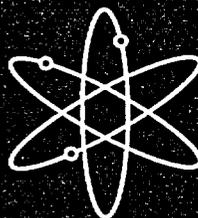# Radiological Surveys for Controlling Release of Solid Materials

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

AVAILABILITY OF REFERENCE MATERIALS
IN NRC PUBLICATIONS

Radiological Surveys for Controlling Release of Solid Materials

Draft Report for Comment

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ABSTRACT

1

2   The U.S. Nuclear Regulatory Commission (NRC) is developing a basis to support decisions on whether
3   to undertake a rulemaking that would set specific requirements on controlling licensees' releases of solid
4   materials. Specifically, the solid materials being evaluated include metals, building concrete, onsite
5   soils, equipment, furniture, etc., which are present at, and/or used in, licensed nuclear facilities during
6   routine operations. Historically, licensees have released solid materials on a case-by-case basis, without
7   a consistent approach to designing and conducting clearance surveys. This draft report provides
8   information about measuring residual radioactivity in materials that are to be cleared from nuclear
9   facilities, including guidance about designing, performing, and documenting radiological surveys of solid
10 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.]

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

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

FOREWORD

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

256 <u>NRC Examination of its Approach for Controlling the Release of Solid Material</u>

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

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

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

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

277 <u>Further Development of Use of the Data Quality Objectives Process</u>

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

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

294 <u>Scope and Approach of this Report</u>

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

301 The alternative of not permitting material to be released if it is located in an area where radioactive
302 materials are used or stored, referred to in the issues paper as "prohibition," would rely principally
303 on process knowledge of where the material originated because it would use that information as a basis
304 for determining disposition of the material. Information on process knowledge is presented
305 in Section 4.3 of this report. This alternative would not be as dependent upon detailed methods
306 for radiological surveys and, thus, much of the information in later sections of this report would not apply
307 to this alternative. The alternatives of continuing current practice or permitting release using dose-based
308 criteria rely upon process knowledge of where the solid materials originated in the facility, as well as
309 comprehensive radiological surveys to demonstrate that the level of radioactivity on the material would
310 meet the required criteria. Information on various survey methodologies is presented in Section 5.
311 The alternative of restricted use may use process knowledge to determine those materials that would be
312 limited to authorized uses, but may be similar to unrestricted use in the need for comprehensive surveys.
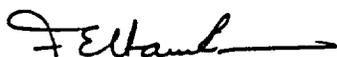
313 Farouk Eltawila, Director
314 Division of Systems Analysis and Regulatory Effectiveness
315 Office of Nuclear Regulatory Research

ACKNOWLEDGMENTS

ABBREVIATIONS AND SYMBOLS

347	ADP	automated data processing
348	AEC	Atomic Energy Commission
349	ALARA	as low as is reasonably achievable
350	ANL	Argonne National Laboratory
351	ANSI	American National Standards Institute
352	ASME	American Society of Mechanical Engineers
353	CSM	conveyorized survey monitor
354	CSS	Compton suppression spectrometer
355	d'	detectability index
356	D&D	decontamination and decommissioning
357	$DCGL_c$	derived concentration guideline level for clearance
358	dpm	disintegrations per minute
359	DQA	data quality assessment
360	DOD	U.S. Department of Defense
361	DOE	U.S. Department of Energy
362	DQI	data quality indicator
363	DQO	data quality objective(s)
364	ε_i	instrument efficiency
365	ε_s	surface efficiency
366	EIC	electret ion chamber
367	EML	Environmental Measurements Laboratory
368	EPA	U.S. Environmental Protection Agency
369	EPRI	Electric Power Research Institute
370	FIDLER	Field Instrument for the Detection of Low-Energy Radiation
371	FPXRF	field-portable x-ray flourescence
372	FWHM	full width at half maximum
373	GDP	gaseous diffusion plant
374	GeLi	germanium-lithium
375	GM	Geiger-Mueller
376	HPGe	high-purity germanium (detector)
377	HVT	half-value thickness
378	i	observation interval
379	ICP-MS	inductively coupled plasma mass spectrometer
380	ISGS	*in situ* gamma spectrometry
381	ISO	International Organization for Standardization
382	IUPAC	International Union of Pure and Applied Chemistry
383	LBGR	lower bound of the gray region
384	LN	liquid nitrogen
385	MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
386	MCNP	Monte Carlo N-Particle
387	MDA	minimum detectable activity
388	MDC	minimum detectable concentration
389	MDCR	minimum detectable count rate
390	MFP	mean-free-path
391	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 $DCGL_c$.

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[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.

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

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

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

473 ## 1.4   Methodology

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

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

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

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

505 ● nuclear power reactor
506 ● sealed source facility
507 ● transuranic facility
508 ● fuel fabrication facility
509 ● broad research and development (R&D) facility
510 ● gaseous diffusion plant
511 ● uranium mill facility
512 ● rare earth facility

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

2 ROADMAP

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

585   r. Reevaluate solid material disposition options.

```
┌─────────────────────────────────────────────────────┐
│ a. Evaluate physical description of solid material (4.2) │
└─────────────────────────────────────────────────────┘
                          │
                          ▼
┌──────────────────────────────────────────────────────────────────┐
│ b. Evaluate and document process knowledge/characterization of solid material (4.3) │
└──────────────────────────────────────────────────────────────────┘
                          │
                          ▼
              No      ◇ c. Is material impacted? (4.3) & (4 4) ◇
Release ◀─────────────
                          │ Yes
                          ▼
┌───────────────────────────────────────────────────────┐
│ d. Specify release criteria and conditions for solid material (4.1) │
└───────────────────────────────────────────────────────┘
                          │
                          ▼
              ┌──────────────────────┐
              │ e  Classify material (4.4) │
              └──────────────────────┘
                          │
                          ▼
```

Cost considerations
- Cost of disposal/WAC survey
- Value of material
- Cost of clearance survey
- Cost of decontamination
- Cost of dismantlement

◇ f Is clearance an option? (3 2) ◇ → No → r. Re-evaluate disposition options

Yes

Figure 2.1: Flow diagram for clearance of solid materials

7

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

588

8

3 DATA QUALITY OBJECTIVES

589

590 The approach used in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM,
591 1997) has proven to be very useful for designing efficient, objective, and defensible final status surveys
592 to collect data to support decisions concerning the release of lands and structures for unrestricted use
593 according to the criteria established by the Commission's final rule (NRC, 1997). Many of the
594 improvements in the design of final status surveys using the MARSSIM were achieved through the
595 extensive use in that document of the Data Quality Objectives (DQO) Process.

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

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

3.1 State the Problem

609

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

617 If solid material has the potential for containing contamination from facility operations, a survey is
618 generally required before the material may be released from controls. The types and sensitivity of
619 equipment, procedures, and resources available for measuring any contamination in or on the material
620 should be also be addressed. The regulatory criteria for preventing the release from control of materials
621 with unacceptable levels of contamination must also be established. These may be either activity-based
622 or dose-based. If the criteria are dose-based, the equivalent criteria in terms of an activity concentration
623 must be obtained from an approved dose modeling procedure; NUREG-1640 provides an example of a
624 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.

(f.1) - Is clearance an option for the survey method chosen? (3 2)

Disposal   Evaluate clearance and disposal options   Clearance

Select Survey Method
- Application of DCGLs
- Determine background
- Determine MDC
- MDC <DCGL

1. Value of material

2 Cost of disposal

Are there inaccessible areas? — No

Yes

1. Cost of dismantlement

3 Cost of waste acceptance survey

No — Is decontamination necessary?

Yes

2. Cost of decontamination

Total cost of disposal option (sum of 1,2, and 3)

3. Cost of clearance survey

Total cost of clearance option (sum of 1,2, and 3)

No — cost of disposal option< cost of clearance option — Yes

Perform clearance survey

No — Is another survey option available? — Yes

dispose

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

660

11

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

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

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

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

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

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

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

703   **3.4 Define the Study Boundaries**

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

711 In some cases, there may be a more natural connection between the "batch size" of a lot of material and
712 the type of survey that should be performed. This is discussed at length in Sections 4.1, 4.2, and 5. Here,
713 the reader should simply note that for material that consists of many small regular pieces, a conveyorized
714 scanning system may be used. In this case, a batch might be the amount of material within the instrument
715 field of view. If the material consists of a few large irregularly shaped pieces, a batch might be a single
716 piece that is hand-scanned, or perhaps a few pieces scanned *in toto* using a box or drum counter, or
717 measured using an *in situ* gamma spectrometer.

718   **3.5   Develop a Decision Rule**

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

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

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

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

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

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

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

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

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

14

3.6   Specify Limits on Decision Errors

781

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

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

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

801   *Null Hypothesis: The solid material surveyed contains no contamination.*

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

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

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

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

862   3.7   Optimize the Design for Obtaining Data

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

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

873   With sufficient scanning sensitivity, direct measurements are not required. Conveyorized survey
874   monitors may be able to efficiently scan 100 percent of the material, again without the need for direct
875   measurements. Measurements of an entire batch of material using *in toto* techniques in essence combine
876   the attributes of a direct measurement with a measurement that has some of the attributes of a
877   100-percent scan.

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

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

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

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

17

4 SURVEY DESIGN CONSIDERATIONS

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

4.1 Release Guidelines

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

4.1.1 Forms of Release Guidelines

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

916 **Table 4.1: Regulatory Guide 1.86 surface activity guidelines**

Radionuclide	Average Total Surface Activity in 1 m² (dpm/100 cm²)	Maximum Surface Activity in 100 cm² (dpm/100 cm²)[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-cm² areas.

928   The application of the surface activity guidelines shown Table 4.1 requires some explanation. First, it is
929   important to understand that surface activity levels may be averaged over 1 m², but no surface activity
930   levels can exceed the maximum surface activity specified for a 100-cm² area. Hence, RG 1.86 provides
931   release criteria for surface activity, as well as averaging conditions for the application of those criteria.
932   Also note that RG 1.86 does not include volumetric release criteria. The standards were to be dose-
933   based; hence, the release criteria should include the dose criterion upon which to base the $DCGL_C$
934   (clearance DCGL), as well as any necessary conditions for the implementation of the $DCGL_C$. For
935   example, any limits on the area or volume averaging of solid materials should be clearly expressed.
936   Restrictions on the averaging area or volume of solid materials will necessarily impact the material
937   survey unit or batch size.

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

945   **4.1.2   Release Guidelines — Averaging Conditions and Survey Unit Considerations**

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

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

970   **4.2   Solid Materials**

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

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

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

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

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

999   **Concrete rubble** consists of crushed concrete of a soil-like consistency from the demolition of buildings
1000   and structures. The reinforcing steel rebar has been removed from the concrete rubble. The primary
1001   assessment techniques include laboratory analysis of a statistically determined number of representative
1002   samples and surface scans, or use of a CSM. The total surface area of the crushed concrete when spread
1003   out to a height of 15 cm (to facilitate scanning) is about 50 m^2. This survey unit is assumed to have no
1004   inaccessible areas.

1005   A **concrete slab** consists of 30-cm thick medium density concrete (2.4 g/cm^3), with surface dimensions
1006   of 1.2 m by 1.8 m. The primary assessment technique is surface activity measurements, perhaps with the
1007   number of measurements statistically determined, and surface scans. This survey unit is assumed to have
1008   no inaccessible areas and only to have contamination surficially. If volumetric contamination is
1009   expected, alternative clearance survey techniques, such as concrete core samples, are warranted.

1010 **Small-bore pipe** (<6 cm diameter) from piping systems and electrical conduit is assumed to be sectioned
1011 into 1.2-m to 1.8-m lengths. It is assumed that conventional survey instrumentation cannot access the
1012 pipe interiors. For Class 2 and 3 survey units—so classified because the pipe interiors are very unlikely
1013 to have contamination—the primary assessment technique is surface activity measurements of pipe
1014 exteriors, with a number of smears from the pipe interiors, and surface scans. Class 1 survey units should
1015 be fully surveyed inside—so either the pipe must be cut open or specialty survey equipment employed.
1016 It may also be possible to evaluate the cut pipe using a CSM or *in situ* gamma spectrometer. The surface
1017 area for pipe section exteriors per survey unit is 17 m^2 (based on a pipe diameter of 6 cm and 1.5-m
1018 lengths).

1019 **Large-bore pipe** (>6 cm diameter) from piping systems is assumed to be sectioned into 1.2-m to 1.8-m
1020 lengths. It is assumed that conventional survey instrumentation can access the pipe interiors. Therefore,
1021 this survey unit is assumed to have no inaccessible areas. The primary assessment technique is surface
1022 activity measurements of pipe interiors and exteriors, and surface scans. The surface area for pipe
1023 section interiors and exteriors per survey unit is 72 m^2 (based on a pipe diameter of 30 cm and 1.5-m
1024 lengths).

1025 **Structural steel** consists of light and heavy gauge steel that may require sizing to fit on a pallet (1.2-m to
1026 1.8-m lengths). The structural steel may consist of I-beams, structural members, decking, ductwork,
1027 tanks, and other containers. This survey unit is assumed to have no inaccessible areas. The primary
1028 assessment technique is surface activity measurements, with the number of measurements statistically
1029 determined, and surface scans. *In toto* clearance techniques may also be useful to assess structural steel.

1030 **Copper wire** consists of insulated and non-insulated wire (0.6 cm or larger), copper windings, and bus-
1031 bars. It is assumed that this amount of copper weighs 0.75 tons. The primary assessment technique is
1032 surface activity measurements, with the number of measurements statistically determined, and surface
1033 scans. *In toto* clearance techniques may also be useful to assess copper wire. This survey unit is
1034 assumed to have no inaccessible areas.

1035 **Copper ingots (bulk)** consist of size-reduced pieces of copper and ingots. The primary assessment
1036 technique is laboratory analysis of a statistically determined number of representative samples and
1037 surface scans, or use of a CSM. The total surface area of the bulk copper when spread out to a height of
1038 5 cm is about 15 m^2. This survey unit is assumed to have no inaccessible areas.

1039 **Soil** includes materials that are soil-like, consisting of a finely divided mesh. The primary assessment
1040 technique is laboratory analysis of a statistically determined number of representative samples and
1041 surface scans. Other clearance survey techniques that might be employed include use of a CSM or
1042 *in toto* techniques. The total surface area of the soil when spread out to a height of 15 cm (to facilitate
1043 scanning) is about 50 m^2. This survey unit is assumed to have no inaccessible areas.

1044 **Large items for reuse** include transformers, specialty equipment (e.g., lathes), electrical panels, and
1045 other complete systems. These materials are assumed to require some amount of disassembly to allow
1046 access to interior surfaces, but consideration must be given to the fact that these items are valued for their
1047 function, so cutting is usually not an option. The nominal weight of a large item is 1.5 tons. The primary
1048 assessment technique is surface activity measurements, with the number of measurements statistically
1049 determined, and surface scans. *In toto* clearance techniques may also be useful to assess large items for
1050 reuse.

1051 **Scrap metal pile** consists of miscellaneous mixed metals with no common configuration. The scrap may
1052 require sizing to fit on a pallet. The nominal weight of the material on a pallet is assumed to be 1 ton.
1053 The primary assessment technique is surface activity measurements, with the number of measurements
1054 statistically determined, and surface scans. *In toto* clearance survey techniques might also prove useful.
1055 The total surface area of the scrap metal pile is assumed to be about 10 m^2. This survey unit is assumed
1056 to have no inaccessible areas.

1057 **Scrap equipment and small items for reuse** include small pumps, motors, hand tools, power tools,
1058 scaffolding, and the like. These materials are often associated with operational releases and are assumed
1059 to require some amount of disassembly to allow access to interior surfaces. The nominal weight of the
1060 material on a pallet is assumed to be 1.5 tons. The primary assessment technique is surface activity
1061 measurements, with the number of measurements statistically determined, and surface scans. Both
1062 *in toto* and CSM clearance survey techniques might be used to release scrap equipment.

1063 As mentioned in Section 4.1.2, survey units should be selected based on the DQO Process, consistent
1064 with the nature of the material, the clearance survey technique selected, the material's potential for
1065 contamination, and considering any relationship between the survey unit size (i.e., batch size) and the
1066 modeling input used to establish the DCGL$_C$. Table 4.2 provides typical survey unit sizes.

1067   **Table 4.2: Typical material survey unit sizes**

Solid Materials	Examples	Survey Unit Sizes
1069 Bulk materials	soil, concrete rubble, copper ingots	1 to 7.5 m^3 (smaller for CSMs)
1070 Few, large pieces of 1071 equipment and material	concrete slabs, large items	item itself
1072 Small items on a pallet	small- and large-bore pipe sections, structural steel, equipment, scrap metal, copper wire	10 to 100 m^2

23

1073
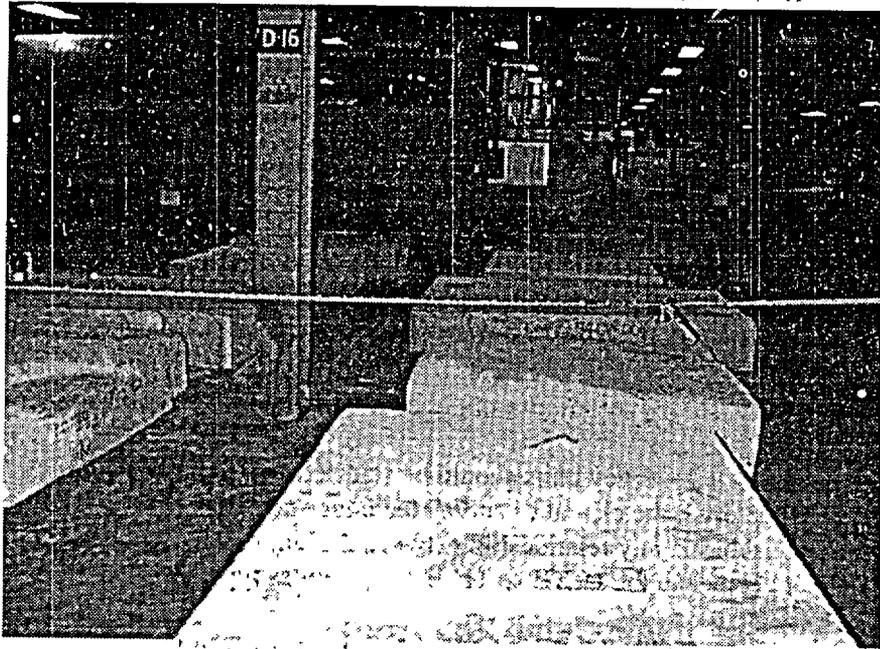
1074
Figure 4.1: Concrete slabs staged for clearance surveys

1075
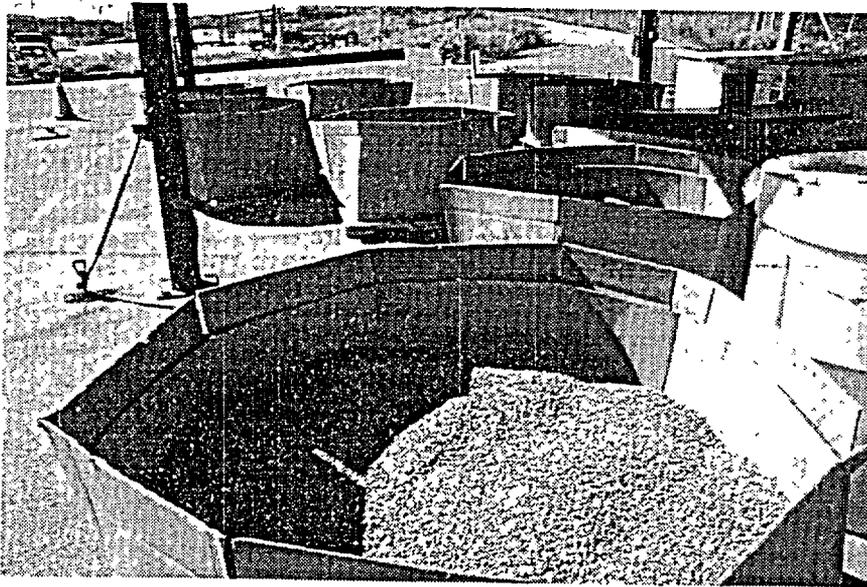
1076
1077
Figure 4.2: Containers of copper chop
(recently surveyed using the conveyorized survey monitor)

24

1078
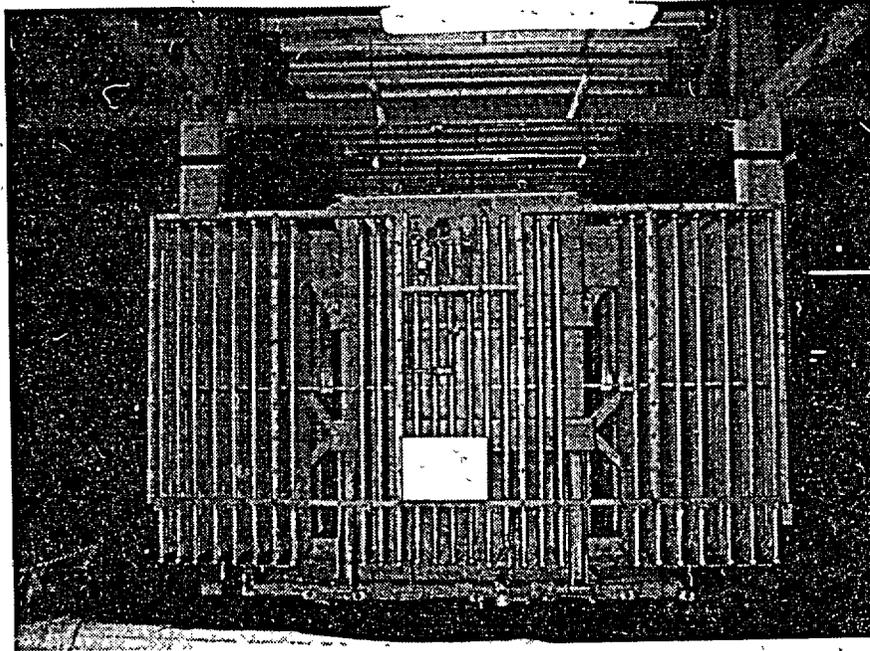
Figure 4.3: Transformer being surveyed for reuse

1079

1080

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

1081

1082

1083

Figure 4.5: Scrap metal piles being prepared for survey

1084
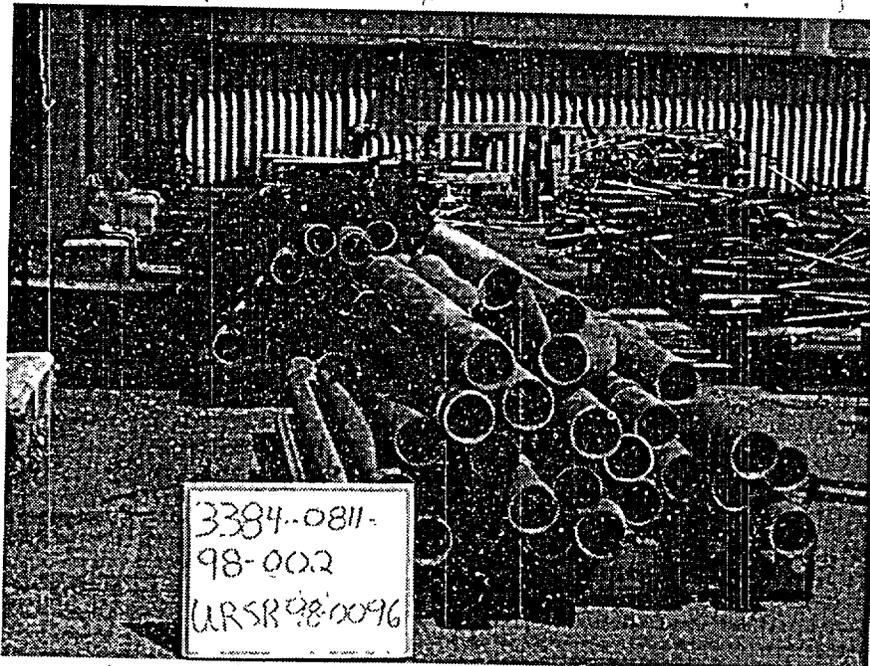
1085

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.

1127　The contamination potential of the solid material is used to further classify the material as either Class 1,
1128　2, or 3 (Section 4.4). The specific classification will assist in defining the survey approach prior to
1129　release. Those materials having the highest potential for contamination would receive the greatest
1130　clearance survey effort.

1131
1132

> 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.

1133　## 4.3.2　Evaluating the Nature of Contamination

1134　Process knowledge can also be used to determine the nature of contamination (i.e., the identity, extent,
1135　and location of the radionuclide contamination on the solid material). The type of facility from which the
1136　materials originated is an important factor. For example, if the solid materials came from a nuclear
1137　power reactor, the likely radioactivity includes fission and activation products; if the materials were from
1138　a gaseous diffusion plant, the radioactivity may include enriched uranium and ^{99}Tc. A number of studies
1139　have investigated screening (release/clearance) levels for key radionuclides associated with clearance
1140　(IAEA 1996, Hill 1995, NRC 1999, ANSI 1999). Rather than develop a new list or augment existing
1141　lists, this section focuses on a few important radionuclides to explore specific issues related to their
1142　presence and detection in solid materials.

1143　The radionuclide mixtures for each facility type (or industry category) should be known in order to
1144　effectively design the clearance survey. The specific facility type provides a general indication of the
1145　expected radionuclides. Short-lived radionuclides (i.e., half-lives from less than a day to several months)
1146　that may be associated with a particular facility are not shown. It is necessary to account for the potential
1147　presence of short-lived radionuclides, which may include justification that the radionuclides are not a
1148　concern because of their expected contamination levels considering radioactive decay. Common
1149　radionuclides at various types of facilities are as follows:

1150　Nuclear Power Reactor　　　^{60}Co
1151　　　　　　　　　　　　　　　　^{137}Cs
1152　　　　　　　　　　　　　　　　^{63}Ni
1153　　　　　　　　　　　　　　　　^{55}Fe
1154　　　　　　　　　　　　　　　　fission and activation products
1155　　　　　　　　　　　　　　　　transuranics
1156
1157　Fuel Fabrication Facility　　　enriched uranium

1158　Sealed Source Facility　　　　^{241}Am
1159　　　　　　　　　　　　　　　　^{60}Co
1160　　　　　　　　　　　　　　　　^{137}Cs
1161　　　　　　　　　　　　　　　　^{90}Sr

1162　Broad R&D Facility　　　　　^{3}H
1163　　　　　　　　　　　　　　　　^{14}C

1164	Transuranic Facility	^{241}Am
1165		^{239}Pu
1166		^{238}Pu
1167	Gaseous Diffusion Plant	^{99}Tc
1168		enriched uranium
1169		transuranics
1170	Uranium Mill Facility	^{238}U
1171		^{230}Th
1172		^{226}Ra
1173		progeny
1174	Rare Earth Facility	Thorium

1175 Scoping and characterization surveys would likely be performed, and may include field measurements
1176 and sample collection with laboratory analysis, to identify the specific radionuclides that are present and
1177 their radiation characteristics. Identification of radionuclides is generally performed through laboratory
1178 analyses, such as alpha and gamma spectrometry, and other radionuclide-specific analyses. For instance,
1179 the radionuclide mixture of contamination on solid materials that originate from a power reactor facility
1180 may be assessed by collecting representative samples, and performing gamma spectrometry analyses to
1181 determine the relative fractions of activation and fission products present. Radionuclide analyses are also
1182 used to determine the relative ratios among the identified radionuclides, as well as to provide information
1183 on the isotopic ratios and percent equilibrium status for common radionuclides like uranium and thorium
1184 decay series. This information is useful in establishing and applying the $DCGL_c$ for the material being
1185 released. Table A.4 in Appendix A provides information on radionuclide characteristics and lists some
1186 standard methods for detecting their radiations.

1187 It is useful to consider the possible contamination scenarios associated with the radionuclide(s) of
1188 concern. Radionuclides that can be connected to a specific function in a power reactor or gaseous
1189 diffusion plant, for example, will have a very specific contamination pattern or scenario based on the
1190 materials and processes involved. For example, ^{55}Fe and ^{54}Mn are activation-corrosion products, which
1191 can be found in irradiated metals from reactors (e.g., core shrouds, support plates, and core barrels), but it
1192 is unlikely that facilities would be attempting to clean (if possible) and release these materials. The more
1193 likely scenario involves materials that are associated with items that are not typically linked with any
1194 process that would expose them to radiation (e.g., neutrons) or radionuclides. Such items include
1195 structural materials (e.g., wood and steel), tools, pipework, heating and ventilation ductwork, and office
1196 equipment. Contamination found on these materials is most likely a result of the inadvertent movement
1197 of radionuclides by personnel and circulating air. However, it is clear in the case of reactor facilities that
1198 the radionuclides ^{60}Co, ^{55}Fe, ^{63}Ni and ^{54}Mn are associated with steel. Tritium (^3H) is the most mobile and
1199 is usually in the form of tritiated water when released. This means it can penetrate porous materials
1200 (such as concrete and wood) and form oxide layers on metals. In general, soluble radionuclides can
1201 penetrate porous materials to create contamination at depth. They can also become airborne and be
1202 transported by air currents to remote and inaccessible areas. Fine particles created by machining
1203 operations can become airborne and be deposited in cracks and on horizontal surfaces. With the
1204 exception of the corrosion-activation products, most of the contamination will reside on surfaces of
1205 various materials.

1206 To summarize, the nature of contamination on solid material can be described in terms of its distribution
1207 on the material. For example, the contamination distribution on most items and materials is generally
1208 spotty, although some materials (particularly those that were designed to have intimate contact with
1209 radioactivity) exhibit a more uniform contamination distribution. This is an important consideration
1210 when selecting the clearance survey approach. Scanning is the preferred clearance survey methodology,
1211 precisely for its ability to detect the predominantly spotty contamination on solid materials.

4.4 Classification

1213 All materials can be divided into two types—non-impacted and impacted. Non-impacted solid materials
1214 have no contamination potential based on process history, while impacted solid materials have some
1215 contamination potential based on operations and process knowledge. Impacted materials are further
1216 subdivided into three classes based on the materials' known contamination levels or contamination
1217 potential, as outlined in the following subsections.

1218 The classification of solid materials is used to determine the clearance survey coverage for that material.
1219 The basic philosophy is that the greater the potential for the material to have contamination, the greater
1220 the clearance survey effort. This is the philosophy in the MARSSIM, as well. The solid material
1221 classification will specify, for example, how much metal scrap on a pallet must be surveyed, or what
1222 fraction of soil must be processed through a conveyorized survey monitor.

1223 Improper classification of materials has serious implications, particularly when it leads to the release of
1224 materials with contamination in excess of clearance criteria. For example, if materials are mistakenly
1225 thought to have a very low potential for having contamination, these materials will be subjected to a
1226 minimal survey rigor. This misclassification results in a higher potential for releasing materials in error.
1227 To minimize these potential errors, investigation levels should be established and implemented to
1228 indicate when additional investigations are necessary. For example, a measurement that exceeds an
1229 appropriately set investigation level may indicate that the material survey unit has been improperly
1230 classified.

4.4.1 Class 1 Solid Materials

1232 Class 1 solid materials are those materials that have (or had) a potential for contamination (based on
1233 process knowledge) or known contamination (based on previous surveys) above the release criterion
1234 (DCGL$_c$). These solid materials include materials that comprise processing equipment or components
1235 that may have been affected by a spill or airborne release.

1236 Basically, Class 1 solid materials are those materials that were in direct contact with radioactive
1237 materials during the operations of the facility or may have become activated. Additionally, solid
1238 materials that have been cleaned to remove contamination are generally considered to be Class 1.
1239 An exception may be considered if there are no inaccessible areas and any contamination is readily
1240 removable using cleaning techniques. Examples of such methods may include vacuuming, wipe downs,
1241 or chemical etching that confidently remove all contamination such that surface activity levels would be
1242 less than the release criteria. Documented process knowledge of these cleaning methods should be
1243 provided to justify this exception to the cognizant regulatory authorities.

30

1244   **4.4.2   Class 2 Solid Materials**

1245   Class 2 solid materials are those materials that have (or had) a potential for or known contamination, but
1246   are not expected to have concentrations above the release criteria. These materials include those items
1247   that are within radiologically posted areas, but are not expected to have contamination. This class of
1248   materials might consist of electrical panels, water pipe, conduit, ventilation ductwork, structural steel,
1249   and other materials that might have come in contact with radioactive materials.

1250   Any Class 2 solid materials that exceed the release criteria, based on previous surveys, should be
1251   reclassified as Class 1 for clearance surveys. For items of unknown or questionable origin, scoping
1252   surveys should be performed to determine whether residual surface contamination is present. Provided
1253   that no activity is identified, the minimum classification for such materials should be Class 2.

1254   **4.4.3   Class 3 Solid Materials**

1255   Class 3 solid materials are those materials that either are not expected to contain any contamination, or
1256   are expected to contain contamination less than some small specified fraction of the release criteria based
1257   on process knowledge or previous surveys. Any solid materials that exceed the specified fraction of the
1258   release criteria, from previous surveys, should be reclassified as Class 2 for clearance surveys.
1259   Additionally, if the historical assessment data are insufficient to clearly document that an item or area is
1260   non-impacted, the minimum classification for such materials would be Class 3.

1261   **4.5   Application of Release Guidelines**

1262   Section 4.1 discussed release guidelines for clearance and the concept of the derived concentration
1263   guideline limit for clearance (DCGL$_c$) based on dose factors, such as from NUREG-1640. This section
1264   addresses how individual DCGLs for clearance can be combined and applied when more than one
1265   radionuclide is potentially present. Options may include the use of gross activity DCGLs for surface
1266   activity compliance and use of surrogate measurements or the unity rule for volume activity compliance.

1267   Regardless of the option used to modify the DCGLs to account for multiple radionuclides, it is necessary
1268   to identify the potential radionuclides, as well as the relative ratios of these radionuclides, if a relative
1269   ratio indeed exists. Section 4.3.2 discusses the approach for determining the nature of the contamination,
1270   as well as calculating the relative ratios among the multiple radionuclides and state of equilibrium for
1271   decay series radionuclides.

1272   **4.5.1   Surface Activity Assessment when Multiple Radionuclides are Present**

1273   Surface activity DCGLs for clearance apply to the total surface activity level. For cases in which the
1274   surface contamination is entirely attributable to one radionuclide, the DCGL$_c$ for that radionuclide is
1275   used for comparison to clearance data. The clearance data may be obtained from direct measurements of
1276   surface activity, scanning with data logging, CSM surveys, etc.

1277 For situations in which multiple radionuclides with their own DCGLs are present, a gross activity
1278 DCGL$_c$ can be developed. This approach enables field measurement of gross activity (using static direct
1279 measurements or scans), rather than determination of individual radionuclide activity, for comparison to
1280 the DCGL$_c$. The gross activity DCGL for surfaces with multiple radionuclides is calculated as follows:

1281 (1)   Determine the relative fraction (f) of the total activity contributed by the radionuclide.
1282 (2)   Obtain the DCGL$_c$ for each radionuclide present.
1283 (3)   Substitute the values of f and DCGL$_c$ in the following equation.

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

1284 For example, assume that 40 percent of the total surface activity was contributed by a radionuclide with a
1285 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
1286 (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).
1287 Using the above equation,

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

1288 $$= 0.3\ Bq/cm^2\ (1,900\ dpm/100\ cm^2)$$

1289 Note that the above equation may not work for sites that exhibit surface contamination from multiple
1290 radionuclides having unknown or highly variable concentrations of radionuclides throughout the site.
1291 In these situations, the best approach may be to select the most conservative surface activity DCGL from
1292 the mixture of radionuclides present. If the mixture contains radionuclides that cannot be measured using
1293 field survey equipment, such as ^{3}H or ^{55}Fe, laboratory analyses of solid materials may be necessary.
1294

1295 Meeting with surface activity DCGLs for radionuclides of a decay series (e.g., radium, thorium, and
1296 uranium) that emit both alpha and beta radiation may be demonstrated by assessing alpha, beta, or both
1297 radiations. However, relying on the use of alpha surface activity measurements often proves problematic
1298 because of the highly variable level of alpha attenuation by rough, porous, and dusty surfaces. Beta
1299 measurements typically provide a more accurate assessment of thorium and uranium contamination on
1300 most building surfaces because surface conditions cause significantly less attenuation of beta particles
1301 than alpha particles. Beta measurements, therefore, may provide a more accurate determination of
1302 surface activity than alpha measurements.

32

1303	The relationship of beta and alpha emissions from decay chains or various enrichments of uranium
1304	should be considered when determining the surface activity for comparison with the DCGL$_C$ values.
1305	When the initial member of a decay series has a long half-life, the radioactivity associated with the
1306	subsequent members of the series will increase at a rate determined by the individual half-lives until all
1307	members of the decay chain are present at activity levels equal to the activity of the parent.
1308	This condition is known as secular equilibrium.

1309 Consider an example in which the radionuclide of concern is ^{232}Th, and all of the progeny are in secular
1310 equilibrium. Assume that a gas proportional detector will be used for surface activity measurements.
1311 The detector's efficiency is dependent upon the radionuclide mixture measured and the calibration source
1312 area (greater than 100 cm^2 area calibration sources are recommended). The ^{232}Th efficiency is calculated
1313 by weighting the individual efficiencies from each of the radionuclides present (see Table 4.3). This
1314 value is greater than 100 percent because of all of the progeny that are assumed to be in equilibrium with
1315 the ^{232}Th. It is important to recognize that if the DCGL$_C$ for ^{232}Th includes the entire ^{232}Th decay series,
1316 the total efficiency for ^{232}Th must account for all of the radiations in the decay series.

1317 **Table 4.3: Detector efficiency for the rare earth facility**
1318 **(^{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

1332 **4.5.2 Volume Activity Assessment when Multiple Radionuclides are Present**

1333 Typically, DCGLs correspond to a release criterion (e.g., a regulatory limit) in terms of dose or risk.
1334 However, in the presence of multiple radionuclides, the total of the DCGLs for all radionuclides could
1335 exceed the release criterion. In this case, the individual DCGLs would need to be adjusted to account for
1336 the presence of multiple radionuclides contributing to the total dose. One method for adjusting the
1337 DCGLs is to modify the assumptions made during exposure pathway modeling to account for multiple
1338 radionuclides. The surrogate measurements discussed in this section describe another method for
1339 adjusting the DCGL to account for multiple radionuclides when radionuclide-specific laboratory analyses
1340 of media samples or *in toto* measurements are performed. Other methods include the use of the unity rule
1341 and development of a gross activity DCGL for surface activity to adjust the individual radionuclide
1342 DCGLs.

1343 The unity rule, represented in the following expression, is satisfied when radionuclide mixtures yield a
1344 combined fractional concentration limit that is less than or equal to one:

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

1345 where

1346   C   =   concentration

1347   DCGL =   clearance guideline value for each individual radionuclide (1, 2, ... n)

1348 For the clearance of solid materials that have potential contamination with multiple radionuclides, it may
1349 be possible to measure just one of the radionuclides and still demonstrate compliance for all of the other
1350 radionuclides present through the use of surrogate measurements. In the use of surrogates, it is often
1351 difficult to establish a "consistent" ratio between two or more radionuclides. Rather than follow
1352 prescriptive guidance on acceptable levels of variability for the surrogate ratio, a more reasonable
1353 approach may be to review the data collected to establish the ratio and to use the DQO Process to select
1354 an appropriate ratio from that data. The $DCGL_C$ must be modified to account for the fact that one
1355 radionuclide is being used to account for one or more other radionuclides.

1356 The following equation illustrates how the DCGL for the measured radionuclide is modified
1357 ($DCGL_{meas,mod}$) to account for the inferred radionuclide:

1358 where

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

1359 C_{infer}/C_{meas} = surrogate ratio for the inferred to the measured radionuclide

1360 When it is necessary for the measured radionuclide to be used as a surrogate for more than one
1361 radionuclide, Equation I-14 on MARSSIM page I-32 can be used to calculate the modified DCGL for the
1362 measured radionuclide:

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

1363 where D_1 is the $DCGL_C$ for the measured radionuclide by itself, D_2 is the $DCGL_C$ for the second
1364 radionuclide (or first radionuclide being inferred) that is being inferred by the measured radionuclide.
1365 R_2 is the ratio of concentration of the second radionuclide to that of the measured radionuclide.
1366 Similarly, D_3 is the $DCGL_C$ for the third radionuclide (or second radionuclide being inferred) that is
1367 being inferred by the measured radionuclide, and R_3 is the ratio of concentration of the third radionuclide
1368 to that of the measured radionuclide.

34

1369 Recall that the benefit of using surrogates is the avoidance of costly laboratory-based analytical methods
1370 to detect radionuclides with weakly penetrating radiation. Surrogates usually emit γ rays, which enable
1371 the use of noninvasive and nondestructive methods. The surrogates come in two forms: (1) surrogates by
1372 virtue of a decay series, and (2) surrogates by virtue of association. The difficulty with surrogates that
1373 are part of a series is that a time for sufficient number of half-lives of the longest lived progeny that
1374 intervenes between and including itself and its parent must pass in order to establish secular equilibrium.
1375 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
1376 of 5.75 years. In the case ^{238}U and ^{226}Ra, the half-lives of the intervening progeny are relatively short.
1377 However, ^{226}Ra possesses a special problem because it decays into ^{222}Rn, which is a noble gas that can
1378 escape the matrix and disrupt equilibrium. Radionuclides that are not part of a decay series have the
1379 potential to be surrogates because they are produced by the same nuclear process (usually fission or
1380 activation) and have similar chemical properties and release mechanisms. However, this type of
1381 surrogate needs some special attention because there must be a consistent ratio between the measured
1382 radionuclide and surrogate, which is not always easy to demonstrate. For example, in the case of
1383 reactors, ^{60}Co can be used as a surrogate of ^{55}Fe and ^{63}Ni because both are activation-corrosion products
1384 with similar chemical properties. Similarly, ^{137}Cs can be used as a surrogate for the β-emitting ^{90}Sr
1385 because both are fission products and are generally found in soluble cationic forms. While ^{137}Cs has
1386 been suggested as a possible surrogate for ^{99}Tc, it must be noted that ^{99}Tc does not have different
1387 chemical properties and, in power reactors, it has different release mechanisms. For a further discussion
1388 of surrogates and establishing ratios, see MARSSIM (1997) and Best and Miller (1987).

1389 ## 4.6   Measurability of Contamination

1390 Detection limits for field survey instrumentation are an important criterion in the selection of appropriate
1391 instrumentation and measurement procedures. For the most part, detection limits need to be determined
1392 in order to evaluate whether a particular instrument or measurement procedure is capable of detecting
1393 residual activity at the regulatory release criteria (DCGLs). For example, the MARSSIM recommends
1394 that the minimum detectable concentration (MDC) should be sufficiently less than the DCGL (e.g., no
1395 greater than 10 to 50 percent of the DCGL). This is a reflection of two concerns. First, when calculated
1396 a priori, the MDC frequently tends to be optimistic in that some factors that may adversely impact
1397 detection sensitivity are either unknown or not included (e.g., surface roughness, interfering
1398 radionuclides, or radiations). Second, the objective is not simply to detect whether radioactivity exists at
1399 levels approaching the DCGL, but to quantify the actual concentration level within a reasonable overall
1400 uncertainty.

1401 Sections 4.6.1 and 4.6.2 address the measurability of contamination under the general survey approaches
1402 of (1) static measurements and (2) scanning, respectively. Static MDCs are calculated when the
1403 clearance survey approach includes conventional direct measurements of surface activity, *in toto*
1404 measurements, or laboratory analyses of media samples. Scan MDCs are calculated when the clearance
1405 survey approach includes scanning with conventional detectors, or when using automated scanning
1406 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.

36

1445   The following equation is used to calculate the MDC for surface activity assessments using conventional
1446   survey instrumentation (NRC 1998a):

$$MDC = \frac{3 + 4.65 \sqrt{C_B}}{KT} \tag{4-2}$$

1447   where C_B is the background count in time, T, for paired observations of the sample and blank. The
1448   quantities encompassed by the proportionality constant, K, include the instrument efficiency, surface
1449   efficiency, and probe geometry. Based on the radionuclides of concern, specific instrument and surface
1450   efficiencies are used to calculate the static MDC for surface activity measurements. The MDC is also a
1451   function of the surface material background level and, therefore, varies with the nature of the surfaces
1452   being surveyed.

1453   The detection and detectability of contamination when using other than the conventional survey approach
1454   must also be considered. Tritium (^{3}H) and ^{14}C create a significant challenge for detection (because of the
1455   associated low instrument efficiency). They each emit a low-energy β radiation, and they are not
1456   amenable to the surrogate approach. Similarly, ^{63}Ni and ^{99}Tc are somewhat difficult to detect because
1457   they too have primary radiations of low-energy betas. Conversely, ^{60}Co, Cs-134, and ^{137}Cs (via Ba-137m)
1458   are easily detected because of their intense and rather energetic gamma-rays and readily-measured beta
1459   radiations. T he evaluation of detectability for these seven radionuclides is more or less independent of
1460   the matrix and nature of the contamination. In general, all of the radionuclides (with the exception of ^{3}H)
1461   can be detected with hand-held devices using standard survey methods. The issue is whether hand-held
1462   devices and standard survey methods can detect these radionuclides, separately or in combination, at the
1463   levels established for release.

1464   Therefore, the recipe to calculate the MDC for any measurement method (such as for an *in toto* technique
1465   or laboratory analysis) is to determine the detection limit, relevant efficiencies, and sample size for the
1466   given instrument and measurement protocol. For some of the more common (conventional) techniques of
1467   measuring radionuclides and materials, these quantities have been either measured, calculated, or
1468   estimated and MDCs are available in the literature (ANSI 1999, MARSSIM 1997, NRC 1998a, EC 1998,
1469   and Goles *et al.* 1991). The reader should note, however, that the MDC provided in these references
1470   apply only to the situation described and must not be construed to be a universal MDC for a particular
1471   instrument or protocol. Rather, they should be viewed only as a general measure of the capability of the
1472   instruments for the application described.

1473   **4.6.2   Scanning-Based MDCs**

1474   Scanning-based MDCs must also be assessed in order to appropriately design the clearance survey
1475   approach. Relevant information on scanning-based MDCs for conventional survey approaches exists in
1476   the MARSSIM (Section 6), NUREG-1507, and Abelquist and Brown, 1999. In general, when planning
1477   surveys, one must often consider minimum detectable count rates (MDCRs) in order to evaluate the
1478   effectiveness of a given scan. An MDCR is an a priori estimate of the signal level that a real surveyor is
1479   expected to recognize as having a signal-to-noise ratio that is distinctly above the ambient detector
1480   background noise. In general, the MDCR is defined as the detector signal level, or count rate for most
1481   equipment, that a surveyor is likely to flag as being "greater than background." The MDCR will depends
1482   on a number of factors, including scan speed, detector type, detector background, and surveyor
1483   performance.

| 1484 | **4.6.2.1 Hand-Held Detector Scan MDCs** |

1485 To illustrate the calculation of scanning-based MDCs, the scanning sensitivity for conventional hand-
1486 held survey instruments is provided for materials being cleared from a gaseous diffusion facility.
1487 [Note: Example 2 in Section 5 of this report pertains to nuclear power plants.] Assuming that a gas
1488 proportional detector is used as the primary instrument used for surface scanning, the instrument
1489 efficiency for scanning is slightly less than that used for static measurements. This is because the
1490 detector is not directly on the surface of the material during scanning. [Note: The fact that the detector
1491 is being moved over the source is separately accounted for in the scan efficiency by determining the
1492 observation interval. The instrument efficiency for scanning is determined based on the detector-surface
1493 geometry for the observation interval, which is on the order of seconds.] Table 4.4 shows the
1494 determination of detection efficiency for a gas proportional detector used for scanning.

1495 **Table 4.4: Detector efficiency when scanning for GDP-enriched uranium (1.2%) and ^{99}Tc**
1496 **using a gas proportional detector (0.4 mg/cm^2 window)**

	Radionuclide	Radiation/Average Energy (MeV)	Activity Fraction	ε_i	ε_s	Weighted Efficiency
1498	^{99}Tc	Beta/0.085	0.7082	0.30	0.25	5.3×10^{-2}
1499	^{238}U	Alpha/4.2	0.1077	0.32	0.25	8.6×10^{-3}
1500	^{234}Th	Beta/0.0435	0.1077	0.20	0.25	5.4×10^{-3}
1501	^{234m}Pa	Beta/0.819	0.1077	0.58	0.50	3.1×10^{-2}
1502	^{234}U	Alpha/4.7	0.1728	0.32	0.25	1.4×10^{-2}
1503	^{235}U	Alpha/4.4	0.0084	0.32	0.25	6.7×10^{-4}
1504	^{231}Th	Beta/0.0764	0.0084	0.29	0.25	6.1×10^{-4}
1505			**Total Weighted Efficiency**			**0.11**

1506   The scan MDC for structure surfaces may be calculated as

$$scan\ MDC = \frac{MDCR}{\sqrt{p}\ \varepsilon_i\ \varepsilon_s} \tag{4-3}$$

1507   where the minimum detectable count rate (MDCR), in counts per minute, can be written as

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

1508   where d' = detectability index (the value can be obtained from Table 6.5 in the MARSSIM),
1509 b_i = background counts in the observation interval,
1510 i = observational interval (in seconds), based on the scan speed and areal extent of the contamination
1511   (usually taken to be 100 cm²),
1512 ε_i is the instrument or detector efficiency (unitless),
1513 ε_s is the surface efficiency (unitless), and
1514 p is the surveyor efficiency (usually taken to be 0.5).

1515   Consider an example that involves determining the gas proportional scan MDC for the GDP mixture on
1516   concrete slabs. The scan MDC will be determined for a background level of 400 cpm and a 1-second
1517   observation interval. For a specified level of performance at the first scanning stage of 95-percent "true
1518   positive" rate and 25-percent "false positive" rate, d' equals 2.32 (from Table 6.5 in the MARSSIM), and
1519   the MDCR is calculated as follows:

1520 $$b_i = (400\ cpm)(1\ s)(1\ min/60\ s) = 6.67\ counts,$$

1521 $$s_i = (2.32)(6.67)^{\frac{1}{2}} = 6.0\ counts,\ and$$

1522 $$MDCR = (6.0\ counts)[(60\ s/min)/(1\ s)] = 360\ cpm.$$

1523   Using a surveyor efficiency of 0.5 and the total weighted efficiency determined in Table 9.1 (0.11), the
1524   scan MDC is calculated as

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

1525   A Geiger-Mueller (GM) detector is often used to scan material surfaces that are difficult (or impossible)
1526   to access using the larger gas proportional detector. The efficiency of a GM detector in scanning this
1527   radionuclide mixture can be determined in a manner similar to that used in Table 4.4. It is important to
1528   note, however, that the scan MDC calculations usually require the assumption that the instrument
1529   efficiencies are determined relative to a 100-cm² calibration source to yield the appropriate units
1530   (dpm/100 cm²). This is in contrast to the static MDC equation, which uses a physical probe area
1531   correction in the calculation of surface activity.

1532 Table 4.5 provides instrument efficiencies that correspond to a 100-cm² calibration source, without
1533 reducing the 2π emission rate for the smaller area subtended by the GM detector. [Note: This is precisely
1534 what would be performed for static measurements of surface activity.] In other words, as long as 100
1535 cm² is used as the size of the postulated small, elevated area, and the instrument efficiency is calculated
1536 for the same area, there is no need for a probe area correction in the scan MDC equation.

1537 **Table 4.5: Detector efficiency when scanning for GDP-enriched uranium (1.2%) and ^{99}Tc**
1538 **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**

1548 As an example, consider evaluating the scanning-based MDC for the gaseous diffusion plant (GDP)
1549 mixture on stainless-steel materials. The scanning-based MDC will be determined for a background level
1550 of 70 cpm and a 1-second interval using a GM detector. For a specified level of performance at the first
1551 scanning stage of 95-percent true positive rate and 25-percent false positive rate, d' equals 2.32 (from
1552 Table 6.5 in the MARSSIM), and the MDCR is calculated as follows:

1553 $$b_i = (70 \ cpm)(1 \ s)(1 \ min/60 \ s) = 1.2 \ counts,$$

1554 $$s_i = (2.32)(1.2)^{\frac{1}{2}} = 2.5 \ counts, \ and$$

1555 $$MDCR = (2.5 \ counts)[(60 \ s/min)/(1 \ s)] = 150 \ cpm.$$

1556 Using a surveyor efficiency of 0.5 and the total weighted efficiency determined in Table 9.2 (0.018), the
1557 scan MDC is calculated as

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

1558   **4.6.2.2 Conveyor Survey Monitor Scan MDCs**

1559   The scan MDC for a CSM can be estimated using Equation 4-1, with some modification to account for
1560   the automated nature of a CSM. That is, the parameters that impact the CSM scan MDC include the
1561   detection limit, efficiency, and sample size. The detection limit is based on the background counts
1562   obtained over the counting interval and the acceptable rate of true (correct detection) and false positives.
1563   The background level depends on the nature of the material, while the counting interval is a function of
1564   both the detector's field-of-view and the system belt speed (i.e., it establishes the length of time that the
1565   detector(s) can respond to a fixed length of material). Basically, the MDCR can be calculated for the
1566   CSM in much the same manner as it is for conventional scans, with the primary difference being that
1567   automated systems interpret the signal stream (data) using a computer-based analysis algorithm rather
1568   than by calculation (Equation 4-4).

1569   Sample or survey unit size is a function of the belt geometry, speed (which establishes the observation
1570   interval), and the detector's field-of-view and, therefore, has a fundamental impact on the scanning
1571   detection limit (cpm) and MDC (Bq/g) of a CSM. The detection efficiency for a CSM depends on the
1572   detector characteristics, nature of the contamination, the material being surveyed, and source-to-detector
1573   geometry. Modeling was performed to support the determination of beta detection efficiencies for
1574   automated scanning systems, as further discussed in Section 5.3.

1575   **4.6.2.3 Empirical Determinations of Scanning-Based MDCs**

1576   Empirical determination of scanning-based MDCs can serve as an alternative to calculation. That is, it is
1577   possible to design experiments to assess (and empirically determine) the scanning-based MDCs for
1578   particular survey instruments and scan procedures. A number of researchers, as well as R&D
1579   professionals, have developed mockups of surfaces with contamination to determine scanning-based
1580   MDCs. For instance, in a study by Goles *et al.* (1991), empirical results included MDCRs as a function
1581   of background levels: 305 net cpm detected in 50-cpm background level, 310 cpm in 250-cpm
1582   background, and 450 cpm in 500-cpm background. It is important to note that these MDCRs were quoted
1583   for detection frequencies of 67 percent (compared to the usual 95 percent). Empirical assessments of
1584   scanning-based MDC can also be valuable for determining the scanning capabilities of specific survey
1585   technicians.

1586   The uncertainty in the scanning-based MDCs calculated using the approaches described in this section
1587   should be viewed in the context of their use. That is, scanning-based MDCs are used to help design the
1588   clearance survey approach, and should represent a "reasonable estimate" of the activity concentration
1589   that can be detected when scanning. In other words, while the scanning-based MDC should be carefully
1590   assessed, it is important to remember that such MDCs are inherently subject to uncertainties (e.g., human
1591   factors, unknown characteristics of contamination prior to survey, variable background levels, etc.).
1592   Recognizing this uncertainty in the scanning-based MDCs, it is worthwhile to consider additional means
1593   of evaluating these values.

1594   Empirical evaluation of scanning-based MDCs can also be an important validation tool. This validation
1595   is performed by assessing the contamination levels that are flagged on solid materials during scanning.
1596   These radionuclide concentrations are evaluated by direct measurements or laboratory analyses, and the
1597   concentrations at the lower end of the range of results should provide a reasonable estimate of the
1598   scanning-based MDC achieved. That is, an empirical evaluation might indicate that the lower values in
1599   the range represent a ballpark estimate of the scanning-based MDC. Obviously, increasing the number of
1600   samples that are actually flagged during the scan, as well as the number of subsequently measured
1601   samples will improve the accuracy of this empirical assessment of scanning-based MDCs.

4.7   Inaccessible Areas

1603   A question that often arises is how to handle the release of materials that have inaccessible areas that may
1604   have contamination. If the material surfaces are inaccessible, then by definition, it is not possible to
1605   demonstrate that release criteria have been satisfied using conventional survey activities. In such cases,
1606   a couple of options exist. First, the material might not be released for unrestricted use; that is,
1607   the surveyor might conclude that since surfaces are not accessible, they must be assumed to have
1608   contamination at levels greater than the release criteria. Thus, the materials might be disposed of as
1609   radioactive waste. In fact, this approach has been used to deal with materials that have inaccessible
1610   surfaces.

1611   A second alternative might be to make the surfaces accessible, either by cutting or dismantling the
1612   material, or by using specialized survey equipment (e.g., small detectors). This option requires additional
1613   resources beyond those required for conventional clearance surveys. The discussion throughout this
1614   report suggests a number of research opportunities for handling materials that have inaccessible areas.

4.7.1 Inaccessible Material Scenarios

1616   It is important to recognize the various inaccessible material scenarios that can occur during the clearance
1617   of materials. Perhaps the most common scenario is when contamination exists on the interior surfaces of
1618   scrap equipment, such pumps, motors, and other equipment. These items can become contaminated
1619   through a number of mechanisms, including their operation in airborne contamination areas where air is
1620   drawn into the equipment, thereby contaminating internal surfaces. Similarly, contaminated lubricating
1621   oil can spread contamination to a number of components within the scrap equipment. Thus, because of
1622   the small openings on these items, it is nearly impossible to use conventional survey activities to assess
1623   the potential for internal contamination.

1624   Another inaccessible material scenario involves contamination on the interior surfaces of pipes that are
1625   difficult to access, such as buried or embedded pipes. Buried and embedded pipes may become
1626   contaminated as a result of their function of transporting radioactive liquids or gases. Buried pipes are
1627   usually at some depth beneath the soil surface and cannot be accessed unless they are excavated. Process
1628   piping, such as that associated with nuclear power reactor systems, can be embedded in concrete, which
1629   further complicates the assessment. In addition, the small diameter of embedded piping typically makes
1630   it extremely difficult to access the interior surfaces.

1631   One final inaccessible material scenario includes some of the material surfaces in a scrap metal (or other
1632   material) pile. This complex geometry is somewhat different from the first two scenarios, in that these
1633   surfaces can be made accessible, but separating the materials for examination might be considered too
1634   labor-intensive to warrant conventional clearance surveys. Therefore, it might be worthwhile to consider
1635   releasing a pile of scrap metal by taking *in situ* gamma spectrometry measurements of the scrap metal
1636   pile. In this case, some of the scrap metal surfaces are considered to be inaccessible because they do not
1637   directly contribute to the detector's response. However, provided that a sufficient fraction of gamma
1638   radiation from the contamination is detected, *in situ* gamma spectrometry might provide a reasonable
1639   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.

1665

1666 As discussed in previous sections of this report, the predominant factor in determining how much effort
1667 should be expended in conducting a clearance survey to release the given solid material is the material's
1668 potential to have contamination in excess of the release criteria. That is, the closer the radionuclide
1669 concentration is to the release criteria, the greater the degree of survey effort that should be expended to
1670 release the material. Process knowledge and characterization activities are used to estimate the
1671 material's contamination potential. The MARSSIM survey approach can be applied to clearance of
1672 materials, by designating the materials as Class 1, 2, or 3 based on each material's contamination
1673 potential.

1674 The decision to implement a particular clearance survey approach depends on the material
1675 characteristics, nature of the contamination, detectability of the emitted radiation, and availability of
1676 survey instrumentation. The reader is encouraged to revisit the DQO Process discussion in Section 3
1677 before selecting a particular clearance survey approach.

1678 ## 5.1   Background Measurements

1679 Release criteria for the clearance of solid materials may be expressed as the concentration of
1680 radioactivity that exceeds background levels. Consequently, an important aspect of clearance surveys is
1681 to adequately assess the background levels associated with specific solid materials. This can be achieved
1682 by selecting background reference materials that are non-impacted (i.e., materials that have no reasonable
1683 potential to be contaminated) and representative of the solid materials being considered for release.
1684 Background measurements are also necessary to calculate the MDC of the selected clearance survey
1685 approach.

1686 The number and type of background measurements that are necessary to support the design of clearance
1687 surveys depends on the particular clearance survey approach, the survey instrument, and the nature of the
1688 solid material. The number of background measurements should be based on the requirements of the
1689 statistical test (if a statistical test is used) or on the DQO Process. [Note: If background levels are a
1690 small fraction of the release criteria, one might consider ignoring the background in demonstrating
1691 compliance. Refer to Section 6 for more information on this conservative practice.]

1692 Background surface activity levels for instrumentation used to measure beta radiation can be expected to
1693 vary in response to a number of influences. The primary variance is attributable to survey conditions
1694 (such as gamma contributions from ambient environmental and building materials), while variations in
1695 the solid materials themselves and temporal fluctuations attributable to sources such as radon can add
1696 additional variance. Backgrounds for alpha-measuring instrumentation can be expected to vary primarily
1697 as a result of natural material contributions and temporal variations in radon, where radon concentrations
1698 tend to be elevated. In all cases, surveys should be performed in areas where instrument backgrounds
1699 from ambient radiation levels allow the detection sensitivity requirements to be met.

1700 Appropriate background data sets should be collected for each detector type, such that all significant
1701 sources of variance are properly accounted for. Background measurements should be collected on
1702 material types representing items that will be surveyed and should also account for fluctuations within
1703 the area where surveys will be performed. Although not required, it is suggested that data sets be formed
1704 for beta-gamma detection equipment by collecting measurements on non-impacted solid materials at
1705 varying locations to establish a good representation of background variance. For those areas where radon
1706 progeny or other external influences on detector response may pose a significant problem, it is suggested
1707 that the materials be moved elsewhere before being surveyed.

45

1708 Dependent upon site- and material-specific considerations, the background data sets may be pooled or
1709 analyzed individually according to material types. The mean and variance of the background
1710 measurements would then be calculated for the complete data set(s). At a minimum, materials with very
1711 dissimilar background radiological properties should not be grouped together. For example, the
1712 background means for various metal types generally should not differ by greater than 30 percent in order
1713 to be considered for grouping.

1714 Background measurements for the conveyor survey monitor should be determined for each type of
1715 non-impacted solid material being considered for release. For example, non-impacted soil could be run
1716 through the CSM repeatedly to develop a background database for that material. (Refer to Section 5.3.)

1717 At least one ambient background measurement for the *in situ* gamma spectrometer (ISGS) should be
1718 performed in the area where clearance surveys will be conducted. This background spectrum should be
1719 collected for a sufficient time to provide the necessary sensitivity for the radionuclide(s) and material
1720 being considered for release. (Refer to Section 5.4.) Provided that the radionuclide(s) being measured
1721 are not naturally present in the solid material being assessed, additional ISGS background measurements
1722 are unwarranted. By contrast, when the radionuclide(s) being measured are naturally present in the solid
1723 material (e.g., uranium, thorium), a number of background measurements should be performed on the
1724 same type of non-impacted solid materials to permit comparison to the materials being released. It is
1725 likely that the number of background measurements required in this case will be based on WRS test data
1726 needs.

1727 **5.2   Survey Approach Using Conventional Instrumentation**

1728 In general, survey methods that use conventional instrumentation can be classified into three survey
1729 categories, which are commonly known as (1) scanning, (2) direct measurements of surface activity, and
1730 (3) smear and miscellaneous sampling. These survey approaches are based on the use of hand-held,
1731 portable field survey instruments, which should have a minimum measurement detection ability, typically
1732 referred to as minimum detectable concentration (MDC), that is less than applicable release criterion
1733 (DCGL$_c$). For difficult-to-detect radionuclides, the survey should use surrogates, or collection methods
1734 and laboratory analysis techniques, that have minimum detection abilities that are less than applicable
1735 release limits for media samples.

1736 **5.2.1   Survey Instrumentation**

1737 To maintain sufficient survey instrument detection capabilities, release surveys should be conducted in
1738 areas with low background radiation levels. Survey instrument parameters to consider include count
1739 times (for direct measurements of surface activity), background levels, and detection efficiencies to
1740 determine if they yield MDCs that are sufficiently below the release criteria to allow unambiguous
1741 decisions regarding the acceptability for release. Section 4.6 provides detailed information on
1742 measurability issues.

1743 All measurement instrumentation should be calibrated and monitored for performance in accordance with
1744 accepted standards applicable to performing surveys before releasing materials from radiological control.
1745 Survey instruments typically include gas proportional, GM, ZnS, and NaI scintillation detectors, coupled
1746 to ratemeters or ratemeter-scalers with audible indicators. Calibration and efficiency data are necessary
1747 to ensure that individual detectors are capable of meeting the minimum performance specifications, as
1748 previously discussed.

1749   **5.2.2   Survey Activities (Measurement Methods)**

1750   As previously mentioned, conventional clearance survey methods include scanning, direct measurements,
1751   and sampling surveys. Given these options, the measurement techniques for a given clearance survey
1752   should be selected on the basis of the radionuclides (radiations) of concern and appropriately sensitive
1753   instrumentation should be selected for field use. The types of measurements, specific portable
1754   instrumentation, and specific measurement methods should be consistent with the appropriate standard
1755   operating procedures (SOPs) and presented in clearance survey plans.

1756   **5.2.2.1 Scanning and Direct Measurements of Surface Activity**

1757   Surface activity surveys are performed using both scans and static, integrated direct measurements.
1758   Clearance materials should be assessed on the basis of process knowledge and other historical
1759   information, and should also be scanned for alpha, beta, or gamma radiation according to the nature of
1760   the potential radionuclides. When pausing during scans, a surveyor should compare the resulting signal
1761   to the expected background level to determine whether the observation indicates an elevated radiation
1762   level. Any locations of elevated direct radiation should be marked for further investigation, which should
1763   include judgmental measurements of surface activity. Scans should be performed using survey
1764   instruments that have been appropriately calibrated for the radiations present. Appropriate investigation
1765   levels should be established and implemented for evaluating elevated radiation.
1766
1767   Direct measurements of surface activity should be performed for materials being considered for release.
1768   The type of surface activity measurement (gross alpha or gross beta) should be selected on the basis of
1769   the potential radionuclides present. Direct measurements should be performed using appropriately
1770   calibrated survey instruments, including gas proportional, GM, and ZnS detectors coupled to ratemeter-
1771   scalars. Material-specific background measurements should also be obtained for each material type.
1772   (Refer to Section 5.1.) In addition, all measurement locations should be properly documented on detailed
1773   survey maps.

1774   **5.2.2.2 Smear and Miscellaneous Sampling**

1775   Materials considered for release may include miscellaneous samplings, such as smear, residue, and/or
1776   swab samples, with the methods chosen on the basis of the inaccessibility of some surfaces. [Note:
1777   Given the significant variations in smear collection efficiencies, smear results are usually considered to
1778   be semi-quantitative]. Smear samples for the determination of removable activity may be collected at
1779   direct measurement locations. Residue and/or swab samples may also be collected at specific locations
1780   where the surface area is inaccessible for direct measurements.

1781   The selected frequency of sampling should be based on the appropriate classification (based on surface
1782   area, minimum number per item), and measurement locations should be properly documented on detailed
1783   survey maps. Procedures and equipment used for sampling (smears, Q-tips, swabs, etc.) should be
1784   appropriate for the assessment of the contamination. A comprehensive reference on the use and purpose
1785   of smears is Frame and Abelquist, 1999.

1786 ### 5.2.3 Clearance Survey Designs Using Conventional Instrumentation

1787 The following sections discuss various applications of the conventional survey approach based primarily
1788 on the capability of the survey instrumentation. These conventional survey applications include (1)
1789 scanning-only, (2) scanning and direct measurements, and (3) statistically based sampling. [Note: In the
1790 following discussion, the statistical term "sample" refers to both direct measurements of surface activity
1791 and media samples (smears, soil, etc.)].

1792 As mentioned in Section 1.3, this report stresses the use of scanning to release materials whenever the
1793 scan MDC is sufficiently sensitive. As such, the conventional survey approaches discussed in the
1794 following sections are ordered in terms of relative ease in performing survey activities. That is,
1795 scanning-only is the most direct survey approach, followed by scanning and direct measurements, and
1796 lastly statistically based sampling. The NRC staff recognizes that constraints in the availability of
1797 specific survey instrumentation, in terms of scan sensitivity or ability to automatically record scanning
1798 results, may limit the conventional survey options that are available to the licensee. However, the reader
1799 should note that each of the techniques discussed in Sections 5.2.3.1 – 5.2.3.3 is equally acceptable for
1800 demonstrating the acceptable release of materials.

1801 ### 5.2.3.1 Scanning-Only

1802 This clearance survey approach can be used to release solid materials only when two conditions are met.
1803 First, the survey instrumentation must exhibit sufficient scan sensitivity. That is, the scan MDC must be
1804 less than the $DCGL_C$. (Refer to Section 4.6 for guidance on determining the scan MDC for comparison
1805 to the $DCGL_C$.) Second, the survey instrumentation must have the capability to automatically document
1806 the survey results, which may be accomplished using a data logger or similar device. This condition
1807 cannot be satisfied by the surveyor manually recording the scan results; automatic documentation is
1808 much more reliable. (Manually recorded scan results are a function of the surveyor's memory.)

1809 The scan coverage should be graded based on the material's classification. That is, 100 percent of
1810 surfaces should be scanned for Class 1 materials, 50 to 100 percent for Class 2, and 10 to 50 percent for
1811 Class 3. The size of the material survey unit may also be a function of the material's classification. That
1812 is, the amount of material comprising Class 1 survey units may be smaller than either Class 2 or 3 survey
1813 units. The size of all survey units may have to be consistent with any dose modeling used to obtain the
1814 $DCGL_C$.

1815 Whenever less than 100 percent of the survey unit is scanned, there is the potential to reintroduce
1816 uncertainty attributable to spatial variability, because the entire population of measurement locations is
1817 not being sampled and the scanning coverage is not random. These factors are expected to be of minimal
1818 consequence in Class 2 and Class 3 survey units because the level of contamination is expected to be
1819 fairly low and not as spotty as in Class 1 survey units. Nonetheless, with less than 100-percent scan
1820 coverage, these measurements should be considered a potentially biased sample, and the resulting
1821 average will be a somewhat biased estimate of the population average.

1822 **5.2.3.2 Scanning and Direct Measurements (and Media Samples)**

1823 This clearance survey approach is possible when the survey instrumentation exhibits sufficient scan
1824 sensitivity (i.e., the scan MDC is less than the $DCGL_C$), but the survey instrumentation does not have the
1825 capability to automatically document the survey results. In this situation, a number of direct
1826 measurements (or media samples) are performed, primarily to document the scan results. The number of
1827 these measurements should be determined using the DQO Process, and may be determined using the
1828 statistically based sampling design discussed in Section 5.2.3.3.

1829 Again, the scan coverage should be graded based on the material's classification. That is, 100 percent of
1830 surfaces should be scanned for Class 1 materials, 50 to 100 percent for Class 2 and 10 to 50 percent for
1831 Class 3. The size of the material survey unit may also be a function of the material's classification. That
1832 is, the amount of material comprising Class 1 survey units may be smaller than either Class 2 or 3 survey
1833 units. Again, the size of all survey units may have to be consistent with any dose modeling used to
1834 obtain the $DCGL_C$.

1835 **5.2.3.3 Statistically Based Sampling**

1836 This clearance survey approach is necessary when the survey instrumentation does not exhibit a
1837 sufficient scan sensitivity (i.e., the scan MDC is greater than the $DCGL_C$). In this instance, scanning is
1838 not capable of demonstrating compliance with the release criteria. Therefore, it is necessary to design the
1839 conventional clearance survey based on a statistical sample size. Scans are still performed to identify
1840 contamination that may exceed the scan MDC, recognizing that areas of contamination falling between
1841 the $DCGL_C$ and the scan MDC in concentration may not always be detected. The scan coverage should
1842 be graded on the basis of the material's classification. That is, 100 percent of surfaces should be scanned
1843 for Class 1 materials, 50 to 100 percent for Class 2, and 10 to 50 percent for Class 3. The size of the
1844 material survey unit may also be a function of the material's classification. That is, the amount of
1845 material comprising Class 1 survey units should be smaller than either Class 2 or 3 survey units. The size
1846 of all survey units should be consistent with any dose modeling used to obtain the $DCGL_C$.

1847 In most cases, the statistical tests used in the MARSSIM are recommended, and for the same reasons.
1848 The criteria for choosing between the Sign test and the Wilcoxon Rank Sum (WRS) test are also the
1849 same. In general, when the radionuclide is not in background (or its background concentration is
1850 negligible) and radionuclide-specific measurements are made, the Sign test is used; otherwise, the WRS
1851 test is used. These nonparametric statistical tests, described below, can be used for both surface activity
1852 assessments and volumetric concentrations in materials. As discussed in Section 3.6, there are two
1853 possible scenarios under which these tests may be conducted. In Scenario A, the survey data are tested
1854 against a specified activity, known as the $DCGL_C$, to determine whether the concentration in the material
1855 survey unit exceeds that value. In Scenario B, the criterion is that no contamination is allowed in
1856 materials that are to be released from radiological controls.

1857   <u>One-Sample Statistical Test (Sign Test)</u>

1858   The Sign test is designed to detect whether there is contamination in the material survey unit in excess of
1859   the $DCGL_C$. This test does not assume that the data follow any particular distribution, such as normal or
1860   log-normal. If any measurement exceeds this $DCGL_C$, additional investigation is recommended, at least
1861   locally, to determine the actual areal extent of the elevated concentration.

1862   The following formal null and alternative hypotheses are tested by the Sign test under Scenario A:

1863   <u>Null Hypothesis</u> -
1864 H_0: The median concentration of contamination in the material survey unit is greater than the $DCGL_C$

1865   *versus*

1866   <u>Alternative Hypothesis</u>
1867 H_a: The median concentration of contamination in the material survey unit is less than the $DCGL_C$

1868   The null hypothesis is assumed to be true unless the statistical test indicates that it should be rejected in
1869   favor of the alternative hypothesis. The null hypothesis states that the probability of a measurement less
1870   than the $DCGL_C$ is less than one-half (i.e., the 50th percentile, or median, is greater than the $DCGL_C$).
1871   Note that some individual survey unit measurements may exceed the $DCGL_C$ even when the survey unit
1872   as a whole meets the release criteria. In fact, a survey unit average that is close to the $DCGL_C$ might have
1873   almost half of its individual measurements greater than the $DCGL_C$. Such a material survey unit may still
1874   not exceed the release criteria.

1875   The assumption is that the survey unit measurements are independent random samples from a symmetric
1876   distribution. If the distribution of measurements is symmetric, the median and the mean are the same.
1877   To the extent that the mean may be larger than the median, there should be some areas of larger
1878   concentration that cause the distribution to be skew. When that is the case, they will be identified by
1879   scanning, and will trigger appropriate investigation levels as described in Section 6. This is the reason
1880   for combining direct measurements with scans in the survey design.

1881   The hypothesis specifies a release criterion in terms of a $DCGL_C$. The test should have sufficient power
1882   ($1-\beta$, as specified in the DQO Process) to detect residual radioactivity concentrations at the lower bound
1883   of the gray region (LBGR). The LBGR should be set at the expected mean contamination level for the
1884   material survey unit. If σ is the standard deviation of the measurements in the material survey unit, then
1885 Δ/σ expresses the size of the shift (i.e., $\Delta = DCGL_C - LBGR$) as the number of standard deviations that
1886   would be considered "large" for the distribution of measurements in the survey unit. Table 5.5 in the
1887   MARSSIM provides sample sizes for the Sign test as a function of relative shift and Type I and II
1888   decision errors.

50

1889   If the criterion specified for controlling the release of material is that there must be no contamination, the
1890   clearance survey requires a different approach, similar to Scenario B described in NUREG-1505. The
1891   following formal null and alternative hypotheses are tested by the Sign test under Scenario B:

1892   Null Hypothesis
1893 H_0: The median concentration of contamination in the material survey unit is zero.

1894   *versus*

1895   Alternative Hypothesis
1896 H_a: The median concentration of contamination in the material survey unit is greater than the upper
1897   bound of the gray region (UBGR).

1898   As in Scenario A, in order to design a survey to test the null hypothesis for Scenario B, it is necessary to
1899   specify a gray region. Since no contamination is the criterion, the LBGR is zero, but it is still necessary
1900   to specify the UBGR. This is essential for determining an appropriate sample size, and for specifying the
1901   needed measurement sensitivity (i.e.,MDC, as discussed in Section 9.1). The width of the gray region, Δ
1902   = UBGR - LBGR = UBGR - 0 = UBGR. If σ is the standard deviation of repeated "blank" measurements
1903   (i.e., measurements on material that is known to contain no contamination), Δ/σ expresses the width of
1904   the gray region as a relative shift. Table 5.5 in the MARSSIM shows that when this relative shift falls
1905   below 1, the sample size required for the test increases dramatically. For example, if $\Delta/\sigma = 1$, and the
1906   DQOs for the Type I and Type II error rates, $\alpha = \beta = 0.05$, 29 measurements are required. If $\Delta/\sigma = 0.5$, 89
1907   measurements are required. If Δ/σ falls as low as 0.1, more than 2,000 measurements are required. Thus,
1908   it is generally recommended that the relative shift Δ/σ be between 1 and 3. Increasing the relative shift
1909   much above 3 does not appreciably reduce the required number of measurements.

1910   There is a direct connection between the UBGR and the MDC. For every instrument and procedure,
1911   there is an associated MDC, which is usually defined to be the concentration that will be detected with a
1912   95-percent probability when it is present, while limiting to 5 percent the probability that a detection
1913   decision will be made when there is actually no contamination. (Refer to Section 4.6.) This decision is
1914   made separately for each measurement. It is a test of the hypothesis that there is no contamination at that
1915   single location on the material. The detection decision is based on whether the instrument signal is above
1916   a critical level corresponding to a concentration equal to about one-half the MDC. The MDC is usually 3
1917   to 4 times the measurement uncertainty, σ. Since the MDC should not exceed the UBGR, the smallest
1918   practical value of the UBGR occurs when it equals the MDC. *Thus, an essential part of the DQO*
1919   *process for this case is setting the required MDC.* This ultimately defines the gray region, the sample
1920   size, and the effort that should be expended to find any contamination that might be present. When the
1921   UBGR = MDC, Δ/σ is about 3. Table 5.5 in the MARSSIM then indicates that between 8 and 20
1922   samples must be taken, depending on the Type I and Type II error rates that are set.

1923   In practice, the very use of the Sign test implies that radionuclide-specific measurements are being made
1924   to detect radionuclides that do not appear in background. Thus, any *unambiguously* detected positive
1925   concentration measured anywhere on the material obviously shows that it does not meet the criterion of
1926   no contamination, even though the *median* added concentration may be zero. This is analogous to the
1927   procedure used in the MARSSIM, namely, if the average concentration exceeds the release criterion,
1928   the survey unit may not be released regardless of the result of the statistical test.

Two-Sample Statistical Test (WRS Test)

1929

1930 Measurements from the reference material and material survey unit are compared using the Wilcoxon
1931 Rank Sum (WRS) test (also called the Mann-Whitney test). The WRS test should be conducted for each
1932 material survey unit. If any measurement in the material survey unit exceeds the average of the reference
1933 material by more than $DCGL_C$ additional investigation is recommended, at least locally, regardless of the
1934 outcome of the WRS test.

1935 The WRS test is most effective when contamination is uniformly present throughout a survey unit.
1936 The test is designed to detect whether this activity exceeds the $DCGL_C$. The advantage of the
1937 nonparametric WRS test is that it does not assume that the data are normally or log-normally distributed.
1938 The WRS test also allows for "less than" measurements to be present in the reference material and the
1939 survey units. As a general rule, the WRS test can be used with up to 40 percent "less than"
1940 measurements in either the reference material or the survey unit. However, the use of "less than" values
1941 in data reporting is not recommended. When possible, report the actual result of a measurement together
1942 with its uncertainty.

1943 The following formal null and alternative hypotheses are tested by the WRS test under Scenario A:

1944 Null Hypothesis
1945 H_0: The median concentration in the material survey unit exceeds that in the reference material by
1946 more than the $DCGL_C$

1947 versus

1948 Alternative Hypothesis
1949 H_a: The median concentration in the material survey unit exceeds that in the reference material by
1950 less than the $DCGL_C$

1951 The null hypothesis is assumed to be true unless the statistical test indicates that it should be rejected in
1952 favor of the alternative. One assumes that any difference between the distributions of the reference
1953 material and material survey unit concentrations is attributable to a shift in the survey unit concentrations
1954 to higher values (i.e., because of the presence of contamination in addition to background).

1955 If the distribution of measurements is symmetric, the median and the mean are the same. To the extent
1956 that the mean may be larger than the median, there should be some areas of larger concentration that
1957 cause the distribution to be skew. When that is the case, they will be identified by scanning, and will
1958 trigger appropriate investigation levels as described in Section 6. This is the reason for combining direct
1959 measurements with scans in the survey design.

1960 The assumptions underlying the WRS test are that (1) the samples from the reference material are
1961 independent random samples from the same reference concentration distribution, (2) samples from the
1962 material survey unit are independent random samples from the same material survey unit concentration
1963 distribution, and (3) each measurement is independent of every other measurement, regardless of which
1964 set of samples it came from.

1965   Note that some or all of the material survey unit measurements may be larger than some reference
1966   material measurements, while still meeting the release criterion. Indeed, some survey unit measurements
1967   may exceed some reference material measurements by more than the $DCGL_c$. The result of the
1968   hypothesis test determines whether or not the material survey unit as a whole is deemed to meet the
1969   release criterion. Individual measurements exceeding the $DCGL_c$ are further investigated to the extent
1970   necessary to ensure that the overall average in the survey unit does not exceed the $DCGL_c$. Additionally,
1971   the test should consider whether any smaller areas with elevated levels of contamination may exceed a
1972   separate criterion set for such areas.

1973   The test should have sufficient power ($1-\beta$, as specified in the DQO Process) to detect residual
1974   radioactivity concentrations at the lower bound of the gray region (LBGR). The LBGR should be set at
1975   the expected mean residual contamination level in the material survey unit. The larger of the two values
1976   of σ estimated from the reference material and material survey unit should be used for the WRS test
1977   sample determination. As described in the MARSSIM, the relative shift, Δ/σ, where $\Delta = DCGL_c -$
1978   LBGR, is calculated. Table 5.3 in the MARSSIM provides sample sizes for the WRS test as a function
1979   of relative shift and Type I and II decision errors.

1980   If the criterion specified for controlling the release of material is that there must be no contamination, the
1981   clearance survey requires an approach similar to Scenario B described in. The following formal null and
1982   alternative hypotheses are tested by the WRS test under Scenario B:

1983   <u>Null Hypothesis</u>
1984 H_0: The median concentration in the material survey unit does not exceed that in the reference
1985   material (i.e., there is no contamination).

1986   *versus*

1987   <u>Alternative Hypothesis</u>
1988 H_a: The median concentration in the material survey unit exceeds that in the reference material by
1989   more than the upper bound of the gray region (UBGR).

1990   For this test, the lower bound of the gray region is set at zero contamination. As for the Sign test using
1991   Scenario B, it is again necessary to specify a UBGR. It is essential for determining an appropriate
1992   sample size and the needed measurement sensitivity. The width of the gray region, $\Delta = UBGR - LBGR =$
1993 $UBGR - 0 = UBGR$. If σ is the standard deviation of repeated "background" measurements (i.e.,
1994   measurements on material known to contain no contamination), Δ/σ expresses the width of the gray
1995   region as a relative shift. Table 5.3 in the MARSSIM shows that when this relative shift falls below 1,
1996   the sample size required for the test increases dramatically. For example, if $\Delta/\sigma = 1$, and the DQOs for
1997   the Type I and Type II error rates, $\alpha = \beta = 0.05$, 32 measurements are required on both the survey material
1998   and on the background reference material. If $\Delta/\sigma = 0.5$, 114 measurements are required on each. If Δ/σ
1999   falls as low as 0.1, more than 2,700 measurements are required on each. Thus, it is generally
2000   recommended that the relative shift Δ/σ be between 1 and 3. Increasing the relative shift much above 3
2001   does not appreciably reduce the required number of samples.

2002 There is a direct connection between the UBGR and the required measurement sensitivity.
2003 To distinguish between a measurement of background on the reference material and a measurement equal
2004 to background plus the UBGR, the instrument or procedure must be able to reliably detect the difference
2005 (i.e., the UBGR). Unless the uncertainty of a typical background measurement, σ_M, is less than the
2006 UBGR, the relative shift $\Delta/\sigma = UBGR/\sigma$ will fall below 1, even if there is no spatial variability
2007 contributing to σ. Conversely, setting the UBGR to be less than σ_M will cause the number of
2008 measurements required to achieve the DQOs to rise dramatically. *Thus, an essential part of the DQO*
2009 *Process for this case is in setting the UBGR, recognizing the implicit demand on the required relative*
2010 *measurement uncertainty at near-background levels.*

2011 Application to Surface Activity Measurements

2012 Either the Sign test or WRS test can also be used for surface activity measurements. Given that many
2013 material survey units are composed of the same material types, using the WRS test should be relatively
2014 straightforward (i.e., same as described in the MARSSIM). In some cases however, the number of
2015 materials present in a batch may make it impractical to use the WRS test. In such cases, it is possible to
2016 perform the Sign test on the difference of paired measurements on similar materials, one from the survey
2017 unit and one from a reference material, as outlined in Section 12 of NUREG-1505 (NRC, 1998b).

2018 When surface activity measurements are performed using non-radionuclide-specific (gross) survey
2019 instruments (e.g., GM and gas proportional detectors), a commonly used procedure is to subtract an
2020 "appropriate average background" from each gross measurement on the solid material, and then analyze
2021 the resulting data using a one-sample statistical test, such as the Sign test. Before doing so, however, the
2022 surveyor should recognize that the WRS test may be more advantageous for the following reasons:

2023 (1) The number of samples taken to compute an appropriate background average is left purely to
2024 judgment. When the WRS test is used, the appropriate number of background measurements has a
2025 statistical basis.

2026 (2) The Sign test will generally not be as powerful as the WRS test (more important as the expected
2027 contamination level approaches the $DCGL_C$).

2028 (3) The same data that are used to calculate the average background can always be used in the WRS test
2029 as well.

2030 The Sign test offers no real savings (compared to the WRS test), with the possible exception of the time
2031 needed to perform the calculations. However, when the material survey unit is very clean, the maximum
2032 survey unit measurement and minimum reference area measurement will likely not exceed the DCGL,
2033 and the survey unit will pass the WRS test without any need for calculations. When the material is
2034 contaminated above the DCGL, a simple comparison of the averages will likely show that the material
2035 cannot be released. It is only in cases where the contamination is near the DCGL that the extra
2036 computations involved in the WRS test will be necessary; however, it is precisely in those cases that the
2037 higher statistical power of the WRS test makes its use more desirable.

2038   <u>Statistical Sample Locations</u>

2039   While many sampling and analysis procedures for solid materials clearance surveys are the same as those
2040   recommended in the MARSSIM, the major exception is the selection of sampling points on a survey unit
2041   consisting of a few large, irregularly-shaped pieces. It is virtually impossible in most cases to identify
2042   random locations on material with odd shapes, simply because such materials are virtually impossible to
2043   grid. Materials consisting of many small regularly shaped pieces can be spread out evenly, as discussed
2044   in Section 4.2. A random start grid (rectangular or triangular) can be used to locate samples. It is
2045   important to emphasize that the objective in this case is to give every portion of the batch the same
2046   opportunity to be sampled. Thus, it is only necessary to locate and lay out the grid sufficiently to ensure
2047   that sampling locations are chosen objectively.

2048   One way to approximate this procedure for a survey unit consisting of a few large, irregularly shaped
2049   pieces is to lay out a grid in the area where measurements are to be made. The batch of material should
2050   be laid out in a single layer on top of this grid. A randomly selected grid node is sampled by measuring
2051   whatever piece (or portion) is nearest that node. If no piece is near, select another point until the required
2052   number are obtained. If there is a well-defined inside and outside (as for a pipe), an additional random
2053   number can be used to determine whether the inside or outside is sampled. Even this procedure may not
2054   be workable for large pieces of equipment that cannot be placed on a grid so that every point has an equal
2055   choice of being sampled. In such cases, there may be no alternative other than to choose biased sampling
2056   locations, giving preference to samples that are more likely to contain radioactivity. This involves
2057   professional judgment, and often results in overestimating the average concentration. This is not a
2058   guarantee, of course, because such judgments are not perfect. It is important to document the criteria
2059   used for selecting sampling locations in a standard operating procedure (SOP), and to document that
2060   these criteria were followed. These criteria, and the associated logic, should be specified before the
2061   actual sampling.

2062   Another possible method for sampling a lot of similarly sized small pieces of material is to systematically
2063   measure every m^{th} piece. This requires some estimate of the total number of pieces, N, so that N/m equals
2064   or exceeds the number, n, required for the statistical tests.

2065   **5.3   Automated Scanning Surveys (conveyorized survey monitors)**

2066   Systems that automate the collection of measurements can offer an appealing alternative to manual
2067   surveys. By design, automated systems require little in the way of human intervention during operation
2068   and analyze the data on-the-fly, while storing the information in digital form. These features can provide
2069   several advantages when compared to manual surveys by personnel using hand-held equipment; however,
2070   such automation typically requires equipment that is both expensive and bulky.

2071   Conveyorized survey monitors (CSMs) offer a form of automation that may be particularly well-suited
2072   for use where significant quantities of bulk material are subject to clearance requirements. As the name
2073   implies, these systems operate by moving materials past radiation detectors using a conveyor system,
2074   while automatically storing and analyzing the resulting signals. The radiation detectors themselves can
2075   be of any type and are chosen to match the application. The most common detectors in use are NaI
2076   crystals for gamma-detection and thin-window proportional counters for beta-detection.

2077 Sections 5.3.1 – 5.3.3 discuss CSM systems and their possible application as a measurement method
2078 when releasing solid materials during clearance surveys. Like all measurement methods, CSMs are
2079 viewed as tools that may be used alone or in tandem with other methods. Although specific
2080 manufacturers' systems are not discussed, Appendix B, "Advanced/Specialized Instrumentation,"
2081 includes a sampling of platforms that are presently being marketed for this application, as well as
2082 supporting information about various types of detectors and materials.

5.3.1 Equipment

2084 Conveyorized survey monitors typically include a motorized conveyor, a detector array, supporting
2085 measurement electronics, and an automated data acquisition subsystem. Monitors may also include
2086 segmented pathways along the conveyor so that suspect material may be transported to a destination
2087 other than that of the non-suspect (or releasable) material.

2088 The conveyor portion of a system consists of a belt that is moved by a variable-speed motor from a
2089 loading area, past a detector assembly or set of assemblies, and onto the final destination, which may be
2090 either a disposal container or an intermediate pile. If a mechanical diverter is used, the system controls
2091 the final material destination based upon user-configured measurement parameters. Without automated
2092 segmentation of the material, a system would need to be used in a "shutdown" mode to allow manual
2093 separation of suspect material.

2094 Since the conveyor operates in a continuous loop, it creates the possibility for cross-contamination on the
2095 belt. When processing materials with a low probability of contamination, as is usually the case during
2096 clearance surveys, this issue is of little concern. For applications where cross-contamination poses a real
2097 issue, however, it would seem reasonable to use a continuously replaced rolled sheeting material as a
2098 protective barrier.

2099 *Automated Data Processing (ADP)* — Measurements collected using a CSM are usually digitized before
2100 being analyzed and digitized. The data are analyzed on-the-fly using a preset algorithm, and decisions
2101 concerning suspect materials are usually made in real-time. The resulting data, together with the analysis
2102 results, are then archived to a long-term digital storage medium.

2103 The counting parameters associated with measuring a stream of material passing near a CSM detector are
2104 very similar to those encountered with other detection systems. Although each manufacturer's system
2105 employs a proprietary analysis mechanism, the fundamental physics and statistical parameters are
2106 independent of the software design. As such, one can estimate the detection sensitivity of a CSM
2107 detector system without detailed knowledge of the analysis methods that are actually used, provided that
2108 the type of detector and electronic configuration are known.

2109 A very interesting capability that is unique to automated systems is the ability to perform multiple,
2110 parallel analyses. As a practical example, a CSM could be configured to monitor over multiple time
2111 intervals, in order to optimize the detection capability for both small and large regions at the same time.
2112 Additionally, the data collected from shorter time intervals could be used to augment the decision
2113 criterion applied to longer time intervals, so that small increases over the long interval may be corrected
2114 for anomalies (e.g., such as from potential hot spots) observed during short-interval measurements.

2115 *Detectors* — The heart of any radiation measurement system is the detector(s). The selection and
2116 configuration of detectors and associated electronics is the single most important aspect of designing any
2117 radiation measurement device, since it defines the system's baseline capability. Auxiliary components,
2118 such as data analysis engines and hardware controls, certainly affect the overall performance of a CSM,
2119 but not to the same degree as the detector(s). The ability of any detector to measure radiation is defined
2120 by physical constraints that cannot be easily manipulated or changed by users, so the initial selection of
2121 this component more-or-less establishes the system's capability.

2122 Gross screening of gamma-emitting radionuclides is usually best performed using scintillation detectors,
2123 such as NaI or plastic scintillators. While these detectors are not the best selection for quantitative
2124 measurement of complex spectra, their excellent detection efficiencies and relatively low cost make them
2125 top candidates for gross gamma measurement applications where CSMs may be desired. Solid-state
2126 gamma-ray detectors, such as high-purity germanium (HPGe) detectors, offer much better assay
2127 capability, but are fairly expensive to purchase and maintain, especially if one is interested in achieving
2128 the same level of detection efficiency offered by large-volume scintillation crystals.

2129 The type, shape, encapsulation, and electronic configuration of a scintillation detector determine its
2130 overall detection efficiency and background response, thereby defining its signal-to-noise ratio.
2131 Consequently, it is important to select detectors that balance background response with detection
2132 efficiency for the suspected radionuclide(s). As an example, a 3" x 3" NaI detector yields a good signal-
2133 to-background ratio for a high-energy gamma-emitter such as ^{60}Co, but is a poor selection for a low-
2134 energy emitter such as ^{241}Am. Beyond the base selection of the detector material and physical design,
2135 one should consider the selection and placement of photodetectors and driving electronics when
2136 considering the optimization of a system. For example, simply reducing (or increasing) the detection
2137 input threshold at the amplifier stage can sometimes critically alter the overall system performance.

2138 High-purity germanium detectors could play an important role in some CSM systems, even though they
2139 are more expensive and difficult to maintain. These detectors are excellent for gamma-ray spectrometry,
2140 as they facilitate an unparalleled capability for nondestructive identification and quantification of
2141 gamma-emitting radionuclides. With the exception of very expensive large-volume crystals, however,
2142 these detectors cannot compete with low-cost scintillation materials when gross sensitivity is desired.
2143 Their use in a CSM system could be warranted in some instances for nuclide identification following a
2144 positive detection during a gross scan. For example, a system could plausibly be configured to
2145 automatically stop a conveyor following a positive *detect*, and then attempt to identify the gamma-
2146 emitting radionuclides present before passing the material to its final destination.

2147 Measurement of beta-emitting radionuclides in (or on) bulk materials may also be possible, depending on
2148 the radionuclide, material type, and release limit. Beta detection can be accomplished using thin-window
2149 gas-filled detectors, such gas proportional and Geiger-Mueller detectors, and thin-windowed scintillators.
2150 The most likely candidate for measuring beta-emitters is large-area gas flow through proportional
2151 detectors with thin Mylar entrance windows; however, large-area sealed proportional and GM detectors
2152 are also expected to perform well. Scintillation materials universally suffer from an inferior signal-to-
2153 background ratio when measuring beta-emitters, but may still be adequate for some applications.

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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.

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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.

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5.3.2 Detection Sensitivity

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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).

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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.

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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).

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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.

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2194 To deal with this fact while using a CSM during clearance surveys, one can assume, for better or worse,
2195 that homogeneity exists within sub-regions of the suspect material and, to be consistent with traditional
2196 survey design, these regions should be labeled as survey units or batches. The desired belt speed should,
2197 therefore, be determined as a function of the release limit (DCGL), the allocated survey unit size, and the
2198 detection efficiency of the system for the target media and expected radionuclide(s).

2199 *Detection Efficiency for Gamma-Emitters using NaI Detectors* — The detection ability of NaI detectors
2200 operating in a gross count rate mode[3] will be dependent on the design, quantity and electronic
2201 configuration of selected detectors. For purposes of providing an example of an expected detection
2202 capability, this section discusses a hypothetical system that has been configured with moderately sized 3"
2203 x 3" cylindrical crystals with supporting electronics. It is assumed that three such detectors will be
2204 operated in tandem in a detector *bank* and that the total detector volume per bank will therefore be about
2205 1000 cm^3.

2206 A common radionuclide that may be measured using such a system would be ^{137}Cs—with a primary
2207 gamma-ray emitted by its daughter (^{137m}Ba) at ~662 keV with an emission ratio of ~0.85. If one assumes
2208 that cesium is mixed relatively homogeneously within each region of a CSM conveyor stream, then a
2209 fairly accurate estimate of detection ability can be calculated by coupling empirical data with modeled
2210 exposure rates. The two empirical parameters that should be known are the total background count rate
2211 and the detection efficiency for ^{137}Cs. In general, although certainly depending on location and
2212 configuration, the background count rate for 3" x 3" cylindrical NaI crystals operating in full-open gross
2213 count rate mode will be in the range of about 8 x 10^3 to 1 x 10^4 counts per minute (cpm) and the detection
2214 efficiency will be approximately 4 x 10^6 cpm per mR/h when measuring ^{137}Cs. For three detectors
2215 ganged into a single electronic bank, these values correlate to a total system background of about 2.7 x
2216 10^4 cpm and a total detection efficiency of about 1.2 x 10^7 cpm per mR/h.

2217 These parameters can be coupled to calculated exposure rates in the vicinity of material passing along a
2218 conveyor system to evaluate detection sensitivity as a function of the material geometry and radionuclide.
2219 As an example application, consider a scenario where a CSM will be used to scan for ^{137}Cs in soil having
2220 a bulk density of 2 g/cm^3. The center-line of the three detectors is assumed to be placed approximately
2221 15 cm above a 76-cm (30-in) wide conveyor belt such that they are evenly spaced across the breadth of
2222 the belt at 13, 38 and 64 centimeters (5, 15 and 25 inches) from one edge. If the soil is assumed to be
2223 2.5-cm (1-inch) thick and to extend on the conveyor for 76-cm (30-inches) along the conveyor to either
2224 side of the detector bank then the expected exposure rate will be about 120 mR/h per μCi/g at the two
2225 outside detectors and approximately 140 mR/h per μCi/g for the center detector. Coupling these data
2226 with the expected detection efficiency previously given, the total efficiency for this geometry—using all
2227 three detectors in an electronically ganged configuration—is expected to be about 1.5 x 10^3 cpm per
2228 pCi/g of ^{137}Cs. If the soil thickness is increased to 10-cm (4-in) and the detectors are positioned 20-cm
2229 (8-in) from the belt, then the system detection efficiency will increase to about 4 x 10^3 cpm per pCi/g of
2230 ^{137}Cs. The latter case represents a count rate increase of 15% above background for each pCi/g of ^{137}Cs.

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[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.

2231	An estimate of the minimum detectable concentration (MDC) can be estimated while operating such a
2232	detector configuration in a scan mode by assuming a false positive detection rate of 1% and a false
2233	negative detection rate of 5% (Currie 1968). These values mean that true contamination will be missed
2234	5 percent of the time, and false alarms will occur 1 percent of the time. For an observation interval of
2235	6 seconds, the MDC for a 2.5-cm (one-inch) thick layer of soil containing ^{137}Cs is expected to be about
2236	2 pCi/g and will decrease to 0.7 pCi/g when the soil thickness is increased to 10 cm.

2237	*Detection Efficiency for Beta-Emitters Using Thin-Window Proportional Detectors* — Beta particles
2238	originating within or on a target media usually undergo significant interaction before reaching the
2239	sensitive volume of a CSM detector. As such, the process for estimating detection ability is significantly
2240	more problematic than is necessary when evaluating detection capability for gamma-emitting
2241	radionuclides. As previously mentioned, the most common type of detector for this application is a thin-
2242	window gas-flow proportional detector. Such detectors have a thin Mylar entrance window with a
2243	density thickness ranging from less than 1 to a few mg/cm^2. Although the mixture may vary, the most
2244	commonly used gas is P-10, containing 90 percent argon and 10 percent methane.

2245	This section provides an analysis of the beta detection ability for gas-flow proportional counters and, in
2246	particular, that which is applicable to a CSM. The first scenario considers surface contamination with
2247	^{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
2248	in soil. These evaluations are summarized in the following paragraphs.

2249	Surface activity refers to contamination on the surface of solid materials. As simple as this sounds, it is
2250	difficult to define what constitutes a "surface," since real-world materials have a thickness when viewed
2251	from the perspective of a radioactive atom deposited *within* their surfaces. One might define surface
2252	contamination as the activity contained within a surface layer that has a thickness equal to that of the
2253	saturation layer (ISO 1988), where the thickness of the saturation layer is defined as the thickness of the
2254	medium (surface material) equal to the maximum range of the specified particulate radiation. While
2255	some materials are more porous than others, all have some level of absorptive capacity. The definition of
2256	"surface," therefore, becomes significant when evaluating the detection ability for charged particles
2257	emitted from the surface of materials, and is amplified significantly when constructing a model.

2258	Consider an 80-cm (31-inch) wide conveyor using five proportional counters with open, or sensitive,
2259	areas of 500-cm^2 each, placed 5 cm above the belt surface. The detectors are rectangular in shape, with
2260	each window region measuring 50 cm x 10 cm (20 in by 4 in), with the long dimension placed parallel to
2261	the direction of belt travel in the CSM. If five such detectors are placed side-by-side across the breadth
2262	of the conveyor, the total sensitive area is 2,500 cm^2 (390 in^2). Each detector is assumed to be configured
2263	individually (not grouped), with 0.8 mg/cm^2 of window material without protective screens, and the
2264	detection capability is assumed to have been maximized for low- to intermediate-energy beta detection.
2265	The background response for such a detector is in the range of 2 to 3 cpm/cm^2 of window area, so each
2266	detector has a non-shielded typical background of about 1,300 cpm. Again, the reader should note that
2267	this configuration is defined for the purpose of estimating beta detection ability as an example; however,
2268	the detection abilities of actual systems will vary by manufacturer (although not very much).

2269 First, the pure beta-emitting radionuclides ^{99}Tc and ^{90}Sr(^{90}Y), having maximum-energy beta emissions of
2270 294 and 546(2280) keV, respectively, are assumed to be placed onto the surface of a thin, flat plane in
2271 contact with a CSM conveyor belt. Although unrealistic for most real-world measurement scenarios, this
2272 finite plane, zero-thickness geometry provides the highest possible beta-detection sensitivity for a system
2273 without improving the detector to belt distance. As an extension to this *pure* geometry, it is then assumed
2274 that the radionuclides are not restricted to the outermost surface, but instead that they have absorbed
2275 homogeneously within the top 50 μm of a masonry-type material (e.g., cement) having a bulk density of 2
2276 g/cm^3. This scenario is much more plausible when evaluating real-world applications. Table 5.1
2277 presents the results of these geometry calculations.

2278 The second geometry places the same isotopes (i.e., ^{99}Tc and ^{90}Sr(^{90}Y)) into a soil matrix and varies
2279 the depth of the material from 0.1 to 1 cm, while keeping the belt to detector distance constant.
2280 The results of this analysis display, both qualitatively and quantitatively, the impact on detection
2281 capability that occurs when beta particles interact within the source-matrix material. Table 5.1 presents
2282 the results.

2283 Finally, the isotope ^{137}Cs, which is both a beta- and a gamma-emitter, is modeled within a soil matrix.
2284 Cesium-137 decays with the emission of a 512-keV$_{max}$ beta 94.6 percent of the time, and decays with the
2285 emission of a 1,173-keV$_{max}$ beta for the remainder. As previously mentioned, ^{137m}Ba is produced by 94.6
2286 percent of ^{137}Cs decays, and it, in turn, emits a 662-keV photon during 90 percent of its decays, yielding
2287 an overall γ-emission ratio of 0.85. Although not previously discussed within this section, gas-flow
2288 proportional counters also detect ionizing electromagnetic radiations (e.g., gamma and x-rays) by
2289 measuring secondary electrons produced both within and outside the gas volume. The probability of
2290 interaction varies; however, the sensitivity is roughly proportional to the mass of intervening material
2291 within the vicinity of the detector, times the probability of interaction within the mass, times the fraction
2292 of those particles carrying enough energy to travel into the detector. For ^{137}Cs, the intrinsic efficiency
2293 expected with a thin-window proportional detector is about 0.01 counts per photon. The photon
2294 detection capability for this scenario was estimated for each CSM detector by calculating the average
2295 solid-angle for the geometry and coupling the result with the activity, source-material absorption
2296 probability and finally the detector interaction probability. Table 5.1 presents the result for the summed
2297 beta and gamma detection capability.

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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-cm² 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).

2366 When small elevated areas of radioactivity are potentially present (e.g., Class 1 materials), their impact
2367 on the average contamination level should be properly addressed during the calibration and efficiency
2368 determination for *in toto* survey techniques. Alternatively, when potential small elevated areas of
2369 radioactivity are a concern, it may be appropriate to consider combining the in toto techniques with
2370 conventional scanning for locations of elevated direct radiation.

2371 When employing *in toto* clearance survey techniques, it is important to consider both the classification of
2372 solid materials and the difference between the material survey unit size and sample size. Consider a
2373 pallet of 1.5-m long steel pipes that is assayed using a calibrated *in situ* gamma spectrometer system.
2374 This pallet represents a material survey unit, which would likely be surveyed via *in situ* gamma
2375 spectrometry in the same manner regardless of its classification.

2376 Consider a large container filled with hundreds of small pieces of equipment and tools that are proposed
2377 for clearance. Assume that a tool monitor will be used to demonstrate compliance with the release
2378 criteria. In this instance, the amount of material (perhaps no more than 10 items at a time) that can be
2379 analyzed by the *in toto* technique represents the sample size, rather than the survey unit size. When
2380 *in toto* survey techniques are used to measure samples, the statistical design methods discussed in
2381 Section 5.2.3.3 should be used to determine the sample size.

2382 The DQO Process should be used to establish the appropriate survey coverage. The material's
2383 classification should be considered when setting the size of the material survey unit. For example, the
2384 amount of material comprising Class 1 survey units may be smaller than either Class 2 or 3 survey units.
2385 Alternatively, it may be reasonable to maintain consistent survey unit sizes for all material classes, while
2386 adjusting the survey coverage based on classification. In this situation, the tool monitor might be used to
2387 assay 100 percent of the materials in Class 1, while smaller fractions of the total material would be
2388 analyzed in Class 2 and 3 survey units. For example, it may not be necessary to survey each and every
2389 brick that comprises a lot of Class 2 bricks. Regardless of the selected approach, the solid materials
2390 having the greatest potential for contamination should receive the highest degree of survey coverage.

2391 Sections 5.4.1 – 5.4.3 discuss *in situ* gamma spectrometry, volume counters (e.g., drum counters, tool and
2392 bag monitors), and portal monitors. Calibration and implementation considerations for using these
2393 systems are also discussed.

5.4.1 *In Situ* Gamma Spectrometry

2395 *In situ* gamma spectrometry (ISGS) measurements for solid materials, particularly in a complex geometry
2396 that renders some of the surfaces inaccessible, may be a viable release survey option. This section
2397 discusses some of the considerations and the overall plan for implementing ISGS as a tool for surveying
2398 solid materials, including experimental results for applying ISGS to surveys of scrap metal. Appendix C
2399 provides a few examples of commercial applications of ISGS.

5.4.1.1 Equipment

2401 An ISGS system typically consists of a semiconductor detector, electronics for pulse amplification and
2402 pulse height analysis, a computer system for data collection and analysis, and a portable cryostat.
2403 The most common detector is the high-purity germanium (HPGe) semiconductor, but other
2404 semiconductors such as developing room temperature variants can be deployed. The HPGe crystal
2405 should be cooled to liquid-nitrogen (LN) temperature for operation, but can be stored at room
2406 temperature without destroying its detection properties.

2407 This is an important distinction between HPGe semiconductor detectors and germanium-lithium (GeLi)
2408 semiconductor detectors, which must be cooled to LN temperature at all times. Scintillating detectors,
2409 such as sodium iodide (NaI), have limited application (e.g., when energy resolution is not a primary
2410 concern). Additionally, depending on the application, lead shielding and collimation may be required.

2411 **5.4.1.2 Technological Advances**

2412 Many technological advances have allowed ISGS to become more of a mainstream survey methodology.
2413 As previously mentioned, one of the most important advancements was the HPGe detector, which only
2414 required cooling to LN temperature during operation. Also, these detectors have increased in volume,
2415 resulting in much higher efficiency, while maintaining excellent energy resolution. These systems can
2416 only be used if the detector is maintained at LN temperature, but the advancements of rugged, multi-
2417 attitude LN cryostats have permitted ISGS systems to be deployed in almost any environment. The
2418 electronics have also been improved by reducing their size, which increased their portability. Typically,
2419 these electronics have been analog in design, which means that they suffered from instability under
2420 certain conditions. Digital electronics packages have overcome the limitations of the analog designs.
2421 The portable computing systems used to collect and analyze the ISGS data have also increased in power
2422 while also decreasing in size.

2424 **5.4.1.3 Sensitivity**

2425 Unlike hand-held detectors used to scan and/or perform direct measurements to qualify or quantify
2426 primarily alpha and beta surface activity, ISGS can be used to quantify volumetric contamination of
2427 gamma-ray-emitting radionuclides. Many factors determine the overall efficiency and sensitivity of an
2428 ISGS system for quantifying volumetric contamination, as follows:

2429 Intrinsic detector efficiency
2430 The intrinsic efficiency of a detector is the measure of how efficient the detector medium absorbs
2431 gamma-ray energy, as a function of energy. At very low energies, gamma-rays are absorbed outside
2432 the detector, in the casing or faceplate. As the energy increases, the intrinsic efficiency increases until
2433 a maximum intrinsic efficiency is reached, typically at an energy of a few hundred keV. After the
2434 maximum is reached, the intrinsic efficiency decreases with increasing energy.

2435 Radionuclide gamma-ray energy and abundance
2436 As discussed above, the intrinsic efficiency of a detector depends on the gamma-ray energy. Also,
2437 attenuation from the material being surveyed increases as the gamma-ray energy decreases. Solid
2438 materials with potential contamination involving radionuclides of low gamma-ray decay abundance,
2439 or yield, require longer count times than radionuclides with high gamma-ray decay abundance.

2440 Background, including shielding and collimation
2441 High background, for the gamma-ray energies of concern, decreases the sensitivity of the ISGS. This
2442 effect is more pronounced at lower energies because of the Compton continuum contributions from
2443 ambient gamma-rays, which are higher in energy than the energy of concern. To reduce the effect of
2444 background, lead shielding and collimation can be used. While generally increasing the sensitivity of
2445 the ISGS measurement, collimation can actually lower the overall efficiency of the ISGS system by
2446 effectively shielding the contamination from the detector. This is a concern when using small-
2447 opening collimators.

	Count time
2448	Count time
2449	Many factors influence the amount of time required to count the material. These include the overall
2450	efficiency, source and background count rates, and desired uncertainty. In general, as the background
2451	increases, the sensitivity decreases. To compensate, increasing count time increases sensitivity. In
2452	order to reduce the uncertainty of the measurement by half, the count time would need to be increased
2453	by a factor of four.
2454	
2455	Geometry
2456	Geometry refers to the orientation of the source material and the detector relative to the source
2457	material. For example, the overall efficiency and, therefore, the sensitivity of the ISGS measurement
2458	would be different if a lot of 25 pipes is stacked in a pyramid, rather than placed flat and unstacked.
2459	The overall efficiency of the ISGS measurement is also affected by the distance the detector is placed
2460	from the source material.

2448 <u>Count time</u>

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2450 efficiency, source and background count rates, and desired uncertainty. In general, as the background

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2455 <u>Geometry</u>

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2457 material. For example, the overall efficiency and, therefore, the sensitivity of the ISGS measurement
2458 would be different if a lot of 25 pipes is stacked in a pyramid, rather than placed flat and unstacked.
2459 The overall efficiency of the ISGS measurement is also affected by the distance the detector is placed
2460 from the source material.

2461 **5.4.1.4 Experimentation to Determine Sensitivity**

2462 Oak Ridge Institute for Science and Education (ORISE) performed an experiment to determine the
2463 magnitude of the ISGS detection capabilities for a release of scrap metal from a nuclear facility. In this
2464 case, 1 metric ton of 12.7-cm (5-in.) diameter steel conduit was selected. To determine how much
2465 radioactivity was required for the experiment, the mass-based, critical-group dose factors reported in
2466 draft NUREG-1640 were used. For comparison with draft NUREG-1640, a normalized unit dose factor
2467 of 10 µSv/y (1 mrem/y) was assumed in the calculations. As the following equation shows, 38 kBq
2468 (1 µCi) of ^{137}Cs on steel would produce approximately 10 µSv/y (1 mrem/y) to the critical member of the
2469 group.

2470
$$\frac{10\,\mu Sv\,y^{-1}}{260\,\mu Sv\,y^{-1}\,Bq^{-1}\,g}\cdot 1E6\,g\cdot\frac{kBq}{1000\,Bq}=38\,kBq$$

2471 Therefore, if the ISGS system can demonstrate a sensitivity less than 38 kBq (1 µCi), this is a candidate
2472 technique. Table 5.2 summarizes the total activity calculations for steel.

2473 **Table 5.2: Calculated total activity for selected radionuclides**
2474 **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

2478 ^{a} To convert to units of mrem y^{-1} pCi^{-1} g, multiply by 3.7x10^{-3}.
2479 ^{b} To convert to units of µCi, divide by 37.

2480 Twenty sources each for ^{137}Cs and ^{60}Co were fabricated; each source was approximately one-twentieth of
2481 38 kBq (1 µCi). The ^{137}Cs sources were randomly placed inside the conduit interiors. A measurement
2482 was performed at the midpoint of each side of the pallet for 10 minutes, for a total of 40 minutes of count
2483 time. The process was repeated for nine additional measurement sets with the ^{137}Cs sources placed
2484 randomly each time. The ^{60}Co measurements were independently performed in the same manner. No
2485 shielding or collimation was used, and the detector was placed 1 meter (vertically) from the floor, and
2486 generally as close as possible to the pallet of steel conduit.

2487   The efficiency, ε, for the region-of-interest (ROI) corresponding to the appropriate total absorption peak
2488   (TAP) for ^{60}Co or ^{137}Cs was calculated. First, the net counts in the TAP ROI were calculated by
2489   subtracting the Compton continuum counts in the ROI from the gross counts in the TAP ROI. Next, the
2490   net counts for the TAP ROI were divided by the total activity of the particular source, and the count time
2491   in minutes to determine efficiency in net counts per minute per kBq. The minimum detectable activity
2492   (MDA), in kBq, for the TAP ROI was calculated by the equation below, using the experimentally
2493   determined efficiency, where the BKG values, or continuum counts, were determined by the gross peak
2494   counts minus the net peak counts.

2495
$$MDA\,[kBq] = \frac{3 + 4.65\sqrt{BKG\,[counts]}}{T\,[min]\;\varepsilon\;[net\;peak\;counts\;per\;min\;per\;kBq]}$$

2496   Table 5.3 below summarizes the results of the ISGS measurements of the steel conduit pallet.

2497   **Table 5.3: Efficiency and MDA summary for ISGS measurements of scrap steel pallet**
2498   **(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

2504   [a] Total propagated uncertainty.
2505   [b] To convert to units of net counts min⁻¹ µCi⁻¹, multiply by 37.
2506   [c] To convert to units of µCi, divide by 37.

2507   Multiple sets of measurements with randomly placed sources (in a non-uniform geometry) were
2508   performed to calculate an unbiased range of efficiencies for this particular geometry. Using the lower
2509   5-percent confidence interval on the 2-sigma range of the efficiency from Table 5.3 allows the MDA to
2510   be conservatively reported for comparison to potential dose limits.

2511   Table 5.3 shows that at an alternative dose criterion of 10 µSv/y (1 mrem/y), ISGS is a viable technology
2512   for 1 metric ton of 5-inch diameter steel conduit released from a nuclear facility. The upper range MDA
2513   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
2514   (1 mrem/y). The upper range MDA for ^{60}Co at 22 kBq (0.6 µCi) is below the total activity of 40 kBq
2515   (1.1 µCi) required to produce 10 µSv/y (1 mrem/y). However, if the more-restrictive dose limit of
2516   1 µSv/y (0.1 mrem/y) is assumed, ISGS would lack the necessary sensitivity to detect 3.8 kBq (0.1 µCi)
2517   of either ^{60}Co or ^{137}Cs.

2518   With the same ^{137}Cs and ^{60}Co sources used with the steel conduit experiment, a second experimental
2519   configuration consisting of a pallet of 148 insulated copper wires with a total weight of 490 kg
2520   (1,080 pounds) was set up. The only difference between the steel and copper experiment was that the
2521   count time was increased from 10 to 30 minutes per measurement to allow for the increased attenuation
2522   of the gamma-rays by the copper. Table 5.4 shows the dose calculation results.

| | | | 2522 |
| | | | 2523 |

Table 5.4: Calculated total activity for selected radionuclides using mass-based, critical-group dose factors for copper (4.9x10⁵ 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	62	78
^{60}Co	1173, 1332	250	19

[a] To convert to units of mrem y^{-1} pCi^{-1} g, multiply by 3.7x10⁷.
[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.

2558 One approach is to perform multiple measurements at different angles around the material, such as all
2559 four sides, and then average the measurement results. Another approach, which is commonly used in
2560 drum counters, is to rotate the material during the measurement time. However, rotating a pallet of pipes
2561 or wire can be unwieldy, if not impossible, so to effectively rotate the material, one might perform part of
2562 one measurement at each location around the material. For example, suppose a count time of 40 minutes
2563 was required to meet the required sensitivity and the material was to be measured from all four sides.
2564 The first 10 minutes of the single measurement would be performed, and then the acquisition would be
2565 paused while the detector was moved to the second measurement location, and then the acquisition would
2566 continue for another 10 minutes. This process would be repeated for the remaining two positions.
2567

2568 ## 5.4.2 Volume Counters

2569 Various designs of volume counters can be used to quantify surface activity or total activity. Volume
2570 counters, while generally designed for specific counting applications, have common characteristics.
2571 These include a counting chamber, array of detectors, and electronic package for analysis.

2572 The counting chambers are designed specifically for the measurement application. The size determines
2573 what type of materials or containers the system is capable of measuring. Volumes range from small
2574 items to large shipping containers. A variety of detectors, including gas proportional, plastic and NaI
2575 scintillators, HPGe semiconductors, and long-range alpha detection configurations, are used in volume
2576 counters, depending on the application. Many designs focus on detecting specific waste streams (e.g.,
2577 transuranic waste, with a high throughput). Systems designed to quantify alpha and/or beta surface
2578 activity use gas proportional and plastic scintillator detectors or long-range alpha detection. Plastic and
2579 NaI scintillators and HPGe semiconductor detectors are used for volumetric gamma radioactivity.

2580 Calibrations are usually performed with standard packages or suitable geometries containing sources of
2581 known activity. Shielded configurations are frequently used to reduce the background, thereby
2582 increasing the signal-to-noise ratio. In many systems, the shielded configuration completely surrounds
2583 the material to be measured (i.e., 4π counting geometry). An example of this configuration is the drum
2584 counter, in which a conveyor belt typically moves the drum into the counting chamber, where the drum is
2585 usually rotated during the measurement to obtain a more representative average. After the count, the
2586 drum is then moved out and another drum counted.

2587 Considerations for applying volume counters do not vary significantly from the individual application of
2588 each of the mentioned detectors. For example, gas proportional detectors need to be calibrated to a
2589 calibration source representative of the radioactivity, and the considerations listed for ISGS apply for
2590 systems using HPGe detectors for volume counting.

2591 ## 5.4.3   Portal Monitors

2592 A common example of a portal monitor is a truck or rail car scrap metal radiation detection system.
2593 These use large-area plastic scintillation detectors to detect buried radioactive sources in scrap metal.
2594 The radioactive sources are identified by detecting small changes in the ambient gamma background.
2595 Entities in the United States have used portal monitors upon receipt of materials in incoming shipments.
2596 Advances in portal monitor technology may one day allow surveyors to use this technique as a primary
2597 material survey technique.

2597 **5.5 Laboratory Analytical Methods**

2598 Sections 5.5.1 – 5.5.3 discuss the laboratory analyses for hard-to-detect nuclides and various media
2599 matrices (i.e., bulk materials). This discussion ties in with the conventional survey approach, in the sense
2600 that some statistical samples (such as ^{3}H in concrete) are much more complex to analyze than others
2601 (such as simple direct measurement of surface activity).

2602 **5.5.1 Representative Sampling and Laboratory Analysis**

2603 Laboratory analysis provides the greatest level of accuracy and precision, with the lowest detection
2604 levels. Indeed, some techniques have remarkable detection limits. For example; an inductively coupled
2605 plasma mass spectrometer (ICP-MS) can have detection limits less than 1 part per quintillion (ppq).
2606 Furthermore, laboratory analyses usually do not suffer from the calibration issues that plague ISGS and
2607 *in toto* systems (namely, the expense associated with producing or obtaining reference materials needed
2608 to develop or validate a calibration).

2609 Laboratory methods for measuring radioactivity cover a broad range of techniques. It is difficult to
2610 reduce all of the standard techniques to a single recipe. However, once the samples are collected, they
2611 are usually subject to a destructive process (gamma spectrometry is a notable exception), which changes
2612 the physical or chemical state of the sample. Next, the samples are usually purified or chemically
2613 separated into a solution to which a tracer is usually added. The sample is then put in a form that will
2614 allow it to be counted efficiently. This preparation can be time-consuming and costly. Table 5.6
2615 provides cost information on routine radiochemical analysis. Ultimately, the decision to follow an
2616 approach that uses laboratory techniques will balance data quality objects against available resources.

2617 **Table 5.6: Cost information on routine radiochemical analysis**

Energy Spectrometry			
Radiation	Technique/Instrumentation	Estimated Cost per Measurement†	Relative degree of precision
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	

2626 † From Appendix H of the MARSSIM
2627 ◊Recent data from commercial laboratories suggest that this value should be closer to the value for alpha spectroscopy

70

| 2628 | ## 5.5.2 Sample Collection |

2629 The assay process actually begins with the collection of samples. The critical issue regarding the use of
2630 laboratory methods is that the object that is sampled must be disturbed; that is, some amount of material
2631 must be removed from the object. The amount can range from a fraction of a gram in the case of a swipe
2632 or wipe sample for removable alpha contamination, to several kilograms in the case of soil sampling.
2633 While extracting samples from surface soil, for example, is relatively simple and involves the use of
2634 trowels and augers, the collection of samples from steel and concrete can be very difficult. Sampling
2635 these materials requires chisels, hammers, drills, and other more specialized equipment. The collection
2636 of samples, specifically the number and location of the samples, is fundamental to characterizing and
2637 quantifying the contamination. Morever, the number and location of the samples should follow the DQO
2638 Process (see Section 3).

2639 ## 5.5.3 Sample Preparation

2640 Most samples that are collected cannot be assayed directly, but should be converted to a suitable form for
2641 assay. The type and energy of the radiation to be measured determine the ultimate form. For example,
2642 samples containing α or low-energy β activity have problems with self-absorption and, therefore, the
2643 form of the sample should be as thin as possible. More importantly, chemical purification may be
2644 required if interferences are anticipated. Table 5.7 provides a general indication of the sample
2645 preparation for α and β assay for low to medium activities in solid samples. The preparation of samples
2646 for gamma-ray analysis is usually less involved. For example, the preparation of soil involves nothing
2647 more than drying and homogenization. For a more complete listing of standard laboratory methods and
2648 instruments, see the MARSSIM; for specific radiochemical techniques, consult the Environmental
2649 Measurements Laboratory (EML) Procedures Manual (U.S. DOE, 1990) and Radiochemical Analytical
2650 Procedures for Analysis of Environmental Samples (EPA, 1979).

2651	**Table 5.7: Sample preparation for α and β assay for low to medium radioactivity levels**	

2652	**Sample preparation for α assay (solid sample)**		
2653	Detector	Sample preparation	Preparation time
2654 2655	Solid-State Semiconductor	If the sample is thin, count directly. If not, dissolve and redeposit as a thin source	a week or more
2656 2657	Liquid Scintillator	Dissolve in suitable solvent and heat as liquid, or count directly as a suspension in a gel	several days to a week
2658	**Sample preparation for β assay (solid sample)**		
2659 2660	Proportional Counter	May be counted directly unless low energy β (< 50 keV) requires pretreatment	day
2661 2662	Solid-State Semiconductor	Same as proportional counter	day
2663 2664	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

2666 Sections 5.6.1 – 5.6.3 address quality assurance (QA) issues involving the measurement systems
2667 associated with clearance surveys, including the calibration process, data quality indicators, and quality
2668 control (QC). In general, any assay or measurement strategy must develop and follow a quality assurance
2669 process, which should be part of an overall quality assurance program. For guidance in establishing
2670 quality assurance programs, see ASME NQA-1-1994, EPA Guidance Document QA/G-5, and Regulatory
2671 Guide 4.15 (NRC, 1979). At a minimum, the quality assurance program should address the quality
2672 following elements:

2673 • organizational structure and responsibilities
2674 • procedures and instruction
2675 • records
2676 • personnel qualifications
2677 • quality control of measurement systems

5.6.1 The Calibration Process

2679 An important consideration associated with the calibration of instrumentation for use in clearance
2680 surveys (see Appendix B) is the lack of appropriate reference materials and guidance on methods to
2681 calibrate these systems. Therefore, a calibration process should be developed and documented in a
2682 standard operating procedure (SOP). For general requirements that apply to calibrations see
2683 ANSI/ASQC M1-1987 and ANSI/ISO/IEC 17025:2000.

2684 The following items should be part of the calibration process and included in a QA document:

2685 • Describe the type of instrument to be calibrated.

2686 • Describe the calibration method in sufficient detail so that others can duplicate the method.

2687 • Justify and document the calibration methods.

2688 • Describe how calibration data will be analyzed.

2689 • List the parameters, quantities, and ranges to be determined.

2690 • Describe any corrective action, including recalibration, that will be taken if calibration data fail to
2691   meet the acceptance criteria.

2692 • Describe the calibration standards. If the standards are not traceable (to NIST or some other national
2693   certifying organization), describe how the standards will be prepared. Any method used to verify the
2694   certified value of the standard independently should also be described.

2695 • Describe the frequency of the calibration and whether the frequency is related to any temporal
2696   variation of the system.

2697 **5.6.2 Data Quality Indicators**

2698 Data quality indicators (DQIs) are qualitative and quantitative descriptors used in interpreting the degree
2699 of acceptability or utility of data. The principal DQIs are *precision, bias, representativeness,*
2700 *comparability, and completeness.* These are referred to as the "PARCC" parameters, where the "A"
2701 refers to accuracy rather than bias, but the two are generally regarded as synonymous. Of the five DQIs,
2702 precision and bias are crucial when evaluating the performance of an instrument or measurement method.
2703 Establishing acceptance criteria for precision and bias sets quantitative goals for the quality of the data
2704 generated by measurement instrument. DQIs are established during the planning phase of the DQO
2705 Process. More information on DQIs may be found in the MARSSIM.

2706 Comparability is also important, in that it can establish the validity of a measurement technique,
2707 calibration method, or instrument. For example, calibrations of CSM, ISGS, and *in toto* systems may
2708 need to establish comparability with representative sampling and laboratory techniques. There are
2709 several examples of this approach involving ISGS (DOE 1999a, DOE 1999b, Kalb *et al.* 2000). Two of
2710 the studies (DOE 1999a and Kalb *et al.* 2000) utilize the DQO Process. The intent of these studies was
2711 not to show that ISGS produces data that is indistinguishable from the baseline approach (sampling and
2712 laboratory analysis) on a sample-to-sample basis, but that the decision drawn from the data is the same.

2713 An effective tool for evaluating sources of bias, providing a mechanism for standardization and
2714 establishing traceability are intercomparison or intercalibration exercises. Such exercises have long been
2715 a key element in quality assurance programs for field measurement techniques.

2716   **5.6.3   Quality Control**

2717   Quality control (QC) is an important element of the quality assurance process. The purpose of QC is to
2718   ensure that the measurements and other data-producing systems operate within defined performance
2719   limits as specified in planning (EPA 1998a). QC activities help to identify sources of error and
2720   uncertainty, as well as the impact these quantities will have on the decisionmaking process. QC activities
2721   involve the use of QC samples to detect when attributes of the measurement process are exceeding their
2722   performance limits so that corrective actions can be initiated. The measurement attributes that QC
2723   samples monitor include contamination, calibration drift, bias, and precision. The following is a brief
2724   description of standard QC samples.

2725   <u>Blanks</u> are samples that contain little or no radioactivity, and none of the radionuclide of interest.

2726   <u>Performance Evaluation (PE) Matrices</u> are samples with enhanced levels of radioactivity (compared to
2727   a surrogate material) at a known concentration of the radionuclide(s) of interest.

2728   <u>Calibration Checks</u> are samples containing a source or radioactive material, which is independent of a
2729   calibration standard, and can ensure that the calibration remains in a state of statistical control.

2730   <u>Replicates</u> are samples that are measured repeatedly to check the precision of the system.

2731   The quality assurance document should describe the QC procedure, which should identify the QC checks
2732   that are to be performed, the frequency with which they will be performed, their acceptance criteria, and
2733   a correction action plan to be followed if the acceptance criteria are not met. Table 5.8 provides
2734   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	

2743 **5.7   Clearance Survey Examples**

2744   The clearance survey examples presented on the following pages illustrate possible clearance survey
2745   approaches for pipe sections being released from a power reactor facility. The flow diagram for
2746   clearance of solid materials (Section 2) served as a guide for developing these examples; the letters in the
2747   examples correspond to the steps in Figure 2.1.

2748   <u>Example 1   Clearance of small-bore pipes from nuclear power reactor</u>

2749   a. Evaluate the physical description of the solid material.

2750   The solid material being considered for release is small-bore pipe (steel). The material survey unit
2751   consists of approximately 60 sections of pipe and conduit, each of which is 1.2 to 1.8 m in length.
2752   The diameter of each pipe section is less than 6 cm, with a total interior surface area of 17 m^2 and a
2753   weight of 2 tons. The pipe interiors are considered to be inaccessible with conventional hand-held
2754   detectors.

2755   b. Evaluate and document process knowledge and characterization of the solid material.

2756   The small-bore pipes are from a nuclear power plant. Process knowledge indicates that the pipes were
2757   used to transport radioactive liquids from the nuclear laundry. The radionuclide mixture for the nuclear
2758   power reactor consists of a number of radionuclides, including fission products, activation products, and
2759   even trace quantities of transuranics.

2760   During characterization, three samples of pipe residue were collected and analyzed from the total pipe
2761   population. The radionuclide mixture was as follows:

2762 | ^{60}Co | 15% |
2763 | ^{137}Cs | 27% |
2764 | ^{90}Sr(^{90}Y) | 8% |
2765 | ^{14}C | 13% |
2766 | ^{55}Fe | 11% |
2767 | ^{63}Ni | 6% |
2768 | ^{3}H | 20% |

2769   Therefore, the radionuclide mixture from characterization confirms the process knowledge that fission
2770   and activation products comprise the contamination. The mixture includes radionuclides that are readily-
2771   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).

2772   c. Is the material impacted?

2773   Yes, these small-bore pipe sections are certainly impacted, given that they were used to transport
2774   radioactive liquids.

2775   d. Specify release criteria and conditions for the solid material.

2776   For this example, Regulatory Guide 1.86 will be used. The surface activity guideline for all
2777   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
2778   1,000 dpm/100 cm^2.

2780 e. Classify the material.

2781 The small-bore pipe sections are Class 1. This classification is based on the fact that the material was
2782 designed to be in contact with radioactivity, as further supported by the characterization results.

2783 f. Is clearance an option?

2784 Yes, the licensee in this example has decided to perform a clearance survey.

2785 g. Consider the survey approach based on the nature of the material and contamination.

2786 Given that the interior of the pipe sections is potentially contaminated, it will be necessary to cut the
2787 pipes along their lengths (resulting in semi-cylindrical sections). The nature of the radioactivity suggests
2788 that beta-sensitive detectors would work well.

2789 h. Can scanning be used to release the material?

2790 Yes, the proposed clearance survey approach is to scan the interior of the semi-cylindrical pipe sections
2791 using GM detectors. Before this approach can be implemented, it is necessary to demonstrate that the
2792 scan MDC is less than the DCGL$_C$.

2793 i. Application of DCGLs.

2794 To demonstrate compliance with the clearance release criteria, the clearance survey will consist of
2795 surface scans with a GM detector. Given the radioactive decay emissions from these radionuclides, the
2796 GM will respond to gross beta radiation. Therefore, it is necessary to calculate the gross activity DCGL$_C$
2797 for surface activity using the following equation:

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

2798 where f$_1$, f$_2$, etc. are the fractional amounts of each radionuclide present.

2799 A simplifying observation is that 92 percent of the radionuclide mixture consists of radionuclides for
2800 which the surface activity guideline is 5,000 dpm/100 cm^2, while ^{90}Sr(^{90}Y) makes up 8 percent with a
2801 guideline of 1000 dpm/100 cm^2. Substituting into the above equation, the gross activity DCGL is 3,800
2802 dpm/100 cm^2.

2803 j. Determine background.

2804 Measurements were performed on similar, non-impacted pipe sections to determine the GM background;
2805 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}**

| 2828 | Using a surveyor efficiency of 0.5 and the total weighted efficiency of 1.9×10^{-2}, the scan MDC is |
| 2829 | calculated as |

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

2830   l. Is the scan MDC less than the $DCGL_C$?

2831   No, the scan MDC of 7,400 dpm/100 cm² (1.2 Bq/cm²) is not less than 3,800 dpm/100 cm² (0.6 Bq/cm²).

2832   m. Can the scan MDC be reduced?

2833	It is not likely that modifying the scanning parameters will lower the scan MDC to a value less than the
2834	$DCGL_C$. (Note: If the scan MDC could be sufficiently reduced below the $DCGL_C$, the next step is to
2835	evaluate the instrument's ability to automatically document scan results (step *o*).)[4]

2836   n. Is another clearance survey design feasible?

| 2837 | Since the scan MDC is not sufficiently sensitive, the next step is to determine whether conventional static |
| 2838 | measurements are feasible. Example 2 provides the details of the design. |

| 2839 | Example 2 | Clearance of small-bore pipes from nuclear power reactor (using statistical design for |
| 2840 | | static direct measurements) |

2841	Based on the information obtained in Example 1, step *h* in the flow diagram of Figure 2.1 results in the
2842	decision that scanning with a GM detector cannot be used to release the pipe sections. This example
2843	continues from step *n* in Example 1 (now at the right side of Figure 2.1).

2844   i. Application of DCGLs.

2845	To demonstrate compliance with the clearance release criteria, the clearance survey will consist of static
2846	direct measurements of surface activity using a GM detector. The gross activity $DCGL_C$ for surface
2847	activity determined in Example 1 is the same for this example (i.e., the gross activity $DCGL_C$ is
2848	3,800 dpm/100 cm²).

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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.

2849   j. Determine background.

2850   Fifteen measurements, as determined based on the WRS test (step p), were performed on non-impacted
2851   pipe sections to determine the GM background. The mean background was 60 cpm, with a standard
2852   deviation of 8 cpm.

2853   k. Determine the static MDC.

2854   The static MDC for the GM detector can be calculated as

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

2855   where C_B is the background count in time, T, for paired observations of the sample and blank, ϵ_i is the
2856   instrument efficiency, and ϵ_s is the surface efficiency. However, before the static MDC can be
2857   calculated, it is necessary to determine the total efficiency for the radionuclide mixture. [Note: The
2858   instrument efficiencies for the GM detector used for static measurements (based on the detector's
2859   response to a source area equal to its physical probe area of 20 cm^2) are higher than instrument
2860   efficiencies for the GM detector used for scanning (based on the detector's response to a source area of
2861   100 cm^2), by a factor of 5.]

	ϵ_i	ϵ_s	Radionuclide Fraction	Weighted Efficiency
2862 ^{60}Co	0.25	0.25	0.15	9.40×10^{-3}
2863 ^{137}Cs	0.40	0.5	0.27	5.40×10^{-2}
2864 ^{90}Sr	0.60	0.5	0.08	2.40×10^{-2}
2865 ^{14}C	0.15	0.25	0.13	4.88×10^{-3}
2866 ^{55}Fe	0	0.25	0.11	0
2867 ^{63}Ni	0.05	0.25	0.06	7.50×10^{-4}
2868 ^{3}H	0	0	0.2	0
2869		**Total Weighted Efficiency**		9.3×10^{-2}

2870   Therefore, the static MDC for the GM for 1-minute counts is

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

2873   l. Is the static MDC less than the DCGL$_C$?

2874   Yes, the static MDC of 2,100 dpm/100 cm² is less than the DCGL$_C$ of 3,800 dpm/100 cm².

2875   p. Perform clearance survey based on statistical sampling design for the number of direct measurements
2876   of surface activity.

2877   The WRS test can be used to determine the number of surface activity measurements needed for the
2878   clearance survey. The number of data points necessary for this material survey unit is determined
2879   through the DQO Process. Specifically, the sample size is based on the DCGL$_C$, the expected standard
2880   deviation of the radionuclides in the pipe sections, and the acceptable probability of making Type I and
2881   Type II decision errors.

2882   ● The gross activity DCGL$_C$ is 3,800 dpm/100 cm².

2883   ● Process knowledge, coupled with results from characterization surveys, was used to estimate the
2884   contamination on the pipe sections. The contamination, as measured in gross cpm with a GM
2885   detector, averaged 82 cpm, with a standard deviation of 18 cpm.

2886   ● Other DQO inputs include the LBGR set at the expected contamination level on the pipe sections (82
2887   − 60 cpm, or 22 cpm), and Type I and II errors of 0.05 and 0.01 respectively.

2888   The DCGL$_C$, and the expected standard deviation of the material survey unit and background
2889   measurements are used to estimate the relative shift, Δ/σ.

2890   First, it is necessary to convert the DCGL$_C$ into the same units as the standard deviation:

2891   *gross activity* DCGL$_C$ = (3,800 *dpm*/100 *cm*²)·(9.3*E*−2) 20/100 = 70.7 *cpm*

2892   The larger of the values of the estimated measurement standard deviations from the survey unit and the
2893   reference area should be used. Since the estimated standard deviation in the survey unit is 18 and that for
2894   the reference area is 8, the survey unit value of σ =18 will be used to calculate the relative shift.
2895   The relative shift can now be calculated: (70.7 − 22)/18 = 2.7.

2896   Table 5.3 in MARSSIM (1997) provides a list of the number of data points to demonstrate compliance
2897   using the WRS test for various values of Type I and II errors and Δ/σ. For α = 0.05 and β = 0.01, the
2898   required sample size is about 15 direct measurements for this material survey unit and 15 measurements
2899   on non-impacted pipe sections (background).

2900   The scan coverage for these pipe sections is 100 percent because of their classification (i.e., Class 1).
2901   Note, however, that the scan MDC is 7,400 dpm/100 cm^2; therefore surface activity levels between the
2902   DCGL$_C$ (3,800 dpm/100 cm^2) and the scan MDC will likely be missed during scanning. At a minimum,
2903   however, scanning can detect surface activity at a level of 7,400/3,800, or about two times the DCGL$_C$[5].
2904   A provision for area factors as a function of specific areas of materials may be appropriate to serve as a
2905   possible driver for collecting additional direct measurements. If not, the DQO Process should be used to
2906   assess the risk of missing an area with concentration between the DCGL$_C$ and the scan MDC, and
2907   whether the material is candidate for release.

2908   Direct measurement locations are determined by random number generation. Fifteen pairs of random
2909   numbers are generated, with the first number specifying the particular pipe section to be measured, and
2910   the second number determining the distance from the end of the pipe section for the direct measurement.

2911   <u>Example 3</u>   <u>Clearance of small-bore pipes from nuclear power reactor (using *in situ* gamma</u>
2912   <u>spectrometry)</u>

2913   This clearance survey approach is similar to the approach illustrated in Example 2, with two major
2914   exceptions. First, this approach does not require the pipes to be cut in half; in fact, the entire material
2915   survey unit is measured and results in minimal handling of the material. Second, the clearance survey is
2916   based on one "total" measurement, rather than a statistically based sampling design. Steps *a* through *f*
2917   are the same in Example 3 as they were for the first two examples. -

2918   g. Consider survey approach based on nature of material and contamination.

2919   Given that the interior of the pipe sections is potentially contaminated with some gamma-emitting
2920   radionuclides among the mix, the use of *in situ* gamma spectrometry (ISGS) is considered as a clearance
2921   survey approach.

2922   h. Can scanning be used to release material?

2923   The proposed clearance survey approach is to use ISGS measurements; therefore, scanning is not used to
2924   release the pipe sections.

2925   i. Application of DCGLs.

2926   Considering the radionuclide mixture provided in step *b* (shown in Example 1), ^{60}Co and ^{137}Cs comprise
2927   42 percent of the radioactivity. Therefore, these two radionuclides are measured using ISGS, and are
2928   used as surrogates for the entire mix of radionuclides. In order to use this approach, it is necessary to
2929   assume that this mixture is representative of the potential contamination on the pipe sections (refer to
2930   step *b*).

2931   It is necessary to convert the surface activity guidelines (from RG 1.86) to total activity limits. This is
2932   performed for each radionuclide by multiplying the surface activity guideline by the total surface area of
2933   the pipes in the material survey unit (17 m^2). For example, the total dpm that corresponds to 5,000
2934   dpm/100 cm^2 can be calculated as

2935 $(5{,}000 \text{ dpm}/100 \text{ cm}^2) \times (17 \text{ m}^2) \times (10{,}000 \text{ cm}^2/ 1 \text{ m}^2) = 8.5\text{E6 dpm}$

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

2934 Each of the radionuclides, with the exception of ^{90}Sr(^{90}Y), has a surface activity guideline of
2935 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,
2936 is 1.7x10^6 dpm.

2937 Returning to the use of ^{60}Co and ^{137}Cs as surrogates, it is necessary to modify the DCGL$_C$ for these two
2938 radionuclides to account for all of the other radionuclides. First, note that the limit for both ^{60}Co and
2939 ^{137}Cs is 8.5x10^6 dpm; therefore, when both are measured, the sum of both radionuclides should not
2940 exceed 8.5x10^6 dpm (when they are the only radionuclides present). Equation I-14 on page I-32 o f the
2941 MARSSIM can be used to calculate the modified DCGL$_C$ for Co+Cs:

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

2942 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
2943 radionuclide (^{90}Sr(^{90}Y)) that is being inferred by ^{60}Co and ^{137}Cs. R_2 is the ratio of concentration of the
2944 ^{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
2945 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
2946 calculated for the mixture as follows:

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

2947 Therefore, to demonstrate compliance, the ISGS result should be less than 2.7x10^6 dpm (1.22 µCi) for the
2948 sum of ^{60}Co and ^{137}Cs.

2949 j. Determine background.

2950 Since neither ^{60}Co nor ^{137}Cs is present naturally in the material (pipe sections), the background value (i.e.,
2951 Compton continuum) for each radionuclide's region of interest (ROI) was determined from an ambient
2952 count at the location where the pipe section clearance measurements will be performed. The count time
2953 should be long enough to result in sufficiently sensitive MDC.

2956   k. Determine static MDC.

2957   The static MDC for the *in situ* gamma spectrometer can be calculated as

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

2958   where BKG is the background continuum counts determined in time T, and ε is the efficiency in net peak
2959   counts per minute per activity (μCi or Bq). This MDC is the general MDC for the measurement process,
2960   rather than an individual MDC for each measurement.

2961   The measurement protocol consisted of four 10-minute measurements at the midpoint of each side of the
2962   material survey unit. The efficiency for a particular distribution of radioactivity within the pipe sections
2963   was determined by randomly positioning a known quantity of ^{60}Co and ^{137}Cs radionuclide sources within
2964   a non-impacted geometry of pipe sections. The efficiencies for the ^{60}Co (1,173 keV) ranged from 7.2 to
2965   17.3 net counts per minute per μCi, while the efficiencies for the ^{137}Cs ranged from 8.8 to 21.8 net counts
2966   per minute per μCi. To be conservative, the MDCs for both ^{60}Co and ^{137}Cs were calculated for the lowest
2967   efficiencies observed. The MDCs for ^{60}Co and ^{137}Cs were 0.6 and 0.5 μCi, respectively.

2968   l. Is the static MDC less than the $DCGL_C$?

2969   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
2970   were greater than the $DCGL_C$ of 1.22 μCi, step *m* would be performed to determine whether the MDCs
2971   could be reduced (e.g., by using longer count times).

2972   p. Perform *in toto* survey.

2973   Perform clearance survey based on ISGS measurements for ^{60}Co and ^{137}Cs. Each measurement consists
2974   of four 10-minute measurements at the midpoint of each side of the material survey unit. The total
2975   activity for both ^{60}Co and ^{137}Cs is summed, and then compared to the $DCGL_C$ of 1.22 μCi. Survey results
2976   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.

3016 When discrete sampling is involved, determining that the sampling design provides adequate power is
3017 important to decisionmaking, particularly in cases where the levels of contamination are near the $DCGL_c$.
3018 This can be done both prospectively, during survey design to test the efficacy of a proposed design, and
3019 retrospectively, during interpretation of survey results to determine that the objectives of the design are
3020 met. The procedure for generating power curves for specific tests is discussed in Appendix I to the
3021 MARSSIM. Note that the accuracy of a prospective power curve depends on estimates of the data
3022 variability, σ, and the number of measurements. After the data are analyzed, a sample estimate of the
3023 data variability, namely the sample standard deviation (s) and the actual number of valid measurements
3024 will be known. The consequence of inadequate power is that a survey unit that actually meets the release
3025 criterion has a higher probability of being incorrectly deemed *not* to meet the release criterion.

3026 ### 6.2.2 Conduct a Preliminary Data Review

3027 To learn about the structure of the data — identifying patterns, relationships, or potential anomalies —
3028 one can review quality assurance (QA) and quality control (QC) reports, prepare graphs of the data, and
3029 calculate basic statistical quantities.

3030 #### 6.2.2.1 Data Evaluation and Conversion

3031 Quality assurance reports that describe the data collection and reporting processes can provide valuable
3032 information about potential problems or anomalies in the data. EPA Report QA/G-9 (EPA 2000)
3033 recommends a review of (1) data validation reports that document the sample collection, handling,
3034 analysis, data reduction, and reporting procedures used; (2) quality control reports from laboratories or
3035 field stations that document measurement system performance, including data from check samples, split
3036 samples, spiked samples, or any other internal QC measures; and (3) technical systems reviews,
3037 performance evaluation audits, and audits of data quality, including data from performance evaluation
3038 samples. This report also suggests that when reviewing QA reports, particular attention should be paid to
3039 information that can be used to check assumptions made in the DQO Process, especially any anomalies in
3040 recorded data, missing values, deviations from standard operating procedures, or the use of nonstandard
3041 data collection methodologies.

3042 Verification of instrument calibrations and calculations of minimum detectable concentrations (MDCs)
3043 are particularly important to surveys of solid materials. Clearly, MDCs must be capable of detecting
3044 contamination at the $DCGL_c$. When making quantitative comparisons of the average of survey data to a
3045 limit, the MARSSIM recommends that the MDC target should be 10–50 percent of the $DCGL_c$. This is
3046 an expression of the fact that a simple detection decision does not address the relative uncertainty of the
3047 data value obtained. The minimum quantifiable concentration (MQC) is often defined as the smallest
3048 concentration that can be measured with a relative standard uncertainty of 10 percent. As a rule of thumb
3049 mentioned previously, the MDC is generally about 3 to 4 times the standard uncertainty of repeated
3050 background or blank measurements. An extension of this rule of thumb is that the MQC is about 10
3051 times the standard uncertainty. Hence, if one wishes to not merely detect but also quantify
3052 concentrations near the $DCGL_c$, the MQC should be no larger than the $DCGL_c$. Combining the
3053 approximations for the MQC as 10 times the uncertainty and the MDC as about 3 or 4 times the
3054 uncertainty, the MDC should be about one-third of the MQC. Thus, the recommendation that the MDC
3055 should be 10–50 percent of the $DCGL_c$ is really an expression of the fact that the MQC should be no
3056 larger than the $DCGL_c$.

3055 These rough guides can sometimes point out inconsistencies or shortcomings in the data analysis. For
3056 example, suppose that the $DCGL_C$ is 200, and the claimed MDC is 100. Data are then reported as
3057 100±75, 50±75, -25±50, and 75±75. The relative uncertainties are rather high. Are they consistent with
3058 the quoted MDC? If the MDC is estimated as 3 to 4 times these uncertainties, we get values of 150 to
3059 300, much higher than the quoted 100. This is an indication that the data quality targets are not being
3060 met.

3061 Radiological survey data are usually obtained in units, such as the number of counts per unit time, that
3062 have no intrinsic meaning relative to DCGLs. For comparison of survey data to DCGLs, the survey data
3063 from field and laboratory measurements are converted to DCGL units.

3064 Basic statistical quantities that should be calculated for the sample data set are as follows:

3065 • mean
3066 • standard deviation
3067 • median

3068 Example:

3069 Suppose the following 10 measurement values are from a survey unit composed of materials:

3070   9.1,   10.7,   13.6, 3.4,   13.3,   7.9,   4.5,   7.7,   8.3,   10.4
3071

3072 First, the average of the data (8.88) and the sample standard deviation (3.3) should be calculated.

3073 These next 10 measurements are from an appropriate matching reference material:

3074   6.2,   13.8,   15.2, 9.3,   6.7,   4.9,   7.1,   3.6,   8.8,   8.9.

3075 The average of these data is 8.45 and the standard deviation is 3.7.

3076 The average of the data can be compared to the reference material average and the $DCGL_C$ to get a
3077 preliminary indication of the survey unit status. The difference in this case is 0.43.

3078 Where there is much added activity, this comparison may readily reveal that the material survey unit
3079 should not be released — even before applying statistical tests. For example, if the difference between
3080 the survey unit and reference material averages of the data exceeds the $DCGL_C$, the survey unit clearly
3081 does not meet the release criterion. On the other hand, if the difference between the largest survey unit
3082 measurement (13.6) and the smallest reference material measurement (3.6) is below the $DCGL_C$, the
3083 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.

3086 The value of the sample standard deviation is especially important. If it is too large (compared to that
3087 assumed during the survey design), this may indicate that an insufficient number of samples were
3088 collected to achieve the desired power of the statistical test. Again, inadequate power can lead to an
3089 increased probability of incorrectly failing a material survey unit.

3090 The median is the middle value of the data set when the number of data points is odd, and is the average
3091 of the two middle values when the number of data points is even. Thus 50 percent of the data points are
3092 above the median, and 50 percent are below the median. Large differences between the mean and
3093 median would be an early indication of a skew in the data. This would also be evident in a histogram of
3094 the data. For the example data above, the median is 8.7 (i.e., (8.3 + 9.1)/2). The difference between the
3095 median and the mean (i.e., 8.45 - 8.7 = -0.25) is a small fraction of the sample standard deviation
3096 (*i.e.*, 3.3). Thus, in this instance, the mean and median would not be considered significantly different.

3097 Examining the minimum, maximum, and range of the data may provide additional useful information.
3098 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
3099 only 3.1 standard deviations. Thus, the range is not unusually large. When there are 30 or fewer data
3100 points, values of the range much larger than about 4 to 5 standard deviations would be unusual.
3101 For larger data sets, the range might be wider.

3102 ### 6.2.2.2 Graphical Data Review

3103 Graphical data review may consist of a posting plot and a histogram or quantile plots. A *posting plot* is
3104 simply a map of the survey unit with the data values entered at the measurement locations. This
3105 potentially reveals heterogeneities in the data, especially possible patches of elevated contamination.
3106 Even in a reference material survey, a posting plot can reveal spatial trends in background data, which
3107 might affect the results of the two-sample statistical tests. Posting plots are most useful when the data
3108 are obtained by discrete measurements.

3109 If the posting plot reveals systematic spatial trends in the survey unit, the cause of the trends would need
3110 to be investigated. In some cases, such trends could be attributable to contamination, but they may also
3111 be caused by inhomogeneities in the survey unit background. Other diagnostic tools for examining
3112 spatial data trends may be found in EPA Guidance Document QA/G-9.

3113 The role of a posting plot for a CSM would be a time series display of the data, showing any trends
3114 between adjacent batches of material being conveyed beneath the detector.

3115 However, the geometric configuration of most survey units composed of a few large irregularly shaped
3116 pieces of material is transitory. The arrangement of tools, piles of scrap, and the like will change as
3117 pallets of material are moved around and even while pieces are lifted to be surveyed. In these cases,
3118 some identifying marks, numbers, or bar-code labels should be used to identify and track where
3119 measurements were made, at least until it is determined that the material can be released. Such marking
3120 or labeling need not be permanent, but may be made with chalk and removable labels.

88

3121 A *frequency plot* (or histogram) is a useful tool for examining the general shape of a data distribution.
3122 This plot is a bar chart of the number of data points within a certain range of values. A frequency plot
3123 reveals any obvious departures from symmetry, such as skewing or bimodality (two peaks), in the data
3124 distributions for the survey unit or reference material. The presence of two peaks in the survey unit
3125 frequency plot may indicate the existence of isolated areas of contamination. In some cases, it may be
3126 possible to determine an appropriate background for the survey unit using this information.
3127 The interpretation of the data for this purpose is generally highly dependent on site-specific
3128 considerations and should only be pursued after a consultation with the responsible regulatory agency.

3129 The presence of two peaks in the background reference material or survey unit frequency plot may
3130 indicate a mixture of background concentration distributions as a result of different soil types,
3131 construction materials, etc. The greater variability in the data caused by the presence of such a mixture
3132 reduces the power of the statistical tests to detect an adequately decontaminated survey unit. These
3133 situations should be avoided whenever possible by carefully matching the background reference
3134 materials to the survey units, and choosing material survey units with homogeneous backgrounds.

3135 Skewness or other asymmetry can impact the accuracy of the statistical tests. A data transformation
3136 (e.g., taking the logarithms of the data) can sometimes be used to make the distribution more symmetric.
3137 The statistical tests would then be performed on the transformed data. When the underlying data
3138 distribution is highly skewed, it is often because there are a few high activity concentration areas. Since
3139 scanning is used to detect such areas, the difference between using the median and the mean as a measure
3140 for the degree to which uniform contamination remains in a survey unit tends to diminish in importance.

3141 When data are obtained from scanning surveys alone using data loggers, a large number of data points is
3142 usually logged. In essence, the entire Class 1 material survey unit is measured and, while the survey
3143 coverage is less for Class 2 and 3 materials, there will still likely be a large number of data points. In this
3144 case, the frequency plot will be close to the population distribution of concentrations in the survey unit.
3145 The mean and standard deviation calculated from these logged values should be very close to their
3146 population values. In other words, when nearly the entire material survey unit has been measured,
3147 statistical sampling is unnecessary.

3148 Similarly, when an *in toto* measurement has been performed, the entire survey unit has been measured.
3149 Again, statistical sampling is not necessary.

3150 For conveyorized survey monitors, the data may be interpreted batch by batch as it is scanned, in which
3151 case, the data treatment would be most similar to an *in toto* measurement. If the data were logged
3152 continuously, the data treatment would be similar to that for a scanning survey using data loggers.

3153 **6.2.3 Select the Tests**

3154 As mentioned above, when data are obtained from scanning surveys alone using data loggers, a large
3155 number of data points is usually logged. In essence, the entire survey unit is measured. The mean and
3156 the standard deviation calculated from these logged values should be very close to their population
3157 values. In other words, when the entire survey unit has been measured, statistical sampling is
3158 unnecessary, as are statistical tests. There is no uncertainty contribution from spatial variability in survey
3159 unit concentrations because the entire survey unit has been measured. The average of the logged values
3160 may simply be compared to the $DCGL_c$. However, there remains an uncertainty component as a result of
3161 the variability in the measurement process. Measurement variability, unlike spatial variability, can often
3162 be modeled realistically using a normal distribution. In that case, parametric statistical tests may be more
3163 appropriate; however, because removing spatial variability is often the major concern in these surveys, it
3164 is suggested that a simple comparison of the mean to the $DCGL_c$ is sufficient. As long as the
3165 measurement uncertainty is a small fraction of the $DCGL_c$, the gray region should be very narrow.

3166 When an *in toto* measurement has been performed, the entire survey unit has been measured. Only a
3167 single measurement is made, and so the decision is really a detection decision. The statistical test is that
3168 used to calculate the MDC. However, assumptions are made about the distribution of activity inherent in
3169 the calibration of such detectors, and the validity of those assumptions determines the appropriateness of
3170 the measurement.

3171 Again, data from conveyorized survey monitors may be treated as a series of detection decisions on a
3172 batch-by-batch basis, or may be analyzed by aggregating the data, much as with a logging scanner.

3173 When conventional surveys are used, they should address the statistical considerations important for
3174 clearance surveys, as presented in Section 5.2.3.3. The statistical tests recommended for conventional
3175 clearance surveys are the same as those recommended by the MARSSIM for final status surveys of lands
3176 and structures.

3177 The most appropriate procedure for summarizing and analyzing the data is chosen based on the
3178 preliminary data review. The parameter of interest is the mean concentration in the material survey unit.
3179 The nonparametric tests recommended in this report, in their most general form, are tests of the median.
3180 If one assumes that the data are from a symmetric distribution — where the median and the mean are
3181 effectively equal — these are also tests of the mean. If the assumption of symmetry is violated,
3182 nonparametric tests of the median only approximately test the mean. Note that the mean and median only
3183 differ greatly when large concentration values skew the distribution. Such areas can be identified while
3184 scanning. This is precisely why the survey strategies in this report emphasize using *both* direct
3185 measurements and scans. In addition, computer simulations (e.g., Hardin and Gilbert, 1993) have shown
3186 that the approximation of the mean by the median implicit in using the nonparametric tests is a fairly
3187 good technique as far as decisionmaking is concerned. That is, the correct decision will be made about
3188 whether the mean concentration exceeds the DCGL, even when the data come from a skewed
3189 distribution. In this regard, the nonparametric tests are found to be correct more often than the
3190 commonly used Student's t test. The robust performance of the Sign and WRS tests over a wide range of
3191 conditions is the reason that they are recommended in this report.

3192 When a given set of assumptions is true, a parametric test designed for exactly that set of conditions will
3193 have the highest power. For example, if the data are from a normal distribution, the Student's t test will
3194 have higher power than the nonparametric tests. It should be noted that for large enough sample sizes
3195 (e.g., large number of measurements), the Student's t test is not a great deal more powerful than the
3196 nonparametric tests. On the other hand, when the assumption of normality is violated, the nonparametric
3197 tests can be very much more powerful than the t test. Therefore, any statistical test may be used,
3198 provided that the data are consistent with the assumptions underlying their use. When these assumptions
3199 are violated, the prudent approach is to use the nonparametric tests, which generally involve fewer
3200 assumptions than their parametric equivalents.

3201 The one-sample statistical test (Sign test) described in Section 5.5.2.3 of the MARSSIM should only be
3202 used if the radionuclide being measured is not present in background and radionuclide-specific
3203 measurements are made. The one-sample test may also be used if the radionuclide is present at such a
3204 small fraction of the $DCGL_C$ value as to be considered insignificant. In this case, background
3205 concentrations of the radionuclide are included with any contamination that may be present (i.e., the
3206 entire amount is attributed to facility operations). Thus, the total concentration of the radionuclide is
3207 compared to the release criterion. This option should only be used if one expects that ignoring the
3208 background concentration will not affect the outcome of the statistical tests. The advantage of ignoring a
3209 small background contribution is that no reference material is needed. This can simplify the survey
3210 considerably.

3211 The one-sample Sign test (Section 6.3.1) evaluates whether the median of the data is above or below the
3212 $DCGL_C$. If the data distribution is symmetric, the median is equal to the mean. In cases where the data
3213 are severely skewed, the mean may be above the $DCGL_C$, while the median is below the $DCGL_C$. In such
3214 cases, the survey unit does *not* meet the release criterion regardless of the result of the statistical tests.
3215 On the other hand, if the largest measurement is below the $DCGL_C$, the Sign test will *always* show that
3216 the survey unit meets the release criterion, provided that enough samples were taken to meet the DQOs.

3217 For clearance surveys, the two-sample statistical test (WRS test, discussed in Section 5.5.2.2 of the
3218 MARSSIM) should be used when the radionuclide of concern appears in background or if measurements
3219 are used that are not radionuclide-specific. The two-sample WRS test (Section 6.4.1) assumes the
3220 reference material and survey unit data distributions are similar except for a possible shift in the medians.
3221 When the data are severely skewed, the value for the mean difference may be above the $DCGL_C$, while
3222 the median difference is below the $DCGL_C$. In such cases, the survey unit does *not* meet the release
3223 criterion regardless of the result of the statistical test. On the other hand, if the difference between the
3224 largest survey unit measurement and the smallest reference material measurement is less than the
3225 $DCGL_C$, the WRS test will always show that the survey unit meets the release criterion, provided that
3226 enough samples were taken to meet the DQOs.

3227 ## 6.2.4 Verify the Assumptions of the Tests

3228 An evaluation to determine that the data are consistent with the underlying assumptions made for the
3229 statistical procedures helps to validate the use of a test. One may also determine that certain departures
3230 from these assumptions are acceptable when given the actual data and other information about the study.
3231 The nonparametric tests described in this chapter assume that the data from the reference material or
3232 survey unit consist of independent samples from each distribution.

3233 Asymmetry in the data can be diagnosed with a stem and leaf display, a histogram, or a Quantile plot.
3234 As discussed in the previous section, data transformations can sometimes be used to minimize the effects
3235 of asymmetry.

3236 One of the primary advantages of the nonparametric tests used in this report is that they involve fewer
3237 assumptions about the data than their parametric counterparts. If parametric tests are used,
3238 (e.g., Student's t test), any additional assumptions made in using them should be verified (e.g., testing for
3239 normality). These issues are discussed in detail in EPA QA/G-9 (EPA 1998b).

3240 One of the more important assumptions made in the survey design is that the sample sizes determined for
3241 the tests are sufficient to achieve the data quality objectives set for the Type I (α) and Type II (β) error
3242 rates. Verification of the power of the tests (1-β) to detect adequate probability for passing material
3243 survey units that meet the criteria for clearance may be of particular interest. Methods for assessing the
3244 power are discussed in Appendix I.9 to the MARSSIM. If the hypothesis that the material survey unit
3245 radionuclide concentration exceeds the clearance criterion is accepted, there should be reasonable
3246 assurance that the test is equally effective in determining that a survey unit has radionuclide
3247 concentrations less than the $DCGL_C$. Otherwise, unnecessary survey unit failures may result. For this
3248 reason, it is better to plan the surveys cautiously, even to the following extents:

3249 ● overestimating the potential data variability
3250 ● taking too many samples
3251 ● overestimating the minimum detectable concentrations (MDCs)

3252 If one is unable to show that the DQOs are met with reasonable assurance, a resurvey may be needed.

3253 When data are obtained from scanning surveys alone using data loggers, the mean of the logged values
3254 may simply be compared to the $DCGL_C$. Because such a large number of data points are obtained,
3255 essentially the entire population of concentrations on the material has been measured. Thus, no formal
3256 statistical test is necessary. It is the assumption of full measurement coverage that is the central issue in
3257 this case. It is also assumed that the measurement uncertainty is small compared to the $DCGL_C$.
3258 The validity of these assumptions should be carefully examined, and the results documented in the SOPs
3259 and QAPP.

3260 When an *in toto* measurement has been performed, the entire survey unit has been measured. Again,
3261 statistical sampling is not necessary. However, assumptions are made about the distribution of activity
3262 inherent in the calibration of such detectors, and the validity of those assumptions determines the
3263 appropriateness of the measurement.

3264 Examples of assumptions and possible methods for their assessment are summarized in Table 6.1.

92

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

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.

3324   **6.3.1   Applying the Sign Test**

3325   The Sign test is applied by counting the number of measurements in the survey unit that are less than the
3326 $DCGL_C$. The result is the test statistic S+. Discard any measurement that is exactly equal to the $DCGL_C$
3327   and reduce the sample size, N, by the number of such measurements. The value of S+ is compared to the
3328   critical values in MARSSIM Table I.3. If S+ is greater than the critical value, k, in that table, the null
3329   hypothesis is rejected.

3330   **6.3.2   Sign Test Example: Class 1 Copper Pipes**

3331   This example illustrates the clearance survey design for copper pipe sections using a gas proportional
3332   counter to measure ^{239}Pu. Since the alpha background on the copper material is essentially zero, it was
3333   decided to use the Sign test to determine whether the material meets the clearance criterion. The sample
3334   size was determined using the DQO Process, with inputs such as the $DCGL_C$, the expected standard
3335   deviation of the radionuclide concentrations in the pipe sections, and the acceptable probability of
3336   making Type I and Type II decision errors. The inputs were as follows:

3337   ●   The gross activity $DCGL_C$ was 100 dpm/100 cm^2. When converted to cpm, the gross activity $DCGL_C$
3338   was 10 cpm.

3339   ●   The LBGR was set at the expected added activity level on the copper pipe sections (i.e., 5 net cpm —
3340   the same as the gross mean for an alpha background of zero).

3341   ●   The standard deviation on the material survey unit was expected to be about 2 cpm.

3342   ●   The relative shift was calculated as (10 - 5)/2 = 2.5

3343   ●   The Type I and II errors were set at 0.05.

3344   Table 5.5 in the MARSSIM (1997) indicates that the number of measurements estimated for the Sign
3345   Test, N, is 15 ($\alpha = 0.05$, $\beta = 0.05$, and $\Delta/\sigma = 2.5$). Therefore, 15 surface activity measurements were
3346   randomly collected from the inside surfaces of the copper pipe sections. Clearance survey results are
3347   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 α = 0.05, β = 0.10, and Δ/σ =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 α = 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.

3381 **6.4   WRS Test**

3382   The statistical tests discussed in this section will be used to compare each material survey unit with an
3383   appropriately chosen, site-specific reference material. Each reference material should be selected on the
3384   basis of its similarity to the survey unit, as discussed in Section 5.2.3.3. Further information on the WRS
3385   Test can be found in Section 8.4 of the MARSSIM and Chapter 6 of NUREG 1505, Rev.1.

3386   **6.4.1   Applying the WRS Test**

3387   The WRS test is applied as outlined in the following six steps and further illustrated by the example in
3388   Section 6.4.2.

3389   (1)   Obtain the adjusted reference material measurements, Z_i, by adding the $DCGL_C$ to each
3390   reference material measurement, X_i. $Z_i = X_i + DCGL_C$.

3391   (2)   The m adjusted reference sample measurements, Z_i, from the reference material and the n sample
3392   measurements, Y_i, from the survey unit are pooled and ranked in order of increasing size from 1
3393   to N, where $N = m+n$.

3394   (3)   If several measurements are tied (i.e., have the same value), they are all assigned the average
3395   rank of that group of tied measurements.

3396   (4)   If there are t "less than" values, they are all given the average of the ranks from 1 to t. Therefore,
3397   they are all assigned the rank $t(t+1)/(2t) = (t+1)/2$, which is the average of the first t integers. If
3398   there is more than one detection limit, all observations below the largest detection limit should be
3399   treated as "less than" values.[7]

3400   (5)   Sum the ranks of the adjusted measurements from the reference material, W_r. Note that since the
3401   sum of the first N integers is $N(N+1)/2$, one can equivalently sum the ranks of the measurements
3402   from the survey unit, W_s, and compute $W_r = N(N+1)/2 - W_s$.

3403   (6)   Compare W_r with the critical value given in Table I.4 of the MARSSIM for the appropriate
3404   values of n, m, and α. If W_r is greater than the tabulated value, reject the hypothesis that the
3405   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.

3406   **6.4.2   WRS Test Example: Class 2 Metal Ductwork**

3407   This example illustrates the use of the WRS test for releasing Class 2 metal ductwork. Assume that a gas
3408   proportional detector was used to make gross (non-radionuclide-specific) surface activity measurements.
3409

3410   The DQOs for this survey unit include $\alpha = 0.05$ and $\beta = 0.05$, and the $DCGL_C$ converted to units of gross
3411   cpm is 2,300 cpm. In this case, the two-sample nonparametric WRS statistical test was used because the
3412   estimated background level (2,100 cpm) was large compared to the DCGL. The estimated standard
3413   deviation of the measurements, σ, was 375 cpm. The estimated added activity level was 800 cpm; the
3414   LBGR will be set at this value. The relative shift can be calculated as $\Delta/\sigma = (DCGL_C - LBGR)/\sigma$, which
3415   equals 4.

3416   The sample size needed for the WRS test can be found in Table 5.3 of the MARSSIM for these DQOs.
3417   The result is nine measurements in each survey unit and nine in each reference material ($\alpha = 0.05$,
3418 $\beta = 0.05$, and $\Delta/\sigma = 4$). The ductwork was laid flat onto a prepared grid, and the nine measurements
3419   needed in the survey unit were made using a random-start triangular grid pattern. For the reference
3420   materials, the measurement locations were chosen randomly on a suitable batch of material. Table 6.4
3421   lists the gross count rate data obtained.

3422   In column B, the code "R" denotes a reference material measurement, and "S" denotes a survey unit
3423   measurement. Column C contains the adjusted data, which were obtained by adding the $DCGL_C$ to the
3424   reference material measurements (see Section 6.4.1, Step 1). The ranks of the adjusted data appear in
3425   Column D. They range from 1 to 18, since there is a total of 9+9 measurements (see Section 6.4.1,
3426   Step 2). Note that the sum of *all* of the ranks is still $18(18+1)/2 = 171$. Checking this value with the
3427   formula in Step 5 of Section 6.4.1 is recommended to guard against errors in the rankings.

3428   Column E contains only the ranks belonging to the reference material measurements. The total is 126.
3429   This is compared with the entry for the critical value of 104 in Table I.4 of the MARSSIM for $\alpha = 0.05$,
3430   with $n = 9$ and $m = 9$. Since the sum of the reference material ranks is greater than the critical value, the
3431   null hypothesis (i.e., that the average survey unit concentration exceeds the $DCGL_C$) is rejected, and the
3432   ductwork is released.

3433   Note that this conclusion could be reached much more quickly by noting that the largest survey unit
3434   measurement, 3,423, differs from the smallest reference material measurement, 1,427, by much less than
3435   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
3437	1					
3438	2	2180	R	4480	15	15
3439	3	2398	R	4698	16	16
3440	4	2779	R	5079	18	18
3441	5	1427	R	3727	10	10
3442	6	2738	R	5038	17	17
3443	7	2024	R	4324	13	13
3444	8	1561	R	3861	11	11
3445	9	1991	R	4291	12	12
3446	10	2073	R	4373	14	14
3447	11	2039	S	2039	3	0
3448	12	3061	S	3061	8	0
3449	13	3243	S	3243	9	0
3450	14	2456	S	2456	7	0
3451	15	2115	S	2115	4	0
3452	16	1874	S	1874	2	0
3453	17	1703	S	1703	1	0
3454	18	2388	S	2388	6	0
3455	19	2159	S	2159	5	0
			Sum =		171	126

3456   **6.5   Evaluating the Results: The Decision**

3457 Once the data and the results of the tests are obtained, the specific steps required to achieve material
3458 clearance depends on the procedures approved by the regulator and specific considerations to ensure that
3459 the contamination is as low as is reasonably achievable (ALARA). The following considerations are
3460 suggested for the interpretation of the test results with respect to the release limit established for
3461 clearance. Note that the tests need not be performed in any particular order.

3462 **6.5.1 Interpreting Data for Each Survey Type**

3463 Clearance survey designs using conventional instrumentation are as follows:

3464 • Scanning-Only

3465 • Calculate the average and compare it to DCGL.
3466 • Investigate measurements exceeding the DCGL.
3467 • Anything above the DCGL will trigger a reevaluation of the classification if Class 2.
3468 • Any contamination will trigger a reevaluation of the classification if Class 3.

3469 • Statistically Based Sampling

3470 • Techniques are similar to those used in MARSSIM.
3471 • Survey unit must pass statistical tests.
3472 • Sampling involves investigations of individual measurements/scans (as for scanning-only).

3473 • Automated Scanning Surveys (conveyorized survey monitors)

3474 • Scan sensitivity and ongoing QA data must be documented.
3475 • The statistical tests are essentially those used to calculate the MDC as discussed in Section 3.
3476 • "Batch-by-batch" segmented gate systems segregate any material above the clearance DCGL.
3477 • Data from continuous scanning of materials can be interpreted in the same way as for
3478 scanning-only surveys.
3479
3480 • *In Toto* Surveys
3481 • Emphasis is on adequate documentation of calibration.
3482 • A single measurement is compared to the DCGL.
3483 • A realistic estimate of the MDC is essential.

3484   **6.5.2   If the Survey Unit Fails**

3485   When a material survey unit fails to demonstrate compliance with the clearance criterion, the first step is
3486   to review and confirm the data that led to the decision. Once this is done, the DQO Process can be used
3487   to identify and evaluate potential solutions to the problem. The level of contamination on the material
3488   should be determined to help define the problem. For example, if only one or two pieces of material in a
3489   Class 1 material survey unit fail, the simplest solution might be to segregate those pieces and either
3490   remove the added activity from them or dispose of them as waste. If such a situation were encountered in
3491   evaluating Class 2 or Class 3 material survey units, it would call into question the entire classification
3492   procedure, and would require that the material at hand be reclassified and treated as Class 1.

3493   As a general rule, it may be useful to anticipate possible modes of failure. These can be formulated as
3494   the problem to be solved using the DQO Process. Once the problem has been stated, the decision
3495   concerning the failing survey unit can be developed into a decision rule (for example, whether to attempt
3496   to remove the radioactivity or simply segregate certain types of units as waste). Next, determine the
3497   additional data, if any, needed to document that a survey unit with elevated pieces removed or areas of
3498   added activity removed demonstrates compliance with the clearance criterion. Alternatives to resolving
3499   the decision rule should be developed for each type of material survey unit that may fail the surveys.
3500   These alternatives can be evaluated against the DQOs, and a clearance survey strategy that meets the
3501   objectives of the project can be selected.

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3617
3618
3619 *calibration*: comparison of a measurement standard, instrument, or item with a standard or instrument of
3620 higher accuracy to detect and quantify inaccuracies, and to report or eliminate those inaccuracies by
3621 making adjustments.

3622 *Class 1 materials*: solid materials that have (or had) a potential for contamination (based on process
3623 knowledge) or known contamination (based on previous surveys) above the release criteria (DCGL$_c$).

3624 *Class 2 materials*: solid materials that have (or had) a potential for or known contamination, but are not
3625 expected to be above the release criteria (DCGL$_c$).

3626 *Class 3 materials*: solid materials that are not expected to contain any contamination, or are expected to
3627 contain contamination less than a small fraction of the release criteria (DCGL$_c$) based on process
3628 knowledge or previous surveys.

3629 *clearance*: release of solid materials that do not require further regulatory control.

3630 *critical level*: the net count, or final instrument measurement result after appropriate calibration and/or
3631 correction factors have been applied, at or above which a decision is made that activity is present in a
3632 sample. When the observed net count is less than the critical level, the surveyor correctly concludes that
3633 no net activity is present in the sample.

3634 *detection limit*: the smallest number of net counts, or final instrument measurement result after
3635 appropriate calibration and/or correction factors have been applied, that will be detected with a
3636 probability (β) of non-detection, while accepting a probability (α) of incorrectly deciding that activity is
3637 present in a sample.

3638 *impacted*: materials that have some contamination potential, and therefore require a clearance survey in
3639 order to be released.

3640 *inaccessible areas*: locations on the surface of a solid material, which are not accessible for direct survey
3641 evaluation without cutting or dismantling the material. These inaccessible areas include the interior
3642 surfaces of pipes and scrap equipment such as pumps, motors, and other equipment.

3643 *instrument efficiency, ε_i*: similar to the intrinsic efficiency of a detector, the instrument efficiency is the
3644 ratio between the instrument net count rate and the surface emission rate of a source under specified
3645 geometric conditions. For a given instrument, the instrument efficiency depends on the radiation energy
3646 emitted by the source and the geometry between the detector and the source. Instrument efficiency is a
3647 2π value and shall only be used in surface activity determinations when multiplied by a surface efficiency
3648 to yield a 4π value of total efficiency.
3649
3650 *in toto*: a clearance survey technique that measures the entire material (or materials) at once.

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Glossary (continued)

3652 *measurement quality objective* (MQO): a statement of performance objective or requirement for a
3653 particular method performance characteristic. Like DQOs, MQOs can be quantitative or qualitative
3654 statements. An example of a quantitative MQO would be a statement of a required method uncertainty at
3655 a specified radionuclide concentration, such as the action level [i.e., "a method uncertainty of 3.7 Bq/kg
3656 (0.10 pCi/g) or less is required at the action level of 37 Bq/kg (1.0pCi/g)"]. An example of a qualitative
3657 MQO would be a statement of the required specificity of the analytical protocol, such as the ability to
3658 anaquantify the amount of ^{226}Ra present given high levels of ^{235}U in the samples.

3659 *minimum detectable concentration (MDC)*: the smallest activity concentration that can be detected with
3660 specific confidence for a given instrument and specific measurement procedure. The MDC is usually
3661 specified as the smallest activity concentration that can be detected with 95 percent confidence (i.e.,
3662 95 percent of the time a given instrument and measurement procedure will detect activity at the MDC).

3663 *minimum detectable count rates (MDCR)*: the detector signal level, or count rate for most equipment, that
3664 is likely to be flagged by a surveyor as being "greater than background."

3665 *non-impacted materials*: materials that have no reasonable possibility of having contamination.
3666 These materials may be used for background reference measurements.

3667 *process knowledge*: the use of operational information to assess the contamination potential of solid
3668 materials considering the location and use of the materials during operations.

3669 *real property*: land and building structures and equipment or fixtures (e.g., ductwork, plumbing, built-in
3670 cabinets) that are installed in a building in a more or less permanent manner.

3671 *scanning*: a survey technique performed by moving a detector over a surface at a specified speed and
3672 distance above the surface to detect radiation, usually via the audible output of the instrument.

3673 *secular equilibrium*: the condition that exists between the parent and other members of a decay series
3674 when the parent radionuclide decays much more slowly than any of the other members of the series.
3675 During secular equilibrium, the activity of the parent and each daughter radionuclide is equal.

3676 *solid materials (also non-real property)*: as opposed to lands and structures, materials such as
3677 tools/equipment, office items, consumable items, and debris that are offered for clearance.

3678 *spectrometer*: a device that measures energy (specifically, radiation energy).

3679 *surface efficiency, ε_s*: ratio between the number of particles of a given radiation type emerging from the
3680 surface per unit time (surface emission rate) and the number of particles of the same type released within
3681 the source per unit time. The surface efficiency is nominally 0.5, but may be increased by backscattered
3682 radiation and reduced by self-absorption.

3683

3684 *surrogate*: a radionuclide that is measured for the purpose of inferring the radionuclide concentration of
3685 one or more radionuclides that are not measured.

3686 *survey unit, material (lots/batches)*: a specified amount of solid material for which a separate decision
3687 will be made as to whether the unit meets the release criteria for clearance.

3688 *total efficiency, ε_T*: similar to the absolute efficiency of a detector, the total efficiency is the ratio of the
3689 detector response (e.g., in counts) and the number of particles emitted by the source. The total efficiency
3690 is contingent not only on detector properties, but also on the details of the counting geometry, surface
3691 characteristics, and other environmental conditions. The total efficiency (a 4π value) is the product of
3692 the instrument and surface efficiencies.

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

3694   **A.1   Introduction**

3695   This appendix introduces some basic properties of radiation, which are relevant to the measurement of
3696   residual radioactivity in and on solid materials. To provide a generic discussion, this appendix avoids
3697   mentioning or referring to a specific amount of radioactivity. Instead, this appendix focuses on some of
3698   the fundamental principles of radiation detection and measurement. It must be understood that the assay
3699   of residual radioactivity in and on solid materials is not simply a matter of radiation detection; rather, it
3700   involves (to some extent), identifying the presence of specific radionuclides, and quantifying their
3701   specific activities, while satisfying quality assessment objectives. This can be accomplished in a variety
3702   of ways, depending on the nature and type of material, the radionuclides involved, and the distribution of
3703   the radioactivity. It is unlikely that any single detector or method can cover all possible scenarios.

3704   Radionuclides are identified by measuring their nuclear properties, which are usually expressed by the
3705   energy of the radiation emitted as a result of nuclear transformations. Measurement of the radiation
3706   energy, along with a nuclear decay table, provides a method of identifying radionuclides. In situations
3707   where the measurement of the energy is difficult or impossible, the measurement of the nuclear mass
3708   (also known as mass spectroscopy) can also be used. This appendix focuses on techniques that use
3709   energy spectroscopy.

3710   **A.2   Measurement of Radioactivity: Decay Counting**

3711   In the majority of applications, radioactivity is usually measured using an indirect method, which
3712   requires a standard of known activity from which a calibration is obtained. Basically, the radioactivity
3713   (decays per unit time) is measured by counting the number of events in a detector for a specified interval
3714   of time (this interval is referred to as the "count time"). These events, which usually take the form of
3715   electronic pulses, result from the interaction of the radiation with the active (sensitive) components of the
3716   detector. The number of events is proportional to the radioactivity of the source. Once the detector is
3717   calibrated, using a standard source under reproducible conditions, the radioactivity can be quantified.
3718   A more complete discussion of radioactivity measurements, both direct and indirect, may be found in
3719   NCRP Report 58.

3720   For the assay of residual radioactivity in and on solid materials, a comprehensive set of reference
3721   materials does not exist to cover the range of conditions needed to develop an instrument calibration.
3722   The range of conditions refers to the geometry of the measurement system and source, as well as the
3723   disposition and quantity of any material absorbing or scattering radiation. The term calibration, in this
3724   context, presumes that the reference material has traceability to a national certifying organization, such as
3725   the U.S. National Institute of Standards and Technology (NIST) or the International Atomic Energy
3726   Agency (IAEA).

3727   The challenge for instrument developers is to extrapolate from the limited supply of available reference
3728   materials enough information and data to produce meaningful results. For example, the calibration of a
3729   radiation detector or detector system for a large-area (or volume) source, in some cases, can be obtained
3730   through a series of measurements using a certified point source (Becker *et al.*, 1999).

3731 The concept of calibration is evolving to encompass techniques that do not use actual sources, but rather
3732 simulate a calibration source. The simulation method relies on knowledge of and experience with
3733 radiation transport coupled with fast and powerful computers. The radiation transport code, called
3734 Monte Carlo N-Particle (MCNP), employs Monte Carlo methods to simulate radiation transport for
3735 neutrons, photons, and electrons for a wide variety of energies, materials, and geometries (Briesmeister,
3736 1993). The MCNP code provides a resource for investigators to test the response of their instruments to
3737 a variety of measurement conditions, which ultimately can lead to a calibration. It must be emphasized,
3738 however, that the quality or accuracy of a calibration developed using a simulation is predicated on the
3739 quality or accuracy of the transport code and the degree to which the simulation reflects the actual
3740 conditions of the measurement.

A.3 Statistical Models of Nuclear Decay

3742 Radioactive decay is a stochastic or random process. Any measurement of radioactivity has an inherent
3743 variation attributable to the random fluctuations associated with the decay process. Three statistical
3744 models are used to describe and quantify these random fluctuations under different circumstances:

3745 • Binomial distribution is the most general, but computationally cumbersome, distribution of the
3746 three models. It is applied when counting short-lived radionuclides with high efficiency.

3747 • Poisson distribution is a special case of the more general binomial distribution. It is applied
3748 when the counting time is short in comparison to the half-life. The Poisson distribution is a
3749 discrete distribution.

3750 • Gaussian distribution is the distribution applied when the number of decays during the count
3751 time is fairly substantial (> 20). The Gaussian distribution is a continuous distribution.

3752 These statistical models can be used to help understand, interpret, and make predictions concerning the
3753 outcome of radiation measurements. For example, if the outcome of a single measurement yields n
3754 counts, then by applying what is known about the distributions, it is possible to predict the results of
3755 subsequent measurements. This reproducibility is an indication of the precision of the measurement.
3756 A system that can be described by a Poisson (or Gaussian) distribution has a variance, equal to the mean,
3757 which is a measure of the dispersion of a distribution. Therefore, a measurement that yields a result of n
3758 counts has a variance of n and a standard deviation of $\sqrt{n}$. Hence, 68 percent of subsequent
3759 measurements under the same conditions will yield results that fall within the range $n+\sqrt{n}$ to $n-\sqrt{n}$.
3760 Another way of expressing the variability in the measurement in terms of the mean and the standard
3761 deviation is $n\pm k\sqrt{n}$ (counts).

3762 The parameter k is known as a coverage factor and the product $k\sqrt{n}$ defines a confidence interval.
3763 If $k = 1$, then 68 percent of the measurements will fall within an interval that is two standard deviations
3764 wide, centered about the mean. If $k = 2$, then 95.5 percent of the results will fall within an interval that is
3765 four standard deviations wide, centered about the mean. The typical or recommended coverage factor is
3766 $k = 1$ (ISO 1995), and the relative uncertainty is the ratio of the standard deviation to the mean.
3767 Figure A-1 shows the relative uncertainty as a function of the number of counts. The more counts, the
3768 smaller the relative uncertainty, and the greater the precision. For more information on the application of
3769 the statistical models to the analysis of decay counting, see ICRU Report 52 and NAS-NSS Report 3109.

The figure shows a log-log plot with "Relative Uncertainty (%)" on the y-axis (ranging from 0.1% to 100.0%) and "Total Counts" on the x-axis (ranging from 1.E+00 to 1.E+05). A diagonal line decreases from upper left to lower right. An annotation reads "(400 counts, 5% rel. uncertainty)".
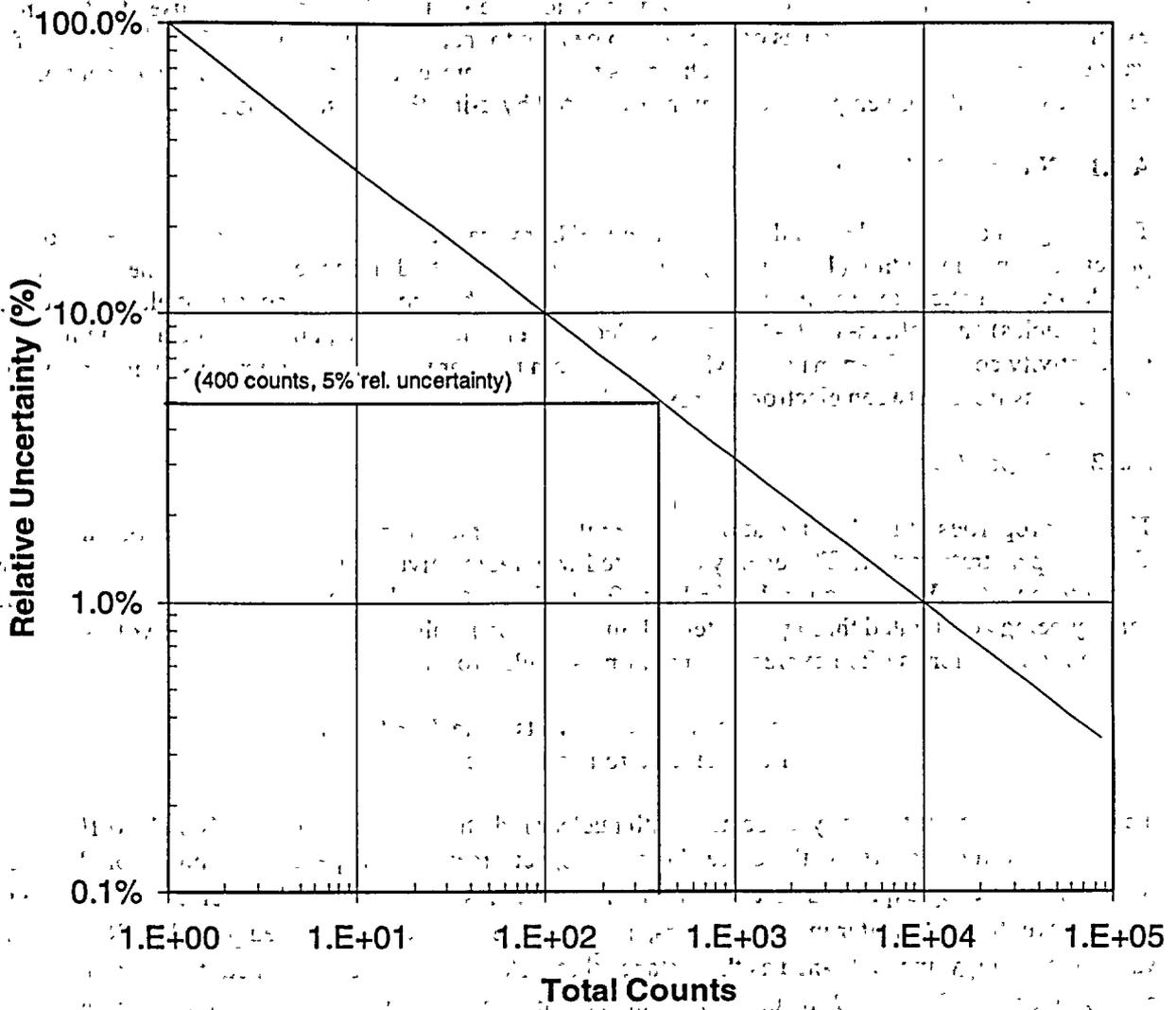
3770 Figure A-1: Relative uncertainty in counting as a function of the total counts for a Poisson process

A-3

3771 If there are requirements specifying a certain precision, the statistical models can be used to determine
3772 experimental parameters, such as count time, to be able to meet the requirements. The suitability of
3773 various instruments or measurement techniques to detect a prescribed or predetermined amount of
3774 radioactivity, with a given precision, can be evaluated by using the statistical models.

3775 **A.3.1 Nuclear Radiation**

3776 The energy and matter released during radioactive decay, called "nuclear radiation," assumes two
3777 principle forms, including (1) charged particles, which are emitted from the nucleus of the atom, and
3778 (2) electromagnetic radiation in the form of photons. The charged particles consist of electrons (called
3779 beta particles) and helium-4 (He-4) nuclei (called alpha particles). The photons associated with
3780 radioactivity consist of gamma rays, which result from nuclear transitions, and x-rays, which result from
3781 atomic transitions between electron energy levels.

3782 **A.3.2 Properties**

3783 The two properties of nuclear radiation that are relevant to radiation detection are its energy and its
3784 ability to penetrate matter. The energy associated with radioactivity is usually expressed in units known
3785 as electron volts (eV), defined as 1 eV $=1.6 \times 10^{-19}$ joules. This is the kinetic energy an electron would
3786 gain by being accelerated through a potential difference of 1 volt. Because the electron volt is a very
3787 small unit, radiation is often expressed in multiples of electron volts.

3788 $$1 \text{ thousand electron volts (1 keV)} = 10^3 \text{ eV}$$
3789 $$1 \text{ million electron volts (1 MeV)} = 10^6 \text{ eV}$$

3790 The energies that are typically associated with nuclear radiation range from about 10 keV to 10 MeV,
3791 and are generally measured with devices known as spectrometers. The penetration power of charged
3792 particles is typically expressed in terms of its range, which is not well-defined for electrons because they
3793 do not travel through matter in straight lines, as is the case with heavier charged particles. Range usually
3794 varies with energy and is defined as the distance that a charged particle will penetrate material before it
3795 ceases to ionize. Figure A-2 illustrates the range of alpha particles in air as a function of energy, while
3796 Figure A-3 shows the maximum range of beta particles as a function of energy for several different
3797 materials. As Figure A-2 illustrates, a 2-Mev alpha particle no longer produces ionizations in air after
3798 traveling only a centimeter distance. Note that the penetrating power of beta particles in metals is also
3799 limited; a 1-MeV beta particle in copper has a maximum range of less than a millimeter. An immediate
3800 consequence of these facts regarding the range of charged particles (alphas and betas) in matter is that
3801 alpha radiation can only be used to assay surficial contamination, while beta radiation can, to a limited
3802 extent, be utilized for volumetric contamination. Also, these two particles produce very different specific
3803 ionization. (The specific ionization is the number of ion pairs produced per unit path length by an
3804 ionizing particle; some detectors exploit this value to discriminate between alpha and beta particles.)
3805 A typical alpha particle traveling through air generates 10,000 to 70,000 ion pairs per centimeter, while
3806 a typical beta particle may produce only 60 to 7,000 ion pairs.
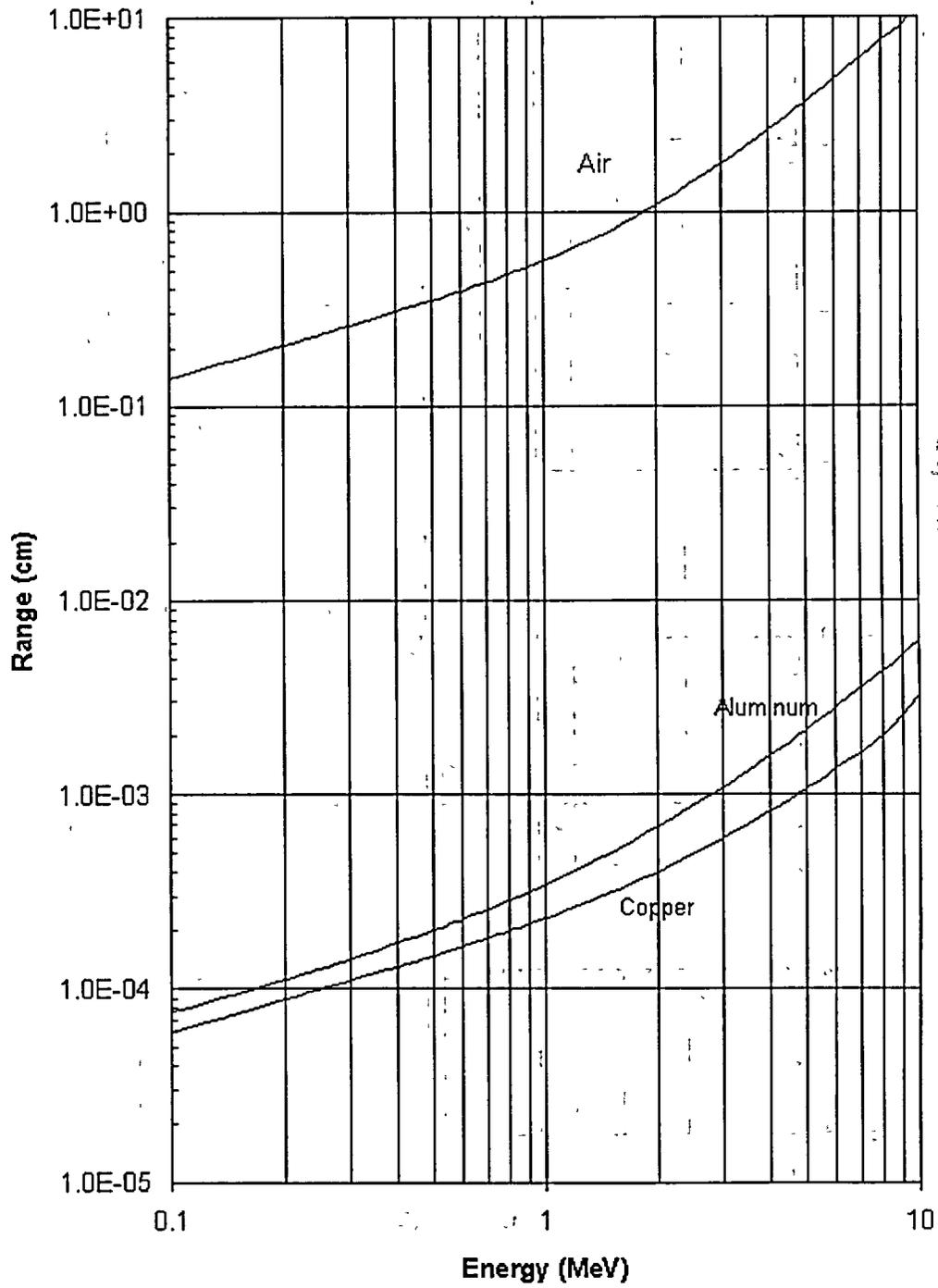
Figure A-2: Range of an alpha particle as a function of energy in several different materials (Data from ICRU Report 49)

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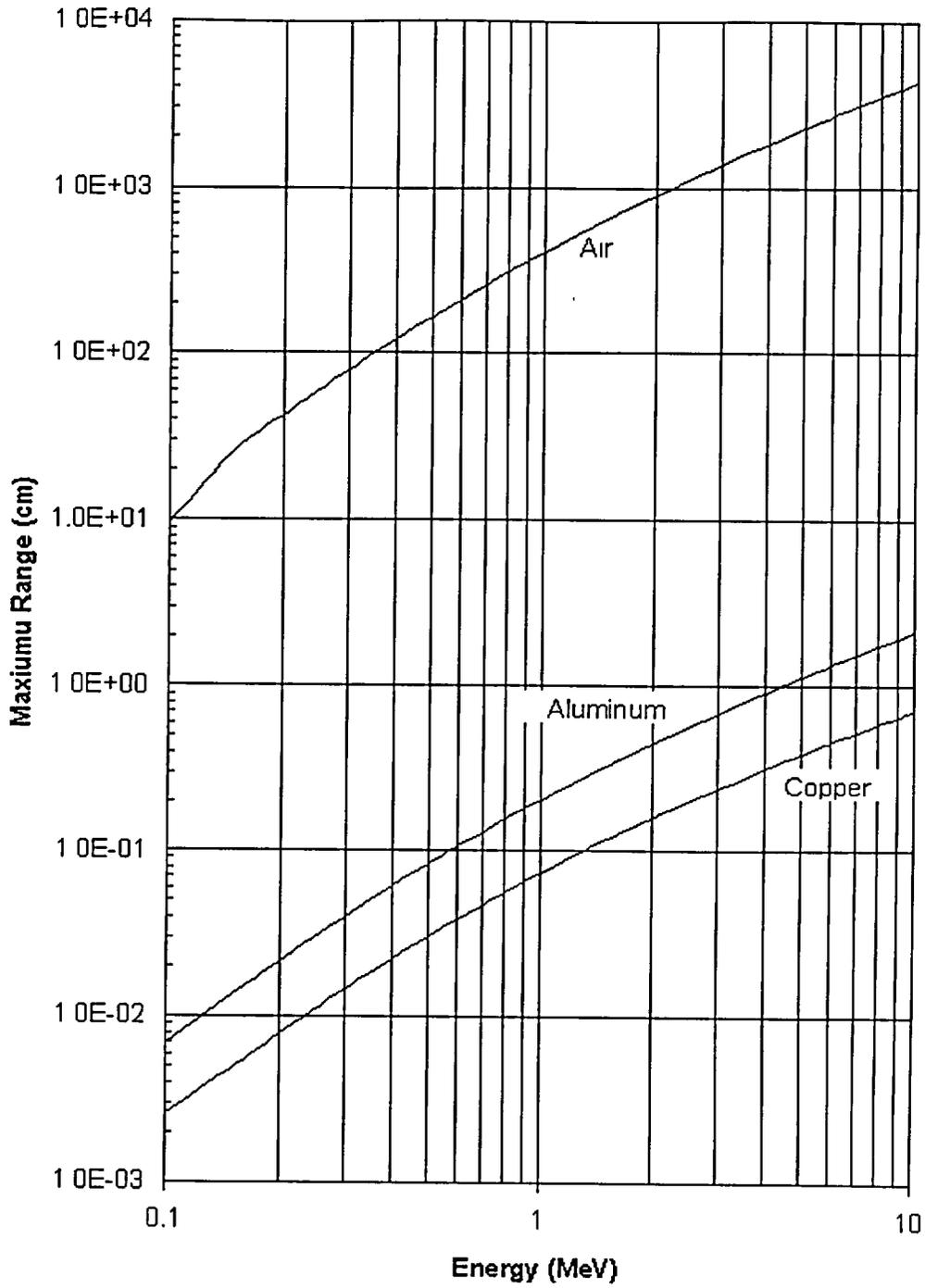
Figure A-3: Range of beta particle as a function of energy in several different materials
(Data from ICRU Report 37)

3811
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A-6

3813 The transport of gamma and x-rays through matter is quite different than for charged particles.
3814 The penetration power of gamma and x-rays in matter is typically expressed in terms of its half-value
3815 thickness (HVT), defined as the thickness of a material necessary to reduce the intensity of an x-ray or
3816 gamma ray beam to one-half of its original value. Figure A-4 is a plot of HVT as a function of energy for
3817 several materials. The HVT in this application can be thought of as an indication of the depth-of-view
3818 for volumetric contamination. Another significant feature of gamma radiation is that, unlike charged
3819 particles, photons can pass through matter without losing energy. The mean-free-path (MFP) is the
3820 average distance a photon can travel before having an interaction. Figure A-5 is a plot of the MFP as a
3821 function of photon energy for several materials. Note that a 1-MeV photon in copper can travel, on
3822 average, almost 2 centimeters without having an interaction. Germanium (Ge) is included in Figure A-5
3823 because it is a common detector material. Here again, a 1-MeV photon can travel, on average,
3824 3 centimeters without having an interaction.

3825 Another form of radiation that comes from the nucleus exists a uncharged particles, called neutrons,
3826 which behave quite differently from gamma rays and charged particles. As previously mentioned,
3827 radiation in the form of gamma rays and charged particles comes from nuclear decay. Neutrons, on the
3828 other hand, are generated by different processes, including the spontaneous fission of heavy elements
3829 such as uranium and plutonium. For most isotopes, the neutron emission rate is low compared to other
3830 forms of radiation.

3831 Table A-1 shows the spontaneous fission for a selected group of heavy elements, along with the
3832 corresponding alpha yield. (For the radionuclides listed in Table A-1, alpha particles are the primary
3833 source of radiation.) While the production of neutrons from the spontaneous fission yield of heavy
3834 elements is considerably less than the number of alpha particles generated from nuclear decay, neutrons
3835 do have a very significant detection advantage over alpha particles in that they can penetrate matter quite
3836 easily. Unlike charged particles, which have a range on the order of centimeters to meters depending on
3837 the type of radiation and the medium of interest (e.g., air, tissue), neutrons, like gamma rays, can have an
3838 indefinite range in matter. This makes neutrons attractive for the assay of volumetric contamination.
3839 Measurements of neutron fluence rates are widely used to assay transuranic waste. Despite this
3840 advantage, the use of neutrons for the assay of residual radioactivity is largely precluded because the
3841 yield is rather small and limited to a handful of heavy elements.
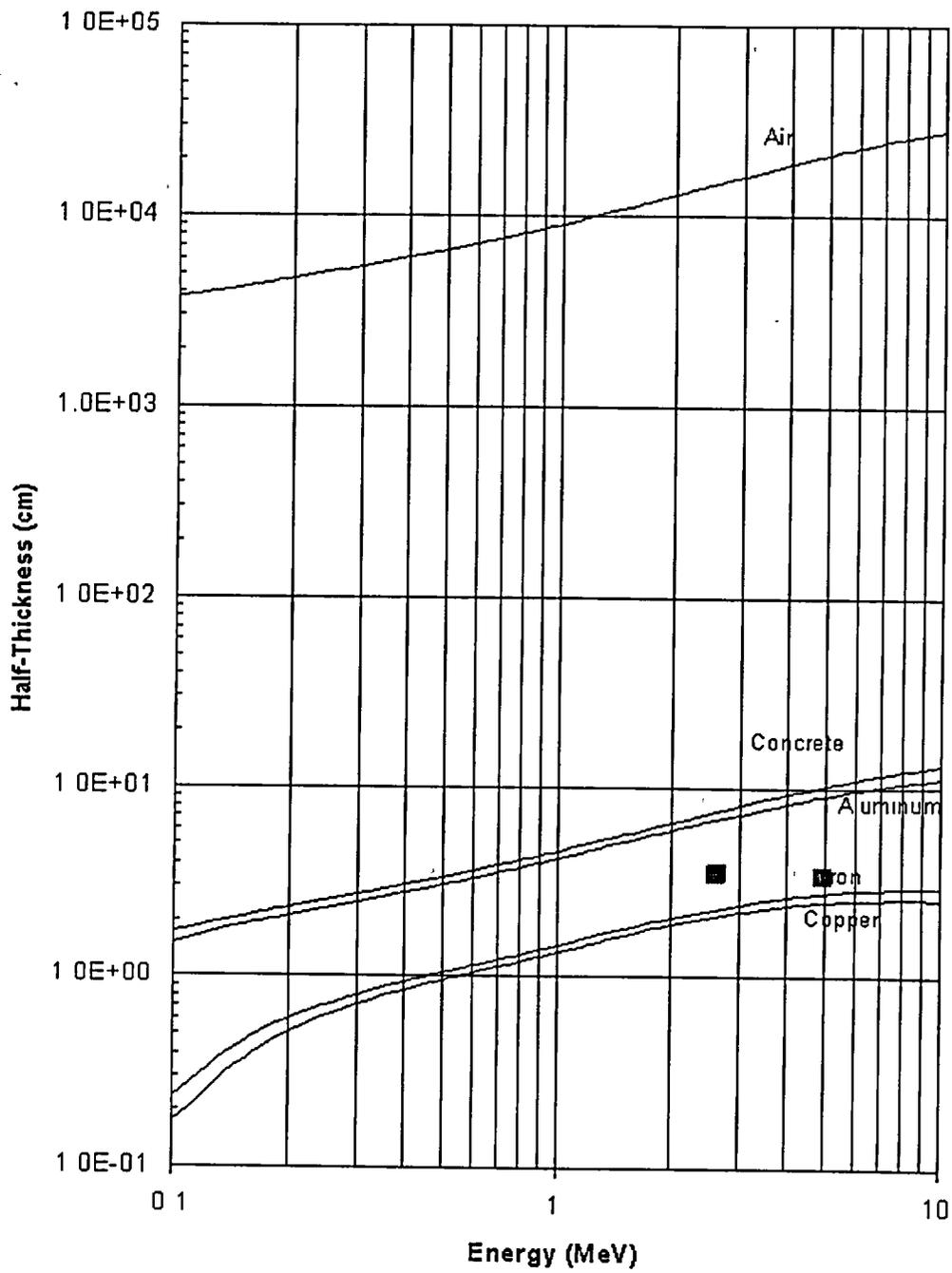
Figure A-4: The half-value thickness of gamma radiation
as a function of energy in several different materials (Hubble and Seltzer, 1995)
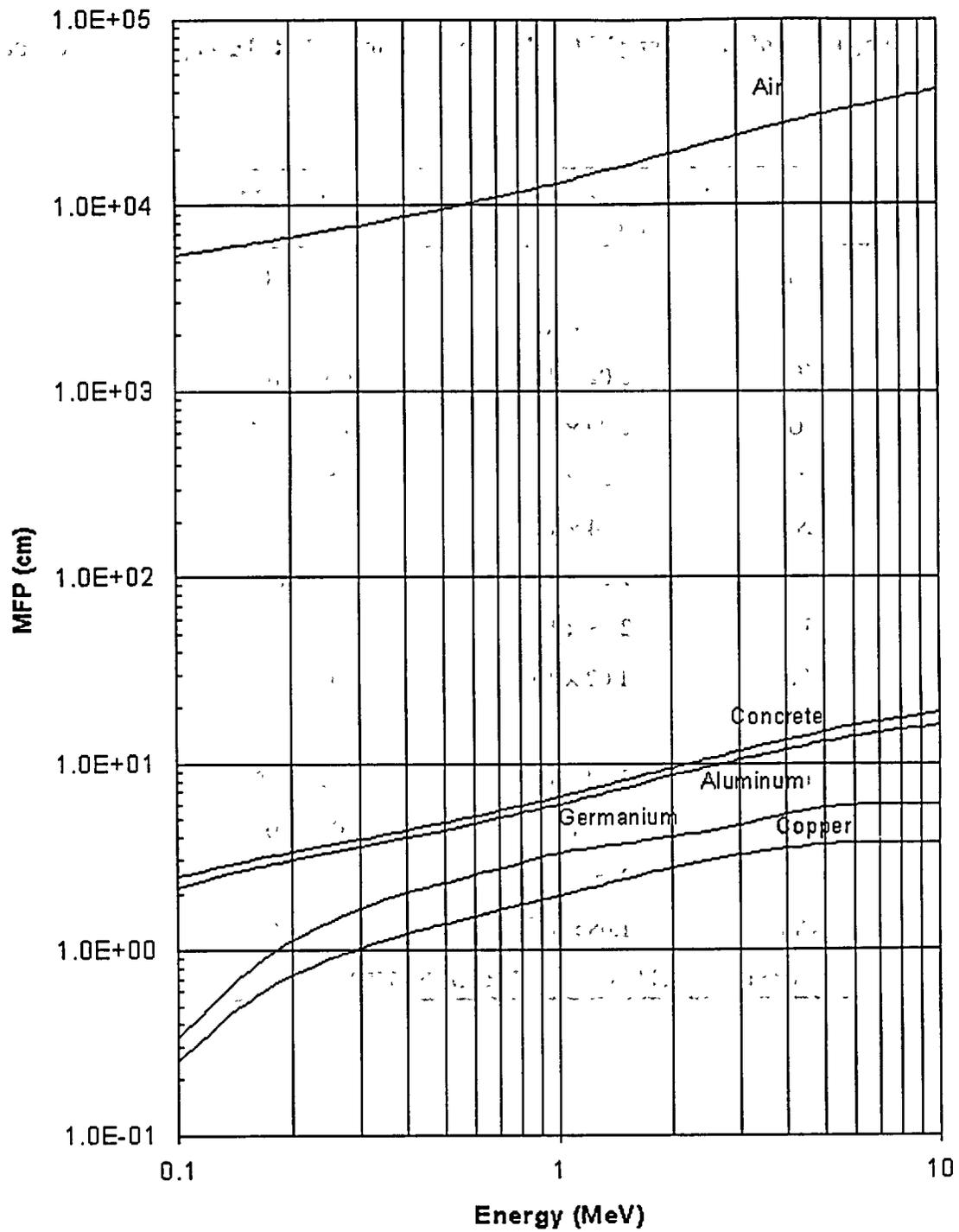
**Figure A-5: The mean-free-path of gamma radiation
as a function of energy in several different materials (Hubble and Seltzer, 1995)**

3844
3845

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.

3847 **A.4 Elements of Radiation Detection**

3848 Radiation detection is a broad field, which covers all types of radiation (e.g., x-ray, gamma-ray, alpha and
3849 beta particles, and neutrons) at levels ranging from background to extremely high levels associated with
3850 operational facilities (e.g., power and research reactors). The methods for detecting radiation are also
3851 quite diverse, ranging from calorimetry (measuring the decay heat) to event counting (counting the
3852 number of radiation interaction events). The purpose of this section is to introduce and discuss some of
3853 the concepts and quantities that are common to most radiation detectors.

3854 **A.4.1 Modes of Operation**

3855 Radiation detectors may be operated in two distinct modes:

3856 ● Current Mode: A radiation detector operated in current mode produces a current that is
3857 proportional to the event rate and the charge produced per event. An event is an interaction of a
3858 single particle (alpha, beta, or gamma ray) in which the particle transfers some or all of its energy
3859 within the sensitive region of the detector. Current mode operation is most often used in high-
3860 activity applications, such as ionization chambers.

3861 ● Pulse Mode: A radiation detector operated in pulse mode produces a pulse associated with
3862 individual events. In many instances, the pulse is proportional to the energy of the incident
3863 radiation. Detectors that utilize this energy proportionality feature are known as spectrometers.
3864 Other detectors, known as gross radiation counters, measure and count pulses regardless of
3865 energy.

3866 **A.4.2 Pulse Height Spectrum**

3867 When detectors that are operated in pulse mode are exposed to radiation, they produce a series of pulses
3868 that can be collected, sorted, and displayed. The result of such a process is a distribution of pulse
3869 heights, which is referred to as a pulse height spectrum. The pulse height can be related to the energy of
3870 the radiation, in which case, the spectrum is called an energy spectrum. The pulse height spectrum
3871 (or energy spectrum) is an important property of the detector output that is used to identify and quantify
3872 the radiation.

3873 **A.4.3 Energy Resolution**

3874 Two fundamental properties of a spectrometer are the precision with which it measures energy and its
3875 ability to distinguish between energies. Together, these properties are known as "energy resolution,"
3876 which is expressed in terms of the full width of a peak at half its maximum value (also referred to as the
3877 full width at half maximum, or FWHM). In some cases, it is expressed in keV; in other cases, it is
3878 expressed as a percentage of the radiation energy. Spectrometers are sometimes characterized as low-,
3879 medium-, or high-resolution detectors. The resolution is a result of statistical processes associated with
3880 the transfer and collection of the energy associated with the radiation. In general, the higher the
3881 resolution, the better — and more expensive — the detector. However, in applications where there is a
3882 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.

3914   <u>Minimum Detectible Concentration and Sensitivity</u>

3915   When discussing limits of detectability, the two expressions that are often used are minimum detectible
3916   concentration (MDC) and sensitivity. The term "minimum detectible concentration" implies a degree
3917   of statistical rigor and mathematical formality, while the term "sensitivity" is generally regarded as a
3918   colloquialism. Even though regulatory bodies, such as the NRC, require the rigor and formality of the
3919   MDC, this appendix uses the term "sensitivity" because it is consistent with the terminology of
3920   instrument manufacturers, and it avoids some of the persistent difficulties associated with the formal
3921   definition of MDC. For example, NUREG-1507 reviewed the literature on the statistical interpretation
3922   of MDC as part of a brief study addressing the consistency of MDC values for five MDC expressions.
3923   The various expressions led to a range of MDC values for a gas proportional counter. While the spread
3924   of MDC values was modest, it illustrates the fact that the MDC is not unique and depends upon the
3925   statistical treatment of the data. Others (MacLellan and Strom, 1999) argue that traditional MDC
3926   formulas (and decision levels) are wrong. In their view, these traditional formulas do not adequately
3927   account for the discrete nature of the Poisson distribution for paired blank measurements at low numbers
3928   of counts. Using the term "sensitivity" retains the concept that is embraced by the MDC, while avoiding
3929   some of the difficulties.

3930   <u>Factors Affecting Sensitivity</u>

3931   The sensitivity of any detection method or system depends on the individual processes and mechanisms
3932   that are particular to that method or system. In broad terms, any process that degrades or absorbs
3933   radiation energy adversely affects sensitivity. The sequence of events that lead to a signal from a
3934   detector begins with the decay of nuclei, or the de-excitation of electrons to produce radiation energy.
3935   The radiation energy must then reach the active or sensitive region of the detector, where it is converted
3936   to information carriers. Any loss of energy that occurs throughout this sequence results in a loss of
3937   sensitivity. Table A-2 addresses the primary energy and information loss mechanisms associated with
3938   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.

3970	Scattered radiation is radiation that has interacted with matter in such a way that its characteristic energy
3971	has changed. Scattered radiation can potentially interfere or obscure energy peaks. The continuum in a
3972	spectrum results from scattered radiation. Radiation can be scattered in the detector, in the source, or
3973	from materials surrounding the detector. While techniques have been developed to extract information
3974	from the continuum, it usually only obscures small peaks and, in some cases, renders the measurement
3975	useless.

3976	Figure A-6 shows the effect of resolution and interference on a gamma ray spectrum. The area under the
3977	peak is the same for all three cases; however, the peak in the bottom spectrum is all but lost to the
3978	continuum. At low radionuclide concentrations, the radiation emitted from most radionuclides competes
3979	with natural background radiation. Many laboratory systems have large and elaborate shields to limit the
3980	interference of natural background radiation. Techniques have been developed to reduce the contribution
3981	of scattered radiation. These techniques include anti-coincident shielding and coincidence counting,
3982	which make use of concurrent or coincident events in multiple or segmented detectors.

3983	Electronic noise is a form of interference that acts on the signal chain. Electronic circuits used to amplify
3984	and process pulses have two basic forms of noise: thermal and shot. Thermal noise refers to noise
3985	occurring in resistors in absence of current flow, while shot noise is associated with a flow of current.
3986	The technology used to process electronic signals is well developed and the instruments are well
3987	designed. Therefore, electronic noise is not typically a limiting factor for detector sensitivity. Rather,
3988	most of the problems with interference come from external sources.
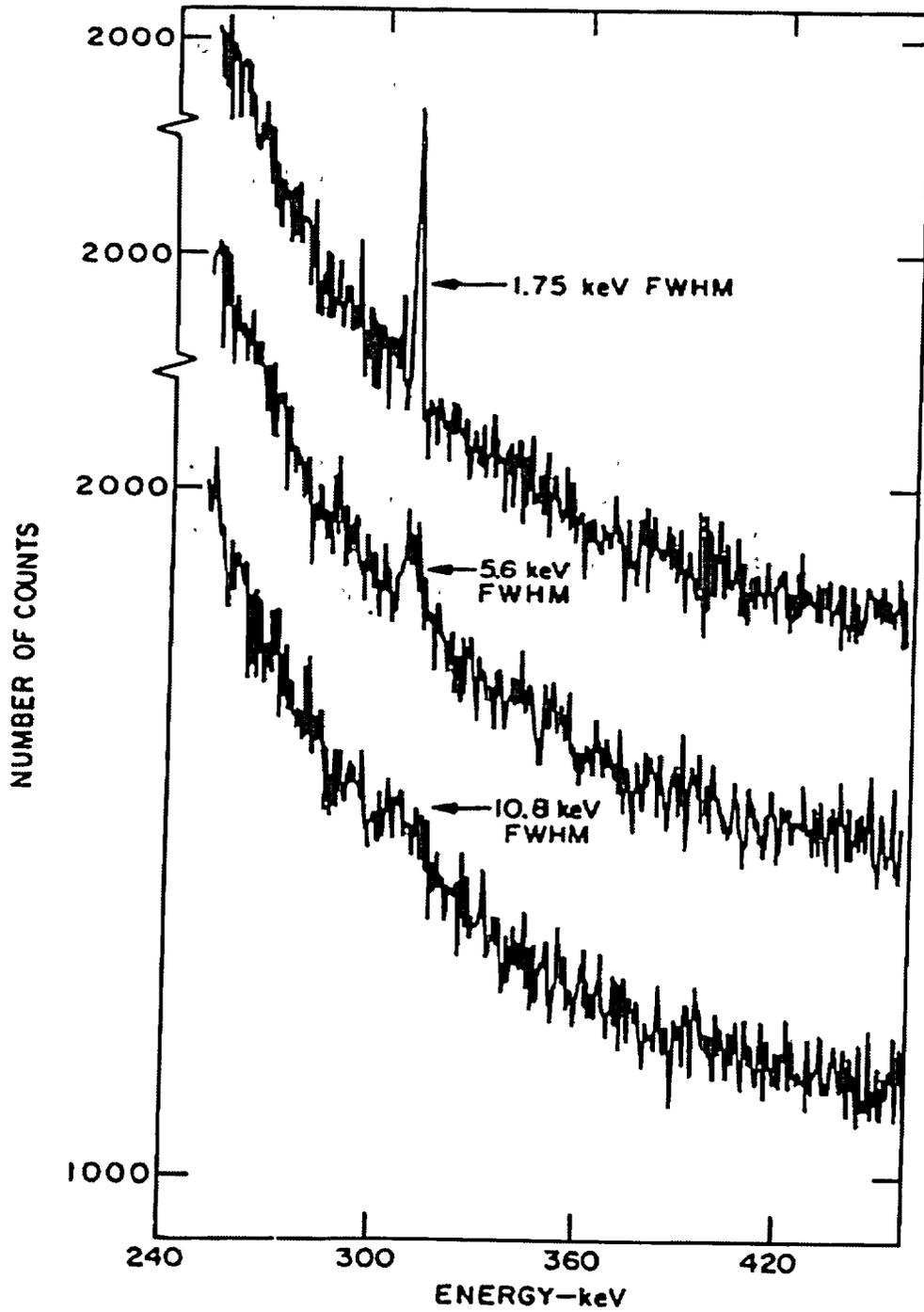
Figure A-6: The effects of interference from scattered radiation
on the ability to detect a peak for several measured energy resolutions (Knoll, 2000)

3992 <u>Sensitivity and Energy Resolution</u>

3993 When spectroscopy is used to measure activity, the sensitivity is affected by the energy resolution.
3994 The issue of energy resolution and its impact on sensitivity is essentially the issue of background
3995 reduction. Recall that the sensitivity represents the minimum amount of activity that produces a
3996 response, in counts, that is statistically significant from background. If the detector has no energy
3997 resolution, any particle that enters the detector's active volume will produce counts. This, in turn, will
3998 increase the amount of activity that must be present in order to establish a response that is statistically
3999 significant from background. Because the decay of a radionuclide often emits radiation with a very
4000 specific energy (e.g., alpha decay), spectroscopy can be used to restrict the response to an energy range
4001 that corresponds to the decay of the radionuclide in question. The better the energy resolution, the
4002 greater the selectivity in the number of counts and the greater the sensitivity. In this way, spectroscopy is
4003 a form of background reduction.

4004 <u>Factors Affecting Energy Resolution</u>

4005 The number of information carriers affects the resolution. That is, the more information carriers that are
4006 produced in the detector's active volume, the greater the energy resolution. This is a result of the
4007 statistical fluctuation in the number of information carriers. Under the assumption of a Poisson process,
4008 the variance in the number of information carriers is equal to the number of information carriers.
4009 Assuming Poisson statistics, the energy resolution, measured in terms of the FWHM, becomes

4010 $$FWHM = 2.35/\sqrt{N}$$

4011 where N is the number of information carriers. Hence, the greater the number of information carriers,
4012 the better the energy resolution. However, measurements of the energy resolution of some types of
4013 radiation detectors have shown that the achievable values for FWHM can be lower than the value
4014 predicted by the above equation. These results indicate that simple Poisson statistics do not describe the
4015 processes that give rise to the formation of each individual charge. The Fano factor has been introduced
4016 in an attempt to quantify the departure of the observed statistical fluctuations in the number of charge
4017 carriers from pure Poisson statistics. The Fano factor is the ratio of the observed variance to the variance
4018 predicted by Poisson statistics. Hence, the smaller the Fano factor, the better the resolution. Fano factors
4019 for semiconductor devices and proportional counters are much less than unity, whereas scintillation
4020 counters have a Fano factor of about unity.

4021 When radiation energy is absorbed in a detector, it must be converted into a form from which information
4022 can be extracted. The term "information carrier" is used to denote, in a general way, the particles that
4023 participate in the conveyance of information. For most detectors, the particles consist of ions, electrons,
4024 and electron-hole pairs. The effectiveness of a detector in terms of producing information carriers relates
4025 to the energy that is lost as a result of their creation. The higher the loss in energy, the less information
4026 that can be extracted. Ultimately, these information carriers deliver their information in the form of a
4027 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)
4067	^{3}H	none	12.28	β (5.69)[b]	none	swipes + liquid scintillation counter
4068	^{14}C	none	5730	β (49.5)[b]	none	thin-window G-M detectors/ GP detectors[f]
4069	^{54}Mn	none	0.85	γ (834.8)	-[d]	gamma or x-ray survey meter
4070	^{55}Fe	none	2.7	x-ray (5.89)	^{60}Co	gamma or x-ray survey meter
4071	^{60}Co	none	5.27	γ (1332)	-[d]	gamma or x-ray survey meter
4072	^{63}Ni	none	100	β (17.1)[b]	^{60}Co	thin-window G-M detectors/ GP detectors[f]
4073	^{90}Sr	decays in ^{90}Y	28.6	β (196)[b]	^{137}Cs	thin-window G-M detectors/ GP[f] detectors
4074	^{99}Tc	none	213000	β (84.6)[b]	^{137}Cs[e]	thin-window G-M detectors/ GP detectors[f]
4075	^{134}Cs	none	2.06	γ (605)	-[d]	gamma or x-ray survey meter
4076	^{137}Cs	decays in Ba-137m	30	β [c]/γ (662)	Ba-137m	gamma or x-ray survey meter
4077	^{232}Th	Th series (parent)	long[a]	α (4010)	^{228}Ac, ^{208}Tl	ZnS/ GP detectors[g h]
4078	^{234}U	U series (progeny)	244500	α (4773)	none	ZnS/ GP detectors[g h] scintillators
4079	^{235}U	Ac series (progeny)	long[a]	α (4389)	-[d]	ZnS/ GP detectors[g h]

4080

	Radionuclide	Series/decay chain	Half-life (y)	Primary radiation (keV)	Potential surrogate	Standard method of detection (survey)
4081						
4082	^{238}U	U series (parent)	long[a]	α (4198)	^{234}Th, ^{234m}Pa	ZnS/ GP detectors[g,h]
4083	^{226}Ra	U series (progeny)	1640	α (4602)	Bi-214, Pb-214	ZnS/ GP detectors[g,h]
4084	^{238}Pu		87.7	α (5499)	none[i]	ZnS/ GP detectors[g,h]
4085	^{239}Pu		24065	α (5156)	none[i]	ZnS/ GP detectors[g,h]
4086	^{240}Pu		6537	α (5168)	none[i]	ZnS/ GP detectors[g,h]

4087 [a] half-life > 10^7 y
4088 [b] average β energy
4089 [c] not used - equilibrium with progeny Ba-137m
4090 [d] not necessary, emits γ
4091 [e] speculative
4092 [f] gas proportional counter operated in $\alpha+\beta$ mode
4093 [g] ZnS - Zinc Sulfide Scintillator
4094 [h] gas proportional counter operated in α mode
4095 [i] does emit gammas of low intensity (<0.1%)

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4147 **Appendix B: Advanced/Specialized Instrumentation**

4148 **B.1   Conventional Radiation Detectors**

4149   This appendix provides information on a wide range of radiation detectors and detection methods.
4150 Beginning with conventional radiation detectors, it profiles various detection systems as they relate to
4151 clearance surveys. While this appendix addresses many examples of commercially available systems, it
4152 could not be, and is not intended to be, exhaustive. It does, however, provide a snapshot of systems that
4153 could have an impact on clearance surveys, and it discusses emerging and advanced radiation detectors
4154 and software programs. While these systems are expected to have an impact on the field of radiation
4155 detection, their impact on clearance surveys is uncertain.

4156   The majority of instruments described in this appendix use one of the following types of radiation
4157 detectors:

4158   ▪   Gas-filled proportional counters and Geiger-Mueller (GM) tubes. Gas proportional detectors
4159   come in two basic types: sealed systems and gas flow proportional systems.

4160   ▪   Scintillation detectors may be either inorganic (e.g., Zinc Sulfide and Sodium Iodide) or organic
4161   (e.g., plastic).

4162   ▪   Solid-state semiconductors include high-purity germanium (HPGe) and cadmium zinc telluride
4163   (CZT).

4164 While a complete discussion of these detectors is beyond the scope of this appendix, the following table
4165 summarizes the properties and features of these detectors. A more comprehensive treatment of these
4166 detectors may be found in Knoll (2000).

4167   **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

4168
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Table B-1: Properties of some common detectors (continued)

	Detector Type		Comments
4174			
4175	**Scintillator**		
4176	ZnS(Ag)	■	limited to thin screens or films
		■	used to detect alpha radiation
4177	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
4178	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.

4194	**Alpha/Beta**

4195   While gas flow proportional detectors can detect alpha and beta radiation, they can distinguish between
4196   the two by adjusting the operating voltage. The active volume of the detector is filled with P-10 gas.
4197   Radiation enters the active volume through an aluminized Mylar window. Typical probe areas cover
4198   about 100 cm^2.

4199   **Beta/Gamma**

4200   Geiger-Mueller detectors or "pancake" detectors are used to detect beta and gamma radiation.
4201   The detector tube is filled with an inert gas, which is a mixture of argon, helium, neon, and a halogen-
4202   quenching gas. Radiation enters this tube through a mica window. A typical probe area covers about
4203   20 cm^2.

4204   **Gamma**

4205   Thallium-activated sodium iodide (NaI(Tl)) scintillation detectors are used to measure gamma radiation.
4206   Since gamma radiation is much more penetrating than alpha and beta radiation, the type of detector
4207   window is not crucial, but these instruments typically use aluminum. The cylindrical crystals range in
4208   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
4209   on a gross count rate mode. However, recent developments in microchips and spectrum analysis
4210   software for NaI(Tl) detectors provide for greater flexibility and expanded use, while still retaining its
4211   portability. These new systems are discussed in the next section.

4212   **B.3   Specialized Instrumentation**

4213   Along with the conventional radiation detection instrumentation, there is a substantial assortment of
4214   instruments that have both generic and specialized uses. While this section addresses many examples of
4215   the commercially available radiation detection systems that are relevant to clearance, it is not intended to
4216   be exhaustive. In addition, it must be noted that the following discussion should not be construed as an
4217   endorsement of any of these products by the U.S. Nuclear Regulatory Commission (NRC).

4218   When available and appropriate, this section provides capital cost information, using the following four
4219   indicators to signify four capital cost ranges; when appropriate and available, estimated measurement
4220   costs may also be provided.

4221 $ -   less than $1k

4222 $$ -   greater than $1k, but less than $10k

4223 $$$ -   greater than $10k, but less than $100k

4224 $$$$-   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.

4264	Electret ion chambers have traditionally been deployed to measure radon concentrations in the air of
4265	homes and businesses. The literature also discusses other applications of EICs, such as measuring alpha
4266	and low-energy beta surface contamination, measuring alpha soil concentration, quantifying alpha
4267	contamination inside piping, and performing gamma dose measurements. EICs can be used for
4268	inexpensive alpha measurements and/or for areas where conventional alpha probes cannot measure.
4269	While the deployment time can be long, the measurement time is very short and sensitivities are much
4270	better compared to traditional detectors such as a gas-proportional counter. Also, EICs can be used in
4271	difficult-to-measure situations, such as tritium contamination or alpha contamination inside piping.
4272	The EICs measure gross alpha, gross beta, gross gamma, or gross radon.

4273	An example of a commercially available EIC is Rad Elec Inc.'s E-PERM alpha radiation monitoring
4274	systems. These systems are available in sizes ranging from 50 to 180 cm^2 and in various electret
4275	thicknesses depending on the required sensitivity.

| 4276 | Capital Cost: $$ |

Alpha Surface Measurements

4278	Oak Ridge National Laboratory (ORNL) has developed a procedure, known as Method RA010, using
4279	Rad Elec's E-PERM alpha radiation monitors for use in decontamination and decommissioning (D&D)
4280	operations (Meyer *et al.*, 1994). Costs for deploying the E-PERM system were reported to be $5 per
4281	measurement for a large-scale survey.

4282	Levinskas *et al.* studied low-level alpha measurements using a 145-ml EIC with a deployment time of
4283	48 hours. They reported that the results were within 5-percent accuracy, compared to NIST-traceable
4284	calibrated gas flow proportional counters. Sensitivity for this measurement method was reported to be
4285	$(1.1 \pm 0.5) \times 10^{-3}$ Bq/cm^2 $(6.4 \pm 3.0$ dpm/100 cm$^2)$ at the 95-percent confidence level.

Alpha Soil Measurements

4287	Meyer *et al.*, 1995, described a method for taking *in situ* measurements of alpha contamination in soils
4288	using EICs. Probe sizes of 50 and 180 cm^2 are used. With a 50-cm^2 EIC, detection limits of 1 Bq/g
4289	(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
4290	deployment times of 6, 12, 24, and 48 hours, respectively. Survey costs ranged from $8 to $25 per
4291	measurement.

Alpha Contaminated Pipes

4293	Direct measurement of alpha contamination inside pipes is difficult because of the short range of
4294	alpha particles. However, measurements of the ionization caused by the alpha radiation in air can be
4295	used to infer alpha contamination. An EIC is placed at the end of the pipe and air is directed through
4296	the pipe to the EIC. The collection of the secondary ions reduces the charge of the electret. Calibration
4297	is performed by locating an alpha source of known strength and determining response factors.
4298	In a 15-minute measurement, uniform alpha contamination in a pipe with a 15-cm diameter can detect
4299	an activity of 0.04 Bq/cm^2 (2.2 dpm/cm^2) (Dua *et al.*, 1997).

4298	**Beta Surface Measurements**

4299 Sensitivities for tritium measurements are reported to be 1 Bq/cm^2 (6,000 dpm/100 cm^2) with a
4300 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
4301 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.

4302 **Gamma Measurements**

4303 The response of this type of detector to gamma radiation is nearly independent for energies ranging from
4304 15 to 1,200 keV. A 30-day deployment with 50-ml chamber is required to quantify an ambient field of
4305 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.
4306 The smaller chamber is generally used for long-term monitoring.

4307 **Portable Gamma-Ray Spectrometers**

4308 There are a wide variety of handheld spectrometers available on the market. They consist of two general
4309 types, including integrated systems and modular systems. The integrated systems have the detector and
4310 electronics contained in a single package. The modular systems separate the detector from the
4311 electronics. These spectrometers employ small scintillators, typically NaI(Tl), and room temperature
4312 solid semiconductors such as CZT. Recently, the systems using NaI(Tl) scintillators utilize special
4313 analysis software to do isotope identification. These systems represent an advancement over the
4314 conventional scintillation probes connected to rate meters. The systems using CZT have superior
4315 resolution (compared to scintillators) and, therefore, perform the standard peak analysis. The preferred
4316 application for the devices tends to be in nuclear non-proliferation, where isotope identification is more
4317 important than sensitivity.

4318 Three systems of note include SAM-935 from Berkeley Nucleonic Corporation, RADSMART from
4319 SAIC, and the GR-130 miniSPEC from Exploranium. All of these systems are handheld and do some
4320 form of isotope identification. The SAM-935 uses an NaI(Tl) scintillator and a spectrum analysis
4321 technique called Quadratic Compression Conversion™ to perform rapid isotope identification.
4322 The RADSMART uses a proprietary CsI scintillator coupled to a photodiode. The isotope identification
4323 is performed using spectrum templates rather than peak analysis, which is often problematic for low-to-
4324 medium resolution spectrometers such as CsI. The GR-130 miniSPEC also uses an NaI(Tl) scintillator,
4325 but performs a peak analysis on the spectrum for isotope identification. These systems are no more
4326 sensitive to radiation than the conventional instruments (e.g., small scintillators operated in a gross count
4327 mode), but they can provide information on radionuclide identity. These systems are rather new and
4328 there is little or no data available to support claims that the spectrum analysis programs can significantly
4329 improve the sensitivity.

4330 Capital Cost: $$$

4331 **X-ray Fluorescence**

4332 X-ray fluorescence (XRF) is a spectroscopic method in which secondary x-ray emission is generated by
4333 the excitation of a sample with x-rays. The x-rays eject inner-shell electrons, then outer-shell electrons
4334 take their place and emit photons in the process. The wavelength of the photons depends on the energy
4335 difference between the outer-shell and inner-shell electron orbitals. The amount of x-ray fluorescence is
4336 sample-dependent, and quantitative analysis requires calibration with standards that are similar to the
4337 sample matrix. The nature of the method does not allow for isotope identification (but rather the element
4338 itself) and is generally not useful for measuring the fluorescence yield in elements with atomic numbers
4339 less than 32.

4340 Recently, field-portable x-ray fluorescence (FPXRF) systems have been developed that are available
4341 commercially. These systems use sealed sources to produce fluorescent x-rays and contain a small x-ray
4342 spectrometer to measure the fluorescent x-rays. The advantage of this technology includes the ability to
4343 measure solids, liquids, thin films, and powders. FPXRF is a useful technique for screening or surveying
4344 materials for their elemental content when portability, short analysis times, and real-time results are
4345 required. For information concerning the performance of FPXRF, see Potts (1999) and U.S. DOE
4346 (1998a).

4347 An FPXRF, known as the Spectrace 9000, is commercially available from Thermo NORAN's
4348 KevexSpectrace. This device uses iron-55 (^{55}Fe), cadmium-109 (Cd-109), and americium-241 (^{241}Am) to
4349 produce a wide range of excitations, capable of exciting atoms of atomic number 16 (sulfur) to 92 .
4350 (uranium). T his particular unit can simultaneously measure 25 elements. The detector uses a mercuric
4351 iodide semiconductor to measure the fluorescent x-rays. The Spectrace 9000 can operate on battery or
4352 110-Vac power. Measurements can be made on a surface, or small samples can be taken and placed in a
4353 small counting chamber attached to the probe.

4354 Capital Cost: $$$

4355 **Compton Suppression Spectrometer**

4356 Background reduction is critical to maximizing detector sensitivity. Typical methods for background
4357 reduction include lead shields and anti-Compton shields made of NaI(Tl) (or bismuth germanate[8]).
4358 Princeton Gamma Tech (PGT) has developed a Compton Suppression Spectrometer (CSS) based on
4359 the Duode detector, which is a transversely segmented single crystal of high-purity germanium.
4360 PGT developed the crystal processing techniques specifically to improve detector performance at low
4361 energies without sacrificing the efficiency of a large HPGe detector. Suppression is achieved by
4362 detection and electronic vetoing of coincident energy deposition events in the rearmost segment of the
4363 crystal. At low energies, most of these coincident events are from background photons, which have
4364 undergone forward Compton scattering from the front "planar" segment. The suppression provided by
4365 this geometry is ideal for rejecting these background events.

4366 In general, the Duode suppression provides significant background reduction across the energy range and
4367 improvement in the signal-to-noise ratio (SNR) and, thus, reduced peak fitting errors in a limited energy
4368 range. For a strong peak, a reduction in background has little effect on the SNR or peak-fitting error.
4369 For a weaker peak, such as 2–3 standard deviations (σ) above background or lower, the improvement in
4370 the SNR and reduced peak fitting error can be significant. The principal benefit of the Duode is for
4371 measurement of those isotopes which would normally be lost in the background (Haskins et al., 2000).

4372 Capital Cost: $$$

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[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.

4375 **B.3.2 Application-Specific Detection Systems**

4376 Responding to the measurement needs of nuclear facilities engaged in D&D activities, instrument
4377 manufacturers have developed specialized detection systems and, in a few instances, services that are
4378 designed to facilitate and expedite radiation measurements associated with the D&D effort. Many of
4379 these systems use traditional detectors (gas proportional counters, plastic scintillators, and NaI(Tl)
4380 scintillators) coupled to rate meters. The design goal of these systems is to optimize throughput while
4381 detecting contamination at guideline levels[9]. These goals have been more-or-less accomplished by using
4382 large shielded detectors and arranging them in a manner to optimize the geometrical efficiency.
4383 Shielding the detectors helps to improve the SNR by reducing the background. This section briefly
4384 addresses the following systems and/or applications:

4385 ■   conveyorized survey monitors

4386 ■   floor and surface contamination monitors

4387 ■   *in situ* gamma-ray spectrometry systems

4388 ■   *in toto* monitors

4389 ■   pipes (interior/exterior)

4390 ■   subsurface

4391 ■   portal monitors

4392 This section does not address systems that have been developed specifically for the assay of transuranic
4393 waste. Some of the systems are quite sophisticated and use active measurement techniques, as discussed
4394 in Section B.4.

4395 <u>**Conveyorized Survey Monitors**</u>

4396 Conveyorized survey monitors (CSMs) automate the scanning or hand-frisking of materials. Current
4397 systems have been designed to measure materials such as clothing (laundry monitors), copper chop
4398 (small pieces of copper), concrete rubble, and soil. A typical CSM consists of a conveyor belt that passes
4399 under or between an array of detectors. Most systems use an array of gas flow proportional counters in a
4400 staggered configuration. The staggered configuration eliminates blind spots (locations where
4401 contamination may be present but cannot be detected because the radiation cannot reach the detectors).
4402 Systems range from small monitors with small belts to large trailer-mounted systems for measuring and
4403 segregating (in terms of activity) rubble, debris, and soil.

4404 **Commercial Systems**

4405 Eberline manufactures several conveyor systems. Model ACM-10 is an automated contamination
4406 monitor utilizing a single conveyor belt. Radiation measurements are performed with an array of
4407 10 large-area (503-cm^2) gas proportional detectors that are located above and below the belt. Model
4408 140A is a larger version of the ACM-10, which utilizes two conveyor belts to compress the material
4409 being measured (typically clothes). This model uses an array of gas flow proportional counters, 14 above
4410 and 14 below. Ludlum manufactures a laundry monitor (Model 329-32) that also utilizes a single
4411 conveyor belt. It uses two arrays of sixteen 100-cm^2 gas proportional detectors each.

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

4412 BNFL markets a CSM that is intended for rubble, debris (e.g., concrete and steel), and soil. This high-
4413 throughput system (~ 1.5 × 10⁴ kg/h) uses a modular detection approach, which means that it has
4414 individual detector modules to measure specific radiation types. For example, the system has a gross
4415 gamma detection module, an alpha/beta surface detection module, a low-resolution gamma spectrometry
4416 module, and a high-resolution gamma spectrometry module. Multiple modules can be linked together
4417 when data from different radiation types are needed. Canberra Industries also markets a CSM for rubble,
4418 debris, and soil. This trailer-mounted system is also a high-throughput system; Canberra reports a
4419 throughput up to 4.5 × 10⁴ kg/h (50 tons/h). The system uses shielded HPGe detectors to perform
4420 spectroscopy on the material. However, for specific situations that do not require the high resolution,
4421 offered by the germanium detectors, large NaI(Tl) detectors can be utilized. An available diverter
4422 mechanism can be used to automatically segregate materials in terms of activity.

4423 A similar system, called the Segmented Gate System (SGS), is available as a service from Eberline
4424 Services. The SGS is primarily a soil characterization and sorting system, which has been in use for a
4425 number of years and has processed more than 176,000 m³ of soil. The system consists of a combination
4426 of conveyor systems, radiation detectors, and computer controls that remove contaminated soil from a
4427 moving feed supply on a conveyor belt. The system uses two sets of gamma radiation detector arrays
4428 housed in shielded enclosures. The two sets of detectors allow for the radiation measurement of two
4429 gamma energy regions of interest. The thin detector array uses 0.160-cm thick NaI(Tl) detectors and
4430 incorporates a 1.9-cm thick lead shield that is fully encased in steel. T he thick detector array uses 5-cm
4431 thick NaI(Tl) detectors and is housed in a similar shield. Eberline Services reports a throughput of
4432 approximately 3.4 × 10⁴ kg/hr (38 tons/hr). While the majority of applications have measured gamma
4433 radiation from radionuclides such as cesium-137 (^{137}Cs), cobalt-60 (^{60}Co), and americium-241 (^{241}Am),
4434 the SGS has been equipped with beta detectors to assay strontium/yttrium-90 (^{90}Sr(^{90}Y)).

4435 **Large-Area Surface Contamination Monitors**

4436 Conventional survey instruments, such as those described previously (e.g., gas proportion counters,
4437 GM tubes, and ZnS scintillators), are very efficient at measuring surface contamination on small items.
4438 However, with a relatively small active area (100 cm² for a gas proportional counter, 20 cm² for G-M
4439 pancake probes and 75 cm² for some ZnS scintillators), these devices are rather inefficient at scanning
4440 large objects such as walls and floors. This section addresses the natural extension of these devices for
4441 the measurement of contamination on large areas. These large-area surface contamination monitors have
4442 active areas that exceed 1,000 cm² and are ideally suited for scanning large, flat areas such as walls,
4443 floors, and soil. The simplest systems mount conventional survey instruments, such as gas proportional
4444 counters with rate meters, on a mobile platform. More sophisticated systems utilize position sensitive
4445 gas proportional counters and/or fiberoptic sensors, and can perform data logging and mapping.

4446 **Commercial Systems**

4447 Several companies market systems that detect contamination on floors. The Ludlum Model 239-1F floor
4448 monitor represents one of the simplest systems available. This modular system features a 16 cm x 47 cm
4449 gas flow proportional counter that can be mated to any one of three survey meters, one of which is a data
4450 logger. The single-handled, two-wheeled cart can accommodate the rate meter and a Matheson size 2 or
4451 Linde Q bottle for the counting gas. The FM-300 floor monitor series, manufactured by Aptec-NRC, is
4452 also a modular floor monitor system. The basic unit features two large, sealed proportional counters.
4453 The detectors have an active area of 504 cm² and a sensitivity of 42–83 Bq (2,500–5,000 dpm) for ^{60}Co in
4454 normal background. The model FM-302 system includes the battery powered omniTrack rate meter.
4455 While the omniTrack rate meter does not currently do data logging, the system is being modified to
4456 support this feature.

4457   Thermo Eberline makes the FCM-4, which is an integrated system that uses four 15.2 cm x 20.3 cm
4458   ZnS(Ag) scintillators. The system, which comes with a computer to allow data logging, is similar to the
4459   Aptec-NRC system in terms of its form; it has four wheels and a tubular handle. Thermo Eberline
4460   reports a sensitivity of 8.3 Bq (500 dpm) alpha and 33 Bq (2,000 dpm) beta from ^{137}Cs.

4461   Shonka Research Associates Inc. produces the Surface Contamination Monitor and Survey Information
4462   Management System (SCM/SIMS). This sophisticated system features a position-sensitive gas
4463   proportional counter mounted on a motor-driven cart. The position-sensitive gas proportional counter
4464   uses a multi-wire electrode configuration to detect the position of the activity within the active volume.
4465   The width of the proportional counter used with the SCM/SIMS is variable, typically from 0.5 to 5 m.
4466   Also, the system can be equipped with a variety of sensors to facilitate the detection of both beta/gamma
4467   and alpha radiation fields.

4468   The SIMS part of the system includes a video camera and a series of software programs that processes
4469   and analyzes the collected survey strip data. The SIMS records both the intensity and location of the
4470   radioactivity in an electronic database and mapping software. STITCHER© is a program that takes the
4471   individual survey strips and positions them relative to each other and the survey area. Once the strips are
4472   positioned, the VISUSPECT program projects and averages the data from the strips onto standard
4473   100-cm² areas typical of manual surveys. The data from this array can then be visually inspected using
4474   various image-processing algorithms, or it can be used to generate a data report that documents the
4475   average contamination present in each 1-m² area and the maximum contamination level in a given 100
4476   cm² within this 1-m² area. Note that 100 cm² is the active area of most hand-held probes that would be
4477   used for scanning applications. More information on the SCM/SIMS and its detection principles can be
4478   found in papers and reports by Shonka (1992, 1995, 1996a, and 1996b) and U.S. DOE (1998b).
4479   The SCM/SIMS is not for sale. It is included as a service that is provided by Millennium Service.

4480   BetaScint Inc. has designed a detector that uses a fiberoptic sensor to determine the concentration of
4481 ^{90}Sr or ^{238}U in soil. The device, called BetaScint™, uses a layered configuration of scintillating fibers to
4482   detect betas from the radioactive decay of Yttrium-90 and Protactinium-234m (the equilibrium progeny
4483   of ^{90}Sr and ^{238}U, respectively). It can also discriminate between high- and low-energy betas and between
4484   beta and gamma-rays. To achieve this discrimination, it exploits the penetrating properties of betas and
4485   gamma rays. The detector measures 1.5 m x 0.35 m x 0.8 m and weighs approximately 20 kg. The
4486   monitor can be placed on or above contaminated soil or surfaces. Once the active window of the
4487   BetaScint™ sensor is placed over a sample of dry homogeneous soil, the beta particles excite electrons in
4488   a plastic fiber doped with fluorescent compounds in the layers of the sensor. The plastic fibers scintillate
4489   when the fluorescent molecules lose energy and return to their ground state. Scintillations in the plastic
4490   fibers are counted by photon detectors to determine the activity of the soil sample. The unit can be
4491   calibrated by exposing it to a soil with a known quantity of ^{90}Sr (or ^{238}U).

4492   The BetaScint™ is specifically designed to measure ^{90}Sr and ^{238}U, but cannot distinguish between beta
4493   radiation from ^{90}Sr and ^{238}U (it measures the sum of ^{90}Sr and ^{238}U). However, except in rare cases, ^{90}Sr
4494   and ^{238}U usually do not occur together because the source of ^{90}Sr contamination is a fission product,
4495   while ^{238}U is associated with the fuel or fuel element (that is, it is not a fission product). If other
4496   radionuclides are known (or suspected) to be present, data from other measurement techniques must be
4497   utilized. For example, high levels of ^{137}Cs in the soil will produce interference (the decay of ^{137}Cs emits
4498   two betas). Demonstrations have shown that ^{137}Cs interference will not become an issue, unless its
4499   concentration exceeds that of ^{90}Sr by many orders of magnitude. When ^{137}Cs and ^{90}Sr levels are
4500   comparable and less than 3.7 Bq/g (100 pCi/g) (i.e., typical soil remediation conditions), the ^{137}Cs
4501   contribution to the sensor background is negligible. More information on the BetaScint™ can be found
4502   in papers and reports by Schilk *et al.* (1994a, 1994b, 1995a, and 1995b) and U.S. DOE (1998c).

4503   *In Situ* **Gamma-Ray Spectrometry**

4504   *In situ* gamma spectroscopy is a measurement technique that uses HPGe detectors to measure gamma-ray
4505   fluence to quantify radionuclide inventories for a variety of source geometries. The technique has been
4506   used most often to measure activity in surface soil with real-time or near-real-time results. The approach
4507   has been commercialized by selling detectors that are calibrated for a specific application or source
4508   geometry.

4509   **Commercial Systems**

4510   The *In Situ* Object Counting System (ISOCS) from Canberra Industries, Inc., uses a computational
4511   process to identify and quantify radioactivity in a variety of geometrical arrangements. While the system
4512   can be calibrated using traditional prepared radioactive sources, the real advantage of the ISOCS
4513   software is the ability to calculate efficiencies by entering parameters such as the elemental composition,
4514   density, standoff distance, and physical dimensions. By using the supplied geometry templates (for
4515   example, boxes, cylinders, pipes, circular planes, rectangular planes, spheres, and wells such as Marinelli
4516   beakers), a calibration curve is generated that can be applied to multiple collected spectra. A more
4517   detailed review of this system may be found in Kasper (1999) and Kalb *et al.* (2000). The M-1 Gamma
4518   Spectroscopy System for *In Situ* Activity Measurements is an *in situ* system, manufactured by
4519   PerkinElmer. This system uses the DOE Environmental Measurement Laboratory characterization
4520   methodology. It is targeted for undisturbed soil measurements in environmental restoration projects,
4521   assessment of radionuclides deposited during emergencies, and routine environmental monitoring.
4522   PerkinElmer also produces an *in situ* system that consists of the ISOTOPICS software program; a mobile
4523   assay system, which includes a detector, collimator, and MCA called ISO-CART; and an HPGe detector.
4524   Of these components, ISOTOPICS and ISO-CART are intended to be used together for the
4525   nondestructive analysis of drums. The M-1 system and ISOCS participated in an intercomparison
4526   exercise, which evaluated the bias of the systems for measuring activity in surface soil. A discussion of
4527   the intercomparison and the results may be found in Miller *et al.* (1998).

4528   Capital Cost: $$$

4529   Eberline Services offers *in situ* spectrometry as a service. The service features a proprietary system,
4530   called Spectral Nondestructive Assay Platform (SNAP), which uses HPGe detectors to measure a variety
4531   of waste packages, including B-25 boxes, "D" boxes, glove boxes, and 208-liter (55-gallon) drums.
4532   Eberline Services claims that its approach enables the system to map contamination levels and locations
4533   with near-real-time results.

4534   *In Toto* **Monitors**

4535   *In toto* monitors covers a range of instruments that measure or assay objects *in toto*. The systems consist
4536   of a counting chamber, an array of detectors, and an electronics package. There is a wide variety of
4537   volume counters ranging from small item monitors to box counters and waste assay systems. A typical
4538   small item monitor has a counting chamber of about $0.08\ m^3$. Box counters and waste assay systems are
4539   designed to measure specific waste containers such B-25 boxes, which have a volume of $2.55\ m^3$. Since
4540   box counters and waste assay systems are designed to measure a specific type of waste (transuranic
4541   waste) utilizing advanced measurement methods, they are addressed in Section B.4. In general, volume
4542   counters use a variety of detectors such as gas proportional counters, plastic scintillators, and NaI(Tl)
4543   scintillators. These detectors are shielded (to reduce background) and surround the counting chamber to
4544   maximize the geometrical efficiency. Calibrations are performed with standard packages or suitable
4545   geometries containing sources of known activity.

4544   **Commercial Systems**

4545   BNFL Instruments has developed the IonSens® 208 Large Item Monitor. The system is called the
4546   "Large Item Monitor" because it has a chamber volume of nearly 1 cubic meter. The IonSens® 208
4547   determines the total alpha activity on objects by measuring the specific activity (number of ion pairs
4548   produced per unit path length by an ionizing particle) created by the alpha particles as they interact with
4549   the air surrounding the item being assayed. Filtered air passes over the object and is drawn to a detector
4550   which measurements the ionization. The system consists of two modules, an air inlet module and a
4551   measurement module. The air inlet module filters ambient air to remove particulates and dust before
4552   entering the measurement module. The measurement module is an airtight 1 m × 1 m × 0.8 m cavity in
4553   which the items to be measured are placed. BNFL reports a limit of detection of 10–15 Bq
4554   (600–900 dpm) for a 100-second count time.

4555   Thermo Eberline produces a series of small item/tool monitors, including the TCM-2, WCM-10,
4556   LRAD-1, and GTM. The TCM-2 is designed to detect hot particles and low-level contamination
4557   distributed on tools. The system uses an array of 6 gas flow proportional detectors, each of which is
4558   electrically divided, resulting in 12 channels or counting zones. The detector geometry is designed to
4559   minimize dead zones and maximize sensitivity. The system features "sumzones," which represent the
4560   combination of detector counts from any two channels. The sumzones are important for detecting
4561   distributed activity. This system has 30 sumzones and an adjustable interior volume. Thermo Eberline
4562   reports a sensitivity of 0.83 Bq/cm^2 (5,000 dpm/100 cm^2) for beta contamination with an approximate
4563   counting time of 10 seconds. The WCM-10 is intended for waste and uses six large area plastic
4564   scintillators. The counting chamber is heavily shielded and lined with polished stainless steel to facilitate
4565   decontamination. Thermo Eberline reports a sensitivity of approximately 74 Bq (2 nCi) of Co-60.
4566   An option to include a weight sensor outputs reported activity in activity per unit mass.

4567   The LRAD-1 uses the long-range alpha detection technique (see the next section for a description) to
4568   measure alpha contamination on surfaces. The detection principle is similar to BNFL IonSens®, which
4569   detects the ions produced by alpha particles. Thermo Eberline reports a sensitivity of approximately
4570   5 Bq (300 dpm) for objects that fit in the counting chamber, which has a volume of 0.08 m^3. The GTM
4571   is another tool monitor that uses a 5-cm thick plastic scintillator on four or six sides of the counting
4572   chamber. Just as with the TCM-2, the system utilizes a signal from the individual detectors as well as
4573   summed signals from any two detectors to measure "hotspots" as well as uniformly distributed sources.

4574   The G35-90 Package Monitor, manufactured by Canberra, is designed to detect the concentration and
4575   type of gamma-emitting radionuclides within small packages. Unlike the other systems, in which the
4576   counting chambers can be closed, the G35-90 has a 90-liter open-ended rotating drum for a counting
4577   chamber. The system is mobile and computer-controlled, and utilizes two shield NaI(Tl) scintillators.
4578   The system comes calibrated from the factory. No MDC or sensitivity data has been reported for the
4579   system. Finally, NE Technology produces the SAM 11 Small Articles Monitor. Like some of the other
4580   systems described in this section, it uses an array of shielded plastic scintillators to detect beta/gamma
4581   radiation. This system has a fairly large counting chamber volume, approximately 0.5 m^3.

4582   <u>**Pipes**</u>

4583   In addition to building debris, D&D activities have produced, and will continue to produce, a
4584   considerable amount of ductwork and piping. Because of their interior surface, long lengths of small-
4585   diameter ductwork and piping are largely inaccessible to conventional survey instrumentation.
4586   Manufacturers have, therefore, developed specialized instrumentation to survey the exterior and interior
4587   of piping.

4588   **Commercial Systems**

4589   The IonSens® Alpha Pipe Monitor, available from BNFL, is a modular system that measures total alpha
4590   contamination on metallic pipe work and/or scaffolding poles. It can accommodate lengths up to 6 m and
4591   diameters up to 15 cm. The detection method and basic operation is very similar to the IonSens® 208.
4592   The IonSens® Alpha Pipe Monitor consists of three basic modules, including the air inlet module,
4593   measurement module, and detection head module. The measurement modules are airtight and can be
4594   configured to accept 6-m lengths by joining three measurement modules. As with the other IonSens®
4595   systems, the detection head module contains the ion detector as well as a HEPA filter, fan, data
4596   processing electronics, iris seal, and PC. BNFL claims a limit of detection of 15 Bq (900 dpm) for a
4597   300-second count time. The detection module has a small standardized source that is used to monitor
4598   performance.

4599   The Pipe Explorer™, available through Science and Engineering Associates Inc., is a pipe
4600   characterization system that employs an airtight membrane deployed from a canister with air pressure to
4601   line the interiors of pipes and to carry a tether to which detectors are attached. As the membrane
4602   deploys, detectors are towed along inside the membrane while measurement data is collected. This
4603   system consists of three primary components, including (1) the deployment canister, which holds the
4604   membrane and detector assembly as well as the necessary transducers and sensors for the operation of the
4605   system, (2) the data acquisition computer, which logs and correlates information from the deployment
4606   and detector systems, and (3) the instrumentation and control box, which is used to control the
4607   deployment of the membrane and survey tools. The heart of the system is an airtight membrane that is
4608   initially spooled inside the deployment canister. Air pressure on the membrane causes it to be pulled
4609   from the spool, and deployed into the pipe. A characterization tool (such as a radiation detector) is
4610   attached to the end of the membrane and is towed into the pipe as the membrane unwinds. Because the
4611   membrane and detector are tethered to the spooler inside the canister, they can be wound back into the
4612   canister. The detector can, thus, be moved freely through the pipe while its output and position are
4613   continuously recorded. The Pipe Explorer™ system can be used to tow any detector that is compact
4614   enough to fit into a pipe. The tether has two coaxial cables available and six single conductor cables,
4615   which are used to provide power and control to the characterization tools. To measure alpha particles
4616   with the Pipe Explorer™, the membrane material itself must be an integral part of the detection system.
4617   An effective solution is to make the membrane material a scintillator, and then tow a photodetector
4618   through the pipe to detect the scintillation events occurring in the membrane. This is the approach
4619   adopted for the alpha measurement capability, which is referred to as the Alpha Explorer™ system.

4620   The Pipe Explorer™ system has been laboratory-tested and tested at a number of DOE locations,
4621   including Idaho National Engineering and Environmental Laboratory and Argonne National Laboratory.
4622   More information on the Pipe Explorer™ System is provided in published reports (Matalucci *et al.*
4623   1995a; Cremers *et al.* 1994, 1995, 1996, and 1997; Cremers and Kendrick 1998; and U.S. DOE 1996b).

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4626 The Pipe Crawler®, developed by Radiological Services, Inc., is a manually deployed pipe inspection
4627 system that consists of a crawler, mounted with a 360° array of thin GM probes connected by cable to an
4628 external data processing and storage system. A family of crawlers is used to accommodate various
4629 piping sizes. The dimensions of a given crawler must closely match the size of pipe to be surveyed; this
4630 ensures the proper counting geometry (the detector surface must be within about 1 cm of the surface),
4631 which is afforded by a spring-loaded wheel suspension system. Each crawler is custom made, employing
4632 commercially available GM tubes. The size and shape of the available GM tubes strongly influence the
4633 configuration and design of a given crawler. The smaller crawlers for pipes with diameters less than
4634 20.3 cm are manually deployed using flexible fiberglass rods attached to either end. The rods are similar
4635 to those used by plumbers. The larger crawlers (for 20.3-cm diameter and larger pipes) employ
4636 pneumatically operated positioning systems. It must be noted that the Pipe Crawler® is utilized by
4637 Radiological Services, Inc. exclusively as a part of a service they provide to customers and, as such, it is
4638 not for sale.

4639 **Subsurface**

4640 While *in situ* spectrometry provides a noninvasive approach to surface soil investigation, the subsurface
4641 remains intractable to such techniques. Current developments in instrumentation seek to reduce the
4642 burden of obtaining subsurface data. This basically involves using small detectors that can be pushed
4643 through the soil and are capable of real-time results. Because of the expense associated with the
4644 sampling equipment, subsurface measurements are typically provided as a service.

4645 One system related to subsurface sampling is the cone penetrometer, which consists of a $2–4 \times 10^5$ kg
4646 (20- to 40-ton) truck equipped with hydraulic rams to push steel cones, one section at a time, into the
4647 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
4648 1.5 m/hr (50 ft/hr). Compared to traditional drilling methods, cone penetrometer techniques are less
4649 costly, allow less-intrusive sampling and analysis, do not result in contaminated soils being brought to
4650 the surface, and minimize worker exposure to potential industrial and chemical hazards. Although cone
4651 penetrometer techniques have existed for many years, most earlier efforts focused on oil exploration and
4652 construction engineering. Only recently has the technique been applied in environmental
4653 characterization and monitoring, with resulting development of many sampling devices and sensors for
4654 use with the cone penetrometer. Applied Research Associates Inc. is a research and engineering
4655 company that provide subsurface sampling using a cone penetrometer.

4656 A spectral gamma probe, developed for DOE by the U.S. Army Corps of Engineers Waterways
4657 Experiment Station, was evaluated and demonstrated under field push (a push is when the penetrometer
4658 is driven into the ground) conditions at the DOE Savannah River Site in 1997. The probe consists of a
4659 2.5 cm x 7.6 cm NaI(Tl) scintillation crystal, a photomultiplier tube, a temperature sensor, and a custom
4660 designed preamplifier. The temperature monitor is used to track temperature changes, which can affect
4661 the performance of the spectrometer. The probe is driven into the subsurface using a cone penetrometer
4662 truck. During a field evaluation, nine pushes were made at three locations, and the gamma probe was
4663 stopped at 7.6-cm (3-in) to 30.5-cm (12-in) intervals for counting during each push. Results of the
4664 gamma probe measurements were compared with results of laboratory analysis of surrounding soils.
4665 Where the sites were primarily contaminated with ^{137}Cs with little beta activity, gamma probe results
4666 corresponded well with laboratory analysis results. However, the gamma probe experienced interference
4667 from the high level of beta activity found at some sites. In general, the lower limit of detection for ^{137}Cs
4668 was found to be in the range of 0.3–0.5 Bq/g (8–11 pCi/g).

4667 To minimize the deleterious effect caused by a high level of beta activity, Sentor Technologies, Inc. is
4668 developing a high-pressure xenon spectrometer device for use with the cone penetrometer. Three
4669 prototype devices have been built and tested in the laboratory; however, they are not commercially
4670 available.

4671 Commercially available radiation detection systems for subsurface measurements include HPGe
4672 detectors that have small diameter endcaps and dewars, typically about 7 cm, that can be lowered into
4673 boreholes. These detectors are available from PerkinElmer.

Portal Monitors

4675 Portal monitors cover a broad range of instrumentation reflecting a wide range of applications. For
4676 purposes of this discussion, a portal monitor is an instrument that detects radioactivity as it passes
4677 through a portal, which is typically an access point to a controlled area or checkpoint through which
4678 people, vehicles, equipment, and waste pass. Just as with many of the other systems previously
4679 discussed, these systems use large detectors to improve sensitivity. Most systems use plastic scintillators
4680 because they are rugged, inexpensive, and can be made with a large surface area. Count or integration
4681 times are very short (typically just a few seconds). The detectors are usually part of a structure which
4682 surrounds the portal on one, two or three sides. Although not strictly a portal monitor, plastic
4683 scintillators can also be attached to the base frame of grapples[10] to detect radioactivity in scrap metal.
4684 These devices have a clear advantage over portal monitors because the scintillator is in contact with the
4685 metal and remains in contact for as long as it takes to grab and move it, which could be several minutes.
4686 Like portal monitors, they are gross radiation detectors and do not provide quantitative information
4687 (e.g., activity per unit mass); they usually signal the operator when a preset threshold has been exceeded.

Commerical Systems

4689 A large number of portal monitoring systems are available from several manufacturers. This section
4690 briefly mentions a few systems. For monitoring small waste items as they pass through doorways,
4691 Ludlum makes a series (3530/3532/3534) of monitors that use NaI(Tl) scintillators. Models 3530 and
4692 3534 use two shielded 7.6 cm x 2.5 cm NaI(Tl) detectors, while Model 3534 uses four detectors. These
4693 detectors are mounted on opposite sides of a doorway or opening through which waste may pass. For
4694 larger waste items that are transported by vehicles, Ludlum makes Model 3500-1000WM, which utilizes
4695 two 7,866-cm^3 shielded plastic scintillation detectors. Exploranium is very active in the area of detecting
4696 radioactivity in scrap metal. They have a series of large portal monitors that detect radioactivity
4697 transported by vehicles, including railcars. These systems also use large plastic scintillators mounted to
4698 large structures.

4699 One portal monitor of note comes from Constellation Technology Corporation. They have developed a
4700 mobile system, known as the HPXe-1000, that performs spectroscopy. The unique feature of this system
4701 is the fact that it uses high-purity xenon gas (HPXe). The use of HPXe for gamma-ray spectroscopy is
4702 covered in the section on detector materials (see Section B.4). Constellation reports a resolution of
4703 3-percent FWHM at 662 keV for a detector that has a linear dimension of 1 m and a mass of almost 2 kg.
4704 The primary application for this system is the detection of special nuclear material for treaty verification.

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

4707	Rad/Comm Systems makes a grapple mounted detectors called the Cricket. The Cricket consists of a
4708	30 cm x 30 cm x 10cm scintillator mounted inside the top of the grapple. The system also has a
4709	protective shield, battery pack, and controller. Detectable source strengths for scrap densities of 0.5,
4710	0.75, and 1.0 g/cm³ range from 30–100 kBq (0.03–2.7 mCi) for ⁶⁰Co, 180–1,000 kBq (4.9 –27 mCi) for
4711	Cs-137, and 80–250 kBq (2.1–6.8 mCi) for ²²⁶Ra (de Beer et al., 1999).

| 4712 | ## B.4 Advanced Radiation Detection Systems |

4713	Advancements in radiation detection instrumentation have resulted from developments in material
4714	science, advances in electronics, and software. This trend shows no sign of slowing down and will
4715	continue to be the driving force behind the innovations in radiation monitoring instruments.

| 4716 | **Detector Materials** |

4717	One of the most important properties of a material that makes it a good radiation detector (and
4718	spectrometer) is its ability to absorb radiation energy. The property of a material to absorb radiation
4719	energy is known as the stopping power, which is defined as an average rate of energy loss of a particle
4720	per unit thickness of a material or per unit mass of material traversed. The higher the stopping power, the
4721	better the detector material. Stopping or absorbing the energy of charged particles is not an issue, but
4722	absorbing high-energy photons is. In general, high-density materials with large atomic numbers (Z) are
4723	ideally suited to absorb high-energy photons. Once a material has absorbed the radiation energy, it must
4724	be converted to information carriers. This conversion is accomplished either by producing ions as in the
4725	case of gas-filled detectors, electron-hole pairs as in the case of solid-state semiconductors, or
4726	photoelectrons as in the case of scintillators. A detector must be able to produce these information
4727	carriers efficiently; that is, with as little loss in energy as possible. The energy that is required to produce
4728	information carriers (ions, electron-hole pairs, photoelectrons) ranges from a few eV to about 100 eV.
4729	In general, the lower the better, in terms of the resolution for a spectrometer.

4730	In the case of solid-state semiconductor detectors, a rather large bias voltage (> 1,000 volts) is applied to
4731	the crystal. T his bias voltage creates a depleted region where electron-hole pairs are created when
4732	radiation energy is absorbed. The electrons and holes are swept from the depleted region and are
4733	collected to create a charge pulse. A good semiconductor material must have a high resistivity in order to
4734	prevent the collection of unwanted current, sometimes called leakage current, in the presence of a high
4735	bias voltage. The resistivity is linked to energy separating the valence and conduction bands, the so-
4736	called bandgap. The larger or wider the bandgap the greater the resistivity. If the bandgap is wide
4737	enough, the leakage current becomes low enough to permit room temperature operation.

4738	When describing the properties of a solid-state semiconducting detector material, the issues of purity and
4739	crystal defects are important. A process known as charge trapping occurs when charge carriers (electron
4740	and holes) recombine in the crystal lattice. This occurs for a number of reasons, but it is often traced to a
4741	lack of purity and crystal defects. The reduction in charge collection attributable to trapping reduces the
4742	size of the charge pulse and, therefore, reduces the resolution and efficiency of the detector. However, a
4743	new technique, which uses microwave photons instead of electrons as the information carriers, avoids
4744	some of the problems associated with charge collection.

4745 ## Cadmium Telluride and Cadmium Zinc Telluride

4746 A radiation spectrometer that operates at high (i.e., room) temperature has obvious advantages over
4747 conventional cryogenic spectrometers for applications where the system has to operate in an unattended
4748 mode or where liquid nitrogen (or a sufficient source of power) is difficult to obtain or too cumbersome
4749 to use. In recent years, the technology of radiation detectors that operate at room temperature has greatly
4750 improved, as a result of the ability to grow a number of semiconductor materials. Cadmium zinc telluride
4751 (CZT) and cadmium telluride (CdTe) are two such semiconductor materials with the properties required
4752 by a high-performance spectrometer. CdTe and CZT have high atomic numbers; however, a chief
4753 concern related to the use of alloy materials (including CdTe and CZT) for detector applications is
4754 degradation of detector resolution as a result of detector matrix heterogeneity. The most significant
4755 drawback of CZT is the insufficient supply of high-quality crystals for spectroscopic systems. This
4756 circumstance results from both uniformity issues and carrier transport properties.

4757 ## Other Detector Materials

4758 While CdTe and CZT are currently receiving most of the attention and focus as room temperature
4759 detectors, several other materials are being researched for this function. The following paragraphs
4760 briefly summarize the current development of four such materials, namely xenon (gaseous and liquid),
4761 mercuric iodide, lead iodide, and diamond.

4762 *Xenon*

4763 The properties of xenon that make it desirable as a detector material are that the energy required to
4764 generate an ion pair is 21.9 eV (which is smaller that argon and neon), and that its Fano factor is about
4765 0.17. This means, for example, that the 662-keV gamma-ray line from ^{137}Cs has an energy resolution of
4766 0.56-percent FWHM in xenon. This excellent intrinsic resolution, combined with a high atomic number
4767 (Z=54), shows that xenon is a suitable medium for high-resolution gamma-ray detection. Tepper *et al.*
4768 (1998) report on a cylindrical ionization chamber filled with highly purified xenon that has an energy
4769 resolution of 1.8 percent at 662 keV.

4770 Xenon does exhibit some nonlinear behavior in its density when its pressure is varied near its critical
4771 point[11], which corresponds to 10^6 dynes/cm^2 (58 bar), $\rho = 1.1$ g/cm^3 and 17°C. In general, at room
4772 temperature, xenon exhibits very little increase in pressure, for significant increases in density.
4773 Nonetheless, the sensitivity of the pressure to temperature must be considered when designing a detector
4774 using xenon (Mahler *et al.*, 1996). A portable gamma-ray system using xenon gas will be discussed later.

4775 *Liquid Xenon*

4776 Liquid xenon (LXe) has been used as a detection medium for an imaging telescope (Aprile *et al.*, 2000).
4777 LXe is an ideal material for high-energy gamma-ray detection because of its high density (3 g/cm^3) and
4778 high atomic number (Z=54). The ionization and excitation of xenon atoms, which result from these
4779 interactions produce a large number of electron-ion pairs (6,400 e-/ 100 keV, whereas gas proportional
4780 counters yield ~4,000 e-/ 100 keV) and a similar number of scintillation photons. However, when
4781 compared to gaseous xenon, the resolution of LXe (approximately 6 percent at 1 MeV) is somewhat
4782 poor.

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

4784   Red mercuric iodide (α-HgI_2) has been researched for almost three decades for use as a room temperature
4785   radiation detector material. Its high atomic number and wide bandgap make α-HgI_2 particularly well-
4786   suited for fabrication of room temperature compact spectrometers. It has been used to produce some of
4787   the highest resolution room temperature x-ray and gamma-ray detectors. However, these positive
4788   properties are balanced by several negative properties, including the fact that the material has a relatively
4789   high vapor pressure at room temperature, and the iodine is generally preferentially sublimed at a faster
4790   rate, yielding a mercury rich surface. Additionally, the material is mechanically very soft, and
4791   delaminates easily at the iodine layers (James 1996, Van Scyoc 1996).

4792   A novel room temperature, high-resolution HgI_2 spectrometer that has the needed performance and yield
4793   of high-quality detectors, with minimal support and maintenance requirements, has been developed
4794   (Van Scyoc, 1997). In particular, the reduction of charge trapping defects has been achieved by
4795   eliminating the material properties most degrading to performance. With these improvements, HgI_2
4796   devices with high-energy resolution over the range of x-ray and gamma-ray photon energies of 1 keV to
4797   1 MeV can be readily produced. Figure B-1 shows the dramatic difference between the [241]Am spectrum
4798   produced with a conventional HgI_2 detector on the left, and the same spectrum produced with the new
4799 HgI_2. Notice that the peaks on the right spectrum are much sharper and more symmetric. Also notice
4800   that while low-energy tailing is still visible, it is at a much lower level, which allows a Compton
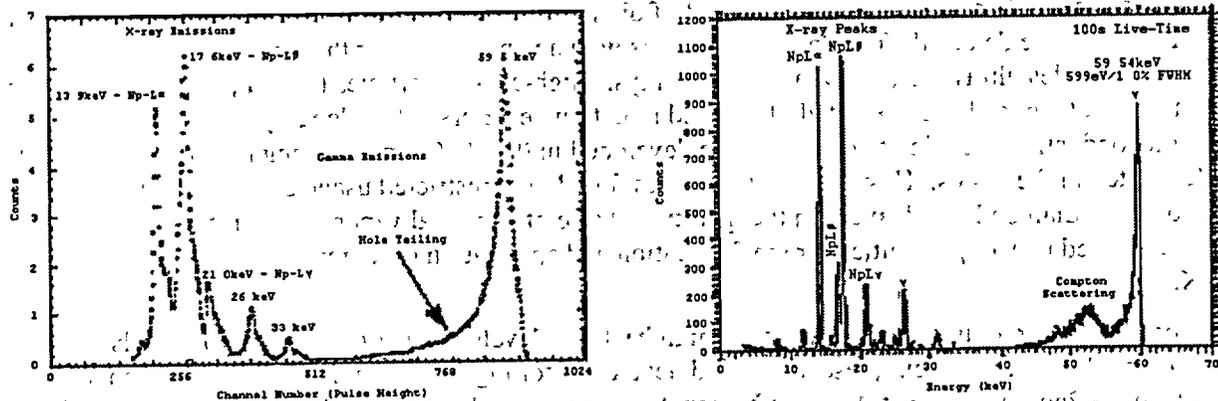4801   scattering peak to become visible.

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

4802	*Lead Iodide*
4803	Lead iodide (PbI_2) was first introduced in the 1970s as a candidate material for nuclear radiation
4804	spectrometry having an extraordinarily high efficiency for gamma rays. In addition, the wide bandgap of
4805	this material makes possible the growth of extremely high resistivity material. Lead iodide has a high Z
4806	and a high density (6.2 g/cm³), which means a high stopping power. Thus, room temperature, and even
4807	above room temperature, operation of gamma-ray spectrometers fabricated in this material is feasible.
4808	Also, the growing of single crystals of lead iodide is simpler in comparison to mercuric iodide or CZT
4809	growth. High-purity (99.9999 percent pure) PbI_2 is commercially available and further purification
4810	(which is crucially important for detector grade material) is accomplished by zone refining for
4811	100 passes[12]. The primary difference between recently demonstrated lead iodide detectors and those
4812	fabricated earlier appears to be the degree of crystal purity. However, one of the obstacles in dealing
4813	with PbI_2 is its poor mechanical behavior resulting from its layered structure.

4814 With the appropriate processing techniques, it has been found that detectors fabricated from high-purity
4815 PbI_2 crystal exhibit significant improvement in performance, compared to those produced from low-
4816 purity crystals. However, problems still exist in lead iodide because of the low charge carrier collection
4817 efficiency, which is probably caused by additional impurities or defects incorporated during crystal
4818 growth and detector fabrication processes (Hermon, 1997).

4819 *Diamond*

4820 For application to radiation detectors, the wide bandgap, radiation hardness, optical transparency, and
4821 low atomic number are important properties of diamond. Any radiation that generates free carriers in
4822 diamond can be detected. This includes photons with an energy greater than the bandgap of 5.5 eV,
4823 which includes ultraviolet, x-ray, and gamma rays. High-energy particles (e.g., alpha particles, electrons,
4824 neutrons, etc.)can also be detected. Diamond radiation detectors have a lengthy history.
4825 Photoconductive ultraviolet detectors were developed in the 1920s and ionizing radiation detectors were
4826 fabricated in the 1940s. However, these devices found only restricted usage because of the limitations of
4827 geological diamonds. Advances in the quality and size of chemical vapor deposition (CVD) diamonds
4828 have created new opportunities for the fabrication and application of diamond radiation detectors
4829 (Kania, 1997).

4830 Because of their ability to withstand very high heat flux levels and very high radiation levels, CVD
4831 diamond detectors are being researched and developed for high-energy physics devices, such as the
4832 Advanced Photon Source at Argonne National Laboratory and the Large Hadron Collider at the European
4833 Laboratory for Particle Physics (Liu *et al.* 1996, Hrubec *et al.* 1998; Friedl *et al.* 1998).

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[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.

4834	The detection of radioactive sources in scrap metal presents a harsh environment that excludes many
4835	traditional detector materials. The lifting magnets used in a scrap yard would be a favorable location to
4836	detect potentially contaminated metal entering the yard. Unfortunately, the presence of magnetic fields
4837	and mechanical vibration prohibits the use of traditional photomultiplier tubes with scintillation
4838	detectors. Moreover, the high temperatures restrict the use of solid-state detectors such as Ge or Si.
4839	Manfredi and Millaud (2000) have proposed that diamond be used as a detector material for
4840	contamination in scrap metal. Since diamond has a low Z, it is unsuitable for the detection of medium to
4841	high energy gamma rays. Manfredi and Millaud have proposed the development of a conversion-type
4842	detector that would be made of alternating layers of converter material and detectors. High-energy
4843	photons would strike the conversion material (tungsten has been suggested) and produce secondary
4844	radiation that could be detected in the diamond.

| 4845 | **Software** |

4846	The role of software in radiation detection is to facilitate the analysis and interpretation of information
4847	that detectors provide. Numerous analytical techniques have been developed, which utilize and optimize
4848	spectrometric information. For example, information in the form of a detector response, which can be
4849	calculated using radiation transport codes, can be combined with spectral information (e.g., count rates
4850	associated with radiation energy) to provide spatial distribution of radioactivity. Still other techniques
4851	improve detector sensitivity by optimizing spectrometric information. Software aids in the
4852	implementation of these analytical techniques, which can improve and extend the abilities of radiation
4853	detectors.

| 4854 | **<u>Gamma Detector Response and Analysis Software</u>** |

4855	The Gamma Detector Response and Analysis Software (GADRAS) is a collection of programs used to
4856	plot and analyze gamma-ray spectra. In contrast to most spectral analysis programs that find radionuclide
4857	concentrations by determining the areas of characteristic photopeaks and ignoring the
4858	continuum,GADRAS uses linear regression to fit the entire spectrum with a combination of computed
4859	spectral templates. Spectra are computed using a semi-empirical response function that was originally
4860	developed for use with sodium iodide detectors (Mitchell, 1986), and was expanded to accommodate
4861	other types of scintillators plus semiconductor detectors such as high-purity germanium. Subsequent
4862	developments that have been incorporated into the current response function enable computation of
4863	spectra based only on the detector material and dimensions. This capability can be applied to evaluation
4864	of detector designs prior to fabrication. GADRAS was developed at Sandia National Laboratory and is
4865	used primarily for safeguard applications (Mitchell, 1992a). It has been used to analyze air filter samples
4866	for the Remote Atmospheric Monitoring Project (Mitchell, 1987 and 1992b). Figure B-2 shows a typical
4867	spectrum analysis of an air filter sample using a modified form of GADRAS called RAMP-PC1.

4868	GADRAS-PC1 is a version of the software that has been written specifically for use on IBM-compatible
4869	personal computers. Routines included in GADRAS-PC1 enable a calibration of the response function
4870	parameters by fitting computed spectra for a set of calibration sources to measured spectra. The template
4871	set used in the analysis of unknown sources can include combinations of the 96 isotopic sources in the
4872	radionuclide library, fluorescence x-rays, or a user-defined library of source templates. The
4873	GADRAS-PC1 response function has been used to characterize a variety of sodium iodide, cesium
4874	iodide, bismuth germanate, and plastic scintillators plus high purity germanium detectors.
4875	GADRAS-PC1 is particularly useful for analysis of spectra recorded by the scintillators because the low
4876	resolution can preclude identification of photopeaks for all but the simplest gamma-ray sources. The
4877	analysis routine also excels for weak sources or measurements with short counting times because the
4878	entire spectrum is utilized, including statistically significant continuum regions.

4879 The GADRAS response function is based on the fundamental interactions of photons with the detector
4880 material. The first-order response is derived from the detector material's crosssections for photoelectric
4881 absorption, Compton scattering, and pair production. As many as 49 adjustable parameters can be used
4882 to include compensation for unusual scattering environments and anticoincidence shields. It is seldom
4883 necessary to use more than about 20 parameters, including those associated with the energy calibration
4884 and detector resolution. The response function also computes the effects of phenomena that are generally
4885 neglected, including: detection of coincident gamma-rays, pileup attributable to high count rates,
4886 bremsstrahlung radiation, escape of fluorescence x-rays, and leakage of high-energy electrons from the
4887 detector. Note that the response function obtained using GADRAS is not necessarily different from a
4888 response function obtained using a radiation transport code such as Monte Carlo N-Particle (MCNP)[13].
4889 GADRAS uses measurements and linear regression to obtain a response function, while a radiation
4890 transport code uses a simulation to determine the same quantity.

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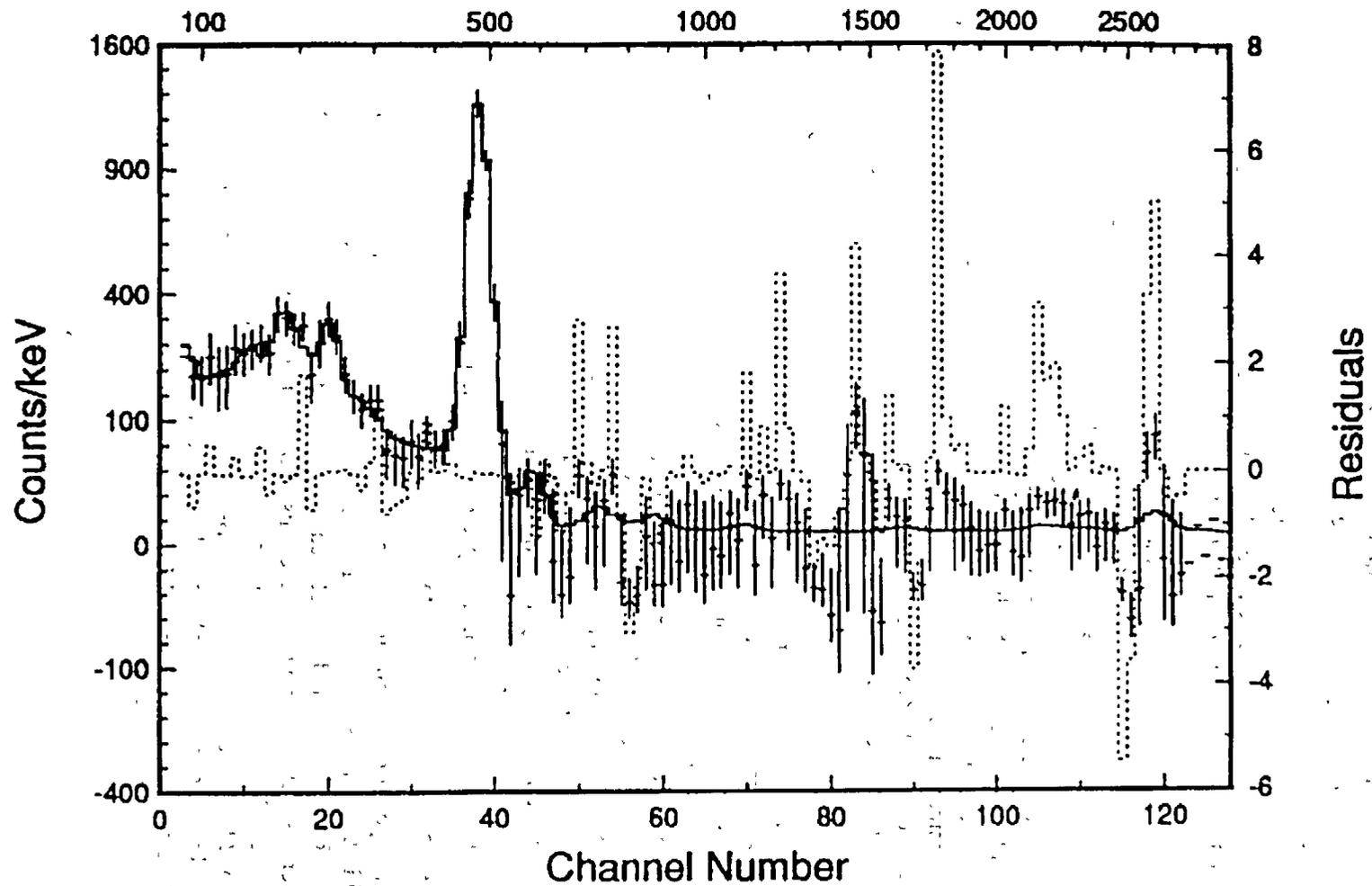
Figure B-2: Analysis of an atmospheric filter sample containing Be-7 using a modified form of GADRAS. The plot shows background subtracted data represented with a 1σ uncertainty. The step histogram gives the compound spectrum for the combination of isotopes including Be-7, ^{212}Pb, Ru-103, and Ce-141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

Gamma Penetration Depth Unfolding Algorithm

4891

4892 The Gamma Penetration Depth Unfolding Algorithm (GPDUA) comprises a computer code and
4893 measurement technique that uses the penetrating properties of gamma-rays to determine the depth of
4894 contamination in materials. The measurement technique uses a typical portable HPGe gamma-ray
4895 spectrometer system, consisting of a multichannel analyzer, high-voltage source, laptop computer (with
4896 appropriate counting software), and a portable HPGe detector with a collimator. The lead collimator
4897 serves two purposes, in that it (1) localizes the field of view, and (2) simplifies the efficiency
4898 calculations. It must be noted that the method is applicable to radionuclides that emit at least two
4899 gamma-rays, or radionuclides that emit a single gamma ray but have gamma-emitting progeny; parent and
4900 progeny must be in secular equilibrium. The peak areas that correspond to the energies of the uncollided
4901 gamma-rays are the only information necessary for GPDUA. It is the ratio of the counts in the peak areas
4902 that contains the necessary information to determine the depth of contamination. GPDUA uses a point
4903 kernel approach and solves an integral equation involving the net counts (from those photons incident on
4904 the detector face), the intrinsic efficiency, the distance from the source to the detector, and the depth of
4905 penetration. GPDUA solves the equation by iterating on the depth, and the depth that solves the equation
4906 is the depth of the contamination. GPDUA has been tested with MCNP and predicts the depth of
4907 contamination to within 10 percent of the actual (simulated) depth, regardless of the type of
4908 contamination distribution (i.e., point, disk, or linear distribution) (Naessens and Xu, 1999).

Microwave-Based Radiation Detector

4909

4910 As previously noted, room temperature semiconductors suffer from material defects, which limit their
4911 potential for high-energy gamma-ray spectrometry. Tepper and Losee (2001) are investigating the
4912 feasibility of using microwaves to measure changes in the conductivity of these wide-bandgap materials
4913 to determine the energy of the absorbed radiation. The method provides a way of extracting the energy
4914 information without having to collect the charge, which has been a problem for these materials. The
4915 method of using microwaves to measure the electrical properties of various materials has been used for
4916 years. This, however, is the first time that microwaves have been used for gamma-ray spectroscopy.
4917 Preliminary results show promise, but the sensitivity must be improved by at least two orders of
4918 magnitude before high-resolution gamma-ray using this technique is a reality. Tepper and Losee are
4919 confident that the sensitivity can be improved; however, it is unclear whether such a system could ever
4920 match the performance of conventional cryogenic spectrometers such as HPGe detectors.

Compressed Xenon Gamma-Ray Spectrometer

4921

4922 A prototype gamma-ray spectrometer utilizing xenon gas at high pressure has been developed at
4923 Brookhaven National Laboratory (Smith, 1996). Known as Compressed Xenon Gamma-Ray
4924 Spectrometer (COXGARS), it was initially developed for safeguards applications. COXGARS is a
4925 portable, battery-powered spectrometer, which functions at ambient temperature with an energy
4926 resolution between semiconductor (Ge) and scintillation (NaI(Tl)) spectrometers; Mahler *et al.* (1997)
4927 reports an FWHM at 662 keV of 2.5 percent. Figure B-3 shows the internal components of the
4928 COXGARS systems, which is capable of prolonged, low-power operation without a requirement for
4929 cryogenic fluids or other cooling mechanisms. Table B-2 provides some of the important characteristics
4930 of the compressed xenon spectrometer.

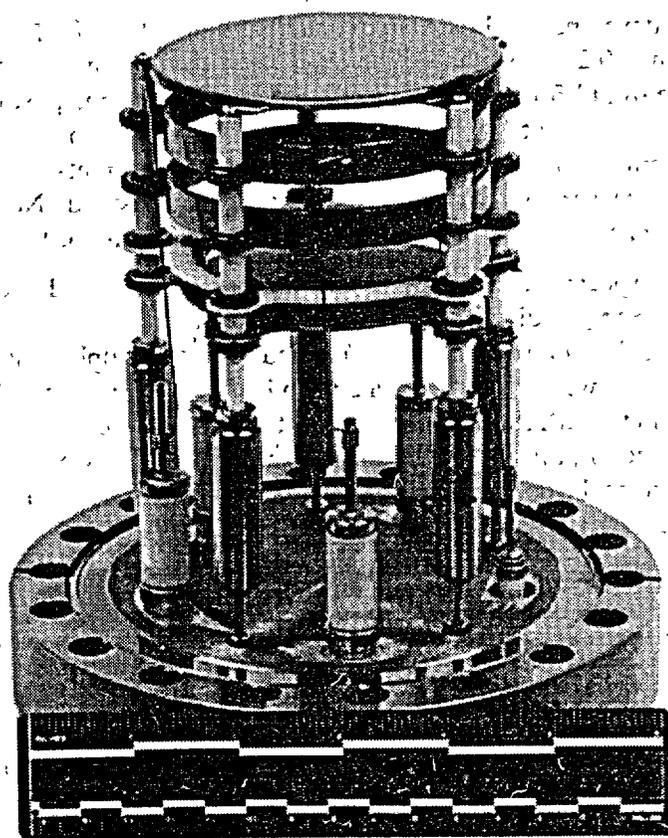|	Table B-2: Characteristics of COXGARS	
4931		
4932	Energy Range	100 keV to ~ 1 MeV
4933	Sensitive Volume	160 cm^3
4934	Sensitive Area	30 cm^2
4935	Energy Resolution @ 662 keV	2.5%
4936	Intrinsic Efficiency @ 200 keV/662 keV	40%/15%
4937	Detector Mass	10 kg
4938	Portable System Mass	Two 20 kg containers
4939	Power Consumption	7 W

Figure B-3: The internal structure of COXGARS

Static and Dynamic Long-Range Alpha Detector

4938

4939 Static and Dynamic Long-Range Alpha Detector (LRAD) systems are designed to monitor alpha
4940 contamination by measuring the number of ions produced by alpha particles as they interact with the air;
4941 a typical alpha particle will generate about 150,000 ion pairs. A key feature of the LRAD detection
4942 principle is that the ion pairs persist long enough so that ions may be collected on a detection electrode,
4943 which is located some tens of centimeters away from an alpha-contaminated surface. The ions may be
4944 transported to the electrode either by an air current or an electric field. Both the static and dynamic
4945 LRAD surface monitors use an electric field. A more detailed description of the LRAD concept and
4946 devices is contained in several reports (MacArthur 1991a, 1991b, 1992a, 1992b, and 1993).

4947 Static LRAD Surface Monitor. In the static LRAD, the ions generated over the surface to be monitored
4948 are collected on the detection electrode by a small electric field generating a bias voltage. This flow of
4949 ions represents a small current which can be detected by a current meter or recording device.
4950 This current is proportional to the total amount of contamination on the surface covered by the enclosure.
4951 The detector enclosure serves two purposes, (1) to define the active area of the detector and (2) to
4952 prevent externally generated ions from reaching the detector electrode and causing a spurious current.

4953 A static LRAD system developed by Los Alamos National Laboratory (LANL) for measuring surface
4954 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
4955 with the detector on the front lift moves the detector between monitoring positions; it places the detector
4956 open face down on the soil. About 15 minutes are required for signals to stabilize after the detector is
4957 moved to a new monitoring position. Once signals are stable, the currents are averaged for about 5
4958 minutes. In this current measuring mode, only alpha activity is measured. Note that the LRAD monitor
4959 relies on the physical connection between the LRAD enclosure and the surface to be monitored.

4960 Since the LRAD is not a spectrometer, it cannot identify radionuclides and, therefore, interference is a
4961 problem. It cannot, for example, distinguish between the alpha activity from naturally occurring alpha-
4962 emitting radionuclides such as uranium and thorium, and man-made alpha emitters such as plutonium.
4963 It also cannot distinguish between surface alpha contamination and radon gas that emanates from the soil
4964 and mixes with air within the LRAD chamber. The static LRAD detection electrode and the surface to
4965 be monitored form a capacitor; this is called a capacitive coupling. Any movement of one surface
4966 relative to the other changes the detector capacitance. This capacitive coupling causes a small current to
4967 flow in the detector, creating an erroneous signal in the detector.

4968 Field tests at various DOE sites have shown that LRAD surface soil monitors (SSMs) are faster and more
4969 sensitive than traditional alpha detectors for measuring alpha contamination (Johnson, 1993). However,
4970 an evaluation of the LRAD, performed at Savannah River, found several limitations to the application of
4971 this technology:

4972 ▪ The signals differed dramatically (factors of 20) above the uncontaminated sample materials.
4973 This likely resulted from differences in concentration of naturally occurring alpha emitters, such
4974 as uranium and thorium.

4975 ▪ The edge seals used in the prototype sometimes allowed radon in-leakage during the
4976 measurement. When this occurs, the LRAD signals do not stabilize.

4977 ▪ Any contact between the LRAD charge collection plate and the ground can result in leakage
4978 currents that are large relative to signals from uncontaminated soil. Great care must be taken to
4979 monitor soil where grass is growing.

4980 It was concluded that if the LRAD is used to locate alpha contamination and map its distribution, results
4981 must be used with caution (Sigg, 1995). Many false-positive indications are likely to be obtained, which
4982 could require additional measurements by other independent methods.

4983 Dynamic LRAD Surface Monitor. Some of the limitations discussed above (capacitive coupling and the
4984 fact that the detector must be in contact with the surface to be monitored) have been addressed by adding
4985 an additional electrode (MacArthur et al., 1998). Externally generated ions can be excluded using an
4986 electrostatic electrode. An electric field between the guard electrode and the surface excludes unwanted
4987 ions from entering the chamber volume. This guard electrode removes the requirement for physical
4988 contact between the enclosure and the surface. The LRAD can be continuously moved relative to the
4989 surface to be monitored.

4990 The guard electrode and gridded detector concepts are combined in the large dynamic surface monitor.
4991 This detector system can be operated in a scan mode with little or no loss of sensitivity. Movement of
4992 the detector relative to the surface includes both "moving-LRAD" applications (e.g., measurements of
4993 walls, floors, and soil), as well as "moving-surface" applications (e.g., soil and/or rubble conveyer belt
4994 systems). Although the grid on the front of the detection chamber makes it more vulnerable, grid wires
4995 as large as 0.5 mm in diameter have been demonstrated, and there is some speculation that larger wires
4996 would work as well. The current supplied to the exposed guard electrodes is limited to about a microamp
4997 without affecting the operation of the electrode.

4998 Waste Assay Systems

4999 Waste Assay for Non-Radioactive Disposal System (WAND). The WAND system scans low-density
5000 waste (mostly paper and plastic). This system is designed to verify that the levels of radioactive
5001 contamination (if present) are low enough so that the waste can be disposed of in public landfills. The
5002 WAND system was developed to reduce the volume of low-level waste that requires disposal from
5003 LANL.

5004 The WAND system consists of a lead-shielded chamber containing six 12.7-cm diameter phoswich
5005 detectors. A phoswich detector is a combination of two scintillators (in this case NaI and CsI) optically
5006 coupled to a single PM tube. The combination of scintillators rejects background events and separates
5007 the full energy x-rays from other signals. The WAND system has a conveyor system that moves a
5008 30.5-cm wide layer of paper through the chamber about 5 cm beneath the detectors and deposits the
5009 screened material into a waste bin. Either pre-shredded paper or packets of paper no more than 30 sheets
5010 thick, are manually placed on the conveyor belt.

5011 The electronic portion of the WAND system consists of electronic modules (needed to process the
5012 signals from the six detectors) and a desktop computer (486/66 PC). The software portion of the system
5013 consists of a custom analysis algorithm (written in C++ language), along with the code by which the
5014 operator controls the system and produces reports. Each phoswhich detector is equipped with a
5015 preamplifier and two electronic nuclear instrument modules (NIMs), which provide the buffering,
5016 amplification, and pulse shaping. To preserve the individual signals from each of the 12 detectors while
5017 using a single analog-to-digital conversion (ADC) module, a custom multiplexer module was designed to
5018 handle the data. With the exception of the multiplexer, the electronics are all commercially available.

5019	While moving the waste material at a speed of 1.27 cm/sec beneath the detector array, the system
5020	software performs a series of consecutive 10-second evaluations of the levels of radioactivity seen in
5021	each detector. If the count rate in any of the four energy regions of interest (ROIs) meets or exceeds the
5022	upper limit of the background, the conveyor belt backs up and does a recount. If excess radioactivity is
5023	detected on the recount, the conveyor belt stops and the software identifies the detector and the ROI that
5024	had the increased count rate. Additional information on the WAND system may be found in papers and
5025	reports by Arnone et al. (1998) and Myers (2000).
5026	**High-Efficiency Radiation Counter for Low Emission Sensitivity System (HERCULES).** The
5027	HERCULES system consists of a vertical array of three phoswich scintillation detectors positioned in a
5028	shielded detection chamber. Low-density waste is placed in a 30-gallon plastic drum, which rotates on a
5029	turntable (12 RPM) approximately 4.0 cm from the detector array. Count times can be varied according
5030	to detection sensitivity requirements, but the standard measurement time for most radionuclides is 1,000
5031	seconds. A sliding door on the top of the detection chamber allows for access to waste in the plastic
5032	drum. The chamber walls are filled with 2 inches of lead shielding and are lined on the interior with
5033	0.08-cm copper and cadmium sheets[14]. The HERCULES system uses the same electronic components
5034	and software packages as the WAND system, which makes the components easily exchangeable.
5035	Additional information on the HERCULES system may be found in Myers (2000).
5036	**Controleur Automatique de DEchets Faiblement Actifs (CADEFA).** The CADEFA is a system designed
5037	by Canberra Industries for assaying large samples, specifically waste containers for the decommissioning
5038	of the Chinon A3 Nuclear Power Plant. The samples can be as large as 1 m³ (250 gal) and weigh as
5039	much as 450 kg (½ ton). Samples that were measured using CADEFA were thermal insulation, steel
5040	pipes and beams, electrical wiring, and concrete. Gamma-ray spectrometry was used to achieve the
5041	desired detection levels in the presence of fluctuating levels of natural radioactivity. Some of the
5042	samples being considered for measurement at Chinon contain radionuclides that emit many gamma-rays
5043	such as Eu-152, Eu-154, and ⁶⁰Co, along with naturally occurring radium, thorium, and potassium. These
5044	radionuclides represent the limit that a NaI(Tl) scintillator and standard gamma-ray analysis software can
5045	reliably detect[15]. Hence, HPGe detectors are being considered, since they have much better resolution
5046	and would provide better results for this radionuclide mixture (Bronson, 1994).

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[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.

5049   <u>Transuranic (TRU)/ Low-Level Waste</u>. A number of requirements govern the disposition of DOE waste
5050   generated at both Federal and commercial disposal sites. These requirements constitute the basis for the
5051   performance of nondestructive waste assay (NDA) systems. The specific requirements for the
5052   disposition of transuranic waste types are defined in the Waste Isolation Pilot Plant (WIPP) Waste
5053   Acceptance Criteria and the associated Quality Assurance Program Plan (U.S. DOE, 1996d). WIPP
5054   requirements essentially force NDA systems to be able to quantitatively determine alpha-emitting
5055   transuranic elements with a half-life greater than 20 years that comprise 95 percent of the hazard. WIPP
5056   also requires NDA systems to have sufficient sensitivity to verify that the total alpha activity per gram of
5057   waste matrix exceeds 3,700 Bq/g (100 nCi/g). I n addition, the NDA technique must have a measurement
5058   range equal to or greater than a 325 fissile gram equivalent[16]. Therefore, a significant amount of
5059   technological development and innovation is being brought to bear on NDA systems for the assay of
5060   TRU waste for storage at WIPP.

5061   **Technologies and Methodologies**

5062   Some aspect of the technologies and methodologies used in this field could be applicable to the
5063   measurement of residual radioactivity in volumes and on surfaces. The following paragraphs discuss
5064   some representative technologies.

5065   <u>Active & Passive Computed Tomography</u>

5066   Computed tomography (CT) is a radiographic method that permits the nondestructive physical and, to a
5067   limited extent, chemical characterization of the internal structure of materials. Since the method is x-ray
5068   based, it applies equally well to metallic and non-metallic specimens.

5069   In conventional radiography, x-rays pass through the object, and the transmitted intensity is recorded as a
5070   two-dimensional image. The information contained in this radiograph is a projection of the absorption
5071   density in the sample onto the plane perpendicular to the x-ray beam direction. When the sample is
5072   imaged several times in different orientations, volumetric information on the sample structure can be
5073   obtained using computer algorithms. Known as a tomographic reconstruction or tomography, this
5074   enables us to look at "slices" of the investigated object without physically cutting it. Figure B-4
5075   illustrates the CT process.

5076   Active and passive computed tomography (A&PCT) is a gamma-ray NDA method, which has been used
5077   to identify and quantify transuranics in 208-liter (55-gallon) waste drum containers (Martz *et al.*, 1996,
5078   1997, and1998). The A&PCT consists of two separate measurements. The first is an active CT (ACT)
5079   scan that can yield quantitative attenuation data (related to density and atomic number) using an external
5080   radiation source. The second measurement is a passive CT (PCT) scan that can, in principle, localize all
5081   detectable radionuclides within a volume (in this case, a drum) and determine their identity if an entire
5082   energy spectrum is obtained.

---

[16]A method of normalizing fissile and fissionable isotopes to plutonium-239 for use in establishing criticality safety limits

5083   For ACT, the function to be imaged is the measured x-ray or gamma-ray attenuation of an external
5084   source, whereas in the case of PCT, the function to be imaged is the measured x-ray or gamma-ray
5085   activity at one or more energies of all detectable radionuclides within a drum. The ACT images are used
5086   to correct the PCT images for attenuation to determine the activity of the internal or external emitting
5087   source. For an A&PCT scanner with gamma-ray spectrometry detection equipment, each radionuclide in
5088   the drum can be identified by the energy of its characteristic radiation. More information on A&PCT can
5089   be found in papers and reports by Decman (1996), Keto (1995), Matalucci (1995b), and Robertson (1997
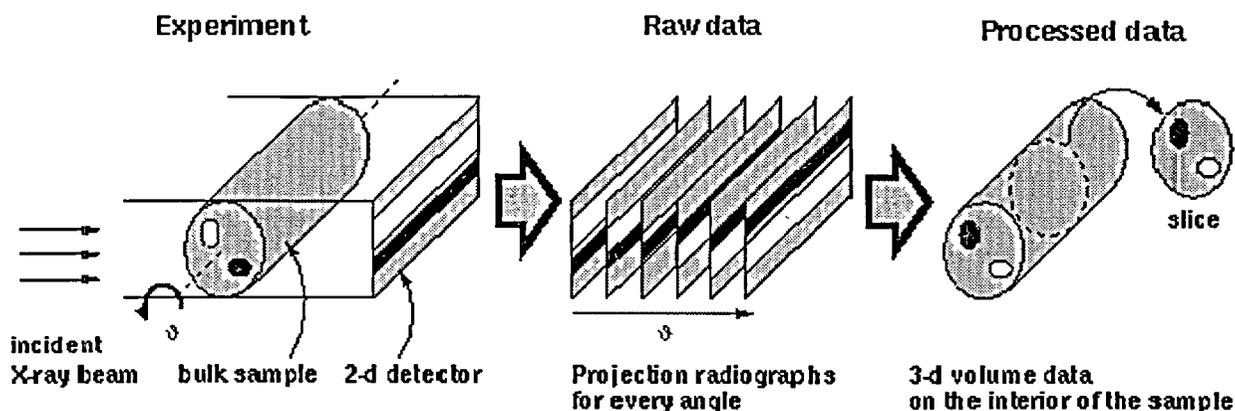5090   and 1998).

Figure B-4: The Computed Tomographic Process

5091   Becker et al. (1999) evaluated 13 (with 1 under development) boxed waste NDA technologies, 2 passive
5092   neutron-based systems, and 7 active/passive neutron-based systems. Some of the technologies for the
5093   boxed waste NDA assays are summarized below. Detailed information from Becker et al. was preserved
5094   to illustrate the level of technology that is used to assay boxed waste containers. Background
5095   information on the technologies was included when provided.

5096   **Canberra's Gamma Box Counter**

5097   The Canberra Gamma Box Counter is designed to accommodate a variety of box container sizes up to the
5098   large (~ 80 m^3) shipping container. The system is typically configured with either two or eight HPGe
5099   detectors, which can be placed close to the container to optimize sensitivity, or at a distance for a far-
5100   field measurement of higher dose rate containers. The system is intended to characterize fission and
5101   activation product waste, as well as waste generated from plutonium, uranium, radium, and thorium
5102   processing applications. These waste forms are typically generated in decommissioning or
5103   environmental restoration applications. Mathematical calibrations are generated using Canberra In Situ
5104   Object Counting Software (ISOCS). Matrix corrections are performed using an average density matrix
5105   correction technique based on the sum of spectral data from all detectors. Corrections for nonuniform
5106   distributions can be accomplished through the calibration and through a differential peak absorption
5107   analysis technique. Qualitative evaluations of nonuniformity can also be made by evaluating the
5108   response of the individual detectors.

5109 **Oak Ridge National Laboratory's Y-12 Box Assay System**

5110 The Y-12 B-25 box NDA system is used to sort "non-radioactive waste" from low-level waste at the
5111 1.3-Bq/g (35 pCi/g) total uranium activity. The system was designed and built at the Y-12 plant and
5112 commenced operation in early 1996. The waste form characterized by the system is produced as a
5113 byproduct of Y-12 plant operations and decontamination and decommissioning activities, and is routinely
5114 packaged in the B-25 type box.

5115 The Y-12 box assay system is composed of two arrays of uncollimated 12.7-cm diameter by 12.7-cm
5116 thick NaI(Tl) detectors. Each array consists of six detectors placed on the long sides of the box.
5117 Detector spacing is determined according to the Nyquist critical spatial frequency[17]. Each detector is also
5118 positioned 31.75 cm from the surface of the waste box. The output of each detector is routed to a
5119 multichannel analyzer for display and analysis. Regions of interest are set for peak area quantification at
5120 the 185.7-keV gamma-ray from ^{235}U and 1,001-keV gamma-ray from ^{234m}Pa. Analysis is performed using
5121 a point-source efficiency response followed by a transmission correction for attenuation, thus quantifying
5122 the radioactivity of ^{235}U and ^{238}U. Four HPGe detectors, two on each side, screen the box for the
5123 presence of non-uranium isotopes to provide information on enrichment. A 5-cm thick iron wall on each
5124 side of the detector arrays provides background radiation shielding.

5125 In a separate measurement station, a three-position gamma-ray transmission measurement is made
5126 through the short, horizontal axis of the box. This measurement allows correction of the uncollided flux
5127 for matrix attenuation. The transmission measurement is acquired via three collimated NaI(Tl) detectors
5128 (7.6-cm diameter by 7.6-cm thick) located on one side of the box, opposite three depleted uranium and
5129 three enriched uranium transmission sources on the other side. Data from the two measurement systems
5130 are fused together in an algorithm that yields measurement results for ^{235}U and ^{238}U.

5131 **East Tennessee Technology Park (ETTP) K-25 Box Assay System**

5132 The East Tennessee Technology Park, formerly the K-25 Site, was a uranium enrichment facility that
5133 processed and stored a large variety of radioactive wastes. These waste forms are generated primarily as
5134 a result of maintenance and decontamination and decommissioning operations in the five gaseous
5135 diffusion plants. The B-25 type box is the predominant container type used for waste packaging. Matrix
5136 types are segregated into two broad categories, including combustibles and metallic waste forms. The
5137 waste is primarily contaminated with uranium at variable enrichments that historically have averaged
5138 approximately 3 percent. Techniques used include NaI(Tl) gamma, HPGe gamma, and passive neutron.
5139 The measurement protocol commences with an assay at the NaI(Tl) detector station, followed by a
5140 passive neutron measurement for metallic type matrices only, and a final measurement via a HPGe
5141 gamma spectroscopy system.

5142 The NaI(Tl) measurement station consists of four 12.7-cm diameter by 7.6-cm thick lead collimated
5143 NaI(Tl) detectors interfaced to a PC-based analyzer equipped with four 1,000-channel analyzers. Two
5144 detectors are centered on each long side of the B-25 box, 45.7 cm from the edge at 91.4 cm, box surface
5145 offset. The system independently processes signals from each of the four detectors. Regions of interest
5146 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
5147 sum response of the four detectors, corrected for efficiency, attenuation, and background, is the basis for
5148 mass determination on either ^{235}U or ^{238}U.

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[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

5149 The radioactive source's spatial and matrix attenuation dependent detector response is modeled for each
5150 NaI(Tl) detector using a program called GAMMAEFF. Corrections for matrix attenuation are based on
5151 the net box weight to determine matrix density and knowledge of the matrix type to arrive at appropriate
5152 gamma attenuation coefficients. The matrix density is determined from the net box weight with the
5153 assumption that the matrix fills the box homogeneously. The GAMMAEFF program uses the matrix
5154 type, density, and associated attenuation coefficients for determination of matrix attenuation correction
5155 factors over a range of matrix types and densities. The matrix correction factor is applied to each of the
5156 NaI(Tl) responses, and the sum of the four detectors are used to arrive at the isotope mass. A 3-percent
5157 uranium enrichment is assumed for the NaI(Tl) measurement when the ^{235}U and ^{238}U masses are less than
5158 0.2 and 30 grams, respectively. Mass values less than these do not allow use of the HPGe system for
5159 enrichment measurements due to sensitivity considerations. Under such conditions, the NaI(Tl) system is
5160 effectively a standalone measure.

5161 A passive neutron measurement station is used to verify that large masses of highly enriched ^{235}U have
5162 not been missed in the heterogeneous steel matrix. The HPGe measurement is used to estimate the
5163 ^{235}U enrichment and identify the presence of other gamma-ray emitting radionuclides. The mass of
5164 ^{235}U or ^{238}U (based on the NaI(Tl) measurements) is used as the reference value for determination of
5165 enrichment and mass of other radionuclides through HPGe measured relative ratios. The system consists
5166 of one collimated HPGe detector positioned to view the long side center of the box. The HPGe detector
5167 is interfaced to a PC data acquisition and analysis system. The results of radionuclide identification and
5168 peak fit routines are input to the ISOTOPICS program, which uses this information with measurement
5169 configuration data to compute geometry and matrix attenuation corrections. Matrix and container
5170 material types are adjusted to ensure applicable mass attenuation coefficients are employed for the
5171 gamma-ray energies of interest. The HPGe results are normalized to the ^{235}U, and occasionally ^{238}U,
5172 mass derived from the NaI(Tl) measurement station. The NaI(Tl) based ^{235}U mass value used as this
5173 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

5175 The specification and preliminary design of a waste assay system for the identification and quantification
5176 of gamma-ray-emitting radionuclides in the B-25 waste box container has been performed at the Oak
5177 Ridge National Laboratory Waste Examination and Assay Facility (WEAF). The system, tentatively
5178 called the B-25 Box Assay System (B-BAS), is designed to address the need to measure the radionuclide
5179 content of a B-25 waste box at its site of residence. This is specifically intended to reduce costs by
5180 minimizing transportation of the box to a facility specifically for nondestructive assay or representative
5181 sampling of its contents.

5182 The B-BAS is based on an array of eight low-resolution/high-efficiency 7.6-cm by 7.6-cm NaI(Tl)
5183 detectors for identification and quantification of waste entrained, gamma-emitting radionuclides.
5184 Four detectors are positioned on one long side of the B-25 box with a symmetrical arrangement of the
5185 remaining four on the opposite side. The eight detectors are mounted to a moveable support structure
5186 with large wheels, allowing the B-BAS assembly to be moved by hand down the long axis of a B-25
5187 waste container. This moveable structure is designed to be easily transportable between measurement
5188 sites. The wheels are removed to insert the B-BAS in the WEAF Real-Time Radiography (RTR) system
5189 for the ultra-high-sensitivity "No Rad Added" type measurements.

5188   The moveable detector assembly positions the detectors at a distance of 30 cm from the surface of the
5189   B-25 box. The detector's spatial configuration is designed to allow a maximum field of view for the
5190   middle two detectors and a minimum field of view for the uppermost and lowermost detectors. The two
5191   middle detectors have the same collimator design (i.e., a 34.2 degree angle from the centerline of the
5192   collimator). The uppermost collimator has a smaller field of view with only a 9.5 degree angle of
5193   collimation with respect to the centerline. The smallest field of view is implemented in the lowest
5194   detector (4.4 degree angle with respect to the centerline). Each collimator has at least 2.5 cm of lead to
5195   shield background gamma rays.

5196   The measurement protocol for the B-BAS is to acquire data in a scanning fashion by movement of the
5197   NaI(Tl) detector array across the B-25 box. This scanning data acquisition mode is performed manually
5198   by operating personnel. When the B-BAS is inserted into the WERF RTR chamber, the wheels of the
5199   B-BAS are removed and the detectors are fixed. Scanning is achieved within the RTR chamber via a
5200   B-25 box transport system, which moves the box past the fixed detector array at a constant speed.

5201   Signals from the NaI(Tl) detectors are routed into two mixer/routers. Each of the two mixer/routers
5202   allows simultaneous acquisition of up to four signals. These mixer/routers have a preamplifier and an
5203   amplifier on each channel. The preamp/amp combination allows the user the ability to "gain match" the
5204   detectors. The purpose of gain matching is to allow spectra summing for the detector arrays by adding
5205   channel to channel. The summed spectra are processed through a PC-based, multichannel analyzer card.

5206 **B.5   A Survey of Reported Minimum Detectable Concentrations for Selected Instruments and**
5207   **Measurement Methods**

5208   For low-level measurements, the minimum detectable concentration (MDC) is an important performance
5209   characteristic. It is usually difficult to make a fair and meaningful comparison of the sensitivity between
5210   various instruments (e.g., a gas proportional counter and a GM tube) and measurement methods
5211   (e.g., total ionization and gamma-ray spectrometry). Yet, some approaches are generally regarded as
5212   more sensitive than others. This section lists MDC values for a collection of instruments and
5213   measurement methods that are relevant to clearance. In most cases, MDC values are provided from
5214   instrument vendors without any explanation concerning the methods and specific formulae used to arrive
5215   at these values; therefore, they should be viewed with caution.

5216   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
5217   technologies that have been applied to volumetric contamination. Table B-3a categorizes the
5218   techniques/technologies according to the application, assay strategy, matrix, source size, assay
5219   technique/technology, and radiation detector. Assay strategies reflect techniques that are used to
5220   quantify activity. They range from simple techniques that measure total ionization to more sophisticated
5221   techniques that involve spectroscopy with passive and active methods of background reduction. Surface
5222   measurements are treated in Tables B-4a and B-4b. Note that, unlike Table B-3a, these tables do not
5223   address applications because (for the technologies listed) the application is exclusively for
5224   decontamination and decommissioning (D&D). Also, note that for surface contamination, the preferred
5225   detection method involves measuring total ionization, which precludes (for the most part) radionuclide
5226   identification.

5227   The range of MDC values for volumetric contamination is rather large. The Compton suppression well
5228   counter (CSWC) has an MDC of a few tenths of a Bq/kg in the case of ^{137}Cs, while scanning for natural
5229   uranium using scintillators has an MDC of several thousand Bg/kg. The situation is similar for surface
5230   contamination; the MDCs range from a few tens of Bq/m^2 for liquid scintillation counting to a few
5231   thousand Bq/m^2. Count times range from 1 second in the case of scanning measurements to a day or
5232   more for laboratory analysis. Sample size (and active area in the case of surface contamination) is one of
5233   the key features in determining the sensitivity. Note that in the case of the CSWC (Table B-3a,
5234   ID nos. 4a, 4b, 4c), the sensitivities are fairly low and somewhat comparable to the MDCs for the *in situ*
5235   measurements of soil taken with a HPGe detector at a standoff distance of 1 m (Table B-3a, ID nos. 5a,
5236   5b, 5c, 5d). The *in situ* soil measurements achieve low MDCs with a relatively short count time
5237   (as compared to the CSWC) because of the large sample size. The CSWC uses just a few grams of
5238   material, while an *in situ* soil measurement has an effective sample size of about 100,000 kilograms.
5239   Compare that situation with the *in situ* measurement of soil; note the MDC for ^{137}Cs is a respectable 0.8
5240   Bq/kg. This situation is similar for surface contamination. The LRAD system (see Table B-4b, ID no. 4)
5241   has an MDC in the range of 12–30 Bq/m^2, compared to a gas proportional counter with an MDC for ^{230}Th
5242   and transuranics of 600 Bq/m^2. While the count time is not given for the LRAD system (it is not
5243   unreasonable to believe that it is commensurate with the count time for the gas proportional counter),
5244   we see that the active area of the LRAD is 100 times greater than that of the gas proportional counter.

5245   The foregoing discussion leads us to a general conclusion that has implications for the design of a
5246   detection system and/or measurement strategy to achieve the appropriate MDC value for a given
5247   application. Specifically, *use the largest practical sample size coupled with the largest practical*
5248   *detector or array of detectors.*

5249   It is clear that measurement of radioactivity associated with the control of solid materials is greatly
5250   facilitated by the development of new radiation detectors and detection systems. Of the systems
5251   addressed, the ones being developed for the assay of transuranic waste are of particular interest.
5252   Although not directly applicable to levels of radiation near background, they do represent the state-of the-
5253   art in radiation detection. This appendix attempted to compare the detection sensitivity for a variety of
5254   systems, with the caveat that many of the reported MDCs are from instrument manufacturers and should
5255   be viewed with caution. The comparison is valuable in the sense that it led to a general conclusion
5256   regarding the sensitivity of radiation detectors for radioactivity associated with the control of solid
5257   materials.

Table B-3a: Measurement technologies for volumetric contamination

ID #	Application	Assay Strategy	Matrix	Source Size (g)	Assay Technique/ Technology	Radiation Detector
1a			water (ρ=1.0 g/cm³)	1000		HPGe (60% rel. efficiency)†
1b						
1c						
2	Routine sample analysis			250	gamma-ray spectrometry with shielded detector	HPGe (115% rel. efficiency)†
3		sampling & lab analysis	soil (ρ=1.0 g/cm³)	—ᵃ		NaI (Tl) (7.6 cm × 7.6 cm)
4a					Compton suppression well detector/gamma-ray spectrometry	HPGe well detector (125 cm³)
4b	Environmental			3		
4c						
5a						
5b					in situ	HPGe
5c				~10⁸	gamma-ray spectrometry at 1 m	(40% rel. efficiency)†
5d		NDA/ direct measurements	soil (ρ=1.5 g/cm³)		in situ	6 HPGe
6	D&D			~10⁹	gamma-ray spectrometry at 8 m	(75% rel. efficiency)†
7				N/A	gamma-ray spectrometry	CZT array
8a				N/A	portable energy dispersive x-ray fluorescence	HgI₂
8b						—ᵃ

Table B-3a: Measurement technologies for volumetric contamination

	ID #	Application	Assay Strategy	Matrix	Source Size (g)	Assay Technique/ Technology	Radiation Detector
5278	9a	D&D					
5279	9b				N/A	laser ablation mass spect.	N/A
5280	9c		NDA/				
5281	10		direct measurements		~700	scintillating fiber optics with anti-coincidence counting	Fiber Optic (Beta-Scint™)
5282	11a			soil			NaI(Tl)
5283	11b			(ρ=1.5 g/cm³)			(3.8 cm × 3.8 cm)
5284	11c		NDA/		N/A	gross radiation counting	
5285	12a		hand-held scanning				NaI(Tl)
5286	12b						(5.1 cm × 5.1 cm)
5287	12c						

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	ID #	Application	Assay Strategy	Matrix	Source Size (g)	Assay Technique/ Technology	Radiation Detector
5288	13a						
5289	13b					WAND system	
5290	13c	Waste Assay	NDA/ *in toto*	low density	N/A		Array of Phoswich Detectors
5291	14a						
5292	14b					HERCULES system	
5293	14c						
5294	15a						
5295	15b			low Z, low density	$\sim 10^7$	*in situ*	HPGe
5296	15c					gamma-ray spectrometry at 1 m	(40% rel. efficiency)†
5297	15d			$(\rho = 0.3 \text{ g/cm}^3)$			
5298	16a						
5299	16b				200 liter (55 gallon)	*in situ*	HPGe
5300	16c			--ª	drum	gamma-ray spectrometry at 1 m	(40% rel. efficiency)†
5301	16d						
5302	17a				5×10^6	CADEFA	
5303	17b			Misc.Waste		gamma-ray spectrometry	--ª
5304	18a				8		
5305	18b						
5306	19	Safeguards		HEU in van	4×10^4 - $2 \times 10^{5\ b}$	portal monitor	plastic scintillators

5307 --ª data not provided

5308 ᵇ represents total mass of radionuclide (e.g , 40 – 200 kg of highly enriched uranium (HEU))

5309 † 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	$\sim10^5$	
5b	^{137}Cs	900	0.8		www canberra com/literature/technical_ref/ga mma/isocs
5c	^{238}U		110	$\sim10^8$	
5d	^{241}Am		3.6	$\sim10^5$	
6	^{241}Am	3600	3.8	$\sim10^{10}$	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 x10⁴	
5346	15b	^{137}Cs	900	12	1 x10⁵	www.canberra.com/literature/technical_ref/gamma/isocs
5347	15c	^{238}U		1100	1 x10⁷	
5348	15d	^{241}Am		1900	2 x10⁷	
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	~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			FIDLER* (NaI(Tl))	--ᵃ
3a		γ ray spectrometry with unshielded detector		
3b			HPGe (40% rel. efficiency)†	N/A
3c				
4		LRAD/ total ionization	ionization chamber	1
5a				
5b			gas proportional counter	0.01
5c	NDA/ direct measurements			
6		total ionization	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			scintillating membrane	
11b		Pipe Explorer™/total ionization	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

5389

Table B-4b: MDC values for surface contamination

ID #	Time (s)	Radionuclide/ Radiation Type	MDC (Bq/m²)	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	www canberra com/literature/technical_ref/gamma/i socs
3b	3600	^{137}Cs	3500	
3c		^{241}Am	310	
4	--ᵃ	α activity	12-30	NUREG-1575, 1997
5a		^{14}C	930	NUREG-1507, 1998
5b	60	^{99}Tc	4.9	
5c		^{90}Sr(^{90}Y)	2.9	
6		^{230}Th and transuranic	600	Goles, R.W., 1991
7		^{230}Th	108	
8a		^{90}Sr(^{90}Y)	10⁴	
8b		fission products	10⁴	
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

5414 <div style="text-align:center">References</div>

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2 TITLE AND SUBTITLE

Criteria for Preparation and Evaluation of Radiological Emergency Plans
Radiological Surveys for Controlling Release of Solid Materials

Draft Report for Comment

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10. SUPPLEMENTARY NOTES

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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.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

survey instruments
radiological surveys
clearance