Thomas Jefferson National Accelerator Facility

Jefferson Lab

Technical Basis for Release of Solid Materials From Radiological Control When Residual Radioactivity Levels are Indistinguishable From Background

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Technical Basis for Release of Material as Indistinguishable From Background

I. Scope

This document contains the Jefferson Lab Radiation Control Department policy for the release of equipment and material from accelerator enclosure areas where there is potential for induced radioactivity to occur in the material. It does not address release protocols applicable to potential surface contamination.

II. Purpose

The purpose of this document is to establish the protocol for release of materials from accelerator enclosures whereby the criterion for release is that no radioactivity above background can be detected in the material using appropriate, commercially available instrumentation. The approach we use involves the following steps. We will:

- Estimate the detection threshold, or "decision level", and Minimum Detectable Concentration for the Bicron/Thermo Microrem survey instrument when used to conduct release surveys.

- Compare these detection capabilities to current industry consensus standards for unrestricted release of material.

- Establish the conditions under which this method of release survey is appropriate, and describe documentation requirements for material release using the protocol discussed.

This document also provides a convenient compilation of some important process knowledge elements related to material release decision-making.

III. Introduction and Executive Summary

As a result of adoption of DOE Order 5400.5 into the JSA contract, it is incumbent on Jlab to ensure the methods used for the release of material from radiological control are consistent with the requirements in the Order. Since O 5400.5 only provides specific release criteria for surface contamination, and materials residing in accelerator enclosures have more potential for volumetrically induced radioactivity than surface contamination, a technical basis is needed to support release decisions for such material.

According to O5400.5, release of property, equipment and items with potential volumetric radioactivity requires the application for and approval of authorized release limits from DOE Headquarters. However, subsequent guidance has provided some clarification on this issue. Though the vast majority of guidance for material release is aimed at surface radioactivity, portions of the guidance applicable to volumetric radioactivity allow for field element approval of release limits and protocols for such materials under some conditions [DOE, 1995; DOE, 1997c; DOE, 2002]. Specifically, when potential doses involved are negligible, and applicable requirements for data reporting and maintenance of records are met, the approval process is more flexible.

In addition, when materials can be demonstrated to contain no detectable radioactivity above background, DOE guidance allows for release of the material without reference to a specific limit. This guidance consistently maintains that *in the absence of detectable radioactivity, materials and equipment can be released after documenting the surveys in accordance with approved procedures.* In this case, the release criterion is defined as "Indistinguishable From Background"(IFB).

Authorized release limits are developed based on the premise that detectable amounts of radioactive material may be present on/in the material, and in many cases, the limits are established for a restricted or prescribed end use of the property [DOE, 1997b; DOE, 1997c]. Such limits are always prescribed in terms of actual quantities of radioactivity. Guidance for release of materials under authorized limits also presumes the presence of measurable radioactivity [DOE, 2002]. It is stressed that materials released under the provisions of our IFB protocol are not expected to contain any residual radioactivity, and that *any detectable activity* disqualifies the material as a candidate for release. This approach is also consistent with DOE directives and activities aimed at satisfying the requirements for the release of metals from radiological areas for recycling [DOE, 2001]. This is significant, as much of the material released from radiological areas is metal.

This document establishes the technical basis for the release protocol used at JLab, under which, released materials are determined to have no detectable residual radioactivity above background. As such, these materials are not considered radioactive effluents or emissions. This document is not intended to be the basis for an authorized release limit, but will be transmitted to the Jefferson Site Office (TJSO) for review, as appropriate. This document does not address clearance of items based on surveys for surface contamination.

When release limits are explicitly stated or derived from dose limits, there is generally one quantity or concentration applicable for a given nuclide/matrix. This value is referred to as the derived concentration guideline level (DCGL) for release. Industry guidance normally recommends that the Minimum Detectable Concentrations (MDCs) for the equipment and methods used be held to some fraction of the DCGL [MARSSIM, 2000; NUREG, 2002]. Again, in the present case, the release criterion is the detection limit; that is, a decision to release material above the detection limit will not be made, in part, because no direct quantification of radioactivity is being made (released materials are not characterized as having activity below some value). This arises due to the complication with potential volumetric activity that the measurement sensitivity depends on the dimensions, mass, volume and type of material being surveyed. Therefore, there is in reality, a range of MDCs related to the variation in those and other factors that is not readily parameterized. The approach used to estimate the MDC must therefore be reasonably conservative, in light of this variation, and the survey protocol must be prescribed such that this variation is constrained.

Historically at Jefferson Lab (JLab), Stapleton studied release criteria and detection sensitivity, and established the technical basis for release surveys, recommending a threshold of about 20 µrem/hr, based on recommendations and practices in place at the time [Stapleton, 1990]. The Thermo/Fisher Microrem® portable survey meter[(1)] has been chosen as the instrument with which release surveys are conducted. Standard operating procedures conservatively constrained the release limit to an assigned detection threshold of twice the ambient background, when surveying in a low background [JLab, 2008a]. This effectively equates to a nominal threshold of about 10 net µrem/hr, in an area with typical background. While this approach is consistent with industry experience and practice [Goles, 1991; NUREG, 1992; NUREG, 1998a; Walker, 1994], the full implementation of O5400.5 provides an opportunity to reassess the detection sensitivity, document the approach used to derive it, to evaluate this in light of the concept of “indistinguishable from background”, and to benchmark the results against the current industry consensus on appropriate criteria for unrestricted release.

[(1)] Instrument operations manuals and technical specifications are on file in the RadCon department

Subsequent to Stapleton's review, a considerable amount of regulatory guidance has been developed containing recommended DCGLs for unrestricted release of solid materials. These recommendations are uniformly derived from the widely adopted upper bound of 1 mrem/y potential dose to a member of the public. DOE employs the 1 mrem/y dose guideline in various guidance documents, but does not currently provide the associated DCGLs for volumetric activity in solids. We here evaluate our IFB protocol against the industry consensus DCGLs for unrestricted material release, demonstrating that the sensitivity of the protocol is conservative (estimated MDC does not exceed the consensus DCGLs), and that any potential exposures are ALARA.

Industry guidance stresses the importance of conducting specific field-testing studies to validate chosen techniques for release surveys whenever possible, although this is an effort acknowledged to be beyond the reach of many radiation protection organizations. We have examined our clearance practices in light of a large body of experience, evaluated computational and statistical approaches, and performed limited empirical testing of performance of individuals conducting the surveys. We also used this opportunity to define and document a set of constraints that enhance detection sensitivity and limit its variation, under which release surveys may be conducted, thus strengthening confidence in the protocol.

IV. Instrumentation

As noted above, the instrument used at Jlab for material release surveys is the Thermo/Fisher (formerly Bicron) Microrem survey meter. This instrument has been in use at Jefferson Lab for two decades, and has been the instrument used exclusively for release surveys for most of that time [May, 1991]. The Microrem employs a nearly tissue equivalent organic (plastic) scintillation detector, with an active volume of about 13 cubic centimeters. The tissue-equivalent response of the meter overcomes problems of energy dependence found in many micro-R meters. In addition to the tissue-like response of the scintillator, the output pulses of the photomultiplier are processed by a charge-integrating circuit to produce an energy-weighted signal that is proportional to the dose equivalent. This signal drives the rate meter and audio counter circuits. Two enhancements of the instrument over the years have improved sensitivity and effectiveness of the meter; the addition of a built-in audio counter, and the addition of the low-energy 'LE' option, which extends the energy response below 20 keV. These instruments exhibit a nearly flat energy response from 17 keV to 1.3 MeV (see Fig. 1). The thin window of this instrument allows detection of low energy gamma and x-rays, and even beta particles, enhancing detection sensitivity. We believe this to be the only "micro-R" type instrument to possess this capability. Although all instruments in the current stock of meters at JLab do not have both features, these features are assumed to be present on any meter used for a release survey (this is discussed later in the context of "constraints" applied to the release protocol).

The detector sensitivity of the Microrem is given by the manufacturer as approximately 100 cpm per µrem/hr. For our purposes a more useful relationship is the audio signal click rate. The audio circuit produces an audible click rate that is proportional to the displayed dose rate. The relationship in click rate to dose rate was found experimentally to be approximately 20 cpm per µrem/hr. In a typical low-background area this results in audio count rates of about 150-250 counts per minute. This will be discussed in more detail later, in the context of scan survey detection sensitivity.

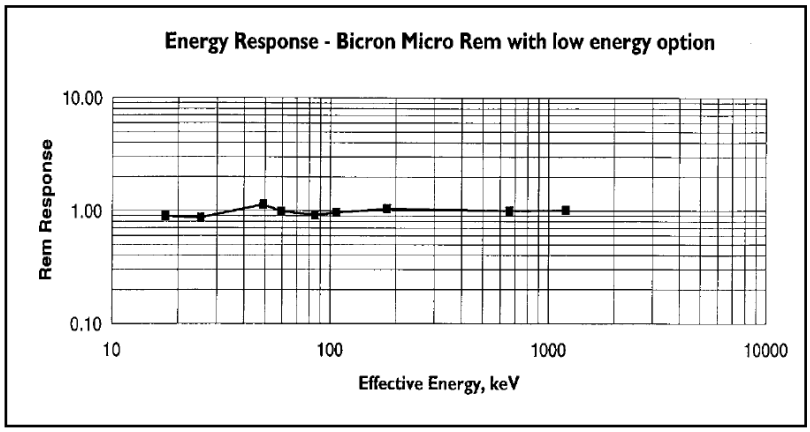

Figure 1 – MicroRem LE Energy Response

V. Description of Source Term and Types of Materials Being Monitored

The potential radionuclides produced in materials and equipment within accelerator enclosures has been studied in great detail. We refer the reader to the references for a full discussion of the theory and computational approaches that have been developed to evaluate activation from electron beams [IAEA, 1979; Barbier, 1969; Saxon, 1969; Ulrici, 2006; DeStaebler, 1963; Moe, 1991]. Swanson gives estimates for production cross sections, saturation activity as a function of beam power, and examples from practice of nuclides detected in various materials [IAEA, 1979]. Swanson's method has become the basis for a standard approach to evaluating radioactivity produced by electron beams. Fasso compared this technique to an approach using the FLUKA radiation transport code, and had statistically better agreement with experimental results [Fasso, 1999]. However, any prediction of potential activation in some real medium must also consider general characteristics of the materials being irradiated and the conditions of exposure. This process knowledge is a key input for evaluating items and equipment for activation.

This document deals only with solid materials that may routinely be candidates for release from control. Some activation products potentially present in these materials may also occur in other media but we will be concerned here only with volumetric radioactivity in solid objects/materials commonly present in - and released from - beam enclosures. Activation products that could be expected to become deposited in or on such items are assessed through surface contamination measurements or analyses of the media through means other than a direct survey of the item.

The materials of concern fall into two general categories; (1) metals, and (2) relatively low-Z organics, glass/ceramics, plastics, composites, etc. Concrete might constitute a third category altogether, partly due to the potential for some material-specific activation products, and due to potential self-shielding (an issue in any large, bulk item). Concrete can be surveyed for release using the direct scan technique, but process knowledge must be applied to evaluate the exposure history for the material in question. Standard operating procedures will identify constraints and triggers for applying additional means of analysis. Concrete activation is discussed further in subsequent sections of this document. Materials to be surveyed can also be classified by their proximity to the beam-line proper, as susceptibility for activation depends heavily on this parameter.

Beam-Line Components

Radiation and radioactivity production at electron accelerators is dominated by photonuclear processes that occur in a fairly limited physical region around the beam path. Secondary particles such as neutrons contribute to activation of a more distributed nature, further from the beam line. Nuclides of concern in beam-line related items are therefore almost all photo-activation products. Materials subject to this type of activation are those present in the beam line vacuum chamber, dumps, targets and their systems, supporting structures, magnets and diagnostic equipment. These components are most commonly made from stainless steel, copper, aluminum and common (carbon) steel. Experience at Jefferson Lab is consistent with published data concerning the radionuclides produced in these materials [May, 1996]. Table 1 depicts some radiological properties of photo-activation products with half-lives greater than a few days that can be produced in beamline components.

Material	Nuclide	Production	A_{sat} [a]	Half-life	Decay modes	Photon energy per decay (MeV)[b]
Carbon steel (iron) [c]	Sc-46	Fe-54 (γ,sp)[f]	7.4	83.9 d	β^-	2.0
	V-48	Fe-54 (γ,sp)	15	16 d	ε, β^+	2.78 (incl. annih. photons)
	Cr-51	Fe-54 (γ,sp)	15	27.8 d	ε	0.03
	Mn-54	Fe-56 (γ,np)	22	303 d	ε	0.83
	Fe-55	Fe-54 (γ,sp)	490	2.6 y	ε	0.0016 [d]
Stainless steel [e]	Sc-46	Fe-54 (γ,sp)	5.1	83.9 d	β^-	2.0
	V-48	Fe-54 (γ,sp)	10.5	16 d	ε, β^+	2.78 (incl. annih. photons)
	Cr-51	Fe-54 (γ,sp) Cr-52 (γ,n)	96	27.8 d	ε	0.03
	Mn-54	Mn-55 (γ,n) Fe-56 (γ,np)	27	303 d	ε	0.83
	Fe-55	Fe-56 (γ,n)	338	2.6 y	ε	0.0016 [d]
	C0-56	Ni-58 (γ,np)	1.8	77 d	ε, β^+	2.74 (incl. annih. photons)
	Co-57	Ni-58 (γ,p)	22	270 d	ε	0.12
	C0-58	Ni-60 (γ,np)	0.7	71.3 d	ε, β^+	0.96 (incl. annih. photons)
	Co-60	Ni-61 (γ,p) Ni-62 (γ,np)	0.46	5.27 y	β^-	2.5
	Ni-63	Ni-64 (γ,n)	0.7	101 y	β^-	No gamma
Copper [c]	C0-58	Cu-63 (γ,sp)	24	71.3 d	ε, β^+	0.96 (incl. annih. photons)
	Co-60	Cu-63 (γ,n2p)	24	5.27 y	β^-	2.5
	Ni-63	Cu-65 (γ, np)	17	101 y	β^-	No gamma
Aluminum [c]	Be-7	Al-27 (γ,sp)	4.8	53.6 d	ε	0.048
	Na-22	Al-27 (γ,3n2p)	9.3	2.62 y	β^+	2.2 (incl. annih. photons)

Table 1 - Properties of nuclides produced in beamline materials

[a] Saturation activity in GBq/kW.
[b] Sum of the yield-weighted energy of photons with yields greater than a few percent
[c] Values for A_{sat} taken from IAEA, 1979
[d] Low energy x-rays (no gamma emission)
[e] A_{sat} calculated from cross section data in Barbier, 1969.
[f] sp = spallation reaction

Several important generalizations can be drawn from these data. The saturation activity can be used as a rough indicator of the overall importance of the nuclide of interest, but it must be remembered that saturation will never be approached for nuclides having long half-lives. Activity approaching saturation can be assumed when buildup times exceed two to three times $T_{½}$. As a group, activation products occurring in these materials are gamma emitters with reasonably high photon energy and abundance (including, in some cases, decay by positron emission, with the accompanying 511 keV annihilation radiation). In practice, after days to weeks of decay, the activation encountered in steel (including stainless) beam-line components is always dominated by Cr-51, Mn-54, Co-57 and Co-58; activity in copper by Co-58 and Co-60, and the activity in aluminum is dominated by Na-22. In the first hours to days after irradiation, a number of shorter-lived nuclides are also present. Since we are examining the detection sensitivity of our release method, the limiting case (most conservative) is to assume that the method depends on detection of the longer-lived nuclides after some decay time (the short-lived nuclides reach saturation quickly, are generally present in higher concentrations, and are almost exclusively positron emitters, making detection easy via the annihilation photons).

Fe-55 and Ni-63 emit no high-energy photons, and detection of these nuclides requires specialized techniques. However, these nuclides will always occur in the presence of other, easier to detect nuclides. In the case of Ni-63, due to its long half-life, the activity build-up rate is extremely slow, and its activity will always be a negligible fraction of the gamma emitters.

In the special case of components made of iron (significant quantities typically only found in the iron cores of magnets), Fe-55 may be the most prevalent long-lived nuclide. However, in real components, several moderating factors reduce the potential that some component containing a significant quantity of Fe-55 may be inadvertently released due to the absence of detectable activity. Iron (or carbon steel) as a beamline component is generally only used for magnetic cores. These components always contain other materials (generally copper) as integral parts of their construction. Also, due to their nature, these components are rarely removed from beam enclosures, and even more rarely permanently decommissioned (and hence potentially released to the public). And, as mentioned above, several other key nuclides are produced in iron. These nuclides reach saturation relatively quickly, and appear in detectable quantities at low beam power absorption rates. The graph shown in figure 2 depicts buildup of activity in iron in the case of an operating regime with 90 days of continuous operation followed by 90 days of shut-down time (roughly analogous to CEBAF operational cycles). The data indicate that the total activity of readily detectable gamma-emitting activation products may be a factor of five to ten lower than that of Fe-55. Components other than magnets are generally made of steel alloys, resulting in higher levels of the gamma emitters noted, and several additional nuclides.

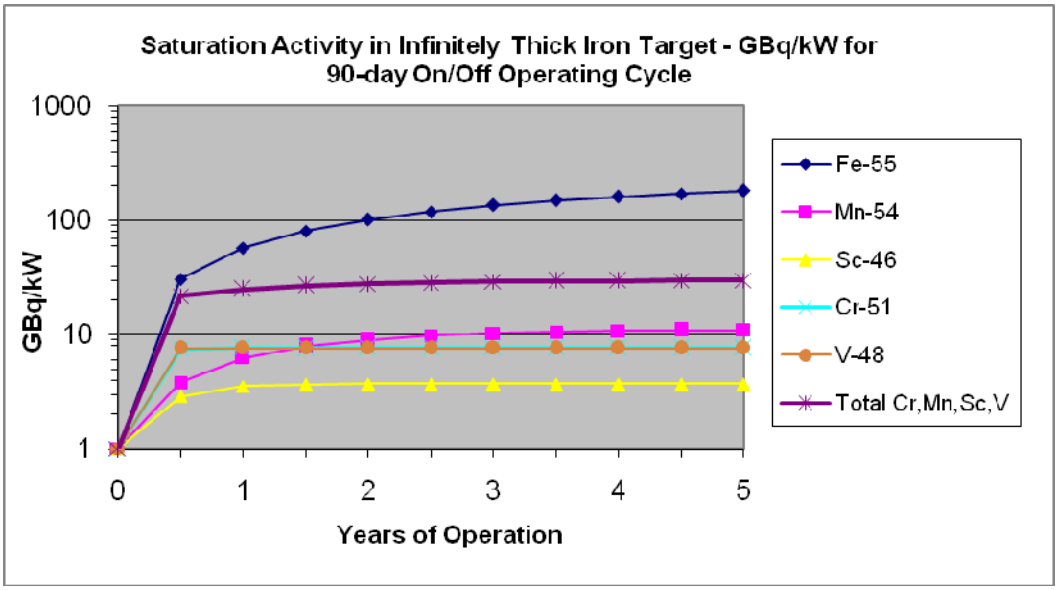

Figure 1 - Average Activity in Iron

Notes:
1 Saturation activity values taken from [IAEA, 1979]
2 Activity averaged over six month periods
3 Buildup of Fe-55 reaches ~ 75% of 20-year concentration in 5 years

Note that the activity values shown are for an infinitely thick target completely absorbing the entire beam energy, and the activity in a real beamline component will be many orders of magnitude lower. The actual specific activity of these nuclides at saturation in iron is expected to approach a few tens of Bq/g per Watt of power lost in nearby thick targets [Fasso, 1999]. Since saturation is reached relatively quickly for these nuclides, and readily detectable levels are produced from small beam losses, their production provides a sensitive detection surrogate for Fe-55. Pathway analyses indicate that potential doses from Fe-55 are on order 100 times lower than from equal concentrations of the gamma emitters under conservative conditions associated with the release of materials to the public [IAEA, 1996a; IAEA, 1998; ANSI, 1999]. The sensitivity of our release methodology ensures that no significant doses can occur from potential undetected activity contained in released materials.

Materials in the second category described above (dominated by low-Z hydro-carbon polymers) are used incidentally in items such as magnets, cabling and diagnostic equipment in the form of conductor insulation, packaging, and electronic components such as circuit boards. The nature of the materials in the second group results in activation dominated by the short-lived isotopes C-11, N-13, and O-15. All have half-lives in the minutes. In practice, activity in components such as cables and magnet windings is dominated by activation of the metal conductor. The combination of target path length, cross sections for activation, and half-life of the activation products results in the bulk of long-lived activity occurring in the metallic components themselves.

The longer-lived nuclides Be-7 and H-3 (tritium) can be produced in these materials, however, production cross-section are much lower than for the short lived nuclides. Be-7 is detectable through gamma surveys, having a photon energy of 477 keV. Activity potential may be roughly analogous to that in water (given similar radiation length, cross sections, effective A and Z, etc.). Reactions leading to production of these nuclides all have yields of order 10^{-3} of the (γ, n)

reactions giving rise to the most prevalent nuclides [Oka 1969, IAEA, 1979]. Experience at JLab indicates that these materials do not readily become activated unless they are used as targets, target holders, or other hardware so close to the beam as to be directly impacted by the primary photon beam created during beam interaction. An important practical factor in this situation is certainly the threshold energy for most of the reactions. Photoproduction of Be-7 in the materials of interest requires photon energies in excess of 30 MeV (about two to three times the threshold energy for typical (γ, n) reactions. Photon energy in the bremsstrahlung beam is drastically reduced after only a single scattering event, therefore limiting the potential for these reactions at much distance from the beam path.

The materials of concern are generally employed as thin films, sheaths or covers on metal objects. Again, due to the combination of important activation parameters it can be concluded that the activity concentration for both Be-7 and H-3 (and in particular for H-3, having a much longer half-life) will almost always be a small fraction of the activity produced in the associated metal components.

In summary, we can conclude that the long-lived radioactivity induced in beam-line components is dominated by gamma-emitting nuclides formed in metal items. Where these components include non-metals, the levels of activity in these materials will be lower than in the metal, such that they are unlikely to contain limiting quantities of activity. In addition, hard-to-detect nuclides are not expected to exist in the absence of readily detected nuclides, and where they are possible, their relative contribution to potential doses is significantly below that of the easily detectable radionuclides. Therefore, surface gamma measurement is an appropriate release method for these materials, assuming an appropriate level of sensitivity. Practical experience at JLab supports this conclusion.

Components Located Far From Beam-Line

As mentioned, the radioactivity formed by photon interactions in the high-energy shower dominates nuclide production at electron accelerators. This shower is significantly forward-peaked, and so the bulk of the activity is formed in close proximity to the beam-line. Of secondary but not negligible concern is the formation of radionuclides by high energy nuclear particles ejected as a consequence of the photonuclear interactions. Of these particles, neutrons are the only significant source of activation outside the region of the shower. It has been estimated that about 0.1% of the beam energy goes to produce these neutrons [DeStaebler, 1963].

Neutrons emitted from beam interactions are semi-isotropic, and therefore present a somewhat uniform activating flux. These neutrons may induce activity in materials at significant distances from the beam-line (to first order, one can conservatively estimate that all the neutrons created inside the enclosure are eventually absorbed in the concrete structure of the enclosure).

Interactions of the neutrons include high-energy nuclear events and low energy capture. The activation products produced by high-energy neutrons in metals are reasonably well represented by the nuclides listed in Table 1 (though the saturation amounts will be somewhat different). Activity produced by slow neutron capture depends strongly on the composition of the material exposed, as the absorption cross sections are not smooth functions of A or Z. But, in common metallic components, when considering long-lived radioactivity, we can add potential activation products Fe-59, Zn-65 and Sn-123 to Table 1. All of these nuclides are beta/positron emitters with associated energetic gamma emission. In stainless steel, Cr-51 will dominate the induced activity produced by slow neutron capture. Relative fractions of activity after one year of buildup are shown in Table 2.

Nuclide	Relative Fraction of Activity after One Year Buildup
Cr-51	0.85
Fe-55	0.13
Fe-59	0.017
Ni-63	0.002

Table 2 - Relative activity from neutron capture in stainless steel after one year buildup

Due to small absorption cross-sections, long-lived activation by slow neutron absorption is insignificant in the low-Z, organics/composites previously discussed. This, in conjunction with the low production from high energy interactions allows us to conclude that induced activity in such materials at large distances from the beam-line is negligible, to the point that these materials, when located sufficiently out of the immediate vicinity of the beam, could be released based on process knowledge alone. However, administrative procedures in place at JLab require surveys on all materials released from locations where activation is likely.

Concrete

In the case of concrete, a well-known set of expected radionuclides is shown in Table 3. These activation products are formed from interactions of high energy photons and neutrons, and from slow neutron capture.

Nuclide	Half-life	Decay Mode	Gamma energy per decay (MeV)
Na-22	2.62 y	β^+	2.2 (incl. annih. Photons)
Eu-152	13.54 y	ε, β^-	0.96
Co-60	5.27 y	β^-	2.5
Mn-54	303 d	E	0.83

Table 3 - Common nuclides produced in normal concrete

In some concretes, the nuclides Eu-154, Cs-134, Ba-133, Ti-44 and others may be formed, depending on the specification of the concrete. The concrete in the CEBAF enclosure is normal density material, however, moveable shielding blocks with different specifications may be used in beam enclosures. In any event, Eu-152 and Co-60 concentrations will generally dominate.

Tritium is also produced in concrete, and its concentration is seen to scale with that of the gamma emitters [EU, 1999; Masumoto, 2003]. Experience at JLab is consistent with various studies indicating that tritium may be expected in concentrations on order of a factor of ten higher than the detectable gamma emitters. However, the potential dose received in a credible exposure scenario involving tritiated concrete, is at least of order 10^4 smaller than from exposure to equal amounts of the gamma emitters. Again, as in the case of Ni-63 or Fe-55, clearance based on detection of the gamma emitters is conservatively protective for H-3.

Given the nuclides of concern, gamma surveys are appropriate for clearance of concrete. The main concern in release of concrete is the potential, in thick pieces, for self-shielding in the material, in which case activation deep in the concrete (or the reinforcement bar) might be missed. However, it is well established that induced activity in concrete as a function of depth is proportional to the high energy particle fluence through the material, and highest concentrations are always seen in the first few centimeters [Kamboj, 2000; EU, 1999; Masumoto, 2003]. Experience at JLab using high-resolution gamma spectroscopy on core-drillings indicates that

when surface dose rate scans detect no activity, these results can be confidently relied on as confirmation of absence of radioactivity in depth, including in re-bar metal [JLab, 2008b]. In addition, the relationship of surface dose rate to activity concentration indicates that these surveys are appropriate for detecting the nuclides of concern at a conservatively sensitive level.

Table 4 shows the activity concentration for a large concrete block with detectable activity. The block has been in storage for many years post-irradiation, and the gamma emission is due primarily to the longer lived nuclides Eu-152, Ti-44 and Co-60. The block is readily identifiable as radioactive via gamma survey, having a surface dose rate of about 10-15 µrem/hr above bkg. The activity concentrations shown are averages of the concentration in the first 20 cm depth of the block. These concentrations are compared to the screening values for release of material in ANSI/HPS N13.12 [ANSI, 1999]. It is notable that more recently irradiated concrete would also contain the readily detected nuclide Na-22, increasing the sensitivity of the survey method.

Nuclide	Average activity*	Screening limit*	Fraction of limit
Co-60	6.46E-6	3E-5	0.215
Ti-44	1.58E-6	3E-5**	0.053
Eu-152	1.01E-6	3E-5	0.034
H-3	8.12E-5	3E-3	0.027
*activities in µCi/g		Sum of fractions	0.33

Table 4 - Data from concrete block reading ~15 µrem/hr above background

** N13.12 does not list screening values for this nuclide, the value was derived by comparison of effective dose factors from [NCRP, 1996]

The concrete block in this example would not be releasable under the criteria defined in this document. However, it can be seen that the activity concentration is well below the screening level in ANSI N13.12 (derived from the 1 mrem/y dose criterion). This illustrates that gamma surveys are adequately sensitive for release of concrete, and that the detection threshold is a very small fraction of the DCGL.

The tritium is not detected by gamma survey. But as mentioned, it is well established that H-3 production generally scales with that of the gamma emitters, and its activity is expected to be negligible when there is no detectable gamma activity. Only in the extreme case of activated concrete which has aged for many years after exposure is there significant likelihood of the presence of tritium above background in the absence of detectable gamma emitters. Knowledge of exposure history should be used to support release decisions for concrete, and where this process knowledge is not robust, additional sampling and analysis may be required.

In summary, materials likely to be routinely released from beam enclosures can be confidently assessed by means of gamma surface dose rate measurements, given that the measurements are sensitive enough to detect the nuclides of concern at sufficiently low levels. We discuss measurement sensitivity in more detail in the following sections.

VI. Statistical Considerations and Measurement Sensitivity

Various studies have been conducted concerning release surveys, and much guidance has been produced, though it focuses heavily on surface contamination monitoring [NUREG, 1998a; NUREG, 1998b, DOE, 2002; MARSSIM, 2000]. These documents review the static counting technique, where actual counts are recorded for a fixed period, and normal (Poisson) counting statistics apply. Currie's technique [Currie, 1968] is the standard statistical method applied to such counting. The scanning technique has also been thoroughly described, usually under the

assumption that the scanning is done in a fashion involving continuous movement of the probe until the signal changes enough to instigate a pause, then a static observation is made. This type of survey is seen as occurring in two stages. An alternative to classic counting statistics that is often used to assess sensitivity in this case is called signal detection theory (SDT), and has elements analogous to normal counting statistics.

Neither of these approaches (Poisson statistics or SDT) applies perfectly to assessing the sensitivity of volumetric surveys of the type performed at JLab. There are some important inherent differences in performing a surface contamination survey and a survey for volumetric activity. The release survey for volume activity is a semi-static measurement (assuming a reasonably small component being monitored). Normally, the detector is placed in contact with the object and held there for a short period, then possibly moved to another location and again held in position. For small (hand-held) components, a single location of measurement is generally sufficient to assess the object (since the activity at depth in the object is of interest, not activity on a particular surface). In addition, on larger objects, the technician will often use process knowledge and experience to conduct static measurements at locations on the component that have the highest probability for activation (in addition to scanning large areas of the object). There is similarity to scanning for contamination when making volume activity measurements, in that the technique involves a period of observation, which is analogous to the "scan" phase of a surface contamination survey. If no activity is detected in this phase of the survey, the technician makes a determination that no further measurement is necessary in that location. The obvious difference in the two techniques is that there is generally much less movement of the detector in the survey for volume activity, and hence, a longer observation period.

Taking these differences into account, and making appropriate adjustments, the general approach used in SDT is reasonable, and we will examine our method under this protocol, noting areas where there are differing assumptions. Human factors enter into any survey of this type, and these are well-considered in the SDT approach.

Below we define the minimum detectable signal level for the release survey method used at JLab. This is the signal level that is expected to be recognized by the surveyor as being greater than the local background. This threshold is often referred to as the *minimum detectable count rate* (MDCR), which, in our case is expressed as a dose rate.

Characteristics of the Instrument

The MicroRem instrument employs an analog dose-equivalent rate meter, in combination with an audio counting circuit. The audible count rate indication is not a "raw" signal output from the detector, as is common with typical contamination monitoring instruments, but is derived from the dose rate signal. The circuitry in the MicroRem produces an audible count rate of about 20 cpm per µrem/hr. In typical background of 10 µrem/hr, the audible count rate is ~ 200 "clicks" per minute. We will consider the normal range of background to be from about 5-15 µrem/hr.

One assumption of SDT is the reliance on the audible signal for detection decisions by the surveyor. In the case of surface contamination surveys, instrument read-out is typically in counts per minute (cpm), and correlates directly to the audio count rate. Under typical monitoring conditions, background meter readings are generally on order of 10 - 20% of the scale range. Detection decisions are assumed to be based almost entirely on the surveyor's interpretation of the audio signal. This is true to an extent in the Jlab survey, however, there are some important differences. Certain features of the meter read-out enhance detection sensitivity. The instrument scale used to conduct release surveys has a range of 0 – 20 µrem/hr

(this constraint is discussed later). On this scale, ambient background produces a meter reading of approximately 50% of scale. In this case, a doubling of the reading results in "pegging" the meter, and forcing a decision of "not releasable" (as this is obviously "distinguishable" from background). The important range of detection decision sensitivity therefore lies nominally between 10 and 20 µrem/hr. The visual input received by the surveyor from the Microrem is therefore inherently more sensitive than from a typical contamination monitoring instrument, and it can be reliably assumed that a doubling of signal will *always* produce a detection decision (the instrument is also equipped with an over-range alarm that alerts the operator of a "pegged" condition). This also brackets our sensitivity assessment, such that we are attempting to determine if a surveyor might be able to detect even smaller signals than a doubling of background. The overall effect of having two sensitive inputs (audible and visual display) for surveyor decision is enhanced sensitivity; however, the enhancement is difficult to quantify. Our discussion of SDT will generally not attempt to account for this, except to note, where appropriate, the *a priori* assumption that activity corresponding to a doubling of background will always be detected.

Statistical Approaches

This document will not attempt a rigorous review of statistical methods, but we will briefly describe the approach used to conduct hypothesis testing for detection sensitivity. It should be noted that the terms "detection limit" and "detection sensitivity" as used here, are independent of field conditions, and represent the value of the instrument reading (eg. in µrem/hr) corresponding to a detectable signal (MDCR). These terms are not interchangeable with the term Minimum Detectable Concentration (MDC), which does depend on field conditions, such as the geometry and composition of the material under survey. We here only evaluate the detection sensitivity, and will later discuss approximate MDCs.

What we have referred to as "normal counting statistics" is the common practice of statistical analysis of integrated counts taken during fixed time periods under essentially static, well-known background conditions. Assumptions for these techniques also include a consistent geometry and sample media characteristics. We will briefly discuss several important statistical concepts arising from this approach.

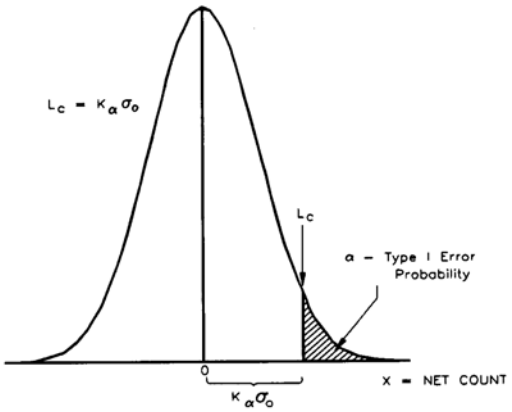

Figure 3 – Representation of a zero net activity background distribution

The curve shown in Figure 3 represents a background distribution minus the mean of the distribution. This results in a Poisson distribution with a mean of zero. Conveniently, values along the X axis then become net counts. In the distribution, L_C is the *critical level*, the number of net counts at which the detector output is considered "above background". A Type I error

("false positive") occurs when such a decision is made (the measurement is deemed to be above background) in the absence of any sources. This level is chosen, such that there is a predetermined probability of such an error, at a known confidence. The probability of such an error is designated α and is traditionally selected as 0.05.

For completeness, we show in Figure 4 the presence of a source signal whose mean net count rate is L_D. L_D is the *detection limit* – or level of activity that can be reliably quantified with a known confidence. This level is usually set based on the probability (β) of making a Type II error also being 0.05. A Type II error ("false negative") occurs when the response is considered to be background when there is actually radiation present above background. When converted to appropriate units, L_D becomes the *minimum detectable activity* (or concentration).

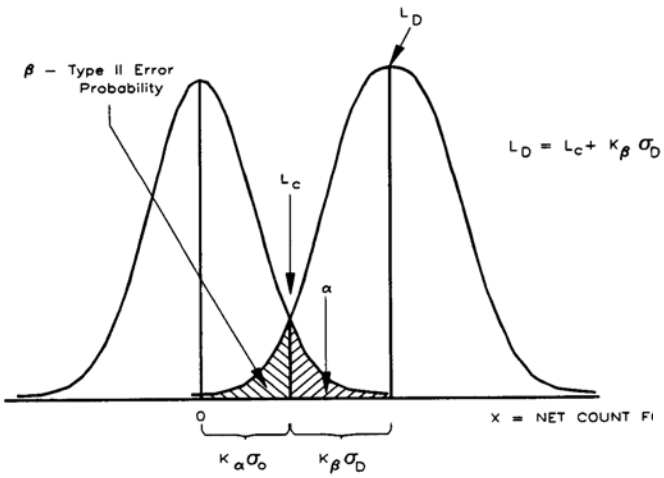

Figure 4 – Background Plus Source Signal

When conducting surveys for release, a decision rule is selected based on a hypothesis test. The MARSSIM [MARSSIM, 2000] describes two scenarios for determining the appropriate null hypothesis. In Scenario A, an activity limit is specified. In this case it is important to show that L_D for the chosen monitoring method is at or below this limit. In this scenario, the Type I and Type II errors are reversed compared to the above descriptions, and the null hypothesis is defined as the material having activity exceeding the release criterion. This requires statistically significant evidence that the residual activity is below the release limit to reject the null hypothesis (and release the material). In this case, it can be seen that the release criterion must be a quantifiable concentration, giving a known and acceptable confidence interval and value for α.

However, in our case, MARSSIM Scenario B is more applicable. In this case, the release criterion is that there must be no added activity above background in the material. The null hypothesis is that the material under survey is indistinguishable from background. There must be statistically significant evidence that activity is present to reject the null hypothesis. In this case, significance is defined by α, and the decision to reject the null hypothesis is made at L_C. In effect, it becomes important only to *detect*, rather than *quantify* activity. The determination of the suitability of the detection threshold (the relationship of L_C to consensus DCGLs, or any quantitative limit) is not directly addressed, but is evaluated separately.

In a static counting scenario, L_C is easy to determine. To obtain a reference value for comparison, a MicroRem meter was adapted to a counter/scaler to allow fixed time interval counting. Background counts were measured and L_C was determined for the count times indicated, based on a Type I error probability of 0.05, by the formula,

$$L_C = k_\alpha \, \sigma_0$$

where,

k_α is the value of the standard normal deviate corresponding to a one-tailed probability level of $1 - \alpha$. With $\alpha = 0.05$, $k_\alpha = 1.645$.

σ_0 is the standard deviation of our zero-mean count distribution, and is obtained by propagating the individual errors associated with population standard deviation (σ_B) and that of the zero-activity samples (s_B). That is,

$$\sigma_0 = \sqrt{\sigma_B^2 + s_B^2}$$

We make the standard assumption that σ_B and s_B are equal, which gives the following;

$$L_C = 1.645 \sqrt{2 \, s_B^2}$$

This further simplifies to: $L_C = 2.33 \, s_B$, which is valid for paired blanks (applicable to field counting). In practice, s_B can be approximated by the square root of the background count (B) if the count rate is reasonably high, resulting in;

$$L_C = 2.33\sqrt{B}$$

In this case, twenty 1-minute background counts were taken, with a mean count of 93.3. L_C is then 22.5 net counts. The background dose rate indication fluctuated between about 4-7 µrem/hr during the counting. This gives a relationship of ~ 17 counts per minute per µrem/hr. Furthermore, the dose rate corresponding to L_C is a net increase of about 1.3 µrem/hr. This demonstrates the inherent difficulty of applying static counting statistics to ratemeter type instruments and scan surveys. Fluctuations in the ratemeter display make discerning such a change very difficult, though the audible output enhances sensitivity somewhat.

In addition, this critical level is defined only for a one minute count. In practice, the detector is usually not held stationary for long periods. L_C can be calculated for different measurement periods, and results for several counting times are shown below. Again, we use these examples only for comparison, as they are only valid under static counting conditions.

Sample Count Time	L_C (net count rate)*	Net Dose Rate (µrem/hr)
5 sec	57	3.4
10 sec	42	2.5
20 sec	32	1.9

Table 5 – Relationship of sample count time to sensitivity for 1 minute mean background of 93.3 cpm

* L_C calculated following [Strom and Stansbury, 1992]

For shorter count times, the decision level approaches the historically employed “twice background” release criterion. For comparison, the L_C for a 5 second count in a range of backgrounds is given in Table 6.

Background (µrem/hr)	Background (cpm)	L_C (net count rate)	Sensitivity* (net µrem/hr)
5	85	54.9	3.2
7	119	64.8	3.8
10	170	77.5	4.6
12	204	84.8	5.0
15	255	94.8	5.6

Table 6 – Calculated L_C for a Range of Background Rates
* 5 second sample count times

It is interesting to note that during evaluations of technicians' release decisions under controlled conditions, most detection decisions were made within about 5-10 seconds. Also, the lower the activity level, the longer the technicians took to make a detection decision. Quantification assessments generally took on order 20 seconds. These tests are described more fully in a subsequent section of this note.

Since static counting is not employed for field release surveys, the above discussion provides only a reference point for release survey sensitivity. We now turn our attention to Signal Detection Theory (SDT). Many of the parameters of SDT are analogous to the static counting statistics discussed. However, this method is applied to ratemeter applications, and specifically to the case where the surveyor is responding to the audible response or "click rate" of the meter. Details of the instrument response characteristics have already been discussed.

SDT is discussed in detail in the references [NUREG, 1998a; NUREG, 2002; MARSSIM, 2000]. This approach is well suited to the task of locating radioactivity on/in materials by field survey. The information available to the surveyor can arise from either noise (background) alone or from signal-plus-noise and can be represented by two (typically overlapping) probability density distributions, as shown in Figure 5. The similarity to the static counting statistics is evident in the figure. The task of the observer is to indicate whether an increase in survey instrument output arises from "noise alone" or a "noise plus signal" event.

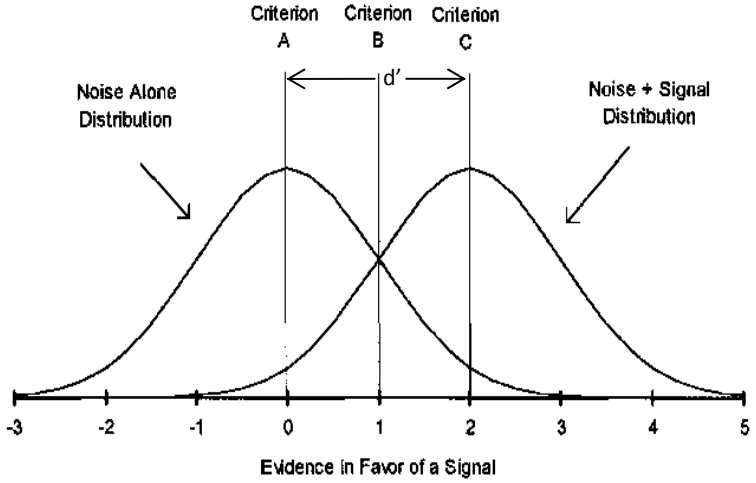

Figure 5 – SDT Measures of Sensitivity (d') shown relative to assumed underlying distributions

To make this decision, a criterion must be established at some point along the continuum – i.e., once the criterion point is set, any measurement greater than (to the right of) the criterion will be interpreted as the presence of radioactivity. Using the assumption that the underlying distributions are normal and of equal variance, an index of sensitivity (*d'*) can be calculated which represents the distance between the means of the distributions in units of their common standard deviation. The index is calculated by transforming the true-positive and false-positive rates to standard deviation units, (i.e., z-scores) and taking the difference.

$$d' = z\,(\textit{false positive}) - z\,(\textit{true positive})$$

Tabulated values of d' are available in the references. It is conventional in SDT analysis to describe performance in terms of the false positive rate (α) and the true positive rate (1-β). True negatives (correct releases) and false negatives (missed activity) are complements of these quantities.

As mentioned, when release decisions are made based on human interpretations of signal levels, one must consider the "human performance" factors affecting the release decision. The sensitivity measure d' is independent of the observer, however, there are at least two human factors bearing on the release decision. One is the willingness to report a positive result – i.e. the criterion for responding "yes". This is affected by the perception of the "cost" of the decision by the observer. For instance, if a Type I error (false positive) is judged to have a significant cost, the surveyor will place the criterion more conservatively (e.g., criterion C in Figure 5). Factors that might sway the observer in this direction might be the impact in expense and effort for management/disposal of the material as radioactive. Alternatively, if the observer perceives that positive results will occur frequently, and are a routine expectation, and that the cost of a false negative is high, they are likely to set very liberal criteria for positives (Figure 5, criterion A). In this case, the decision comes at the cost of a high number of false positives.

Additionally, in practice, observers do not operate with ideal efficiency, and an observer efficiency value is recommended. Guidance suggests a value no greater than 0.75 for this parameter. It should be noted that the ideal observer is not considered to be error free, but operates at the optimum level – that is, at the minimum calculated error rate.

As mentioned previously, some assumptions of SDT are not perfectly applicable the survey performed for volumetric activity. Perhaps the most important one is the assumption of a two-step survey, in which scanning is conducted until a signal is detected that causes the surveyor to pause in an area. The "search – pause" approach to the survey is applicable, but with the important difference that during our search phase, the detector is usually not being moved (or if so, is being moved very slowly, such that observation intervals are fairly long). So the assumptions about observation intervals in SDT need to be modified. In a typical survey for volumetric activity, the detector is held semi-stationary for a period of several seconds (more analogous to the second stage of the scan survey in the SDT approach), while a decision is formulated about the presence of activity, and may be held in place for an extended period once a possible detection event is perceived (this phase of the measurement is used not only to confirm detection but to make a quantitative assessment of dose rate). This second phase may take 10 – 30 seconds in practice.

We will consider only a single scan phase, with an assumed observation time of 5 seconds. Observations shorter than this are not adequate to conduct a valid survey, and survey protocols reinforce the need to use observation periods that allow thorough investigation of the material. As mentioned, longer observation times are usually used when no immediate indication of activity is noted, thus increasing sensitivity.

Determining MDCR and Employing Surveyor Efficiency

Detectability for a Poisson distribution can be expressed as

$$D \approx \frac{s_i}{\sqrt{b_i}}$$

where b_i is the average number of background counts in an interval. For background count rates, b, in cpm and observation interval length, i, in seconds, $b_i = b(i/60)$. The detectability index (D) is asymptotically equal to d'. The minimum detectable number of net source counts in the interval is given by s_i. For an ideal observer, the number of source counts required for a specified level of performance (defined by the selection of d') can be found by

$$s_i = d' \sqrt{b_i}$$

As mentioned, *d'* values are tabulated, and selected values are depicted in Table 7. Since we are assuming a single stage of scanning, d' should be chosen such that the false positive rate is acceptable, while maintaining a high true positive rate (in two-stage scanning, the true positive rate must be equally high at both stages). The conventionally accepted true positive proportion is 0.95. Following [NUREG, 1998a] for the final scan stage false positive proportion, we select a false positive rate of 0.20 (false positives are reduced in practice through longer observation).

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
0.50	0.38	0.68	1.04	1.64
0.55	0.26	0.54	0.91	1.51
0.60	0.13	0.42	0.82	1.38

Table 7 – Values of *d'* for Selected True Positive and False Positive Proportions

Assuming a background dose rate of 10 µrem/hr, the background count rate from the instrument audible is ~170 cpm. Our observation interval is 5 seconds, and our selected value of *d'* is 2.48. This gives a value for s_i of (2.48) $(14.167)^{1/2}$ = 9.3. To convert this to a count rate, we multiply by (60/i) for a net MDCR of 112 cpm, or ~ 6.6 net µrem/hr.

Table 8 gives sensitivities (MDCR) for a range of background levels under which release surveys may be conducted (procedurally limited to background between 5 and 15 µrem/hr). These values are all based on five second count times, and *d'* of 2.48.

Background (µrem/hr)	Background (cpm)	Net MDCR (cpm)	Sensitivity* (net µrem/hr)
5	85	79	4.6
7	119	94	5.5
10	170	112	6.6
12	204	123	7.2
15	255	137	8.1

Table 8 – Survey Sensitivity (MDCR) of the Ideal Observer for Selected Backgrounds
* 5-second observation interval

Finally, adjusting for the less-than-ideal observer, we correct these sensitivity values by the following formula

$$s_i = \frac{d' \cdot \sqrt{b_i}}{\sqrt{p}}$$

or equivalently, MDCR/ $\sqrt{p}$, where *p* is the efficiency of the observer. As mentioned, an upper bound of 0.75 is suggested for the value of *p*, resulting in the values shown in Table 9 for detection sensitivity.

Background (µrem/hr)	Sensitivity* (net µrem/hr)
5	5.3
7	6.4
10	7.6
12	8.3
15	9.4

Table 9 – Survey Sensitivity for a Non-Ideal Observer
* 5-second observation interval

The implied precision of fractions of µrem/hr in these estimates is not meaningful in practice. These values are not largely different from the historically employed criterion at JLab of twice background. When conducting a release survey under the criterion of indistinguishability from background, it is likely that in many cases, technicians will identify signal levels lower than those listed above (as we have observed in field testing). But these values represent a reasonable statistically based upper bound on the release criterion, useful for evaluating screening levels.

In practice, we will simplify this range of values based on survey constraints. One such constraint requires the measurement for release surveys to be made using the lowest range of the instrument (0-20 µrem/hr). Under this condition, when surveying in the maximally permitted background (15 µrem/hr), the effective release limit is 5 µrem/hr (i.e. "pegging" the meter). This bounds the release criterion to a range of about 5-8 µrem/hr. Given the fluctuation of the instrument reading at these levels, we conservatively assign a value of 5 µrem/hr above background as the release criterion. This is believed to be an achievable detection threshold

under actual survey conditions, considering that observation periods normally exceed 5 seconds, especially when activity is not immediately indicated. Field protocol for release will stress "any detectable activity" as the limit, allowing the technician to decide on the presence of activity even if levels are below the assigned criterion. The limited validation study discussed in the following section demonstrates efficacy of this approach.

VII. Field Testing

We conducted trials to evaluate technician performance, using actual accelerator hardware exhibiting some level of activation. The items were chosen with a range of activity levels, from non-detectable to just above background in order to evaluate detection sensitivity at the threshold of detectability. All the items tested were relatively small, hand-held objects. Surveys of this type of item represent the most conservative case, from the standpoint of detecting a target screening level, since the total activity in the item is small when activity is present at the screening concentration. Larger objects, and collections of smaller objects cause higher dose rates for the same activity concentrations, improving detection sensitivity. Therefore, for the activity concentration present, these objects provide a good example of the limiting case for detection sensitivity.

The objects of interest were counted on the modified MicroRem instrument, to establish a known reference count/dose rate. Counts were taken for one minute, and results are shown in Table 10. All items with detectable activity indicated count rates above L_D (MDA) for a one minute count. However, three of the five items fall below the sensitivity levels for SDT and static counting statistics for 5-second observations – that is, they should not be detectable in a five second field scan. One item falls below the L_C for a one minute count.

Item	Mass (g)	Net Counts	Net Dose Rate*
7 cm flange	238	87.7	6.6 ± 1.8
Socket-head bolt	81.4	55.7	4.3 ± 1.4**
Hex-head bolt	84.5	5.7	< L_C
Aluminum plate	693	142.7	8.4 ± 2
Ion pump power supply	1976	65.7	3.9 ± 1.3**

Table 10 – Analysis of Accelerator Hardware Used for Performance Evaluation

* Dose Rates in µrem/hr, 2σ estimate of error, 1 minute count

** Item activity is below calculated MDCR for a 5 second count

Four technicians were asked to evaluate each item and determine (1) if the item was measurably radioactive, and (2), if so, what the contact net dose rate of the item was. The technicians were instructed to indicate when they had made a decision regarding criterion (1), so that observation times could be qualitatively evaluated. Table 11 depicts the results of the trials. All of the technicians classified all the items as radioactive except for the hex-head bolt. Two technicians initially classified the Hex-head bolt as radioactive, but after additional observation, decided it did not have detectable activity. As mentioned, longer times were taken to make decisions on the items with the lowest readings. Typical decision times were on order 5-10 seconds, for the items with net positive activity. The table shows the readings each technician reported for each item. All the surveys were conducted in the same physical location, and each technician reported a background dose rate of about 7 µrem/hr.

Item	Net Dose Rates (μrem/hr)					
	Measured	Tech #1	Tech #2	Tech #3	Tech #4	Mean*
7 cm flange	6.6	3	4	8	8	5.75
Socket-head bolt	4.3	4	3	8	10	6.25
Hex-head bolt	< L_C	ND**	ND	ND	ND	ND
Aluminum plate	8.4	13	13	13	11	12.5
Ion pump power supply	3.9	9	3	4	4	5

Table 11 – Results of Technician Trials

* Mean of the technician readings

** ND = Not Detectable

Note that technicians often over-reported the readings compared to the "true" dose rates. This reflects the effect of meter fluctuation, and the practice of reporting instrument readings conservatively, and includes any differences in individual instrument response due to calibration tolerances. The results also confirm the ability of technicians to consistently identify very low levels of activity, at or below that which is statistically expected based on signal detection theory. Additionally, in this test, the technicians exhibited the ability to reliably detect activity at or below the L_C for static counting. We conclude that an assigned upper limit on detection sensitivity of 5 μrem/hr is achievable and practical. We call this an "upper limit" because, as noted, our field protocol will stress that any indication of activity during the survey will disqualify the item for release. Trained and experienced RCTs can often identify the presence of very small amounts of activity, below the level for which statistical methods can be rigorously applied.

VIII. Estimation of the Activity Concentration (MDC) at the MDCR, and Comparison to the Clearance Standards

As discussed in section VI, the null hypothesis for our release decisions is the absence of radioactivity above background (Scenario B in the MARSSIM approach). When this is the case, a quantitative assessment of the activity in the sample is not made, but rather, a decision that no activity detectable above background exists. For this reason among others, the survey result for a cleared item does not convey the conventional message that "the activity is less than X", where X is the compliance criterion. Instead, in this case, the compliance criterion is the absence of activity above background, as discussed in section III.

However, we must still estimate the approximate activity concentration that would cause us to reject the null hypothesis (a "working" MDC). This activity estimate is needed in order to evaluate the acceptability of the measurement sensitivity in terms of accepted consensus standards for clearance screening levels. As mentioned, in the case of volumetric activity, the concentration of activity giving rise to an instrument signal exceeding the detection threshold may vary due to several physical parameters associated with the material under survey, and with various mixtures of radionuclides.

We showed in Table 4 (section V) that in practice, activated concrete can be detected when activity levels are significantly below recommended clearance levels. If we simply scale the measured dose rate to the MDCR for a non-ideal observer, we would estimate the detection threshold for this concrete to be about 20% of the effective DCGL (for the nuclide mixture in this example) found in ANSI/HPS N13.12. Scaling to our assigned upper limit of detection (we will refer to this value as *MDCR** or *Assigned Release Limit – ARL*), our estimated detection sensitivity is about 10% of the effective DCGL. In practice, as we have seen, it is likely that we could distinguish activity at even lower levels, but we wish to maintain conservatism in our estimate.

Similarly, we can compare Stapleton's results [Stapleton, 1990] for activation in accelerator components to the consensus standards (which were all developed subsequent to his work). Stapleton calculated that objects having 10 pCi/g uniform activity, consisting of a mixture of typical activation products would exhibit a surface dose rate of about 13 µrad/hr (we will take rad ~ rem). Using the same approach for an object containing only Co-60, the dose rate was calculated as 26 µrad/hr. Again, scaling these values we would arrive at a threshold activity for the nuclide mixture of about 4 – 7 pCi/g, and for the Co-60, about 2 – 4 pCi/g. In the case of Co-60, the ANSI DCGL is 30 pCi/g. Again, we see that our estimated survey sensitivity is on order 10 – 20% of the clearance DCGL.

Stapleton's nuclide mixture was based on weighted photon energies and not specific percentages of various nuclides. However, clearance DCGLs for the predominant activation products described in Tables 1 – 3 are all at or above the DCGL for Co-60 (not all nuclides listed in the tables are contained in the ANSI standard, but applicable values can be derived from the references). Therefore, typically observed mixtures of these nuclides would have an effective DCGL above the Co-60 guideline.

It should be noted that a range of clearance levels have been recommended by various agencies and authors. The references provide comparisons of this range. Differences in recommended values often occur due to different grouping of nuclides into categories, for which there is often a factor of ten difference in recommended DCGL. On the whole, these differences do not affect the outcome of our estimations of detectable activity in an important way, considering the nuclides of concern, overall quantities, and level of precision with which these estimates are made.

We should keep in mind that these are idealized characterizations. To evaluate the matter further, we conducted measurements aimed at verifying the reasonableness of these calculations. The components used for our field evaluations of technician performance were subjected to in-situ gamma spectrometry using the Canberra ISOCS® analysis system.

In-Situ Measurements

Analysis was performed on each item used for field evaluation of the technicians. Given the small size of the items, and the limitations of the geometry modeling in the ISOCS program, we estimate the overall measurement uncertainty to be on order of 50%. The measurements demonstrated the effect of object size on detection sensitivity. As discussed earlier, the limiting case is for a small object, since total activity in such an object is very small, even when activity concentration may be at or above the estimated MDC. This effect is understood by technicians performing release surveys, and survey protocols take this into account (small items such as nuts and bolts are grouped together for survey whenever possible).

The results of the ISOCS analysis are shown in Tables 12(a) – 12(e). For each item, the detected nuclides, their activity concentrations and the relevant ANSI clearance DCGLs and DCGL fractions are shown. The sum of the DCGL fraction is the total effective activity with respect to the effective clearance level. A value greater than one indicates the item exceeds the recommended clearance level.

The "Activity Fraction at ARL" indicates the effective activity fraction when values are adjusted to the assigned release limit of 5 urem/hr – a qualitative indicator of whether the survey sensitivity is above or below the effective clearance DCGL.

Ion Pump Power Supply (1976) Table 12(a)

Nuclides Detected	Activity Concentration (pCi/g)	ANSI DCGL (pCi/g)	DCGL Fraction
Na-22	1.3	30	0.04
Mn-54	7.7	30	0.26
Co-57	0.6	300*	0.002
Co-60	8.4	30	0.28
Zn-65	1.4	30	0.05
		Sum of Fractions	0.63
		Activity Fraction at ARL	0.8

* DCGL derived from [ANSI, 1999 and NCRP, 1996]

Aluminum Plate (693g) Table 12(b)

Nuclides Detected	Activity Concentration (pCi/g)	ANSI DCGL (pCi/g)	DCGL Fraction
Na-22	32	30	1.07
		Sum of Fractions	1.07
		Activity Fraction at ARL	0.6

Beamline Flange (238 g) Table 12(c)

Nuclides Detected	Activity Concentration (pCi/g)	ANSI DCGL (pCi/g)	DCGL Fraction
Mn-54	41	30	1.37
Co-57	19	300*	0.06
Co-60	6	30	0.2
		Sum of Fractions	1.63
		Activity Fraction at ARL	1.2

* DCGL derived from [ANSI, 1999 and NCRP, 1996]

Socket-head Bolt (81.4 g) Table 12(d)

Nuclides Detected	Activity Concentration (pCi/g)	ANSI DCGL (pCi/g)	DCGL Fraction
Mn-54	18	30	0.6
Co-57	19	300*	0.06
Co-60	34	30	1.13
		Sum of Fractions	1.79
		Activity Fraction at ARL	2.1

* DCGL derived from [ANSI, 1999 and NCRP, 1996]

Hex-head Bolt (84.5g) Table 12(e)

Nuclides Detected	Activity Concentration (pCi/g)	ANSI DCGL (pCi/g)	DCGL Fraction
Mn-54	6.3	30	0.2
Co-57	3.5	300*	0.01
Co-58	0.34	30	0.01
Co-60	3.3	30	0.1
		Sum of Fractions	0.3
		Activity Fraction at ARL	N/A**

* DCGL derived from [ANSI, 1999 and NCRP, 1996]

** Not calculable due to absence of measurable dose rate

These data provide insight into several areas of the clearance surveys. First, as described in Section III, there exists a range of MDCs when surveying for volumetric activity. The limiting extreme of this range occurs with very small objects, which is apparent in the data. The demonstration of reasonably good detectability even in the limiting case provides good assurance that in most cases, detection sensitivity is very conservative (this was further demonstrated by the concrete block data). Care has to be taken in the specific procedures used so that there is adequate consideration given to small objects. As mentioned, JLab procedures address this condition, requiring the pooling of small objects for survey. Secondly, it should be noted that in two cases (socket-head bolt and ion pump power supply), the observed detectability is better than the calculated effective detectable activity at the ARL. This is evident in the fact that activity was actually detected below the ARLs, and indicates conservatism in the assigned sensitivity values.

In the case of the hex head bolt, no activity was detectable with the field survey equipment, but very small levels of activity were detected with gamma spectroscopy, although the activity was well below the effective DCGL. As shown in the calculations, the activity is not statistically detectable with the MicroRem, even with a one-minute, static count. As noted, during the field trial, two technicians initially identified this object as having activity, but then changed their decision after further monitoring. This gives a qualitative indication that even in the extreme case of a small object with activity well below the consensus DCGLs, there is some potential for detection.

For completeness, we should note that the total amount of materials cleared from accelerator enclosures is relatively small. Statistics specific to the actual volume of materials removed and eventually released from such areas are not available, but a conservative estimate can be made. Removal of large apparatus is rare, and most hardware that can be reused is retained on site. Material most likely to impact the public is in the form of waste, disposed in a municipal waste landfill. Waste statistics for the accelerator site indicate an annual disposal weight of approximately 56 tons for all sources of waste onsite. This includes all refuse generated on the accelerator site. We might conservatively estimate that 5% of this material actually originates in beam enclosure areas, giving us a value of 2 – 3 tons of waste annually from these areas. IAEA considers this a moderate quantity of impacted waste, and sets the clearance DCGL for this quantity of material at the same value as its exemption level in the Basic Safety Standards [IAEA, 1998; IAEA, 1996b]. These values are consistent with the ANSI DCGLs.

Taken together, these data indicate that detection sensitivity for components typically evaluated for clearance is very good; consistent with consensus standards on clearance, and that clearance protocols at JLab conservatively protect the public and environment. Constraints in monitoring protocols ensure conservative application of surveys and process knowledge. Ultimately, the goal of minimizing the potential for exposure to members of the public is met, through ensuring that materials released from control have no radioactivity distinguishable from background. Potential doses to members of the public are insignificant, and well below the 1 mrem/yr consensus standard.

IX. Constraints on Monitoring

Several monitoring constraints have already been mentioned, but here we summarize the parameters under which the release surveys described are conducted. The method of survey described in this document applies only to solid materials of moderate mass and volume. As discussed, very small items (less than a few hundred grams) represent a limiting case, and should be grouped together whenever possible for survey. Very thick items may also require additional analysis and/or careful process knowledge assessment, including evaluation of inaccessible activation, for instance in the case of large magnet cores in which activation patterns may cause induced activity in locations that cannot be reached with a survey meter. Dispersible materials (i.e. pulverized rubble, vacuum cleaner waste, etc.) should be sampled and analyzed, rather than direct surveyed.

The release survey consists of a near-static measurement, or a series of semi-static measurements, rather than a scan of a large area. For large items (structural materials, long sections of pipe, etc), the survey should involve very slow scanning, and periodic pauses to survey for several seconds in a static fashion (this should be done periodically, whether or not there is an indication of activity at that point during the scan, and process knowledge should be applied to ensure that local points of interest that may have higher potential for activity are assessed carefully).

High density materials such as lead or tungsten should be evaluated by other means (in addition to a survey), including process knowledge, representative sampling, etc. Materials that have a likelihood of containing tritium as a primary activation product require sampling/smears. Process knowledge and procedural guidance should be followed to determine if contamination surveys are needed in addition to the activation survey, or in some cases possibly needed instead of activation survey (i.e. items from beam dump cooling water system). Concrete should be evaluated based on its content (i.e. high density vs. low density) and time since irradiation. Normal-density concrete which has aged many years post irradiation may contain tritium as a primary contaminant. Recently irradiated concrete, and high-density concrete are readily assessable by survey.

The survey should be done where background is low and stable – less than 15 µrem/hr – and must be done on the X 0.1 scale (the X 1 scale does not provide adequate resolution on the meter). In addition, all surveys must be conducted with the audible on, and the response set to Medium. All release surveys must be conducted with the LE version of the instrument.

In some cases, it may be practical to conduct a conditional release assessment. In some cases, certain components or materials may be difficult to unconditionally release due to the item's physical properties, or it may be more convenient not to attempt a full unconditional release if the item can be controlled for on-site use (e.g. vacuum pumps taken from beam enclosure to shop areas for maintenance). In such cases, adequate administrative controls on the items must be established to identify the item as restricted from release until further evaluation is conducted.

X. Procedures and Documentation

Documentation for release must meet the requirements of O 5400.5 for identification of personnel, instrumentation, and other pertinent information concerning release surveys. Guidance and format for survey documentation is found in the references.

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