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DOE STANDARD

CLEARANCE AND RELEASE OF PERSONAL PROPERTY FROM ACCELERATOR FACILITIES

U.S. Department of Energy
Washington, D.C. 20585

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FOREWORD

This Department of Energy (DOE) Technical Standard (STD) is approved for use by all DOE components, including the National Nuclear Security Administration (NNSA), and their contractors.

Beneficial comments (recommendations, additions, and deletions), as well as any pertinent data that may be of use in improving this document, should be addressed to:

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This Technical Standard is developed to support the control, clearance, and release of personal property (materials, equipment, and items) from accelerator, accelerator facilities, and modules thereof. This Technical Standard focuses on materials and equipment that have the potential to be impacted by accelerator operations and addresses volumetric residual radioactivity issues and establishes a three tier approach that can be used for decision-making relative to the overall clearance process. This Standard facilitates implementation of requirements in DOE Order (O) 458.1, *Radiation Protection of the Public and the Environment*, and uses the dose constraint of 1 mrem/y, in ANSI N13.12-2013 Standard, *Surface and Volume Radioactivity Standards for Clearance*, [ANSI2013]. ANSI N13.12-2013 Standard derives and establishes the screening levels (SLs) for the clearance of materials that contain, or may contain, residual surface or volume radioactivity from radiological control which generally satisfies the criteria set forth in the Order.

Throughout this Standard, the word "shall" is used to denote requirements essential to satisfy the intent of this Standard. "Should" is used for recommendations intended to ensure quality objectives in the Order are met. "May" is used to denote permission, but not a requirement or recommendation.

This Technical Standard was prepared following the requirements for due process, consensus, and approval as required by the U.S. Department of Energy Technical Standards Program. Consensus was established with substantial agreement by all members of the writing and review teams, and the Technical Standard was approved by the DOE directives approval process (RevCom).

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1. Purpose and Scope

Purpose

The purpose of this Standard is to provide Department of Energy (DOE) accelerator facilities or modules thereof an acceptable approach for managing and dispositioning personal property that may be radiologically impacted (mainly by activation) by accelerator operations.

This Standard facilitates implementation of requirements in DOE Order 458.1 *Radiation Protection of the Public and the Environment* (hereafter O458.1 or the Order) [DOE2013], by providing criteria and guidance for DOE accelerator facilities to develop and implement site-specific programs (including management, technical, and operational aspects) for the radiological clearance and release of personal property. Implementing this Standard should enable DOE accelerator facilities to satisfy applicable requirements in O458.1 for clearance and release of personal property. The provisions of this Standard are also intended to implement the DOE guidance on improving the site's monitoring and release practices [DOE2001].

Use of this Standard is not mandatory. Where a site's property clearance and release program deviates from the requirements in this standard, documentation providing equivalent assurance of compliance with the requirements in the O458.1 shall be available and approved by DOE. It is recommended that deviations from this Standard and equivalent assurance be made part of the technical bases documentation for clearance protocols.

Scope

In this Standard, following the definition as provided in O458.1, *clearance* is the removal of personal property that contains or may contain residual radioactive material from DOE radiological control. *Release* is not specifically defined and is used here to refer to that material has gone through the clearance process and is released (off-site or on-site) from radiological control.

This Standard addresses both surface and volumetric radioactivity issues. Applicable requirements and criteria should already be in place for clearance of materials with potential for surface radioactivity. Therefore, emphasis is placed on volumetric radioactivity due to potential for volumetric activation in accelerator facilities.

This Standard addresses materials and equipment (M&E) defined as personal property in O458.1. Further, this standard applies to M&E released from a radiological area in accordance with 10 CFR Part 835.1101. The terms materials, M&E, personal property and property are used interchangeably in this Standard. This Standard applies to, but is not limited to, solid materials comprised mainly of common metal such as beam pipes, magnets, collimators, beam dumps, RF cavities and waveguides, detector components, electronics, power supplies, cables, racks, supporting stands, shielding, etc.

The approach described in this Standard and its appendices may be applicable to other types of solid materials such as concrete shield blocks or uncommon metals such as niobium. However,

sites shall: document their process knowledge of equipment, produce site-specific technical bases documentation for clearance of such materials, and provide the characteristics and parameters that are necessary for clearance determinations. Site specific documentation should specify when these site and material characteristics may not apply.

This Standard addresses programs and protocols for off-site releases without restriction on use (i.e., unrestricted release) as well as releases that may involve use restrictions on the property (i.e., restricted release).

This Standard applies to special projects that release a large amount of materials, e.g., from a decommissioned or non-operating facility (non-routine releases) as well as facility operations that release a smaller amount of materials from an operating facility (routine releases).

This Standard could be used as a model for non-accelerator facilities particularly those that have volumetric activation issues. If this Standard is used as a template in determining applicability, some of the process knowledge, technical basis and measurement methods in this Standard may no longer be applicable. Examples of non-applicable volumetric activation conditions would be presence of alpha emitters or dominance of pure beta emitters. These situations shall nonetheless be addressed in any approved clearance approach.

This Standard does not apply to real property, Naturally Occurring Radioactive Material (NORM), or Technologically Enhanced NORM (TENORM). Although the measurement methods in this Standard do not apply to non-solid materials (liquid, gas, powder, etc.), the approach and criteria described herein may be used to develop clearance protocols for such materials, provided such protocols are supported by appropriate technical bases and measurement methods, which may be different from those for solid materials.

Throughout this Standard, the word “shall” is used to denote requirements essential to satisfy the intent of this Standard. “Should” is used for recommendations intended to ensure quality objectives in the Order are met. “May” is used to denote permission, but not a requirement or recommendation.

2. Introduction

For meeting the O458.1 personal property clearance and release requirements, this Standard provides an approach to allow sites to establish personal property clearance and release programs as they relate to the unique radiological conditions encountered at accelerator facilities.

The site's material clearance and release program shall include, at a minimum, the clearance protocols and the technical basis that can be used to justify and support the clearance protocols. The clearance protocols in this Standard include three elements: 1) clearance criteria, 2) process knowledge, and 3) measurement methods. Process knowledge and/or measurements shall be used to demonstrate that the materials released meet the selected clearance criteria. Materials that have been characterized based on process knowledge evaluation or measurements to meet one of the clearance criteria may be released, provided that the appropriate DOE approval is in effect for the criterion used for the release.

General requirements of the three elements of the clearance protocols (criteria, process knowledge and measurements) and the accompanying technical basis are described in sections 2.1 through 2.4. Chapter 3 provides more details for guidance on clearance protocols and technical basis.

The appendices to this Standard provide general process knowledge, technical bases and measurement methods for volumetric activation in solid metals that may be applicable to, and utilized by, the accelerator facilities that have similar beam parameters and operational modes.

Clearance Criteria

O458.1 prescribes a Total Effective Dose (TED) constraint of 1 mrem (0.01 mSv) above background in any calendar year for each specific clearance of personal property with potential residual radioactivity. Based on a dose constraint of 1 mrem/y, the ANSI N13.12-2013 Standard, *Surface and Volume Radioactivity Standards for Clearance*, [ANSI2013] derives and establishes the screening levels (SLs) for the clearance of materials that contain or may contain residual levels of surface or volume radioactivity of radioactive materials (in terms of radioactivity per unit surface area or mass) from radiological control which generally satisfies the criteria set forth in the Order.

For clearance of personal property with potential surface activity, O458.1 allows previously approved guidelines and limits (such as activity guidelines or approved site-specific limits) to be used until updated or replaced. Section 3.1 provides details for guidance on the clearance criteria for surface radioactivity. Appendix A provides rationale and guidance for the use of ANSI surface SLs as the DOE pre-approved Authorized Limits (ALs) for surface radioactivity.

For clearance of personal property with potential volumetric radioactivity, O458.1 requires the process of DOE Authorized Limits be followed, but no specific activity values are provided. However, the Authorized Limit shall be based on the applicable dose constraint. For clearance of property with potential volumetric radioactivity, this Standard adopts the following 3-tiered clearance criteria that can be related to ANSI N13.12-2013 volume SLs:

1) A criterion of indistinguishable from background (IFB) at a level lower than the SLs,
2) A criterion equal to the SLs, and
3) A criterion higher than the SLs.

Section 3.2 provides details for guidance on the 3-tiered clearance criteria for volumetric radioactivity. Appendix A provides rationale and guidance for the use of ANSI volume SLs as potential DOE pre-approved Authorized Limits for volumetric radioactivity.

Use of Process Knowledge

Evaluation of potential surface activity and volumetric activation based on process knowledge is an essential element of a site's clearance protocol. Process knowledge may include mechanisms of radioactive contamination or activation, characteristics of materials and induced radionuclides, historical facility and accelerator operation parameters and conditions, history of use of the property, etc. Documented process knowledge to determine if materials potentially contain residual radioactivity may be used as a basis for clearance decision.

Section 3.3 provides details for the use of process knowledge to support clearance protocols. Appendix B provides general process knowledge that may allow the evaluation of induced radioactivity and its characteristics in accelerator facilities.

If there is sufficient process knowledge to establish that materials have no potential for either surface or volumetric radioactivity, the materials may be released without subjecting them to the comprehensive clearance process, including the measurement methods, described in this Standard. Examples include materials that have always stayed in areas in which there is no physics potential for volumetric activation (such areas may include klystrons, x-ray radiation generating devices, synchrotron radiation and free-electron-laser photon beam lines that operate at energies less than the activation thresholds) or surface contamination (such areas may include areas not classified as Contamination Areas). Such process knowledge shall be included in a site's technical basis or clearance documentation. In some cases, some limited level of radiological measurement may be useful to validate process knowledge evaluations.

Measurement Methods

If process knowledge cannot demonstrate that property does not contain residual radioactivity, radiation measurements shall be conducted to support the clearance decision. Measurement methods (instruments and techniques) shall be adequate for the radionuclides of interest and the selected clearance criteria.

Sections 3.4 and 3.5 provide guidance on the measurement methods for surface and volumetric radioactivity, respectively. Section 3.6 gives guidance for the conduct of measurements in a graded approach by considering both process knowledge and the recommendations in the *Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual* (MARSAME) [MAR2009].

Technical Basis

The site's material clearance and release program shall include technical basis documentation that can justify and support the site's material clearance protocols. The technical basis documentation shall clearly identify what, why and how property may be radiologically impacted (i.e., surface contaminated or volumetrically activated) and the induced radionuclide characteristics, define clearance criteria, as well as describe measurement methods.

Section 3.7 provides more guidance for technical basis documentation. Appendix C and Appendix D provide the technical bases for volumetric activation that may be used by high energy electron and proton accelerator facilities, respectively, that operate in similar beam parameters and modes.

Appendix E provides the technical basis and examples of appropriate measurement methods for volumetric radioactivity and the estimations of associated detection capabilities.

These technical bases may be utilized by accelerator facilities that have similar beam parameters and operational modes.

3. Material Clearance Protocols

As a minimum, the site's material clearance protocols shall include the following three elements:

1) Clearance criteria that are approved by DOE:
    a. For surface activity, the criteria as prescribed in O458.1 (expressed in surface activity in units of dpm / 100 cm^2),
    b. For volumetric activation, 3-tiered criteria that can be related to ANSI N13.12-2013 volume SLs (expressed in terms of volumetric activity in units of pCi/g),
2) Evaluation of potential surface activity and volumetric activation of materials based on process knowledge. The evaluation includes identifying conditions or operations that may potentially contaminate or activate materials. The evaluation may be used to set the requirements and a graded approach for the measurements, and
3) Adequate measurement methods, which include measurements for surface or volumetric radioactivity, as well as additional confirmatory measurements that may be warranted.

Process knowledge evaluation or measurements may be used to demonstrate that one of the clearance criteria is satisfied and materials may be released. In many cases, process knowledge itself is not sufficient to establish the clearance decision. In that case, materials shall be measured with appropriate instruments and techniques for surface or volumetric radioactivity to determine that the selected clearance criteria are met.

Clearance Criteria for Surface Radioactivity

Per O458.1, previously approved guidelines and limits may be used for clearance of personal property with potential surface activity. One of the previously approved surface contamination guidelines are from 10 CFR 835 and they are summarized in the Appendix A.

Based on a dose criterion of 1 mrem/y to the public, ANSI N13.12-2013 establishes the volume and surface SLs for volumetric and surface radioactivity levels, respectively, for clearance of personal property. ANSI N13.12-2013 surface activity SLs are summarized in Table 1. Note that ANSI N13.12 derived the surface SL from the corresponding volume SL based on a mass-to-surface ratio of 1 g/cm^2 with the understanding that the actual mass-to-surface ratio of the released material can be used to derive the surface SLs.

The 10 CFR 835 surface contamination guidelines may be used until the ANSI N13.12-2013 surface activity SLs are approved by DOE as pre-approved Authorized Limits for clearance of materials with potential surface contamination.

Appendix A provides rationale for the endorsement of ANSI N13.12-2013 surface activity SLs and volume activity SLs as the DOE pre-approved Authorized Limits.

Release based on a clearance criterion above the ANSI N13.12-2013 surface SLs shall follow the DOE process for the approval and use of Authorized Limits.

Table 1: ANSI N13.12-2013 screening levels (SLs) for surface activity and volumetric activity.

Radionuclide Groups	**Surface SLs (dpm /100 cm^2)**	**Volume SLs (pCi/g)**
Group 1: High-energy gamma, radium, thorium, transuranics, and mobile beta-gamma emitters (e.g., ^{22}Na, ^{46}Sc, ^{54}Mn, ^{56}Co, ^{60}Co, ^{65}Zn, ^{125}Sb, ^{134}Cs, ^{152}Eu, ^{154}Eu)	600	3
Group 2: uranium and selected beta-gamma emitters (e.g., ^{57}Co, ^{58}Co, ^{59}Fe, ^{113}Sn, ^{124}Sb)	6,000	30
Group 3: General beta-gamma emitters (e.g., ^{7}Be)	60,000	300
Group 4: Low-energy beta-gamma Emitters (e.g., ^{3}H, ^{45}Ca, ^{63}Ni)	600,000	3,000
Group 5: Low-energy beta emitters (e.g., ^{55}Fe)	600,000	30,000

Clearance Criteria for Volumetric Radioactivity

Table 1 summarizes the ANSI volume screening levels for radionuclides in 5 groups: 3 pCi/g for Group 1 radionuclides, 30 pCi/g for Group 2 radionuclides, 300 pCi/g for Group 3 radionuclides, 3,000 pCi/g for Group 4 radionuclides, and 30,000 pCi/g for Group 5 radionuclides. Groups 4 and 5 radionuclides generate much lower dose risk (at least a factor of 1,000 lower than the Group 1 radionuclides) because they emit only low-energy beta or gamma rays.

Because no specific clearance guidelines are provided by O458.1 for volumetric radioactivity, this Standard prescribes 3-tiered clearance criteria related to ANSI N13.12-2013 volume SLs that may be used by DOE sites:

1) A criterion of indistinguishable from background (IFB) at a level lower than the SLs,
2) A criterion equal to the SLs, and
3) A criterion higher than the SLs.

The use of the 3-tiered clearance criteria and the associated DOE approval process for clearance and releases of materials with potential volumetric radioactivity are summarized in Table 2. For all clearance criteria, approval by DOE Field Element for the site's material clearance and release program is required. The technical basis and measurement methods to support the program should be peer reviewed.

Table 2: Three-tiered clearance criteria for materials with potential volumetric radioactivity.

Clearance Criterion	**Basis, Requirements, Usage**	**Approval of Clearance Criterion and Release Plan**
IFB [1]	DTs < SLs. [2,4] Measurements are IFB (material has no detectable radioactivity other than background). IFB releases satisfy paragraph 4.k(3)(a) of O458.1. IFB is the preferred clearance criterion. No radiological control is warranted and is already ALARA.	Approval for releases is not needed if the releases are covered by the scope of the material clearance program.
ANSI N13.12-2013 SL [4]	DLs < SLs. [3,4] Measurements meet SLs (derived based on the ANSI dose criterion of 1 mrem/y). ALARA analysis is required because materials with detectable radioactivity may be released. Non-IFB release can satisfy paragraph 4.k(3)(b) of O458.1 under certain conditions.	SLs are subject to DOE pre-approved AL process (satisfy O458.1 dose constraint of 1 mrem/y). [5] DOE Field Element approval of each release plan, which includes a qualitative ALARA analysis.
> SL	Follow DOE AL process to obtain approval for release under a site-specific Authorized Limits. Mainly for restricted release.	DOE approval of the clearance criterion as AL. DOE Field Element approval of each release plan, which includes a quantitative ALARA analysis and notification of DOE program elements and EHSS consistent with O458.1. [6]

[1] IFB: Indistinguishable From Background.
[2] DTs: Detection thresholds of the IFB measurement methods for proxy radionuclides.
[3] DLs: Detection limits of the measurement methods for proxy radionuclides.
[4] SLs: ANSI N13.12-2013 volume Screening Levels for proxy radionuclides.
[5] ALs: DOE Order 458.1 Authorized Limits.
[6] EHSS: Environment, Health, Safety and Security.

3.1.1 IFB Clearance Criterion

The IFB criterion is consistent with the O458.1 paragraph 4.k.(3)(a) in releasing property not containing residual radioactive material. The IFB clearance criterion is applied to property that is expected to have no residual volumetric radioactivity. This expectation shall be confirmed by

clearance measurements with results that are indistinguishable from ambient background radiation signals.

The IFB level of a measurement method depends on the detection capability. Sites using different measurement methods in different background levels can have different IFB levels. Therefore, when IFB is used as the clearance criterion, the measurement methods shall have sufficient detection capability (which is less than the ANSI N13.12-2013 volume SLs) to provide reasonable assurance that M&E released on the basis of this criterion satisfy the 1 mrem/y dose constraint in O458.1. Detection capability should be evaluated with conservatism (e.g., using the minimum sensitivity or the highest background for all measurement conditions); otherwise the uncertainty should be evaluated to show that the detection capability is below the ANSI volume SLs with uncertainty included.

The measurement methods described in the Section 3.5 and Appendix E of this Standard are designed to meet these objectives. Release with IFB clearance criterion requires only that radioactivity is *not detectable*, using appropriate instruments and techniques, and does not specifically invoke a quantitative radioactivity determination. In this case, detection thresholds (DTs), defined in Appendix E, for proxy radionuclides shall be less than the ANSI N13.12-2013 volume SLs. The measurement methods and its detection threshold evaluation shall be documented in the site's technical basis.

The IFB clearance criterion is the lowest and most restrictive criterion, but release based on IFB criterion requires the least justification and DOE review and is the preferred clearance criterion unless other considerations warrant application of the other more rigorous approaches. ALARA analysis is not required for releases using the IFB criterion. Measurements and clearance based on IFB criterion are more straight-forward to be conducted than those with non-IFB clearance criteria. Materials meeting the IFB clearance criterion need not be treated as radioactive materials and are not subject to radiological control and may have unrestricted release. DOE guidance and accelerator community practices have established that materials with no detectable radioactivity, other than background, are acceptable for release when clearance is conducted using approved procedures, including adequate measurement methods.

The site's material clearance and release program may include protocols and practices for non-routine and routine releases, and each release plan of a material release project or activity may apply to a batch of releases that has one of the three clearance criteria. The DOE process that evaluates and approves requests from DOE sites for material clearance and releases shall be followed. Each site should prepare the clearance and release proposals in accordance with DOE guidance and submit them to the DOE Field Element Manager. The IFB clearance process is to demonstrate that no detectable residual radioactive material is present and hence the property conforms to the clearance requirements of O458.1 subject to DOE approval of the protocols. Approval for each release using the IFB criterion is not needed if the releases are covered by the scope of the site's material clearance and release program that is approved by DOE. The independent verification requirements in O458.1 still apply. DOE is responsible for independently verifying that the IFB is being properly implemented.

3.1.2 ANSI N13.12-2013 SL as Clearance Criterion

The second tier clearance criterion is the ANSI N13.12-2013 volume radioactivity screening levels. For main gamma-emitting radionuclides of interest at accelerators, the ANSI N13.12-2013 volume SLs represent volumetric radioactivity levels that may be measurable above ambient background with sensitive, portable survey instruments. Therefore, clearance based on ANSI SL criterion may allow releases of some materials with detectable radioactivity (i.e., the material is not IFB). The non-IFB criterion is consistent with the O458.1 paragraph 4.k.(3)(b) in releasing property containing residual radioactive material. This criterion is consistent with the use of surface activity guidelines (in units of dpm/100 cm^2) that allows for unrestricted releases of materials with small amounts of surface radioactivity above background to be present, as long as the dose consequence is no more than the dose constraint of 1 mrem/y.

When ANSI volume SLs are used as the clearance criterion, the measurement methods shall have sufficient detection limits (DLs as defined in Appendix E) which is less than the ANSI N13.12-2013 volume SLs.

Because materials released with the ANSI N13.12-2013 SL criterion may contain detectable radioactivity, an ALARA analysis of each release plan associated with the SL clearance criterion shall be conducted in accordance with O458.1. Since the ANSI SLs already demonstrate compliance with the O458.1 dose constraint of 1 mrem/y and have included a generic consideration of the ALARA process, this ALARA analysis should utilize a graded process that optimizes releases of radioactive material to the environment and exposure to the work force and to members of the public. The level of effort in evaluating alternative operations, processes, and other measures should be commensurate with the potential benefit of the dose reduction. Release based on the SL criterion should consider societal, environmental, technical, economic and public policy considerations, and may include restrictions or conditions on future use of the material.

Approval by DOE Field Element is required for each release plan based on the ANSI volume SL criterion including associated ALARA analysis. If appropriate, a site's pre-existing ALARA analysis may be used to support the release plan rather than developing a new one specifically for the release plan. DOE-HDBK-1215-2014 provides further guidance on optimizing radiation protection in support of O458.1 requirements.

3.1.3 Clearance Criterion Higher Than ANSI N13.12-2013 SLs

Release based on a clearance criterion above the ANSI N13.12-2013 SLs shall follow the DOE process for the approval and use of Authorized Limits. In general, clearance criteria higher than the SLs should only be used for restricted release or designated use where application of applicable controls provides additional assurance that the O458.1 criteria are met at the higher concentrations. However, there are situations where physical or chemical properties of the specific M&E ensure DOE dose constraints and ALARA considerations can be met without use restrictions and option 3 may be used for such property.

Process Knowledge and its Application

Appendix B provides guidance on what general process knowledge is and may allow sites opportunities to evaluate induced radioactivity and its characteristics in accelerator facilities. Appendix B may be useful to those non-accelerator facilities working to develop Authorized Limits for property with potential for volumetric radioactivity as well.

General process knowledge such as beam particle type and beam energy (described in more details in Appendix B2) is applicable to accelerator facilities with similar characteristics and operational modes (in terms of producing induced radioactivity).

Specific process knowledge includes operational information such as beam energies and beam losses that are relevant to the specific facility, location and use of equipment within the facility as well as any temporary or long term storage decisions. Specific process knowledge should be used to support the clearance measurements and decisions.

General and specific process knowledge, though non-quantitative in many cases, may still be used to identify the areas, conditions and operations that potentially can contaminate or activate materials, as well as to set a graded approach for the measurement methods, including the need of confirmatory measurements.

When process knowledge is used in part as a basis for the clearance process, a documented evaluation process for applying process knowledge to determine if property potentially contains residual radioactivity shall be established. The process shall include evaluation of facility records for operations or activities that may have the ability to potentially contaminate or activate the materials.

This evaluation shall be documented. Property management requires that handling, use, storage and release decisions are predicated on knowledge of the property and its use. All known hazards are to be identified and any potential release decision should address notification and disposal considerations.

If process knowledge cannot demonstrate that property does not contain residual radioactivity, radiological measurements shall be conducted to supplement process knowledge evaluations.

If not supplemented by radiological measurements, process knowledge evaluations shall be adequate to determine:

1) Whether the property has ever been used for radiological activities or in areas that could have resulted in the presence of residual radioactive material within or on the property, or
2) Whether property formerly containing residual radioactive material has been decontaminated and demonstrated to meet the O458.1 Authorized Limits (ALs), and has not been used in a manner or in areas that could have resulted in the re-contamination of the property.

Measurement Methods for Surface Radioactivity

Prior to being released, property shall be surveyed to determine that its surface activity meets the ANSI N13.12-2013 surface screening levels.

Property with the potential for surface radioactivity shall be surveyed using instruments and techniques with detection capability that is less than the ANSI N13.12-2013 surface activity screening levels or the O458.1 Authorized Limits for surface activity. Detection capability should be evaluated with conservatism (e.g., using the minimum sensitivity or the highest background for all measurement conditions); otherwise the uncertainty should be evaluated to show that the detection capability is below the ANSI surface SLs with uncertainty included.

The measurement methods for surface activity for accelerator facilities are the same as the standard methods established throughout the DOE complex, e.g., surface scanning using a pancake GM and/or surface swipe samples for gross beta-gamma counting. Many commercially available beta-gamma survey meters for surface activity measurements have detection capabilities that are lower than the O458.1 Authorized Limits for surface activity.

Measurement Methods for Volumetric Radioactivity

ANSI N13.12-2013 has placed gamma radionuclides such as ^{22}Na, ^{54}Mn, ^{60}Co, ^{65}Zn and ^{152}Eu in Group 1, because they emit high-energy and high-yield gamma rays. For the measurements of potential volumetric radioactivity in this Standard, many of these gamma radionuclides may serve as "proxy" radionuclides, because they are easy to be measured with common survey instruments. Group 4 (e.g., ^{3}H and ^{63}Ni) and Group 5 (e.g., ^{55}Fe) radionuclides emit only low-energy beta or gamma rays, which are hard to measure with common survey instruments. These hard-to-measure radionuclides (Groups 4 and 5) have volume SLs that are at least 1,000 times higher than that of proxy radionuclides in Group 1.

Measurements for proxy radionuclides are necessary and may be sufficient (as justified by the site's technical basis document), to characterize levels of all radionuclides of interest (including the hard-to-measure radionuclides).

The technical bases for volumetric radioactivity from activation at electron and proton accelerators are described in Appendices C and D, respectively. The key conclusions include the following:

1) Many radionuclides can be produced by activation from prompt radiation, but most have short half-lives and decay quickly, within days or weeks. The most abundant radionuclides are those with a half-life of the order of the irradiation time, e.g., months to years (such prolonged time is more relevant to material release projects). *This justifies the measurements of radionuclides with long half-lives.*
2) Radionuclides with either their atomic number or mass number lower than their parent nuclides can be produced (except radionuclides produced from thermal neutron absorption). However, there are no productions of radionuclides that emit alpha particles. *This justifies the measurements of beta-gamma radionuclides.*

3) Radionuclides that are hard to measure (e.g., Groups 4 and 5 radionuclides in Table 1) are in most cases accompanied by proxy radionuclides that can be measured easily (e.g., Group 1 radionuclides). Measurement of the proxy radionuclides can be used to estimate levels of the hard to detect radionuclides that cannot be measured in most cases.
4) Induced radioactivity profile in an item is volumetric and the maximum radioactivity is near the surface that faces the beam loss point. *This justifies using surface (scanning or fixed position) measurements for volumetric radioactivity*.
5) Many sensitive, portable instruments that are commercially available (e.g., scintillator-based survey meters) have capability to detect the proxy radionuclides at levels that are less than the corresponding ANSI N13.12-2013 SL values. *This justifies the use of sensitive, portable survey meters for material clearance measurements*.

The measurement of proxy radionuclides and maximum surface activity are two essential elements of the process that serve as the technical bases for the surface survey method for volumetric radioactivity recommended by this Standard:

1) Measurements for proxy radionuclides are necessary and sufficient.
2) Surface surveys (i.e., measurements over an object's surfaces for volumetric radioactivity) for proxy radionuclides using portable instruments and techniques with high sensitivity in an environment with low background (i.e., sufficient detection capability to satisfy the clearance criterion) are adequate and sufficient.
3) Surface surveys (scanning or fixed-position measurements) of all surfaces of an object, regardless of its shape and size, are sufficient. Measurement of the surface of an object that faces the beam loss point is also acceptable, if the beam loss geometry is known.

3.1.4 Measurement Methods

Volumetric radioactivity measurements are to be conducted when potential volumetric radioactivity cannot be excluded by process knowledge. The measurement methods shall utilize instruments with sufficient sensitivity in a sufficiently low ambient background environment to achieve the required detection capability, i.e., detection thresholds (DTs) for the IFB criterion or detection limits (DLs) for other clearance criteria, for proxy radionuclides.

When the IFB criterion is used, the measurement methods shall have DTs less than the ANSI N13.12-2013 SLs. When a non-IFB criterion is used, the measurement methods shall have DLs that are less than the ANSI N13.12-2013 SLs.

The detection capability (DT or DL) shall be estimated and documented based on the instrument's sensitivity to proxy radionuclides in the materials and the acceptable background in the measurement environment. The sensitivity depends on factors such as the material type, types of radionuclides, volumetric radioactivity distribution, the detector-sample geometry, and the measurement mode such as scanning or fixed-position, etc. The minimum sensitivity and appropriate background level should be used such that the detection capability can be estimated conservatively.

Many sensitive, survey instruments that are commercially available have detection capabilities in typical ambient environment that are less than the ANSI N13.12-2013 volume SL values.

Appendix E provides examples of appropriate instruments and techniques and the estimations of associated detection capabilities (DT or DL). It shows that common scintillator-probe survey instruments can have detection capabilities that are less than the ANSI SLs. For example, when using the 1"x1" NaI detector in the surface scanning mode to measure proxy radionuclides, the DLs can be less than the ANSI volume SLs of 3 pCi/g for Group 1 radionuclides. Appendix E may be used by the site as a guide to estimate the detection capabilities associated with the site's measurement methods.

Surface surveys (measurements over an object's surfaces for volumetric radioactivity) may be sufficient such that measurements for surface contamination are not needed, when it can be demonstrated that the surface surveys for volumetric radioactivity are as protective as measurements for surface contamination. For example, if process knowledge shows that surface contamination is minimal and the induced radiation field on an object's surface is dominated by the volumetric radioactivity of the proxy radionuclides (e.g., there are no radionuclides emitting alpha, pure beta or low-energy photons), surface surveys for volumetric radioactivity alone may demonstrate that both ANSI volume and surface SLs are satisfied.

For surface surveys, either scanning over the whole surface or fixed-position measurements may be conducted, depending on process knowledge for the radioactivity profile and gradient. For example, vacuum chamber and magnets which are close to beam loss points (e.g., collimators or dumps) can have high-gradient volumetric radioactivity profile and, thus, may need to be surface-scanned. On the other hand, the volumetric radioactivity profiles in the items that are not close to any beam loss points (e.g., shielding, cables, and supporting structures) are slowly varying and, thus, fixed-position measurements may be appropriate and sufficient.

If the material to be surveyed has inaccessible surfaces which may be shielded from an effective survey by overlying material, the material should be disassembled to allow access to those surfaces to be surveyed, unless other means are used to measure the inaccessible surface or process knowledge can demonstrate that the inaccessible surfaces have lower radioactivity levels. If inaccessible surfaces cannot be accessed and Process Knowledge is not adequate for material release, it may be prudent to not release the item(s).

Instrument response (e.g., in cpm) is proportional to the volume (size) of the item being surveyed and would reach a maximum when the volume reaches a certain size. The detection capability becomes worse when volume of the item is reduced. Therefore, to achieve a better detection capability (i.e., lower DT or DL), the item being surveyed should be of a sufficient volume. Small items such as bolts or thin wires, which also have similar activation potential, should be surveyed together as a group to achieve a sufficient volume.

Averaging is inherent to the process of measurements or evaluations for surface activity (dpm averaged over a surface area of 100 cm^2) or volumetric activity (pCi averaged over the mass of interest in grams). For volumetric activity measurements, ANSI N13.12-2013 prescribes that: 1) the average should be done over a total volume or mass not to exceed 1 m^3 or 1,000 kg, and 2) for smaller items, the average shall be done over the entire mass. These ANSI guidelines should be followed.

Attention should be paid to a potential "mass effect" when releasing a large amount of material. Mass effect occurs when many items, each with an IFB or very low radioactivity levels, can collectively generate a higher radiation signal. This condition may occur in particular when the collection of material consists of relatively small items (see discussion above regarding measurement sensitivity). When material handling processes produce such aggregate collections, the site's QA practices should be tailored to address the mass effect. Methods such as bulk or portal monitoring, or additional surveys on aggregate collections of such material may be used to address this issue.

3.1.5 Confirmatory Measurements

The surface survey methods may be supplemented by one of the confirmatory measurements (as warranted), which should have comparable or lower DLs than the surface survey methods.

Confirmatory measurements should be conducted in a graded approach, which is based on process knowledge and field survey results. Confirmatory measurements may include gamma spectrometry in the field, laboratory sample analysis, portal gate monitor, or bulk monitoring systems.

The guidelines for the selection and use of confirmatory measurements include:

1) For large objects, gamma spectrometry at strategic surface locations or laboratory measurements of representative core samples may be used.
2) For individual small items, additional surface surveys may be sufficient (e.g., a slower scanning or a stationary survey with a longer time can be used to lower the DL).
3) For small items (such as cables, wires or bolts) grouped in a batch or bin, gamma spectrometry or a bulk monitoring system may be used.
4) The frequency of confirmatory measurements should be higher (e.g., > 10%) when materials have higher potential to be mischaracterized by the surface survey methods.

A protocol should be developed when confirmatory measurements identify materials with detectable radioactivity, which are not detected by the surface surveys. The protocol should include the following: 1) a review of all related field survey and confirmatory measurements, and 2) investigation of the causes of the situation and a review of the extent of the condition. In any case, materials shall not be released with detectable radioactivity if the IFB clearance criterion is used.

Graded Approach and MARSAME Considerations

The details of measurement methods and process shall be documented in operating procedures and should be consistent with consensus measurement standards and methods for release of personal property such as ANSI N13.12-2013 or MARSAME. Guidance in MARSAME and other references should be used to establish the extent of measurement process and coverage required at a given site. However, every provision in MARSAME is not necessarily applicable to all clearance measurement activities. For instance, the MARSAME process includes a step

involving preliminary surveys, which may not be necessary for facilities with mature programs using clearance protocols consistent with this Standard.

Process knowledge evaluation may allow for "area zoning" and "component identification," which may be used for planning and optimizing the measurement process in a graded approach in the field. This Standard applies and extends the MARSAME concept and terminology for classifying areas or M&E as either impacted or non-impacted.

Areas with no credible potential for surface or volumetric radioactivity above background levels may be classified as non-impacted areas. Non-impacted areas may include the areas outside accelerator housings, klystron areas, and low-energy photon beam line areas. Materials from such areas are not normally candidates for clearance measurements. However, periodic surveys and assessments of such areas and materials may be a useful element of a site's clearance process.

Areas with some potential for surface or volumetric radioactivity should be classified as impacted areas. Such areas include areas within accelerator housing or radiological areas (particularly Contamination Areas). Impacted areas may be further divided into sub-categories depending on the degree of potential surface radioactivity or volumetric activation (based on process knowledge or historical radiological surveys). Sites should evaluate impacted areas and design the site's clearance and release program using an appropriate graded approach. M&E with the highest potential to exceed clearance criteria (in MARSAME, such materials are called Class 1) should receive the greatest scrutiny when establishing clearance procedures. For example, areas near normal beam loss points such as beam dumps, collimators and targets normally have the greatest potential for surface or volumetric radioactivity and should receive the highest degree of clearance survey effort (e.g., 100% of items from such areas would require surveys). Impacted areas with lower potential for activation or surface radioactivity may not need 100% survey coverage.

In addition to area classification, specific components may be candidates for designation as either impacted or non-impacted. These designations should be determined on a site-specific basis and the justifications for the designations should be described in the site's technical basis documents. Note that an area may have different designations with respect to surface or volumetric radioactivity.

Program and Technical Basis

The provisions of 10 CFR 835 cover the release of materials from radiological areas (particularly Contamination Areas) to controlled areas on-site. The off-site releases of materials need to follow the requirements in O458.1 and this Technical Standard. The off-site release program should be consistent and have no gaps with the on-site material control program and requirements of both 10 CFR 835 and O458.1 need to be satisfied.

The site's material clearance and release program should be an integrated element of the site's environmental and public radiation protection programs (that satisfy O458.1) and occupational

protection programs (that satisfy 10 CFR 835). The technical, management and operational aspects of the site's material clearance and release program shall be established and documented, and should include the release of materials from controlled areas.

A technical basis shall be developed and documented to justify and support the site's material clearance and release program. The documentation shall clearly identify the areas and operations that potentially can contaminate property, define clearance criteria, and describe measurement methods. The technical basis documentation should at least answer the following questions: "What radionuclides can be produced from activation?", "What radionuclides are of interest and what are their characteristics?", "How can the proxy radionuclides be measured?", and "What are the detection thresholds and detection limits of the measurement methods for proxy radionuclides?"

The technical basis documentation should include the following:

1) Process knowledge, including descriptions of the facility, M&E and information (e.g., beam parameters, operation modes and beam losses) that are relevant to the estimation of potential surface activity and volumetric activation, the physics and characteristics, as well as the potential, for surface activity and volumetric radioactivity,
2) The selected clearance criteria,
3) Measurement methods,
4) Consideration of MARSAME guidance, tailored to the site using a graded approach,
5) Descriptions or examples of impacts and impacted areas,
6) The conditions under which confirmatory measurements are warranted and conducted,
7) Estimation of the detection capability (detection thresholds or detection limits) of the measurement methods for proxy radionuclides,
8) Any bounding conditions that may limit the application of the technical basis (including measurement methods). Examples may include special types of materials that the induced radioactivity characteristics have not been documented, existence of radionuclides with short half-lives, measurement conditions (e.g., minimum sample size/volume required, maximum scan speed and maximum background allowed, and locations of fixed-position measurements), and
9) An integrated ALARA process evaluation program.

Appendices C and D summarize the technical bases for volumetric activation and volumetric radioactivity profiles for electron and proton accelerator facilities, respectively. Two key characteristics, "proxy radionuclides" and "surface maximum", are described to justify and support the measurement methods described in this Standard. These technical bases may be used by DOE sites with similar accelerator and operation characteristics to develop their site-specific technical basis documentation and to evaluate and define areas and operations that can potentially activate materials, and the appropriateness of the measurement methods.

4 Documentation and Reporting Requirements

The site's material clearance and release program is supported by having documentation that is complete, of high quality, and readily retrievable. General program documentation includes site operation history and process knowledge, program manuals and procedures, measurements and instrumentation, training, etc. In addition to general program documentation, recordkeeping also includes release specific documentation which pertains to each release of individual items or batches of material.

The following documents and records related to the material clearance and release program shall be produced, reviewed for quality and archived.

General Program Documentation:

- Relevant regulations and standards,
- Site-specific material clearance and release program documentation,
- Documents concerning accelerator and facility parameters and operation history,
- Documents concerning evaluations for both surface activity and induced volumetric radioactivity,
- Procedures and records for Radiological Area classification, radiation survey and posting,
- Clearance criteria selected,
- Descriptions of concept and approach of proxy radionuclides and hard-to-measure radionuclides,
- Procedures for clearance measurements, particularly for proxy radionuclides,
- Detection capability (detection thresholds or detection limits) for proxy radionuclides,
- Procedures and records for instrument calibration and testing,
- Procedures and records for personnel training and qualification for the use of instruments and the conduct of measurements,
- Procedures and records for material storage on-site prior to release, and
- Stakeholder communication plan.

Clearance and Release Specific Documentation:

- Specific process knowledge used to support or supplement clearance measurements, i.e., beam parameters and operating conditions, material usage, etc.,
- Historical survey records of the property to be released,
- Description of facility and item(s) surveyed,
- Radiological survey report: clearance measurement conditions and results, including date and location of measurements, instrument models and serial numbers, calibration date, background signals, response check results, and measurement conditions and results,
- Confirmatory measurement reports such as gamma spectroscopy and portal gate monitor records, if applicable.
- Comments and other correction factors that have effects on the results,
- Date and name of surveyor, as well as the name and signature of the reviewer/approver,
- Records of release or disposal decisions (e.g., no potential reuse) and authorization, and
- Records of stakeholder communication (if any), including reporting to management and regulators.

When non-IFB clearance criteria are used, the records shall include:

1) The types and quantities of residual radioactive material within the property,
2) The surface activity of the items to be released,
3) Results of an evaluation of residual radioactive material for property with surfaces that are difficult to access for surveys of surface or volumetric radioactivity, if any, and
4) Documentation demonstrating that any residual radioactive material within or on the property is in compliance with applicable specific DOE Authorized Limits.

The documents and records shall be managed and maintained for accountability and in a manner that is auditable.

Large items (such as accelerator magnets and detector components) should be individually identified, labeled, surveyed and recorded. Small items (such as cables, wires, fasteners, breakage, support brackets, etc.) may be surveyed individually or as a group, collectively identified and labeled, and collectively recorded.

A database should be used for large-scale material clearance and release projects to archive the records of process knowledge, item inventory, survey results, on-site reuse or storage, and off-site disposal and release.

Information regarding materials released shall be reported to DOE in the site's Annual Site Environmental Report (ASER) as required by O458.1. When IFB clearance criterion is used, the types and quantities of material shall be reported. When non-IFB clearance criteria are used, the types and quantities of residual radioactive material, as well as their specific radioactivity and total radioactivity for the radionuclides, shall also be reported.

5 DOE Independent Verification, Stakeholder Communication

DOE Independent Verification

The requirements of the DOE oversight and independent verification in O458.1 shall be followed. Independent Verification should include a review of the written approvals for radiation protection documentation and the site's material clearance and release program (the latter may be part of the approval of the site's environmental and public protection program to satisfy O458.1). The adequacy of property clearance programs and its implementation are to be verified by DOE.

The DOE independent verification may include review or approval of release-specific documentation and decision. Specific DOE approvals are not required for releases of materials under the IFB criterion if the clearance protocol based on the IFB criterion is approved by DOE and has been incorporated into the site's material clearance and release program.

For releases with the ANSI N13.12-2013 SL clearance criterion, DOE Field Element approval for each release plan and associated qualitative ALARA analysis is required. The level of independent oversight to verify that the approved plans are being implemented effectively is the responsibility of the DOE FEM.

Release based on a clearance criterion that is above the ANSI N13.12-2013 SLs shall follow the DOE process for the development, approval, and use of DOE Authorized Limits.

Stakeholder Communication

When non-IFB clearance criteria are used, a review of potential impacts or conflicts with both internal and external stakeholders shall be undertaken to ensure that the planned releases are compliant with all applicable regulations, requirements or agreements and the stakeholders are properly notified as necessary.

Stakeholder communication should begin with identification of relevant stakeholders, which may include DOE, other federal or state agencies, subcontractors including local recycling companies, members of the public, and site's management and departments for operation, property control, non-radiological safety, security, etc.

The appropriate and relevant requirements for stakeholders should be identified. Examples include:

1) The Secretarial memorandum entitled "Managing the Release of Surplus and Scrap Materials" dated January 19, 2001,
2) Property control processes including asset disposition,
3) State or local regulations for transport of oversized materials on public roads,
4) Site security processes which may limit transport of materials at certain times,
5) Local recyclers which may have limits on material size, weight, or commodity type restrictions,

6) Another DOE site which has an agreement to collect or transfer materials for use at their site,
7) Prime contracting agencies which may have unique relationships with community groups or sustainability goals that a material clearance and release program will enhance,
8) Site management which makes decisions on the use of any unexpected revenues created by the material clearance and release program, and
9) O458.1and other regulatory requirements.

The communication strategy may include development of appropriate means of communication (MOU, Letters, public website, FAQ, etc.). The strategy should also include elements of content, purpose, author or representative, and priority. For example, an FAQ may require content from a technical author, such as definition of IFB in language appropriate to the audience.

6 References

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7 Common Acronyms and Definitions

AL	Authorized Limit (DOE Order 458.1)
ALARA	As Low As Reasonably Achievable
DLA	Decision Level Activity
DL	Detection Limit
DT	Detection Threshold
EHSS	Environment, Health, Safety and Security
IFB	Indistinguishable from Background
SL	ANSI N13.12-2013 Screening Level

ALARA (As Low As Reasonably Achievable): An approach to radiation protection to manage and control releases of radioactive material to the environment, and exposure to the work force and to members of the public so that the levels are as low as is reasonably achievable, taking into account societal, environmental, technical, economic, and public policy considerations. As used in this Standard, ALARA is not a specific release or dose limit but a process which has the goal of optimizing control and management of releases of radioactive material to the environment and doses so that they are as far below the applicable limits of O458.1 as reasonably achievable. [adapted from DOE HDBK-1215-2014 definition].

ALARA Process: A graded process for evaluating alternative operations, processes, and other measures, for optimizing releases of radioactive material to the environment, and exposure to the work force and to members of the public taking into account societal, environmental, technical, economic, and public policy considerations to make a decision concerning the optimum level of public health and environmental protection. A graded approach provides the flexibility to perform qualitative or quantitative ALARA analyses. For low doses, qualitative evaluations normally will suffice. [adapted from DOE HDBK-1215-2014 definition].

Authorized Limit (AL): A limit on the concentration or quantity of residual radioactive material on the surfaces or within property that has been derived consistent with DOE directives including the ALARA process requirements. An Authorized Limit may also include conditions or measures that limit or control the disposition of property. [adapted from DOE Order 458.1 definition].

Clearance (of Property): The removal of property that contains or may contain residual radioactive material from DOE radiological control under 10 CFR Part 835 and DOE Order 458.1. [adapted from DOE Order 458.1 definition].

Clearance Protocol: A set of processes and rules for determining the disposition of material, which includes three elements: clearance criteria, process knowledge and measurement methods.

Confirmatory Measurement: Additional radiological measurement that is used to supplement the surface survey method. Confirmatory measurements may include gamma spectrometry in the field, laboratory analysis of representative samples of the item using low background HPGe or LSC measurements, portal gate monitor, bulk monitoring systems, or other means. Confirmatory

measurement is a best management practice and is conducted, as warranted, in a graded approach depending on considerations of process knowledge and ALARA.

Critical Level (also see Detection Threshold): In counting statistics, the Critical Level (L_C) is the count (or count rate) in a zero-mean count distribution having a defined probability (α) of being exceeded. It can be stated: the measured result at which one can decide whether or not the result indicates detection. It is common practice to set α equal to 0.05 and accept a 5% probability of incorrectly concluding that activity is present when it is not (a false positive result). This level is associated with determining if a measurement is IFB.

Critical Concentration: The Critical Concentration is the L_C corrected for detector efficiency and other factors to yield an estimate of the radioactivity concentration. The Critical Concentration has units of pCi/g, and it may also be called the decision level activity (DLA).

Detection Limit (DL): The smallest amount of radioactivity (in pCi/g for volumetric radioactivity measurements, or dpm / 100 cm^2 for surface activity) that can be detected and quantified by a measurement method. The DL may also be called the Minimum Detectable Concentration (MDC). Statistically, the detection limit is based on the L_D; the mean net count from a sample having a specified probability (β) of escaping detection (a false negative result). This probability is usually chosen to be 5%.

Detection Threshold (DT): The instrument signal level (in cpm, µR/h, etc.) associated with a determination that a measurement is different than background. The DT is analogous to L_C and is used to establish measurement protocols for IFB clearance processes. It should not be confused with the detection limit (DL), which is an estimate of the quantity of radioactivity that can be measured with a specified confidence.

Impacted Area: Areas at an accelerator site (normally within accelerator hosing or radiological areas) in which there is a reasonable potential for materials within the area to become activated or contaminated with potential radioactivity above background levels.

Indistinguishable from Background (IFB): The absence of detectable radioactivity as determined by evaluation or measurements.

Materials and Equipment (M&E): A generic term for personal property that includes materials, equipment, apparatus, components, articles, etc. to which this Standard applies.

Measurement Method: A set of processes and rules for making appropriate radiological measurements of potential volumetric radioactivity or surface activity for material clearance purpose, which includes the types of instruments and the measurement techniques. The measurement method for volumetric radioactivity in this Standard includes surface survey and, as warranted, the confirmatory measurements.

Measurement Quality Objective (MQO): A statement of a performance objective or requirement for a particular measurement method performance characteristic.

Non-impacted Area: Areas at an accelerator site in which there is no reasonable potential for materials present in the area to become activated or contaminated with radioactivity above background.

Personal Property (or Property): Property of any kind, except for real property. [Adapted from DOE Order 458.1 definition].

Process Knowledge: A collective, objective description of the physical, operational, administrative, and radiological conditions associated with material being considered for clearance such that a reasonable and defendable decision can be made regarding whether the material could or could not have become activated or contaminated. Examples of process knowledge include, but are not limited to: the size, geometry composition, and physical properties of the item, physics of radionuclide production in a facility, beam parameter and beam loss information, which allow for the evaluation of induced radioactivity, potential radioactivity distribution, radiological measurement results, etc.

Proxy Radionuclide: An easy-to-measure radionuclide which may be used to infer the existence, activity, or dose of a hard-to-measure radionuclide, based on the established relationship between the two types of radionuclides.

Real Property: Land and anything permanently affixed to the land such as buildings, fences and those things attached to the buildings, such as light fixtures, plumbing and heating fixtures, or other such items, that would be personal property if not attached. [Adapted from DOE Order 458.1 definition].

Restricted Release: A release of material for limited and controlled off-site use.

Screening Level (SL): Activity concentrations (for either surface or volume radioactivity) that are designed to determine compliance with the primary dose criterion through comparison with radiation survey results. [Adapted from ANSI N13.12-2013 definition]. The primary dose criterion is a Total Effective Dose (TED) of 1 mrem/y, above background, for clearance of materials from regulatory control.

Site: A land or property upon which DOE facilities or activities are located and access to which is subject to DOE or DOE contractor control. [Adapted from DOE Order 458.1 definition].

Surface Activity: Radioactivity residing on, or near, the surface of an item. This activity can be adequately quantified in units of radioactivity per unit surface area.

Surface Surveys (for Volumetric Radioactivity): Measurements over the surfaces of an object to determine the presence or quantity of potential volumetric radioactivity within the object due to activation. The measurements may be conducted using a portable survey meter in a scanning or fixed-position measurement mode or using a gamma spectrometer in a fixed-position mode. This may be different from the measurements for surface radioactivity.

Volumetric Radioactivity: Radioactivity residing in and throughout the volume of an item. At accelerator facilities, volumetric radioactivity in a material can result from volumetric activation by primary beam or secondary particles.

Appendix A: Rationale for Endorsement of ANSI N13.12-2013 SLs as DOE Pre-Approved Authorized Limits

A1: Rationale

In 1996, the International Atomic Energy Agency (IAEA) recommended a set of radionuclide-specific clearance levels for unrestricted release of solid materials, based on an annual dose criterion of 1 mrem (0.01 mSv) to general public members [IAEA1996]. In 1998, the European Commission (EC) recommended specific clearance levels for metal recycling, also based on a dose criterion of 1 mrem/y [EC1998]. In 1999, ANSI issued the N13.12-1999 Standard, "Surface and volume radioactivity standards for clearance" [ANSI1999], which was extensively revised in 2013 [ANSI2013], to establish Screening Levels (SLs) for radionuclides for clearance of materials from radiological control. The ANSI SL values for both surface and volumetric radioactivity were based on a dose criterion of 1 mrem/y to the public from exposure scenarios for potential use of released materials.

The ANSI annual dose criterion of 1 mrem is consistent with that endorsed by the IAEA and the EC. The clearance criterion of 1 mrem/y is also supported by the National Council on Radiation Protection and Measurements [NCRP2002] and the American Nuclear Society [ANS2008]. The ANSI SL levels satisfy the dose constraint of 1 mrem in a calendar year in the DOE Order 458.1 for clearance and release of personal property [DOE2013].

The potential radiological impact resulting from the unrestricted release of the materials that satisfy the criterion of ANSI N13.12-2013 SLs is no more than the dose of 1 mrem/y to a member of the public. This represents a potential exposure level that is well below the average dose of 310 mrem/y to the US population from natural background radiation [NCRP2009] and is within the normal fluctuation of annual background radiation levels. Potential radiological impacts to the general public and the environment due to the release of the materials are negligible and these materials should not be subject to radiological control and can have unrestricted release. Therefore, this Standard endorses the ANSI N13.12-2013 SL values for both surface activity and volume activity as the DOE pre-approved Authorized Limits.

A2: Use of Previously Approved Surface Activity Guidelines

DOE Order 458.1 allows the use of previously approved guidelines or limits (such as the surface activity guidelines) as the pre-approved Authorized Limits for unrestricted release of personal property that may be potentially surface contaminated. The surface contamination guidelines given in the now canceled DOE Order 5400.5 Figure IV-1 are considered previously approved guidelines. Property may be released if the results of a survey with appropriate instruments indicate that the potential activity of the property is less than the surface contamination guidelines given in DOE Order 5400.5 Figure IV-1.

DOE Order 5400.5 Figure IV-1 does not have surface contamination guidelines for radionuclides that emit pure beta radiation such as ^{3}H. 10 CFR Part 835, *Occupational Radiation Protection*, [DOE2011] sets a removable surface contamination value of 10,000 dpm / 100 cm^2 for tritium below which the materials can be removed from contamination areas to controlled areas. This

limit is based on the assumption that tritium will permeate the volume of whatever material is contaminated, so no value is given for total contamination (fixed + removable). DOE guidance [DOE1991] recommends the use of 10,000 dpm / 100 cm^2 for removable surface contamination as an acceptable limit for unrestricted release.

The surface contamination guidelines for beta-gamma emitters from DOE Order 5400.5 Figure IV-1 (and 10 CFR 835 for tritium) are summarized in Table A1. These values may be used before the ANSI N13.12-2013 surface activity SLs are approved by DOE as pre-approved Authorized Limits for the unrestricted release of materials with only potential surface contamination.

Table A1: Surface contamination guidelines.

Radionuclide	**DOE Order 5400.5 Figure IV-1 Allowable Total Residual Surface Contamination (dpm / 100 cm^2)**		
	Average	**Maximum**	**Removable**
Beta-gamma emitters, except ^{90}Sr and others noted in Figure IV-1	5,000	15,000	1,000
	10 CFR Part 835 Removal Surface Contamination (dpm / 100 cm^2)		
Tritium	10,000		

A3: Use of ANSI N13.12-2013 Volume SLs

Radionuclides have different radiological hazards (or dose risk), depending on the energy and yield of radiation (photon, beta, alpha or neutron) emitted per disintegration and the half-life. ANSI N13.12-2013 has evaluated the dose risk for various radionuclides to members of the public and assigned a Screening Level (SL) that is equivalent to a dose risk of 1 mrem/y for unrestricted use scenarios of the released material, which also includes a generic consideration of ALARA process. The radionuclides are separated into 5 groups, each with an ANSI SL value.

For volumetric radioactivity, the ANSI volume SL value is expressed in radioactivity per unit mass (in units of pCi/g) for residual radionuclides within a material. Table 1 of this Standard shows that the volume SL is:

- 3 pCi/g for Group 1: high-energy gamma emitters, radium, thorium, transuranics and mobile beta-gamma emitters (e.g., ^{22}Na, ^{54}Mn, ^{60}Co, ^{65}Zn and ^{152}Eu),
- 30 pCi/g for Group 2: uranium and selected beta-gamma emitters (e.g., ^{57}Co, ^{58}Co and ^{59}Fe),
- 300 pCi/g for Group 3: general beta-gamma emitters (e.g., ^{7}Be),
- 3,000 pCi/g for Group 4: low-energy beta-gamma emitters (e.g., ^{3}H and ^{63}Ni), and
- 30,000 pCi/g for Group 5: low-energy beta emitters (e.g., ^{55}Fe).

ANSI N13.12-2013 provides a more complete list of radionuclides. Group 1 radionuclides emit high-energy and high-yield gamma rays, which are easy to measure with common survey instruments. Some of these radionuclides are called proxy radionuclides in this Standard and are the main radionuclides of interest to be measured. Groups 4 and 5 radionuclides generate much lower dose risk (a factor of 1,000 and 10,000, respectively, lower than the Group 1 radionuclides) because they emit only beta or very low energy gamma rays, which are also harder to measure with common instruments.

Because more than one radionuclide can be present in a material, it is the ratio of the radioactivity to SL for each radionuclide (called the SL fraction in this Standard) that is important. To satisfy a clearance criterion using the ANSI N13.12-2013 SLs, the sum of the SL fraction for all potential radionuclides (called R value in this Standard) shall not be more than one, as shown in Eq. A1:

$$R = \sum_i (A_i / SL_i) \qquad \text{(Eq. A1)}$$

where:
A_i = Radioactivity per unit mass for radionuclide *i* (in pCi/g)
SL_i = ANSI N13.12-2013 Screening Level for radionuclide *i* (in pCi/g)

References

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Appendix B: Process Knowledge for Volumetric Activation in Accelerator Facilities

This Appendix provides the types of process knowledge that may allow for evaluation of volumetric induced radioactivity and its characteristics in accelerator facilities.

The process knowledge may include:

1) Physics of radionuclide production (activation and radioactive decay and progeny in-growth, types and yields of radionuclides that can be produced as a function of radiation particle type and energy, radioactivity distribution, etc.),
2) Accelerator and beam line characteristics (e.g., layout of accelerator and beam line components, beam particle type, beam energy, beam loss locations and amounts, etc.),
3) Material and component characteristics (e.g., type, size, geometry, composition and physical property of material and component, as well as common impurities),
4) Accelerator and facility operation history (e.g., the period and length of beam-on operations, the cool down time or decay time, the maintenance/repair activities that may cause surface activity, etc.).

Process knowledge can be of general type that is applicable to most (if not all) accelerator facilities with similar characteristics and operational modes (in terms of producing induced radioactivity). Examples of general process knowledge include:

1) Klystrons that operate at no more than a few MeV do not cause activation and, if they are not located inside any accelerator housing they will not be activated either. Though the immediate area surrounding a klystron is generally posted conservatively as a Radiation Area when the klystron is on for the purpose of personnel exposure control for prompt radiation; there should be no induced radioactivity issues.
2) Synchrotron radiation beam lines at synchrotron light facilities and the photon beam lines at Free Electron Laser (FEL) facilities operated at less than a few MeV do not have activation potential (same reason as the klystrons).
3) Low-energy x-ray devices do not have activation potential.
4) The higher the particle beam energy (when exceeding the activation thresholds of the materials), the higher the beam loss, or the longer the beam loss period, the higher the activation.
5) Areas with potential activation are within the accelerator housings, in which the access is prohibited with an interlocked access control system to prevent personnel exposure to prompt radiation coming from accelerator beam losses.
6) The activation inside the accelerator housing is limited to locations near normal beam loss points such as beam targets, beam dumps, collimators and slits (these components are designed to intercept a fraction of or the entire beam). Most components inside accelerator housings do not have activation.
7) Large experimental detector regions (except in regions of target or collision point) are expected to have lower activation potential than regions with accelerator beam lines, because the beam losses near a detector have to be very small, if not zero, to avoid detector background interference problems.
8) Storage ring accelerators have lower activation potential than linear accelerators due to their lower integral beam losses.

9) There is minimal possibility of surface activity in solid materials, unless they are submitted to destructive process such as machining, drilling, or grinding.

Specific process knowledge, such as information of beam losses and operational history, may be applicable to only a specific facility, a project or an activity within the site. Most facilities likely have used normal beam loss information for shielding design and evaluation of the radioactive air production for annual National Emissions Standard for Hazardous Air Pollutants (NESHAPs) reporting. However, in many cases, these beam loss estimates, though conservative, are subject to large uncertainty or not well known such that an accurate estimation of the absolute radioactivity profile for object of interest is not feasible.

Appendix C: Technical Basis for Volumetric Activation at Electron Accelerator Facilities

This Appendix describes the technical basis for the material clearance at high-energy electron accelerator facilities. More technical basis details are available in [JLAB2009; SLAC2011]. The characteristics of the induced radionuclides and volumetric radioactivity in materials due to activation, as well as the process knowledge and technical basis supporting the measurement methods for volumetric radioactivity, are summarized in this Appendix.

In summary, the key technical basis includes the following:

1) Radionuclides with either their atomic number or mass number that is lower than their parent nuclides can be produced (except radionuclides produced from thermal neutron absorption), but none are alpha emitters. *This justifies that the beta-gamma radionuclides are the radionuclides of interest.*
2) Most abundant radionuclides of interest are those with half-lives on the order of the beam irradiation time (about 1 to 10 years). *This justifies the radionuclides with medium to long half-lives are the radionuclides of interest.*
3) Radionuclides that emit pure beta or low energy x rays (which are hard to measure) are accompanied by "proxy" radionuclides that emit high-yield, high-energy gamma rays (which can be measured easily). *This justifies the measurements of proxy radionuclides, instead of measurements of all potential radionuclides including the hard-to-measure radionuclides that can be produced.*
4) Induced radioactivity profile in an object is volumetric (non-uniform in general) and the maximum radioactivity is near the surface that faces the beam loss point. *This justifies using surface (scanning or fixed positions) measurements for volumetric radioactivity*.
5) There are commercially available portable instruments, such as the scintillator-based survey meters, that have detection thresholds or limits for proxy radionuclides that are less than the corresponding ANSI N13.12-2013 SL values. *This justifies the use of sensitive, portable survey meters for volumetric radioactivity measurements*.

C1: Characteristics of Induced Radionuclides

Common metal in accelerator structures include aluminum, iron, copper and steel. From the activation point of view, the main elements present in steel are iron and nickel. The accelerator enclosure walls and floor are commonly made of concrete (mainly SiO_2).

Activation formula shows that the activity of a radionuclide will approach saturation after an irradiation time of about 3 half-lives, and its activity will be reduced to a small fraction of the original radioactivity for approximately 4 half-lives. Therefore, the most abundant radionuclides that can be produced from activation are those with a half-life of the order of the typical irradiation time, e.g., a few months to a few years. For example, for release from a facility decommissioning, the radioactivity of radionuclides with half-lives of the order of a few weeks or months will have reached saturation after a few years of accelerator operation, and likely decay to negligible levels after facility shutdown and prior to release. Therefore, short half-life radionuclides may be neglected for releases from facility dismantling and decommissioning. On the other hand, the activities of radionuclides with half-lives longer than 100 years are built up to

only negligible levels after a few years of operation, and therefore their radioactivity levels may also be ignored.

Table C1 summarizes the radionuclides with a half-life longer than 200 days and shorter than 100 years that can be produced in common accelerator materials [SLAC2011]. The radionuclides where the production yield is the highest for each element have been indicated with "main".

At high energy particle accelerators, induced radioactivity in materials can occur as a direct or indirect consequence to exposure to a particle beam [IAEA1979; SUL1992]. High energy electron beam losses at a normal beam loss point create high energy bremsstrahlung photons, which in turn create hadrons (particularly neutrons). These secondary photons and neutrons can have energies as high as the electron beam energy. The activation in electron accelerators is mainly due to photonuclear reactions with thresholds around 10 MeV (in which a high energy bremsstrahlung photon is absorbed by a nucleus and at least one neutron or proton is ejected) and spallation reaction with threshold around hundreds of MeV (in which a nucleus is broken into more than one nucleus by a high energy photon or hadron). In these cases, the atomic numbers and mass numbers of the daughter radionuclides are generally much lower than their parent nuclides. A few nuclides (e.g., cobalt and europium trace elements in concrete) have large thermal neutron absorption cross sections. In this case, the daughter radionuclides have their mass numbers higher than their parent nuclides (with the same atomic number). The end result is that most induced radionuclides are beta-gamma emitters, while no alpha emitters and very few pure beta emitters are produced.

A few radionuclides are difficult to measure because they emit only betas (e.g., ^{3}H and ^{63}Ni) or low energy x-ray (e.g., 5.9 keV x-ray from ^{55}Fe). However, these hard-to-measure radionuclides always occur in the presence of other radionuclides (which emit high-energy and high-yield gamma rays) with sufficient amounts, which are easy to measure with common instruments [SLAC2011]. These easy-to-measure radionuclides are called "proxy" radionuclides in this Standard, because measurements of these proxy radionuclides can be used to infer indirectly the radioactivity levels of the hard-to-measure radionuclides. For instance, ^{55}Fe in iron is accompanied by proxies of ^{22}Na (1.27 MeV γ), ^{54}Mn (emitting 835 keV γ) and ^{57}Co (122 keV γ). ^{22}Na, which has a half-life (2.6 years) similar to ^{55}Fe (2.7 years), serves as the best proxy for ^{55}Fe. Similarly, pure beta emitter ^{63}Ni in copper is accompanied by the proxy radionuclide of ^{60}Co (1.17 and 1.34 MeV γ) in copper in addition to the proxy radionuclides in iron.

Table C2 summarizes the main radionuclides and proxy radionuclides that can be produced in stainless steel, copper, iron, aluminum and concrete [SLAC2011]. The important proxy radionuclides in metal are ^{22}Na, ^{54}Mn and ^{60}Co. In concrete, radionuclides ^{152}Eu and ^{60}Co, whose radioactivity level depends strongly on thermal neutron fluence and the trace amounts of europium and cobalt nuclides (which have very large thermal neutron cross sections), can be measured in most concrete samples and thus can also serve as proxy radionuclides.

It is not coincidental that proxy radionuclides belong to the ANSI N13.12-2013 [ANSI2013] Group 1 radionuclides (e.g., ^{22}Na, ^{54}Mn, ^{60}Co, ^{65}Zn and ^{152}Eu, which have a volume SL of 3 pCi/g) while the hard-to-measure radionuclides belong to the ANSI N13.12-2013 Groups 4 radionuclides (e.g., ^{3}H and ^{63}Ni, whose SL is 3,000 pCi/g) or Group 5 radionuclides (e.g., ^{55}Fe,

whose SL is 30,000 pCi/g). This is because proxy radionuclides emit high-yield and high-energy gammas such that they dominate the dose rate on the item's surface and simultaneously create higher dose risk.

Because more than one radionuclide can be present in a single material, it is the ratio of the radioactivity concentration to SL for each radionuclide (called SL fraction in this Standard) that is important. The SL ratio between the ANSI Group 4 and Group 1 radionuclides is a factor of 1,000. Therefore, if the ratio of volumetric radioactivity between the hard-to-measure and proxy radionuclides is evaluated to be $\leq$ 1,000, then the measurements of proxy radionuclides would be sufficient.

The characteristic of "proxy radionuclides" is essential and important for clearance measurements: if the proxy radionuclides are not detected (i.e., measurements are IFB), the hard-to-measure radionuclides will not be present at a level that would create a radiation risk either (i.e., higher than their corresponding ANSI N13.12-2013 SLs). Therefore, measurements of proxy radionuclides are sufficient to characterize the materials for clearance.

C2: Volumetric Radioactivity Profile

A well benchmarked Monte Carlo code that can calculate induced radioactivity, such as FLUKA [FLUK2003; FLUK2005], can be used to evaluate the characteristics of induced radioactivity due to prompt radiation. Such evaluation can help to optimize the measurement process in a graded approach. If operational information such as beam losses in components or areas is known, the calculated results can be normalized to estimate the absolute induced radioactivity, which then can be used in the clearance decision process based on this process knowledge.

FLUKA has been used extensively to calculate radioactivity profiles in metals and concrete at accelerator facilities [SLAC2011]. A few calculation and measurement examples are summarized below to illustrate two main characteristics of volumetric radioactivity profiles, proxy radionuclides and surface maximum, at accelerator facilities.

The center of the 3-floor-high BaBar detector in the SLAC PEP-II ring is where the circulating electron and positron beams collide for particle physics study. The BaBar detector has thousands of components in a complex 3-dimensional geometry. Figure C1 shows the FLUKA-calculated induced radioactivity profiles in one quadrant of BaBar detector components. The source term in this calculation is from the low-probability electron and positron collisions at the BaBar center location. Therefore, it is expected that only the central portion of the BaBar detector, e.g., vertex detector and drift chamber, will have measurable radioactivity, if any, from this type of beam losses. This non-normalized, 3-dimensional profile shows only relative radioactivity and is meant to illustrate the "surface maximum" characteristics, i.e., the maximum radioactivity in any potentially activated component within the BaBar detector is always on the surface that faces the source (i.e., e+ and e- collision point in the BaBar detector center).

Table C3 summarizes the induced radioactivity of a BaBar detector component (a large steel plug) calculated with FLUKA at decay times of 1, 2 and 5 years after 10 years of operation. In metal, the hard-to-measure ^{3}H radioactivity is less than the proxy radionuclides of ^{60}Co and

^{54}Mn. Although the radioactivity of the hard-to-measure ^{55}Fe is about twice as high as that of the ^{60}Co proxy, the SL fraction for ^{55}Fe is still a factor of 4,000 lower than that of ^{60}Co. Therefore, measurements of proxy radionuclides are sufficient.

Though concrete is not the main material for this Standard, its induced radioactivity characteristics and profiles may be relevant and applicable to metal, as concrete contains some common metal elements such as manganese and iron. The radioactivity profiles for radionuclides of interest as a function of depth inside a cylindrical concrete tunnel wall in a laterally stray radiation field 2 meters away from a thick iron target at four electron beam energies (0.025, 0.1, 1 and 10 GeV) have been calculated using FLUKA for an irradiation time of 10 years with 3 decay times (4 months, 1 year and 5 years). The four electron energies were chosen to evaluate the different activation mechanisms as a function of photon and neutron energies (e.g., photonuclear reaction between 10-100 MeV and spallation reaction above 100 MeV). Figure C2 shows the radioactivity profiles for a decay time of 1 year with ^{22}Na (from spallation of silicon) as the main proxy and ^{54}Mn as the secondary proxy, whose radioactivity is no less than 1/10 of the hard-to-measure radionuclides of ^{55}Fe and ^{3}H.

The radioactivity profiles for radionuclides of interest as a function of depth inside concrete at electron accelerators with electron energy between 45 MeV and 1.3GeV as well as proton energy between 17 MeV and 12 GeV (with unknown operation history) have been measured [MASU2003; MASU2008]. The results show that the proxies of ^{22}Na and ^{54}Mn from fast neutron reaction can be seen and radionuclides from thermal neutron capture reaction such as ^{60}Co and ^{152}Eu, which are not in the FLUKA-calculated results, can also been seen. Radionuclide ^{3}H has similar depth profiles to those of ^{60}Co and ^{152}Eu in concrete and, thus, ^{3}H is produced from the reaction of ^{6}Li(n,α)^{3}H. It is also because of the build-up of thermal neutron fluence inside concrete, the maximum radioactivity of ^{3}H, ^{60}Co and ^{152}Eu are near, but not at, the surface of concrete.

These calculations and measurements (Figures C1 and C2) illustrate the "surface maximum" characteristics, which allow for the use of surface measurements over an item (scanning or fixed-position measurements) for volumetric radioactivity. If the surface facing the beam loss point is known, only that surface needs to be surveyed. This approach is particularly useful for measurements of large objects as it is generally easier to identify the surface of a large object that faces the beam loss points and it is not easy and safe to flip the large objects in order to conduct an all-surface measurement. This approach has a great advantage and cost implication for dismantling operations. The radioactivity profiles may also be used to derive the volumetric radioactivity averaged over the volume of interest.

C3: Summary of Characteristics

The above FLUKA calculations and measurements [SLAC2011; MASU2008] illustrate the following characteristics of "proxy radionuclides" for metals and concrete:

1) Proxy radionuclides in metals include ^{22}Na, ^{54}Mn and ^{60}Co.
2) Proxy radionuclides in concrete include ^{22}Na and ^{54}Mn (same as metals) as well as ^{60}Co and ^{152}Eu from thermal neutron capture reaction of trace elements.

3) Proxy radionuclides contribute to most (if not all) of the surface dose rate due to their high-energy and high-yield gamma rays.
4) The hard-to-measure radionuclides ^{3}H and ^{55}Fe can exist in metals and concrete. However, their radioactivity is no more than 100 times of those of proxy radionuclides and their dose risks (when normalized to the ANSI SL values) are at least 100 times smaller than the proxy radionuclides. Therefore, measurements of proxy radionuclides are sufficient.
5) In concrete, ^{3}H radioactivity is no more than 100 times of ^{152}Eu radioactivity (13.5-y half-life). Therefore, ^{152}Eu will have a higher SL fraction regardless of decay time and ^{60}Co will have a higher SL fraction for short decay time. This fact supports the use of proxies of ^{152}Eu and ^{60}Co, in addition to ^{22}Na and ^{54}Mn, for the clearance measurements of concrete. Note that the amounts of ^{152}Eu and ^{60}Co depend on not only the amounts of trace elements, but also on the beam energy, the decay time, and the room geometry to produce thermal neutrons. Measurements show that the radioactivity ratio of ^{152}Eu or ^{60}Co to ^{22}Na varies between 1 and 10 [MASU2008].
6) The proxy radionuclide approach may have limitations. For examples, if a concrete block has decayed for over 30 years, ^{3}H radioactivity (12.3-y half-life) may become 1,000 times higher than ^{22}Na radioactivity (2.6-y half-life) and then has a higher SL fraction. Solutions are to collect core samples or surface swipe and analyze with laboratory LSC measurements for ^{3}H, or use other proxy radionuclides such as ^{152}Eu and ^{60}Co.
7) The radionuclides of interest are limited to no more than ten such that the summation in Equation A1 (or E1) can be practically done.

The following characteristics for "surface maximum" can be noted for concrete:

1) At low energy electron accelerators (< 100 MeV) where activation is from photonuclear reaction, the radioactivity is reduced to ~1/1,000 of its surface maximum for every one meter depth; consistent with the attenuation length of 42 g/cm^2 for bremsstrahlung photons in concrete.
2) At high energy electron and proton accelerators (> 100 MeV) where activation is mainly from spallation reactions, the radioactivity is only reduced to ~1/10 of its surface maximum for every one meter depth; consistent with the attenuation length of 120 g/cm^2 for high energy neutrons in concrete.
3) The radioactivity profiles of thermal-neutron-capture products such as ^{60}Co and ^{152}Eu follow the high energy neutron attenuation profile at high energy electron accelerators and follow the attenuation length of bremsstrahlung photons at low energy electron accelerators. However, their absolute magnitudes also depend on amounts of trace elements of cobalt and europium in concrete.

Table C1: Radionuclides with a half-life longer than 200 days and shorter than 100 years that can be produced in common accelerator materials.

Radionuclide	Half-life	Decay modes	O	Al	Si	Fe	Ni	Cu
^{3}H	12.3 y	β^-, no γ	main	✓	✓	✓	✓	✓
^{22}Na	2.6 y	β^+, γ		main	main	✓	✓	✓
^{54}Mn	312 d	ε, γ				main	✓	✓
^{55}Fe	2.73 y	ε, x, no γ				main	✓	✓
^{57}Co	272 d	ε, γ				main	main	✓
^{60}Co	5.26 y	β^-, γ					main	main
^{63}Ni	100 y	β^-, no γ						main

Table C2: Main and proxy radionuclides in metal and concrete materials

Material	Radio-nuclide	Half-life	Main production	Major γ energy	Proxies, Remarks
Carbon steel (Fe, C), Cast iron (Fe, C, Si, Mn)	^{22}Na	2.6 y	Fe, Si, Mn spallation	1.27 MeV (100%)	
	^{54}Mn	312 d	^{55}Mn(γ,n)	835 keV (100%)	
	^{55}Fe	2.73 y	^{56}Fe(γ,n)	5.9 keV x-ray	^{22}Na
	^{57}Co	272 d	^{58}Fe(p,2n)	122 keV (86%) 133 keV (10%)	
Austenitic stainless steel (Fe, C, Cr, Ni) & Copper	the same radionuclides as carbon steel and cast iron, plus:				
	^{60}Co	5.26 y	^{60}Ni(n,p), Cu spallation	1.17 MeV (100%) 1.33 MeV (100%)	
	^{63}Ni	100 y	^{64}Ni(γ,n), Cu spallation	no γ	^{60}Co
Superaustenitic, Ferritic, Martensitic SS (Fe, C, Cr, Ni, Mo)	the same radionuclides as austenitic SS and copper, plus:				
	^{93m}Nb	16.3 y	Mo spallation	31 keV x-ray	^{60}Co
	^{90}Sr	28.8 y	Mo spallation	no γ	^{60}Co
	^{85}Kr	10.8 y	Mo spallation	514 keV (0.4%)	^{60}Co
	^{65}Zn	244 d	Mo spallation	1116 keV (51%)	
Aluminum	^{22}Na	2.6 y	Al spallation	1.27 MeV (100%)	
Concrete (trace amounts of Eu and Co, with large thermal neutron cross sections)	^{3}H	12.3 y	O, Si spallation	no γ	^{22}Na, ^{60}Co, ^{152}Eu
	^{22}Na	2.6 y	Si spallation	1.27 MeV (100%)	
	^{54}Mn	312 d	^{55}Mn(γ,n)	835 keV (100%)	
	^{55}Fe	2.73 y	^{56}Fe(γ,n)	5.9 keV x-ray	^{22}Na
	^{57}Co	272 d	^{58}Fe(p,2n)	122 keV (86%) 133 keV (10%)	
	^{60}Co	5.26 y	^{59}Co(n,γ)	1.17 MeV (100%) 1.33 MeV (100%)	thermal neutron capture
	^{152}Eu	13.5 y	^{151}Eu(n,γ)	many (sum > 140%)	
	^{154}Eu	8.59 y	^{153}Eu(n,γ)	many (sum > 150%)	

Table C3: Induced volumetric radioactivity of a BaBar component (IFR forward steel plug) calculated with FLUKA at three decay times after 10-y operation.

Radionuclide	**Half-life**	**Radioactivity Calculated with FLUKA in pCi/g (% of total)**		
		1 year	**2 years**	**5 years**
^{60}Co (proxy)	5.3 y	1.0×10^{-6} (22%)	8.9×10^{-7} (27%)	6.0×10^{-7} (38%)
^{57}Co	272 d	9.4×10^{-8} (2%)	3.7×10^{-8} (1.1%)	2.3×10^{-9} (0.1%)
^{55}Fe	2.7 y	2.4×10^{-6} (53%)	1.9×10^{-6} (57%)	8.8×10^{-7} (55%)
^{54}Mn (proxy)	313 d	6.5×10^{-7} (14%)	2.9×10^{-7} (9%)	2.5×10^{-8} (1.6%)
^{49}V	338 d	2.7×10^{-7} (6%)	1.3×10^{-7} (4%)	1.4×10^{-8} (0.9%)
^{3}H	12.3 y	6.9×10^{-8} (1.5%)	6.5×10^{-8} (2%)	5.5×10^{-8} (3.4%)
Remaining	—	2.3×10^{-8} (0.5%)	1.0×10^{-8} (0.3%)	6.1×10^{-9} (0.4%)

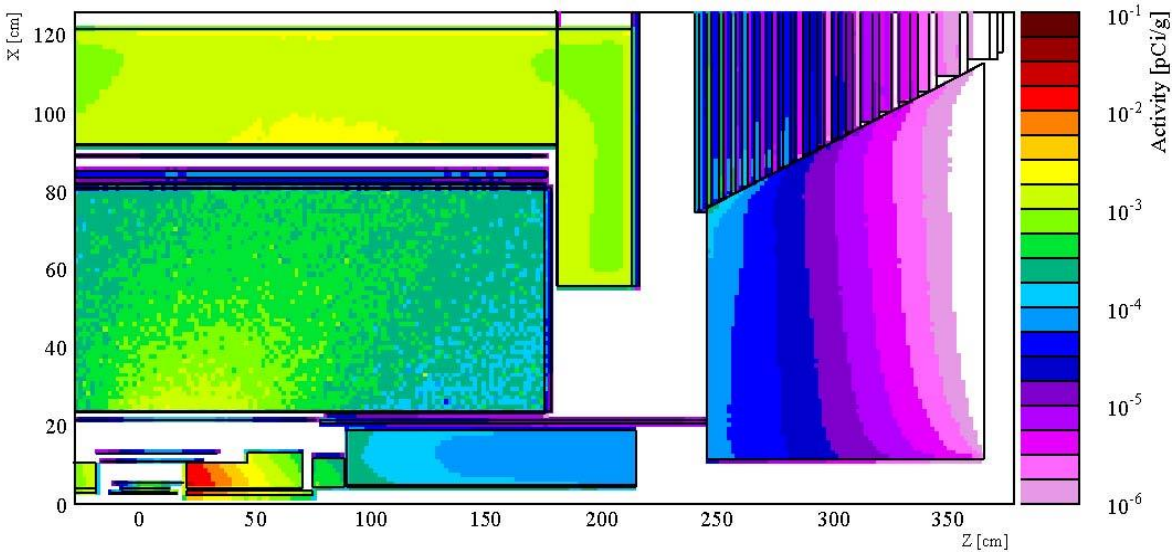

Figure C1: FLUKA-calculated relative volumetric radioactivity profiles in one quadrant of BaBar detector components to show the "surface maximum" characteristics. Source term is the electron and positron collision at the BaBar center location (0, 0, 0).

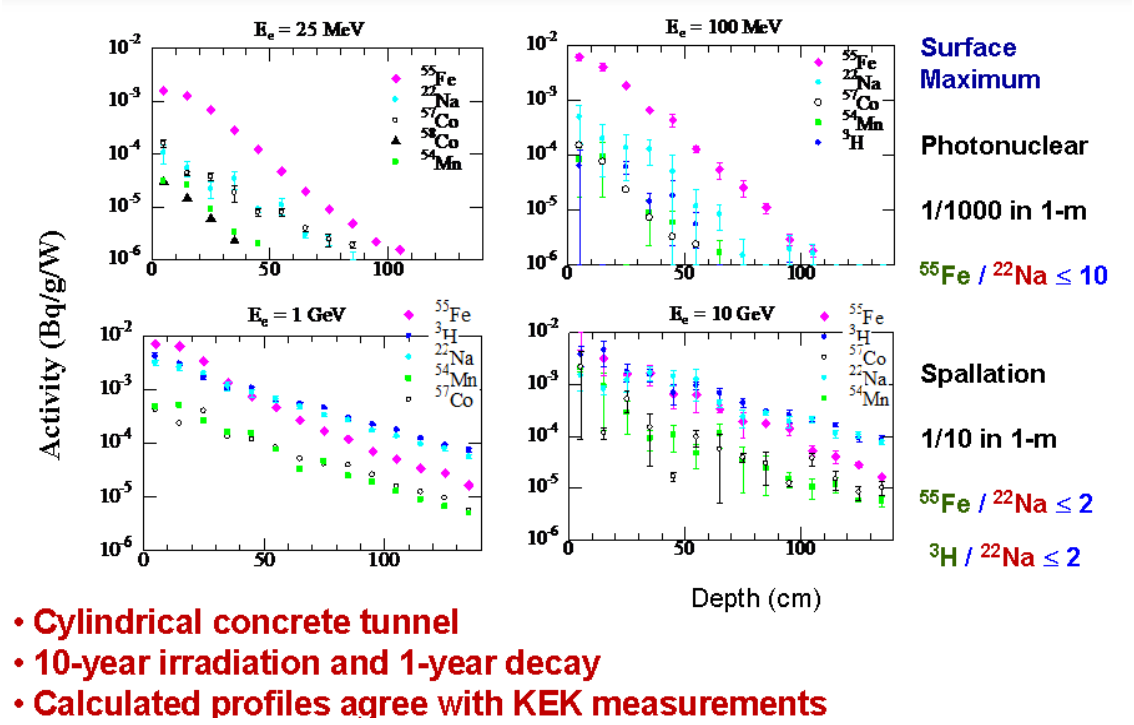

Figure C2: FLUKA-calculated volumetric radioactivity profiles in concrete wall at four electron beam energies (irradiation time of 10 years and a decay time of 1 year).

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Appendix D: Technical Basis for Volumetric Activation at Proton Accelerator Facilities

Proton accelerator activation mechanisms have been well identified in literature [IAEA1988; SUL1992] and tools are available to estimate their production [MCNP2010; NNDC2013].

Many characteristics of volumetric activation at proton accelerators that are relevant to material release are the same as those for electron accelerators provided in Appendix C. For example,

1) No alpha emitters of significance in common metals' activation products.
2) The radionuclides with medium to long half-lives (1-10 years) are the radionuclides of interest.
3) Beta-gamma radionuclides are the radionuclides of interest.
4) Radionuclides that emits pure beta or low energy x-rays (which are hard to measure) are accompanied by proxy radionuclides that emit high-yield, high-energy gamma rays (which can be measured easily).
5) Induced radioactivity profile in an object is volumetric (non-uniform in general) and the maximum radioactivity is at, or near, the surface that faces the beam loss point.

The reason for this similarity is because the dominant reactions producing the induced radionuclides of interest are the same for electron and proton accelerators [IAEA1979; IAEA1988; SUL1992]. This is true particularly for proton beams higher than 100 MeV when spallation reaction becomes the dominant reaction for induced radioactivity production. For beam energy at only tens of MeV, nucleon-ejecting reactions such as (p,np) produce similar radionuclides as the reactions of (γ,n) and (γ,p) in electron accelerators.

D1: General Production Process and Radionuclide Characteristics

For the purposes of this Appendix, only common metals in proton accelerators will be considered, namely, aluminum, iron, steel, stainless steel, and copper.

Production mechanisms are chiefly identified by material types, their proximity to the actual proton beams, and the energy and power of the proton beam. As the target atomic number increases, so does the range of activation products.

Metals directly exposed to a proton beam above approximately 10 MeV can become activated. As the energy of the proton beam increases, intranuclear cascades will develop, and both primary interactions (spallation) and secondary interactions from excited nuclei (evaporation) will cause activation products that range from (p,2p) and (p,n) products to (p,XnXp) products of nearly any total value of X. Components near or in the beam (beam pipe, valves, flanges, and some magnets) can be activated by these reactions. These components are termed "proximal" in this Appendix.

Metals that are farther away from the proton beam, or which are activated by secondary products, such as neutron beam lines at the Spallation Neutron Source (SNS), are not activated in the same way that the proximal metals are. Their activation is dominated chiefly by neutron interactions and, to a lesser extent, high energy gamma interactions. High energy neutron activation products include the same radionuclides as the proton beam products, but as the

neutron energies decrease, the source term becomes dominated by neutron absorption radionuclides, so the mix of expected activation products is different for this source term. These components are termed “distal” in this Appendix.

Table D1 lists activation products that have half-lives between 30 days and 101 years in metals in a proton accelerator [IAEA1979; SCHW2012]. Though the radionuclides in the table consider spallation, evaporation, and absorption products, the specific mix of radionuclides is dependent on both the energy of the proton beam and the distance to the metals being considered.

Table D1: Common activation products in metals in proton accelerators

Metal	**Induced Radionuclide**
Al	^{7}Be, ^{45}Ca, ^{3}H, ^{22}Na
Fe, Steel	As above, plus ^{56}Co, ^{57}Co, ^{58}Co, ^{60}Co, ^{55}Fe, ^{59}Fe, ^{54}Mn, ^{124}Sb, ^{125}Sb, ^{46}Sc, ^{113}Sn, ^{44}Ti, ^{88}Y, ^{88}Zr
Copper	As above, plus ^{63}Ni,^{65}Zn

D2: Proxy Radionuclides

The beta or low energy photon radionuclides (e.g., ^{3}H, ^{7}Be, ^{45}Ca, ^{55}Fe, and ^{63}Ni) are referred to as “hard to measure”. In most cases – especially in the proximal metals – the radionuclides that are hard to measure are accompanied by proxy radionuclides that are reasonably easy to detect. It should not be assumed, however, that a proxy radionuclide that is adequate in one part of the accelerator facility is adequate in another technically distinct area.

Table D2 presents the radionuclides in Table D1 along with an analysis of their hazard and ability to be detected. “Relative Hazard” is based on the ANSI N13.32 Screening Levels for each radionuclide compared with ^{60}Co (Group 1). The “Detectability/Hazard” column is based on the relative response of a 2”×2” NaI detector for each radionuclide compared to the relative hazard for that radionuclide, compared to that for ^{60}Co [REAV2009]. “Hard to Measure” was determined conservatively when the Detectability/Hazard value was less than 1% of the Relative Hazard value (compared to the SL ratio of 1,000 between Group 1 and Group 4 radionuclides).

Using this approach, the hard-to-measure radionuclides in common metals for which a proton facility needs to be concerned for finding a proxy radionuclide are: ^{3}H, ^{7}Be, ^{45}Ca, ^{55}Fe and ^{63}Ni, if they are a significant part of the source term.

Because the radionuclide mix in metals at proton accelerator facilities can vary widely, each facility should consider of what their possible source term could consist. Calculations or analytical measurements should be used at each facility to estimate the ratio of radioactivity for each of these hard-to-measure radionuclides to one or more of the proxy radionuclides. If these calculations or analytical measurements, along with a derived detection capability for the instruments for the proxy radionuclides can demonstrate that the relevant clearance criterion could be met, then a sufficient burden of proof exists for the requested clearance package.

Table D2: Detectability versus hazard.

Radionuclide	Half-life	Relative Hazard	Detectability / Hazard	Hard to Measure?
^{26}Al	7.2E5 y	100%	173%	
^{7}Be	53.2 d	1.0%	0.1%	Yes
^{45}Ca	162.6 d	0.1%	0.0%	Yes
^{56}Co	77.2 d	100%	146%	
^{57}Co	271.8 d	10%	13%	
^{58}Co	70.9 d	10%	8%	
^{60}Co	5.27 y	100%	100%	
^{55}Fe	2.74 y	0.01%	0.0%	Yes
^{59}Fe	44.5 d	10%	5.7%	
^{3}H	12.3 y	0.1%	0.0%	Yes
^{54}Mn	312.2 d	100%	57%	
^{22}Na	2.6 y	100%	190%	
^{63}Ni	101 y	0.1%	0.0%	Yes
^{124}Sb	60.2 d	10%	11%	
^{125}Sb	2.76 y	100%	65%	
^{46}Sc	83.8 d	100%	109%	
^{113}Sn	115.1 d	10%	5.3%	
^{44}Ti	60.0 y	0.1%	0.3%	
^{88}Y	106.6 d	100%	100%	
^{88}Zr	83.4 d	1.0%	0.8%	
^{65}Zn	243.6 d	100%	28%	

The residual radioactivity in material depends on the material type, activation type (directly activated by proton beam or activated by secondary particles, and at what energy), irradiation time, and decay time. Whether or not the hard-to-measure radionuclides have an adequate proxy depends on all of these factors.

The MCNPX Monte Carlo code was coupled with the analytic activation code CINDER to estimate radionuclide concentrations in materials. Extensive calculations were performed for a proton beam loss events at energies 17 MeV, 45 MeV, 65 MeV, 400 MeV, and 12 GeV for three types of materials (316 stainless steel, copper, and 6061 aluminum). Irradiation times of ½ year, 1 year, 5 years, 15 years, and 30 years were considered, each with decay times of 90 days, 1 year, 5 years, 10 years, and 50 years. Results were normalized to a proton beam loss of 1 W.

The “distal” calculations were for materials (316 stainless steel, copper, and 6061 aluminum) immediately surrounding a steel block (1 meter long, 15 cm radius; where a full hadron cascade can occur), in which a proton beam hit the end of a large steel block. The calculations show that the maximum radioactivity for materials occurs at the surface closest to the beam loss.

The "proximal" calculations were for materials directly impacted by the proton beam. In this case, the simulation was for a beam pipe of each of the respective materials, which was simulated by an end strike on a 1-m long, 0.3175 cm volume of the material (316 stainless steel, copper, or 6061 aluminum).

For example, a copper beam pipe directly activated by a proton beam irradiated for 30 years and decayed for 30 days has the following activation profile as a function of proton energy between 17 MeV and 12 GeV, shown in Figure D1 (a proximal case). A similar profile for 316 stainless steel alloy may be seen in Figure D2.

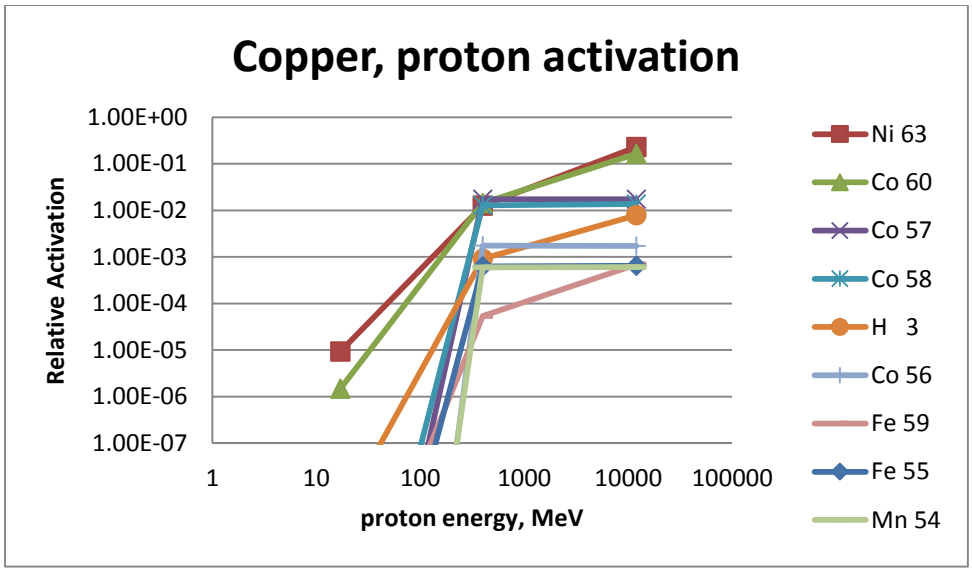

Figure D1: Activation as a function of energy in copper (30-y irradiation and 30-day decay).

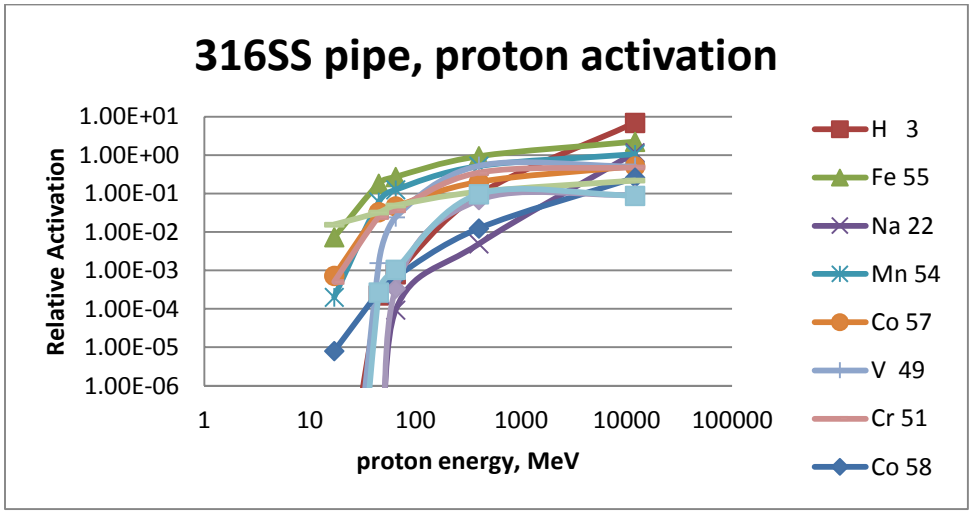

Figure D2: Activation as a function of energy in 316SS (30-y irradiation and 30-day decay).

In Figure D1, it can be seen that ^{60}Co at all activation energies serves as an adequate proxy for ^{63}Ni (whose activity is 10 times higher than ^{60}Co) and ^{3}H (whose activity is < 1/10 of ^{60}Co) for

30 days decay. However, after 30-40 years decay, there may no longer be an adequate proxy for them at lower beam energies, as the easily-detectable radionuclides decay more quickly (e.g., ^{60}Co proxy decays to a level that is a factor of 1,000 less than ^{63}Ni at 30 years, ~6 half-life of ^{60}Co, for the worst case of 17-MeV proton beam). For higher beam energies, ^{60}Co is a good proxy for at least 50 years decay. In this case, as long as the detection technique is demonstrated to be sufficient for ^{60}Co, it is sufficient for all radionuclides in most proton beam energy, irradiation time and decay time conditions.

Tables D3, D4, and D5 summarize the calculated results for both distal and proximal cases. The hard-to-measure radionuclides and the proxy radionuclide that may be used, along with limitations, are identified. The last column in each table identifies whether or not the hard-to-measure radionuclide is less than the ANSI SL values based on the conditions (including 1-W beam loss) described in the above paragraph. Clearly, the decision about whether or not the hard-to-measure radionuclides have the potential to exceed ANSI SLs is dependent on the particular situation.

Limitations on the use of proxy radionuclide approach are based on their relative radioactivity, along with their relative hazard, as identified in Table D2. Some limitations that may be drawn from Tables D3, D4, and D5 include:

1) For 316 stainless steel, ^{55}Fe may be a problem for long decay times for most proton energies. Its long half-life and production mechanisms cause it to be the predominant radionuclide by far when other radionuclides may decay away after 10-20 years. For distal case at 17 MeV proton beam, ^{58}Co can serve as proxy.
2) For copper, the proxy for hard-to-measure radionuclides covers a fairly wide range of irradiation and decay periods, in most cases for greater than 50 years decay.
3) For 6061 aluminum alloy, the proxies for ^{3}H cover a wide range of conditions. If pure aluminum is considered, the unusual case of ^{55}Fe as the only hard-to-measure radionuclide of concern in one case is no longer an issue.

Table D3: 316 SS

	Energy	Hard-to-Measure **Radionuclide**	**Proxy**	**Limitations**	**< SL?***
Distal	17 MeV	^{55}Fe	^{58}Co	Strongly dependent on irradiation/decay times. ½ y irradiation: 50 y decay. 30 y irradiation: 5 y decay.	Always
	65 MeV – 12 GeV	^{55}Fe, ^{3}H	^{54}Mn	Up to 10-20 y decay	Decay times > 5 y
Proximal	17 MeV	^{55}Fe	^{54}Mn	Up to 10 y decay	Decay times > 40 y
	65 MeV	^{55}Fe, ^{3}H	^{54}Mn	Up to 15-20 y decay	
	400 MeV	^{55}Fe, ^{3}H	^{22}Na	Up to 20-30 y decay	
	12 GeV	^{55}Fe, ^{3}H	^{22}Na	Up to 25-30 y decay	

Table D4: Copper

	Energy	Hard-to-Measure **Radionuclide**	**Proxy**	**Limitations**	**< SL?***
Distal	17 MeV	^{63}Ni	^{60}Co	Up to 30-40 y decay	Always
	65 MeV – 12 GeV	^{63}Ni	^{60}Co	None	
Proximal	17 MeV	^{63}Ni	^{60}Co	Up to 30-40 y decay	Irradiation times < 10 y
	45 MeV	^{3}H, ^{63}Ni	^{60}Co	None	No
	400 MeV	^{3}H, ^{55}Fe, ^{63}Ni	^{60}Co	None	No
	12 GeV	^{3}H	^{60}Co	None	No

Table D5: 6061 aluminum

	Energy	Hard-to-measure **Radionuclide**	**Proxy**	**Limitations**	**< SL?***
Distal	17 MeV	^{3}H	^{26}Al	None	Always
	65 MeV – 12 GeV	^{3}H	^{22}Na, ^{26}Al	None	
Proximal	17 MeV	^{55}Fe	^{54}Mn, ^{26}Al	None	> 20 y decay
	65 MeV	^{3}H	^{22}Na, ^{26}Al	None	> 30 y decay
	400 MeV	^{3}H	^{22}Na, ^{26}Al	None	> 50 y decay
	12 GeV	^{3}H, ^{7}Be	^{22}Na	Up to 20 y decay	> 50 y decay

Using the FLUKA code, comprehensive calculations have been conducted for the induced radioactivity profiles in common beamline and shielding materials in several typical beam loss geometries for proton beam from 160 MeV to TeV as a function of irradiation and decay time periods [CERN2011]. The calculated results are compiled into an analytic code ActiWiz that can calculate the radionuclide inventory and dose risk, when compared to appropriate clearance limits.

The induced radioactivity profiles calculated with FLUKA and ActiWiz for proton accelerators confirm the similarity between electron and proton accelerators, which support the concepts of "proxy radionuclides" and "surface maximum" in most cases.

Examples of induced radioactivity using FLUKA and ActiWiz are shown in Tables D6 and D7, which give the induced radioactivity profiles for steel shielding surrounding a 160-MeV proton beam dump (made of graphite and stainless steel) after 30-y irradiation and 2-y decay.

Table D6: Induced radioactivity for steel shielding surrounding a 160-MeV proton beam dump after 30-y irradiation and 2-y decay.

Radionuclide	Half-life	Volumetric Radioactivity (Bq/g)	ANSI SL (Bq/g)	SL Fraction
^{3}H	12.3 y	17	100	0.17
^{49}V	338 d	14	100	0.14
^{54}Mn	313 d	50	0.1	500
^{55}Fe	2.7 y	430	1,000	0.43
^{57}Co	272 d	10	1	10
^{60}Co	5.3 y	340	0.1	3400
^{63}Ni	100 y	14	100	0.14

1) Radionuclides contributed to more than 1% of the total radioactivity or SL are shown.

Table D7: Induced radioactivity profiles for steel and concrete shielding surrounding a 160-MeV proton beam dump (graphite and stainless steel) after 30-y irradiation and 2-y decay.

Component	Dump Core		Shielding	
Material	Graphite	Stainless Steel	Steel	Concrete
Radionuclide (Radioactivity in %)	^{3}H (100%)	^{55}Fe (76%)	^{55}Fe (49%)	^{3}H (75%)
		^{54}Mn (11%)	^{60}Co (39%)	^{55}Fe (11%)
Radionuclide (SL Fraction in %)	^{3}H (100%)	^{54}Mn (88%)	^{60}Co (87%)	^{22}Na (87%)
		^{60}Co (7%)	^{54}Mn (13%)	^{54}Mn (12%)

Tables D6 and D7 show that, for a steel dump and steel shielding:

1) The radionuclides of interest and proxy radionuclides are the same as those in electron accelerators,
2) Proxy radionuclides (^{54}Mn and ^{60}Co in steel; ^{22}Na and ^{54}Mn in concrete) dominate the total dose risk (> 99%), when normalized to the corresponding ANSI SLs, while hard-to-measure radionuclides (^{3}H, ^{55}Fe and ^{63}Ni) are insignificant though they may have higher radioactivity.

3) One exception is the graphite dump, which has only ^{3}H produced and no proxies can be used, due to graphite's low atomic number.

The exception of graphite case again illustrates the importance for individual sites to note that there are limitations (or bounding conditions) to the rules of the technical basis for volumetric radioactivity described in Appendicies C and D. Each site shall evaluate and identify such exceptions and address that in the site's clearance and release program, particularly in the measurement methods.

D3: Surface Maximum

The maximum radioactivity concentration in metals at proton facilities is on or very near to the surface. Surface measurements are sufficient to estimate volume activation throughout metals in proton accelerators.

As described in Appendix C, activation of concrete shielding walls in some beam energies produces a maximum radioactivity concentration away from the surface with the maximum radioactivity at ~10 cm depth that is only 2-3 times higher than that at surface, due to radionuclides from thermal neutron capture reaction [MASU2003; MASU2008; CARR2001]. Some heavy ion facilities have similar situations [STRA2008; STRA2010; FERT2007]. In these cases, surface measurements are likely to be adequate for materials in accelerators with a nominal adjustment to account for the situation that the maximum activation is not far from surface and is only a factor of 2-3 higher than the radioactivity on the surface.

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Appendix E: Technical Basis for Measurement Methods of Volumetric Radioactivity and Determination of Detection Capabilities

This Appendix shows examples of appropriate measurement methods (instruments and techniques), determination of the associated detection capabilities (i.e., detection thresholds and detection limits), and the IFB approach. Some technical details are available in [JLAB2009; SLAC2011a; SLAC2011b].

In this Standard, the term *detection limit*, (DL) is used to denote the smallest concentration of radioactivity that will produce an instrument response greater than the detection threshold with a specified statistical power (usually 95%). The term *detection threshold* (or decision threshold), (DT) is used to specify the net signal level at which it can be said that a measurement differs from background. These concepts have subtle but important distinctions, especially important when invoking IFB measurement protocols.

E1: Requirements for Detection Capability

Measurements for material clearance shall use appropriate radiation detection instruments and procedures. The ANSI N13.12-2013 SL values can serve as guidelines for selecting appropriate instruments and techniques with sufficiently low detection capabilities.

To satisfy a clearance criterion using the ANSI SLs, the sum of the SL fraction (called R value) for all potential radionuclides shall not be more than one, as shown in Eq. E1:

$$R = \sum_i (A_i / SL_i) \qquad \text{(Eq. E1)}$$

where:
A_i = Radioactivity per unit mass for radionuclide *i* (in pCi/g)
SL_i = ANSI N13.12-2013 volume Screening Level for radionuclide *i* (in pCi/g)

Implicit in Eq. E1 is the requirement that measurement methods for this clearance criterion have *detection limits* (DLs) for given proxy radionuclides less than the proxy SLs, such that Eq. E2 is also satisfied, *and* that in practice, the protocols provide conservative assurance that the total activity fraction (R) limitation is met based on measurement of proxy radionuclides.

$$DL_i / SL_i < 1 \qquad \text{(Eq. E2)}$$

where:
DL_i = Detection Limit (DL) for proxy radionuclide *i* (in pCi/g)
SL_i = ANSI N13.12-2013 Screening Level for radionuclide *i* (in pCi/g)

When IFB is used for the clearance criterion, the *detection threshold* (DT) must also be sufficiently low to satisfy the condition in Eq. E1. Here, a subtle distinction between DL and DT can be seen, in that it is possible in theory that an IFB measurement protocol may satisfy Eq. E1 (because the DT of the protocol has been shown to have sufficient sensitivity), but not satisfy Eq. E2 (because the DL of the protocol is > the SL). Another way to state this is that the DT is a

qualitative decision level, at which no claim of a quantitative *concentration measurement* is made; whereas the requirement in Eq. E2 implies a quantitative result that is compared to a limit. In practice, this condition is not normally an issue, as typical measurement protocols have DLs below the SLs, and therefore the DTs associated with the protocols are by definition low enough to satisfy detection sensitivity requirements, even though measurement results < DL cannot technically be reported in terms of quantitative concentration values. Section E5 contains further discussion of these criteria.

There are sensitive, hand-held instruments such as the scintillator-based survey meters that have detection capabilities for proxy radionuclides that are lower than the corresponding ANSI N13.12-2013 SL values. Quantitative estimation for the detection capabilities of the surface survey method, gamma spectrometry, and laboratory sample analysis have been estimated [SLAC2011a; SLAC2011b]. The results show that the detection limits for proxy radionuclides can be < 3 pCi/g for a 1"x1" scintillator probe, < 1 pCi/g for a gamma spectrometry in the field, and < 0.1 pCi/g for laboratory HPGe sample analysis (low-background environmental counting protocol).

Clearly the gamma spectrometry and the laboratory sample analysis have lower detection limits than surface surveys in the field. The gamma spectrometry and laboratory sample analysis, and other means such as portal gate monitor and bulk monitoring systems, can be used as benchmark or confirmatory measurements for field surface surveys when warranted, at the expense of higher operating cost and time.

E2: Surface Survey

For material clearance and release purposes, surface surveys shall be used to determine the potential presence of volumetric radioactivity.

When the IFB or ANSI SL clearance criterion is used, the surface survey method is the minimally required material clearance measurement for volumetric radioactivity.

When the IFB clearance criterion is used, the measurement methods shall have detection thresholds (DTs) that are less than the ANSI N13.12-2013 SLs. When the ANSI SL clearance criterion is used, the measurement methods shall have detection limits (DLs) that are less than the ANSI N13.12-2013 SLs.

The surface surveys shall use a sensitive radiation detection instrument operated in an acceptable ambient environment to achieve the needed detection capability. An example includes a NaI or plastic scintillator probe (1" diameter and 1" thick) for gross beta-gamma counting rate or micro-rem measurements. The survey may be in either the scanning or fixed-position measurement mode or a combination thereof at a specified distance above the surface. The measurements shall cover at least the material surface that is expected to have the maximum radioactivity concentration.

An analytic estimation of the detection limit for surface survey may be made based on the following information:

1) A rule-of-thumb [SLAC2011a] is that a volumetric radioactivity of 1 pCi/g for a proxy radionuclide (ANSI Group 1 gamma emitters) in metal or concrete produces about 1 μrem/h at 2 cm from the surface of an activated object (which is the effective distance for a 1"x1" scintillator meter at 1 cm from the surface).
2) An object with 3 pCi/g of proxy radionuclides (one SL for Group 1 radionuclides) would produce about 3 μrem/h on the object surface.
3) A 1"x1" scintillator meter, when operated in counting rate mode, has a nominal sensitivity of 200 cpm per 1 μrem/h (or 1 pCi/g) for proxy radionuclides, and it can detect ~140 cpm above a background of 700 cpm. This corresponds to a detection limit that is < 3 pCi/g of proxy radionuclides.
4) A micro-R meter can detect ~a few μrem/h above an ambient background of 5 μrem/h. This corresponds to a detection limit of a few pCi/g of proxy radionuclides.

Therefore, the detection limits for proxy radionuclides using common, sensitive hand-held survey meters can be less than the ANSI SL of 3 pCi/g for Group 1 emitters.

E3: Confirmatory Measurements

Confirmatory measurements may be used to supplement the surface surveys when warranted. Confirmatory measurements include, but are not limited to:

1) Measurements using a gamma spectrometer in the spectrometry mode or the gross count mode in the field.
2) Laboratory analysis of representative samples collected from the objects using the low-background HPGe or Liquid Scintillation Counter (LSC) analysis.
3) Bulk measurements, e.g., using portal gate monitor or a bulk monitoring system.

Field gamma spectrometry measurements may utilize a gamma spectrometer (e.g., either HPGe or NaI probe) to identify the potential existence of gamma emitters in materials.

Laboratory sample analysis involves collecting one or more representative core samples or surface swipe samples for the gamma spectrometry or LSC beta measurements using environmental low-background laboratory counting protocols.

Field gamma spectrometry and the laboratory sample analysis should have lower DLs for gamma emitters (typically 0.1-1 pCi/g and 0.01-0.1 pCi/g, respectively) than the surface surveys. However, they may not be practical, particularly the laboratory gamma spectrometry, to be used as routine or production type measurements for material clearance and release. Therefore, the field gamma spectrometry and the laboratory sample analysis may serve as confirmatory or benchmark measurements to the surface survey method.

Table E1 summarizes the measurement methods (surface survey and confirmatory measurements) and typical detection limits for volumetric radioactivity measurements.

The detection capability of a portal gate monitor is typically about 1 μCi of Co-60 point source or in the range of 0.1-1 pCi/g for proxy radionuclides with a uniform volumetric radioactivity in truckload or large volume container quantities.

There are a few special laboratory methods that may be used to measure the hard-to-detect radionuclides for confirmatory purposes. For example, tritium in concrete may be measured using laboratory processing techniques (e.g., combustion oxidation) in combination with LSC. DLs for such methods may be a small fraction of the ANSI N13.12-2013 volume SL value of 3,000 pCi/g for ^{3}H. The removable surface activity of ^{3}H in concrete may also be measured using a surface swipe with LSC measurement. DLs for these measurements are also typically below the ANSI SL and 10 CFR 835 surface activity limit of 10,000 dpm / 100 cm^2 for ^{3}H. The low-energy x-ray emitters such as ^{55}Fe may also be measured if a powdered sample is available for LSC measurement.

Some of these techniques may not be available to a given site. Sites should document types of confirmatory monitoring used and materials to which they apply, frequency of measurements or conditions triggering use in the technical basis.

E4: Estimation of Detection Capability for Surface Surveys

The detection capability (DT and DL) of a surface survey method for volumetric radioactivity depends on sensitivity of the measurement method (i.e., the instrument and technique), the composition of the material being surveyed, and the ambient background. This section shows how the detection capability for the surface scanning survey using a 1”x1” NaI probe operated in the counting rate mode may be estimated. More details are described in the references [SLAC2011a; SLAC2011b; CURR1968; ABEL1999].

Various definitions are in use to describe detection capability. This Standard invokes two conventional concepts in its approach to defining detectability. These are called the *detection threshold* (DT) and the *detection limit* (DL). These concepts are analogous to the *Critical Level* L_C (also called “decision level”) and the *Detection Limit* L_D respectively, popularized by Currie and widely adopted in analytical protocols [CURR1968].

Equations employing these concepts are published in numerous variations in the literature. The formulations below are reasonable approximations when background count rates are reasonably high [CURR1968; ANSI2004].

$$DT \approx 2.33\ \sigma_B \quad \text{(cpm)} \qquad \text{(Eq. E3)}$$

$$DL \approx 4.66\ \sigma_B / \eta \quad \text{(pCi/g)} \qquad \text{(Eq. E4)}$$

where σ_B is standard deviation of the background count rate (cpm) and η is the sensitivity in cpm/(pCi/g). Implied in these equations is that the standard deviation of the background is estimated for the same count time as the material measurement.

The units for DT are typically the “native” output of the instrument (e.g., cpm). The units for DL are in pCi/g. It can be seen that the DL is simply twice the DT count rate corrected for efficiency. In this Standard, DL is associated with a quantitative estimate of the smallest radioactivity that can be measured above the detection threshold with a specified statistical power; often called minimum detectable activity. DT is not associated with a quantitative radioactivity

determination, but only a *decision about the presence of radioactivity* above background. However, for purposes of quantifying detection capability and comparison with ANSI SLs (in units of pCi/g), the DT can be associated with a "critical concentration" or "decision level activity (DLA)", given by Eq. E5 [ANSI2004]. But this should never be described as minimum detectable activity, as it has little statistical power.

$$DLA \approx 2.33\ \sigma_B / \eta \quad (pCi/g) \qquad \text{(Eq. E5)}$$

It should be noted that these equations represent a starting point for determining the true detection capability (DT and DL) of a counting system. The sensitivity, background rate (and its variation), scan speed (e.g., ≤ 5 cm/s), meter response time (fast or slow time constant), as well as human performance factors, all affect the detection capability for a field measurement with portable instrumentation. In general, the detection capability becomes worse with a lower sensitivity, a higher background, a faster scan speed, or a fast response time. The "true" detection capability (or more realistic estimates) is obtained by making appropriate corrections to the above formulas associated with these "non-ideal" factors.

An assessment of the DT and DL (as applicable for field survey protocols used) should be conducted for a particular site's protocols. General detection capability parameters are discussed in this Appendix, but each site shall document details involving survey methods and site procedures in its technical basis. For a full treatment on the use of the concepts introduced here, see the references to this Appendix, in particular [NRC2009; CURR1968; ANSI2004; NRC2004].

In evaluating detection capability, the sensitivity η (in units of cpm/(pCi/g)) needs to be estimated. The sensitivity depends on the detector type, the material being surveyed (type, size and shape), the potential volumetric radioactivity profile of the proxy radionuclides inside the material, and the yields and energies of gamma radiation emitted by proxy radionuclides. Since potential volumetric radioactivity profile is likely not known, reasonable assumptions have to be made based on the process knowledge in order to estimate conservatively the detection capability in units of pCi/g. Therefore, the sensitivity can be better estimated with the calculations using a Monte Carlo code such as MCNP [MCNP2010] for a material with a proxy radionuclide in various volumetric radioactivity profiles. Other means of estimating sensitivity are also acceptable.

Figure E1 shows the calculations of the sensitivity for a 1"x1" NaI probe at 1 cm from the surface of a large iron piece (2 meter diameter and 20 cm depth) using MCNP [SLAC2011b]. The volumetric radioactivity profiles are expressed by an exponential-decreasing depth profile characterized by $\lambda = 0$ (i.e., a uniform depth profile), 0.1, 0.2 and 0.7 cm, as well as a Gaussian surface profile characterized by $\sigma = 5$, 10, 20 and ∞ cm (i.e., a uniform surface distribution). The surface profile depends on the beam loss condition (a point or line source) and the distance between the beam loss point and material.

Figure E2 shows the calculated sensitivity of 1"x1" NaI probe for various ^{60}Co radioactivity distributions within the large iron piece, which varies between 162 and 383 cpm/(pCi/g), using MCNP. Note that the uniform radioactivity profile gives the lowest sensitivity of 162

cpm/(pCi/g), and the value is also consistent with the aforementioned nominal sensitivity of 200 cpm per 1 pCi/g (or 1 µrem/h).

Based on the lowest sensitivity of 162 cpm/(pCi/g), Figure E2 also shows the "critical concentration" for a 1"x1" NaI probe operated at the slow response mode at a background rate between 200 and 700 cpm, and a scan speed of 2.5 cm/s or 5 cm/s. Note that the term "critical concentration" was used for the IFB clearance measurements [SLAC2011b], which is the same as the detection threshold in Eq. 3 (or DLA in Eq. 5). However, to account for the non-ideal factors, the formula of $4.66\sigma_B/\eta$ (instead of $2.33\sigma_B/\eta$) was used to calculate critical concentration [SLAC2011b]. It can be seen that the critical concentration for a 1"x1" NaI probe measuring potential ^{60}Co proxy in a large iron block can be less than the corresponding ANSI SL of 3 pCi/g for Group 1 emitters.

Table E2 summarizes the critical concentration for proxies of ^{22}Na or ^{60}Co for surface scanning using a 1"x1" NaI probe in aluminum, iron or copper. The required detection capability (3 pCi/g of proxies) can be satisfied in most cases. The slower 2.5 cm/s scan speed gives a critical concentration that is ~1.4 times lower than a 5 cm/s scan. Static measurements could have a factor of 2-3 lower critical concentration than the survey at 2.5 cm/s scanning [SLAC2011b].

The volumetric radioactivity in this MCNP study is the radioactivity averaged over an inverse-cone-shape region of 6-cm-depth and 20-cm-radius (about 4,000 cm^3 or 30 kg of iron) below the detector location. This region is called the "detectable region", a region from which gamma rays can be detected by the detector (which depends on material type and gamma ray energy). Note that the 1"x1" NaI probe cylinder in the calculations has a 6-mm-thick lead collar to reduce the background count rate, which also limits the "detectable region" from the sides.

If a smaller detectable region of 2-cm-depth and 8-cm-radius is considered, the sensitivity for a uniform radioactivity profile is reduced to 74 cpm/(pCi/g), which results in a critical concentration of 6 pCi/g at a background rate of 700 cpm.

The concept of "detectable region" has two implications for surface surveys of volumetric radioactivity:

1) If the potentially activated item being surveyed has a volume that is smaller than the "detectable region", the instrument response in cpm would be lower than the measurement of material with the same concentration of radioactive material that is larger than the detectable region. The detection sensitivity is then proportional to the volume of the item. The implication is that, to achieve an acceptable detection sensitivity, the object being surveyed should not be much smaller than the "detectable region". Therefore, it is a good practice to put together small items such as bolts or thin wires that have similar activation potential as a batch for survey.
2) To detect radioactivity beyond the "detectable region", scanning over the item surface or fixed-position measurements at locations separated by a distance no larger than the "detectable region" needs to be conducted. For fixed-position measurements the minimum distance between 2 neighboring fixed positions may be 10 cm for the case studied. A survey method can also be a combination of scanning and fixed-position

measurements (e.g., scanning along X direction and fixed positions along Y direction, i.e., $\Delta X=0$ and $\Delta Y=10$ cm). The above minimum distance between 2 fixed positions is for components close to a beam loss point. Components that are not close to the beam loss points will have a less-variant gradient of the induced radioactivity spatial profile and, therefore, can have a larger distance based on the actual field beam-target-component geometry.

To ensure the surface survey has sufficiently low detection capability, the survey should be conducted at acceptable background levels, e.g., no more than 700 cpm for a 1”x1” NaI probe [SLAC2010].

A rough comparison of the 1”x1” and 3”x3” NaI detectors (or gamma spectrometer) can be made based on the detector’s surface area and the counting time. For example, a 3”x3” NaI probe has a nominal sensitivity about 10 times higher than that of 1”x1” NaI probe due to its larger surface. However, the background signal detected also increases by ~10 times for 3”x3” NaI probe. Therefore, with an equal counting time, the 3”x3” detector would have 3 times better detection capability than the 1”x1” detector, if both detectors are operating in the same mode, e.g., gross counting mode.

Table E1: Measurement methods and their typical detection limits for volumetric radioactivity in metal and concrete (for both proxy and hard-to-measure radionuclides).

Method	Radionuclides of Interest	Instrument & Method	Detection Limit
Surface Survey	Proxies (^{22}Na, ^{54}Mn, ^{60}Co, ^{152}Eu) [1]	Surface survey for gross beta-gamma counting rate using 1"x1" scintillator-based meter in scanning or fixed-position measurement mode	< 3 pCi/g
Field Gamma Spectrometry	Gamma emitters with energy > tens of keV	Surface measurements using a gamma spectrometer in a fixed position measurement mode	< 1 pCi/g
Laboratory Core Sample HPGe Measurements	Gamma emitters with energy > tens of keV	At least one core sample collected per item and counted using low-background HPGe system with an environmental counting protocol	< 0.1 pCi/g
Laboratory Core Sample LSC Measurements	Beta or low-energy x-ray emitters (^{55}Fe, ^{3}H) [2]	At least one powdered sample collected per item and counted using LSC with an environmental counting protocol	^{3}H in concrete: 10 pCi/g
Laboratory Surface Swipe Sample LSC Measurements	^{3}H in concrete	At least one surface swipe sample collected per item and counted using LSC with an environmental counting protocol	15 dpm / 100 cm^2

1) ANSI N13.12-2013 Screening Level (SL) for proxy radionuclides is 3 pCi/g.
2) ANSI N13.12-2013 SL for ^{3}H is 3,000 pCi/g for volumetric radioactivity and 1.3x10^6 dpm / 100 cm^2 for surface activity.

Table E2: Critical concentration for surface survey method using a 1"x1" NaI probe for metals calculated with MCNP.

Material	Scan Rate (cm/s)	Gamma Energy (MeV)	Proxy Radionuclide	Critical Concentration (pCi/g) [1]
Aluminum	2.5	0.511 and 1.275	^{22}Na	2.3
Aluminum	5	0.511 and 1.275	^{22}Na	3.2
Iron	5	0.122	^{57}Co [2]	25
Iron	2.5	1.17 and 1.33	^{60}Co	1.5
Iron	5	1.17 and 1.33	^{60}Co	2.1
Copper	2.5	1.17 and 1.33	^{60}Co	2.4
Copper	5	1.17 and 1.33	^{60}Co	3.4

1) Critical concentration estimated with the formula of $4.66\sigma_B/\eta$ (instead of $2.33\sigma_B/\eta$) at a background rate of 400 cpm.
2) ^{57}Co is not a proxy radionuclide and its critical concentration is ~10 times higher than that for proxies.

MDA Calculations Using MCNP

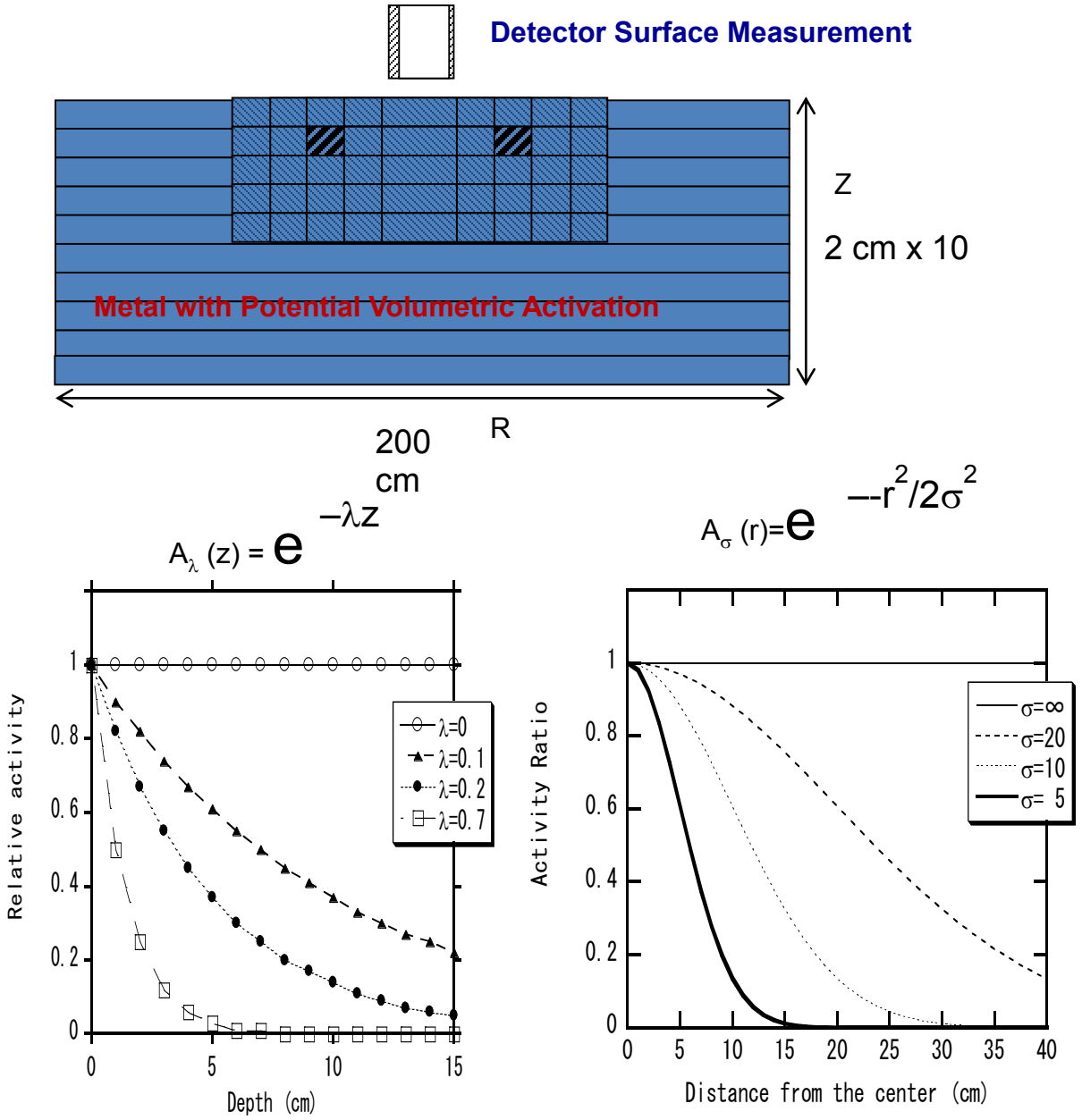

Figure E1: The geometry and volumetric radioactivity profiles used in MCNP calculations of the sensitivity for surface survey of a large iron piece using a 1”x1” NaI probe.

Radioactivity Profile		Sensitivity, η [cpm/(pCi/g)]
Depth	Radial	
λ = 0	σ = 5	383
λ = 0	σ = 10	235
λ = 0	σ = 20	181
uniform	uniform	162
λ = 0.1	σ = ∞	162
λ = 0.2	σ = ∞	163
λ = 0.7	σ = ∞	164

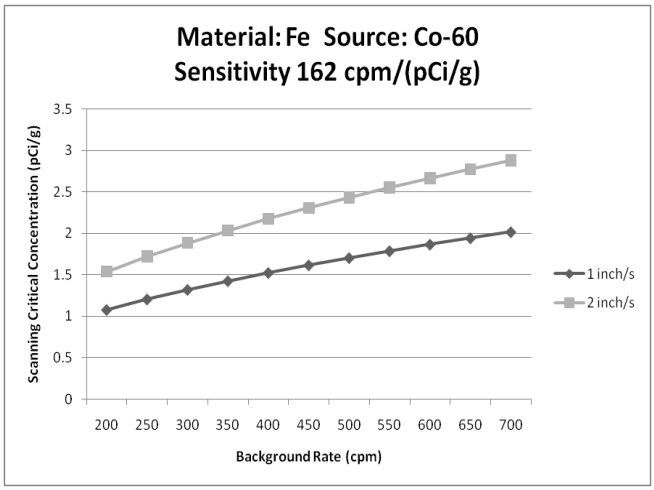

Figure E2: MCNP calculations of the sensitivity of the 1"x1" NaI probe for ^{60}Co in various volumetric radioactivity profiles in a large iron piece and the derived critical concentration in two scan speeds.

E5: Definitions of Detection Capabilities and IFB Approach

In contrast to measuring surface activity, when conducting surface surveys for volumetric radioactivity, parameters needed for determining actual radionuclide concentrations are not usually readily available. Although Section E4 of this Appendix demonstrates that the detection capability of surface survey technique can be less than ANSI SLs for proxy radionuclides, quantitative measurements of radioactivity concentration in M&E are not practical or necessary for most volumetric clearance survey methods. In keeping with the clearance criteria hierarchy of this Standard, the first level of clearance is that of indistinguishable from background (IFB). When radioactivity is detected above background, the requirements for clearance are more onerous. These include quantification of the radioactivity, satisfying additional ALARA objectives and meeting stakeholder expectations. Due to the difficulty of quantifying volumetric radionuclide concentrations using field instrumentation, for most clearance efforts, IFB is the simplest and most appropriate approach.

It is vital to note the IFB criterion is not based on the same concept as the "detection limit" or Minimum Detectable Concentration (MDC). Instead, as described in Section E4, IFB is directly analogous to the concept of the critical level (L_C) in statistical evaluations. When using an IFB approach, one applies a hypothesis test in which the null hypothesis (H_0) is that there is no radioactivity above background in the sample. To reject the null hypothesis, one needs to establish only the "net signal level" (DT) at which there is a specified probability that the result is greater than background. The probability is customarily chosen to give a 5% likelihood of deciding radioactivity is present when in fact it is not (false positive). This concept is well developed in the literature and described fully in MARSAME (where it is referred to as

"Scenario B") and other references. By contrast, in the hypothesis test usually used for release decisions based on a quantifiable detection limit (L_D), the null hypothesis is that the radioactivity exceeds the detection limit. In this case, measurements below the L_D provide evidence allowing rejection of the null hypothesis. This does *not* mean that activity has not been detected.

As recommended in MARSAME and other references, when a *detection decision* is required, the detection limit (or MDC) should only be used as a measurement quality objective (MQO) for the measurement method. The detection decision should be made by comparing the measurement result to the detection threshold (DT) and never to the detection limit (DL). For an IFB survey method, the DT (as determined for a specific survey type) should be used for making determinations of the presence of radioactivity above background.

In the IFB scenario, a statistical Type I decision error results in a "false positive" determination from a measurement, whereas a Type II error results in a "false negative" outcome. In Figure E3, it can be seen that setting the IFB decision level at L_C results in a Type I error probability of α, which is traditionally chosen to be 0.05. Smaller values for α give a larger value of L_C (and lower false positive error rate). Technically, the probability of making a false negative decision depends on the amount of radioactivity in the sample. But it is clear that choosing larger DT values increases the false negative error probability. Also, the meaning of false negative is not the same when using IFB as it is when using the DL as the release limit. A false negative for IFB means the item was declared *free of radioactivity above background* when in fact it has such activity. A false negative when using the DL means the item was declared to have activity *below the DL value* when in fact the activity is at or above the DL. This is an important distinction when defining a site's clearance criteria. If the clearance criterion is defined based on DL, then items measuring less than the DL should not be referred to as IFB. The error of using L_D (or the DL) as a *detection threshold* can clearly be seen in Figure E3. This would result in a false-negative detection decision for half of the measurements when the actual radioactivity is at L_D.

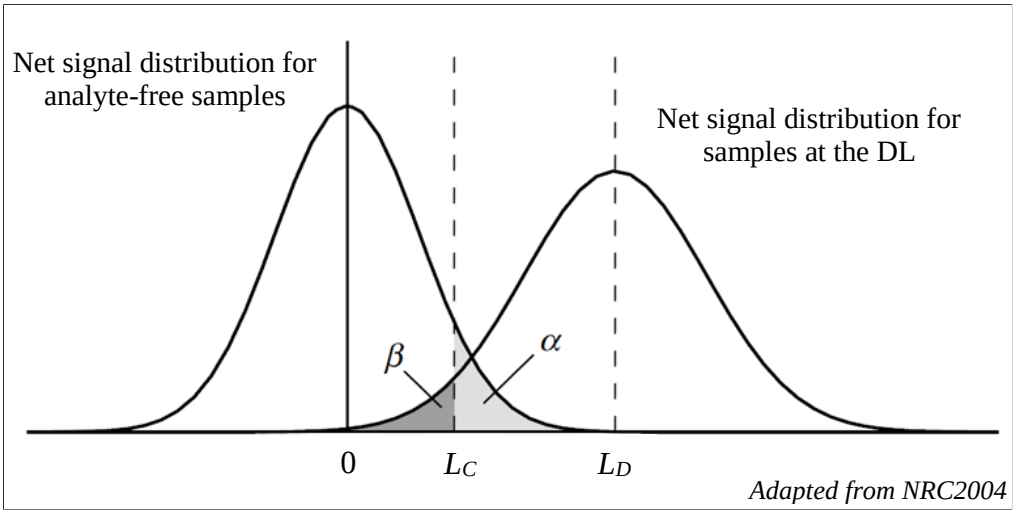

Figure E3: Distribution of net counts related to detection and quantification levels.

By definition, in the IFB clearance criterion, there is little statistical power to quantify levels of radioactivity in a sample with net signal ≤ L_C. But the relationship of the statistical parameters provides good assurance that using the L_C detection decision yields a conservative result when using field measurements as described, under the constraints of appropriate survey procedures. The detection threshold is of order one half the traditionally obtained "detection limit", in terms of net signal strength (counts or dose rate). Use of the DT rather than DL produces a higher statistical power to avoid Type II (false negative) error and helps ensure that potential quantities of radioactivity in M&E that is released are significantly below the applicable Screening Level quantities related to the primary dose constraint of 1 mrem/y.

IFB measurement decision levels are based on the estimation of standard deviation of the background count during the measurement, but when conducting field surveys, should also include corrections and adjustments due to human factors and survey technique (scan versus fixed position counting, etc.) as discussed in Section E4.

The references provide guidance for developing specific formulae to address site-specific survey methods. Using these guidelines, an example for determining the IFB detection threshold is given below. Other technically sound approaches may be used to determine detection capability.

Example

We assume a background of 800 cpm using a 1"x1" NaI detector with a 5 cm/sec scan speed, and an activated region 10 cm in radius, such that the probe is in the vicinity of the region of interest for about 4 sec. For comparison, using the simple assumptions for a static count, we can calculate DT from Eq. E3, and obtain DT = 2.33 $\times \sqrt{\frac{800\ cpm}{0.067 min}} = 254\ cpm.$ Using the minimum sensitivity of 162 cpm/(pCi/g) given in Section E4, this implies a qualitative "DLA" of about 1.6 pCi/g and a DL of about 3.1 pCi/g. This result is acceptable for an IFB clearance decision where the DL is used only as a measurement quality objective (see above). However, setting the IFB level at this level of performance may be unrealistic for use with field survey instruments and techniques, resulting in an unacceptably high false positive rate.

Application of signal detection theory (SDT) is one acceptable approach to account for the non-uniformities inherent in field surveys. The theory applies statistical decision techniques to the detection of signals in noise by human observers [ABEL1999]. In this approach, a value called the index of sensitivity, d' represents the number of normal deviates between the background and net signal means of normally distributed radioactivity (analogous to the points of interest in Figure E3). These indices are tabulated for chosen values of true positive and false positive error rates. Table E3 below gives an excerpt of these values.

Table E3: values of d' for selected true positive and false positive proportions [ABEL1999]

False Positive Proportion	True Positive Proportion			
	0.65	**0.75**	**0.85**	**0.95**
0.05	2.02	2.32	2.68	3.28
0.10	1.66	1.96	2.32	2.92
0.15	1.42	1.72	2.08	2.68
0.20	1.22	1.52	1.88	2.48
0.25	1.06	1.35	1.72	2.32
0.30	0.91	1.20	1.56	2.16
0.35	0.77	1.06	1.42	2.02
0.40	0.64	0.93	1.30	1.90
0.45	0.52	0.80	1.17	1.77

For an ideal observer, the number of source counts in an interval (s_i) required for a specified level of performance (represented by d') is given by:

$$s_i = d' \sqrt{b_i} \qquad \text{(Eq. E6)}$$

Where b_i is the number of background counts in the observation interval. The similarity to the standard Poisson formulation for DT can be seen.

The next factor we need to address is surveyor efficiency. Above, we assumed an ideal observer, but field surveys must take human efficiency into account. It is noted that an ideal observer is not error-free, but operates at the optimum efficiency. Detectability is a function of the square root of efficiency [EGAN1975], where efficiency is defined as the proportion *p* of events registered. Estimates of surveyor efficiency should take into account the specifics of the human interface with the survey meter being used and typical conditions of use. Suggested values range from 0.25 to 0.8. In this example, an efficiency value of 0.50 is used, following [ABEL1999]. Taking efficiency into account, detectable source counts in the interval of interest become:

$$s_i = d' \frac{\sqrt{b_i}}{\sqrt{p}} \qquad \text{(Eq. E7)}$$

Continuing the example, a value is chosen for d'. We will simplify this example, using a single value that might be appropriate for the scan phase of the survey. In most cases, scan surveys are treated as two-step processes, where a surveyor scans and pauses over an area. We will assume constant scanning, with an observation interval of 4 sec as discussed above. In choosing d', the true positive rate is chosen as 0.95 (consistent with conventional levels of Type II error rate) and we choose a false positive rate of 0.30. This is in light of the two-step scan process, which will further reduce false positives in the second step. Higher levels of false positive rates may be acceptable, depending on the type of survey and the consequences of false positive errors. The value for d' is then 2.16. The average background counts in the 4-sec observation interval are:

$800cpm \times \frac{4sec}{60s/min} = 53.3$ counts. We then have the following;

$$s_i = 2.16 \frac{\sqrt{53.3cnt}}{\sqrt{0.5}} = 22.3 \text{ counts in 4 s} = 335 \text{ cpm}$$

This estimate of detection sensitivity does not perfectly represent the statistically derived DT or DL, but provides an estimate of the expected performance of a surveyor in the field. This result would be used as the IFB criterion for clearance and imply a qualitative "DLA" of about 2.1 pCi/g for the proxy. This sensitivity could be improved by lowering the background count rate or slowing the scan speed. The DLA may be adjusted by other factors related to the specific survey conditions.

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