issues are discussed in detail in EPA QA/G-9 (EPA 1998b).

One of the more important assumptions made in the survey design is that the sample sizes determined for the tests are sufficient to achieve the data quality objectives set for the Type I (α) and Type II (β) error rates. Verification of the power of the tests ($1-\beta$) to detect adequate probability for passing material survey units that meet the criteria for clearance may be of particular interest. Methods for assessing the power are discussed in Appendix I.9 to the MARSSIM. If the hypothesis that the material survey unit radionuclide concentration exceeds the clearance criterion is accepted, there should be reasonable assurance that the test is equally effective in determining that a survey unit has radionuclide concentrations less than the $DCGL_C$. Otherwise, unnecessary survey unit failures may result. For this reason, it is better to plan the surveys cautiously, even to the following extents:

- overestimating the potential data variability
- taking too many samples
- overestimating the minimum detectable concentrations (MDCs)

If one is unable to show that the DQOs are met with reasonable assurance, a resurvey may be needed.

When data are obtained from scanning surveys alone using data loggers, the mean of the logged values may simply be compared to the $DCGL_C$. Because such a large number of data points are obtained, essentially the entire population of concentrations on the material has been measured. Thus, no formal statistical test is necessary. It is the assumption of full measurement coverage that is the central issue in this case. It is also assumed that the measurement uncertainty is small compared to the $DCGL_C$. The validity of these assumptions should be carefully examined, and the results documented in the SOPs and QAPP.

When an *in toto* measurement has been performed, the entire survey unit has been measured. Again, statistical sampling is not necessary. However, assumptions are made about the distribution of activity inherent in the calibration of such detectors, and the validity of those assumptions determines the appropriateness of the measurement.

Examples of assumptions and possible methods for their assessment are summarized in Table 6.1.

3263 **Table 6.1: Issues and assumptions underlying survey results**

3264 Survey Type	Issue
3265 Conventional survey	Appropriateness of the statistical test
3266 Scanning only	Data logging and calibration geometry
3267 Automated scanning	Data logging and calibration geometry
3268 *In toto* survey	Calibration model and source geometry

3269 6.2.5 Draw Conclusions from the Data

3270 The types of conventional measurements that can be made on a survey unit are (1) direct measurements
3271 at discrete locations, (2) samples collected at discrete locations, and (3) scans. The statistical tests are
3272 only applied to measurements made at discrete locations. Specific details for conducting the statistical
3273 tests are given in Sections 6.3 and 6.4. When the data clearly show that a survey unit meets or exceeds
3274 the release criterion, the result is often obvious without performing the formal statistical analysis. This is
3275 the expected outcome for Class 2 and Class 3 material survey units. Table 6.2 summarizes examples of
3276 circumstances leading to specific conclusions based on a simple examination of the data.

3277 Scans may uncover potential areas that exceed the $DCGL_C$. Unless a scanning-only survey with a data
3278 logger or an *in toto* measurement is made, any such area will require further investigation. Note that
3279 there may be, as discussed in Section 3.3, separate criteria established for small areas of elevated activity.
3280 The investigation may involve taking further measurements to determine whether the area and level of
3281 contamination are such that the resulting average over the material survey unit meets the release
3282 criterion. The investigation should also provide adequate assurance, using the DQO Process, that there
3283 are no other undiscovered areas of elevated radioactivity in the survey unit that might otherwise result in
3284 a dose or risk exceeding the established criterion. In some cases, this may lead to reclassifying all or part
3285 of a survey unit.

3286 Section 6.3 describes the Sign test used to evaluate the material survey units, and Section 6.4 describes
3287 the WRS test used to evaluate the material survey units where the radionuclide being measured is present
3288 in background. Section 6.5 discusses the evaluation of the results of the statistical tests and the decision
3289 regarding compliance with the release criterion.

Table 6.2: Summary of statistical tests

Radionuclide not in background and radionuclide-specific measurements made:	
Survey Result	**Conclusion**
All measurements less than $DCGL_C$	Survey unit meets release criterion
Average greater than $DCGL_C$	Survey unit does not meet release criterion
Any measurement greater than $DCGL_C$ and the average less than $DCGL_C$	Conduct Sign test and elevated measurement comparison
Radionuclide in background or radionuclide non-specific (gross) measurements made:	
Survey Result	**Conclusion**
Difference between largest survey unit measurement and smallest reference material measurement is less than $DCGL_C$	Survey unit meets release criterion
Difference of survey unit average and reference material average is greater than $DCGL_C$	Survey unit does not meet release criterion
Difference between any survey unit measurement and any reference material measurement greater than $DCGL_C$ and the difference of survey unit average and reference material average is less than $DCGL_C$	Conduct WRS test and elevated measurement comparison

6.3 Sign Test

The statistical test discussed in this section is used to compare each material survey unit directly with the applicable release criterion. A reference material is not included because the measurement technique is radionuclide-specific and the radionuclide of concern is not present in background. In this case, the contamination levels are compared directly with the $DCGL_C$. *The method in this section should only be used if the radionuclide being measured is not present in background or is present at such a small fraction of the $DCGL_C$ value as to be considered insignificant.* In addition, one-sample tests are applicable only if radionuclide-specific measurements are made to determine the concentrations. Otherwise, the method in Section 6.4 is recommended.

Reference materials and reference samples are not needed when there is sufficient information to indicate that there is essentially no background concentration for the radionuclide being considered. With only a single set of survey unit samples, the statistical test used here is called a one-sample test. Further information on the Sign Test can be found in Section 8.3 of the MARSSIM and Chapter 5 of NUREG 1505, Rev.1.

6.3.1 Applying the Sign Test

The Sign test is applied by counting the number of measurements in the survey unit that are less than the $DCGL_C$. The result is the test statistic S+. Discard any measurement that is exactly equal to the $DCGL_C$ and reduce the sample size, N, by the number of such measurements. The value of S+ is compared to the critical values in MARSSIM Table I.3. If S+ is greater than the critical value, k, in that table, the null hypothesis is rejected.

6.3.2 Sign Test Example: Class 1 Copper Pipes

This example illustrates the clearance survey design for copper pipe sections using a gas proportional counter to measure ^{239}Pu. Since the alpha background on the copper material is essentially zero, it was decided to use the Sign test to determine whether the material meets the clearance criterion. The sample size was determined using the DQO Process, with inputs such as the $DCGL_C$, the expected standard deviation of the radionuclide concentrations in the pipe sections, and the acceptable probability of making Type I and Type II decision errors. The inputs were as follows:

- The gross activity $DCGL_C$ was 100 dpm/100 cm^2. When converted to cpm, the gross activity $DCGL_C$ was 10 cpm.
- The LBGR was set at the expected added activity level on the copper pipe sections (i.e., 5 net cpm — the same as the gross mean for an alpha background of zero).
- The standard deviation on the material survey unit was expected to be about 2 cpm.
- The relative shift was calculated as $(10 - 5)/2 = 2.5$
- The Type I and II errors were set at 0.05.

Table 5.5 in the MARSSIM (1997) indicates that the number of measurements estimated for the Sign Test, N, is 15 ($\alpha = 0.05$, $\beta = 0.05$, and $\Delta/\sigma = 2.5$). Therefore, 15 surface activity measurements were randomly collected from the inside surfaces of the copper pipe sections. Clearance survey results are shown on Table 12.3.

Table 6.3: Example sign test results

Data (cpm)	Surface activity (dpm/100 cm²)	< $DCGL_C$?
4	40	Yes
3	30	Yes
3	30	Yes
1	10	Yes
1	10	Yes
4	40	Yes
6	60	Yes
3	30	Yes
9	90	Yes
6	60	Yes
14	140	No
1	10	Yes
4	40	Yes
3	30	Yes
2	20	Yes
Number of measurements less than $DCGL_C$ =14 (= S+)		

The surface activity values on Table 6.3 were determined by dividing the measured cpm by the efficiency (0.10). No probe area correction was necessary. The average count rate on this material survey unit was 4.3 (we had estimated a residual cpm of 5 cpm). The median of the data was 3 cpm. The mean surface activity level was 43 dpm/100 cm². The standard deviation was 3.5, which was higher than the value of 2 that was estimated for the survey design. Thus, the power of the test will be lower than planned for. With the actual value of the relative shift (10 - 5)/3.5 = 1.4, N = 20 measurements would be required. With the 15 measurements, the actual Type II error rate is a little over 0.10. (The closest table entry is for $\alpha = 0.05$, $\beta = 0.10$, and $\Delta/\sigma = 1.4$ with N=16.)

One measurement exceeded the $DCGL_C$ value of 100 dpm/100 cm². The portion of the material survey unit containing that location merits further investigation.

The value of S+, 14, was compared to the appropriate critical value in Table I.3 of the MARSSIM. In this case, for $N = 15$ and $\alpha = 0.05$, the critical value is 11. Since S+ exceeds this value, the null hypothesis that the survey unit exceeds the release criterion is rejected. In this case, the slight loss of power attributable to underestimating the standard deviation did not affect the result. Pending the outcome of the investigation on the one elevated measurement, this material survey unit satisfies the release criteria established for clearance.

6.4 WRS Test

The statistical tests discussed in this section will be used to compare each material survey unit with an appropriately chosen, site-specific reference material. Each reference material should be selected on the basis of its similarity to the survey unit, as discussed in Section 5.2.3.3. Further information on the WRS Test can be found in Section 8.4 of the MARSSIM and Chapter 6 of NUREG 1505, Rev.1.

6.4.1 Applying the WRS Test

The WRS test is applied as outlined in the following six steps and further illustrated by the example in Section 6.4.2.

(1) Obtain the adjusted reference material measurements, Z_i, by adding the $DCGL_C$ to each reference material measurement, X_i. $Z_i = X_i + DCGL_C$

(2) The m adjusted reference sample measurements, Z_i, from the reference material and the n sample measurements, Y_i, from the survey unit are pooled and ranked in order of increasing size from 1 to N, where $N = m+n$.

(3) If several measurements are tied (i.e., have the same value), they are all assigned the average rank of that group of tied measurements.

(4) If there are t "less than" values, they are all given the average of the ranks from 1 to t. Therefore, they are all assigned the rank $t(t+1)/(2t) = (t+1)/2$, which is the average of the first t integers. If there is more than one detection limit, all observations below the largest detection limit should be treated as "less than" values.[7]

(5) Sum the ranks of the adjusted measurements from the reference material, W_r. Note that since the sum of the first N integers is $N(N+1)/2$, one can equivalently sum the ranks of the measurements from the survey unit, W_s, and compute $W_r = N(N+1)/2 - W_s$.

(6) Compare W_r with the critical value given in Table I.4 of the MARSSIM for the appropriate values of n, m, and α. If W_r is greater than the tabulated value, reject the hypothesis that the survey unit exceeds the release criterion.

[7] If more than 40 percent of the data from either the reference material or survey unit are "less than," the WRS test *cannot* be used. Such a large proportion of non-detects suggest that the DQO Process must be revisited for this survey to determine whether the survey unit was properly classified or the appropriate measurement method was used. As stated previously, the use of "less than" values in data reporting is not recommended. Wherever possible, the actual result of a measurement, together with its uncertainty, should be reported.

6.4.2 WRS Test Example: Class 2 Metal Ductwork

This example illustrates the use of the WRS test for releasing Class 2 metal ductwork. Assume that a gas proportional detector was used to make gross (non-radionuclide-specific) surface activity measurements.

The DQOs for this survey unit include $\alpha = 0.05$ and $\beta = 0.05$, and the $DCGL_C$ converted to units of gross cpm is 2,300 cpm. In this case, the two-sample nonparametric WRS statistical test was used because the estimated background level (2,100 cpm) was large compared to the DCGL. The estimated standard deviation of the measurements, σ, was 375 cpm. The estimated added activity level was 800 cpm; the LBGR will be set at this value. The relative shift can be calculated as $\Delta/\sigma = (DCGL_C - LBGR)/\sigma$, which equals 4.

The sample size needed for the WRS test can be found in Table 5.3 of the MARSSIM for these DQOs. The result is nine measurements in each survey unit and nine in each reference material ($\alpha = 0.05$, $\beta = 0.05$, and $\Delta/\sigma = 4$). The ductwork was laid flat onto a prepared grid, and the nine measurements needed in the survey unit were made using a random-start triangular grid pattern. For the reference materials, the measurement locations were chosen randomly on a suitable batch of material. Table 6.4 lists the gross count rate data obtained.

In column B, the code "R" denotes a reference material measurement, and "S" denotes a survey unit measurement. Column C contains the adjusted data, which were obtained by adding the $DCGL_C$ to the reference material measurements (see Section 6.4.1, Step 1). The ranks of the adjusted data appear in Column D. They range from 1 to 18, since there is a total of 9+9 measurements (see Section 6.4.1, Step 2). Note that the sum of *all* of the ranks is still $18(18+1)/2 = 171$. Checking this value with the formula in Step 5 of Section 6.4.1 is recommended to guard against errors in the rankings.

Column E contains only the ranks belonging to the reference material measurements. The total is 126. This is compared with the entry for the critical value of 104 in Table I.4 of the MARSSIM for $\alpha = 0.05$, with $n = 9$ and $m = 9$. Since the sum of the reference material ranks is greater than the critical value, the null hypothesis (i.e., that the average survey unit concentration exceeds the $DCGL_C$) is rejected, and the ductwork is released.

Note that this conclusion could be reached much more quickly by noting that the largest survey unit measurement, 3,423, differs from the smallest reference material measurement, 1,427, by much less than the $DCGL_C$ of 2,300 cpm.

Table 6.4: WRS test for Class 2 ductwork

	A	B	C	D	E
	Data (cpm)	Area	Adjusted Data	Ranks	Reference Material Ranks
1					
2	2180	R	4480	15	15
3	2398	R	4698	16	16
4	2779	R	5079	18	18
5	1427	R	3727	10	10
6	2738	R	5038	17	17
7	2024	R	4324	13	13
8	1561	R	3861	11	11
9	1991	R	4291	12	12
10	2073	R	4373	14	14
11	2039	S	2039	3	0
12	3061	S	3061	8	0
13	3243	S	3243	9	0
14	2456	S	2456	7	0
15	2115	S	2115	4	0
16	1874	S	1874	2	0
17	1703	S	1703	1	0
18	2388	S	2388	6	0
19	2159	S	2159	5	0
		Sum =		171	**126**

6.5 Evaluating the Results: The Decision

Once the data and the results of the tests are obtained, the specific steps required to achieve material clearance depends on the procedures approved by the regulator and specific considerations to ensure that the contamination is as low as is reasonably achievable (ALARA). The following considerations are suggested for the interpretation of the test results with respect to the release limit established for clearance. Note that the tests need not be performed in any particular order.

6.5.1 Interpreting Data for Each Survey Type

Clearance survey designs using conventional instrumentation are as follows:

- Scanning-Only
  - Calculate the average and compare it to DCGL.
  - Investigate measurements exceeding the DCGL.
  - Anything above the DCGL will trigger a reevaluation of the classification if Class 2.
  - Any contamination will trigger a reevaluation of the classification if Class 3.
- Statistically Based Sampling
  - Techniques are similar to those used in MARSSIM.
  - Survey unit must pass statistical tests.
  - Sampling involves investigations of individual measurements/scans (as for scanning-only).
- Automated Scanning Surveys (conveyorized survey monitors)
  - Scan sensitivity and ongoing QA data must be documented.
  - The statistical tests are essentially those used to calculate the MDC as discussed in Section 3.
  - "Batch-by-batch" segmented gate systems segregate any material above the clearance DCGL.
  - Data from continuous scanning of materials can be interpreted in the same way as for scanning-only surveys.
- *In Toto* Surveys
  - Emphasis is on adequate documentation of calibration.
  - A single measurement is compared to the DCGL.
  - A realistic estimate of the MDC is essential.

6.5.2 If the Survey Unit Fails

When a material survey unit fails to demonstrate compliance with the clearance criterion, the first step is to review and confirm the data that led to the decision. Once this is done, the DQO Process can be used to identify and evaluate potential solutions to the problem. The level of contamination on the material should be determined to help define the problem. For example, if only one or two pieces of material in a Class 1 material survey unit fail, the simplest solution might be to segregate those pieces and either remove the added activity from them or dispose of them as waste. If such a situation were encountered in evaluating Class 2 or Class 3 material survey units, it would call into question the entire classification procedure, and would require that the material at hand be reclassified and treated as Class 1.

As a general rule, it may be useful to anticipate possible modes of failure. These can be formulated as the problem to be solved using the DQO Process. Once the problem has been stated, the decision concerning the failing survey unit can be developed into a decision rule (for example, whether to attempt to remove the radioactivity or simply segregate certain types of units as waste). Next, determine the additional data, if any, needed to document that a survey unit with elevated pieces removed or areas of added activity removed demonstrates compliance with the clearance criterion. Alternatives to resolving the decision rule should be developed for each type of material survey unit that may fail the surveys. These alternatives can be evaluated against the DQOs, and a clearance survey strategy that meets the objectives of the project can be selected.

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Glossary

calibration: comparison of a measurement standard, instrument, or item with a standard or instrument of higher accuracy to detect and quantify inaccuracies, and to report or eliminate those inaccuracies by making adjustments.

Class 1 materials: solid materials that have (or had) a potential for contamination (based on process knowledge) or known contamination (based on previous surveys) above the release criteria ($DCGL_C$).

Class 2 materials: solid materials that have (or had) a potential for or known contamination, but are not expected to be above the release criteria ($DCGL_C$).

Class 3 materials: solid materials that are not expected to contain any contamination, or are expected to contain contamination less than a small fraction of the release criteria ($DCGL_C$) based on process knowledge or previous surveys.

clearance: release of solid materials that do not require further regulatory control.

critical level: the net count, or final instrument measurement result after appropriate calibration and/or correction factors have been applied, at or above which a decision is made that activity is present in a sample. When the observed net count is less than the critical level, the surveyor correctly concludes that no net activity is present in the sample.

detection limit: the smallest number of net counts, or final instrument measurement result after appropriate calibration and/or correction factors have been applied, that will be detected with a probability (β) of non-detection, while accepting a probability (α) of incorrectly deciding that activity is present in a sample.

impacted: materials that have some contamination potential, and therefore require a clearance survey in order to be released.

inaccessible areas: locations on the surface of a solid material, which are not accessible for direct survey evaluation without cutting or dismantling the material. These inaccessible areas include the interior surfaces of pipes and scrap equipment such as pumps, motors, and other equipment.

instrument efficiency, ε_i: similar to the intrinsic efficiency of a detector, the instrument efficiency is the ratio between the instrument net count rate and the surface emission rate of a source under specified geometric conditions. For a given instrument, the instrument efficiency depends on the radiation energy emitted by the source and the geometry between the detector and the source. Instrument efficiency is a 2π value and shall only be used in surface activity determinations when multiplied by a surface efficiency to yield a 4π value of total efficiency.

in toto: a clearance survey technique that measures the entire material (or materials) at once.

measurement quality objective (MQO): a statement of performance objective or requirement for a particular method performance characteristic. Like DQOs, MQOs can be quantitative or qualitative statements. An example of a quantitative MQO would be a statement of a required method uncertainty at a specified radionuclide concentration, such as the action level [i.e., "a method uncertainty of 3.7 Bq/kg (0.10 pCi/g) or less is required at the action level of 37 Bq/kg (1.0pCi/g)"]. An example of a qualitative MQO would be a statement of the required specificity of the analytical protocol, such as the ability to anaquantify the amount of ^{226}Ra present given high levels of ^{235}U in the samples.

minimum detectable concentration (MDC): the smallest activity concentration that can be detected with specific confidence for a given instrument and specific measurement procedure. The MDC is usually specified as the smallest activity concentration that can be detected with 95 percent confidence (i.e., 95 percent of the time a given instrument and measurement procedure will detect activity at the MDC).

minimum detectable count rates (MDCR): the detector signal level, or count rate for most equipment, that is likely to be flagged by a surveyor as being "greater than background."

non-impacted materials: materials that have no reasonable possibility of having contamination. These materials may be used for background reference measurements.

process knowledge: the use of operational information to assess the contamination potential of solid materials considering the location and use of the materials during operations.

real property: land and building structures and equipment or fixtures (e.g., ductwork, plumbing, built-in cabinets) that are installed in a building in a more or less permanent manner.

scanning: a survey technique performed by moving a detector over a surface at a specified speed and distance above the surface to detect radiation, usually via the audible output of the instrument.

secular equilibrium: the condition that exists between the parent and other members of a decay series when the parent radionuclide decays much more slowly than any of the other members of the series. During secular equilibrium, the activity of the parent and each daughter radionuclide is equal.

solid materials (also non-real property): as opposed to lands and structures, materials such as tools/equipment, office items, consumable items, and debris that are offered for clearance.

spectrometer: a device that measures energy (specifically, radiation energy).

surface efficiency, ε_s: ratio between the number of particles of a given radiation type emerging from the surface per unit time (surface emission rate) and the number of particles of the same type released within the source per unit time. The surface efficiency is nominally 0.5, but may be increased by backscattered radiation and reduced by self-absorption.

surrogate: a radionuclide that is measured for the purpose of inferring the radionuclide concentration of one or more radionuclides that are not measured.

survey unit, material (lots/batches): a specified amount of solid material for which a separate decision will be made as to whether the unit meets the release criteria for clearance.

total efficiency, ε_T: similar to the absolute efficiency of a detector, the total efficiency is the ratio of the detector response (e.g., in counts) and the number of particles emitted by the source. The total efficiency is contingent not only on detector properties, but also on the details of the counting geometry, surface characteristics, and other environmental conditions. The total efficiency (a 4π value) is the product of the instrument and surface efficiencies.

3693 **Appendix A: Fundamentals of Radiation and Radiation Detection**

A.1 Introduction

This appendix introduces some basic properties of radiation, which are relevant to the measurement of residual radioactivity in and on solid materials. To provide a generic discussion, this appendix avoids mentioning or referring to a specific amount of radioactivity. Instead, this appendix focuses on some of the fundamental principles of radiation detection and measurement. It must be understood that the assay of residual radioactivity in and on solid materials is not simply a matter of radiation detection; rather, it involves (to some extent), identifying the presence of specific radionuclides, and quantifying their specific activities, while satisfying quality assessment objectives. This can be accomplished in a variety of ways, depending on the nature and type of material, the radionuclides involved, and the distribution of the radioactivity. It is unlikely that any single detector or method can cover all possible scenarios.

Radionuclides are identified by measuring their nuclear properties, which are usually expressed by the energy of the radiation emitted as a result of nuclear transformations. Measurement of the radiation energy, along with a nuclear decay table, provides a method of identifying radionuclides. In situations where the measurement of the energy is difficult or impossible, the measurement of the nuclear mass (also known as mass spectroscopy) can also be used. This appendix focuses on techniques that use energy spectroscopy.

A.2 Measurement of Radioactivity: Decay Counting

In the majority of applications, radioactivity is usually measured using an indirect method, which requires a standard of known activity from which a calibration is obtained. Basically, the radioactivity (decays per unit time) is measured by counting the number of events in a detector for a specified interval of time (this interval is referred to as the "count time"). These events, which usually take the form of electronic pulses, result from the interaction of the radiation with the active (sensitive) components of the detector. The number of events is proportional to the radioactivity of the source. Once the detector is calibrated, using a standard source under reproducible conditions, the radioactivity can be quantified. A more complete discussion of radioactivity measurements, both direct and indirect, may be found in NCRP Report 58.

For the assay of residual radioactivity in and on solid materials, a comprehensive set of reference materials does not exist to cover the range of conditions needed to develop an instrument calibration. The range of conditions refers to the geometry of the measurement system and source, as well as the disposition and quantity of any material absorbing or scattering radiation. The term calibration, in this context, presumes that the reference material has traceability to a national certifying organization, such as the U.S. National Institute of Standards and Technology (NIST) or the International Atomic Energy Agency (IAEA).

The challenge for instrument developers is to extrapolate from the limited supply of available reference materials enough information and data to produce meaningful results. For example, the calibration of a radiation detector or detector system for a large-area (or volume) source, in some cases, can be obtained through a series of measurements using a certified point source (Becker *et al.*, 1999).

The concept of calibration is evolving to encompass techniques that do not use actual sources, but rather simulate a calibration source. The simulation method relies on knowledge of and experience with radiation transport coupled with fast and powerful computers. The radiation transport code, called Monte Carlo N-Particle (MCNP), employs Monte Carlo methods to simulate radiation transport for neutrons, photons, and electrons for a wide variety of energies, materials, and geometries (Briesmeister, 1993). The MCNP code provides a resource for investigators to test the response of their instruments to a variety of measurement conditions, which ultimately can lead to a calibration. It must be emphasized, however, that the quality or accuracy of a calibration developed using a simulation is predicated on the quality or accuracy of the transport code and the degree to which the simulation reflects the actual conditions of the measurement.

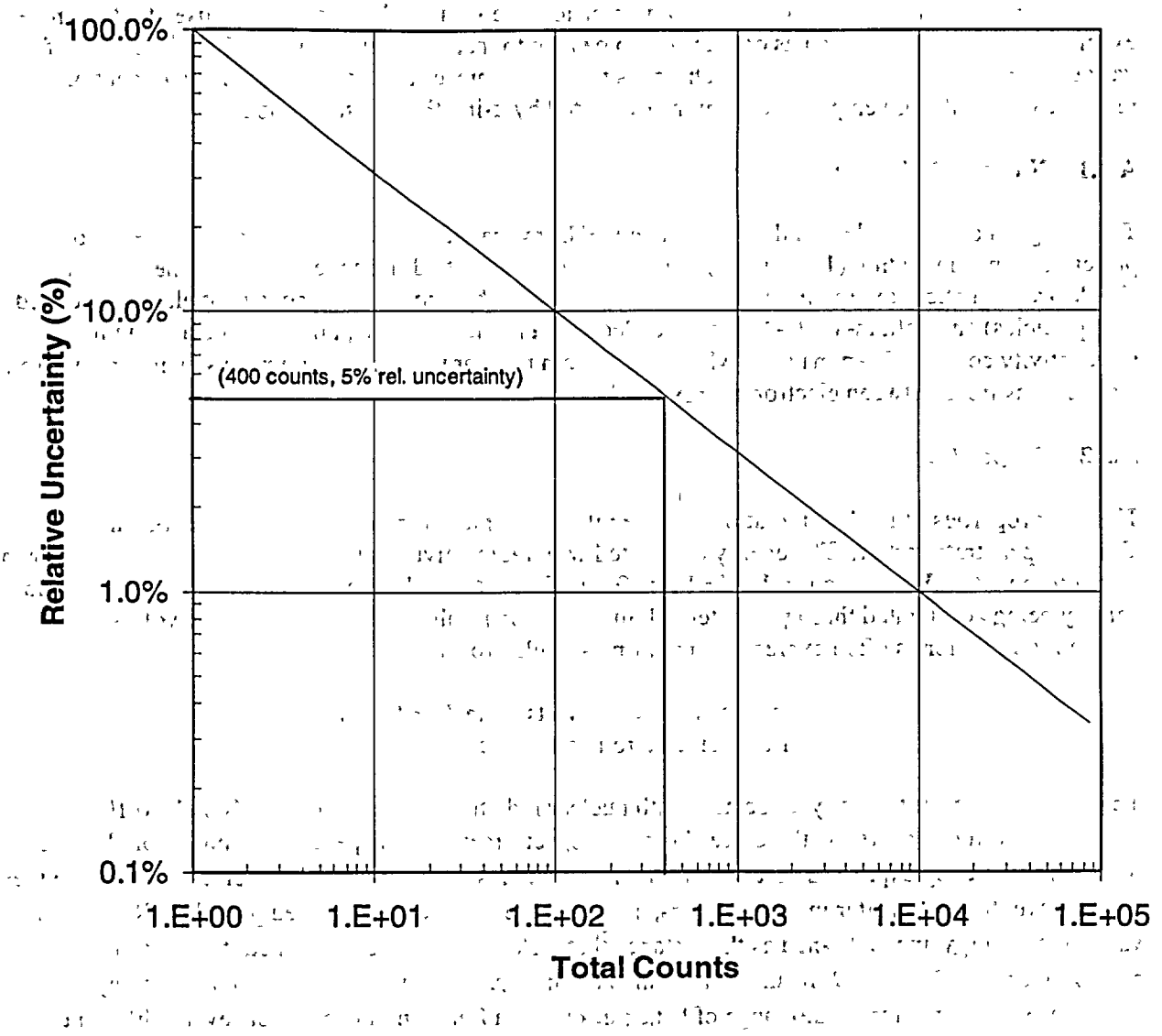
A.3 Statistical Models of Nuclear Decay

Radioactive decay is a stochastic or random process. Any measurement of radioactivity has an inherent variation attributable to the random fluctuations associated with the decay process. Three statistical models are used to describe and quantify these random fluctuations under different circumstances:

- Binomial distribution is the most general, but computationally cumbersome, distribution of the three models. It is applied when counting short-lived radionuclides with high efficiency.
- Poisson distribution is a special case of the more general binomial distribution. It is applied when the counting time is short in comparison to the half-life. The Poisson distribution is a discrete distribution.
- Gaussian distribution is the distribution applied when the number of decays during the count time is fairly substantial (>20). The Gaussian distribution is a continuous distribution.

These statistical models can be used to help understand, interpret, and make predictions concerning the outcome of radiation measurements. For example, if the outcome of a single measurement yields n counts, then by applying what is known about the distributions, it is possible to predict the results of subsequent measurements. This reproducibility is an indication of the precision of the measurement. A system that can be described by a Poisson (or Gaussian) distribution has a variance, equal to the mean, which is a measure of the dispersion of a distribution. Therefore, a measurement that yields a result of n counts has a variance of n and a standard deviation of $\sqrt{n}$. Hence, 68 percent of subsequent measurements under the same conditions will yield results that fall within the range $n+\sqrt{n}$ to $n-\sqrt{n}$. Another way of expressing the variability in the measurement in terms of the mean and the standard deviation is $n \pm k\sqrt{n}$ (counts).

The parameter k is known as a coverage factor and the product $k\sqrt{n}$ defines a confidence interval. If $k = 1$, then 68 percent of the measurements will fall within an interval that is two standard deviations wide, centered about the mean. If $k = 2$, then 95.5 percent of the results will fall within an interval that is four standard deviations wide, centered about the mean. The typical or recommended coverage factor is $k = 1$ (ISO 1995), and the relative uncertainty is the ratio of the standard deviation to the mean. Figure A-1 shows the relative uncertainty as a function of the number of counts. The more counts, the smaller the relative uncertainty, and the greater the precision. For more information on the application of the statistical models to the analysis of decay counting, see ICRU Report 52 and NAS-NSS Report 3109.



3770 **Figure A-1: Relative uncertainty in counting as a function of the total counts for a Poisson process**

If there are requirements specifying a certain precision, the statistical models can be used to determine experimental parameters, such as count time, to be able to meet the requirements. The suitability of various instruments or measurement techniques to detect a prescribed or predetermined amount of radioactivity, with a given precision, can be evaluated by using the statistical models.

A.3.1 Nuclear Radiation

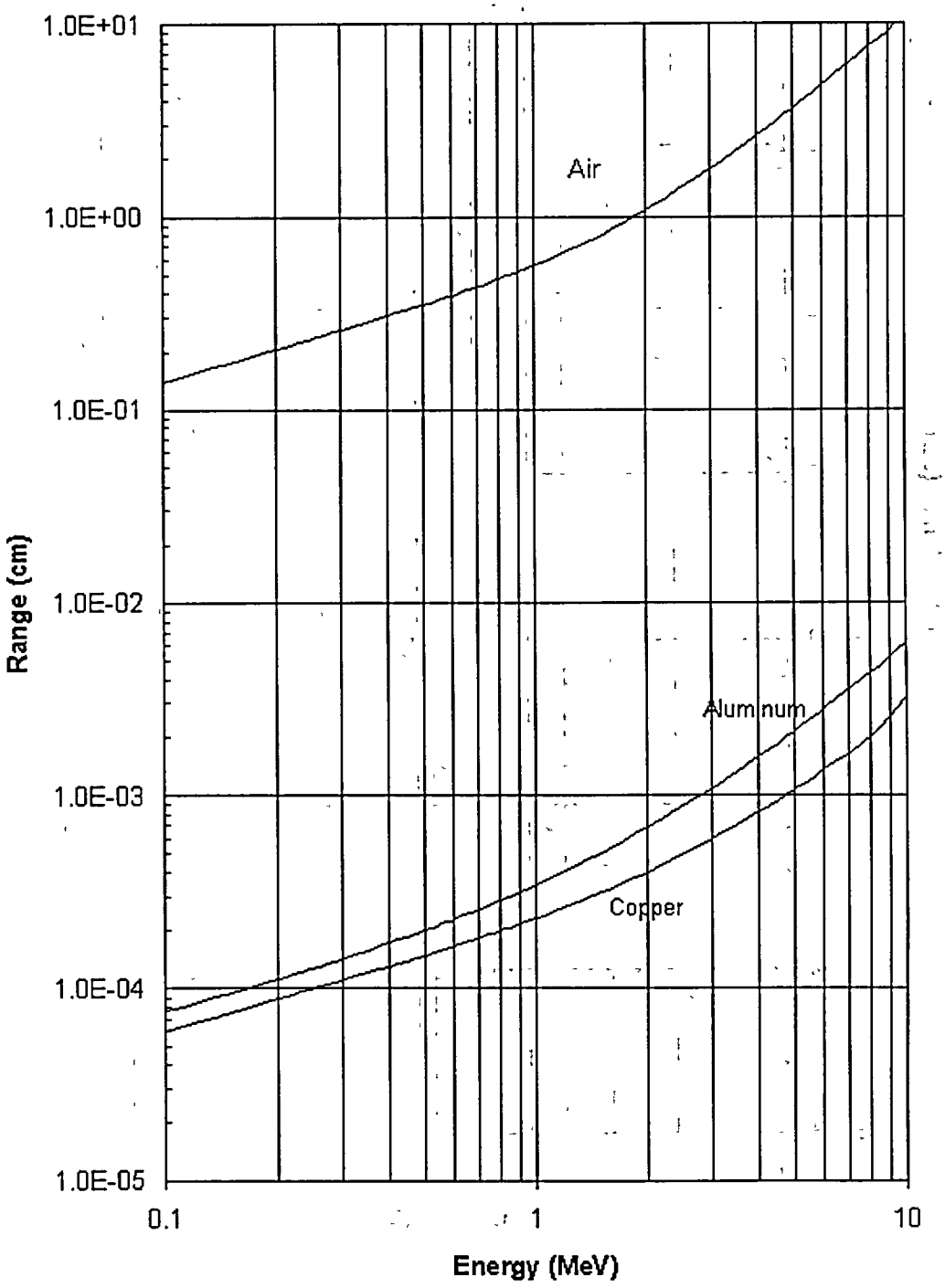
The energy and matter released during radioactive decay, called "nuclear radiation," assumes two principle forms, including (1) charged particles, which are emitted from the nucleus of the atom, and (2) electromagnetic radiation in the form of photons. The charged particles consist of electrons (called beta particles) and helium-4 (He-4) nuclei (called alpha particles). The photons associated with radioactivity consist of gamma rays, which result from nuclear transitions, and x-rays, which result from atomic transitions between electron energy levels.

A.3.2 Properties

The two properties of nuclear radiation that are relevant to radiation detection are its energy and its ability to penetrate matter. The energy associated with radioactivity is usually expressed in units known as electron volts (eV), defined as 1 eV $=1.6 \times 10^{-19}$ joules. This is the kinetic energy an electron would gain by being accelerated through a potential difference of 1 volt. Because the electron volt is a very small unit, radiation is often expressed in multiples of electron volts.

1 thousand electron volts (1 keV) = 10^3 eV
1 million electron volts (1 MeV) = 10^6 eV

The energies that are typically associated with nuclear radiation range from about 10 keV to 10 MeV, and are generally measured with devices known as spectrometers. The penetration power of charged particles is typically expressed in terms of its range, which is not well-defined for electrons because they do not travel through matter in straight lines, as is the case with heavier charged particles. Range usually varies with energy and is defined as the distance that a charged particle will penetrate material before it ceases to ionize. Figure A-2 illustrates the range of alpha particles in air as a function of energy, while Figure A-3 shows the maximum range of beta particles as a function of energy for several different materials. As Figure A-2 illustrates, a 2-Mev alpha particle no longer produces ionizations in air after traveling only a centimeter distance. Note that the penetrating power of beta particles in metals is also limited; a 1-MeV beta particle in copper has a maximum range of less than a millimeter. An immediate consequence of these facts regarding the range of charged particles (alphas and betas) in matter is that alpha radiation can only be used to assay surficial contamination, while beta radiation can, to a limited extent, be utilized for volumetric contamination. Also, these two particles produce very different specific ionization. (The specific ionization is the number of ion pairs produced per unit path length by an ionizing particle; some detectors exploit this value to discriminate between alpha and beta particles.) A typical alpha particle traveling through air generates 10,000 to 70,000 ion pairs per centimeter, while a typical beta particle may produce only 60 to 7,000 ion pairs.



3807 **Figure A-2: Range of an alpha particle as a function of energy in several different materials**
3808 **(Data from ICRU Report 49)**

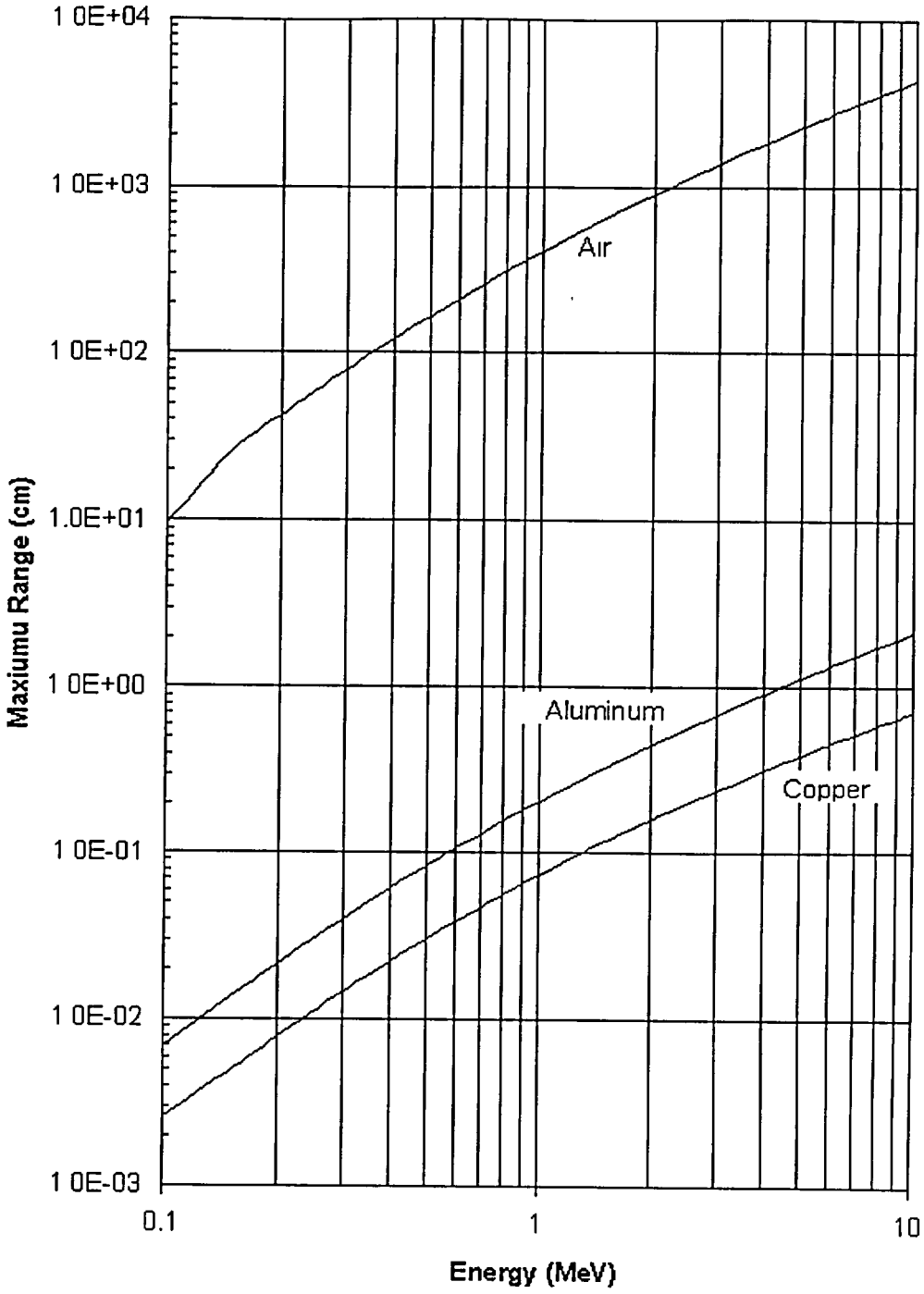


Figure A-3: Range of beta particle as a function of energy in several different materials (Data from ICRU Report 37)

The transport of gamma and x-rays through matter is quite different than for charged particles. The penetration power of gamma and x-rays in matter is typically expressed in terms of its half-value thickness (HVT), defined as the thickness of a material necessary to reduce the intensity of an x-ray or gamma ray beam to one-half of its original value. Figure A-4 is a plot of HVT as a function of energy for several materials. The HVT in this application can be thought of as an indication of the depth-of-view for volumetric contamination. Another significant feature of gamma radiation is that, unlike charged particles, photons can pass through matter without losing energy. The mean-free-path (MFP) is the average distance a photon can travel before having an interaction. Figure A-5 is a plot of the MFP as a function of photon energy for several materials. Note that a 1-MeV photon in copper can travel, on average, almost 2 centimeters without having an interaction. Germanium (Ge) is included in Figure A-5 because it is a common detector material. Here again, a 1-MeV photon can travel, on average, 3 centimeters without having an interaction.

Another form of radiation that comes from the nucleus exists a uncharged particles, called neutrons, which behave quite differently from gamma rays and charged particles. As previously mentioned, radiation in the form of gamma rays and charged particles comes from nuclear decay. Neutrons, on the other hand, are generated by different processes, including the spontaneous fission of heavy elements such as uranium and plutonium. For most isotopes, the neutron emission rate is low compared to other forms of radiation.

Table A-1 shows the spontaneous fission for a selected group of heavy elements, along with the corresponding alpha yield. (For the radionuclides listed in Table A-1, alpha particles are the primary source of radiation.) While the production of neutrons from the spontaneous fission yield of heavy elements is considerably less than the number of alpha particles generated from nuclear decay, neutrons do have a very significant detection advantage over alpha particles in that they can penetrate matter quite easily. Unlike charged particles, which have a range on the order of centimeters to meters depending on the type of radiation and the medium of interest (e.g., air, tissue), neutrons, like gamma rays, can have an indefinite range in matter. This makes neutrons attractive for the assay of volumetric contamination. Measurements of neutron fluence rates are widely used to assay transuranic waste. Despite this advantage, the use of neutrons for the assay of residual radioactivity is largely precluded because the yield is rather small and limited to a handful of heavy elements.

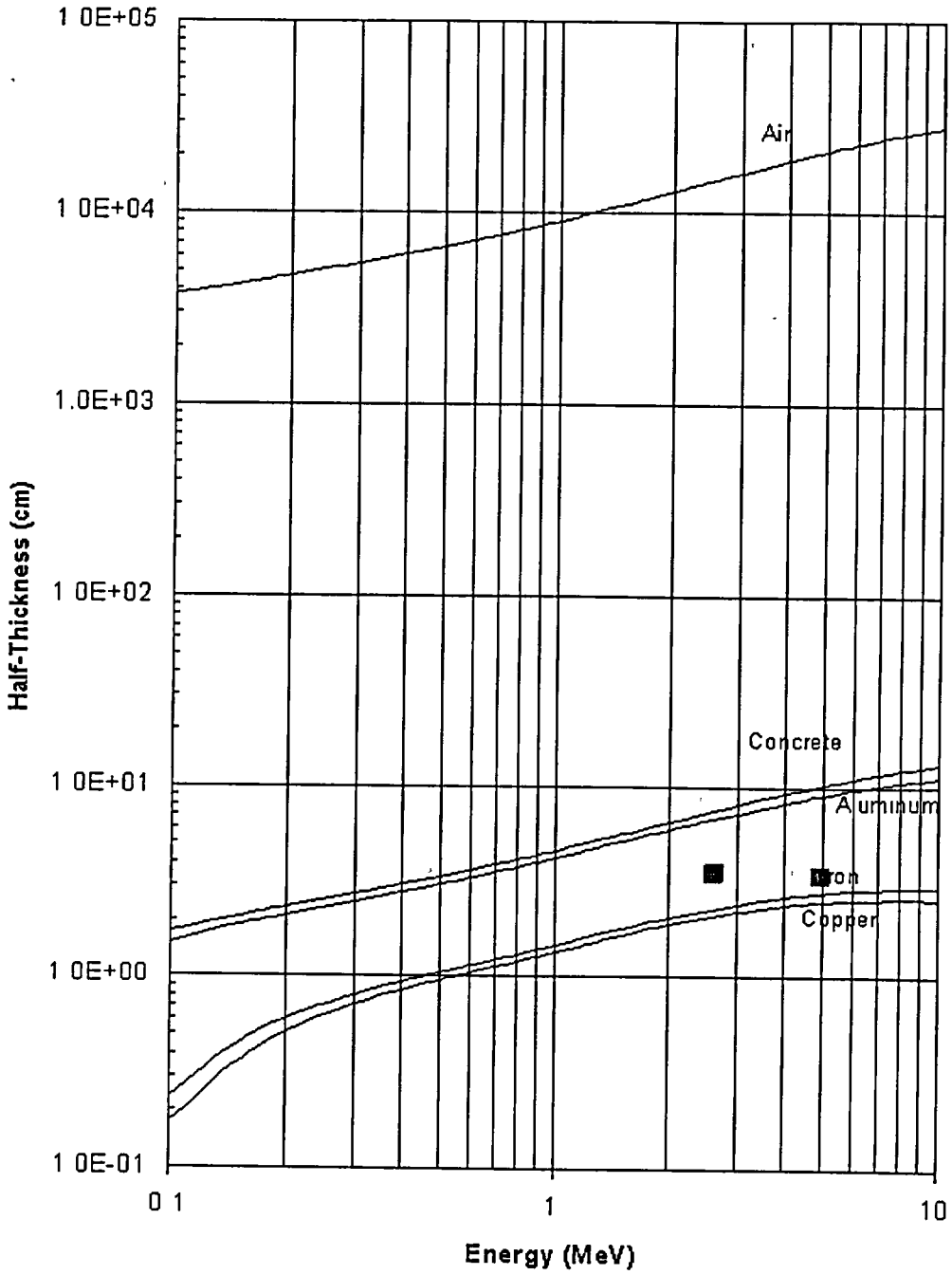


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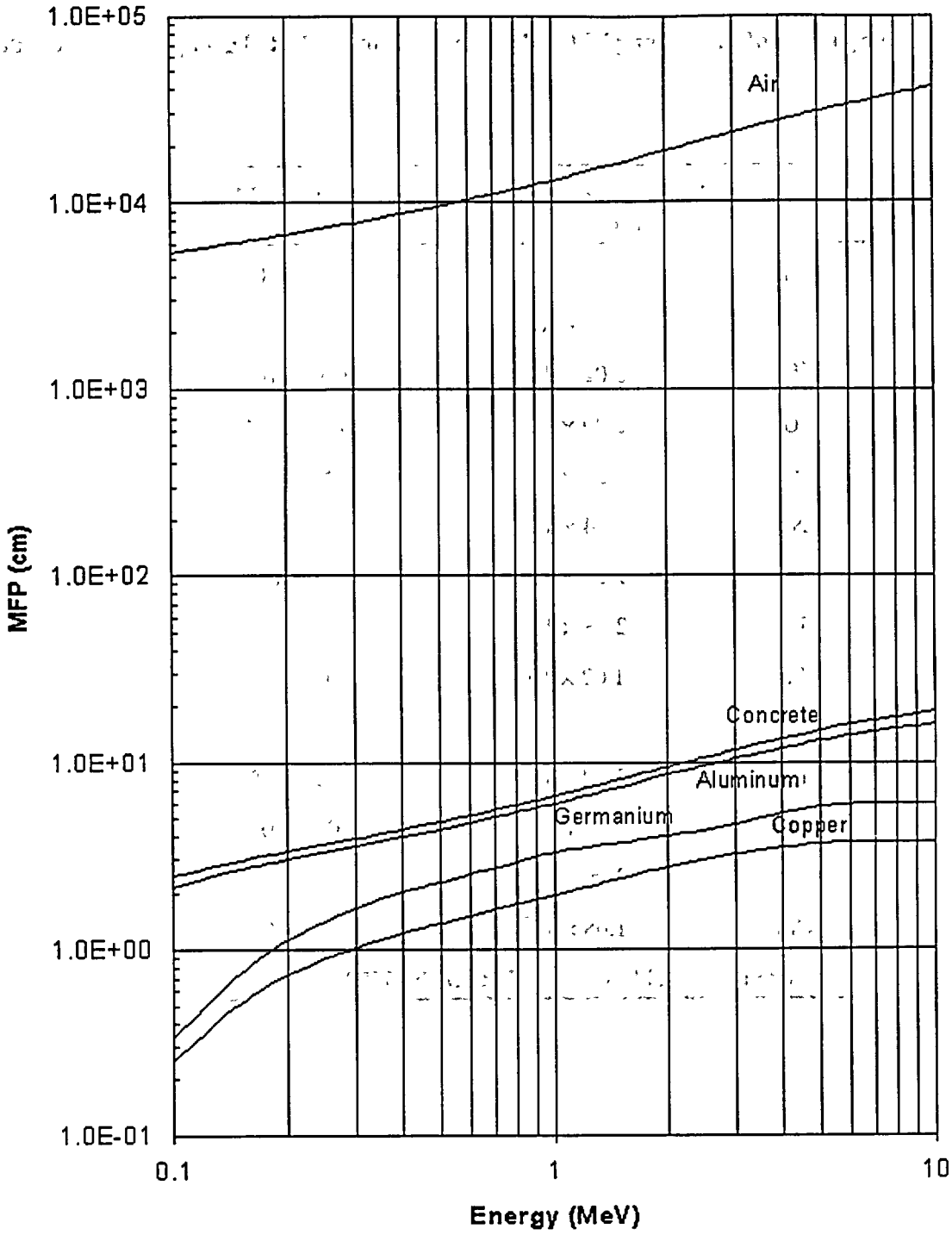
Figure A-4: The half-value thickness of gamma radiation as a function of energy in several different materials (Hubble and Seltzer, 1995)



3844 **Figure A-5: The mean-free-path of gamma radiation**
3845 **as a function of energy in several different materials (Hubble and Seltzer, 1995)**

3846 **Table A-1: A comparison of the fission yield and alpha yield for a selected group of radionuclides**

Isotope	Spontaneous fission yield† (neutron/s-g)	Alpha yield (alpha/s-g)
^{232}Th	6×10^{-8}	3.11×10^{3}
^{233}U	8.6×10^{-4}	3.01×10^{8}
^{234}U	5.02×10^{-3}	1.66×10^{8}
^{235}U	2.99×10^{-4}	3.98×10^{4}
^{238}U	1.36×10^{-2}	9.52×10^{3}
^{237}Np	1.14×10^{-4}	1.23×10^{7}
^{238}Pu	2.59×10^{3}	4.53×10^{11}
^{239}Pu	2.18×10^{-2}	1.70×10^{9}
^{240}Pu	1.02×10^{3}	6.17×10^{9}
^{241}Pu	5×10^{-2}	7.78×10^{7}
^{242}Pu	1.72×10^{3}	1.12×10^{8}
^{241}Am	1.18	1.08×10^{11}
^{242}Cm	2.10×10^{7}	9.11×10^{13}
^{244}Cm	1.08×10^{7}	2.28×10^{14}

† Adapted from Table 11-1 of NUREG/CR-5550.

A.4 Elements of Radiation Detection

Radiation detection is a broad field, which covers all types of radiation (e.g., x-ray, gamma-ray, alpha and beta particles, and neutrons) at levels ranging from background to extremely high levels associated with operational facilities (e.g., power and research reactors). The methods for detecting radiation are also quite diverse, ranging from calorimetry (measuring the decay heat) to event counting (counting the number of radiation interaction events). The purpose of this section is to introduce and discuss some of the concepts and quantities that are common to most radiation detectors.

A.4.1 Modes of Operation

Radiation detectors may be operated in two distinct modes:

- Current Mode: A radiation detector operated in current mode produces a current that is proportional to the event rate and the charge produced per event. An event is an interaction of a single particle (alpha, beta, or gamma ray) in which the particle transfers some or all of its energy within the sensitive region of the detector. Current mode operation is most often used in high-activity applications, such as ionization chambers.

- Pulse Mode: A radiation detector operated in pulse mode produces a pulse associated with individual events. In many instances, the pulse is proportional to the energy of the incident radiation. Detectors that utilize this energy proportionality feature are known as spectrometers. Other detectors, known as gross radiation counters, measure and count pulses regardless of energy.

A.4.2 Pulse Height Spectrum

When detectors that are operated in pulse mode are exposed to radiation, they produce a series of pulses that can be collected, sorted, and displayed. The result of such a process is a distribution of pulse heights, which is referred to as a pulse height spectrum. The pulse height can be related to the energy of the radiation, in which case, the spectrum is called an energy spectrum. The pulse height spectrum (or energy spectrum) is an important property of the detector output that is used to identify and quantify the radiation.

A.4.3 Energy Resolution

Two fundamental properties of a spectrometer are the precision with which it measures energy and its ability to distinguish between energies. Together, these properties are known as "energy resolution," which is expressed in terms of the full width of a peak at half its maximum value (also referred to as the full width at half maximum, or FWHM). In some cases, it is expressed in keV; in other cases, it is expressed as a percentage of the radiation energy. Spectrometers are sometimes characterized as low-, medium-, or high-resolution detectors. The resolution is a result of statistical processes associated with the transfer and collection of the energy associated with the radiation. In general, the higher the resolution, the better — and more expensive — the detector. However, in applications where there is a single energy or a very simple energy spectra, low or medium resolution is adequate.

A.4.4 Detection Efficiency

The two basic types of detector efficiency are absolute and intrinsic. Absolute efficiency is defined as

$$\varepsilon_{abs} = \text{response/number of particles emitted}$$

where the response is usually defined in terms of the number of pulses (or counts) recorded by the detector. The absolute efficiency depends not only on detector properties, but also on the *details of* the counting geometry. It can also be affected by environmental conditions, such as temperature and humidity.

By contrast, intrinsic efficiency is defined as

$$\varepsilon_{int} = \text{response/number of particles incident on the detector}$$

The intrinsic efficiency usually depends on the detector material, the radiation energy, and the physical thickness of the detector in the direction of the incident radiation.

A.4.5 Geometrical Efficiency

Geometrical efficiency is not a property of the detector and can only be defined in the context of the source-detector configuration. In that context, the geometrical efficiency is the fraction of radiation emitted from the source that intercepts the detector. It is expressed in terms of the solid angle, Ω, subtended by the detector with respect to the source:

$$\varepsilon_{geom} = \frac{4\pi}{\Omega}$$

The geometrical efficiency is closely related to the intrinsic and absolute efficiencies. For a source that emits radiation isotropically (i.e., in all directions) with no losses from attenuation, the relationship between ε_{abs}, ε_{int}, and ε_{geom} is expressed as

$$\varepsilon_{abs} = \varepsilon_{geom}\varepsilon_{int}$$

A.4.6 Sensitivity

The sensitivity of a detector has a formal definition, which involves "the ratio of the variation of the observed variable to the corresponding variation of the measured quantity, for a given value of the measured quantity" (ANSI N323A-1997). However, this is never the intended meaning when the term is used. Instead, the sensitivity of an instrument represents the minimum amount of activity or activity concentration that will produce a response from the detector that is statistically significant from the response in the absence of radioactivity.

Minimum Detectible Concentration and Sensitivity

When discussing limits of detectability, the two expressions that are often used are minimum detectible concentration (MDC) and sensitivity. The term "minimum detectible concentration" implies a degree of statistical rigor and mathematical formality, while the term "sensitivity" is generally regarded as a colloquialism. Even though regulatory bodies, such as the NRC, require the rigor and formality of the MDC, this appendix uses the term "sensitivity" because it is consistent with the terminology of instrument manufacturers, and it avoids some of the persistent difficulties associated with the formal definition of MDC. For example, NUREG-1507 reviewed the literature on the statistical interpretation of MDC as part of a brief study addressing the consistency of MDC values for five MDC expressions. The various expressions led to a range of MDC values for a gas proportional counter. While the spread of MDC values was modest, it illustrates the fact that the MDC is not unique and depends upon the statistical treatment of the data. Others (MacLellan and Strom, 1999) argue that traditional MDC formulas (and decision levels) are wrong. In their view, these traditional formulas do not adequately account for the discrete nature of the Poisson distribution for paired blank measurements at low numbers of counts. Using the term "sensitivity" retains the concept that is embraced by the MDC, while avoiding some of the difficulties.

Factors Affecting Sensitivity

The sensitivity of any detection method or system depends on the individual processes and mechanisms that are particular to that method or system. In broad terms, any process that degrades or absorbs radiation energy adversely affects sensitivity. The sequence of events that lead to a signal from a detector begins with the decay of nuclei, or the de-excitation of electrons to produce radiation energy. The radiation energy must then reach the active or sensitive region of the detector, where it is converted to information carriers. Any loss of energy that occurs throughout this sequence results in a loss of sensitivity. Table A-2 addresses the primary energy and information loss mechanisms associated with various processes involved in radiation detection.

Table A-2: Loss mechanisms for radiation detection

Process	Loss Mechanism	Significance
transport from source to sensitive region detector	radiation scattering and absorption	very significant for weakly penetrating radiation, potential loss of all energy
conversion of radiation energy to information carriers	energy to create information carriers	the lower the energy loss, the more information carriers and the better the energy resolution
charge collection	recombination (gases+ semiconductors), trapping (semiconductors), and quenching (scintillators)	significant, in the sense that these processes determine the size of the detector
pulse handling	pileup and ballistic deficit	very minor for low count rates
pulse counting and storage	conversion and storage time	very minor for low count rates
spectrum analysis	peak-fitting algorithm and continuum subtraction[a]	potentially significant, if small peaks on large continua

[a]With the exception of this item, all of the listed loss mechanisms represent physical processes.

Table A-2 does not reflect one of the most significant losses, which does not involve any physical mechanism. Specifically, that loss occurs when the emitted radiation does not intercept the detector. Most conventional detectors have relatively small active areas and intercept only a small fraction of the emitted radiation. The one key to improving sensitivity involves designing detection systems with large active areas that optimize the geometrical efficiency.

The sensitivity has two components, both of which involve the detector response. One focuses on the response to radiation from the source; the other deals with the response to everything else. (In this case, "everything else" is referred to as "background.") Optimizing the sensitivity means maximizing the signal from the source, while minimizing the contribution from background. Maximizing the signal is a matter of energy conservation; the more radiation energy that reaches the detector, the greater the potential for producing a signal and, consequently, the greater the sensitivity. Minimizing the contribution from background is a matter of background reduction, which works not by absorbing energy, but by rendering unusable the information that the energy produces. Background is an interference mechanism.

Interference affects two components of the detection and measurement process: (1) the characteristic radiation from the source (external) and (2) the signal chain (internal). Some examples of external interference come from spectroscopy, where two or more radionuclides can emit characteristic radiation at essentially the same energy. For example, both ^{226}Ra and ^{235}U emit approximately a 186-keV gamma ray and both occur in natural uranium. Another form of interference, which is related to spectrometry, concerns the loss of spectral information (in the form of peaks) from scattered radiation.

Scattered radiation is radiation that has interacted with matter in such a way that its characteristic energy has changed. Scattered radiation can potentially interfere or obscure energy peaks. The continuum in a spectrum results from scattered radiation. Radiation can be scattered in the detector, in the source, or from materials surrounding the detector. While techniques have been developed to extract information from the continuum, it usually only obscures small peaks and, in some cases, renders the measurement useless.

Figure A-6 shows the effect of resolution and interference on a gamma ray spectrum. The area under the peak is the same for all three cases; however, the peak in the bottom spectrum is all but lost to the continuum. At low radionuclide concentrations, the radiation emitted from most radionuclides competes with natural background radiation. Many laboratory systems have large and elaborate shields to limit the interference of natural background radiation. Techniques have been developed to reduce the contribution of scattered radiation. These techniques include anti-coincident shielding and coincidence counting, which make use of concurrent or coincident events in multiple or segmented detectors.

Electronic noise is a form of interference that acts on the signal chain. Electronic circuits used to amplify and process pulses have two basic forms of noise: thermal and shot. Thermal noise refers to noise occurring in resistors in absence of current flow, while shot noise is associated with a flow of current. The technology used to process electronic signals is well developed and the instruments are well designed. Therefore, electronic noise is not typically a limiting factor for detector sensitivity. Rather, most of the problems with interference come from external sources.

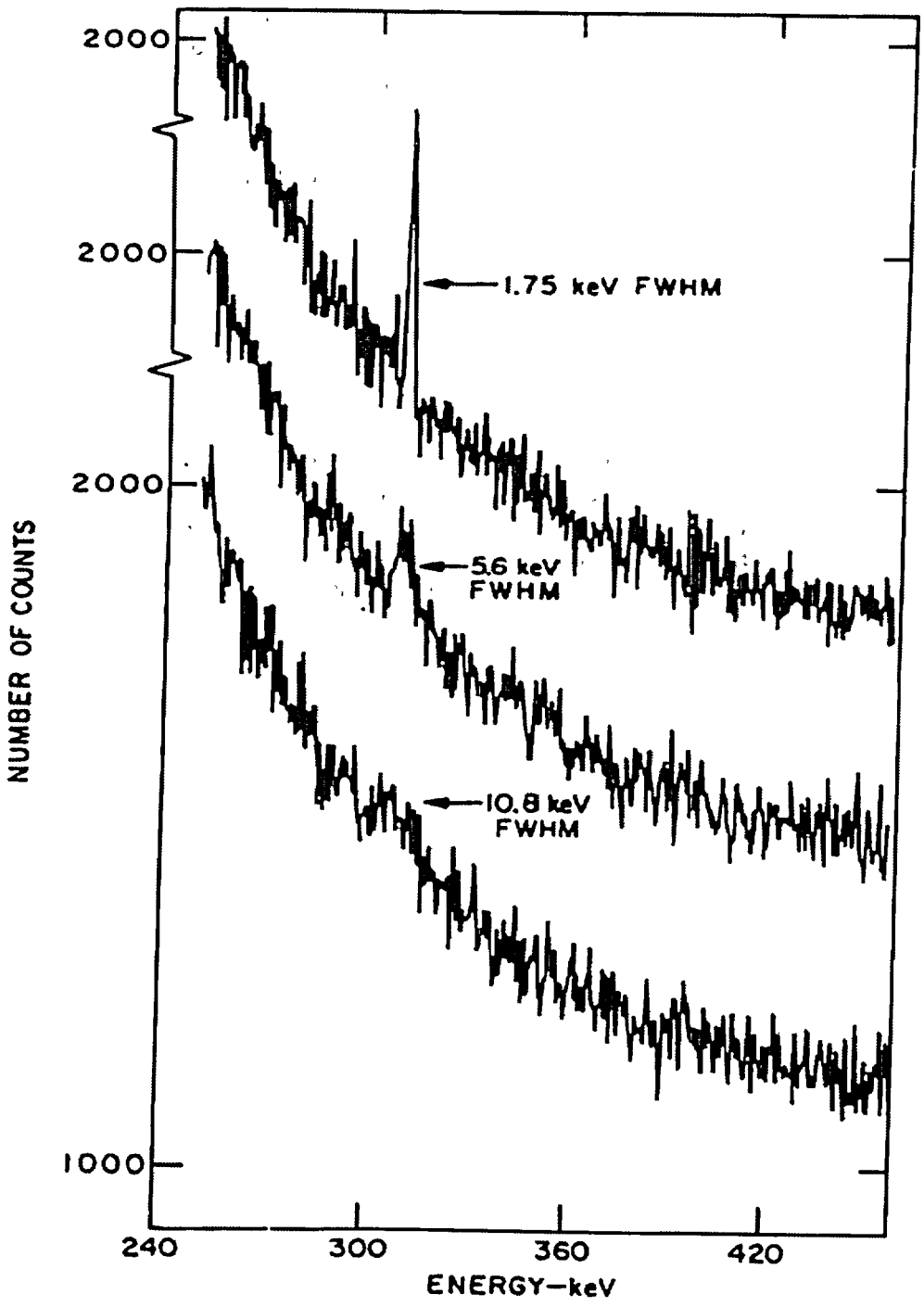


Figure A-6: The effects of interference from scattered radiation on the ability to detect a peak for several measured energy resolutions (Knoll, 2000)

Sensitivity and Energy Resolution

When spectroscopy is used to measure activity, the sensitivity is affected by the energy resolution. The issue of energy resolution and its impact on sensitivity is essentially the issue of background reduction. Recall that the sensitivity represents the minimum amount of activity that produces a response, in counts, that is statistically significant from background. If the detector has no energy resolution, any particle that enters the detector's active volume will produce counts. This, in turn, will increase the amount of activity that must be present in order to establish a response that is statistically significant from background. Because the decay of a radionuclide often emits radiation with a very specific energy (e.g., alpha decay), spectroscopy can be used to restrict the response to an energy range that corresponds to the decay of the radionuclide in question. The better the energy resolution, the greater the selectivity in the number of counts and the greater the sensitivity. In this way, spectroscopy is a form of background reduction.

Factors Affecting Energy Resolution

The number of information carriers affects the resolution. That is, the more information carriers that are produced in the detector's active volume, the greater the energy resolution. This is a result of the statistical fluctuation in the number of information carriers. Under the assumption of a Poisson process, the variance in the number of information carriers is equal to the number of information carriers. Assuming Poisson statistics, the energy resolution, measured in terms of the FWHM, becomes

$$\text{FWHM} = 2.35/\sqrt{N}$$

where N is the number of information carriers. Hence, the greater the number of information carriers, the better the energy resolution. However, measurements of the energy resolution of some types of radiation detectors have shown that the achievable values for FWHM can be lower than the value predicted by the above equation. These results indicate that simple Poisson statistics do not describe the processes that give rise to the formation of each individual charge. The Fano factor has been introduced in an attempt to quantify the departure of the observed statistical fluctuations in the number of charge carriers from pure Poisson statistics. The Fano factor is the ratio of the observed variance to the variance predicted by Poisson statistics. Hence, the smaller the Fano factor, the better the resolution. Fano factors for semiconductor devices and proportional counters are much less than unity, whereas scintillation counters have a Fano factor of about unity.

When radiation energy is absorbed in a detector, it must be converted into a form from which information can be extracted. The term "information carrier" is used to denote, in a general way, the particles that participate in the conveyance of information. For most detectors, the particles consist of ions, electrons, and electron-hole pairs. The effectiveness of a detector in terms of producing information carriers relates to the energy that is lost as a result of their creation. The higher the loss in energy, the less information that can be extracted. Ultimately, these information carriers deliver their information in the form of a charge pulse. Table A-3 lists some key properties of some common detectors.

Table A-3: Important parameters associated with common radiation detectors

Detection system	Information carrier	Energy loss per information carrier (eV)	Number of information carriers per 100 keV	Charge pulse amplitude per 100 keV (coulombs)
NaI (Tl) + PMT[a]	Photoelectron	~120	800	10^{-11}
Proportional tube	Ion pair	25 – 35	3000 – 4000	10^{-12}
Germanium (Ge) detector	Electron-hole pair	3	33000	10^{-14}

[a] Refers to a sodium iodide (NaI) gamma detector with thallium (Tl) as an activator or doping agent. The solid crystalline detector is physically connected to a photomultiplier tube (PMT). Refer to the following text for further information.

A sodium iodide (NaI) gamma detector with thallium (Tl) as an activator or doping agent is a "scintillator," which means that the radiation produces light in a crystalline solid when absorbed. The scintillator is coupled, optically, to a photocathode, which is part of a photomultiplier tube (PMT) assembly, a device that converts the light to electrons (photoelectrons). The "cost" (or loss in energy) for producing these photoelectrons is approximately 120 eV. A 100-keV photon produces about 800 photoelectrons. Further amplification by the PMT results in a charge pulse of 10^{-11} coulombs.

A proportional counter is a gas-filled detector that converts radiation energy to ions. The loss in energy for producing these ions is much less than for the NaI(Tl) detector, resulting in many more information carriers for a 100-keV photon. Note in Table A-3 that an increase in the number of information carriers does not translate to a larger charge pulse.

The germanium detector consists of a very pure crystal of germanium. The crystalline structure conveys special conducting properties. The germanium detector is a solid-state semiconducting diode, which produces electron-hole pairs when radiation energy is absorbed. Note that the energy loss is very small, resulting in a huge number of information carriers for a 100-keV photon. Again, Table A-3 shows that, despite the large number of information carriers, the associated charge pulse is relatively small. While increasing the detector size improves sensitivity, it must be noted that the detector size can have a deleterious effect on resolution. There are loss mechanisms (see Table A-2) that affect the information carriers as they migrate through the material to be collected. The larger the detector, the greater the chance that the information carriers will be neutralized. The loss of information carriers means that a decrease in resolution will occur.

Radionuclides Commonly Identified with Clearance

Of the 1,500 radionuclides, only about 10 to 15 percent present a long-term risk to the public. A number of studies have investigated screening levels for radionuclides associated with clearance (NCRP 129, AEC 1974, Hill 1995, IAEA 1996, EPA 1997, NCRP 1999, NRC 1999, ANSI 1999, EUR 2000). Rather than develop a new list or augment existing lists, Table A.4 lists radionuclides that are common to all of the aforementioned studies and provides some basic information about them. The last column refers to specific radiation detectors, a brief description of which is presented in Appendix B.

Table A-4: Information on selected radionuclides

Radionuclide	Series/decay chain	Half-life (y)	Primary radiation (keV)	Potential surrogate	Standard method of detection (survey)
^{3}H	none	12.28	β (5.69)[b]	none	swipes + liquid scintillation counter
^{14}C	none	5730	β (49.5)[b]	none	thin-window G-M detectors/ GP detectors[f]
^{54}Mn	none	0.85	γ (834.8)	-[d]	gamma or x-ray survey meter
^{55}Fe	none	2.7	x-ray (5.89)	^{60}Co	gamma or x-ray survey meter
^{60}Co	none	5.27	γ (1332)	-[d]	gamma or x-ray survey meter
^{63}Ni	none	100	β (17.1)[b]	^{60}Co	thin-window G-M detectors/ GP detectors[f]
^{90}Sr	decays in ^{90}Y	28.6	β (196)[b]	^{137}Cs	thin-window G-M detectors/ GP[f] detectors
^{99}Tc	none	213000	β (84.6)[b]	^{137}Cs[e]	thin-window G-M detectors/ GP detectors[f]
^{134}Cs	none	2.06	γ (605)	-[d]	gamma or x-ray survey meter
^{137}Cs	decays in Ba-137m	30	β [c]/γ (662)	Ba-137m	gamma or x-ray survey meter
^{232}Th	Th series (parent)	long[a]	α (4010)	^{228}Ac, ^{208}Tl	ZnS/ GP detectors[g h]
^{234}U	U series (progeny)	244500	α (4773)	none	ZnS/ GP detectors[g h] scintillators
^{235}U	Ac series (progeny)	long[a]	α (4389)	-[d]	ZnS/ GP detectors[g h]

Table A-4: Information on Selected Radionuclides (continued)

Radionuclide	Series/decay chain	Half-life (y)	Primary radiation (keV)	Potential surrogate	Standard method of detection (survey)
^{238}U	U series (parent)	long[a]	α (4198)	^{234}Th, ^{234m}Pa	ZnS/ GP detectors[g,h]
^{226}Ra	U series (progeny)	1640	α (4602)	Bi-214, Pb-214	ZnS/ GP detectors[g,h]
^{238}Pu		87.7	α (5499)	none[i]	ZnS/ GP detectors[g,h]
^{239}Pu		24065	α (5156)	none[i]	ZnS/ GP detectors[g,h]
^{240}Pu		6537	α (5168)	none[i]	ZnS/ GP detectors[g,h]

[a] half-life > 10^7 y
[b] average β energy
[c] not used - equilibrium with progeny Ba-137m
[d] not necessary, emits γ
[e] speculative
[f] gas proportional counter operated in α+β mode
[g] ZnS - Zinc Sulfide Scintillator
[h] gas proportional counter operated in α mode
[i] does emit gammas of low intensity (<0.1%)

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Appendix B: Advanced/Specialized Instrumentation

B.1 Conventional Radiation Detectors

This appendix provides information on a wide range of radiation detectors and detection methods. Beginning with conventional radiation detectors, it profiles various detection systems as they relate to clearance surveys. While this appendix addresses many examples of commercially available systems, it could not be, and is not intended to be, exhaustive. It does, however, provide a snapshot of systems that could have an impact on clearance surveys, and it discusses emerging and advanced radiation detectors and software programs. While these systems are expected to have an impact on the field of radiation detection, their impact on clearance surveys is uncertain.

The majority of instruments described in this appendix use one of the following types of radiation detectors:

- Gas-filled proportional counters and Geiger-Mueller (GM) tubes. Gas proportional detectors come in two basic types: sealed systems and gas flow proportional systems.
- Scintillation detectors may be either inorganic (e.g., Zinc Sulfide and Sodium Iodide) or organic (e.g., plastic).
- Solid-state semiconductors include high-purity germanium (HPGe) and cadmium zinc telluride (CZT).

While a complete discussion of these detectors is beyond the scope of this appendix, the following table summarizes the properties and features of these detectors. A more comprehensive treatment of these detectors may be found in Knoll (2000).

Table B-1: Properties of some common detectors

Detector Type	Comments
Gas-Filled	
gas flow proportional counters	▪ use thin windows (aluminized Mylar 0.2 mg/cm^2) to detect alpha and low-energy beta particles
	▪ require a supply of P-10 gas (a mixture of argon and methane gas)
sealed proportional counters	▪ depending on the mass density of the window, can respond to alpha, beta, and gamma radiation
	▪ can be attached to a multichannel analyzer to perform spectroscopy
GM	▪ used primarily for gross radiation measurements
	▪ depending on instrument design, can detect alpha, beta, and gamma radiation

Table B-1: Properties of some common detectors (continued)

Detector Type	Comments
Scintillator	
ZnS(Ag)	▪ limited to thin screens or films ▪ used to detect alpha radiation
NaI(Tl)	▪ used to detect gamma radiation ▪ has superior light output ▪ hygroscopic (absorbs moisture); must be sealed ▪ can be fabricated into a variety of shapes and sizes ▪ can be attached to a multichannel analyzer to perform spectroscopy
organic (plastic)	▪ Responds well to charged particles (e.g., beta particles) ▪ non-hygroscopic and rugged ▪ inexpensive ▪ can be made fairly large (large-area detector) ▪ low density and low atomic number make it inefficient for medium- and high-energy gammas

Table B-1: Properties of some common detectors (continued)

Detector Type	Comments
Solid-State Semiconductor	
HPGe	▪ used for gamma-ray spectroscopy ▪ has superior energy resolution ▪ large volume; high purity crystals can be grown with volumes exceeding 400 cm^3 ▪ high density and atomic number make it well-suited for medium- and high-energy gammas ▪ must be maintained at liquid nitrogen temperatures (77 °K) ▪ expensive
CZT	▪ can be operated at room temperature ▪ used for medium-resolution gamma-ray spectroscopy ▪ small volume (< 1 cm^3)

B.2 Conventional Field Survey Instrumentation

This section briefly describes conventional radiation detection instruments for field surveys. These instruments typically are small, portable systems that have a radiation detector, such as one of those mentioned above, coupled to an electronic data collection and visualization package. The instruments are categorized below in terms of the radiation for which they have the greatest efficiency. For more detailed information on these instruments, see the MARSSIM. For an evaluation of their MDCs, see NUREG-1507.

Alpha

These detectors use silver-activated Zinc sulfide (ZnS(Ag)) to detect alpha radiation. Alpha particles enter the scintillator through an aluminized Mylar window. A typical probe area covers about 75 cm^2.

Alpha/Beta

While gas flow proportional detectors can detect alpha and beta radiation, they can distinguish between the two by adjusting the operating voltage. The active volume of the detector is filled with P-10 gas. Radiation enters the active volume through an aluminized Mylar window. Typical probe areas cover about 100 cm^2.

Beta/Gamma

Geiger-Mueller detectors or "pancake" detectors are used to detect beta and gamma radiation. The detector tube is filled with an inert gas, which is a mixture of argon, helium, neon, and a halogen-quenching gas. Radiation enters this tube through a mica window. A typical probe area covers about 20 cm^2.

Gamma

Thallium-activated sodium iodide (NaI(Tl)) scintillation detectors are used to measure gamma radiation. Since gamma radiation is much more penetrating than alpha and beta radiation, the type of detector window is not crucial, but these instruments typically use aluminum. The cylindrical crystals range in size from 2.5 cm x 2.5 cm (height x diameter) to 7.6 cm x 7.6 cm. Integrated systems are often operated on a gross count rate mode. However, recent developments in microchips and spectrum analysis software for NaI(Tl) detectors provide for greater flexibility and expanded use, while still retaining its portability. These new systems are discussed in the next section.

B.3 Specialized Instrumentation

Along with the conventional radiation detection instrumentation, there is a substantial assortment of instruments that have both generic and specialized uses. While this section addresses many examples of the commercially available radiation detection systems that are relevant to clearance, it is not intended to be exhaustive. In addition, it must be noted that the following discussion should not be construed as an endorsement of any of these products by the U.S. Nuclear Regulatory Commission (NRC).

When available and appropriate, this section provides capital cost information, using the following four indicators to signify four capital cost ranges; when appropriate and available, estimated measurement costs may also be provided.

$ - less than $1k

$$ - greater than $1k, but less than $10k

$$$ - greater than $10k, but less than $100k

$$$$- greater than $100k

B.3.1 General Detectors

Alpha Track Detectors

An alpha track detector is a passive, integrating detector used to measure gross alpha surface contamination on flat surfaces such as concrete, metal, and wood. It can also be used to determine soil activity levels. The 1-mm thick polycarbonate material is deployed on or close to the surface to be measured. Microscopic damage to the plastic matrix occurs when alpha particles strike the surface. This damage is then made visible by etching the material in a caustic solution. After etching the plastic, an optical reader is used to count the number and density of tracks. The track density is then related to the source activity through appropriate calibrations. The standard detector size is 2 cm^2. Alpha track detectors provide gross alpha measurements with no measurable response to beta or gamma radiation.

Sensitivities for surface contamination are 0.03 Bq/cm^2 (200 dpm/100 cm^2), 0.005 Bq/cm^2 (30 dpm/100 cm^2), and 0.002 Bq/cm^2 (10 dpm/100 cm^2) for deployment times of 1, 8, and 48 hours, respectively. For soil contamination, sensitivities are 11 Bq/g (300 pCi/g), 3.7 Bq/g (100 pCi/g), and 0.7 Bq/g (20 pCi/g) for deployment times of 1, 8, and 96 hours, respectively. If deployed along the side of a trench, the alpha track detector can provide depth profile information of the contamination. Alpha track detectors can also be deployed in pipes and on or inside of equipment.

Advantages of alpha track detectors over conventional electronic survey instrumentation are that (1) plastic can be molded into various shapes and sizes to accommodate locations that are not easily accessible for measurements, (2) detectors are passive with no electronic failures, (3) they are inexpensive and rugged, (4) they have no measurable response to beta or gamma radiation, and (5) activities down to background levels can be determined depending upon deployment times and site conditions.

Disadvantages include (1) the etching and counting must be performed by a vendor, requiring shipping to the vendor in a timely manner; (2) measured surfaces must be free of dust, dirt, water, oil, or other material that will attenuate alpha emissions; (3) the plastic is sensitive to scratching, abrasion, oils, perspiration, and radon; and (4) measured surfaces must be relatively flat.

Capital Cost: $$$

Unless an optional automated scanner is provided, each detector is returned to the vendor for reading, at a cost of $5 to $10 per measurement.

Electret Ion Chambers

An electret ion chamber (EIC) is a passive, integrating ionization chamber made from electrically conducting plastic. Ionizing radiation enters the ion chamber through a thin aluminized Mylar window. The electret is a positively charged piece of Teflon®, which produces an electric field that collects the electrons produced by the alpha ionization. As the electrons collect over time on the electret, the charge on the electret becomes neutralized. After the predetermined deployment time, the electret is removed and a charge reader is used to measure the remaining charge of the electret. Knowing the original and final charges, an activity calculation can be performed. An EIC does not require electrical power to operate. An adequate sampling plan is the only technical requirement for using this system, as deployment does not require specially trained technicians.

Electret ion chambers have traditionally been deployed to measure radon concentrations in the air of homes and businesses. The literature also discusses other applications of EICs, such as measuring alpha and low-energy beta surface contamination, measuring alpha soil concentration, quantifying alpha contamination inside piping, and performing gamma dose measurements. EICs can be used for inexpensive alpha measurements and/or for areas where conventional alpha probes cannot measure. While the deployment time can be long, the measurement time is very short and sensitivities are much better compared to traditional detectors such as a gas-proportional counter. Also, EICs can be used in difficult-to-measure situations, such as tritium contamination or alpha contamination inside piping. The EICs measure gross alpha, gross beta, gross gamma, or gross radon.

An example of a commercially available EIC is Rad Elec Inc.'s E-PERM alpha radiation monitoring systems. These systems are available in sizes ranging from 50 to 180 cm^2 and in various electret thicknesses depending on the required sensitivity.

Capital Cost: $$

Alpha Surface Measurements

Oak Ridge National Laboratory (ORNL) has developed a procedure, known as Method RA010, using Rad Elec's E-PERM alpha radiation monitors for use in decontamination and decommissioning (D&D) operations (Meyer *et al.*, 1994). Costs for deploying the E-PERM system were reported to be $5 per measurement for a large-scale survey.

Levinskas *et al.* studied low-level alpha measurements using a 145-ml EIC with a deployment time of 48 hours. They reported that the results were within 5-percent accuracy, compared to NIST-traceable calibrated gas flow proportional counters. Sensitivity for this measurement method was reported to be $(1.1 \pm 0.5)\times10^{-3}$ Bq/cm^2 (6.4 ± 3.0 dpm/100 cm^2) at the 95-percent confidence level.

Alpha Soil Measurements

Meyer *et al.*, 1995, described a method for taking *in situ* measurements of alpha contamination in soils using EICs. Probe sizes of 50 and 180 cm^2 are used. With a 50-cm^2 EIC, detection limits of 1 Bq/g (27 pCi/g), 0.7 Bq/g (18 pCi/g), 0.5 Bq/g (13 pCi/g), and 0.3 Bq/g (9 pCi/g) were achieved for deployment times of 6, 12, 24, and 48 hours, respectively. Survey costs ranged from $8 to $25 per measurement.

Alpha Contaminated Pipes

Direct measurement of alpha contamination inside pipes is difficult because of the short range of alpha particles. However, measurements of the ionization caused by the alpha radiation in air can be used to infer alpha contamination. An EIC is placed at the end of the pipe and air is directed through the pipe to the EIC. The collection of the secondary ions reduces the charge of the electret. Calibration is performed by locating an alpha source of known strength and determining response factors. In a 15-minute measurement, uniform alpha contamination in a pipe with a 15-cm diameter can detect an activity of 0.04 Bq/cm^2 (2.2 dpm/cm^2) (Dua *et al.*, 1997).

Beta Surface Measurements

Sensitivities for tritium measurements are reported to be 1 Bq/cm^2 (6,000 dpm/100 cm^2) with a deployment time of 1 hour, and 0.05 Bq/cm^2 (300 dpm/100 cm^2) for 24 hours. ^{99}Tc sensitivities are 0.08 Bq/cm^2 (500 dpm/100 cm^2) for 1 hour and 0.003 Bq/cm^2 (20 dpm/100 cm^2) for 24 hours.

Gamma Measurements

The response of this type of detector to gamma radiation is nearly independent for energies ranging from 15 to 1,200 keV. A 30-day deployment with 50-ml chamber is required to quantify an ambient field of 6.9×10^{-13} C kg^{-1} s^{-1} (10 μR/hr). Using a 1,000-ml chamber can reduce the deployment time to 2 days. The smaller chamber is generally used for long-term monitoring.

Portable Gamma-Ray Spectrometers

There are a wide variety of handheld spectrometers available on the market. They consist of two general types, including integrated systems and modular systems. The integrated systems have the detector and electronics contained in a single package. The modular systems separate the detector from the electronics. These spectrometers employ small scintillators, typically NaI(Tl), and room temperature solid semiconductors such as CZT. Recently, the systems using NaI(Tl) scintillators utilize special analysis software to do isotope identification. These systems represent an advancement over the conventional scintillation probes connected to rate meters. The systems using CZT have superior resolution (compared to scintillators) and, therefore, perform the standard peak analysis. The preferred application for the devices tends to be in nuclear non-proliferation, where isotope identification is more important than sensitivity.

Three systems of note include SAM-935 from Berkeley Nucleonic Corporation, RADSMART from SAIC, and the GR-130 miniSPEC from Exploranium. All of these systems are handheld and do some form of isotope identification. The SAM-935 uses an NaI(Tl) scintillator and a spectrum analysis technique called Quadratic Compression Conversion™ to perform rapid isotope identification. The RADSMART uses a proprietary CsI scintillator coupled to a photodiode. The isotope identification is performed using spectrum templates rather than peak analysis, which is often problematic for low-to-medium resolution spectrometers such as CsI. The GR-130 miniSPEC also uses an NaI(Tl) scintillator, but performs a peak analysis on the spectrum for isotope identification. These systems are no more sensitive to radiation than the conventional instruments (e.g., small scintillators operated in a gross count mode), but they can provide information on radionuclide identity. These systems are rather new and there is little or no data available to support claims that the spectrum analysis programs can significantly improve the sensitivity.

Capital Cost: $$$

X-ray Fluorescence

X-ray fluorescence (XRF) is a spectroscopic method in which secondary x-ray emission is generated by the excitation of a sample with x-rays. The x-rays eject inner-shell electrons, then outer-shell electrons take their place and emit photons in the process. The wavelength of the photons depends on the energy difference between the outer-shell and inner-shell electron orbitals. The amount of x-ray fluorescence is sample-dependent, and quantitative analysis requires calibration with standards that are similar to the sample matrix. The nature of the method does not allow for isotope identification (but rather the element itself) and is generally not useful for measuring the fluorescence yield in elements with atomic numbers less than 32.

Recently, field-portable x-ray fluorescence (FPXRF) systems have been developed that are available commercially. These systems use sealed sources to produce fluorescent x-rays and contain a small x-ray spectrometer to measure the fluorescent x-rays. The advantage of this technology includes the ability to measure solids, liquids, thin films, and powders. FPXRF is a useful technique for screening or surveying materials for their elemental content when portability, short analysis times, and real-time results are required. For information concerning the performance of FPXRF, see Potts (1999) and U.S. DOE (1998a).

An FPXRF, known as the Spectrace 9000, is commercially available from Thermo NORAN's KevexSpectrace. This device uses iron-55 (^{55}Fe), cadmium-109 (Cd-109), and americium-241 (^{241}Am) to produce a wide range of excitations, capable of exciting atoms of atomic number 16 (sulfur) to 92 (uranium). T his particular unit can simultaneously measure 25 elements. The detector uses a mercuric iodide semiconductor to measure the fluorescent x-rays. The Spectrace 9000 can operate on battery or 110-Vac power. Measurements can be made on a surface, or small samples can be taken and placed in a small counting chamber attached to the probe.

Capital Cost: $$$

Compton Suppression Spectrometer

Background reduction is critical to maximizing detector sensitivity. Typical methods for background reduction include lead shields and anti-Compton shields made of NaI(Tl) (or bismuth germanate[8]). Princeton Gamma Tech (PGT) has developed a Compton Suppression Spectrometer (CSS) based on the Duode detector, which is a transversely segmented single crystal of high-purity germanium. PGT developed the crystal processing techniques specifically to improve detector performance at low energies without sacrificing the efficiency of a large HPGe detector. Suppression is achieved by detection and electronic vetoing of coincident energy deposition events in the rearmost segment of the crystal. At low energies, most of these coincident events are from background photons, which have undergone forward Compton scattering from the front "planar" segment. The suppression provided by this geometry is ideal for rejecting these background events.

In general, the Duode suppression provides significant background reduction across the energy range and improvement in the signal-to-noise ratio (SNR) and, thus, reduced peak fitting errors in a limited energy range. For a strong peak, a reduction in background has little effect on the SNR or peak-fitting error. For a weaker peak, such as 2–3 standard deviations (σ) above background or lower, the improvement in the SNR and reduced peak fitting error can be significant. The principal benefit of the Duode is for measurement of those isotopes which would normally be lost in the background (Haskins *et al.*, 2000).

Capital Cost: $$$

[8]Bismuth Germanate ($Bi_4Ge_3O_{12}$ or BGO) is a scintillation material that has a high density (7.13 g/cm^3) and large atomic number (83), which makes it a preferred detector material for high-energy gamma-rays and anti-Compton shields.

B.3.2 Application-Specific Detection Systems

Responding to the measurement needs of nuclear facilities engaged in D&D activities, instrument manufacturers have developed specialized detection systems and, in a few instances, services that are designed to facilitate and expedite radiation measurements associated with the D&D effort. Many of these systems use traditional detectors (gas proportional counters, plastic scintillators, and NaI(Tl) scintillators) coupled to rate meters. The design goal of these systems is to optimize throughput while detecting contamination at guideline levels[9]. These goals have been more-or-less accomplished by using large shielded detectors and arranging them in a manner to optimize the geometrical efficiency. Shielding the detectors helps to improve the SNR by reducing the background. This section briefly addresses the following systems and/or applications:

- conveyorized survey monitors
- floor and surface contamination monitors
- *in situ* gamma-ray spectrometry systems
- *in toto* monitors
- pipes (interior/exterior)
- subsurface
- portal monitors

This section does not address systems that have been developed specifically for the assay of transuranic waste. Some of the systems are quite sophisticated and use active measurement techniques, as discussed in Section B.4.

Conveyorized Survey Monitors

Conveyorized survey monitors (CSMs) automate the scanning or hand-frisking of materials. Current systems have been designed to measure materials such as clothing (laundry monitors), copper chop (small pieces of copper), concrete rubble, and soil. A typical CSM consists of a conveyor belt that passes under or between an array of detectors. Most systems use an array of gas flow proportional counters in a staggered configuration. The staggered configuration eliminates blind spots (locations where contamination may be present but cannot be detected because the radiation cannot reach the detectors). Systems range from small monitors with small belts to large trailer-mounted systems for measuring and segregating (in terms of activity) rubble, debris, and soil.

Commercial Systems

Eberline manufactures several conveyor systems. Model ACM-10 is an automated contamination monitor utilizing a single conveyor belt. Radiation measurements are performed with an array of 10 large-area (503-cm^2) gas proportional detectors that are located above and below the belt. Model 140A is a larger version of the ACM-10, which utilizes two conveyor belts to compress the material being measured (typically clothes). This model uses an array of gas flow proportional counters, 14 above and 14 below. Ludlum manufactures a laundry monitor (Model 329-32) that also utilizes a single conveyor belt. It uses two arrays of sixteen 100-cm^2 gas proportional detectors each.

[9] Guideline levels depend on the actual application and may be site specific.

BNFL markets a CSM that is intended for rubble, debris (e.g., concrete and steel), and soil. This high-throughput system (~ 1.5×10^4 kg/h) uses a modular detection approach, which means that it has individual detector modules to measure specific radiation types. For example, the system has a gross gamma detection module, an alpha/beta surface detection module, a low-resolution gamma spectrometry module, and a high-resolution gamma spectrometry module. Multiple modules can be linked together when data from different radiation types are needed. Canberra Industries also markets a CSM for rubble, debris, and soil. This trailer-mounted system is also a high-throughput system; Canberra reports a throughput up to 4.5×10^4 kg/h (50 tons/h). The system uses shielded HPGe detectors to perform spectroscopy on the material. However, for specific situations that do not require the high resolution offered by the germanium detectors, large NaI(Tl) detectors can be utilized. An available diverter mechanism can be used to automatically segregate materials in terms of activity.

A similar system, called the Segmented Gate System (SGS), is available as a service from Eberline Services. The SGS is primarily a soil characterization and sorting system, which has been in use for a number of years and has processed more than 176,000 m^3 of soil. The system consists of a combination of conveyor systems, radiation detectors, and computer controls that remove contaminated soil from a moving feed supply on a conveyor belt. The system uses two sets of gamma radiation detector arrays housed in shielded enclosures. The two sets of detectors allow for the radiation measurement of two gamma energy regions of interest. The thin detector array uses 0.160-cm thick NaI(Tl) detectors and incorporates a 1.9-cm thick lead shield that is fully encased in steel. T he thick detector array uses 5-cm thick NaI(Tl) detectors and is housed in a similar shield. Eberline Services reports a throughput of approximately 3.4×10^4 kg/hr (38 tons/hr). While the majority of applications have measured gamma radiation from radionuclides such as cesium-137 (^{137}Cs), cobalt-60 (^{60}Co), and americium-241 (^{241}Am), the SGS has been equipped with beta detectors to assay strontium/yttrium-90 ($^{90}Sr(^{90}Y)$).

Large-Area Surface Contamination Monitors

Conventional survey instruments, such as those described previously (e.g., gas proportion counters, GM tubes, and ZnS scintillators), are very efficient at measuring surface contamination on small items. However, with a relatively small active area (100 cm^2 for a gas proportional counter, 20 cm^2 for G-M pancake probes and 75 cm^2 for some ZnS scintillators), these devices are rather inefficient at scanning large objects such as walls and floors. This section addresses the natural extension of these devices for the measurement of contamination on large areas. These large-area surface contamination monitors have active areas that exceed 1,000 cm^2 and are ideally suited for scanning large, flat areas such as walls, floors, and soil. The simplest systems mount conventional survey instruments, such as gas proportional counters with rate meters, on a mobile platform. More sophisticated systems utilize position sensitive gas proportional counters and/or fiberoptic sensors, and can perform data logging and mapping.

Commercial Systems

Several companies market systems that detect contamination on floors. The Ludlum Model 239-1F floor monitor represents one of the simplest systems available. This modular system features a 16 cm x 47 cm gas flow proportional counter that can be mated to any one of three survey meters, one of which is a data logger. The single-handled, two-wheeled cart can accommodate the rate meter and a Matheson size 2 or Linde Q bottle for the counting gas. The FM-300 floor monitor series, manufactured by Aptec-NRC, is also a modular floor monitor system. The basic unit features two large, sealed proportional counters. The detectors have an active area of 504 cm^2 and a sensitivity of 42–83 Bq (2,500–5,000 dpm) for ^{60}Co in normal background. The model FM-302 system includes the battery powered omniTrack rate meter. While the omniTrack rate meter does not currently do data logging, the system is being modified to support this feature.

Thermo Eberline makes the FCM-4, which is an integrated system that uses four 15.2 cm x 20.3 cm ZnS(Ag) scintillators. The system, which comes with a computer to allow data logging, is similar to the Aptec-NRC system in terms of its form; it has four wheels and a tubular handle. Thermo Eberline reports a sensitivity of 8.3 Bq (500 dpm) alpha and 33 Bq (2,000 dpm) beta from ^{137}Cs.

Shonka Research Associates Inc. produces the Surface Contamination Monitor and Survey Information Management System (SCM/SIMS). This sophisticated system features a position-sensitive gas proportional counter mounted on a motor-driven cart. The position-sensitive gas proportional counter uses a multi-wire electrode configuration to detect the position of the activity within the active volume. The width of the proportional counter used with the SCM/SIMS is variable, typically from 0.5 to 5 m. Also, the system can be equipped with a variety of sensors to facilitate the detection of both beta/gamma and alpha radiation fields.

The SIMS part of the system includes a video camera and a series of software programs that processes and analyzes the collected survey strip data. The SIMS records both the intensity and location of the radioactivity in an electronic database and mapping software. STITCHER© is a program that takes the individual survey strips and positions them relative to each other and the survey area. Once the strips are positioned, the VISUSPECT program projects and averages the data from the strips onto standard 100-cm^2 areas typical of manual surveys. The data from this array can then be visually inspected using various image-processing algorithms, or it can be used to generate a data report that documents the average contamination present in each 1-m^2 area and the maximum contamination level in a given 100 cm^2 within this 1-m^2 area. Note that 100 cm^2 is the active area of most hand-held probes that would be used for scanning applications. More information on the SCM/SIMS and its detection principles can be found in papers and reports by Shonka (1992, 1995, 1996a, and 1996b) and U.S. DOE (1998b). The SCM/SIMS is not for sale. It is included as a service that is provided by Millennium Service.

BetaScint Inc. has designed a detector that uses a fiberoptic sensor to determine the concentration of ^{90}Sr or ^{238}U in soil. The device, called BetaScint™ , uses a layered configuration of scintillating fibers to detect betas from the radioactive decay of Yttrium-90 and Protactinium-234m (the equilibrium progeny of ^{90}Sr and ^{238}U, respectively). It can also discriminate between high- and low-energy betas and between beta and gamma-rays. To achieve this discrimination, it exploits the penetrating properties of betas and gamma rays. The detector measures 1.5 m x 0.35 m x 0.8 m and weighs approximately 20 kg. The monitor can be placed on or above contaminated soil or surfaces. Once the active window of the BetaScint™ sensor is placed over a sample of dry homogeneous soil, the beta particles excite electrons in a plastic fiber doped with fluorescent compounds in the layers of the sensor. The plastic fibers scintillate when the fluorescent molecules lose energy and return to their ground state. Scintillations in the plastic fibers are counted by photon detectors to determine the activity of the soil sample. The unit can be calibrated by exposing it to a soil with a known quantity of ^{90}Sr (or ^{238}U).

The BetaScint™ is specifically designed to measure ^{90}Sr and ^{238}U, but cannot distinguish between beta radiation from ^{90}Sr and ^{238}U (it measures the sum of ^{90}Sr and ^{238}U). However, except in rare cases, ^{90}Sr and ^{238}U usually do not occur together because the source of ^{90}Sr contamination is a fission product, while ^{238}U is associated with the fuel or fuel element (that is, it is not a fission product). If other radionuclides are known (or suspected) to be present, data from other measurement techniques must be utilized. For example, high levels of ^{137}Cs in the soil will produce interference (the decay of ^{137}Cs emits two betas). Demonstrations have shown that ^{137}Cs interference will not become an issue, unless its concentration exceeds that of ^{90}Sr by many orders of magnitude. When ^{137}Cs and ^{90}Sr levels are comparable and less than 3.7 Bq/g (100 pCi/g) (i.e., typical soil remediation conditions), the ^{137}Cs contribution to the sensor background is negligible. More information on the BetaScint™ can be found in papers and reports by Schilk *et al.* (1994a, 1994b, 1995a, and 1995b) and U.S. DOE (1998c).

In Situ Gamma-Ray Spectrometry

In situ gamma spectroscopy is a measurement technique that uses HPGe detectors to measure gamma-ray fluence to quantify radionuclide inventories for a variety of source geometries. The technique has been used most often to measure activity in surface soil with real-time or near-real-time results. The approach has been commercialized by selling detectors that are calibrated for a specific application or source geometry.

Commercial Systems

The *In Situ* Object Counting System (ISOCS) from Canberra Industries, Inc., uses a computational process to identify and quantify radioactivity in a variety of geometrical arrangements. While the system can be calibrated using traditional prepared radioactive sources, the real advantage of the ISOCS software is the ability to calculate efficiencies by entering parameters such as the elemental composition, density, standoff distance, and physical dimensions. By using the supplied geometry templates (for example, boxes, cylinders, pipes, circular planes, rectangular planes, spheres, and wells such as Marinelli beakers), a calibration curve is generated that can be applied to multiple collected spectra. A more detailed review of this system may be found in Kasper (1999) and Kalb *et al.* (2000). The M-1 Gamma Spectroscopy System for *In Situ* Activity Measurements is an *in situ* system, manufactured by PerkinElmer. This system uses the DOE Environmental Measurement Laboratory characterization methodology. It is targeted for undisturbed soil measurements in environmental restoration projects, assessment of radionuclides deposited during emergencies, and routine environmental monitoring. PerkinElmer also produces an *in situ* system that consists of the ISOTOPICS software program; a mobile assay system, which includes a detector, collimator, and MCA called ISO-CART; and an HPGe detector. Of these components, ISOTOPICS and ISO-CART are intended to be used together for the nondestructive analysis of drums. The M-1 system and ISOCS participated in an intercomparison exercise, which evaluated the bias of the systems for measuring activity in surface soil. A discussion of the intercomparison and the results may be found in Miller *et al.* (1998).

Capital Cost: $$$

Eberline Services offers *in situ* spectrometry as a service. The service features a proprietary system, called Spectral Nondestructive Assay Platform (SNAP), which uses HPGe detectors to measure a variety of waste packages, including B-25 boxes, "D" boxes, glove boxes, and 208-liter (55-gallon) drums. Eberline Services claims that its approach enables the system to map contamination levels and locations with near-real-time results.

In Toto Monitors

In toto monitors covers a range of instruments that measure or assay objects *in toto*. The systems consist of a counting chamber, an array of detectors, and an electronics package. There is a wide variety of volume counters ranging from small item monitors to box counters and waste assay systems. A typical small item monitor has a counting chamber of about 0.08 m^3. Box counters and waste assay systems are designed to measure specific waste containers such B-25 boxes, which have a volume of 2.55 m^3. Since box counters and waste assay systems are designed to measure a specific type of waste (transuranic waste) utilizing advanced measurement methods, they are addressed in Section B.4. In general, volume counters use a variety of detectors such as gas proportional counters, plastic scintillators, and NaI(Tl) scintillators. These detectors are shielded (to reduce background) and surround the counting chamber to maximize the geometrical efficiency. Calibrations are performed with standard packages or suitable geometries containing sources of known activity.

Commercial Systems

BNFL Instruments has developed the IonSens® 208 Large Item Monitor. The system is called the "Large Item Monitor" because it has a chamber volume of nearly 1 cubic meter. The IonSens® 208 determines the total alpha activity on objects by measuring the specific activity (number of ion pairs produced per unit path length by an ionizing particle) created by the alpha particles as they interact with the air surrounding the item being assayed. Filtered air passes over the object and is drawn to a detector which measurements the ionization. The system consists of two modules, an air inlet module and a measurement module. The air inlet module filters ambient air to remove particulates and dust before entering the measurement module. The measurement module is an airtight 1 m × 1 m × 0.8 m cavity in which the items to be measured are placed. BNFL reports a limit of detection of 10–15 Bq (600–900 dpm) for a 100-second count time.

Thermo Eberline produces a series of small item/tool monitors, including the TCM-2, WCM-10, LRAD-1, and GTM. The TCM-2 is designed to detect hot particles and low-level contamination distributed on tools. The system uses an array of 6 gas flow proportional detectors, each of which is electrically divided, resulting in 12 channels or counting zones. The detector geometry is designed to minimize dead zones and maximize sensitivity. The system features "sumzones," which represent the combination of detector counts from any two channels. The sumzones are important for detecting distributed activity. This system has 30 sumzones and an adjustable interior volume. Thermo Eberline reports a sensitivity of 0.83 Bq/cm^2 (5,000 dpm/100 cm^2) for beta contamination with an approximate counting time of 10 seconds. The WCM-10 is intended for waste and uses six large area plastic scintillators. The counting chamber is heavily shielded and lined with polished stainless steel to facilitate decontamination. Thermo Eberline reports a sensitivity of approximately 74 Bq (2 nCi) of Co-60. An option to include a weight sensor outputs reported activity in activity per unit mass.

The LRAD-1 uses the long-range alpha detection technique (see the next section for a description) to measure alpha contamination on surfaces. The detection principle is similar to BNFL IonSens®, which detects the ions produced by alpha particles. Thermo Eberline reports a sensitivity of approximately 5 Bq (300 dpm) for objects that fit in the counting chamber, which has a volume of 0.08 m^3. The GTM is another tool monitor that uses a 5-cm thick plastic scintillator on four or six sides of the counting chamber. Just as with the TCM-2, the system utilizes a signal from the individual detectors as well as summed signals from any two detectors to measure "hotspots" as well as uniformly distributed sources.

The G35-90 Package Monitor, manufactured by Canberra, is designed to detect the concentration and type of gamma-emitting radionuclides within small packages. Unlike the other systems, in which the counting chambers can be closed, the G35-90 has a 90-liter open-ended rotating drum for a counting chamber. The system is mobile and computer-controlled, and utilizes two shield NaI(Tl) scintillators. The system comes calibrated from the factory. No MDC or sensitivity data has been reported for the system. Finally, NE Technology produces the SAM 11 Small Articles Monitor. Like some of the other systems described in this section, it uses an array of shielded plastic scintillators to detect beta/gamma radiation. This system has a fairly large counting chamber volume, approximately 0.5 m^3.

Pipes

In addition to building debris, D&D activities have produced, and will continue to produce, a considerable amount of ductwork and piping. Because of their interior surface, long lengths of small-diameter ductwork and piping are largely inaccessible to conventional survey instrumentation. Manufacturers have, therefore, developed specialized instrumentation to survey the exterior and interior of piping.

Commercial Systems

The IonSens® Alpha Pipe Monitor, available from BNFL, is a modular system that measures total alpha contamination on metallic pipe work and/or scaffolding poles. It can accommodate lengths up to 6 m and diameters up to 15 cm. The detection method and basic operation is very similar to the IonSens® 208. The IonSens® Alpha Pipe Monitor consists of three basic modules, including the air inlet module, measurement module, and detection head module. The measurement modules are airtight and can be configured to accept 6-m lengths by joining three measurement modules. As with the other IonSens® systems, the detection head module contains the ion detector as well as a HEPA filter, fan, data processing electronics, iris seal, and PC. BNFL claims a limit of detection of 15 Bq (900 dpm) for a 300-second count time. The detection module has a small standardized source that is used to monitor performance.

The Pipe Explorer™, available through Science and Engineering Associates Inc., is a pipe characterization system that employs an airtight membrane deployed from a canister with air pressure to line the interiors of pipes and to carry a tether to which detectors are attached. As the membrane deploys, detectors are towed along inside the membrane while measurement data is collected. This system consists of three primary components, including (1) the deployment canister, which holds the membrane and detector assembly as well as the necessary transducers and sensors for the operation of the system, (2) the data acquisition computer, which logs and correlates information from the deployment and detector systems, and (3) the instrumentation and control box, which is used to control the deployment of the membrane and survey tools. The heart of the system is an airtight membrane that is initially spooled inside the deployment canister. Air pressure on the membrane causes it to be pulled from the spool, and deployed into the pipe. A characterization tool (such as a radiation detector) is attached to the end of the membrane and is towed into the pipe as the membrane unwinds. Because the membrane and detector are tethered to the spooler inside the canister, they can be wound back into the canister. The detector can, thus, be moved freely through the pipe while its output and position are continuously recorded. The Pipe Explorer™ system can be used to tow any detector that is compact enough to fit into a pipe. The tether has two coaxial cables available and six single conductor cables, which are used to provide power and control to the characterization tools. To measure alpha particles with the Pipe Explorer™, the membrane material itself must be an integral part of the detection system. An effective solution is to make the membrane material a scintillator, and then tow a photodetector through the pipe to detect the scintillation events occurring in the membrane. This is the approach adopted for the alpha measurement capability, which is referred to as the Alpha Explorer™ system.

The Pipe Explorer™ system has been laboratory-tested and tested at a number of DOE locations, including Idaho National Engineering and Environmental Laboratory and Argonne National Laboratory. More information on the Pipe Explorer™ System is provided in published reports (Matalucci *et al.* 1995a; Cremers *et al.* 1994, 1995, 1996, and 1997; Cremers and Kendrick 1998; and U.S. DOE 1996b).

The Pipe Crawler®, developed by Radiological Services, Inc., is a manually deployed pipe inspection system that consists of a crawler, mounted with a 360° array of thin GM probes connected by cable to an external data processing and storage system. A family of crawlers is used to accommodate various piping sizes. The dimensions of a given crawler must closely match the size of pipe to be surveyed; this ensures the proper counting geometry (the detector surface must be within about 1 cm of the surface), which is afforded by a spring-loaded wheel suspension system. Each crawler is custom made, employing commercially available GM tubes. The size and shape of the available GM tubes strongly influence the configuration and design of a given crawler. The smaller crawlers for pipes with diameters less than 20.3 cm are manually deployed using flexible fiberglass rods attached to either end. The rods are similar to those used by plumbers. The larger crawlers (for 20.3-cm diameter and larger pipes) employ pneumatically operated positioning systems. It must be noted that the Pipe Crawler® is utilized by Radiological Services, Inc. exclusively as a part of a service they provide to customers and, as such, it is not for sale.

Subsurface

While *in situ* spectrometry provides a noninvasive approach to surface soil investigation, the subsurface remains intractable to such techniques. Current developments in instrumentation seek to reduce the burden of obtaining subsurface data. This basically involves using small detectors that can be pushed through the soil and are capable of real-time results. Because of the expense associated with the sampling equipment, subsurface measurements are typically provided as a service.

One system related to subsurface sampling is the cone penetrometer, which consists of a 2–4 × 10^5 kg (20- to 40-ton) truck equipped with hydraulic rams to push steel cones, one section at a time, into the ground. Penetration rates can be as high as 5.5 m/hr (180 ft/hr), but are typically 1.2 m/hr (40 ft/hr) to 1.5 m/hr (50 ft/hr). Compared to traditional drilling methods, cone penetrometer techniques are less costly, allow less-intrusive sampling and analysis, do not result in contaminated soils being brought to the surface, and minimize worker exposure to potential industrial and chemical hazards. Although cone penetrometer techniques have existed for many years, most earlier efforts focused on oil exploration and construction engineering. Only recently has the technique been applied in environmental characterization and monitoring, with resulting development of many sampling devices and sensors for use with the cone penetrometer. Applied Research Associates Inc. is a research and engineering company that provide subsurface sampling using a cone penetrometer.

A spectral gamma probe, developed for DOE by the U.S. Army Corps of Engineers Waterways Experiment Station, was evaluated and demonstrated under field push (a push is when the penetrometer is driven into the ground) conditions at the DOE Savannah River Site in 1997. The probe consists of a 2.5 cm x 7.6 cm NaI(Tl) scintillation crystal, a photomultiplier tube, a temperature sensor, and a custom designed preamplifier. The temperature monitor is used to track temperature changes, which can affect the performance of the spectrometer. The probe is driven into the subsurface using a cone penetrometer truck. During a field evaluation, nine pushes were made at three locations, and the gamma probe was stopped at 7.6-cm (3-in) to 30.5-cm (12-in) intervals for counting during each push. Results of the gamma probe measurements were compared with results of laboratory analysis of surrounding soils. Where the sites were primarily contaminated with ^{137}Cs with little beta activity, gamma probe results corresponded well with laboratory analysis results. However, the gamma probe experienced interference from the high level of beta activity found at some sites. In general, the lower limit of detection for ^{137}Cs was found to be in the range of 0.3–0.5 Bq/g (8–11 pCi/g).

To minimize the deleterious effect caused by a high level of beta activity, Sentor Technologies, Inc. is developing a high-pressure xenon spectrometer device for use with the cone penetrometer. Three prototype devices have been built and tested in the laboratory; however, they are not commercially available.

Commercially available radiation detection systems for subsurface measurements include HPGe detectors that have small diameter endcaps and dewars, typically about 7 cm, that can be lowered into boreholes. These detectors are available from PerkinElmer.

Portal Monitors

Portal monitors cover a broad range of instrumentation reflecting a wide range of applications. For purposes of this discussion, a portal monitor is an instrument that detects radioactivity as it passes through a portal, which is typically an access point to a controlled area or checkpoint through which people, vehicles, equipment, and waste pass. Just as with many of the other systems previously discussed, these systems use large detectors to improve sensitivity. Most systems use plastic scintillators because they are rugged, inexpensive, and can be made with a large surface area. Count or integration times are very short (typically just a few seconds). The detectors are usually part of a structure which surrounds the portal on one, two or three sides. Although not strictly a portal monitor, plastic scintillators can also be attached to the base frame of grapples[10] to detect radioactivity in scrap metal. These devices have a clear advantage over portal monitors because the scintillator is in contact with the metal and remains in contact for as long as it takes to grab and move it, which could be several minutes. Like portal monitors, they are gross radiation detectors and do not provide quantitative information (e.g., activity per unit mass); they usually signal the operator when a preset threshold has been exceeded.

Commerical Systems

A large number of portal monitoring systems are available from several manufacturers. This section briefly mentions a few systems. For monitoring small waste items as they pass through doorways, Ludlum makes a series (3530/3532/3534) of monitors that use NaI(Tl) scintillators. Models 3530 and 3534 use two shielded 7.6 cm x 2.5 cm NaI(Tl) detectors, while Model 3534 uses four detectors. These detectors are mounted on opposite sides of a doorway or opening through which waste may pass. For larger waste items that are transported by vehicles, Ludlum makes Model 3500-1000WM, which utilizes two 7,866-cm^3 shielded plastic scintillation detectors. Exploranium is very active in the area of detecting radioactivity in scrap metal. They have a series of large portal monitors that detect radioactivity transported by vehicles, including railcars. These systems also use large plastic scintillators mounted to large structures.

One portal monitor of note comes from Constellation Technology Corporation. They have developed a mobile system, known as the HPXe-1000, that performs spectroscopy. The unique feature of this system is the fact that it uses high-purity xenon gas (HPXe). The use of HPXe for gamma-ray spectroscopy is covered in the section on detector materials (see Section B.4). Constellation reports a resolution of 3-percent FWHM at 662 keV for a detector that has a linear dimension of 1 m and a mass of almost 2 kg. The primary application for this system is the detection of special nuclear material for treaty verification.

[10]Grapples are pneumatic devices with "fingers" or tines that are used to pick up and move scrap metal.

Rad/Comm Systems makes a grapple mounted detectors called the Cricket. The Cricket consists of a 30 cm x 30 cm x 10cm scintillator mounted inside the top of the grapple. The system also has a protective shield, battery pack, and controller. Detectable source strengths for scrap densities of 0.5, 0.75, and 1.0 g/cm^3 range from 30–100 kBq (0.03–2.7 mCi) for ^{60}Co, 180–1,000 kBq (4.9 –27 mCi) for Cs-137, and 80–250 kBq (2.1–6.8 mCi) for ^{226}Ra (de Beer *et al.*, 1999).

B.4 Advanced Radiation Detection Systems

Advancements in radiation detection instrumentation have resulted from developments in material science, advances in electronics, and software. This trend shows no sign of slowing down and will continue to be the driving force behind the innovations in radiation monitoring instruments.

Detector Materials

One of the most important properties of a material that makes it a good radiation detector (and spectrometer) is its ability to absorb radiation energy. The property of a material to absorb radiation energy is known as the stopping power, which is defined as an average rate of energy loss of a particle per unit thickness of a material or per unit mass of material traversed. The higher the stopping power, the better the detector material. Stopping or absorbing the energy of charged particles is not an issue, but absorbing high-energy photons is. In general, high-density materials with large atomic numbers (Z) are ideally suited to absorb high-energy photons. Once a material has absorbed the radiation energy, it must be converted to information carriers. This conversion is accomplished either by producing ions as in the case of gas-filled detectors, electron-hole pairs as in the case of solid-state semiconductors, or photoelectrons as in the case of scintillators. A detector must be able to produce these information carriers efficiently; that is, with as little loss in energy as possible. The energy that is required to produce information carriers (ions, electron-hole pairs, photoelectrons) ranges from a few eV to about 100 eV. In general, the lower the better, in terms of the resolution for a spectrometer.

In the case of solid-state semiconductor detectors, a rather large bias voltage (> 1,000 volts) is applied to the crystal. T his bias voltage creates a depleted region where electron-hole pairs are created when radiation energy is absorbed. The electrons and holes are swept from the depleted region and are collected to create a charge pulse. A good semiconductor material must have a high resistivity in order to prevent the collection of unwanted current, sometimes called leakage current, in the presence of a high bias voltage. The resistivity is linked to energy separating the valence and conduction bands, the so-called bandgap. The larger or wider the bandgap the greater the resistivity. If the bandgap is wide enough, the leakage current becomes low enough to permit room temperature operation.

When describing the properties of a solid-state semiconducting detector material, the issues of purity and crystal defects are important. A process known as charge trapping occurs when charge carriers (electron and holes) recombine in the crystal lattice. This occurs for a number of reasons, but it is often traced to a lack of purity and crystal defects. The reduction in charge collection attributable to trapping reduces the size of the charge pulse and, therefore, reduces the resolution and efficiency of the detector. However, a new technique, which uses microwave photons instead of electrons as the information carriers, avoids some of the problems associated with charge collection.

Cadmium Telluride and Cadmium Zinc Telluride

A radiation spectrometer that operates at high (i.e., room) temperature has obvious advantages over conventional cryogenic spectrometers for applications where the system has to operate in an unattended mode or where liquid nitrogen (or a sufficient source of power) is difficult to obtain or too cumbersome to use. In recent years, the technology of radiation detectors that operate at room temperature has greatly improved, as a result of the ability to grow a number of semiconductor materials. Cadmium zinc telluride (CZT) and cadmium telluride (CdTe) are two such semiconductor materials with the properties required by a high-performance spectrometer. CdTe and CZT have high atomic numbers; however, a chief concern related to the use of alloy materials (including CdTe and CZT) for detector applications is degradation of detector resolution as a result of detector matrix heterogeneity. The most significant drawback of CZT is the insufficient supply of high-quality crystals for spectroscopic systems. This circumstance results from both uniformity issues and carrier transport properties.

Other Detector Materials

While CdTe and CZT are currently receiving most of the attention and focus as room temperature detectors, several other materials are being researched for this function. The following paragraphs briefly summarize the current development of four such materials, namely xenon (gaseous and liquid), mercuric iodide, lead iodide, and diamond.

Xenon

The properties of xenon that make it desirable as a detector material are that the energy required to generate an ion pair is 21.9 eV (which is smaller that argon and neon), and that its Fano factor is about 0.17. This means, for example, that the 662-keV gamma-ray line from ^{137}Cs has an energy resolution of 0.56-percent FWHM in xenon. This excellent intrinsic resolution, combined with a high atomic number (Z=54), shows that xenon is a suitable medium for high-resolution gamma-ray detection. Tepper *et al.* (1998) report on a cylindrical ionization chamber filled with highly purified xenon that has an energy resolution of 1.8 percent at 662 keV.

Xenon does exhibit some nonlinear behavior in its density when its pressure is varied near its critical point[11], which corresponds to 10^6 dynes/cm^2 (58 bar), ρ = 1.1 g/cm^3 and 17 °C. In general, at room temperature, xenon exhibits very little increase in pressure, for significant increases in density. Nonetheless, the sensitivity of the pressure to temperature must be considered when designing a detector using xenon (Mahler *et al.*, 1996). A portable gamma-ray system using xenon gas will be discussed later.

Liquid Xenon

Liquid xenon (LXe) has been used as a detection medium for an imaging telescope (Aprile *et al.*, 2000). LXe is an ideal material for high-energy gamma-ray detection because of its high density (3 g/cm^3) and high atomic number (Z=54). The ionization and excitation of xenon atoms, which result from these interactions produce a large number of electron-ion pairs (6,400 e-/ 100 keV, whereas gas proportional counters yield ~4,000 e-/ 100 keV) and a similar number of scintillation photons. However, when compared to gaseous xenon, the resolution of LXe (approximately 6 percent at 1 MeV) is somewhat poor.

[11] The critical point is where two phases (e g., liquid and gas) have exactly the same density and are indistinguishable.

Mercuric Iodide

Red mercuric iodide (α-HgI_2) has been researched for almost three decades for use as a room temperature radiation detector material. Its high atomic number and wide bandgap make α-HgI_2 particularly well-suited for fabrication of room temperature compact spectrometers. It has been used to produce some of the highest resolution room temperature x-ray and gamma-ray detectors. However, these positive properties are balanced by several negative properties, including the fact that the material has a relatively high vapor pressure at room temperature, and the iodine is generally preferentially sublimed at a faster rate, yielding a mercury rich surface. Additionally, the material is mechanically very soft, and delaminates easily at the iodine layers (James 1996, Van Scyoc 1996).

A novel room temperature, high-resolution HgI_2 spectrometer that has the needed performance and yield of high-quality detectors, with minimal support and maintenance requirements, has been developed (Van Scyoc, 1997). In particular, the reduction of charge trapping defects has been achieved by eliminating the material properties most degrading to performance. With these improvements, HgI_2 devices with high-energy resolution over the range of x-ray and gamma-ray photon energies of 1 keV to 1 MeV can be readily produced. Figure B-1 shows the dramatic difference between the ^{241}Am spectrum produced with a conventional HgI_2 detector on the left, and the same spectrum produced with the new HgI_2. Notice that the peaks on the right spectrum are much sharper and more symmetric. Also notice that while low-energy tailing is still visible, it is at a much lower level, which allows a Compton scattering peak to become visible.

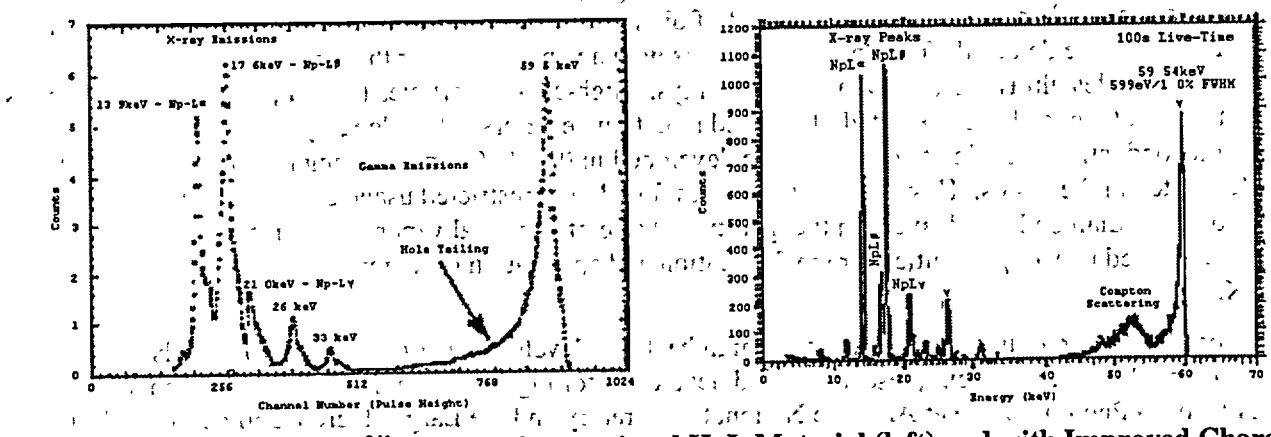

Figure B-1: Spectrum of ^{241}Am with Conventional HgI_2 Material (left) and with Improved Charge Transport HgI_2 (right) (Van Scyoc, 1997)

Lead Iodide

Lead iodide (PbI_2) was first introduced in the 1970s as a candidate material for nuclear radiation spectrometry having an extraordinarily high efficiency for gamma rays. In addition, the wide bandgap of this material makes possible the growth of extremely high resistivity material. Lead iodide has a high Z and a high density (6.2 g/cm^3), which means a high stopping power. Thus, room temperature, and even above room temperature, operation of gamma-ray spectrometers fabricated in this material is feasible. Also, the growing of single crystals of lead iodide is simpler in comparison to mercuric iodide or CZT growth. High-purity (99.9999 percent pure) PbI_2 is commercially available and further purification (which is crucially important for detector grade material) is accomplished by zone refining for 100 passes[12]. The primary difference between recently demonstrated lead iodide detectors and those fabricated earlier appears to be the degree of crystal purity. However, one of the obstacles in dealing with PbI_2 is its poor mechanical behavior resulting from its layered structure.

With the appropriate processing techniques, it has been found that detectors fabricated from high-purity PbI_2 crystal exhibit significant improvement in performance, compared to those produced from low-purity crystals. However, problems still exist in lead iodide because of the low charge carrier collection efficiency, which is probably caused by additional impurities or defects incorporated during crystal growth and detector fabrication processes (Hermon, 1997).

Diamond

For application to radiation detectors, the wide bandgap, radiation hardness, optical transparency, and low atomic number are important properties of diamond. Any radiation that generates free carriers in diamond can be detected. This includes photons with an energy greater than the bandgap of 5.5 eV, which includes ultraviolet, x-ray, and gamma rays. High-energy particles (e.g., alpha particles, electrons, neutrons, etc.)can also be detected. Diamond radiation detectors have a lengthy history. Photoconductive ultraviolet detectors were developed in the 1920s and ionizing radiation detectors were fabricated in the 1940s. However, these devices found only restricted usage because of the limitations of geological diamonds. Advances in the quality and size of chemical vapor deposition (CVD) diamonds have created new opportunities for the fabrication and application of diamond radiation detectors (Kania, 1997).

Because of their ability to withstand very high heat flux levels and very high radiation levels, CVD diamond detectors are being researched and developed for high-energy physics devices, such as the Advanced Photon Source at Argonne National Laboratory and the Large Hadron Collider at the European Laboratory for Particle Physics (Liu *et al.* 1996, Hrubec *et al.* 1998; Friedl *et al.* 1998).

[12]Some solids can be purified by a process known as zone-refining. The impure solid is packed tightly in a glass tube, and the tube is lowered slowly through a heating coil that melts the solid As the melted solid cools slowly in the region of the tube below the heating loop, pure crystals separate out, leaving most of the impurities behind in the molten zone. This process can be repeated as often as necessary to achieve the desired purity of the recrystallized solid.

The detection of radioactive sources in scrap metal presents a harsh environment that excludes many traditional detector materials. The lifting magnets used in a scrap yard would be a favorable location to detect potentially contaminated metal entering the yard. Unfortunately, the presence of magnetic fields and mechanical vibration prohibits the use of traditional photomultiplier tubes with scintillation detectors. Moreover, the high temperatures restrict the use of solid-state detectors such as Ge or Si. Manfredi and Millaud (2000) have proposed that diamond be used as a detector material for contamination in scrap metal. Since diamond has a low Z, it is unsuitable for the detection of medium to high energy gamma rays. Manfredi and Millaud have proposed the development of a conversion-type detector that would be made of alternating layers of converter material and detectors. High-energy photons would strike the conversion material (tungsten has been suggested) and produce secondary radiation that could be detected in the diamond.

Software

The role of software in radiation detection is to facilitate the analysis and interpretation of information that detectors provide. Numerous analytical techniques have been developed, which utilize and optimize spectrometric information. For example, information in the form of a detector response, which can be calculated using radiation transport codes, can be combined with spectral information (e.g., count rates associated with radiation energy) to provide spatial distribution of radioactivity. Still other techniques improve detector sensitivity by optimizing spectrometric information. Software aids in the implementation of these analytical techniques, which can improve and extend the abilities of radiation detectors.

Gamma Detector Response and Analysis Software

The Gamma Detector Response and Analysis Software (GADRAS) is a collection of programs used to plot and analyze gamma-ray spectra. In contrast to most spectral analysis programs that find radionuclide concentrations by determining the areas of characteristic photopeaks and ignoring the continuum,GADRAS uses linear regression to fit the entire spectrum with a combination of computed spectral templates. Spectra are computed using a semi-empirical response function that was originally developed for use with sodium iodide detectors (Mitchell, 1986), and was expanded to accommodate other types of scintillators plus semiconductor detectors such as high-purity germanium. Subsequent developments that have been incorporated into the current response function enable computation of spectra based only on the detector material and dimensions. This capability can be applied to evaluation of detector designs prior to fabrication. GADRAS was developed at Sandia National Laboratory and is used primarily for safeguard applications (Mitchell, 1992a). It has been used to analyze air filter samples for the Remote Atmospheric Monitoring Project (Mitchell, 1987 and 1992b). Figure B-2 shows a typical spectrum analysis of an air filter sample using a modified form of GADRAS called RAMP-PC1.

GADRAS-PC1 is a version of the software that has been written specifically for use on IBM-compatible personal computers. Routines included in GADRAS-PC1 enable a calibration of the response function parameters by fitting computed spectra for a set of calibration sources to measured spectra. The template set used in the analysis of unknown sources can include combinations of the 96 isotopic sources in the radionuclide library, fluorescence x-rays, or a user-defined library of source templates. The GADRAS-PC1 response function has been used to characterize a variety of sodium iodide, cesium iodide, bismuth germanate, and plastic scintillators plus high purity germanium detectors. GADRAS-PC1 is particularly useful for analysis of spectra recorded by the scintillators because the low resolution can preclude identification of photopeaks for all but the simplest gamma-ray sources. The analysis routine also excels for weak sources or measurements with short counting times because the entire spectrum is utilized, including statistically significant continuum regions.

The GADRAS response function is based on the fundamental interactions of photons with the detector material. The first-order response is derived from the detector material's crosssections for photoelectric absorption, Compton scattering, and pair production. As many as 49 adjustable parameters can be used to include compensation for unusual scattering environments and anticoincidence shields. It is seldom necessary to use more than about 20 parameters, including those associated with the energy calibration and detector resolution. The response function also computes the effects of phenomena that are generally neglected, including: detection of coincident gamma-rays, pileup attributable to high count rates, bremsstrahlung radiation, escape of fluorescence x-rays, and leakage of high-energy electrons from the detector. Note that the response function obtained using GADRAS is not necessarily different from a response function obtained using a radiation transport code such as Monte Carlo N-Particle (MCNP)[13]. GADRAS uses measurements and linear regression to obtain a response function, while a radiation transport code uses a simulation to determine the same quantity.

[13]MCNP is distributed within the United States by the Radiation Safety Information Computational Center (RSICC), formerly the Radiation Shielding Information Center (RSIC), Oak Ridge, Tennessee.

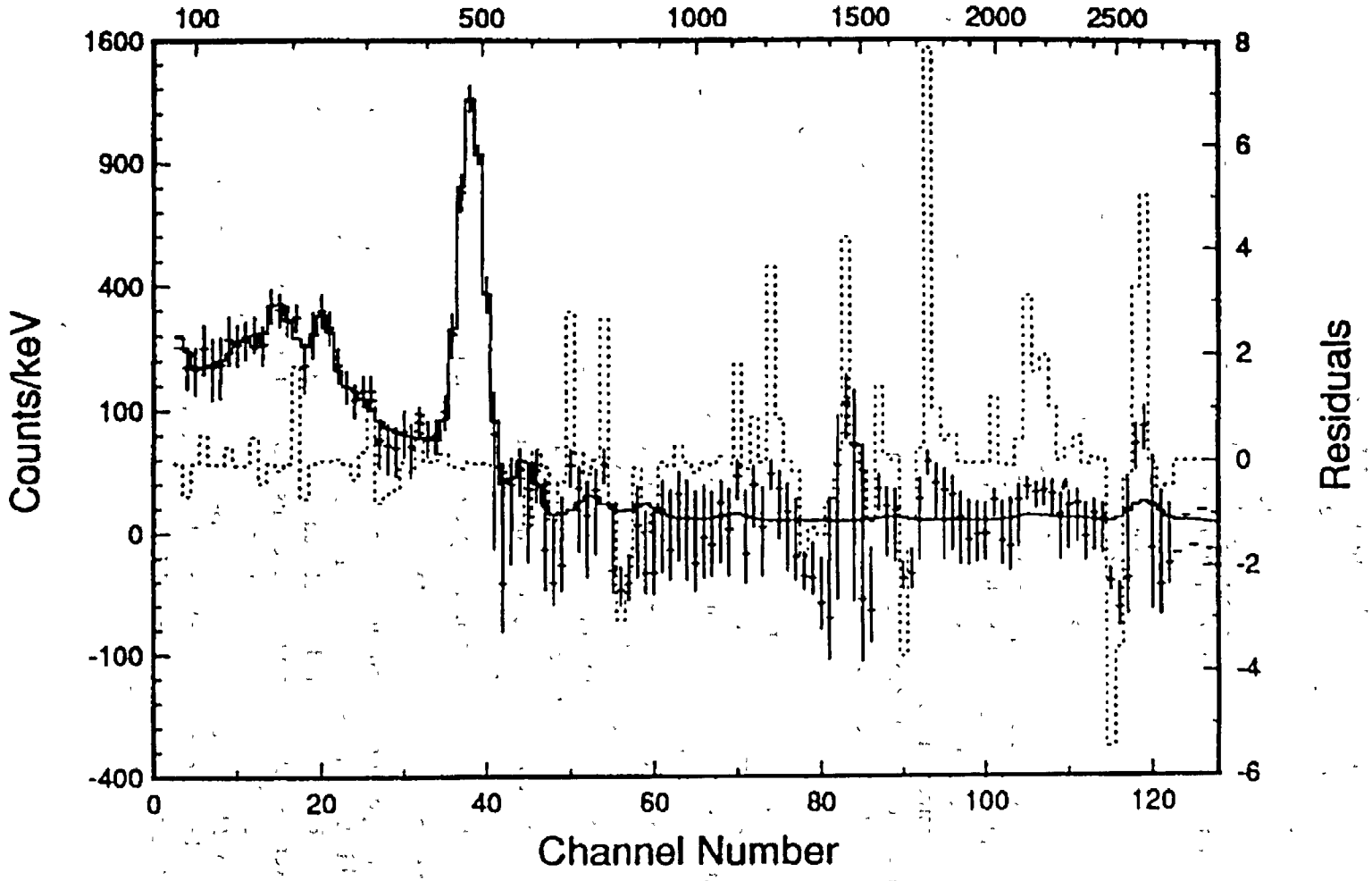

Figure B-2: Analysis of an atmospheric filter sample containing Be-7 using a modified form of GADRAS. The plot shows background subtracted data represented with a 1σ uncertainty. The step histogram gives the compound spectrum for the combination of isotopes including Be-7, ^{212}Pb, Ru-103, and Ce-141 (Mitchell 1992a)

Gamma Penetration Depth Unfolding Algorithm

The Gamma Penetration Depth Unfolding Algorithm (GPDUA) comprises a computer code and measurement technique that uses the penetrating properties of gamma-rays to determine the depth of contamination in materials. The measurement technique uses a typical portable HPGe gamma-ray spectrometer system, consisting of a multichannel analyzer, high-voltage source, laptop computer (with appropriate counting software), and a portable HPGe detector with a collimator. The lead collimator serves two purposes, in that it (1) localizes the field of view, and (2) simplifies the efficiency calculations. It must be noted that the method is applicable to radionuclides that emit at least two gamma-rays, or radionuclides that emit a single gamma ray but have gamma-emitting progeny; parent and progeny must be in secular equilibrium. The peak areas that correspond to the energies of the uncollided gamma-rays are the only information necessary for GPDUA. It is the ratio of the counts in the peak areas that contains the necessary information to determine the depth of contamination. GPDUA uses a point kernel approach and solves an integral equation involving the net counts (from those photons incident on the detector face), the intrinsic efficiency, the distance from the source to the detector, and the depth of penetration. GPDUA solves the equation by iterating on the depth, and the depth that solves the equation is the depth of the contamination. GPDUA has been tested with MCNP and predicts the depth of contamination to within 10 percent of the actual (simulated) depth, regardless of the type of contamination distribution (i.e., point, disk, or linear distribution) (Naessens and Xu, 1999).

Microwave-Based Radiation Detector

As previously noted, room temperature semiconductors suffer from material defects, which limit their potential for high-energy gamma-ray spectrometry. Tepper and Losee (2001) are investigating the feasibility of using microwaves to measure changes in the conductivity of these wide-bandgap materials to determine the energy of the absorbed radiation. The method provides a way of extracting the energy information without having to collect the charge, which has been a problem for these materials. The method of using microwaves to measure the electrical properties of various materials has been used for years. This, however, is the first time that microwaves have been used for gamma-ray spectroscopy. Preliminary results show promise, but the sensitivity must be improved by at least two orders of magnitude before high-resolution gamma-ray using this technique is a reality. Tepper and Losee are confident that the sensitivity can be improved; however, it is unclear whether such a system could ever match the performance of conventional cryogenic spectrometers such as HPGe detectors.

Compressed Xenon Gamma-Ray Spectrometer

A prototype gamma-ray spectrometer utilizing xenon gas at high pressure has been developed at Brookhaven National Laboratory (Smith, 1996). Known as Compressed Xenon Gamma-Ray Spectrometer (COXGARS), it was initially developed for safeguards applications. COXGARS is a portable, battery-powered spectrometer, which functions at ambient temperature with an energy resolution between semiconductor (Ge) and scintillation (NaI(Tl)) spectrometers; Mahler *et al.* (1997) reports an FWHM at 662 keV of 2.5 percent. Figure B-3 shows the internal components of the COXGARS systems, which is capable of prolonged, low-power operation without a requirement for cryogenic fluids or other cooling mechanisms. Table B-2 provides some of the important characteristics of the compressed xenon spectrometer.

Table B-2: Characteristics of COXGARS	
Energy Range	100 keV to ~ 1 MeV
Sensitive Volume	160 cm^3
Sensitive Area	30 cm^2
Energy Resolution @ 662 keV	2.5%
Intrinsic Efficiency @ 200 keV/662 keV	40%/15%
Detector Mass	10 kg
Portable System Mass	Two 20 kg containers
Power Consumption	7W

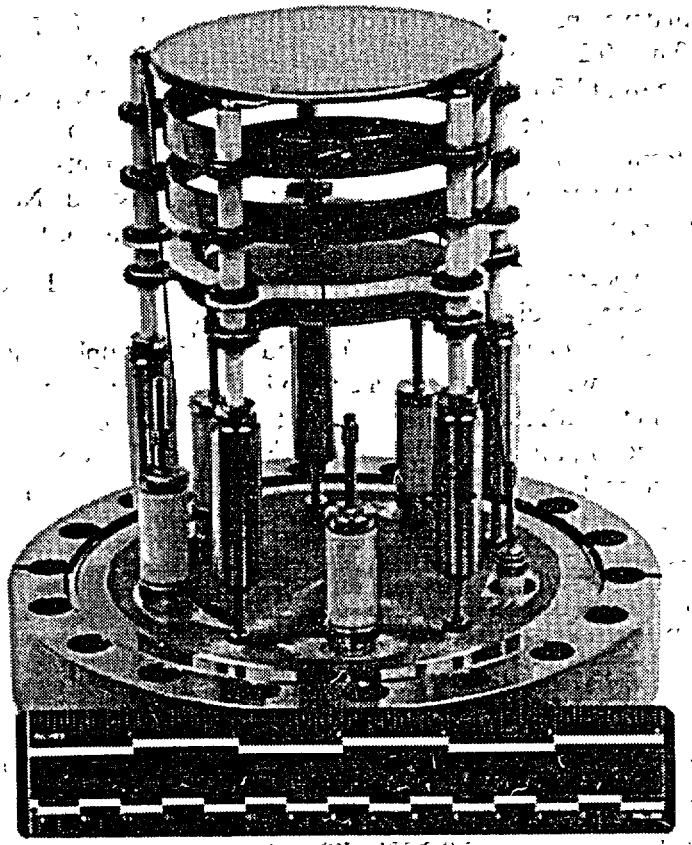

Figure B-3: The internal structure of COXGARS

Static and Dynamic Long-Range Alpha Detector

Static and Dynamic Long-Range Alpha Detector (LRAD) systems are designed to monitor alpha contamination by measuring the number of ions produced by alpha particles as they interact with the air; a typical alpha particle will generate about 150,000 ion pairs. A key feature of the LRAD detection principle is that the ion pairs persist long enough so that ions may be collected on a detection electrode, which is located some tens of centimeters away from an alpha-contaminated surface. The ions may be transported to the electrode either by an air current or an electric field. Both the static and dynamic LRAD surface monitors use an electric field. A more detailed description of the LRAD concept and devices is contained in several reports (MacArthur 1991a, 1991b, 1992a, 1992b, and 1993).

Static LRAD Surface Monitor. In the static LRAD, the ions generated over the surface to be monitored are collected on the detection electrode by a small electric field generating a bias voltage. This flow of ions represents a small current which can be detected by a current meter or recording device. This current is proportional to the total amount of contamination on the surface covered by the enclosure. The detector enclosure serves two purposes, (1) to define the active area of the detector and (2) to prevent externally generated ions from reaching the detector electrode and causing a spurious current.

A static LRAD system developed by Los Alamos National Laboratory (LANL) for measuring surface soil uses a 1.0 m x 1.0 m x 0.2 m box-shaped ion chamber with an open bottom face. A small tractor with the detector on the front lift moves the detector between monitoring positions; it places the detector open face down on the soil. About 15 minutes are required for signals to stabilize after the detector is moved to a new monitoring position. Once signals are stable, the currents are averaged for about 5 minutes. In this current measuring mode, only alpha activity is measured. Note that the LRAD monitor relies on the physical connection between the LRAD enclosure and the surface to be monitored.

Since the LRAD is not a spectrometer, it cannot identify radionuclides and, therefore, interference is a problem. It cannot, for example, distinguish between the alpha activity from naturally occurring alpha-emitting radionuclides such as uranium and thorium, and man-made alpha emitters such as plutonium. It also cannot distinguish between surface alpha contamination and radon gas that emanates from the soil and mixes with air within the LRAD chamber. The static LRAD detection electrode and the surface to be monitored form a capacitor; this is called a capacitive coupling. Any movement of one surface relative to the other changes the detector capacitance. This capacitive coupling causes a small current to flow in the detector, creating an erroneous signal in the detector.

Field tests at various DOE sites have shown that LRAD surface soil monitors (SSMs) are faster and more sensitive than traditional alpha detectors for measuring alpha contamination (Johnson, 1993). However, an evaluation of the LRAD, performed at Savannah River, found several limitations to the application of this technology:

- The signals differed dramatically (factors of 20) above the uncontaminated sample materials. This likely resulted from differences in concentration of naturally occurring alpha emitters, such as uranium and thorium.
- The edge seals used in the prototype sometimes allowed radon in-leakage during the measurement. When this occurs, the LRAD signals do not stabilize.
- Any contact between the LRAD charge collection plate and the ground can result in leakage currents that are large relative to signals from uncontaminated soil. Great care must be taken to monitor soil where grass is growing.

It was concluded that if the LRAD is used to locate alpha contamination and map its distribution, results must be used with caution (Sigg, 1995). Many false-positive indications are likely to be obtained, which could require additional measurements by other independent methods.

Dynamic LRAD Surface Monitor. Some of the limitations discussed above (capacitive coupling and the fact that the detector must be in contact with the surface to be monitored) have been addressed by adding an additional electrode (MacArthur *et al.*, 1998). Externally generated ions can be excluded using an electrostatic electrode. An electric field between the guard electrode and the surface excludes unwanted ions from entering the chamber volume. This guard electrode removes the requirement for physical contact between the enclosure and the surface. The LRAD can be continuously moved relative to the surface to be monitored.

The guard electrode and gridded detector concepts are combined in the large dynamic surface monitor. This detector system can be operated in a scan mode with little or no loss of sensitivity. Movement of the detector relative to the surface includes both "moving-LRAD" applications (e.g., measurements of walls, floors, and soil), as well as "moving-surface" applications (e.g., soil and/or rubble conveyer belt systems). Although the grid on the front of the detection chamber makes it more vulnerable, grid wires as large as 0.5 mm in diameter have been demonstrated, and there is some speculation that larger wires would work as well. The current supplied to the exposed guard electrodes is limited to about a microamp without affecting the operation of the electrode.

Waste Assay Systems

Waste Assay for Non-Radioactive Disposal System (WAND). The WAND system scans low-density waste (mostly paper and plastic). This system is designed to verify that the levels of radioactive contamination (if present) are low enough so that the waste can be disposed of in public landfills. The WAND system was developed to reduce the volume of low-level waste that requires disposal from LANL.

The WAND system consists of a lead-shielded chamber containing six 12.7-cm diameter phoswich detectors. A phoswich detector is a combination of two scintillators (in this case NaI and CsI) optically coupled to a single PM tube. The combination of scintillators rejects background events and separates the full energy x-rays from other signals. The WAND system has a conveyor system that moves a 30.5-cm wide layer of paper through the chamber about 5 cm beneath the detectors and deposits the screened material into a waste bin. Either pre-shredded paper or packets of paper no more than 30 sheets thick, are manually placed on the conveyor belt.

The electronic portion of the WAND system consists of electronic modules (needed to process the signals from the six detectors) and a desktop computer (486/66 PC). The software portion of the system consists of a custom analysis algorithm (written in C++ language), along with the code by which the operator controls the system and produces reports. Each phoswhich detector is equipped with a preamplifier and two electronic nuclear instrument modules (NIMs), which provide the buffering, amplification, and pulse shaping. To preserve the individual signals from each of the 12 detectors while using a single analog-to-digital conversion (ADC) module, a custom multiplexer module was designed to handle the data. With the exception of the multiplexer, the electronics are all commercially available.

While moving the waste material at a speed of 1.27 cm/sec beneath the detector array, the system software performs a series of consecutive 10-second evaluations of the levels of radioactivity seen in each detector. If the count rate in any of the four energy regions of interest (ROIs) meets or exceeds the upper limit of the background, the conveyor belt backs up and does a recount. If excess radioactivity is detected on the recount, the conveyor belt stops and the software identifies the detector and the ROI that had the increased count rate. Additional information on the WAND system may be found in papers and reports by Arnone *et al.* (1998) and Myers (2000).

High-Efficiency Radiation Counter for Low Emission Sensitivity System (HERCULES). The HERCULES system consists of a vertical array of three phoswich scintillation detectors positioned in a shielded detection chamber. Low-density waste is placed in a 30-gallon plastic drum, which rotates on a turntable (12 RPM) approximately 4.0 cm from the detector array. Count times can be varied according to detection sensitivity requirements, but the standard measurement time for most radionuclides is 1,000 seconds. A sliding door on the top of the detection chamber allows for access to waste in the plastic drum. The chamber walls are filled with 2 inches of lead shielding and are lined on the interior with 0.08-cm copper and cadmium sheets[14]. The HERCULES system uses the same electronic components and software packages as the WAND system, which makes the components easily exchangeable. Additional information on the HERCULES system may be found in Myers (2000).

Controleur Automatique de DEchets Faiblement Actifs (CADEFA). The CADEFA is a system designed by Canberra Industries for assaying large samples, specifically waste containers for the decommissioning of the Chinon A3 Nuclear Power Plant. The samples can be as large as 1 m^3 (250 gal) and weigh as much as 450 kg (½ ton). Samples that were measured using CADEFA were thermal insulation, steel pipes and beams, electrical wiring, and concrete. Gamma-ray spectrometry was used to achieve the desired detection levels in the presence of fluctuating levels of natural radioactivity. Some of the samples being considered for measurement at Chinon contain radionuclides that emit many gamma-rays such as Eu-152, Eu-154, and ^{60}Co, along with naturally occurring radium, thorium, and potassium. These radionuclides represent the limit that a NaI(Tl) scintillator and standard gamma-ray analysis software can reliably detect[15]. Hence, HPGe detectors are being considered, since they have much better resolution and would provide better results for this radionuclide mixture (Bronson, 1994).

[14]Shielding with Cu and Cd is a well known technique to reduce the backscattering of fluorescent lead x-rays into the low-energy end of the NaI(Tl) spectra.

[15]Recall that NaI(Tl) has a resolution of about 7 - 8 % at 662 keV. This limits the ability of a NaI(Tl) spectroscopy system to distinguish between a radionuclides based on their gamma-ray spectra. Only radionuclides with intense spectral lines that don't coincide with the characteristic lines associated with natural background can be reliably identified with a NaI(Tl) detector.

Transuranic (TRU)/ Low-Level Waste. A number of requirements govern the disposition of DOE waste generated at both Federal and commercial disposal sites. These requirements constitute the basis for the performance of nondestructive waste assay (NDA) systems. The specific requirements for the disposition of transuranic waste types are defined in the Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria and the associated Quality Assurance Program Plan (U.S. DOE, 1996d). WIPP requirements essentially force NDA systems to be able to quantitatively determine alpha-emitting transuranic elements with a half-life greater than 20 years that comprise 95 percent of the hazard. WIPP also requires NDA systems to have sufficient sensitivity to verify that the total alpha activity per gram of waste matrix exceeds 3,700 Bq/g (100 nCi/g). I n addition, the NDA technique must have a measurement range equal to or greater than a 325 fissile gram equivalent[16]. Therefore, a significant amount of technological development and innovation is being brought to bear on NDA systems for the assay of TRU waste for storage at WIPP.

Technologies and Methodologies

Some aspect of the technologies and methodologies used in this field could be applicable to the measurement of residual radioactivity in volumes and on surfaces. The following paragraphs discuss some representative technologies.

Active & Passive Computed Tomography

Computed tomography (CT) is a radiographic method that permits the nondestructive physical and, to a limited extent, chemical characterization of the internal structure of materials. Since the method is x-ray based, it applies equally well to metallic and non-metallic specimens.

In conventional radiography, x-rays pass through the object, and the transmitted intensity is recorded as a two-dimensional image. The information contained in this radiograph is a projection of the absorption density in the sample onto the plane perpendicular to the x-ray beam direction. When the sample is imaged several times in different orientations, volumetric information on the sample structure can be obtained using computer algorithms. Known as a tomographic reconstruction or tomography, this enables us to look at "slices" of the investigated object without physically cutting it. Figure B-4 illustrates the CT process.

Active and passive computed tomography (A&PCT) is a gamma-ray NDA method, which has been used to identify and quantify transuranics in 208-liter (55-gallon) waste drum containers (Martz *et al.*, 1996, 1997, and1998). The A&PCT consists of two separate measurements. The first is an active CT (ACT) scan that can yield quantitative attenuation data (related to density and atomic number) using an external radiation source. The second measurement is a passive CT (PCT) scan that can, in principle, localize all detectable radionuclides within a volume (in this case, a drum) and determine their identity if an entire energy spectrum is obtained.

[16]A method of normalizing fissile and fissionable isotopes to plutonium-239 for use in establishing criticality safety limits

For ACT, the function to be imaged is the measured x-ray or gamma-ray attenuation of an external source, whereas in the case of PCT, the function to be imaged is the measured x-ray or gamma-ray activity at one or more energies of all detectable radionuclides within a drum. The ACT images are used to correct the PCT images for attenuation to determine the activity of the internal or external emitting source. For an A&PCT scanner with gamma-ray spectrometry detection equipment, each radionuclide in the drum can be identified by the energy of its characteristic radiation. More information on A&PCT can be found in papers and reports by Decman (1996), Keto (1995), Matalucci (1995b), and Robertson (1997 and 1998).

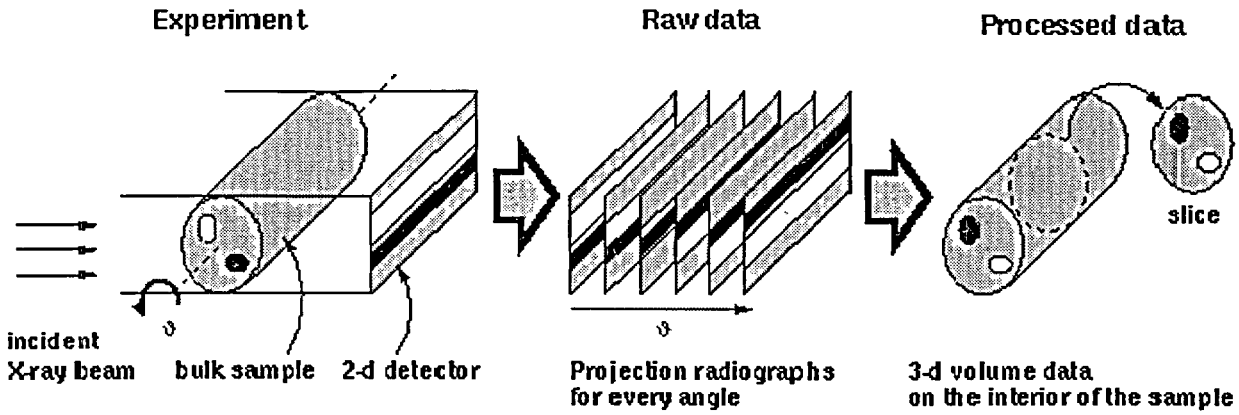

Figure B-4: The Computed Tomographic Process

Becker *et al.* (1999) evaluated 13 (with 1 under development) boxed waste NDA technologies, 2 passive neutron-based systems, and 7 active/passive neutron-based systems. Some of the technologies for the boxed waste NDA assays are summarized below. Detailed information from Becker *et al.* was preserved to illustrate the level of technology that is used to assay boxed waste containers. Background information on the technologies was included when provided.

<u>Canberra's Gamma Box Counter</u>

The Canberra Gamma Box Counter is designed to accommodate a variety of box container sizes up to the large (~ 80 m^3) shipping container. The system is typically configured with either two or eight HPGe detectors, which can be placed close to the container to optimize sensitivity, or at a distance for a far-field measurement of higher dose rate containers. The system is intended to characterize fission and activation product waste, as well as waste generated from plutonium, uranium, radium, and thorium processing applications. These waste forms are typically generated in decommissioning or environmental restoration applications. Mathematical calibrations are generated using Canberra *In Situ* Object Counting Software (ISOCS). Matrix corrections are performed using an average density matrix correction technique based on the sum of spectral data from all detectors. Corrections for nonuniform distributions can be accomplished through the calibration and through a differential peak absorption analysis technique. Qualitative evaluations of nonuniformity can also be made by evaluating the response of the individual detectors.

Oak Ridge National Laboratory's Y-12 Box Assay System

The Y-12 B-25 box NDA system is used to sort "non-radioactive waste" from low-level waste at the 1.3-Bq/g (35 pCi/g) total uranium activity. The system was designed and built at the Y-12 plant and commenced operation in early 1996. The waste form characterized by the system is produced as a byproduct of Y-12 plant operations and decontamination and decommissioning activities, and is routinely packaged in the B-25 type box.

The Y-12 box assay system is composed of two arrays of uncollimated 12.7-cm diameter by 12.7-cm thick NaI(Tl) detectors. Each array consists of six detectors placed on the long sides of the box. Detector spacing is determined according to the Nyquist critical spatial frequency[17]. Each detector is also positioned 31.75 cm from the surface of the waste box. The output of each detector is routed to a multichannel analyzer for display and analysis. Regions of interest are set for peak area quantification at the 185.7-keV gamma-ray from ^{235}U and 1,001-keV gamma-ray from ^{234m}Pa. Analysis is performed using a point-source efficiency response followed by a transmission correction for attenuation, thus quantifying the radioactivity of ^{235}U and ^{238}U. Four HPGe detectors, two on each side, screen the box for the presence of non-uranium isotopes to provide information on enrichment. A 5-cm thick iron wall on each side of the detector arrays provides background radiation shielding.

In a separate measurement station, a three-position gamma-ray transmission measurement is made through the short, horizontal axis of the box. This measurement allows correction of the uncollided flux for matrix attenuation. The transmission measurement is acquired via three collimated NaI(Tl) detectors (7.6-cm diameter by 7.6-cm thick) located on one side of the box, opposite three depleted uranium and three enriched uranium transmission sources on the other side. Data from the two measurement systems are fused together in an algorithm that yields measurement results for ^{235}U and ^{238}U.

East Tennessee Technology Park (ETTP) K-25 Box Assay System

The East Tennessee Technology Park, formerly the K-25 Site, was a uranium enrichment facility that processed and stored a large variety of radioactive wastes. These waste forms are generated primarily as a result of maintenance and decontamination and decommissioning operations in the five gaseous diffusion plants. The B-25 type box is the predominant container type used for waste packaging. Matrix types are segregated into two broad categories, including combustibles and metallic waste forms. The waste is primarily contaminated with uranium at variable enrichments that historically have averaged approximately 3 percent. Techniques used include NaI(Tl) gamma, HPGe gamma, and passive neutron. The measurement protocol commences with an assay at the NaI(Tl) detector station, followed by a passive neutron measurement for metallic type matrices only, and a final measurement via a HPGe gamma spectroscopy system.

The NaI(Tl) measurement station consists of four 12.7-cm diameter by 7.6-cm thick lead collimated NaI(Tl) detectors interfaced to a PC-based analyzer equipped with four 1,000-channel analyzers. Two detectors are centered on each long side of the B-25 box, 45.7 cm from the edge at 91.4 cm, box surface offset. The system independently processes signals from each of the four detectors. Regions of interest are set on the MCA for the 185.7-keV gamma-ray of ^{235}U and the 1,001-keV gamma-ray of ^{234m}Pa. The sum response of the four detectors, corrected for efficiency, attenuation, and background, is the basis for mass determination on either ^{235}U or ^{238}U.

[17]The distance between adjacent detectors is the sum of the distances corresponding to that point where the detector response is one-half the maximum for a point source response at 31 75 cm from the detector face

The radioactive source's spatial and matrix attenuation dependent detector response is modeled for each NaI(Tl) detector using a program called GAMMAEFF. Corrections for matrix attenuation are based on the net box weight to determine matrix density and knowledge of the matrix type to arrive at appropriate gamma attenuation coefficients. The matrix density is determined from the net box weight with the assumption that the matrix fills the box homogeneously. The GAMMAEFF program uses the matrix type, density, and associated attenuation coefficients for determination of matrix attenuation correction factors over a range of matrix types and densities. The matrix correction factor is applied to each of the NaI(Tl) responses, and the sum of the four detectors are used to arrive at the isotope mass. A 3-percent uranium enrichment is assumed for the NaI(Tl) measurement when the ^{235}U and ^{238}U masses are less than 0.2 and 30 grams, respectively. Mass values less than these do not allow use of the HPGe system for enrichment measurements due to sensitivity considerations. Under such conditions, the NaI(Tl) system is effectively a standalone measure.

A passive neutron measurement station is used to verify that large masses of highly enriched ^{235}U have not been missed in the heterogeneous steel matrix. The HPGe measurement is used to estimate the ^{235}U enrichment and identify the presence of other gamma-ray emitting radionuclides. The mass of ^{235}U or ^{238}U (based on the NaI(Tl) measurements) is used as the reference value for determination of enrichment and mass of other radionuclides through HPGe measured relative ratios. The system consists of one collimated HPGe detector positioned to view the long side center of the box. The HPGe detector is interfaced to a PC data acquisition and analysis system. The results of radionuclide identification and peak fit routines are input to the ISOTOPICS program, which uses this information with measurement configuration data to compute geometry and matrix attenuation corrections. Matrix and container material types are adjusted to ensure applicable mass attenuation coefficients are employed for the gamma-ray energies of interest. The HPGe results are normalized to the ^{235}U, and occasionally ^{238}U, mass derived from the NaI(Tl) measurement station. The NaI(Tl) based ^{235}U mass value used as this measure has a smaller geometry dependent correction versus the HPGe system.

Oak Ridge National Laboratory's Waste Examination and Assay Facility B-25 Box Assay System

The specification and preliminary design of a waste assay system for the identification and quantification of gamma-ray-emitting radionuclides in the B-25 waste box container has been performed at the Oak Ridge National Laboratory Waste Examination and Assay Facility (WEAF). The system, tentatively called the B-25 Box Assay System (B-BAS), is designed to address the need to measure the radionuclide content of a B-25 waste box at its site of residence. This is specifically intended to reduce costs by minimizing transportation of the box to a facility specifically for nondestructive assay or representative sampling of its contents.

The B-BAS is based on an array of eight low-resolution/high-efficiency 7.6-cm by 7.6-cm NaI(Tl) detectors for identification and quantification of waste entrained, gamma-emitting radionuclides. Four detectors are positioned on one long side of the B-25 box with a symmetrical arrangement of the remaining four on the opposite side. The eight detectors are mounted to a moveable support structure with large wheels, allowing the B-BAS assembly to be moved by hand down the long axis of a B-25 waste container. This moveable structure is designed to be easily transportable between measurement sites. The wheels are removed to insert the B-BAS in the WEAF Real-Time Radiography (RTR) system for the ultra-high-sensitivity "No Rad Added" type measurements.

The moveable detector assembly positions the detectors at a distance of 30 cm from the surface of the B-25 box. The detector's spatial configuration is designed to allow a maximum field of view for the middle two detectors and a minimum field of view for the uppermost and lowermost detectors. The two middle detectors have the same collimator design (i.e., a 34.2 degree angle from the centerline of the collimator). The uppermost collimator has a smaller field of view with only a 9.5 degree angle of collimation with respect to the centerline. The smallest field of view is implemented in the lowest detector (4.4 degree angle with respect to the centerline). Each collimator has at least 2.5 cm of lead to shield background gamma rays.

The measurement protocol for the B-BAS is to acquire data in a scanning fashion by movement of the NaI(Tl) detector array across the B-25 box. This scanning data acquisition mode is performed manually by operating personnel. When the B-BAS is inserted into the WERF RTR chamber, the wheels of the B-BAS are removed and the detectors are fixed. Scanning is achieved within the RTR chamber via a B-25 box transport system, which moves the box past the fixed detector array at a constant speed.

Signals from the NaI(Tl) detectors are routed into two mixer/routers. Each of the two mixer/routers allows simultaneous acquisition of up to four signals. These mixer/routers have a preamplifier and an amplifier on each channel. The preamp/amp combination allows the user the ability to "gain match" the detectors. The purpose of gain matching is to allow spectra summing for the detector arrays by adding channel to channel. The summed spectra are processed through a PC-based, multichannel analyzer card.

B.5 A Survey of Reported Minimum Detectable Concentrations for Selected Instruments and Measurement Methods

For low-level measurements, the minimum detectable concentration (MDC) is an important performance characteristic. It is usually difficult to make a fair and meaningful comparison of the sensitivity between various instruments (e.g., a gas proportional counter and a GM tube) and measurement methods (e.g., total ionization and gamma-ray spectrometry). Yet, some approaches are generally regarded as more sensitive than others. This section lists MDC values for a collection of instruments and measurement methods that are relevant to clearance. In most cases, MDC values are provided from instrument vendors without any explanation concerning the methods and specific formulae used to arrive at these values; therefore, they should be viewed with caution.

The focus of this section is the data in Tables B-3a, B-3b, B-4a, and B-4b. Tables B-3a and B-3b cover technologies that have been applied to volumetric contamination. Table B-3a categorizes the techniques/technologies according to the application, assay strategy, matrix, source size, assay technique/technology, and radiation detector. Assay strategies reflect techniques that are used to quantify activity. They range from simple techniques that measure total ionization to more sophisticated techniques that involve spectroscopy with passive and active methods of background reduction. Surface measurements are treated in Tables B-4a and B-4b. Note that, unlike Table B-3a, these tables do not address applications because (for the technologies listed) the application is exclusively for decontamination and decommissioning (D&D). Also, note that for surface contamination, the preferred detection method involves measuring total ionization, which precludes (for the most part) radionuclide identification.

The range of MDC values for volumetric contamination is rather large. The Compton suppression well counter (CSWC) has an MDC of a few tenths of a Bq/kg in the case of ^{137}Cs, while scanning for natural uranium using scintillators has an MDC of several thousand Bg/kg. The situation is similar for surface contamination; the MDCs range from a few tens of Bq/m^2 for liquid scintillation counting to a few thousand Bq/m^2. Count times range from 1 second in the case of scanning measurements to a day or more for laboratory analysis. Sample size (and active area in the case of surface contamination) is one of the key features in determining the sensitivity. Note that in the case of the CSWC (Table B-3a, ID nos. 4a, 4b, 4c), the sensitivities are fairly low and somewhat comparable to the MDCs for the *in situ* measurements of soil taken with a HPGe detector at a standoff distance of 1 m (Table B-3a, ID nos. 5a, 5b, 5c, 5d). The *in situ* soil measurements achieve low MDCs with a relatively short count time (as compared to the CSWC) because of the large sample size. The CSWC uses just a few grams of material, while an *in situ* soil measurement has an effective sample size of about 100,000 kilograms. Compare that situation with the *in situ* measurement of soil; note the MDC for ^{137}Cs is a respectable 0.8 Bq/kg. This situation is similar for surface contamination. The LRAD system (see Table B-4b, ID no. 4) has an MDC in the range of 12–30 Bq/m^2, compared to a gas proportional counter with an MDC for ^{230}Th and transuranics of 600 Bq/m^2. While the count time is not given for the LRAD system (it is not unreasonable to believe that it is commensurate with the count time for the gas proportional counter), we see that the active area of the LRAD is 100 times greater than that of the gas proportional counter.

The foregoing discussion leads us to a general conclusion that has implications for the design of a detection system and/or measurement strategy to achieve the appropriate MDC value for a given application. Specifically, *use the largest practical sample size coupled with the largest practical detector or array of detectors.*

It is clear that measurement of radioactivity associated with the control of solid materials is greatly facilitated by the development of new radiation detectors and detection systems. Of the systems addressed, the ones being developed for the assay of transuranic waste are of particular interest. Although not directly applicable to levels of radiation near background, they do represent the state-of the-art in radiation detection. This appendix attempted to compare the detection sensitivity for a variety of systems, with the caveat that many of the reported MDCs are from instrument manufacturers and should be viewed with caution. The comparison is valuable in the sense that it led to a general conclusion regarding the sensitivity of radiation detectors for radioactivity associated with the control of solid materials.

Table B-3a: Measurement technologies for volumetric contamination

ID #	Application	Assay Strategy	Matrix	Source Size (g)	Assay Technique/ Technology	Radiation Detector
1a	Routine sample analysis	sampling & lab analysis	water (ρ=1.0 g/cm^3)	1000	gamma-ray spectrometry with shielded detector	HPGe (60% rel. efficiency)†
1b						
1c						
2				250		HPGe (115% rel. efficiency)†
3			soil (ρ=1.0 g/cm^3)	—[a]		NaI (Tl) (7.6 cm × 7.6 cm)
4a	Environmental			3	Compton suppression well detector/gamma-ray spectrometry	HPGe well detector (125 cm^3)
4b						
4c						
5a	D&D	NDA/ direct measurements	soil (ρ=1.5 g/cm^3)	~10^8	*in situ* gamma-ray spectrometry at 1 m	HPGe (40% rel. efficiency)†
5b						
5c						
5d				~10^9	*in situ* gamma-ray spectrometry at 8 m	6 HPGe (75% rel. efficiency)†
6						
7				N/A	gamma-ray spectrometry	CZT array
8a				N/A	portable energy dispersive x-ray fluorescence	HgI_2
8b						—[a]

Table B-3a: Measurement technologies for volumetric contamination

ID #	Application	Assay Strategy	Matrix	Source Size (g)	Assay Technique/ Technology	Radiation Detector
9a	D&D	NDA/ direct measurements	soil (ρ=1.5 g/cm^3)	N/A	laser ablation mass spect.	N/A
9b						
9c						
10				~700	scintillating fiber optics with anti-coincidence counting	Fiber Optic (Beta-Scint™)
11a		NDA/ hand-held scanning		N/A	gross radiation counting	NaI(Tl) (3.8 cm × 3.8 cm)
11b						
11c						
12a						NaI(Tl) (5.1 cm × 5.1 cm)
12b						
12c						

Table B-3a: Measurement technologies for volumetric contamination

ID #	Application	Assay Strategy	Matrix	Source Size (g)	Assay Technique/ Technology	Radiation Detector
13a	Waste Assay	NDA/ *in toto*	low density	N/A	WAND system	Array of Phoswich Detectors
13b						
13c						
14a						
14b					HERCULES system	
14c						
15a			low Z, low density (ρ=0.3 g/cm^3)	~10^7	*in situ* gamma-ray spectrometry at 1 m	HPGe (40% rel. efficiency)†
15b						
15c						
15d						
16a			--[a]	200 liter (55 gallon) drum	*in situ* gamma-ray spectrometry at 1 m	HPGe (40% rel. efficiency)†
16b						
16c						
16d						
17a			Misc.Waste	5×10^6	CADEFA gamma-ray spectrometry	--[a]
17b						
18a				8		
18b						
19	Safeguards		HEU in van	4×10^4 - 2×10^5 [b]	portal monitor	plastic scintillators

--[a] data not provided

[b] represents total mass of radionuclide (e.g , 40 – 200 kg of highly enriched uranium (HEU))

† rel. efficiency: efficiency relative to a 7.6 cm x 7 6 cm NaI(Tl) detector

Table B-3b: MDC values for volumetric contamination

ID #	Radionuclide	Time (s)	MDC (Bq/kg)	MDA* (Bq)	Reference
1a	^{60}Co		0.64	0.64	
1b	^{137}Cs	600	0.70	0.70	ANSI/HPS N13.12-1999
1c	^{241}Am		4.2	4.2	
2	^{137}Cs	6000	1.4	0.35	Koch, P., *et al.*, 1997.
3	^{40}K	36000	15	N/A[c]	Ibeanu, I., 1999.
4a	^{137}Cs		0.32	9.6×10^{-4}	
4b	^{238}U	86400	18	N/A[d]	Harbottle, G., *et al.*, 1994
4c	^{241}Am		0.44	0.0013	
5a	^{60}Co		1.1	$\sim 10^{5}$	
5b	^{137}Cs	900	0.8		www canberra com/literature/technical_ref/gamma/isocs
5c	^{238}U		110	$\sim 10^{8}$	
5d	^{241}Am		3.6	$\sim 10^{5}$	
6	^{241}Am	3600	3.8	$\sim 10^{10}$	Reimann, R.T, private communication
7	Uranium	--[a]	27		Metzger, R *et al.*, 1998
8a	^{40}K	--[a]	6500	N/A[d]	Potts, P.J., 1999
8b	^{238}U	--[a]	1900		

Table B-3b: MDC values for volumetric contamination

	ID #	Radionuclide	Time (s)	MDC (Bq/kg)	MDA* (Bq)	Reference
5329	9a	^{60}Co		37		
5330	9b	^{137}Cs	--[a]	4	N/A	NUREG-1575, 1997
5331	9c	^{238}U		0.04		
5332	10	^{90}Sr / ^{238}U	300	37	26	U.S. DOE, 1998a
5333	11a	^{137}Cs		380		
5334	11b	Nat U	~ 1	4300		
5335	11c	^{241}Am		1700	N/A	Abelquist, E.W., and W.S. Brown, 1999
5336	12a	^{137}Cs		240		
5337	12b	Nat U	~ 1	2700		
5338	12c	^{241}Am		1200		
5339	13a	^{137}Cs			52	
5340	13b	^{238}U	1000	<190	52	Myers, S.C., 2000
5341	13c	^{241}Am			30	
5342	14a	^{137}Cs			104	
5343	14b	^{238}U	1000	<190	181	Myers, S.C., 2000
5344	14c	^{241}Am			22	
5345	15a	^{60}Co		7.8	8×10^4	
5346	15b	^{137}Cs	900	12	1×10^5	www.canberra.com/literature/technical_ref/gamma/isocs
5347	15c	^{238}U		1100	1×10^7	
5348	15d	^{241}Am		1900	2×10^7	
5349	16a	^{60}Co	900	48	N/A	www.canberra.com/literature/technical_ref/gamma/isocs
5350	16b	^{137}Cs		28		

Table B-3b: MDC values for volumetric contamination

ID #	Radionuclide	Time (s)	MDC (Bq/kg)	MDA* (Bq)	Reference
16c	^{238}U		3500		
16d	^{241}Am		2700		
17a	^{60}Co	180	2	1000	Bronson, F., 1994
17b	^{137}Cs		2	1000	
18a	^{60}Co	180	25	200	Bronson, F., 1994
18b	^{137}Cs		25	200	
19	HEU	~ 1-5	N/A	$\sim 10^8$	York, R.L., *et al.*, 1996

*MDA - minimum detectable activity

N/A[c] - Not applicable because no sample mass provided.

N/A[d] - Not applicable because not enough data was provided (mass and/or count time).

Table B-4a: Measurement technologies for surface contamination

ID #	Assay Strategy	Assay Technique/Technology	Detector	Active area (m^2)
1	sampling & lab analysis	liquid scintillation counting	NaI(Tl)	N/A
2	NDA/ direct measurements	γ ray spectrometry with unshielded detector	FIDLER* (NaI(Tl))	–[a]
3a			HPGe (40% rel. efficiency)†	N/A
3b				
3c				
4		LRAD/ total ionization	ionization chamber	1
5a		total ionization	gas proportional counter	0.01
5b				
5c				
6			gas proportional counter	0.01
7			zinc sulfide	0.01
8a			Geiger-Muller tube	0.002
8b				
9a		LRAD/ total ionization	ionization chamber	0.01
9b				
9c		total ionization	large-area monitor	0.01
9d				
10a	NDA/ scanning measurements (manual & conveyorized)	SCM/SIMS/total ionization	position-sensitive proportional counter	--[a]
10b				
11a		Pipe Explorer™/total ionization	scintillating membrane	
11b			NaI(Tl)	
12	NDA/ *in toto* measurements	IONSENS™ 28 Large Item Monitor	ionization chamber	-b

--[a] data not provided

* Field Instrument for the Detection of Low Energy Radiation (FIDLER) The FIDLER consists of a thin Be and Al window with a NaI detector coupled to a PMT (see NUREG-1575 for more information)

†rel. efficiency - efficiency relative to a 7 6 cm x 7.6 cm NaI(Tl) detector

Table B-4b: MDC values for surface contamination

ID #	Time (s)	Radionuclide/ Radiation Type	MDC (Bq/m^2)	Reference
1	--[a]	^{90}Sr	0.18	ANSI/HPS N13.12-1999
2	--[a]	^{241}Am	19000	Kirby, J., *et al.*, 1976
3a		^{60}Co	350	
3b	3600	^{137}Cs	3500	www canberra com/literature/technical_ref/gamma/isocs
3c		^{241}Am	310	
4	--[a]	α activity	12-30	NUREG-1575, 1997
5a		^{14}C	930	
5b	60	^{99}Tc	4.9	NUREG-1507, 1998
5c		$^{90}Sr(^{90}Y)$	2.9	
6		^{230}Th and transuranic	600	
7		^{230}Th	108	
8a		$^{90}Sr(^{90}Y)$	10^4	
8b		fission products	10^4	Goles, R.W., 1991
9a	60	$^{90}Sr(^{90}Y)$	750	
9b		U (nat), ^{235}U, ^{238}U & progeny	600	
9c		^{230}Th and transuranic	600	
9d		fission products	750	
10a	--[a]	β/γ activity	500	Pulsford, S.K., *et al.*, 1998
10b	--	α activity	50	
11a	~3	α activity/ ^{238}U	8300	Cremer, C.D., and D.T. Kendrick, 1998
11b	--	β/γ activity/ Co-60	1100	
12	100	α activity	4000	www.bnfl-instruments.com

--[a] data not provided

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10. SUPPLEMENTARY NOTES

G.E. Powers, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The U.S. Nuclear Regulatory Commission (NRC) is developing a basis to support decisions on whether to undertake a rulemaking that would set specific requirements on controlling licensees' releases of solid materials. Specifically, the solid materials being evaluated include metals, building concrete, onsite soils, equipment, furniture, etc., which are present at, and/or used in, licensed nuclear facilities during routine operations. Historically, licensees have released solid materials on a case-by-case basis, without a consistent approach to designing and conducting clearance surveys. This draft report provides information about measuring residual radioactivity in materials that are to be cleared from nuclear facilities, including guidance about designing, performing, and documenting radiological surveys of solid materials to address the need for consistency in the surveys.

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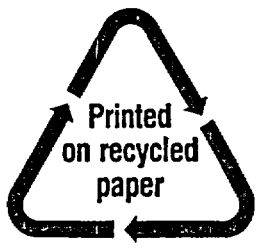

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