American National Standard

ANSI/HPS N13.12-2013

Surface and Volume Radioactivity Standards for Clearance

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American National Standards Institute, Inc.

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Foreword

(This foreword is not part of American National Standard ANSI/HPS N13.12-2013.)

The need for comprehensive clearance[1] criteria for items, equipment, and facilities with surface or volume radioactive materials has been recognized for several decades. Initial attempts to develop this standard began in 1964 and were limited in scope to the consideration of radioactive materials on surfaces. Volume radioactivity, including radioactive materials dispersed throughout the material, materials activated by neutrons, and radioactive materials in soils were all beyond the scope of the initial efforts.

In 1999, the final version of this standard was formally issued. It provided both the primary dose[2] standard for clearance and derived screening levels for surface and volume sources of radioactivity, based on the primary dose standard. However, the 1999 version of this standard was not fully adopted by U.S. federal or state agencies because, although judgment and consensus were used in its development, the technical basis was not deemed robust enough (because the technical basis used information from the reported literature instead of conducting internally consistent scenario and pathway analysis) to warrant broad application.

During the 1980s and 1990s, exposure pathway assessment was used to support the development of new regulations or standards. Of particular note have been the efforts of the U.S. Nuclear Regulatory Commission (USNRC) to develop revised decommissioning regulations, guidance from the National Council on Radiation Protection and Measurements (NCRP) on the development and application of screening models, and the International Atomic Energy Agency (IAEA) efforts to develop clearance (formerly exemption) criteria. The intent of these efforts was to link secondary (derived) criteria with a primary dose (or risk) criterion.

Since 1999, federal and state agencies have developed proposals related to clearance. In an April 26, 2010, *Federal Register* notice the USNRC reported the status of their efforts. "The staff provided a draft proposed rule package on Controlling the Disposition of Solid Materials to the Commission on March 31, 2005, which the Commission disapproved. The Commission's decision was based on the fact that the Agency is currently faced with several high priority and complex tasks, that the current approach to review specific cases on an individual basis is fully protective of public health and safety, and that the immediate need for this rule has changed due to the shift in timing for reactor decommissioning. The Commission has deferred action on this rulemaking." The schedule for the notice of proposed rulemaking is to be determined.

Internationally, the IAEA published Safety Guide No. RS-G-1.7, "Application of the Concepts of Exclusion, Exemption and Clearance," along with Safety Report Series No. 44, "Derivation of Activity Concentration Values for Exclusion, Exemption and Clearance" (IAEA 2004, 2005). These two reports provide definitive

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[1]**Clearance** is the removal of items or materials that contain or *may* contain residual levels of radioactive materials within authorized practices from further radiological control. Clearance is distinct from authorized radioactive discharges. Clearance implies that the subject materials or objects were under regulatory control—exclusion and exemption do not. **Exclusion** is the designation by a regulatory authority that the magnitude or likelihood of an exposure is essentially unamenable to control through requirements of standards and such exposures are typically outside the scope of requirements or standards (e.g., exposure from ^{40}K in the body, from cosmic radiation at the surface of the earth, and from unmodified concentrations of radionuclides in most raw materials). **Exemption** is the designation by a regulatory authority that specified uses of radioactive materials or sources of radiation are not subject to regulatory control because the radiation risks to individuals and the collective radiological impact are sufficiently low. [Sources: Principles for the Exemption of Radiation Sources and Practices from Regulatory Control; Safety Series No. 89; 1988; and International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, 1996; International Atomic Energy Agency, Vienna.]

[2]Where radiation dose in this context is the total effective dose. Radiation dose, in general, is the energy deposited per unit mass. The SI unit for total effective dose is joule per kilogram, given the special name sievert (Sv). The conventional unit for total effective dose in the United States is the rem, which equals 0.01 joule per kilogram.

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international recommendations for clearance of items or materials that contain levels of radioactive materials within authorized practices from further radiological control.

In response to the IAEA recommendations, the U.S. federal agencies concerned with controlling radiation exposures requested that the writing group for this standard be reformed to determine whether it would be possible to harmonize, or bring into accord, the 1999 version of this standard and the IAEA recommendations. The writing group was reformed in 2005, and this revised standard is a result of the harmonization effort. It is noted that this standard focuses only on clearance, and not on exemption and exclusion, as included in the IAEA recommendations. It is also noted that the IAEA recommendations are based on modeling that uses the dosimetric recommendations of the International Commission on Radiological Protection (ICRP) found in their publication 60 (ICRP 1991). The writing group considered using the dosimetric recommendations from the newer ICRP Publication 103 (ICRP 2007) but concluded that, lacking scenario-specific dose conversion factors, it was not possible at this time.

The purpose and scope of this standard are provided in Section 1.0 and basic definitions are provided in Section 2.0. Section 3.0 contains the basic dose criterion (in terms of the primary dose standard for clearance) and the derived screening levels (in terms of the activity per unit surface area or mass). Section 4.0 provides information useful in the implementation of this standard and covers a wide variety of topics including the role of process knowledge, instrument selection, surface versus volume measurements, summing radionuclide fractions, concentration averaging, removable radioactivity levels, radiological measurements, representative sampling and testing, and quantitative versus qualitative measurement techniques. Section 5.0 provides a discussion of records, and Section 6.0 provides the references cited in this standard. A more complete discussion of the details and technical basis for the harmonization of the previous version of this standard with the IAEA recommendations for clearance are found in Annex A. Annex B contains a discussion of as low as reasonably achievable (ALARA) considerations for clearance. A comparison of the derived screening levels with other guidance is provided in Annex C. Annex D contains informative references supporting the information in Annexes A, B, and C.

Suggestions for improvement of this standard are welcome. Send suggestions to the Health Physics Society, 1313 Dolley Madison Blvd., Suite 402, McLean, VA 22101.

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Surface and Volume Radioactivity Standards for Clearance

1.0 Purpose and Scope

1.1 Purpose

This standard is intended to provide guidance for protecting human health from radiation exposure by specifying a primary radiation dose criterion and derived screening levels for the clearance of items or materials that could contain radioactive materials, in a manner consistent with the recommendations of the International Atomic Energy Agency (IAEA) on clearance.

1.2 Scope

This standard applies to the clearance of items and materials from areas controlled to protect individuals from exposure to radiation or radioactive material during or after operations. This standard establishes a primary radiation dose criterion and derived screening levels for surface and volume radioactivity for groups of radionuclides.

The following are not included in the scope of this standard:

1. naturally occurring radioactive materials (NORM) in bulk;
2. radioactive materials in or on persons, including ^{40}K in the body;
3. release of a licensed or regulated site or facility;
4. radioactive materials on or in foodstuffs;
5. release of land or soil intended for agricultural purposes;
6. clearance issues related to national defense or security;
7. process gases and liquids, including those discharged in accordance with federal or state license requirements;
8. radioactive material in transport in accordance with Department of Transportation, U.S. Nuclear Regulatory Commission (USNRC), or U.S. Department of Energy (DOE) regulations; and
9. radioactive materials at concentrations or quantities governed by regulations or exemption or exclusion.

This clearance standard is not intended as a substitute for radiological criteria for decommissioning or for intervention criteria during cleanup projects. This standard provides screening levels, which are protective of the public health, for the clearance of items or materials.

2.0 Definitions

The following basic definitions are provided to aid in the understanding and interpretation of this standard.

ALARA: The acronym means As Low As is Reasonably Achievable. ALARA is not a standard of care; it is the process through which the Optimization Principle of the radiation protection philosophy is addressed. ALARA means making every reasonable effort to maintain exposures to radiation as far below dose limits as is practical and consistent with the authorized purpose of the practice, taking into account the state of technology, the economics of improvements in relation to the technology, the economics of improvements in relation to benefits to worker and public health and safety, and other societal and socioeconomic considerations.

Background: Background radiation includes both natural and anthropogenic sources. Natural background radiation sources include primordial radionuclides (i.e., radionuclides belonging to the three radioactive decay series headed by ^{238}U, ^{235}U, ^{232}Th), ^{40}K, and ^{87}Rb cosmogenic radionuclides (i.e., radionuclides produced by interactions of cosmic nucleons with target atoms in the atmosphere or in the earth including ^{14}C, ^{3}H, ^{7}Be, and ^{22}Na), and cosmic radiation (i.e., radiation from the secondary particles, mostly high-energy muons and electrons, produced by interactions of charged particles, primarily protons, from extraterrestrial sources with the Earth's atmosphere). Anthropogenic background radiation and radioactive material in the environment includes human-made or human-induced sources such as nuclear fallout from previous weapons testing or significant accidents with major geographic impacts and industrial releases such as emissions from coal-fired combustion that are dispersed to the

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environment. Naturally occurring radioactive material in bulk that has been technologically enhanced is not considered background for purposes of this standard.

Clearance: The removal of items or materials that contain or *may* contain residual levels of radioactive materials within authorized practices from further radiological control for radiation protection purposes.

Committed Effective Dose: The dosimetric quantity formed by multiplying the time integral of the equivalent dose rate in a tissue or organ following an intake by the appropriate tissue weighting factor and then summing the resultant products, as defined by the International Commission on Radiological Protection (ICRP 1991).

Critical Group: A group of individuals in the population expected to receive the highest dose.

Decay Series: The decay of a radionuclide (parent) producing a decay product (progeny) that itself is radioactive and subsequently undergoes radioactive decay, producing its own decay product. This process *may* be repeated several times with each succeeding radioactive decay product undergoing radioactive decay until a stable (i.e., nonradioactive) decay product is produced. The entire sequence from the original parent radionuclide through all of the succeeding decay products to the final stable decay product is a decay series.

DQO: The acronym means Data Quality Objectives, which are an established set of qualitative and quantitative criteria, including acceptable tolerance levels, to specify the quality and quantity of data necessary to support a given decision (e.g., conformance with acceptable clearance screening levels).

Effective Dose: Formed by weighting the equivalent dose in a tissue or organ by the tissue weighting factor, w_T, and summing over the tissues or organs, as defined by the International Commission on Radiological Protection (ICRP 1991). The equivalent dose to the whole body at a depth of 1.0 cm *may* be used as the effective dose for external exposures.

Equivalent Dose: The product of the absorbed dose in a tissue or organ and the radiation weighting factor, w_R, as defined by the International Commission on Radiological Protection (ICRP 1991).

Exclusion: The designation by a regulatory authority that the magnitude or likelihood of an exposure is essentially unamenable to control through requirements of standards and such exposures are outside the scope of standards (e.g., exposure from ^{40}K in the body, from cosmic radiation at the surface of the earth, and from unmodified concentrations of radionuclides in most raw materials).

Exemption: The designation by a regulatory authority that specified uses of radioactive materials or sources of radiation are not subject to regulatory control because the radiation risks to individuals and the collective radiological impact are sufficiently low.

Ionizing Radiation: Includes alpha particles, beta particles, gamma rays, x-rays, and other particles capable of causing ionization of target materials.

May: The term *may* is used in this standard to identify elements of optional guidance potentially useful in interpreting and implementing the requirements of this standard.

MQO: The acronym means Measurement Quality Objectives, which are an established set of performance criteria for a given measurement method necessary to meet the established DQOs for a given clearance action.

Process Knowledge: A collection of facts, information, and data characterizing or describing past or current processes or practices.

Screening Level: 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.

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Shall: The word *shall* is used to identify mandatory elements of this standard needed for full implementation of the standard.

Should: The word *should* is used to identify elements of guidance in this standard. If the guidance is not followed, technical equivalency in outcome must be demonstrated in alternative manners.

Site: A licensed or regulated location that includes one or more (1) facilities designated for the use of radioactive materials or radiation generating devices or (2) radiologically controlled or restricted areas.

Surface Radioactivity: Radioactivity residing on or near the surface of items or materials that can be adequately quantified in units of activity per unit area. When an item or material (including structural components and shielding at nuclear reactors) has been exposed to neutron irradiation, or when an item or material could have cracks or interior surfaces allowing the distribution of radioactivity within the interior matrix, it is conservatively considered to be a volume source of radioactivity.

Survey: A systematic evaluation and documentation of radiological measurements with a correctly calibrated and operated instrument or instruments that meet the detection levels required by the objective of the evaluation.

Technologically Enhanced Naturally Occurring Radioactive Material (TENORM): Naturally occurring radioactive material disturbed or altered from natural settings or present in a technologically enhanced state due to past or present human activities and practices that could result in a relative increase in radionuclide concentrations, radiation exposures and risks to the public, and threat to the accessible environment above background radiation levels. "Technologically enhanced" means that the radiological, physical, and chemical properties of the radioactive material have been altered. In comparing background radioactivity or radiation levels with TENORM, the definition does not provide a point of reference, such as a level equivalent to a multiple of background, because this could lead to situations in which a material not considered TENORM in a high-background area could become TENORM when relocated to a lower-background area. TENORM does not include radiation emanating from or radioactivity present in ores, rocks, soils, and materials containing uranium and thorium subject to regulations under the Atomic Energy Act (ANSI/HPS N13.53-2009).

Total Effective Dose (TED): The sum of the effective dose (for external exposures) and the committed effective dose (for internal exposures) from exposures during a single year.

Volume (or Volumetric) Radioactivity: Radioactive material residing in or throughout the volume of items or materials. Volume radioactivity can result from activation (e.g., through neutron irradiation) or from the penetration or absorption of radioactive materials into porous materials, cracks, or on interior surfaces within the interior matrix of items or materials. A common example is wood that has absorbed fluids containing radioactive materials.

3.0 Dose Criteria and Derived Screening Levels

The following section defines the primary (dose) and secondary screening (derived) criteria that comprise this standard.

3.1 Primary Dose Criterion

The primary criterion of this standard is to provide for public health and safety to an average member of a critical group such that the dose *shall* be limited to 10 μSv/y (1.0 mrem/y) TED for clearance of materials from regulatory control. When justified on a case-by-case basis, clearance *may* be permitted at higher dose levels when it can be reasonably assured that exposures to multiple sources (including those that are beyond the scope of this standard) will be maintained ALARA below established public dose limits (e.g., 1 mSv/y [100 mrem/y]) TED. Justification and application of a higher dose standard for clearance is beyond the scope of this standard.

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3.2 Derived Screening Levels

Table 1 provides screening levels, above background, for the clearance of items or materials that contain surface radioactivity or items or materials that contain volume radioactivity. The screening levels *shall* apply, irrespective of future use or application of the item or material after clearance, given the exclusions noted in Section 1.2. A generic consideration of the ALARA process has been applied in the development of the derived screening levels. Based on a detailed ALARA evaluation, it *shall* be permissible to derive less-restrictive screening levels on a case-by-case basis using the primary dose criterion.

The screening levels for TENORM are equivalent to the surface and volume administrative release levels provided in Table 2-1 of ANSI/HPS N13.53-2009.

A discussion of how the screening levels in Table 1 were derived is found in Annex A. As shown in Table 1, the radionuclides have been divided into five groups based on similarity of screening levels, ranging from 0.1 to 1,000 Bq/cm^2 or Bq/g, depending on the group considered. Surface radioactivity screening levels, in units of Bq/cm^2 (or in conventional units of disintegrations per minute per 100 cm^2) were derived using a surface-to-mass conversion factor (i.e., cm^2/g) of 1. This approach was verified by comparing the scenario results for surface radioactivity only against the scenario results for volume sources for representative radionuclides. For materials where this ratio is significantly different, the surface radioactivity screening levels *shall* be adjusted accordingly, as discussed in Annex A. Note that within a specific radiation protection program, the surface screening *may* be modified based on ALARA considerations or based on operational requirements to limit the amount of surface radioactivity encountered (i.e., radioactivity control requirements).

Annex B contains a discussion of ALARA considerations for clearance. Annex C contains comparisons of the screening levels with previous guidance provided by the U.S. Atomic Energy Commission (AEC) in Regulatory Guide 1.86 (USAEC 1974), and clearance levels published by the IAEA in Safety Series No. RS-G-1.7, “Application of the Concepts of Exclusion, Exemption and Clearance” (IAEA 2004) based on pathway analyses described in a companion report, Safety Report Series No. 44, “Derivation of Activity Concentration Values for Exclusion, Exemption and Clearance” (IAEA 2005).

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Table 1. Screening levels for clearance[a]

Radionuclide groups[b]	SI units Surface (Bq/cm^2) Volume (Bq/g)	Conventional units Surface (dpm/100 cm^2)	 Volume (pCi/g)
Group 1 High-energy gamma, radium, thorium, transuranics, and mobile beta-gamma emitters: ^{22}Na, ^{46}Sc, ^{54}Mn, ^{56}Co, ^{60}Co, ^{65}Zn, ^{94}Nb, ^{106}Ru, ^{110m}Ag, ^{125}Sb, ^{129}I[c], ^{134}Cs, ^{137}Cs, ^{152}Eu, ^{154}Eu, ^{182}Ta, ^{207}Bi, ^{210}Po, ^{210}Pb, ^{226}Ra, ^{228}Ra, ^{228}Th, ^{229}Th, ^{230}Th, ^{232}Th, ^{232}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{244}Pu, ^{241}Am, ^{243}Am, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, ^{249}Cf, ^{251}Cf, ^{254}Es, and associated decay chains[d], and others[b]	0.1	600	3
Group 2 Uranium and selected beta-gamma emitters: ^{14}C, ^{36}Cl, ^{59}Fe, ^{57}Co, ^{58}Co, ^{75}Se, ^{85}Sr, ^{90}Sr, ^{95}Zr, ^{99}Tc, ^{105}Ag, ^{109}Cd, ^{113}Sn, ^{124}Sb, ^{123m}Te, ^{139}Ce, ^{140}Ba, ^{155}Eu, ^{160}Tb, ^{181}Hf, ^{185}Os, ^{190}Ir, ^{192}Ir, ^{204}Tl, ^{206}Bi, ^{233}U, ^{234}U, ^{235}U, ^{238}U, natural uranium[e], ^{237}Np, ^{236}Pu, ^{243}Cm, ^{244}Cm, ^{248}Cf, ^{250}Cf, ^{252}Cf, ^{254}Cf, and associated decay chains[d],and others[b]	1	6,000	30
Group 3 General beta-gamma emitters: ^{7}Be, ^{74}As, ^{93m}Nb, ^{93}Mo, ^{93}Zr, ^{97}Tc, ^{103}Ru, ^{114m}In, ^{125}Sn, ^{127m}Te, ^{129m}Te, ^{131}I, ^{131}Ba, ^{144}Ce, ^{153}Gd, ^{181}W, ^{203}Hg, ^{202}Tl, ^{225}Ra, ^{230}Pa, ^{233}Pa, ^{236}U, ^{241}Pu, ^{242}Cm, and others[b]	10	60,000	300
Group 4 Low-energy beta-gamma emitters: ^{3}H, ^{35}S, ^{45}Ca, ^{51}Cr, ^{53}Mn, ^{59}Ni, ^{63}Ni, ^{86}Rb, ^{91}Y, ^{97m}Tc, ^{115m}Cd, ^{115m}In, ^{125}I, ^{135}Cs, ^{141}Ce, ^{147}Nd, ^{170}Tm, ^{191}Os, ^{237}Pu, ^{249}Bk, ^{253}Cf, and others[b]	100	600,000	3,000
Group 5 Low-energy beta emitters: ^{55}Fe, ^{73}As, ^{89}Sr, ^{125m}Te, ^{147}Pm, ^{151}Sm, ^{171}Tm, ^{185}W, and others[b]	100 (surface)[f] 1,000 (volume)	600,000[f]	30,000

[a]The screening levels for clearance have been rounded to one significant figure and are assigned to both surface and volume radioactivity (assuming an average surface to mass ratio of 1:1, as discussed in Annex A), unless otherwise noted. Note: regulatory authorities *may* increase all volume and surface screening levels by one order of magnitude when clearing bulk quantities of less than 1 metric ton or 1 m^2.

[b]To determine the specific group for radionuclides not shown, a comparison of the screening factors, by exposure scenario, listed in Tables B. 1, C.1, and D.1 of NCRP Report No. 123I (NCRP 1996) for the radionuclides in question and the radionuclides in the general groups above *should* be performed and a determination of the proper group made, as described in Annex A.

[c]Because of potential ground-water concerns, the volume or surface radioactivity values for ^{129}I *should* be lowered by one order of magnitude when disposal to landfills or direct disposal to soil is anticipated.

[d]For decay chains, the screening levels represent the total activity (i.e., the activity of the parent plus the activity of all progeny) present.

[e]The natural uranium screening levels for clearance *shall* be lowered from Group 2 to Group 1 if decay-chain progeny are present (i.e., uranium ore versus process or separated uranium, for example, in the form of yellowcake). The natural uranium activity equals the activity from uranium isotopes (48.9% from ^{238}U, plus 48.9% from ^{234}U, plus 2.2% from ^{235}U). This approach is consistent with summing radionuclide fractions discussed in Section 4.4.

[f]For radioactivity control considerations, surface radioactivity screening levels for Group 5 radionuclides are controlled to the Group 4 surface radioactivity screening levels.

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3.3 Screening Level Discussion

The radionuclide groupings for the screening levels for clearance shown in Table 1 were determined based on the potential of the radionuclides in the group to deliver similar radiation doses for the pathways and scenarios evaluated by the IAEA (2004, 2005). The titles of Groups 1 through 5 were developed as a shorthand notation, which *may* not describe the radiations emitted by all of the radionuclides assigned to a particular group. For example, Group 2, uranium and selected beta gamma emitters, also contains selected transuranic radionuclides that are alpha emitters. However, all of the radionuclides in Group 2 have a similar potential to deliver 10 μSv (1 mrem) TED per year for the scenarios and pathways identified, independent of the type of radiation emitted.

As noted in footnote a to Table 1, the surface screening levels were derived based on a surface-to-mass ratio of 1 to 1 (cm^2/g). The surface screening levels *should* be reduced proportionally for items or materials where this ratio is greater than 1 to 1 (for items such as thin sheet metal), and increased proportionally when the ratio is less than one (for items such as heavy metal components). A discussion of surface to mass ratios is included in Annex A. Footnote a to Table 1 also provides guidance for establishing volume screening levels when small quantities are encountered, indicating that regulatory authorities *may,* with appropriate justification and scenario documentation, increase the volume screening levels by one order of magnitude when quantities are less than 1 metric ton. This statement is in recognition of the reduced ability of small quantities of cleared material to deliver radiation doses to individuals and is consistent with the guidance provided by the IAEA (2004, 2005).

As stated in Footnote b to Table 1, to determine the specific group for radionuclides not shown, a comparison of the screening factors, by exposure scenario, listed in Tables B.1, C.1, and D.1 of NCRP Report No. 123I (NCRP 1996) for radionuclides in question and the radionuclides in the general groups *should* be performed and a determination of the proper group made, as described in Annex A. It is recognized that the selection of reference radionuclides in the groups shown in Table 1 could influence the outcome of this process. For this reason, at most three radionuclides from the relevant groups from Table 1 for comparison *shall* be selected that (1) have similar half-lives, (2) have similar chemical properties (i.e., are from the same elemental groups on the periodic table), and (3) represent similar types of radiation. As also discussed in Annex A, the IAEA guidance (2004) incorporated derived levels for exemption and exclusion, when appropriate, in addition to clearance. This means that IAEA includes values for numerous short-lived radionuclides assuming exposure to a continuously replenishing source (such as an effluent stream from an operating facility), rather than clearance where the radioactivity associated with cleared items or materials would rapidly decrease with time. Values from the IAEA that were derived for exemption and exclusion are not included in this standard because the focus is on clearance; thus, direct use of the IAEA clearance levels *should* be done with caution. As a practical matter, *should* short-lived radionuclides be encountered, the items or material could be stored for radioactive decay prior to clearance.

As noted in footnote c to Table 1, the IAEA lowered the clearance level for ^{129}I by one order of magnitude because of its potential environmental mobility through the ground-water pathway. This is a concern when disposal of bulk materials to soil is encountered. For this reason, the surface and volume screening levels for clearance of ^{129}I-bearing items or materials *should* be lowered by one order of magnitude when bulk disposal to landfills or direct disposal to soil is anticipated. It is also noted that release of land or soil intended for agriculture is beyond the scope of this standard because this situation is associated with release of land following decommissioning or license termination activities.

Footnote d to Table 1 indicates that the screening levels for clearance for decay chains represent the total activity of the

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parent plus all of the progeny present. The screening level could appropriately be derived using the sum of fractions methods outlined in Section 4.4.

For Footnote e to Table 1, because natural uranium and other radionuclides of natural origin are ubiquitous in the environment with varying concentrations, the IAEA concluded that the resulting exposures from the same radionuclides that are present from practices are often unamenable to control through the requirements of their Basic Safety Standards (IAEA 2004). The IAEA activity concentrations for clearance were determined on the basis of consideration of the worldwide distribution of activity concentrations for these radionuclides, not a dose criterion of 10 μSv/y (1.0 mrem/y) TED with activity concentrations derived through a scenario analysis. However, several naturally occurring radionuclides are individually included in licensed or controlled sources and screening levels for clearance are needed. The clearance screening levels for radionuclides of natural origin would be consistent with those derived for artificial (human-made) radionuclides. For this reason, individual naturally occurring radionuclides are included based on N13.12-1999 (in Group 1) and uranium radionuclides that have been separated from their decay chain progeny (in Group 2). As a result, footnote e to Table 1 indicates that the screening levels for clearance are for the uranium radionuclides only, that is, assuming that the uranium had been processed to remove non-uranium progeny. When those progeny are present in equilibrium with the uranium decay-chain parents, the screening levels *shall* be lowered by one order of magnitude to reflect the screening level that would be derived using the sum of fractions methods outlined in Section 4.4. This approach is consistent with a companion ANSI/HPS standard (N13.53-2009) and takes into consideration cases in which individual naturally occurring radionuclide decay-chain members could be in licensed or controlled materials, independent of the entire decay chain. This general guidance is applicable when decay progeny are in a more restrictive clearance screening group than the parent and is consistent with the summing radionuclide fractions information presented in Section 4.4.

Finally, the derived surface radioactivity screening levels for Group 5 radionuclides was inconsistent with the principles of radioactivity control for operating facilities. Therefore, the surface radioactivity screening level for Group 5 radionuclides were lowered to the Group 4 surface radioactivity screening level.

4.0 Implementation

Measurement of radioactivity and the quantity of radioactive material on or in items or materials suitable for clearance *shall* be performed to verify conformance with the screening levels established in Section 3.0 of this standard. Radiological measurements performed for the purpose of clearance *shall* include direct field surveys of the item or material, laboratory analysis of representative samples of the item or material, or a combination of both, with measurement results characterizing radioactivity levels from accessible surface areas and internal volumes.

4.1 Process Knowledge

When invoked, process knowledge *shall* be documented, as is appropriate, with supporting technical information, including plans, procedures, specifications, assumptions, algorithms, definitions, expert testimony, and historical measurements. It is recognized that some degree of professional judgment is used in all situations; however, it is often most relevant to situations in which information or data are not reasonably available by collection, survey, literature search, or experimentation. Radiological measurements performed for the purpose of clearance *shall* take into consideration the following:

a) all relevant operational history of the items or materials to identify the known or potentially present radionuclides and associated activity levels;
b) the potential radionuclide distribution (on the surface of the item or material and/or within the volume of the item or material);

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c) the size, geometry, composition, and physical properties of the item or material;
d) any associated radionuclides that result from radioactive decay and progeny in-growth; and
e) the use of surrogate radionuclides in interpreting the activity of hard-to-detect radionuclides when their respective relationships are well established.

4.2 Volume and Surface Measurements

Volume measurements for clearance *shall* be used when volume radioactivity in items or materials is known or potentially present. The number and location of samples taken and analyzed *shall* be sufficient to provide reasonable confidence that they are representative for the item or material being cleared (see Section 4.9).

Volume measurements for clearance *may* be used in lieu of surface levels provided that:

a) the size or shape or the item or material makes it unreasonable to perform radiological surveys representative of the radioactivity on all of the surfaces of the item or material;
b) the item or material can be representatively sampled for laboratory analysis; and
c) it can be demonstrated that the use of volume measurements for clearance is as protective as using surface measurements (e.g., the volume sampling is not used to dilute excessive surface activity to meet the clearance screening levels).

Surface screening levels *shall* be used when the size or shape of the item or material reasonably allows direct radiological surveys for surface radioactivity.

Items or materials known to have activity levels that are in excess of the clearance screening levels *should* not be:

a) intentionally blended with lower activity material for the purpose of meeting this standard; or
b) intentionally coated, plated, encased, or covered to reduce the apparent surface or volume radioactivity.

4.3 Concentration Averaging

Averaging is inherent to the radiological measurement process for determining both surface activity and volume activity concentrations. When performing radiological measurements of items or materials for clearance, a determination of the average radionuclide concentrations *should* be performed such that:

a) Multiple surface measurements are averaged over an established item or material surface area not to exceed 1 m^2, or an area already specified in applicable regulations, or an area determined by the DQO/MQO process. For items or materials with a surface area less than the established surface area for averaging measurements, an average over the entire surface *shall* be derived for each item or material. The intent is that there must be a reasonable expectation that average concentration of radioactive material is less than the applicable surface screening level over (1) the established surface area for averaging measurements for items or materials or (2) the total surface area of each item or material when that area is less than the applicable surface screening level. This does not necessarily require physical measurement of each item or material over every specified area for averaging (see Sections 4.8, 4.9, and 4.10).
b) Multiple volume measurements are averaged over an established total volume or mass not to exceed 1 m^3 or a mass of 1 metric ton, or the volume or mass specified in applicable regulations, or the mass or volume determined by the DQO/MQO process (which could be larger than 1 m^3 or a mass of 1 metric ton). For items or materials with mass less than the established total volume or mass for averaging measurements, an average over the entire mass *shall* be derived for each item or material. The intent is that there must be reasonable confidence that the volume concentration of radionuclides in any established

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volume or mass for averaging measurements for items or materials will be less than the applicable volume or mass screening level. This does not necessarily require multiple physical measurements of every specified volume or mass for averaging of the material cleared.

c) No single measurement made to calculate an average surface activity *shall* exceed 10 times the surface screening level.

d) No single measurement made to calculate an average volume activity concentration *shall* exceed 10 times the volume screening level.

In lieu of multiple representative measurement sampling, single in situ or in toto measurement techniques *may* be used to determine conformance with the surface and volume screening levels in Section 3.0 of this standard. Process knowledge *may* be used to support the determination that the surface or volume activity concentrations are homogeneously distributed on or throughout the material.

4.4 Summing Radionuclide Fractions

When radiological measurements, or determinations of relative abundance of radionuclides, are performed for the purpose of clearance for a mixture of radionuclides that is known or potentially present, a determination of whether or not the radionuclide mixture meets the screening levels provided in Section 3.0 of this standard *shall* be made.

If the radionuclides in the mixture are from the same group, as shown in Table 1, the following sum of fractions *may* be used to determine whether the mixture meets the applicable screening level:

$$\sum_{i=1}^{n} C_i / SL_g \leq 1, \quad (1)$$

where C_i is the measured or determined activity concentration of radionuclide, i, in or on the material; SL_g is the screening level of the radionuclide group of interest, g; and n is the number of radionuclides in the radionuclide mixture for the group of interest. Eqn (1) presumes that radiological measurements or determinations are made for each individual radionuclide in the mixture. If measurements or determinations are made of the total activity concentration of all radionuclides in the mixture, then eqn (1) *may* be simplified to the following:

$$C_T / SL_g \leq 1 \quad (2)$$

where C_T is the measured or determined total activity concentration of all radionuclides in or on the material and SL_g is the screening level of the applicable group of interest, g. If measurements are used to demonstrate compliance within a group where both alpha and beta gamma emitters are present and individual measurements of total alpha and total beta gamma are made, the sum of the fractions will apply, and the sum of the total alpha and total beta gamma measurements divided by SL_g must be less than or equal to 1.

When a mixture contains radionuclides from more than one group, the sum of fractions *shall* be applied across all applicable radionuclide groups. If radiological measurements or determinations are made for each individual radionuclide in the mixture, the ratio of the concentration of each radionuclide to the associated screening level for its group is summed over all of the radionuclides in the mixture. If the sum of these ratios is less than or equal to 1, the screening level requirements are met:

$$\sum_{i=1}^{n} (C_i / SL_{gi}) \leq 1, \quad (3)$$

where C_i is the measured or determined activity concentration of radionuclide i in or on the material; SL_{gi} is the screening level of the applicable group, g, for a given radionuclide i; and n is the number of radionuclides in the radionuclide mixture.

If the specific groups in a mixture are known, and measurements or determinations are made of the total activity concentrations for each group, the ratio of the total activity concentration of each group to the associated screening level for that group is summed over all of the groups in the mixture. If the sum of these ratios is less

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than or equal to one, the screening level requirements are met:

$$\sum_{m=1}^{k} (C_m/SL_{gm}) \leq 1, \quad (4)$$

where C_m is the measured or determined total activity concentration of a given group, m; SL_{gm} is the screening level of this group, m; and k is the number of groups in the mixture. If measurements or determinations are made that are considered representative of the total activity concentrations in more than one group, the measured or determined total activity concentration is apportioned among the groups for which the measurement or determination is considered representative. This apportionment is based on the known or presumed relative abundance of total activity concentration (for a given type of measurement or determination, e.g., alpha or low-energy beta) in each group to the total activity concentration (for the same type of measurement or determination) of the mixture. The ratio of the total activity concentration of each group to the associated screening level for that group, times the relative abundance of that group, is summed over all of the groups in the mixture. If the sum of these ratios is less than or equal to 1, the screening level requirements are met:

$$\sum_{m=1}^{k} f_m(C_M/SL_{gm}) \leq 1, \quad (5)$$

where C_M is the measured or determined total activity concentration of the mixture; f_m is the measured or documented fractional abundance of a given group, m, in the mixture; SL_{gm} is the screening level of a given group, m; and k is the number of groups in the mixture.

The application of the sum of fractions method across groups is illustrated in the following example. Based on process knowledge, the surface radioactivity associated with an item or material proposed for clearance is composed primarily of ^{63}Ni with lower levels of ^{90}Sr, ^{137}Cs, and $^{239/240}$Pu. Radiological measurements of the radioactivity on the item or material give the following results: ^{63}Ni – 50 Bq/cm^2 (300,000 dpm/100 cm^2), ^{90}Sr – 0.05 Bq/cm^2 (300 dpm/100 cm^2), ^{137}Cs – 0.05 Bq/cm^2 (300 dpm/100 cm^2), and $^{239/240}$Pu – 0.02 Bq/cm^2 (120 dpm/100 cm^2). The screening levels from Table 1 for these radionuclides are as follow: ^{63}Ni (Group 4) – 100 Bq/cm^2 (600,000 dpm/100 cm^2), ^{90}Sr (Group 2) – 1 Bq/cm^2 (6,000 dpm/100 cm^2), and ^{137}Cs and $^{239/240}$Pu (Group 1) – 0.1 Bq/cm^2 (600 dpm/100 cm^2). Because ^{137}Cs and $^{239/240}$Pu are in the same group, their results *may* be summed (total of 0.07 Bq/cm^2 [420 dpm/100 cm^2]) for comparison with the group screening level. Comparison of the radiological measurements against the applicable group screening levels shows each radionuclide and grouping (^{137}Cs and $^{239/240}$Pu – Group 1) is less than the applicable screening level. However, because these radionuclides represent different groups in Table 1, it is necessary to assess the total radioactivity from these radionuclides using a sum of fractions. Because radiological measurements for each individual radionuclide are available, the sum of fractions is assessed using eqn 3:

{(Result for ^{63}Ni/Group 4 screening level)
+ (Result for ^{90}Sr /Group 2 screening level)
+ (Result for ^{137}Cs and $^{239/240}$Pu /Group 1 screening level)}
= [50 Bq/cm^2 (300,000 dpm/100 cm^2)]
/ [100 Bq/cm^2 (600,000 dpm/100 cm^2)]
+ [0.05 Bq/cm^2 (300 dpm/100 cm^2)]
/ [1 Bq/cm^2 (6,000 dpm/100 cm^2)]
+ [0.07 Bq/cm^2 (420 dpm/100 cm^2)
/ [0.1 Bq/cm^2 (600 dpm/100 cm^2)]
= [0.5] + [0.05] + [0.7] = 1.25 (6)

This sum of fractions exceeds 1. Therefore, the total radioactivity associated with this item or material exceeds the screening levels, and the item or material could not be cleared under this standard without a specific technical justification (i.e., specific dose assessment and ALARA evaluation).

A sum of fractions determination is not necessary for progeny radionuclides with half-lives that are short compared to the half-life of the parent (i.e., those that reach a constant ratio of activity with the parent as a

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function of time that is short when compared to 1 y, the exposure duration used in the scenario analysis in the calculation of TED). These radionuclides have been included with their parent radionuclides (e.g., the short-lived progeny associated with ^{90}Sr, ^{137}Cs, ^{144}Ce, ^{210}Pb, ^{226}Ra, ^{235}U, ^{238}U, and others) as established in Section 3.0 this standard.

The known or potentially present radionuclide mixture *shall* be supported by historical and process knowledge and laboratory sample analysis, as necessary. To account for radionuclides that are known to be or are potentially present in the mixture, but are not detectable by the methods used to detect their presence at the screening levels, the sum of fractions equations above *may* be modified by adding a factor, D, which is the detectable fraction of the radionuclide mixture for the methods used, for each specific radionuclide, D_i, or for each radionuclide group, D_m, in the mixture. For this case, eqn (1) is modified as follows:

$$\sum_{i=1}^{n} (C_i/SL_g)/D_i \leq 1, \qquad (7)$$

where D_i is the detectable fraction of each radionuclide in the mixture. The detectable fraction represents the actual fraction of radionuclides that are measured by the survey method.

Application of these sum-of-fractions' equations *should* focus on the most significant radionuclides present. If a mixture contains radionuclides that are calculated to contribute less than 1% of the total dose for the mixture, these radionuclides *may* be ignored unless it is determined that the total dose from all such "insignificant" contributors exceeds 10% of the total dose for the mixture.

4.5 Removable Radioactivity Levels

In most situations, measurement of removable surface activity levels is not necessary or appropriate to demonstrate compliance with this standard. However, measurements of removable surface activity levels could be appropriate and *may* be included as part of survey programs for radioactivity control purposes, when justified, to support ALARA evaluations and associated decisions, and under special circumstances to support conformance with applicable screening levels (see Section 4.8.2.2).

4.6 Measuring Radionuclides Above Background

It is recognized that difficulties in distinguishing radionuclide activity concentrations resulting from an authorized process from variations in natural background levels could occur. When measuring materials that contain naturally occurring radionuclides (e.g., ^{226}Ra, ^{232}Th, ^{238}U, and associated decay chains), a determination *shall* be made as to whether individual radionuclides are present at levels above natural background, or exceeding licensed or regulated quantities. Measurements *should* be of sufficient quality and quantity to determine whether such radionuclides that have been technologically enhanced are present in concentrations above the screening levels.

4.7 Instrument Selection

Instruments used for radiological measurements *shall* be:

a) selected based upon the instrument's detection capability for each known or potential radionuclide or mixture of radionuclides;
b) selected and calibrated for the chosen surrogate radionuclide(s) when hard-to-detect radionuclides are present in the radionuclide mixture;
c) capable of measuring the quantity of the detectable radionuclides on or in the item or material;
d) capable of distinguishing from background radionuclides at or below the screening levels established in Section 3.0 of this standard for a specified background level;
e) calibrated (National Institute of Standards and Technology [NIST] or internationally traceable, potentially using International Organization for Standardization [ISO] reference radiations) for the known or

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potential radionuclide spectrum and distribution;

f) operated and maintained by qualified personnel, in accordance with an appropriate Quality Assurance program (e.g., split samples, cross-checks, response/operational checks); and

g) capable of withstanding the conditions encountered during performance of measurements (e.g., temperature, humidity, vibration, and/or shock).

4.8 Radiological Measurements

Radiological measurements *shall* be performed to verify conformance with the screening levels established in Section 3.0 of this standard. Whenever such measurements are performed to determine radionuclide activities or activity concentrations, DQOs *shall* be established. The purpose of these DQOs is to establish a set of qualitative and quantitative criteria, including acceptable tolerance levels, to specify the quality and quantity of data necessary to support a given decision regarding conformance with acceptable clearance screening levels.

As an important part of the DQOs, MQOs *shall* be established for a given measurement method. MQOs provide the performance criteria for a measurement method necessary to meet the purposes of the DQOs. MQOs *shall* include consideration of the measurement methods:

a) measurement uncertainty at specified radionuclide activities and/or activity concentrations;

b) detection capability for specified radionuclides;

c) practical constraints on the time required to perform the measurement;

d) quantification capability for specified radionuclides;

e) range, that is, the measurement method's ability to measure the specified radionuclides or radiations over some specified range of surface or volume radioactivity or concentrations;

f) specificity, that is, the ability of the measurement method to detect or quantify the specified radionuclides or radiations in the presence of interferences; and

g) robustness, that is, the stability of the measurement method's performance relative to small variations in measurement method parameter values.

MQOs are also useful in selecting and evaluating potential measurement methods or instruments (see Section 4.7 of this standard).

4.8.1 Direct Measurements Measurement methods can generally be categorized as either direct or indirect measurement methods. Typical direct measurement methods which could be performed include (1) in situ measurements and (2) scanning.

4.8.1.1 In Situ Measurements In situ measurements *shall* be performed in accordance with established DQOs and MQOs. In situ measurements *should* be performed by placing the measurement instrument in a fixed position at a specified distance from the surface of a given item or material and taking a discrete measurement, using a scaler or other radioanalytical measuring device, for a predetermined time interval. Single in situ measurements *may* be performed on individual items, groups of items, or materials. Multiple in situ measurements *may* be combined to provide several different measurements of the same item or material, or *may* be used to provide measurements for a specified fraction of a group of items or material. This type of measurement allows for easy calculation of the measurement sensitivity and detection capabilities. The uncertainty of a single measurement is also easily evaluated for quantitative measurements.

In situ measurements are generally used to provide an estimate of the average radioactivity or radionuclide concentration over a certain area or volume defined by the instrument sensitivity and calibration. Determining the instrument sensitivity and calibration for situations other than radionuclides uniformly distributed on a plane or through a regularly shaped volume (e.g., a disk or cylinder) can be complicated and could contribute to the total measurement method uncertainty.

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4.8.1.2 Scanning Scanning measurements *shall* be performed in accordance with established DQOs and MQOs. Scanning measurements *should* be performed by moving the measurement instrument at a specified speed at a specified distance above the surface of a given item or material while monitoring the count rate (e.g., with a rate meter or other radioanalytical measuring device). Alternatively, the item or material *may* be moved past a stationary instrument at a specified distance and speed (e.g., convey systems or certain portal monitors). Audio monitors *may* also be useful to identify areas of elevated activity. Scanning *may* also be done using the scalar counter to demonstrate that a scanned area meets the clearance screening criteria. Scanning measurements can more readily provide thorough coverage of a given item or material and are often relatively quick and inexpensive to perform.

Scanning measurements typically involve many variables such as scan speed and distance and variable background as the instrument or item or material is moved. It could be difficult to control these variables precisely, especially with hand-held instruments and irregularly shaped items or materials, which could lead to increased measurement uncertainty. Determining a calibration function for geometries other than surface radioactivity uniformly distributed on a plane can be complicated and could contribute to the total measurement uncertainty. Therefore, this type of measurement does not allow for easy calculation of detection limits or typical counting statistics.

Although counting statistics related to a single measurement are not typically calculated when performing scanning measurements, this type of measurement *may* be used to survey items or materials for clearance with reasonable confidence. Although the level of confidence might not be easily quantified by calculation, it can be demonstrated through repetitive measurements of a radioactive standard designed such that the standard's activity or activity concentration is consistent with the screening levels established in Section 3.0 of this standard.

4.8.2 Indirect Measurements Indirect measurement methods typically involve the collection and analysis of representative samples of the items or materials being cleared. Such measurements are typically performed with laboratory instrumentation or instrumentation setup in laboratory-like conditions with sample preparation and measurements conducted using established procedures and trained personnel.

4.8.2.1 Sampling Sampling *shall* be performed in accordance with established DQOs and MQOs. Sampling consists of removing a portion of the item or material for separate laboratory analysis. This measurement method typically has detection capabilities that are more sensitive than direct measurement methods, particularly for the analysis of complicated radionuclide mixtures, difficult-to-measure radionuclides, and extremely low concentrations of radioactivity. Sampling is typically used to provide an estimate of the average radioactivity or radionuclide concentration for a specified item or volume of material. Sampling *may* also be used to validate data collected using other measurement methods. Sampling typically requires the most time for data generation and is often the most expensive measurement method.

4.8.2.2 Swipes Swipes, alternatively termed *wipes* or *smears*, are a specialized form of sampling used to measure removable surface radionuclides or radioactivity. This standard does not specify screening criteria for an allowable removable fraction and hence, swipes are not a required tool in determining conformance to the screening levels for most radionuclides. However, they are useful for radioactivity control and to support ALARA decisions.

Except as noted elsewhere, swipes *should* be performed by using a dry filter paper or other absorbent material or suitable substance to wipe a specified area of a surface. The filter paper or other substance is then appropriately analyzed for specific radionuclides or radioactivity.

Swipe results can vary because the fraction of surface radioactivity transferred to the dry filter paper or other material is unknown and

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variable, and the surface area covered by the swipe could be variable. In addition, the results can vary with time due to environmental factors or interactions of surface activity with the surface itself. To minimize variability of results, it is important that individuals responsible for collecting swipes be appropriately trained so as to ensure a consistent application of the sampling approach. Using a template or cutout with a known area can help control the variability in the area covered by a swipe. Using a tool that applies consistent pressure while collecting swipes can reduce the variability in the fraction of radioactivity removed. Implementing a protocol for preparing surfaces and sorting items or materials prior to swiping can reduce variability in surface textures and conditions, resulting in lower variability in swipe results.

In special situations in which direct measurement of radioactivity could be impractical, swipes *may* be an effective tool (e.g., situations in which tritium is a major radionuclide of concern). Direct measurement of tritium is difficult and use of swipes *may* be the most practical approach for comparison to the screening level. Swipes for tritium *should* be taken with moist filter paper or styrofoam media because they are more effective in capturing tritium than dry filter paper.

It is noted that for some surfaces and for some radionuclides (such as tritium), removable surface activity levels can vary over time as the radionuclides seep from porous surfaces. Thus, caution *should* be used when obtaining and interpreting removable surface radioactivity results.

4.9 Statistically Based Representative Measurements

Appropriate statistical methods for determining conformance with both surface and volume screening levels *may* be used when the volume or size of the item or material makes it unreasonable to perform 100% direct radiological surveys, and

a) the radioactivity is known, based on process knowledge, to be generally homogeneous in distribution throughout the item, or group of items, or material;
b) the origin and physical form of the item or material is such that any radioactivity present would be generally homogeneous in distribution on the surface of an item or throughout the material; or
c) process knowledge indicates that the potential for presence of radioactivity is low.

The statistical methods or tests, random or stratified as appropriate, chosen *should*

a) test the assumed null hypothesis (e.g., typically, the null hypothesis is that radioactivity exists above the clearance levels);
b) take into account the uncertainty of background measurements;
c) be sufficient to detect known or potentially present radionuclides at the screening level with a reasonable confidence level;
d) perform adequately for data sets that include values detectable above background; and
e) perform adequately with non-normally distributed data.

As with any radiological measurement, whenever representative measurements are performed to determine radionuclide activities or activity concentrations, applicable DQOs and MQOs *shall* be established. All representative measurements *shall* be performed in accordance with these established DQOs and MQOs.

4.10 Implementation Approach: MARSAME

Implementation of the screening levels established in Section 3.0 of this standard for clearance of items and materials, including consideration of the factors discussed in Sections 4.1–4.4 and 4.6–4.9 above, *should* be conducted using an established, technical framework for such activities.

The *Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual* (MARSAME) (MARSAME 2009) is an example of one such acceptable technical framework as well as the

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associated methods for the clearance of items and materials. MARSAME is a supplement to the *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) (MARSSIM 2002), and like MARSSIM is a joint effort by the Department of Defense (DOD), DOE, Environmental Protection Agency (EPA), and U.S. Nuclear Regulatory Commission (USNRC). MARSAME also incorporates information, in particular statistical methodologies, for the measurement of radioactivity from the *Multi-Agency Radiological Laboratory Analytical Protocols Manual* (MARLAP 2004).

MARSAME provides technical information and graded approaches for planning, implementing, assessing, and documenting surveys for the appropriate disposition, including clearance, of items and materials. Included is information on selecting appropriate measurement methods as well as information on the development and use of appropriate DQOs and MQOs. Figure 1 provides a diagram of the MARSAME approach to the disposition of items and materials. In this diagram the screening levels established in Section 3.0 of this standard would be considered "Action Levels." MARSAME provides statistical approaches that could be used to establish sampling frequencies and averaging areas or volumes, as described in Section 4.3 of this standard. Additional details and information on the MARSAME approach can be obtained from the references cited above.

MARSAME does not represent the only acceptable approach for clearance of items and materials. Discussion of MARSAME in this standard is not considered an endorsement of this approach, but rather, an example of an established technical framework, which the signatory agencies find acceptable and useful for most item and material disposition situations. A careful assessment of the applicability of any technical approach, such as MARSAME, must be performed before use in implementing the screening levels established in Section 3.0 of this standard.

5.0 Records

The following records, in addition to those required for radiation survey reports in "Practice for Occupational Radiation Exposure Records Systems" (ANSI/HPS N13.6-2010), *shall* be maintained:

a) description and results (e.g., specific criteria and results of the DQO and MQO processes for each clearance action, as applicable);
b) description of item or material cleared or dispositioned. Note: the description *should* be sufficiently complete to permit a knowledgeable person to identify the item or material and to associate the description with the measurement results for the item or material;
c) detector and/or rate meter/scaler identification numbers;
c) measurement results, date, and identity of the person(s) who performed the measurement;
d) archived procedures or records that specify pertinent details of calibration, operating instructions, personnel training, derivation of efficiencies and conversion factors, and other technical details of the clearance method; and
e) summary descriptions of disposition methods or end-use of materials cleared under the provisions and criteria of this standard, as implemented under operational procedures.

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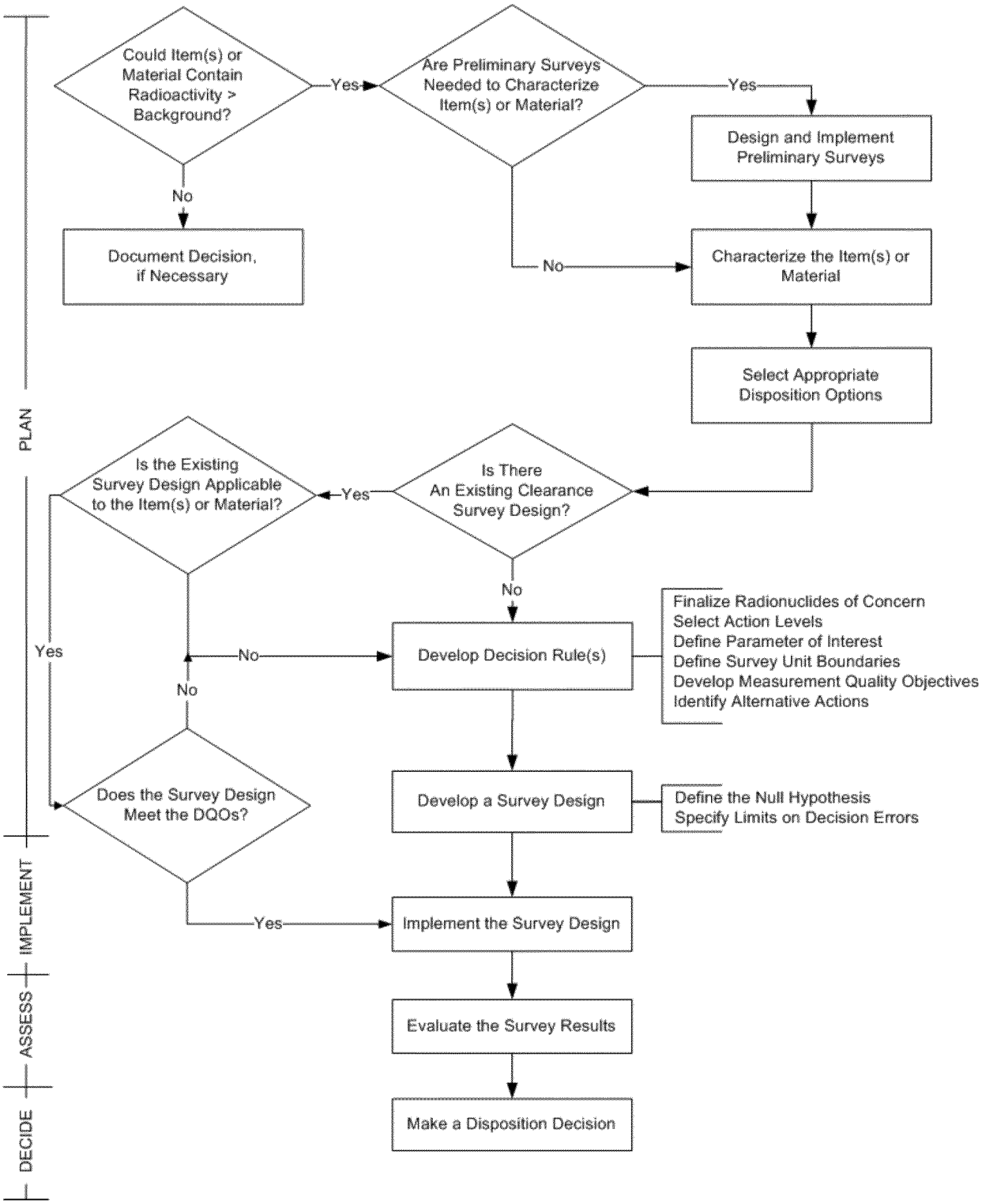
Figure 1. MARSAME Approach to Clearance[a]



[a]Adapted from Roadmap Figure 2 of MARSAME (2009).

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6.0 Normative References

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Annex A

(Informative)

IAEA Clearance Recommendations and Harmonization with This Standard

A.1 Introduction

During 2005, the ANSI/HPS N13.12 (1999) writing group was reconvened to consider new information provided by the IAEA in their Safety Standards Series, Safety Guide No. RS-G-1.7, "Application of the Concepts of Exclusion, Exemption and Clearance) (2004). Specifically, the writing group was tasked to evaluate the IAEA reference and determine whether it would be possible to harmonize its content with an updated version of N13.12-1999. To that end, a comparison of the IAEA Safety Standards Series and N13.12-1999 was conducted. The following paragraphs summarize the IAEA recommendations and the results of the comparisons and analysis and provide the specific recommendations that best achieved harmonization of the technical content.

A.2 IAEA Recommendations

The objective of IAEA RS-G-1.7 (IAEA 2004) was to "provide guidance to national authorities, including regulatory bodies, and operating organizations on the application of the concepts of exclusion, exemption and clearance as established in the BSS" (Basic Safety Standards) (IAEA 1996). In this context, exclusion relates to exposures from sources that are impractical or essentially impossible to control, such as those from some sources of naturally occurring radioactive material or those from cosmic radiation. Exemption means exempt from the requirements for practices covered by the IAEA BSS, as determined by national or regulatory authorities. The IAEA provided exemption values for moderate quantities of material in Schedule I of their BSS, which apply to materials that pose sufficiently low radiation risks to be of no regulatory concern, with no appreciable likelihood of exposure scenarios that could lead to radiation exposures of concern. The IAEA exemption values have not been adopted by the United States. Although clearance is similar to exemption, it specifically relates to the removal of radioactive material within authorized practices from further control by the regulatory body, such as for bulk materials. Although exemption may apply to radionuclides of both natural and artificial (human-made) origin, the IAEA limited clearance to consider only those of artificial origin. The clearance levels were derived generally as the lowest values obtained from a number of exposure scenarios involving direct exposure to or handling of cleared materials and accounting for potential foodstuffs and drinking water pathways of intake. For a number of short-lived radionuclides, the IAEA exposure scenario results for clearance led to levels that are higher than the exemption levels given in the BSS. For these radionuclides, the values chosen were the exemption levels of Schedule I of the BSS, and not those derived from the clearance scenarios. The resulting values of activity concentration for radionuclides of artificial origin in bulk (i.e., equal to or greater than 1 metric ton) from Table 2 of IAEA RS-G-1.7 are listed in Table A.1, with those derived from the IAEA exemption levels shown in ***bold italics***. The IAEA did not provide recommendations for surface radioactivity in IAEA RS-G-1.7, nor did it provide values for uranium and thorium decay chain members since these were considered to be of natural origin.

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Table A.1. Values of activity concentration for radionuclides of artificial origin in bulk (i.e., equal to or greater than 1 metric ton) from IAEA RS-G-1.7[a]

Radionuclide	Half-life	IAEA bulk levels (Bq/g)	Radionuclide	Half-life	IAEA bulk levels (Bq/g)
^{3}H	12.35 y	100	***^{135}I***	***6.61 h***	***10***
^{7}Be	53.3 d	10	***^{129}Cs***	***32 h***	***10***
^{14}C	5,730 y	1	***^{131}Cs***	***9.7 h***	***1,000***
^{18}F	***44.5 d***	***10***	***^{132}Cs***	***6.5 d***	***10***
^{22}Na	2.6 y	0.1	^{134}Cs	2.1 y	0.1
^{24}Na	***15 h***	***1***	***^{134m}Cs***	***2.9 h***	***0.1***
^{31}Si	***156 m***	***1,000***	^{135}Cs	2.3×10^{6} y	100
^{32}P	***14.3 d***	***1,000***	***^{136}Cs***	***13.1 d***	***1***
^{33}P	***25.4 d***	***1,000***	^{137}Cs	30 y	0.1
^{35}S	87 d	100	***^{138}Cs***	***32 m***	***10***
^{36}Cl	3.0×10^{5} y	1	^{131}Ba	11.7 d	10
^{38}Cl	***37.2 m***	***10***	^{140}Ba	12.8 d	1
^{42}K	***12.4 h***	***100***	***^{140}La***	***40.3 h***	***1***
^{43}K	***22.6 h***	***10***	^{139}Ce	138 d	1
^{45}Ca	163 d	100	^{141}Ce	32.5 d	100
^{47}Ca	***4.53 d***	***10***	***^{143}Ce***	***33 h***	***10***
^{46}Sc	83.8 d	0.1	^{144}Ce	284 d	10
^{47}Sc	***3.35 d***	***100***	***^{142}Pr***	***19.1 h***	***100***
^{48}Sc	***43.7 h***	***1***	***^{143}Pr***	***13.6 d***	***1,000***
^{48}V	***16.2 d***	***1***	^{147}Nd	11 d	100
^{51}Cr	27.7 d	100	***^{149}Nd***	***1.7 h***	***100***
^{51}Mn	***46.2 m***	***10***	^{147}Pm	2.62 y	1,000
^{52}Mn	***5.6 d***	***1***	***^{149}Pm***	***53.1 d***	***1,000***
^{52m}Mn	***21.1 m***	***10***	^{151}Sm	93 y	1,000
^{53}Mn	3.7×10^{6} y	100	***^{153}Sm***	***46.7 m***	***100***
^{54}Mn	312.5 y	0.1	^{152}Eu	13.3 y	0.1
^{56}Mn	***2.58 h***	***10***	***^{152m}Eu***	***9.32 h***	***100***
^{52}Fe	***8.3 h***	***10***	^{154}Eu	8.8 y	0.1
^{55}Fe	2.7 y	1,000	^{155}Eu	4.96 y	1
^{59}Fe	44.5 y	1	^{153}Gd	242 d	10
^{55}Co	***17.5 h***	***10***	***^{159}Gd***	***18.6 h***	***100***
^{56}Co	79 d	0.1	^{160}Tb	72.3 d	1
^{57}Co	271 d	1	***^{165}Dy***	***2.33 h***	***1,000***
^{58}Co	71 d	1	***^{166}Dy***	***81.6 h***	***100***
^{58m}Co	***9.15 h***	***1,0000***	***^{166}Ho***	***26.8 h***	***100***
^{60}Co	5.27 y	0.1	***^{169}Er***	***9.3 d***	***1,000***
^{60m}Co	***10.47 m***	***1,000***	***^{171}Er***	***7.52 h***	***100***
^{61}Co	***1.65 h***	***100***	^{170}Tm	129 d	100
^{62m}Co	***13.9 m***	***10***	^{171}Tm	1.92 y	1,000
^{59}Ni	7.5×10^{4} y	100	***^{175}Yb***	***4.19 d***	***100***
^{63}Ni	96 y	100	***^{177}Lu***	***6.71 d***	***100***
^{65}Ni	***2.52 h***	***10***	^{181}Hf	42.2 d	1
^{64}Cu	***12.7 h***	***100***	^{182}Ta	115 d	0.1
^{65}Zn	243.9 d	0.1	^{181}W	121 d	10
^{69}Zn	***57 m***	***1,000***	^{185}W	75.1 d	1,000

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Radionuclide	Half-life	IAEA bulk levels (Bq/g)	Radionuclide	Half-life	IAEA bulk levels (Bq/g)
^{69m}Zn	***13.76 h***	***10***	***^{187}W***	***23.9 h***	***10***
^{72}Ga	***14.1 h***	***10***	***^{186}Re***	***90 h***	***1,000***
^{71}Ge	***11.8 d***	***1,0000***	***^{188}Re***	***17 h***	***100***
^{73}As	80.3 d	1,000	^{185}Os	94 d	1
^{74}As	17.8 d	10	^{191}Os	15.3 d	100
^{76}As	***26.3 h***	***10***	***^{191m}Os***	***13.03 h***	***1,000***
^{77}As	***38.8 h***	***1,000***	***^{193}Os***	***30 h***	***100***
^{75}Se	119.8 d	1	^{190}Ir	12.2 d	1
^{82}Br	***35.3 h***	***1***	^{192}Ir	74 d	1
^{86}Rb	18.7 d	100	***^{194}Ir***	***19.5 h***	***100***
^{85}Sr	64.8 d	1	***^{191}Pt***	***2.8 d***	***10***
^{85m}Sr	***69.5 m***	***100***	***^{193m}Pt***	***4.3d***	***1,000***
^{87m}Sr	***2.8 h***	***100***	***^{197}Pt***	***18.3 h***	***1,000***
^{89}Sr	50.5 d	1,000	***^{197m}Pt***	***94.4 m***	***100***
^{90}Sr	29.12 y	1	***^{198}Au***	***2.7 d***	***10***
^{91}Sr	***9.5 h***	***10***	***^{199}Au***	***3.14 d***	***100***
^{92}Sr	***2.71 h***	***10***	***^{197}Hg***	***64.1 h***	***100***
^{90}Y	***64 h***	***1,000***	***^{197m}Hg***	***23.8 h***	***100***
^{91}Y	58.5 d	100	^{203}Hg	46.4 d	10
^{91m}Y	***49.7 m***	***100***	***^{200}Tl***	***26.1 h***	***10***
^{92}Y	***3.54 m***	***100***	***^{201}Tl***	***26.1 h***	***10***
^{93}Y	***10.1 h***	***100***	^{202}Tl	12.2 d	10
^{93}Zr	1.53×10^{6} y	10	^{204}Tl	3.78 y	1
^{95}Zr	64 d	1	***^{203}Pb***	***52.05 h***	***10***
^{97}Zr	***16.9 h***	***10***	^{206}Bi	6.24 d	1
^{93m}Nb	13.6 y	10	^{207}Bi	38 y	0.1
^{94}Nb	2.03×10^{4} y	0.1	***^{203}Po***	***36.7 m***	***10***
^{95}Nb	***35 d***	***1***	***^{205}Po***	***1.80 h***	***10***
^{97}Nb	***72.1 m***	***10***	***^{207}Po***	***350 m***	***10***
^{98}Nb	***51.5 m***	***10***	***^{211}At***	***7.2 h***	***1,000***
^{90}Mo	***5.67 h***	***10***	^{225}Ra	14.8 d	10
^{93}Mo	3.5×10^{3} y	10	***^{227}Ra***	***42.2 m***	***100***
^{99}Mo	***66 h***	***10***	***^{226}Th***	***30.9 m***	***1,000***
^{101}Mo	***14.6 m***	***10***	^{229}Th	7,340 y	0.1
^{96}Tc	***4.3 d***	***1***	^{230}Pa	17.4 d	10
^{96m}Tc	***51.1 m***	***1,000***	^{233}Pa	27 d	10
^{97}Tc	2.6×10^{6} y	10	***^{230}U***	***20.8 d***	***10***
^{97m}Tc	90 d	100	***^{231}U***	***4.2 d***	***100***
^{99}Tc	2.13×10^{5} y	1	^{232}U	72 y	0.1
^{99m}Tc	***6.02 h***	***100***	^{233}U	1.58×10^{5} y	1
^{97}Ru	***2.9 d***	***10***	^{236}U	2.34×10^{7} y	10
^{103}Ru	39.2 d	10	***^{237}U***	***6.75 d***	***100***
^{105}Ru	***4.4 h***	***10***	***^{239}U***	***23.5 m***	***100***
^{106}Ru	368 d	0.1	***^{240}U***	***14.1 m***	***100***
^{103m}Rh	***56.1 m***	***1,0000***	^{237}Np	2.14×10^{6} y	1
^{105}Rh	***35.4 h***	***100***	***^{239}Np***	***2.36 d***	***100***
^{103}Pd	***16.96 d***	***1,000***	***^{240}Np***	***656 m***	***10***
^{109}Pd	***13.43 h***	***100***	***^{234}Pu***	***8.8 h***	***100***

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Radionuclide	Half-life	IAEA bulk levels (Bq/g)	Radionuclide	Half-life	IAEA bulk levels (Bq/g)
^{105}Ag	41 d	1	***^{235}Pu***	***25.3 m***	***100***
^{110m}Ag	250 d	0.1	^{236}Pu	2.85 y	1
^{111}Ag	***7.4 d***	***100***	^{237}Pu	45.3 d	100
^{109}Cd	453 d	1	^{238}Pu	87.7 y	0.1
^{115}Cd	***53.5 h***	***10***	^{239}Pu	24,065 y	0.1
^{115m}Cd	44.6 d	100	^{240}Pu	6,537 y	0.1
^{111}In	***2.83 d***	***10***	^{241}Pu	14.4 y	10
^{113m}In	***1.66 h***	***100***	^{242}Pu	3.76×10^5 y	0.1
^{114m}In	49.5 d	10	***^{243}Pu***	***4.96 h***	***1,000***
^{115m}In	5.1×10^5 y	100	^{244}Pu	8.26×10^7 y	0.1
^{113}Sn	115.1 d	1	^{241}Am	432 y	0.1
^{125}Sn	9.7 d	10	***^{242}Am***	***16.02 h***	***1,000***
^{122}Sb	***2.7 d***	***10***	***^{242m}Am***	***26 m***	***0.1***
^{124}Sb	60.2 d	1	^{243}Am	7,380 y	0.1
^{125}Sb	2.77 y	0.1	^{242}Cm	163 d	10
^{123m}Te	119.7 d	1	^{243}Cm	28.5 y	1
^{125m}Te	58 d	1,000	^{244}Cm	18.11 y	1
^{127}Te	***9.4 h***	***1,000***	^{245}Cm	8,500 y	0.1
^{127m}Te	109 d	10	^{246}Cm	4,730 y	0.1
^{129}Te	***69.6 m***	***100***	^{247}Cm	1.56×10^7 y	0.1
^{129m}Te	33.4 d	10	^{248}Cm	3.39×10^5 y	0.1
^{131}Te	***25 m***	***100***	^{249}Bk	320 d	100
^{131m}Te	***30 h***	***10***	***^{246}Cf***	***35.7 h***	***1,000***
^{132}Te	***78.2 h***	***1***	^{248}Cf	333.5 d	1
^{133}Te	***12.45 m***	***10***	^{249}Cf	350.6 y	0.1
^{133m}Te	***55.4 m***	***10***	^{250}Cf	13.1 y	1
^{134}Te	***41.8 m***	***10***	^{251}Cf	898 y	0.1
^{123}I	***13.2 h***	***100***	^{252}Cf	2.64 y	1
^{125}I	60.1 d	100	^{253}Cf	17.8 d	100
^{126}I	***13.02 d***	***10***	^{254}Cf	60.5 d	1
^{129}I	1.57×10^7 y	0.01	***^{253}Es***	***20.47 d***	***100***
^{130}I	***12.4 h***	***10***	^{254}Es	276 y	0.1
^{131}I	8.04 d	10	***^{254m}Es***	***39.3 h***	***10***
^{132}I	***2.30 h***	***10***	***^{254}Fm***	***3.24 h***	***1,0000***
^{133}I	***20.8 h***	***10***	***^{255}Fm***	***20.07 h***	***100***
^{134}I	***52.6 h***	***10***			

[a]The Bq/g values shown in ***bold italics*** were derived from the IAEA exemption levels in Schedule I of the BSS.

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A.3 Similarities between ANSI N13.12-1999 and IAEA RS-G-1.7

A comprehensive review and comparison of the 1999 version of N13.12 and the IAEA (2004, 2005) recommendations revealed several similarities:

- both used a 10 μSv/y (1.0 mrem/y) individual dose basis;
- both used scenarios and pathway analysis, a rounding convention, plus judgment, to establish order of magnitude activity concentrations;
- both provide activity concentrations in units of Bq/g;
- both allow for modifications; for example, bulk disposal in excess of about 1 metric ton may have more restrictive activity concentrations;
- both had many similar exclusions:
  - radioactive materials in foodstuffs (drinking water, animal feed, and other material intended for use in food or animal feed);
  - radioactive discharges of liquid or airborne effluents; and
  - radioactive residues in the environment (use as decommissioning or intervention criteria); and
- both used the sum of fractions approach for mixtures.

A.4 Differences Between ANSI N13.12-1999 and IAEA RS-G-1.7

The comprehensive review and comparison revealed a number of differences between the 1999 version of this standard and the IAEA (2004, 2005) recommendations:

- IAEA (2004) included exclusion, exemption, and clearance of bulk material with volume radioactivity, whereas the 1999 version of this standard included clearance levels for both volume and surface radioactivity assuming a surface to mass ratio of 1:1;
- IAEA (2004) used the "lowest values obtained from scenarios" for activity concentrations, whereas the 1999 version of this standard was based on judgment within the range of scenario results (typically, near the more restrictive end of the range);
- IAEA (2004) used log averaging to select the screening levels for clearance, whereas the 1999 version of this standard used arithmetic averaging to select the screening levels for clearance;
- IAEA (2004) used scenarios that included foodstuff and drinking water pathways, as appropriate, without providing separate guidance for food and drinking water, whereas the 1999 version of this standard generally avoided agricultural or environmental transport scenarios;
- IAEA (2004) reached the general conclusion that individual dose will almost always be more limiting than collective total effective dose commitments, whereas the 1999 version of this standard did not discuss potential collective doses beyond a general ALARA discussion;
- IAEA (2004) considered direct and continual handling of material without allowance for radioactive decay (similar to exemption), whereas the 1999 version of this standard considered these types of scenarios to be for radiation workers, not members of the public;
- IAEA (2004) contains values for radionuclides of natural origin based on worldwide distribution of activity concentrations, not a dose value of 10 μSv/y, whereas the 1999 version of this standard contains these radionuclides based on scenario results and published scenario evaluations because they may be part of licensed or regulated activities;
- IAEA (2004) used ICRP 60 (ICRP 1991) dosimetry methods, whereas the 1999 version of this standard used ICRP 26 (1977) dosimetry methods;
- for many short-lived radionuclides, the exemption levels of Schedule I of the BSS (IAEA 1996) were chosen by the IAEA as the activity concentrations because derived clearance levels were higher than the exemption levels, whereas the 1999 version of this standard does not contain short-lived radionuclides because they were determined not to be relevant to the general practice of clearance (i.e.,

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sufficient delays before release to allow for radioactive decay);

- the 1999 version of this standard was based on three considerations: individual dose (10 μSv/y), radiation detection, and radioactivity control (i.e., are clearance levels so high as to conflict with sound radiation protection practices?), whereas the IAEA (2004) is silent on radiation detection and radioactivity control issues; and
- there were several different exclusions:
  - the 1999 version of this standard excluded naturally occurring radioactive materials;
  - the 1999 version of this standard excluded radioactive materials on or in persons;
  - the 1999 version of this standard excluded land or soil intended for agricultural purposes;
  - the 1999 version of this standard excluded clearance issues related to national defense or security;
  - IAEA (2004) excluded radon in air;
  - IAEA (2004) excluded potassium-40 in the body (also excluded by the BSS); and
  - IAEA (2004) excluded material in transport in accordance with IAEA transportation regulations.

A.5 Comparison

The IAEA Safety Standards Series (IAEA 2004) provides a listing of activity concentrations for 257 radionuclides of artificial origin in bulk. Upon inspection, the clearance levels for about 138 short-lived radionuclides, as shown in the ***bold italicized*** entries in Table A.1, were set by the IAEA to be the exemption levels of Schedule I of the Basic Safety Standards (IAEA 1996). This means that it is difficult to perform a direct comparison against the N13.12-1999 clearance levels. To make a comparison possible, only those radionuclides with clearance levels derived from dose calculations, and not exemption levels from Schedule I of the BSS, can be considered. This results in a list of 119 radionuclides common to both references, as shown in the unitalicized entries in Table A.1.

The activity concentrations provided by the IAEA (2004) were compared against the clearance levels provided in the 1999 version of this standard. Note that the values in the 1999 version of this standard for numerous radionuclides were derived based on a comparison of the screening factors, by exposure pathway listed in Tables B.1 through D.1 of NCRP Report No. 123I (NCRP 1996). This was done in a conservative manner to ensure that the appropriate clearance level would not be overestimated. A summary of the radionuclides considered in this comparison, the IAEA (2004) activity concentrations for clearance and those from N13.12-1999, and the ratio of the two levels (i.e., IAEA/N13.12-1999) are provided in Table A.2. A review of the activity concentrations provided in Table A.2 shows exact agreement for about 50% of the 119 radionuclides selected.

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Table A.2. Radionuclides and activity concentrations considered in the initial comparison

Radionuclide[a]	IAEA[b] (Bq/g)	N13.12-1999[c] (Bq/g)	Ratio: IAEA/ N13.12-1999	Radionuclide[a]	IAEA[b] (Bq/g)	N13.12-1999[c] (Bq/g)	Ratio: IAEA/ N13.12-1999
^{3}H	100	100	1	^{139}Ce	1	10	0.1
^{7}Be	10	10	1	^{141}Ce	100	100	1
^{14}C	1	100	0.01	^{144}Ce	10	10	1
^{22}Na	0.1	1	0.1	^{147}Pm	1,000	100	0.1
^{35}S	100	100	1	^{147}Nd	100	100	1
^{36}Cl	1	1	1	^{151}Sm	1,000	100[d]	10
^{45}Ca	100	100	1	^{152}Eu	0.1	1	0.1
^{46}Sc	0.1	1	0.1	^{154}Eu	0.1	1	0.1
^{51}Cr	100	100	1	^{155}Eu	1	1	1
^{53}Mn	100	100	1	^{153}Gd	10	10	1
^{54}Mn	0.1	1	0.1	^{160}Tb	1	10	0.1
^{55}Fe	1,000	100[d]	10	^{170}Tm	100	100	1
^{59}Fe	1	10	0.1	^{171}Tm	1,000	100[d]	10
^{56}Co	0.1	1	0.1	^{181}Hf	1	10	0.1
^{57}Co	1	1	1	^{182}Ta	0.1	1	0.1
^{58}Co	1	1	1	^{181}W	10	10	1
^{60}Co	0.1	1	0.1	^{185}W	1,000	100[d]	10
^{59}Ni	100	100	1	^{185}Os	1	10	0.1
^{63}Ni	100	100	1	^{191}Os	100	100	1
^{65}Zn	0.1	1	0.1	^{190}Ir	1	1	1
^{73}As	1,000	100[d]	10	^{192}Ir	1	1	1
^{74}As	10	10	1	^{203}Hg	10	10	1
^{75}Se	1	1	1	^{202}Tl	10	10	1
^{86}Rb	100	100	1	^{204}Tl	1	1	1
^{85}Sr	1	1	1	^{207}Bi	0.1	1	0.1
^{89}Sr	1,000	100[d]	10	^{225}Ra	10	10	1
^{90}Sr	1	1	1	^{229}Th	0.1	0.1	1
^{91}Y	100	100	1	^{230}Pa	10	10	1
^{93}Zr	10	100	0.1	^{233}Pa	10	10	1
^{95}Zr	1	1	1	^{232}U	0.1	0.1	1
^{93m}Nb	10	10	1	^{233}U	1	1	1
^{94}Nb	0.1	1	0.1	^{236}U	10	10	1
^{93}Mo	10	10	1	^{237}Np	1	0.1	10
^{97}Tc	10	10	1	^{236}Pu	1	1	1
^{97m}Tc	100	100	1	^{237}Pu	100	100	1
^{99}Tc	1	100	0.01	^{238}Pu	0.1	0.1	1
^{103}Ru	1	0.1	1	^{239}Pu	0.1	0.1	1
^{106}Ru	0.1	1	0.1	^{240}Pu	0.1	0.1	1
^{105}Ag	1	10	0.1	^{241}Pu	10	10	1
^{110m}Ag	0.1	1	0.1	^{242}Pu	0.1	0.1	1
^{109}Cd	1	1	1	^{244}Pu	0.1	0.1	1
^{115m}Cd	100	100	1	^{241}Am	0.1	0.1	1
^{114m}In	10	10	1	^{243}Am	0.1	0.1	1
^{115m}In	100	100	1	^{242}Cm	10	10	1
^{113}Sn	1	10	0.1	^{243}Cm	1	1	1
^{125}Sn	10	10	1	^{244}Cm	1	0.1	10

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Radionuclide[a]	IAEA[b] (Bq/g)	N13.12-1999[c] (Bq/g)	Ratio: IAEA/ N13.12-1999	Radionuclide[a]	IAEA[b] (Bq/g)	N13.12-1999[c] (Bq/g)	Ratio: IAEA/ N13.12-1999
^{124}Sb	1	1	1	*^{245}Cm*	0.1	0.1	1
^{125}Sb	0.1	1	0.1	*^{246}Cm*	0.1	0.1	1
^{123m}Te	1	10	0.1	*^{247}Cm*	0.1	0.1	1
^{125m}Te	1,000	100	10	*^{248}Cm*	0.1	0.1	1
^{127m}Te	10	10	1	*^{249}Bk*	100	100	1
^{129m}Te	10	10	1	*^{248}Cf*	1	1	1
^{125}I	100	100	1	*^{249}Cf*	0.1	0.1	1
^{129}I	0.01	0.1	0.1	*^{250}Cf*	1	1	1
^{131}I	10	10	1	*^{251}Cf*	0.1	0.1	1
^{134}Cs	0.1	1	0.1	*^{252}Cf*	1	1	1
^{135}Cs	100	100	1	*^{253}Cf*	100	100	1
^{137}Cs	0.1	1	0.1	*^{254}Cf*	1	1	1
^{131}Ba	10	10	1	*^{254}Es*	0.1	0.1	1
^{140}Ba	1	10	0.1				

[a]Where radionuclides in italics indicate N13.12-1999 values were derived using NCRP 123I (1996).

[b]Where IAEA indicates screening levels from IAEA (2004).

[c]Where ANSI/HPS screening levels are from N13.12-1999.

[d]For the radionuclide shown, the N13.12-1999 screening level was artificially included in the 100 Bq/g group based on radioactivity control concerns.

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The ratio developed by dividing the IAEA activity concentration by the activity concentration from the 1999 version of this standard for each radionuclide provides the following observations:

- For numerous radionuclides for which external exposure is the primary exposure pathway, the IAEA (2004) activity concentrations are one order of magnitude lower than the corresponding activity concentrations from N13.12-1999. These radionuclides include ^{22}Na, ^{46}Sc, ^{54}Mn, ^{59}Fe, ^{56}Co, ^{60}Co, ^{65}Zn, ^{93}Zr, ^{94}Nb, ^{106}Ru, ^{105}Ag, ^{113}Sn, ^{125}Sb, ^{123m}Te, ^{134}Cs, ^{137}Cs, ^{139}Ce, ^{152}Eu, ^{154}Eu, ^{160}Tb, ^{181}Hf, ^{182}Ta, ^{185}Os, and ^{207}Bi.
- For the majority of these radionuclides, evaluation of the scenario results in IAEA Safety Report Series No. 44 (IAEA 2005) indicated that that there was a wide distribution of results across the scenarios considered, often spanning five or six orders of magnitude. Thus, when IAEA selected the most restrictive value, it was often lower than the value selected by the 1999 version of this standard.
- For a few radionuclides, the IAEA (2004) included "environmental mobility" by considering ingestion of food and drinking water. The resulting IAEA activity concentrations are about a factor of 100 less than derived for the 1999 version of this standard, where environmental mobility was not a strong consideration. These radionuclides include ^{14}C, ^{99}Tc, and ^{129}I.
- The 1999 version of this standard limited the application to four screening level groups, ranging from 0.1 to 100 Bq/g, and eliminated a proposed fifth group (1,000 Bq/g) based on radioactivity control concerns (i.e., merged the proposed 1,000 Bq/g group radionuclides into the 100 Bq/g group), whereas the IAEA included six screening level groups, ranging from 0.01 to 10,000 Bq/g. The IAEA 10,000 Bq/g group only applied to a few radionuclides with extremely short half-lives (on the order of minutes to hours), which were eliminated from the initial comparison because they were based on the IAEA exemption values. A few of the IAEA activity concentrations were listed as 1,000 Bq/g, with the corresponding N13.12-1999 values listed as 100 Bq/g based on radioactivity control.

Each of these observations was determined by reviewing the scenarios, data, assumptions, and methods used by each study in determining the clearance level for each radionuclide.

A.6 Evaluation

From this comparison, the following conclusions were reached:

- Scenario Approach. N13.12-1999 relied largely on published studies, which included studies by DOE and EPA on the effects of metal recycle and concentration of radionuclides in metal, slag, and bag house dust. However, this means that there was no control over scenario definition and the results of similar scenarios (i.e., building re-use) were used to develop ranges, without consideration of the basic conceptual or data differences among the scenarios. The IAEA (2005) developed a set of internally consistent scenarios with similar data assignments to develop clearance results that included doses to workers at metal foundries.
- Number of Radionuclides Considered. Because the 1999 version of this standard relied on published studies, there were a limited number of radionuclides that could be directly compared. It was not possible to develop a comprehensive list as was done in the IAEA (2004) study. Instead, the 1999 version of this standard outlined a method for determining clearance screening levels for radionuclides not shown based on a comparison of the screening factors, by exposure pathway listed in Tables B.1 through D.1 of NCRP Report No. 123I (NCRP 1996).
- Selection of Clearance Screening Levels. Because of the conceptual and data differences among the studies considered in the 1999 version of this standard, the ranges were often large and "outliers" were sometimes dropped to narrow the range. However, this was not a consistent practice and several radionuclides were found to have less

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restrictive screening levels than the corresponding IAEA (2004) clearance screening levels. Further, because the studies considered by the 1999 version of this standard did not include all of the radionuclides, some of the ranges consisted of only a few data points.

- Radioactivity Control. The 1999 version of this standard initially had five proposed groups, with the highest group at 1,000 Bq/g. The 1,000 Bq/g group contained only a few radionuclides. It was agreed by the writing group that this radionuclide group was not needed based on surface radioactivity control concerns, and those radionuclides were included in the 100 Bq/g group. As stated previously, the IAEA included six groups, ranging from 0.01 to 10,000 Bq/g, where the highest group only applied to a few radionuclides with extremely short half-lives, which were eliminated from this comparison because they were based on the IAEA exemption levels.

A.7 Selection of Revised Volume Clearance Screening Levels

Figure A.1 shows the volume clearance screening level ranges, based on the range of scenario dose results, for the radionuclides in common to both IAEA (2004) (shown with a dashed line) and N13.12-1999 (shown with a solid line). The screening levels selected for both publications are shown with vertical solid lines. This figure does not include radionuclides with screening levels derived using NCRP 123I (1996), as recommended by the 1999 version of this standard. As shown in Figure A.1, for the majority of 39 radionuclides in common to both studies, the N13.12-1999 and IAEA (2004) ranges showed significant overlap. In fact, in almost all cases the IAEA (2004) volume clearance screening levels would be within the range identified by N13.12-1999. Based on this graphical comparison of the ranges of volume clearance screening levels from both studies, it is concluded that the IAEA (2004) clearance levels for volume activity sources can be adopted for all radionuclides, except for ^{129}I and naturally occurring radionuclides in this revised version of N13.12. Additional considerations were included:

- General Selection of Clearance Levels. As shown in Figure A.1, the IAEA (2004) selected clearance levels that were near the lowest end of the range of scenario results generated for each radionuclide. The scenarios were intended to be internally consistent with similar data assignments, which were conservative in nature. However, it could be argued that the selection of clearance levels derived from a distribution of scenario results should be more centrally located, given the goal of being protective of 10 μSv per year. The writing group concluded that the selected clearance levels were conservative assignments as compared to the distribution of scenario results for most radionuclides.

- ^{129}I Considerations. Special difficulties for ^{129}I have been recognized. The IAEA (2004) volume clearance screening levels derived from the typical scenarios resulted in a range from 3.3 to 38,000 Bq/g per 10 μSv/y. However, the final IAEA (2004) volume clearance screening level was set at 0.01 Bq/g per 10 μSv/y. This was done in recognition of potential ground-water (i.e., drinking water) protection issues involving the clearance of rather large quantities of material to landfills or soil. The writing group considered the situation, and concluded the volume clearance screening level for ^{129}I could be set at 0.1 Bq/g per 10 μSv/y if a limitation on disposal to landfills or soil is included.

- Naturally Occurring Radionuclides. Finally, the IAEA (2004) did not include naturally occurring radionuclides in the derivation of volume activity concentration levels at 10 μSv/y, but rather based them on worldwide distribution of activity concentrations for naturally occurring radioactive materials. However, because these radionuclides could also be included in licensed or controlled materials, the writing group concluded that they deserved inclusion in this standard. In reviewing the modeling basis, and in reviewing a companion ANSI/HPS standard (ANSI/HPS N13.53-2009), it was decided to include the clearance screening levels previously developed

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for these radionuclides in the 1999 version of standard. The writing group included individual naturally occurring radionuclides based on N13.12-1999 (in Group 1) and uranium radionuclides that have been separated from their decay-chain progeny (in Group 2).

- Surface Radioactivity Considerations. Finally, clearance of items or materials with surface radioactivity can cover a wide range of conditions that represent reuse, recycle, and disposal. N13.12-1999 considered two scenarios, evaluated by several different methods. The first scenario considered release of a room with surface area sources of radioactivity. The scenario considered that the room is occupied after clearance, which results in exposures of an individual during the first year after clearance. The second scenario considered the doses resulting from the clearance for reuse of tools and equipment with residual surface radioactivity. Both scenarios considered external, inhalation, and ingestion exposure pathways. The surface radioactivity was assumed to be in a 1-to-1 correspondence with mass. The scenarios conservatively considered that all of the surface radioactivity was totally fixed (important for analysis of the external exposure pathway), or totally removable (important for the inhalation or ingestion pathways) to maximize the dose potential for each radionuclide. The results of these surface radioactivity scenarios are included in the ranges of scenario dose results in Figure A.1 for N13.12-1999. The results showed a relatively good agreement between scenarios that considered volume or surface radioactivity. Thus, the writing group concluded that the volume radioactivity scenarios considered by the IAEA would bound the results for surface radioactivity scenarios for a mass-to-surface ratio of 1.0.

A.8 Surface Radioactivity Clearance Levels

Unlike the 1999 version of this standard, the IAEA (2004) did not provide surface radioactivity clearance levels stating that this would be the subject of future guidance. The 1999 version of this standard assumed that the average relationship between surface and volume radioactivity would be a 1:1 ratio; that is, average items or materials would have about the same surface area in square centimeters as mass in grams. This assumption was consistent with earlier recommendations from the IAEA (1988, 1992, 1996).

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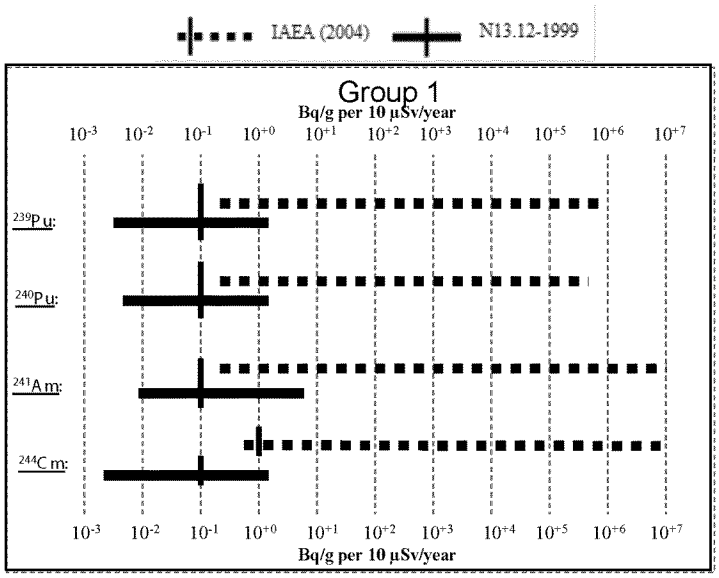
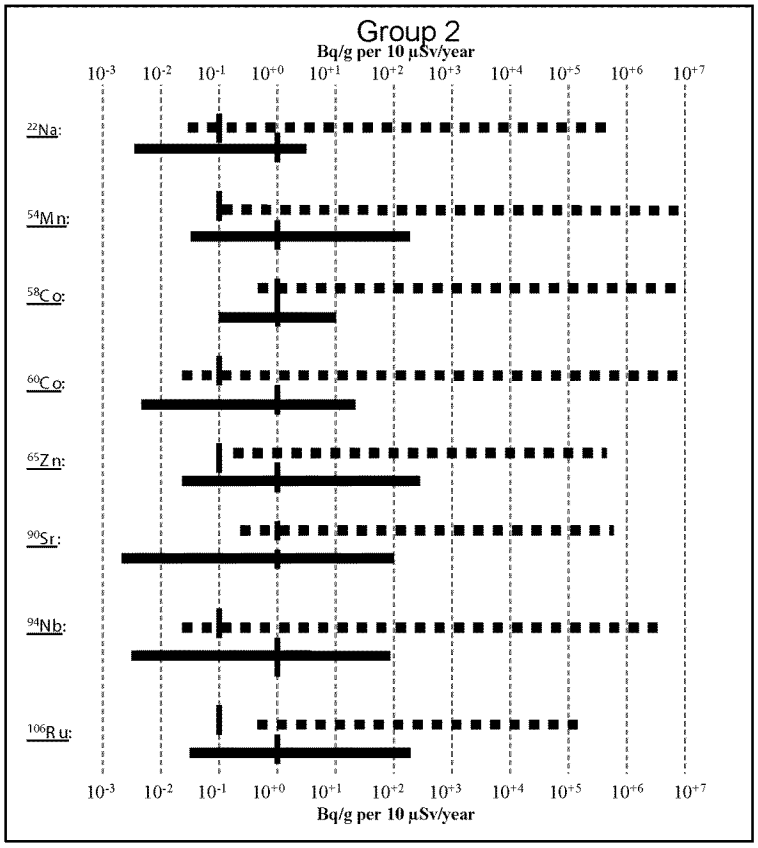
Figure A.1. Comparison of clearance level ranges between IAEA (2004) and N13.12-1999, in units of Bq/g per 10 µSv/y.





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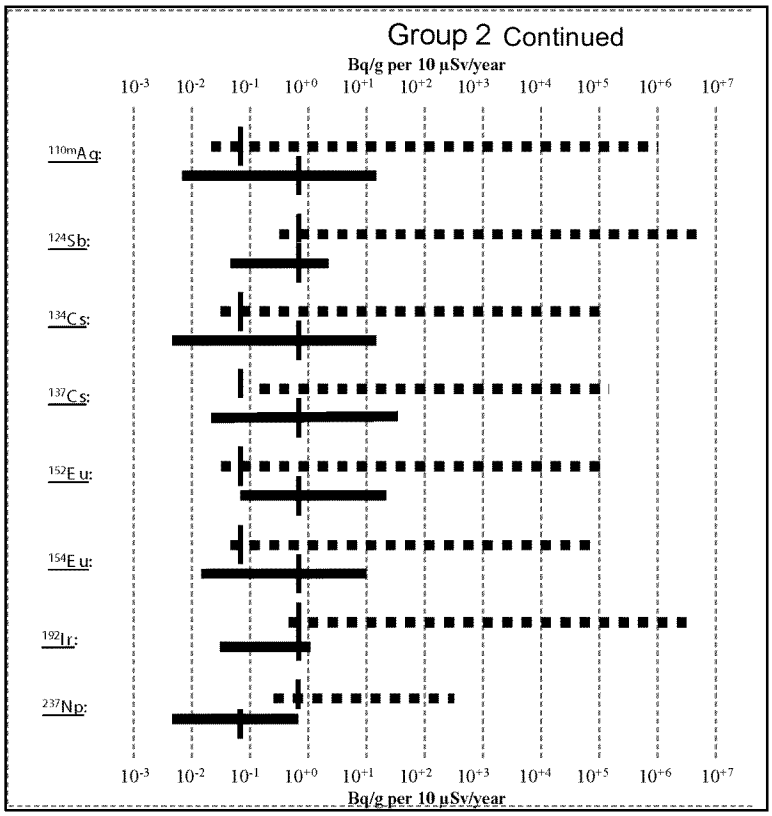

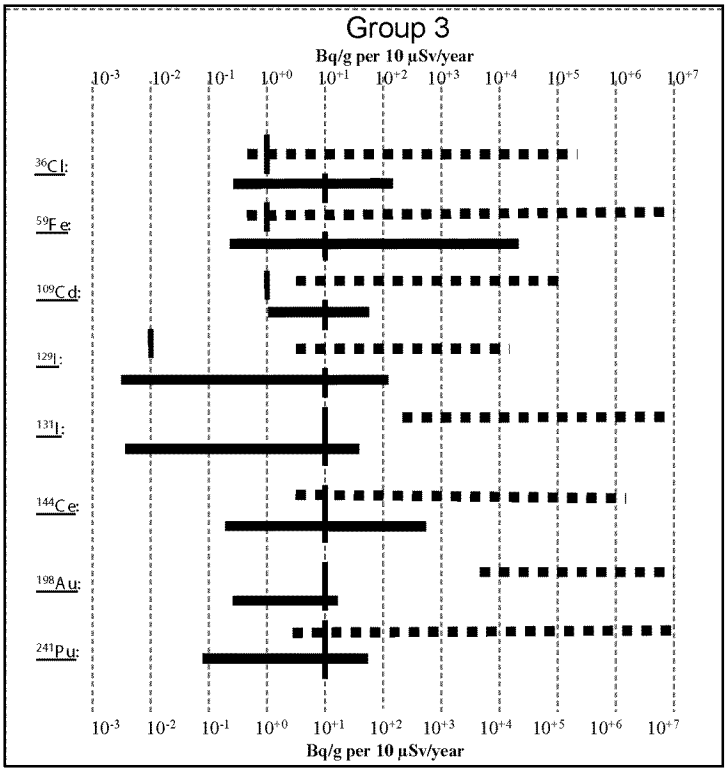

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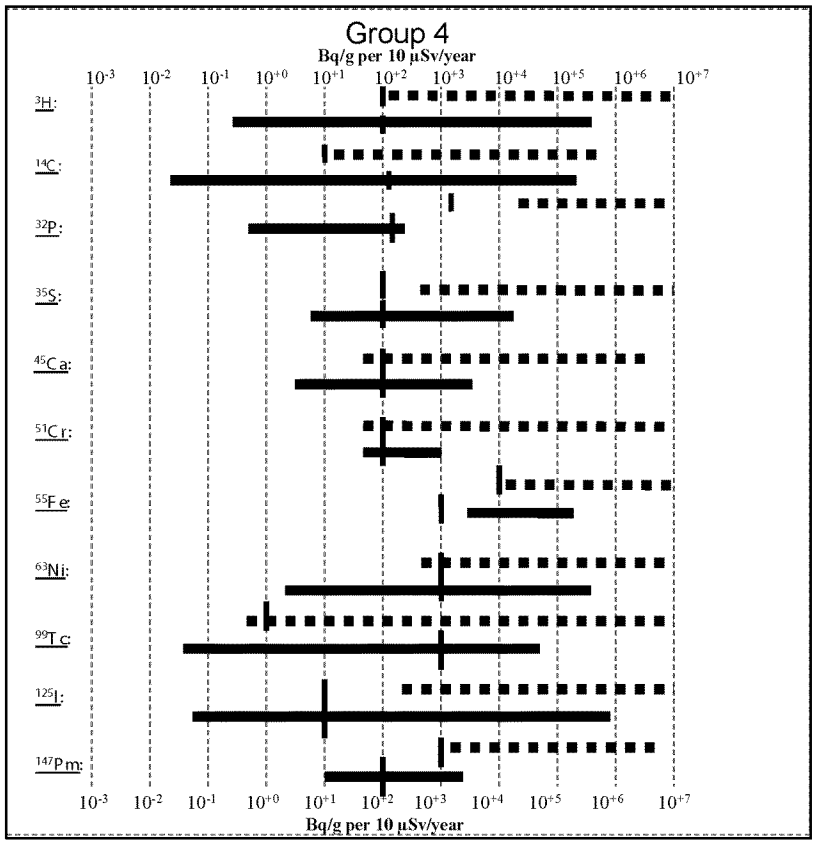

Although most survey programs are designed around detecting surface radioactivity, there are noted difficulties that must be addressed. The nature of surface radioactivity will vary from item to item and material to material. For example, surface radioactivity on concrete or wood surfaces may actually be a distributed source with a shallow depth compared to that on steel or plastic surfaces. The ability to remove surface radioactivity from an item or material during handling may be an important consideration when attempting to establish radiation exposure scenarios and parameters. For example, the inhalation and ingestion pathways rely on the presence of removable radioactivity to be resuspended into air or transferred to hands (then to the mouth). Perhaps most importantly, the relationship between surface and volume radioactivity can be difficult to describe for many situations.

Studies conducted by the USNRC attempted to evaluate the relationship between surface and volume radioactivity as related to clearance during decommissioning. In their report "Radiological Assessments for Clearance of Materials from Nuclear Facilities" (USNRC 2003), the USNRC provided literature surveys and data for recycling steel, copper, aluminum, and concrete. For each type of material, they provided ranges of mass-to-surface ratios (i.e., the ratio of the mass in grams to the surface area in square centimeters) considering items and materials encountered during decommissioning of commercial nuclear power reactors. In several cases, mean values for specific metals were obtained using 10,000 Monte Carlo simulations over a variety of different item or material sizes and geometries. A summary of the ranges of mass-to-surface ratios provided by the USNRC (2003) is provided in Table A.3.

In addition to items or materials from nuclear power reactor decommissioning, other typical items or materials *may* normally be cleared from radiological controls at nuclear facilities. A summary of the mass-to-surface ratios for various items or materials that could be cleared is provided in Table A.4.

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Although there is a high degree of variability as seen in the information in Tables A.3 and A.4, the following general conclusions are made:

Table A.3. Summary of mass-to-surface information from the USNRC (2003)[a]

Material/item	Mass-to-surface ratio (g/cm^2)	Discussion
Ferrous metals		
Reinforcing steel (rebar)		In general, as the cross section of rebar increases, the mass-to-surface ratio increases as shown by the summary data in the previous column.
Metric #10	1.87	
Metric #22	4.36	
Metric #43	8.44	
Pipe hangers		Pipe hangers are commonly found in nuclear power reactors and vary over a wide range of sizes depending on the application. In general, these are massive items with relatively little surface area.
37 kg	1.81	
168 kg	9.96	
1,177 kg	19.90	
Mean values	4.53– 5.34	PWR and BWR mean values for mass-to-surface ratios across a number of different items that could be encountered.
Concrete		
Reactor building		Reactor building concrete typically provides shielding and structure, and in most areas is quite thick compared to other industrial situations.
Structural concrete	1,846	
Cylindrical wall	351	
Dome	264	
Interior	297	
Turbine building		Turbine building concrete includes both structural and support, and other areas requiring concrete fill.
Concrete fill	37	
Structural concrete	292	
Superstructure	81	
Reactor auxiliaries		
Structural concrete	245	Interior walls and floors.
Fuel storage		
Structural concrete	424	Interior walls, floors, and shielding and support areas.
Superstructure	142	
Total average	280	Reflects high-density, small surface applications at nuclear power reactor.
Aluminum sheet		
Sheet		
0.160 cm	0.432	
0.203 cm	0.549	As a lower-density metal, the typical surface-to-mass ratios are a fraction of those for steel.
0.245 cm	0.686	
0.406 cm	1.097	
0.635 cm	1.715	
Mean value	0.90	Monte Carlo-generated mean across several thicknesses.
Copper		
Mean value	0.52	Monte Carlo-generated mean across several items including copper bus bars and wiring.

[a]For non-containerized materials only.

- The volume screening levels for clearance can be converted into surface screening levels for clearance by multiplying by the mass-to-surface ratio. This means that the lower the mass-to-surface ratio, the lower the surface screening levels for clearance, and the higher the mass-to-surface ratio, the higher the surface screening levels for clearance.

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Table A.4. Summary of mass-to-surface ratios for typical items

Material/item	Mass-to-surface ratio (g/cm^2)	Discussion
Notebook computer	0.45–0.66	Considering all surfaces and considering two alternative power supply sizes
Paint can (full)	3.7	Exterior surfaces only
Paint can (empty)	0.2	Exterior surfaces only
Plywood sheet	0.21–0.61	Standard 118 by 235 cm (4 foot by 8 foot) sheets from 0.63 cm (one-quarter inch) to 1.9 cm (three-quarters inch) thick
Hammer, carpenter's	1.25	Standard 0.45 kg (1 pound) hammer, and handle
Hammer, sledge	6.76	Standard 4.5 kg (10 pound) sledge hammer and handle
Hand truck (large wood)	0.5	Includes four wheels
Lead acid battery	12.2	Twelve-volt medium duty

- For most materials encountered during operations and decommissioning, and for many of the typical bulk items or materials considered, the mass-to-surface ratio is greater than 1.0. This means that defaulting to a mass-to-surface ratio of 1.0 will be conservative for most situations.

From this information, the writing group concluded that continued use of a mass-to-surface ratio of 1:1 for determining surface screening levels for clearance is usually both conservative and warranted, except in cases with a thin sheet-like geometry, such as sheet metal. In the exceptional case of sheet-like geometries, a mass-to-surface ratio of 0.2 is usually conservative and acceptable. However, the derived surface radioactivity screening levels for a specific practice should be modified, if information exists to justify the modification, to either higher or lower values.

A.9 Screening Levels for Clearance for Additional Radionuclides

Screening levels for clearance are shown in Table 1 for over 130 individual radionuclides and selected decay chains. Footnote b to Table 1 indicates that the NCRP screening factors, by exposure scenario, listed in Tables B.1, C.1, and D.1 of NCRP Report No. 123I (NCRP 1996), should be used with comparison to the radionuclides shown in Table 1 of this standard to determine the proper group for radionuclides not shown in Table 1. This section provides an example of how to make this determination. In general, the process used is to evaluate the radiations emitted by the additional radionuclide, select comparison radionuclides from the relevant screening levels for clearance groups from Table 1 (according to emitted radiations), and then obtain the NCRP screening factors from Report No. 123I (1996), by exposure scenario, for the additional radionuclide and the comparison radionuclides selected from Table 1. By inspection of these factors, identify those for the comparison radionuclides that most closely match those for the additional radionuclide, for each scenario. From this comparison, assign the clearance screening level group that most closely matches the additional radionuclide to the comparison radionuclides, for each scenario. If there is disagreement among the most comparable factors across the NCRP scenarios, assign the clearance screening level group to the lowest (i.e., most restrictive) group shown.

For this example, four radionuclides not included in Table 1 of this standard are considered: ^{195}Au, ^{247}Bk, ^{182}Ta, and ^{231}Pa. A summary of the radioactive half-lives, primary modes of radioactive decay, and clearance screening level from the Table 1 radionuclide groups that have modes of radioactive decay similar to those of the additional radionuclides, are summarized in

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Table A.5. Eight comparison radionuclides from Table 1, or about twice as many comparison radionuclides as additional radionuclides, are selected, including ^{210}Po, ^{241}Am, ^{60}Co, ^{137}Cs, ^{103}Ru, ^{144}Ce, ^{45}Ca, and ^{141}Ce. The radioactive half-lives, primary modes of decay, assigned screening level group, and volume screening levels (in Bq/g) for these comparison radionuclides are shown in Table A.6. For this example, the comparison radionuclides cover the first four radionuclide groups from Table 1; however, if the additional radionuclides could include low-dose beta emitters, then radionuclides from Group 5 should be included in the list of comparison radionuclides as well.

Table A.5. Radionuclides not included in Table 1 of this standard

Radionuclide	Half-life	Primary mode of decay	Similar mode of decay from Table 1 groups
^{195}Au	183 d	β, γ	2, 3, 4
^{247}Bk	1,380 y	α	1, 2
^{182}Ta	115.5 d	β, γ	2, 3, 4
^{231}Pa	3.3×10^4 y	α	1, 2

Table A.6. Comparison radionuclides from Table 1 of this standard

Radionuclide	Half-life	Primary mode of decay	N13.12 screening level group	Volume screening levels (Bq/g)
^{210}Po	138.4 d	α, γ	1	0.1
^{241}Am	432 y	α, γ	1	0.1
^{60}Co	5.3 y	γ	2	1
^{137}Cs	30 y	β, γ	2	1
^{103}Ru	39 d	β, γ	3	10
^{144}Ce	284 d	β, γ	3	10
^{45}Ca	163 d	β, γ	4	100
^{141}Ce	32.5 d	β, γ	4	100

Next, the screening factors, by scenario, from Tables B.1 through D.1 of NCRP Report No. 123I (NCRP 1996) are obtained for the additional and comparison radionuclides, as shown in Table A.7. The screening factors are for scenarios that consider releases to the atmosphere, fresh surface water, and ground.

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Table A.7. Screening factors, by scenario, from the NCRP Report No. 123I

Radionuclide	NCRP No. 123I screening values		
	Atmospheric ($Sv \cdot Bq^{-1} \cdot m^{-3}$ in air)[a]	Fresh surface water ($Sv \cdot Bq^{-1} \cdot m^{-3}$ in surface water)[b]	Ground (Sv/Bq)[c]
^{210}Po	1.6×10^{-1}	8.2×10^{-8}	3.3×10^{-14}
^{241}Am	1.0	2.0×10^{-6}	8.4×10^{-12}
^{60}Co	1.7×10^{-1}	6.1×10^{-7}	6.6×10^{-12}
^{137}Cs	2.2×10^{-1}	1.1×10^{-6}	1.4×10^{-11}
^{103}Ru	1.1×10^{-3}	2.1×10^{-9}	1.3×10^{-20}
^{144}Ce	5.2×10^{-3}	3.7×10^{-8}	3.6×10^{-15}
^{45}Ca	1.0×10^{-3}	1.7×10^{-8}	4.2×10^{-15}
^{141}Ce	5.0×10^{-4}	5.6×10^{-9}	9.6×10^{-23}
^{195}Au	9.1×10^{-4}	3.3×10^{-9}	8.2×10^{-16}
^{247}Bk	1.3	3.0×10^{-6}	1.1×10^{-9}
^{182}Ta	6.0×10^{-4}	1.9×10^{-7}	1.6×10^{-14}
^{231}Pa	3.0	5.1×10^{-6}	1.7×10^{-10}

[a]From Table B.1 of NCRP No. 123I (NCRP 1996).
[b]From Table C.1 of NCRP No. 123I (NCRP 1996).
[c]From Table D.1 of NCRP No. 123I (NCRP 1996).

By inspection, the screening factor values from the list of comparison radionuclides from Table 1 of this standard that most closely match those for the additional radionuclides are determined, and the clearance screening level group for each radionuclide is assigned, as summarized in Table A.8.

Table A.8. Clearance screening level group determination for the additional radionuclides considered

Additional radionuclide	Most closely matching NCRP screening levels and (clearance screening level group)			Assigned clearance screening level group
	NCRP atmospheric	NCRP fresh surface water	NCRP ground	
^{195}Au	^{141}Ce (Group 4)	^{103}Ru (Group 3)	^{144}Ce (Group 3)	3
^{247}Bk	^{241}Am (Group 1)	^{241}Am (Group 1)	^{137}Cs (Group 2)	1
^{182}Ta	^{141}Ce (Group 4)	^{60}Co (Group 2)	^{60}Co (Group 2)	2
^{231}Pa	^{241}Am (Group 1)	^{241}Am (Group 1)	^{137}Cs (Group 2)	1

As shown in Table A.8, for all of the additional radionuclides considered, the most closely matching comparison radionuclide from Table 1 of this standard varies by the type of NCRP screening value considered, and in all cases the comparison radionuclides are from different clearance screening level groups. The assigned clearance screening level group was conservatively determined to be the lowest (i.e., most restrictive) group shown.

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Annex B

(Informative)

As Low As Reasonably Achievable Considerations

The ICRP has provided recommendations for radiation protection that have been adopted by many of its member countries, with the notable exception of the United States. The ICRP has recommended a system for radiation protection that consists of three parts (1991):

- justification of a practice: no practice shall be adopted unless it introduces a net positive benefit;
- optimization of radiation protection: all exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account (the ALARA principle); and
- limitation of individual risk: through regulation, individuals shall not exceed the limits established for protection of the public.

The IAEA, under the IAEA Statute approved by the United Nations in 1956, is empowered to recommend the radiation safety standards to be followed by personnel in all IAEA programs. The IAEA's standards largely reflect the recommendations of the ICRP, including their three-part system for radiation protection. In 1988, the IAEA published the final version of Safety Series No. 89, "Principles for the Exemption of Radiation Sources and Practices from Regulatory Control" (IAEA 1988). In this document, the IAEA recognized that there was "no internationally unified policy for excluding or exempting (i.e., clearing) sources from regulatory control." The IAEA indicates that clearance should be conducted within the context of its basic three elements for radiation protection principles. Further, the IAEA identified two basic criteria for determining whether a practice is a candidate for clearance from continued regulation: (1) the individual risks must be sufficiently low as not to warrant regulatory concern and (2) radiation protection, including the cost of regulatory control, must be optimized. The first criterion required setting a level of trivial dose and the second criterion required the application of an optimization, or ALARA analysis.

In Safety Series 89 (IAEA 1988), the IAEA concluded that "an individual radiation dose, regardless of its origin, is likely to be regarded as trivial if it is on the order of some tens of microsieverts per year." Because an individual could be exposed to radiation doses from several cleared sources or practices, the IAEA concluded that doses on the order of 10 μSv/y (1.0 mrem/y) per practice would be reasonable. The IAEA also applied the guidance of about 10 μSv/y (1.0 mrem/y) in Safety Series No. 111-P-1.1 (IAEA 1992) for the recycle and reuse of materials from nuclear facilities, and in recent documents on application of the concepts of exclusion, exemption, and clearance (IAEA 2004, 2005). The IAEA guidance has been carefully considered and applied in this standard (Annex A).

Supporting its position, the IAEA provided an overview of risk-based considerations. They commented that few people would commit their own resources to reduce an annual risk of death below 10^{-5}, and that even fewer would take action at an annual level of 10^{-6} (Royal Society of London 1983). They also noted that several authors who proposed values of trivial individual dose have set the level of annual risk of death that is seen to be of no public concern at about 10^{-6} to 10^{-7} (Baker et al. 1983; Clarke and Fleishman 1984; Meinhold 1984; Spangler 1987; Travis et al. 1987). Using a rounded risk factor of 10^{-2} per sievert for whole-body exposures, the IAEA concluded that the level of trivial individual effective dose would be in the range of 10 to 100 μSv/y (1 to 10 mrem/y). . This range was further justified as being a level of dose that was a few percent of natural background and a small fraction of normal variations in natural background.

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Current studies on the Biological Effects of Ionizing Radiation (BEIR) are contained in the BEIR VII study "Health Risks from Exposures to Low Levels of Ionizing Radiation" (NRC 2006). BEIR VII is the seventh report in a series of publications from the National Research Council (NRC) on the biological effects of ionizing radiation. BEIR VII focuses on the health effects from low levels of low linear energy transfer (low-LET) ionizing radiation, including x-rays and gamma radiation. The main purpose of the study was to update BEIR V (NRC 1990) using new information from epidemiologic and experimental research that had accumulated since the 1990 BEIR V review. BEIR VII recommended some small revisions to the factors for health effects calculations from a given amount of radiation exposure. It also offered agreement with the linear no-threshold (LNT) theory, which is that the relationship of health effects to radiation exposure is the same at low doses and dose rates as it is at high doses and dose rates, and that the relationship applies down to zero dose. The press release for BEIR VII stated, "The scientific research base shows that there is no threshold of exposure below which low levels of ionizing radiation can be demonstrated to be harmless or beneficial."

By establishing the primary dose criterion of 10 μSv/y (1.0 mrem/y) for clearance as a small fraction of the primary dose limit, consistency with the dose limit principle is ensured, even if individual exposures to multiple "cleared" sources were possible. In developing this standard, it has been assumed that licensed or otherwise regulated practices to which this standard *may* be applied have been justified and, therefore, it is not necessary to address the justification principle. Further, it is recognized that dose limits are established to ensure that the average member of a critical group will not receive doses in excess of the general 1 mSv/y (100 mrem/y) limit established as the allowable dose limit for all sources and pathways combined. The ALARA principle is based on the assumption, presumed to be conservative, that detriment resulting from any radiation exposure is proportional to the dose associated with such exposures, consistent with the recent BEIR conclusions. The ALARA principle is applied to ensure that doses resulting from an activity or practice are sufficiently below the established dose limit that the cumulative effects (suggested by the conservative LNT assumption) on the affected population(s) are ALARA, considering economic, technical, and social factors.

In establishing a dose limit for clearance of items and materials containing radioactive material, a full quantitative optimization (ALARA) analysis would entail the evaluation of various alternatives (means of controlling exposures), the associated collective dose reduction of each alternative, and the associated cost (including capital and other implicit costs) of implementing the controls, along with consideration of other factors. The result of the analysis would identify those alternatives that reduce doses to levels that are as far below the primary dose criterion as is reasonably achievable. The cost of such an analysis would likely be prohibitive, perhaps on the order of $100,000 or more in today's market. In general, the ALARA process can be considered a cost–benefit process in which the selection of the appropriate controls for reducing doses below the dose limit is based on the optimization of differential costs (expenses and risks) and benefits (risk reduction and other factors). The details associated with such evaluations are themselves limited by the benefit that can be achieved through use of the analysis in risk-management decision making. That is, if potential detriment (e.g., collective dose) is high, then there could be substantial benefit to be gained in optimizing the controls through rigorous quantitative analyses and comparison of alternatives. However, if the potential collective dose is low, then the cost of a quantitative ALARA analysis is not warranted, and little more than qualitative considerations can be justified.

In developing this standard, the screening levels have been selected in accord with the IAEA recommendations (2004, 2005) such that associated individual dose is well below the primary individual dose limit. The screening levels are based on the consideration of reasonably conservative estimates of the maximum dose to an individual under exposure conditions that are more likely to overestimate than

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underestimate potential doses. Thus, one can reasonably expect typical doses to members of the public from releases associated with the screening levels to be less than about 10 μSv/y (1.0 mrem/y). . In general, these doses will likely affect a very small fraction or subset of the population.

Assuming an average individual in the United States receives a nonmedical background radiation dose of about 3.2 mSv/y (320 mrem/y) (NCRP 2009, Figure 8.2), the individual and collective doses to the critical group resulting from clearance of items or materials using the criterion from this standard will be no more than about 0.2% of the dose the same population would receive from natural background in any year. The magnitude of the potential collective doses to the critical group associated with clearance of items or materials in accordance with this standard is so low that additional ALARA evaluations or analyses, or further reduction in the primary dose standard, are not deemed necessary. As a result, there is no need to assess quantitatively ALARA conditions to support the application of the clearance levels established in this standard. Rather, use of the screening levels consistent with the standard can be assumed to satisfy the ALARA requirements (the optimization principle).

The screening levels in this standard should not be considered as the sole basis for clearance. It is appropriate to develop clearance levels for release-specific materials, activities, or practices using site- or activity-specific factors that diverge from the screening levels provided in this standard. It is reasonable to expect that the use of less conservative, more specific, or more accurate parameters and assumptions, and possibly a reasonable expectation or knowledge of expected use (or at least first-use or disposition of materials to be released), will result in higher screening levels. It is presumed that such determinations will be shown to be justified based on a more rigorous application of the ALARA principle.

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Annex C

(Informative)

Comparisons with Other Guidance

The screening levels for clearance developed for this standard are next compared with previous guidance for clearance or unrestricted release of materials with volume or surface radioactivity. The first comparison is for materials with volume radioactivity, and considers the values from Table 1 of this standard and the values provided by the IAEA in Safety Guide No. RS-G-1.7 (IAEA 2004). This comparison is shown in Table C.1. For most radionuclides considered, there is an exact correlation between the clearance levels provided by this standard and the IAEA. However, the exception is ^{129}I, which emits a low-energy beta particle that is difficult to detect with field instruments. The rationale for increasing the clearance screening level for this standard was based on the overall IAEA scenario results and the IAEA's guidance to protect ground water should disposal to land of relatively large quantities of material be considered. The guidance provided in footnote c to Table 1 of this standard was developed to indicate that the clearance level should be lowered by one order of magnitude when disposal to landfills or direct disposal to soil is anticipated.

The second comparison is for materials with surface radioactivity and considers the values from Table 1 of this standard and the values provided by the U.S. Atomic Energy Commission (USAEC) in Regulatory Guide 1.86 (USAEC 1974), shown in Table C.2. As described in the 1999 version of this standard, it is expected that there is little agreement between Regulatory Guide 1.86 and the screening levels of this standard. This is in part because this standard is dose-based, whereas Regulatory Guide 1.86 was primarily based on instrumentation detectability, inhalation (through association with the maximum permissible concentrations in air found in 10 CFR Part 20), and ingestion (through association with the maximum permissible concentrations in water found in 10 CFR Part 20), which produced rather broad radionuclide groupings (USNRC 1991). For example, the general beta-gamma category in Regulatory Guide 1.86 included radionuclides that produce highly different scenario results, such as ^{60}Co and ^{99}Tc.

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Table C.1. Radionuclides and volume activity concentrations comparisons

Radionuclide	N13.12-2011[a] (Bq/g)	IAEA[b] (Bq/g)	Radionuclide	N13.12-2011[a] (Bq/g)	IAEA[b] (Bq/g)
^{3}H	100	100	^{147}Pm	1,000	1,000
^{7}Be	10	10	^{147}Nd	100	100
^{14}C	1	1	^{151}Sm	1,000	1,000
^{22}Na	0.1	0.1	^{152}Eu	0.1	0.1
^{35}S	100	100	^{154}Eu	0.1	0.1
^{36}Cl	1	1	^{155}Eu	1	1
^{45}Ca	100	100	^{153}Gd	10	10
^{46}Sc	0.1	0.1	^{160}Tb	1	1
^{51}Cr	100	100	^{170}Tm	100	100
^{53}Mn	100	100	^{171}Tm	1,000	1,000
^{54}Mn	0.1	0.1	^{181}Hf	1	1
^{55}Fe	1,000	1,000	^{182}Ta	0.1	0.1
^{59}Fe	1	1	^{181}W	10	10
^{56}Co	0.1	0.1	^{185}W	1,000	1,000
^{57}Co	1	1	^{185}Os	1	1
^{58}Co	1	1	^{191}Os	100	100
^{60}Co	0.1	0.1	^{190}Ir	1	1
^{59}Ni	100	100	^{192}Ir	1	1
^{63}Ni	100	100	^{203}Hg	10	10
^{65}Zn	0.1	0.1	^{202}Tl	10	10
^{73}As	1,000	1,000	^{204}Tl	1	1
^{74}As	10	10	^{207}Bi	0.1	0.1
^{75}Se	1	1	^{225}Ra	10	10
^{86}Rb	100	100	^{226}Ra	0.1	1
^{85}Sr	1	1	^{228}Th	0.1	1
^{89}Sr	1,000	1,000	^{229}Th	0.1	0.1
^{90}Sr	1	1	^{232}Th	0.1	1
^{91}Y	100	100	^{230}Pa	10	10
^{93}Zr	10	10	^{233}Pa	10	10
^{95}Zr	1	1	^{232}U	0.1	0.1
^{93m}Nb	10	10	^{233}U	1	1
^{94}Nb	0.1	0.1	^{234}U	1	1
^{93}Mo	10	10	^{235}U	1	1
^{97}Tc	10	10	^{236}U	10	10
^{97m}Tc	100	100	^{238}U	1	1
^{99}Tc	1	1	^{237}Np	1	1
^{103}Ru	10	10	^{236}Pu	1	1
^{106}Ru	0.1	0.1	^{237}Pu	100	100

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Radionuclide	N13.12[a] (Bq/g)	IAEA[b] (Bq/g)	Radionuclide	N13.12[a] (Bq/g)	IAEA[b] (Bq/g)
^{105}Ag	1	1	^{238}Pu	0.1	0.1
^{110m}Ag	0.1	0.1	^{239}Pu	0.1	0.1
^{109}Cd	1	1	^{240}Pu	0.1	0.1
^{115m}Cd	100	100	^{241}Pu	10	10
^{114m}In	10	10	^{242}Pu	0.1	0.1
^{115m}In	100	100	^{244}Pu	0.1	0.1
^{113}Sn	1	1	^{241}Am	0.1	0.1
^{125}Sn	10	10	^{243}Am	0.1	0.1
^{124}Sb	1	1	^{242}Cm	10	10
^{125}Sb	0.1	0.1	^{243}Cm	1	1
^{123m}Te	1	1	^{244}Cm	1	1
^{125m}Te	1,000	1,000	^{245}Cm	0.1	0.1
^{127m}Te	10	10	^{246}Cm	0.1	0.1
^{129m}Te	10	10	^{247}Cm	0.1	0.1
^{125}I	100	100	^{248}Cm	0.1	0.1
^{129}I[c]	0.1	0.01	^{249}Bk	100	100
^{131}I	10	10	^{248}Cf	1	1
^{134}Cs	0.1	0.1	^{249}Cf	0.1	0.1
^{135}Cs	100	100	^{250}Cf	1	1
^{137}Cs	0.1	0.1	^{251}Cf	0.1	0.1
^{131}Ba	10	10	^{252}Cf	1	1
^{140}Ba	1	1	^{253}Cf	100	100
^{139}Ce	1	1	^{254}Cf	1	1
^{141}Ce	100	100	^{254}Es	0.1	0.1
^{144}Ce	10	10			

[a]Where ANSI/HPS screening levels are from this standard.

[b]Where IAEA screening levels are from IAEA 2004.

[c]Because of potential ground-water concerns, the values for ^{129}I should be lowered by one order of magnitude when disposal to landfills or direct disposal to soil is anticipated.

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Table C.2. Radionuclides and surface radioactivity comparisons

Radionuclide	N13.12[a] (Bq/cm^2)	Regulatory Guide 1.86 (Bq/cm^2)[b]	Radionuclide	N13.12[a] (Bq/cm^2)	Regulatory Guide 1.86 (Bq/cm^2)[b]
^{3}H	100	0.83	^{147}Pm	1,000	0.83
^{7}Be	10	0.83	^{147}Nd	100	0.83
^{14}C	1	0.83	^{151}Sm	1,000	0.83
^{22}Na	0.1	0.83	^{152}Eu	0.1	0.83
^{35}S	100	0.83	^{154}Eu	0.1	0.83
^{36}Cl	1	0.83	^{155}Eu	1	0.83
^{45}Ca	100	0.83	^{153}Gd	10	0.83
^{46}Sc	0.1	0.83	^{160}Tb	1	0.83
^{51}Cr	100	0.83	^{170}Tm	100	0.83
^{53}Mn	100	0.83	^{171}Tm	1,000	0.83
^{54}Mn	0.1	0.83	^{181}Hf	1	0.83
^{55}Fe	1,000	0.83	^{182}Ta	0.1	0.83
^{59}Fe	1	0.83	^{181}W	10	0.83
^{56}Co	1	0.83	^{185}W	1,000	0.83
^{57}Co	1	0.83	^{185}Os	1	0.83
^{58}Co	0.1	0.83	^{191}Os	100	0.83
^{60}Co	0.1	0.83	^{190}Ir	1	0.83
^{59}Ni	100	0.83	^{192}Ir	1	0.83
^{63}Ni	100	0.83	^{203}Hg	10	0.83
^{65}Zn	0.1	0.83	^{202}Tl	10	0.83
^{73}As	1,000	0.83	^{204}Tl	1	0.83
^{74}As	10	0.83	^{207}Bi	0.1	0.83
^{75}Se	1	0.83	^{225}Ra	10	0.83
^{86}Rb	100	0.83	^{226}Ra	0.1	0.017
^{85}Sr	1	0.83	^{228}Th	0.1	0.017
^{89}Sr	1,000	0.83	^{229}Th	0.1	0.017
^{90}Sr	1	0.17	^{232}Th	0.1	0.17
^{91}Y	100	0.83	^{230}Pa	10	0.83
^{93}Zr	10	0.83	^{233}Pa	10	0.83
^{95}Zr	1	0.83	^{232}U	0.1	0.17
^{93m}Nb	10	0.83	^{233}U	1	0.83
^{94}Nb	0.1	0.83	^{234}U	1	0.83
^{93}Mo	10	0.83	^{235}U	1	0.83
^{97}Tc	10	0.83	^{236}U	10	0.83
^{97m}Tc	100	0.83	^{238}U	1	0.83
^{99}Tc	1	0.83	^{237}Np	1	0.017
^{103}Ru	10	0.83	^{236}Pu	1	0.017
^{106}Ru	0.1	0.83	^{237}Pu	100	0.017
^{105}Ag	10	0.83	^{238}Pu	0.1	0.017
^{110m}Ag	0.1	0.83	^{239}Pu	0.1	0.017
^{109}Cd	1	0.83	^{240}Pu	0.1	0.017
^{115m}Cd	100	0.83	^{241}Pu	10	0.017
^{114m}In	10	0.83	^{242}Pu	0.1	0.017
^{115m}In	100	0.83	^{244}Pu	0.1	0.017
^{113}Sn	1	0.83	^{241}Am	0.1	0.017
^{125}Sn	10	0.83	^{243}Am	0.1	0.017

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Radionuclide	N13.12[a] (Bq/cm^2)	Regulatory Guide 1.86 (Bq/cm^2)[b]	Radionuclide	N13.12[a] (Bq/cm^2)	Regulatory Guide 1.86 (Bq/cm^2)[b]
^{124}Sb	1	0.83	^{242}Cm	10	0.017
^{125}Sb	0.1	0.83	^{243}Cm	1	0.017
^{123m}Te	1	0.83	^{244}Cm	1	0.017
^{125m}Te	1,000	0.83	^{245}Cm	0.1	0.017
^{127m}Te	10	0.83	^{246}Cm	0.1	0.017
^{129m}Te	10	0.83	^{247}Cm	0.1	0.017
^{125}I	100	0.017	^{248}Cm	0.1	0.017
^{129}I	0.1	0.017	^{249}Bk	100	0.017
^{131}I	10	0.017	^{248}Cf	1	0.017
^{134}Cs	0.1	0.83	^{249}Cf	0.1	0.017
^{135}Cs	100	0.83	^{250}Cf	1	0.017
^{137}Cs	0.1	0.83	^{251}Cf	0.1	0.017
^{131}Ba	10	0.83	^{252}Cf	1	0.017
^{140}Ba	1	0.83	^{253}Cf	100	0.017
^{139}Ce	1	0.83	^{254}Cf	1	0.017
^{141}Ce	100	0.83	^{254}Es	0.1	0.017
^{144}Ce	10	0.83			

[a]Where ANSI/HPS screening levels are from this standard.

[b]Where the Regulatory Guide 1.86 (USAEC 1974) values are “average” values converted from units of dpm/100 cm^2 to units of Bq/cm^2.

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Annex D

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