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Assessing the Radioactivity in Materials used for SMR Construction (ARMSC)

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1 TERMS AND DEFINITIONS

Term	Definition
Activation products	Also referred to as neutron activation products. These are materials that become radioactive due to neutron absorption (capture) by a stable isotope. Isotopes have different neutron capture probabilities that are dependent on neutron energy. For example, a stable ^{59}Co atom becomes activated to radioactive ^{60}Co after absorbing a neutron.
Activity Units	The SI unit for activity is Becquerel (Bq) and is defined as one disintegration (decay) per second. It is defined in terms of reciprocal seconds (s^{-1}). The older imperial unit Curie (Ci) is based on the activity of 1g of ^{226}Ra. The conversion is 1 Ci = 3.7×10^{10} Bq.
Alpha radiation, alpha particles (α)	A ^{4}He nucleus consisting of 2 protons and 2 neutrons. May also be referred to as a helium ion. The most common source of alpha radiation is the decay of heavy radionuclides (uranium for example).
Background radiation	A constant source of ionizing radiation resulting from a variety of sources such as cosmic and terrestrial radiation in the environment. Also defined as the radiation that is detected by an instrument when no radioactive source is present.
Beta radiation, beta particles (β)	A high energy electron (β-) or positron (β+) emitted during the process of radioactive decay. For example, tritium (^{3}H) decays to ^{3}He through the emission a low energy β- particle.
Contamination	The presence of unwanted radioactive material.
Counts & Count Rate	Counts are the number of individual events detected in a detector. The count rate is the number of events over a defined time interval typically measured in counts per minute (cpm) or counts per second (cps).
Daughter products, decay products, progeny	The nuclide that results following radioactive decay.
Fission Products	Atomic fragments that result following the fission (splitting) of a heavy atom such as ^{235}U. During the fission process the uranium atom splits into two smaller nuclei, several neutrons, gamma rays as well as energy in the form of heat.
Fixed contamination	Radioactive material that is adhered to a non-radioactive surface and cannot be removed by non-abrasive means.
Gamma radiation, gamma photons, gamma rays (γ)	High energy electromagnetic radiation commonly produced in the radioactive decay process.

Term	Definition
Half-life	Defined in terms of probability, time required for the activity of a given radioactive substance to decay to half of its initial value.
Isotope	An element that has the same number of protons but different number of neutrons.
Loose contamination	Radioactive material deposited on a surface that is easily removable through means of wiping
Moderator	Material in a reactor used to reduce the energy of neutrons to facilitate a fission chain reaction.
Parent Radionuclide	The original nuclide that undergoes radioactive decay.
Radioactive decay	The process whereby an unstable atomic nucleus spontaneously emits radiation to achieve stability. Achieving stability may require one or multiple decay steps. The decaying isotope is referred to as the parent and the resulting isotope (which may still be radioactive) is referred to as the daughter. Long decay chains are referred to as decay series (the uranium series for example) where subsequent daughters are referred to as progeny.
Radioisotope, radionuclide	An isotope that undergoes radioactive decay at some point in time.
Secular Equilibrium	Occurs when the half-life of a radioactive daughter is significantly shorter than the half-life of the parent. In this case, the quantity (or activity) of the daughter builds up until the number of daughter atoms being produced is equal to the number decaying. Once equilibrium is established, the total activity of the daughter decays at the same rate as the parent.
X-Rays	High energy electromagnetic radiation. The main distinction between x-rays and gamma rays is the point or origin. X-rays are produced by processes outside of the nucleus while gamma rays originate from within the nucleus. Generally, x-rays have lower energies compared gamma rays, however they do overlap on the electromagnetic spectrum. Once emitted, x-rays and gamma rays of a given energy are indistinguishable.

ACRONYMS

ALARA	As Low As Reasonably Achievable
ARMSC	Assessing the Radioactivity in Materials used for SMR Construction
BC	Before Christ
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
CO	Carbon Monoxide
CO_2	Carbon Dioxide
CPM	Counts Per Minute
CSA	Canadian Standards Association
DCGL	Derived Concentration Guideline Level
$DCGL_w$	Weighted Derived Concentration Guidance Level
DL	Detection Limit
DQO	Data Quality Objective
DT	Decision Threshold
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
GSR	General Safety Requirements
HPGe	Hyper Pure Germanium
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
ISO	International Organization for Standardization
IT	Information Technology
LBGR	Lower Boundary of the Gray Region
LSC	Liquid Scintillation Counting
MARLAP	Multi-Agency Radiological Laboratory Analytical Protocols
MARSAME	Multi-Agency Radiation Survey and Assessment of Materials and Equipment manual
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MS	Mass Spectrometry
MTR	Mill Test Report
NB	New Brunswick
NORM	Naturally Occurring Radioactive Material
NRCAN	Natural Resources Canada
NSRDR	Nuclear Substances and Radiation Device Regulations
NUREG	Nuclear Regulatory
PMT	Photo Multiplier Tubes
PPM	Parts Per Million

PWR	Pressurized Water Reactor
QA	Quality Assurance
REGDOC	Regulatory Document
RMS	Records Management System
SMR	Small Modular Reactor
TECDOC	Technical Document
UCL	Unconditional Clearance Levels
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
US	United States
UV	Ultraviolet
VBA	Visual Basic for Applications
WRS	Wilcoxon Rank Sum
XRF	X-ray Fluorescence

2 INTRODUCTION

2.1 Background

As Canada advances toward a low-carbon economy, many forms of energy are needed to power the growing demand for clean, affordable, and reliable electricity. To bring clean power to more Canadians, the Government of Canada has committed to working with the provinces and territories to enable deployment of Small Modular Reactors (SMR).

Natural Resources Canada (NRCan) launched the Enabling SMR Program in February 2023. Knowledge generated from projects under this program will serve to advance the objectives of Canada’s SMR Action Plan [1]; and can inform policies and decision-making pertaining to the life cycle of SMRs. The program aims to fund and support research and development in areas of:

- Waste management for SMRs; and
- Support the creation of SMR supply chains

This document, Assessing the Radioactivity in Materials used for SMR Construction (ARMSC), falls under the waste management category, which has the overall goal of advancing knowledge and understanding in the management, characterization, and minimization of SMR waste. The focus of this document is to provide guidance to support radioactive waste minimization during decommissioning through the characterization of pre-existing radioactive substances in common construction materials.

Contamination or activation of building materials from the operation of the facility may be indistinguishable from the radioactive elements originally contained within the building materials. Without adequate baseline characterization data from when the facility was constructed, waste generated from the decommissioning and demolition of these facilities at the end of life could be incorrectly classified leading to increased quantities of radioactive waste.

The regulatory expectation of assessing the baseline radioactivity in building materials prior to commissioning is outlined in Canadian Nuclear Safety Commission (CNSC) REGDOC-2.11.2 [2] under clause 10.1. However, there is little guidance on what the challenges are to acquiring this data prior to commissioning, how such baseline surveys can be performed, and how the data can later be used to support decommissioning. The ARMSC aims to address these issues.

2.2 Objective

The objective of this document is to provide guidance to SMR vendors and planned owners or operators in Canada, that outlines the characterization of radiological elements in the most common building materials during the construction phase. This guide will support decommissioning and waste management activities by providing a strategy to characterize the baseline activity of radioactive substances within building materials. Characterization will determine the baseline activity of radioactive materials before the facility is licenced and operated thereby fulfilling the requirements of CNSC REGDOC 2.11.2 [2]. The acquired data may be compared against characterization data during facility decommissioning to ensure materials are appropriately managed as radioactive waste or unconditionally released based upon pre-existing quantities of radioactive materials.

2.3 Scope

The ARMSC document provides an overview of the sources of radiation, sampling techniques, and analyses methods associated with determining the baseline or pre-construction radioactivity of SMR building materials. The current revision of this document focuses on the two most common building materials, namely concrete and steel, which make up the significant majority. The pre-construction characterization of these two materials can result in a significant impact to the clearance and classification of waste during decommissioning. The concepts and methodologies presented can be easily extrapolated to other building materials of interest. Based on industry feedback, this document may be expanded to include other building materials in subsequent revisions. It is important to note that site specific characterization and the assessment of non-radiological contaminants are outside of the scope of this guide.

This document aims to consolidate industry information, identify issues in acquiring data, and offer practical guidance that help overcome the challenges associated with determining this baseline radioactivity. In addition, the following practical challenges encountered during decommissioning are also addressed:

1. How the regulatory criteria for the clearance of Naturally Occurring Radioactive Materials (NORM) differ from other radioactive substances, and how this impacts the baseline characterization of construction materials [4] [5];
2. Recommendations on what to do if NORM concentrations exceeding the exemption quantities or unconditional clearance levels are encountered during the construction phase;
3. How alpha emitters and other hard to detect radionuclides are typically handled in operating power plants [6], and the challenges of using the same method for use of recycled material and during decommissioning;
4. The issues associated with the decision threshold and detection limit for low counts samples [7] [8];
5. The practical application of the MARSSIM and MARSAME criteria for the number of samples required in an investigation area [9]; and
6. Data retention challenges during the life of the facility.

3 ENVIRONMENTAL SOURCES OF RADIATION

Environmental sources of radiation are an inherent, historical, and continuing feature of life on Earth. They are found in various concentrations in common media ranging from construction materials, to food, and even within the human body itself. For the purposes of this guide, environmental sources of radiation related to the construction of SMRs are divided into two main categories: Naturally Occurring Radioactive Material (NORM) and artificial and synthesized radioactive materials.

3.1 Naturally Occurring Radioactive Material

NORM are divided into two broad categories. Cosmogenic radionuclides (such as tritium and ^{14}C) that are produced by cosmic ray interaction in the upper atmosphere and terrestrial radionuclides that are in located the Earth’s crust. In the context of construction materials, NORM consists almost entirely of terrestrial sources that migrate from the Earth’s crust to the surface through either natural processes (such as diffusion through cracks, or transportation by groundwater) or the result of human activities (such as mining or mineral extraction). In addition, higher concentrations of NORM can exist due to activities such as refinement, material processing, and fabrication. Terrestrial radionuclides of interest consist of isotopes of uranium, thorium and potassium whose half-lives are comparable to the age of the Earth itself.

The term NORM is primarily used to distinguish natural sources of radiation from those that are synthesized by or a product of the nuclear fuel cycle. In developed countries, exposure to NORM constitutes about half of the average person’s yearly radiation dose (the other half being nuclear medicine). Radiation doses resulting from NORM are a natural part of living on the planet and cannot be prevented. Therefore, with the exception of import, export and transportation of material, NORM is exempt from regulation by the CNSC. Rather, guidelines for the management of NORM are provided by Health Canada [10] and its regulation is the responsibility of individual provinces and territories.

3.1.1 Uranium, Thorium, & Actinium Decay Series

The uranium and thorium isotopes form characteristic decay chains whose daughter products not only emit different types of radiation (alpha, beta, gamma) but also undergo a phase change from solid to gas, which is the case for radon, that can further result in diffusion and dispersal. The three main decay chains found in nature are known as the Uranium Series, Thorium Series, and Actinium Series and are shown below in Figure 3-1, Figure 3-2 and Figure 3-3 respectively. Each of the diagrams below originates with the radionuclide that has the longest half-life. The figures show the main radionuclides within the decay chain, the type of radiation emitted from each decay, the half-life, the physical state and terminating (stable) nuclide. Two common characteristics of all three decay series are that each has a gaseous member (an isotope of radon), and each terminates with a stable isotope of lead.

It is important to note that during the early years of the solar system, the chains were longer with more unstable heavier elements at the top, with relatively shorter half-lives, that have since decayed to extinction. The long-lived uranium and thorium isotopes, currently shown at the top of the series, essentially represent bottle necks that allow the series to exist today. Of the three decay series, the actinium series is least abundant on earth. This is because the half-life of its parent radionuclide ^{235}U (0.7 billion years) is relatively short compared to the age of the earth (about 4.5 billion years). Thus, a significant portion of the original ^{235}U on the planet has decayed, and currently ^{235}U comprises about 0.7% of the uranium found in nature. While ^{235}U has significant power generation and military applications, its concentrations are well below detection thresholds when it comes to building materials. Some of these extinct radionuclides have since been synthesized through nuclear reactions or are byproducts in the nuclear fuel cycle. These will continue to decay down their respective decay series, however, they are not categorized as NORM. For example, ^{239}Pu (half-life of 24,110 years) is commonly found in used nuclear fuel and despite being fissile, also undergoes alpha decay to ^{235}U and would follow the actinium decay series shown in Figure 3-3, however it is not treated as NORM. Therefore, it is the uranium and thorium series that are the dominant decay chains found in building materials.

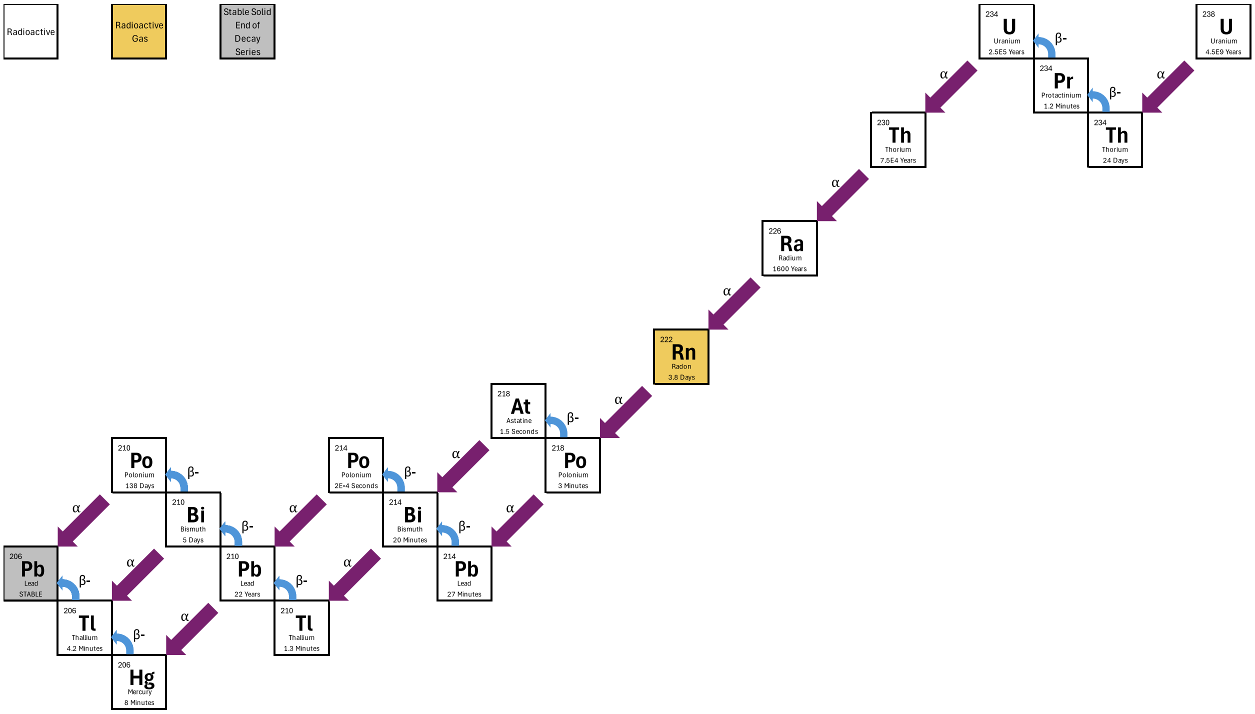

Figure 3-1: Uranium Series

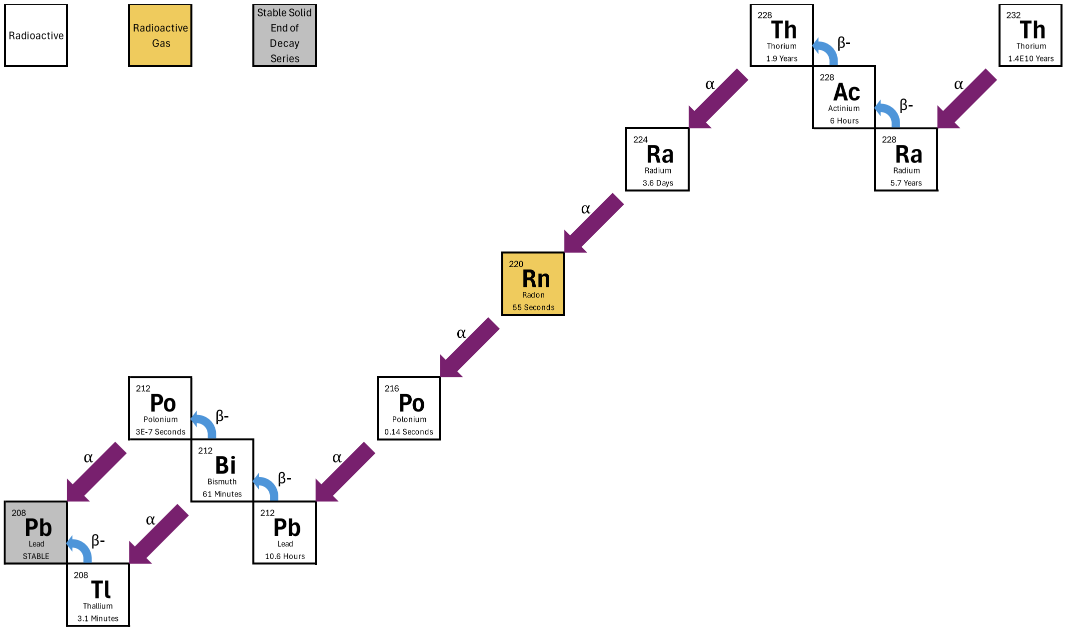

Figure 3-2: Thorium Series

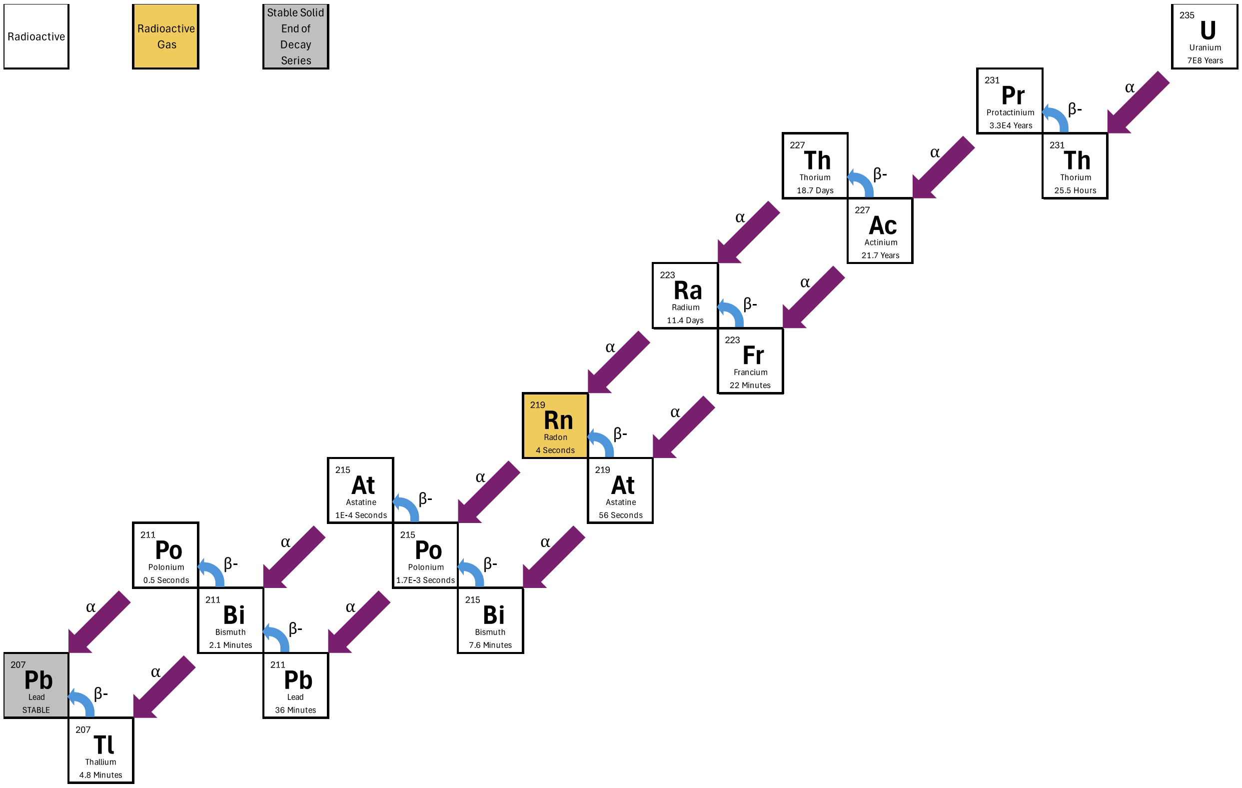

Figure 3-3: Actinium (^{235}U) Series

3.1.2 Potassium-40

Another major source of NORM is Potassium-40. Potassium is one of the most abundant elements on earth and is essentially found everywhere in nature. Of particular importance is its presence in soil and, as a result, in many common building materials. Also, since it is an essential nutrient for plant growth, potassium is found in all plant and animal tissue. Natural potassium consists of 3 isotopes, two of which are stable and one radioactive. The stable isotopes ^{39}K and ^{41}K account for almost all the natural potassium present. The radioactive ^{40}K comprises only a small fraction, about 0.012%. Despite being in trace quantities relative to the stable isotopes, the sheer abundance of potassium in nature means ^{40}K can be readily detected. With a half-life of 1.25 billion years, ^{40}K decays to either stable ^{40}Ca by way of beta minus decay 89% of the time, or stable ^{40}Ar by way of electron capture and emission of a high energy gamma ray (1.46 MeV) 11% of the time as shown in Figure 3-4. It is this high energy characteristic gamma that allows the easy detection of ^{40}K. On rare occasions (0.001%), ^{40}K decays by positron (beta plus) emission to ^{40}Ar.

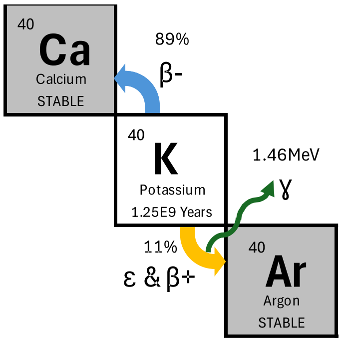

Figure 3-4: Decay of ^{40}K

3.1.3 Quantities of Naturally Occurring Radioactive Material

Typical quantities of NORM found in common building materials are shown below in Table 3-1. The data presented was obtained from Reference [11] and primarily collected in Europe. The quantities of NORM are presented as a range of activity concentrations in Bq/kg. Note that concentrations of the uranium series are presented in terms of the ^{226}Ra concentration since the 185 keV gamma is readily identifiable in gamma spectrometry measurements. This is further discussed in the analysis techniques section (Annex B). It is important to note that the activity concentrations presented in Table 3-1 show a large degree of variation and are meant to provide the reader with an idea of expected range of NORM quantities found in common building materials. The table is not intended to be a substitute for performing actual measurements but rather highlights the need for site and sample specific quantification during planning and construction phases for future decommissioning purposes.

Table 3-1 Activity Concentrations (Bq/kg) of NORM in Building Materials [11]

Material	^{226}Ra	^{232}Th	^{40}K
Concrete	1-250	1-190	5-1570
Aerated concrete	9-2200	1-220	180–1600
Clay bricks	1-200	1-200	60–2000
Sand-lime bricks and limestone	6-550	1-30	5–700
Natural building stones	1-500	1-310	1–4000
Natural gypsum	1-70	1-100	7–280
Cement	7-180	7-240	24–850
Tiles	30-200	20-200	160-1410
Phosphogypsum	4-700	1-53	12-120
Blast furnace slag stone and cement	30-120	30-220	-

Currently detailed distributions of NORM concentrations in construction materials sourced from the various regions of Canada are not available. A study of radioactivity in Canadian building materials was conducted in 1992 [11], that focused on materials that were found on construction sites in the Montreal area and imported materials purchased from local stores. That study found that NORM concentrations of the materials examined were within the ranges of Table 3-1. However, large variations in NORM concentrations across the country are expected, with potentially elevated levels in the uranium rich areas. In addition, regions focused on oil & gas production, fertilizer production, and mining & underground workings may have elevated, albeit more localized, concentrations of NORM that can impact levels in building materials. As an extension to this document, a NORM activity assessment across Canadian building material suppliers is recommended. Efforts may be focused on potential SMR construction sites and locations where raw materials (such as aggregate) for those sites would be sourced.

3.2 Artificial and Synthesized Radioactive Material

Artificial or synthesized radioactive materials consist of radionuclides that have been created on earth, by human beings, through the use of nuclear weapons, nuclear reactors or particle accelerators. In most cases there are no natural mechanisms or processes through which they are produced. While many are byproducts of the military or civilian generation sectors, specific synthesized radioisotopes have vast medical, industrial, and research applications. This guide focuses a small subset of artificial radionuclides that may be encountered during the construction phase of SMRs. The artificial radionuclides discussed are divided into two categories: fallout and recycled materials.

Radionuclides from both categories may be encountered during construction but are expected to be in relatively small quantities compared to NORM. These radionuclides are fission and/or activation products that are produced in nuclear power plants and are readily detected during operation and outages. For an operating facility they are typically found within radioactive systems or other areas of the plant resulting from invasive maintenance work and from incidents of contamination spread or spills from radioactive systems. The challenge with respect to decommissioning is distinguishing between baseline quantities present during construction and what has been introduced from plant operation. Provided that radionuclide activities remain below clearance thresholds, unrestricted release as a part of the decommissioning process will be possible. However, if these radionuclides are detected during the construction phase, and accounting for radioactive decay, valuable insight to support decommissioning initiatives and strategies may be gained by being able to distinguish what was there in the beginning from what was introduced during operations.

3.2.1 Fallout

Nuclear fallout refers to residual radioactive material that is dispersed globally following the detonation of a nuclear weapon or following a nuclear incident in which containment of a nuclear substance was compromised. In either case, the event results in radioactive particulate material of various sizes and radioactive vapours that are propelled into the atmosphere. While heavier material falls to the ground near the incident site, lighter material may be carried into the upper atmosphere where it can circulate for years eventually falling back to the earth as precipitation or dust. Weather patterns and particle size greatly impact the path and location of fallout, however the longer this material stays in the atmosphere the more widespread it can become. Most of the nuclear weapon testing and all nuclear accidents have been in the northern hemisphere, however fallout has been observed globally.

3.2.1.1 Cesium-137

The radionuclide of particular importance from a construction point of view is ^{137}Cs, which is a semi-volatile fission product that results from the splitting of the ^{235}U atom. Being an alkali metal (Group 1 of the periodic table), it is highly reactive and readily bonds with chlorides to form a salt. In the environment, it can easily dissolve in water and bind to soil, which is how small quantities can end up in building materials such as concrete. The quantities of ^{137}Cs that were released from weapons testing and nuclear accidents are presented in various reports published by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), References [12] [13] and [14], and are summarized below in Table 3-2.

Table 3-2 Quantities of ^{137}Cs Released from Various sources [12] [13] [14]

Source	^{137}Cs Activity Released (PBq)	Year of Occurrence
Weapons Testing	960	Peaked in 1960s
Chernobyl	10	1986
Fukushima	20	2011

With a half-life of about 30 years, ^{137}Cs decays via beta emission to stable ^{137}Ba 5.4% of the time and to metastable ^{137m}Ba 94.6% of the time. The ^{137m}Ba has a half-life of about 2.5 minutes and subsequently decays to stable ^{137}Ba through the emission of a 0.662 MeV gamma photon. It is this characteristic gamma that allows for the easy identification and quantification of ^{137}Cs (provided secular equilibrium is maintained). The ^{137}Cs decay scheme is shown below in Figure 3-5.

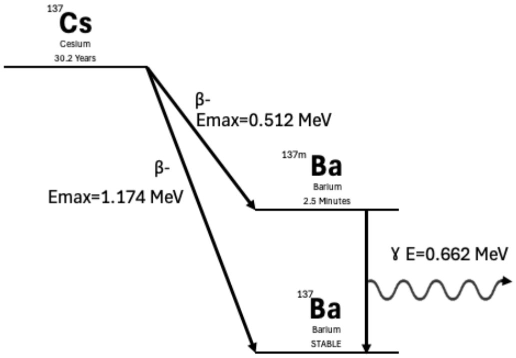

Figure 3-5: Decay Scheme of ^{137}Cs

3.2.2 Recycled Materials

In recent years waste management initiatives have been focused around the circular economy as a means to achieve sustainability. The main premise behind the circular economy is that waste material is not only to be prevented or minimized but cycled back into the various production processes and reused. Due to the additional hazards associated with radioactive material, specific regulations, discussed later in this report, must be met before any such material can be a viable candidate for recycling. Nuclear substances are regulated by the CNSC who issues specific licenses for their possession and use. For materials to be eligible for recycling their radioactivity contents would have to be below the minimum quantity for regulatory control. That is, the activity must be below a defined threshold for any material to be eligible for release to the public domain. This threshold quantity is referred to as the clearance criteria and specified in terms of an activity, activity concentration, or public dose limit in the Nuclear Substances and Radiation Devices Regulations [4]. Material clearance criteria are further discussed in Section 5.

While recycled nuclear materials may contain any number of activation and/or fission products, special consideration should be given to ^{60}Co and hard to detect radionuclides as described below.

3.2.2.1 Cobalt-60

Cobalt-60 is one of the most common radionuclides found in operating nuclear power plants. Its main source is the activation naturally occurring stable ^{59}Co, which is an unavoidable impurity found in the iron and steel that is ubiquitously used in plant construction. In the presence of neutron radiation (during reactor operation) the ^{59}Co undergoes a (n,γ) reaction and becomes activated to ^{60}Co.

Iron and steel have vast applications in nuclear facilities ranging from structural supports, to piping and valves, and maintenance equipment. Corrosion material and deposits containing ^{59}Co that circulate through the core may become activated and subsequently deposited in areas of the reactor heat transport circuit. During maintenance activities, this material may be released as radioactive contamination. In addition, structural components in relative low neutron fluxes for extended periods of time may become slightly activated. Therefore, trace amounts, below clearance thresholds, of ^{60}Co may be present in recycled steel or iron from decommissioned nuclear facilities in the form of slight activation or low-level fixed contamination.

In addition to being a byproduct of nuclear power generation, ^{60}Co is also produced and harvested for use in medical and industrial equipment. There have been several cases globally where such equipment, at the end of its life, with its radioactive source(s) intact was introduced into the scrap recycling process. One of the most significant cases was in the 1980s in Taiwan when several hundred buildings were constructed from steel contaminated with elevated levels of ^{60}Co [15] [16]. To detect the presence of radioactive materials and prevent such contamination, many international ports of entry, scrap metal receivers and steel foundries are equipped with drive through radiation detectors, referred to as portal monitors. Every vehicle entering such a facility must first be scanned by the portal monitor for gamma emitting radionuclides before its payload is cleared and received.

With a half life of 5.27 years, ^{60}Co decays to stable ^{60}Ni via beta minus decay accompanied by the emission of two gamma rays with energies of 1.17 MeV and 1.33 MeV. The corresponding decay scheme is shown below in Figure 3-6. The two characteristic high energy gamma rays make ^{60}Co one of the most recognizable and easily identifiable radionuclides. Also, ^{60}Co may be a surrogate for other (hard to detect) radionuclides, and its presence could warrant additional analyses if the history of the material is not well known.

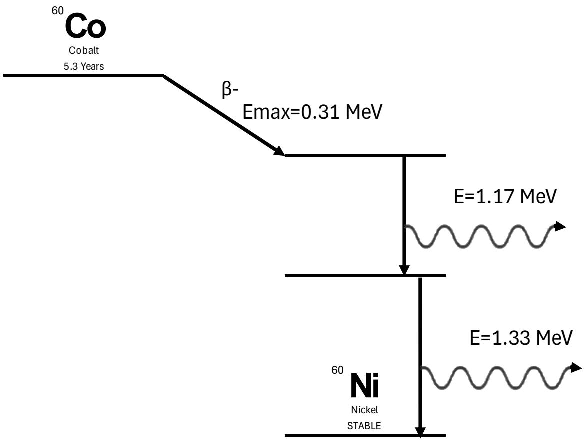

Figure 3-6: Decay Scheme of ^{60}Co

3.2.2.2 Hard to Detect Radionuclides

Hard to detect radionuclides are those species whose activity cannot be readily detected and quantified through direct in-the-field measurement methods due to the energy, type, and abundance of the radiation they emit. Both ^{137}Cs and ^{60}Co, described above, are considered easily detectable due to their high energy characteristic gammas. Hard to detect radionuclides, on the other hand, have no such characteristic gammas. They consist mostly of low energy pure beta (β-) emitters or alpha emitters, and as a result can be easily shielded (or masked) with increasing material depth. For some hard to detect radionuclides, even the presence of layer of film or moisture on the surface can inhibit detection. Many of these species may be present in porous material such as concrete despite having no evidence of other (easily detectable) fission or activation products. Therefore, a thorough understanding of the history and source of such material is imperative since their detection often requires destructive lab testing which can be costly and time consuming.

Tritium (^{3}H) and ^{14}C are two commonly encountered examples low energy pure beta emitters. While both are cosmogenic radionuclides and found in nature, significant quantities were released during nuclear weapons testing and are produced during reactor operation such that from a material clearance point of view they are treated as artificial radionuclides rather than NORM.

3.2.2.2.1 Tritium

In pressurized heavy water reactors, such as the Canada Deuterium Uranium (CANDU) reactor, significant quantities of ^{3}H are produced from the activation of deuterium in the form of D_2O utilized as the coolant and moderator. Spills and excursions can occur inside containment during refuelling, and maintenance activities. The resulting tritium oxide behaves no different than water or water vapour and may be absorbed to great depths (10s of inches) in porous materials such as concrete. With a half-life of 12.3 years tritium decays via the emission of a low energy beta radiation (average energy of 5.7 keV and maximum energy of 18.6 keV) which cannot be detected with contamination survey meters. In comparison, significantly smaller quantities of tritium are produced in other reactor types such as Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR) that do not utilize deuterium. Non-water Generation IV reactors do not produce tritium in their cooling or moderator systems, however smaller quantities may still be produced by other mechanisms (e.g. ternary fission).

3.2.2.2.2 Carbon-14

In nuclear reactors, ^{14}C is formed through neutron interactions with isotopes of impurities found in the fuel components, coolant, moderator, and structural material. The dominant production mechanisms are:

- $^{13}C\ (n,\gamma)\ ^{14}C$;
- $^{14}N\ (n,p)\ ^{14}C$; and
- $^{17}O\ (n,\alpha)\ ^{14}C$.

The majority of ^{14}C produced during operation is captured in the various purification systems. Excursions can occur during open system maintenance and venting operations. ^{14}C that binds with hydrogen to form particulate material that can penetrate porous media while the small quantity gaseous radiocarbon ($^{14}CO_2$) is released as a monitored effluent. With a half-life of 5730 years ^{14}C quantification is very relevant for decommissioning purposes. Although its beta radiation is substantially more energetic than tritium, with an average energy of 49 keV and a maximum energy of 156 keV, detection efficiencies of ^{14}C with in-the-field survey instruments are fractions of 1%. Therefore, the use of lab equipment is required for quantification.

3.2.2.2.3 Transuranic Alpha Emitters

Of all the hard to detect nuclides, transuranic alpha emitters pose the most significant biological hazards due the high energy (4 MeV to 6 MeV) alpha particles they emit. The principal transuranics of interest include isotopes of americium, curium and plutonium. They are formed by the neutron activation and subsequent decay of nuclear fuel, trace quantities of fuel material on external surfaces of the fuel cladding (tramp uranium), and from fuel defects releasing transuranics into the reactor coolant system. As such, they can be found in areas associated with the heat transport system (reactor coolant), used fuel cooling systems, fuelling systems & equipment, and any corresponding radioactive material storage areas. Historical fuel defects and fuel failures therefore play a significant role in the circulating inventory of transuranics, most of which have very long half-lives. Transuranics may become liberated from the system through spills or contamination spread during open system inspection and maintenance activities. Based on operating experience and the potential for significant internal exposures, during operation, defined alpha classifications are applied to impacted areas which stipulate that additional protection measures and alpha specific monitoring be implemented. The alpha classifications are based on Electric Power Research Institute (EPRI) guidelines [6] in which a given area or system is assigned an alpha level according to the abundance of loose alpha contamination relative loose beta-gamma contamination. In practice, classification is based on determining the loose beta-gamma to alpha ratio as measured by smears. The three alpha classifications are shown in Table 3-3[1].

Table 3-3: EPRI Alpha Level Classification

	Level I	Level II	Level III
Alpha Risk	Minimal	Significant	Elevated
Internal Exposure from Loose alpha	<10%	10% - 90%	>90%
Loose Beta-Gamma to Loose Alpha Ratio ($\beta\gamma$)/α	>30 000	30 000 - 300	<300

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[1] Note that beta-gamma to alpha ratios are derived from International Commission on Radiological Protection (ICRP) Publication 30 dose coefficients. Applying dose coefficients from ICRP Publication 60 or later will result in different ratios (typically a factor of two reduction)

The method for the quantification of transuranic alpha emitters, as described the EPRI guide [6], is through the establishment of radionuclide fingerprints of impacted areas and the use of surrogate radionuclides such as ^{60}Co, ^{137}Cs and ^{144}Ce and established ratios.

Most transuranic alpha emitters have half-lives that are much longer than beta-gamma emitters. Therefore, as the contamination ages, and is not replenished, the beta-gamma to alpha ratio will tend to shift towards Alpha Level III. This has significant implications in the context of decommissioning since the use of surrogate radionuclides to identify and quantify the hard to detect transuranics becomes diminished over time. This necessitates the need for alpha specific characterization and accurate classification of impacted areas based on operational history. Areas where aged contamination, requiring specific characterization, may be encountered include:

- Radioactive material storage areas for material and equipment removed from the reactor for a long period of time;
- Waste treatment systems;
- Tanks and vessels containing reactor coolant;
- Areas where spills or loss of contamination control events may have occurred and contamination could have migrated (such as floor drains, piping, and sumps); and
- Areas where fixatives have been applied to seal in historical contamination.

While this guide focuses on building materials, the importance of accurately documenting and maintaining operational records, particularly for events associated with systems that interface with nuclear fuel, is imperative from a facility decommissioning perspective in defining alpha impacted areas. More information on records management is found in Section 9.2.

4 RADIATION DETECTION LIMITS

Radioactive decay is a random process and is defined as the spontaneous transformation of unstable nuclei as they release energy to (in some cases eventually) achieve stability. Thus, any measurement based on the detection of radiation emitted from the decay process is subject to statistical fluctuation. These fluctuations are an inherent and unavoidable source of uncertainty in all radiation detection systems. The detection capability, or detection limit, of a radiation monitoring system is one of its most important performance parameters. It is defined as the quantity of radioactive material that can be detected reliably within defined confidence level (typically set to 95%).

The collection of reference data during the construction phase, and its eventual use during facility decommissioning, require that measured values be compared against a release criterion to make a decision. The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) [9] recommends that the detection limit be less than the release criterion (10% to 50% is considered good practice).

Detection limits are based on the premise that there is always error associated with every measurement. For environmental radioactivity measurements, which typically involve attempting to quantify a small amount of radiation, the uncertainty makes it difficult to distinguish the source from the background radiation. The primary variables that impact detection capability are the background count rate, the detector efficiency, and the duration of the measurement (count time). It is through the control of these three parameters that significantly higher detection capabilities can be achieved for lab equipment over in-field equipment. Effective survey planning therefore requires taking detection capabilities into account.

This section provides some background on the concepts of decision threshold (DT), detection limit (DL), minimum detectable activity (MDA) or minimum detectable concentration (MDC) as well as the basis for establishing these values. Additional details on statistical methods, and derivations are found in Annex A.

It is recommended that the equations presented in CSA N288.0 Appendix D [7], and shown below, be used to calculate detection limits of a given system:

$$R_C = \frac{k_\alpha^2}{2t_0}\left[1 + \sqrt{1 + \frac{4N_0 t_g^2}{k_\alpha^2 t_0}\left(\frac{1}{t_0} + \frac{1}{t_g}\right)}\right] \quad \textbf{Eq. (1)}$$

$$R_D = R_C + \frac{k_\beta^2}{2t_g}\left[1 + \sqrt{1 + \frac{4R_C^2 t_g^2}{k_\alpha^2 k_\beta^2}}\right] \quad \textbf{Eq. (2)}$$

Where,

R_C = Decision threshold, which is defined as the net signal level (in terms of count rate) above which an observed signal can be reliably considered detected (i.e. considered above background)
R_d = Detection limit, which is defined as the net signal level (in terms of count rate) that can be expected to be observed with a defined level of certainty
k_α = Set to equal 1.645 to ensure a 95% confidence level
k_β = Set to equal 1.645 to ensure a 95% confidence level
t_0 = Sample time
t_g = Background time
N_0 = Background number of counts for the interval of t_0

The decision threshold and the detection limit are conventionally defined in terms of counts or count rate. This value can be converted to an activity (Bq), or activity concentration (Bq/kg) with a suitable conversion factor that accounts for the efficiency of the detector. It is a good practice to compare the measurement with the decision threshold in terms counts or count rate because those values have the least uncertainty. When the decision threshold is expressed as an activity, or activity concentration, the uncertainty of the efficiency of the detector is implicitly added to the intrinsic uncertainty of the counting process. As a result, the detection limit is higher than it could be.

4.1 Minimum Detectable Activity/Concentration

It is advantageous, particularly when comparing to release criteria or defining instrument capabilities to express the detection limit (count rate) in terms of an activity (Bq), surface activity (Bq/cm^2) or activity concentration (Bq/kg). This corresponds to the Minimum Detectable Activity (MDA) or Minimum Detectable Concentration (MDC). The expression for the MDC is shown below where C is an appropriate factor to convert count rate to the desired activity units. The factor accounts for parameters such as detector efficiency, detector area, and branching ratio of the radionuclide of interest.

$$MDA\ or\ MDC = CR_D$$ **Eq. (3)**

Where,

C = Factor to convert count rate to the desired activity units
R_d = Detection limit as defined by Eq. (2)

The MDA/MDC is therefore an a priori activity above the critical level that an instrument can be expected to detect 95% of the time. When calculating the MDA/MDC, it is important to utilize a background that is at the high end of what is expected. However, higher than expected backgrounds will result in an increased MDA/MDC that may exceed the release criterion and not meet the data quality objectives of the survey.

This guide uses the terms MDA and MDC somewhat interchangeably in this discussion of detection limits. The selection of one or the other will depend on the form of final release criterion the measurement will be compared against, as discussed in the next section.

5 APPLICATION OF DERIVED CONCENTRATION GUIDELINE LEVEL AND CLEARANCE CRITERIA

The previous section described the concept and how to determine the MDA/MDC of a given radiation detection system. This section provides guidance on defining the release or clearance criteria for the purposes of decommissioning. As previously mentioned, the aim of this guide is to aid in the collection of a compressive dataset that quantifies the radioactivity present in building material at the time of construction. The dataset can be thought of as pre-decommissioning surveys that will eventually be utilized to support the decision for the final disposition of this material. Thus, for direct comparison, the MDAs/MDCs should be defined as they would be during the decommissioning process. Note that the equipment and detection technique used during the decommissioning phase does not need to be the same as that used during initial characterization, consistency in the output data is the only requirement.

The MARSSIM Manual [9] describes the concept of a Derived Concentration Guideline Level (DCGL) as it relates to a clearance criterion. The DCGL either derived based on analyses or obtained directly from established regulatory limits. The term DCGL may be misleading since Canadian regulations specify clearance criteria in terms of an activity (Bq) or an activity concentration (Bq/g) depending on conditions that will be discussed below. As a result, this guide utilizes the term clearance criterion which is analogous to the DCGL concept described in MARSSIM. Based on Reference [9], achieving an MDA/MDC that is between 10%-50% of the release criterion is considered good practice.

The Nuclear Substances and Radiation Device Regulations (NSRDR) [4] define the criteria under which a regulatory control of a nuclear substance is not required or no longer required. These are provided in terms of activity, activity concentrations, or dose limits that cannot be exceeded at any one time. The criteria are defined as:

- Exemption Quantities (activity or activity concentration);
- Unconditional Clearance Levels (activity concentration); and
- Conditional Clearance Levels (dose limits for deriving activity concentrations).

At their core, all three criteria are derived (using pathways analyses) based on the same dose limits, namely an annual effective dose to the public of 10 μSv (1 mrem) for realistic scenarios or an annual effective dose of 1 mSv (100 mrem) for low probability scenarios.

At this point, it is important to define two key concepts:

- **Exemption** refers to materials that are exempt from regulatory control when the expected risk to individuals is sufficiently low (i.e. a licence is not required); and
- **Clearance** refers to the process of confirming, through monitoring, that a material or substance, that is under regulatory control, can be removed from further regulatory control (i.e. license no longer required).

Decommissioning strategies are developed around the concept of clearance (i.e. removing material that was once under regulatory control from any future control). While all three criteria listed above are used to establish clearance levels depending on the situation, exemption quantities also apply to the concept of exemption (i.e. those materials and substances that do not require a licence to begin with). Knowing which criteria apply to a given situation requires understanding the difference between the three criteria, the quantity of material being released and the final disposition pathway. The three clearance criteria are described below and Figure 5-1 summarizes when a given criteria can be used. Additional guidance on the exemption and clearance of material that contain nuclear substances is provided in CSA standard N292.5 [5].

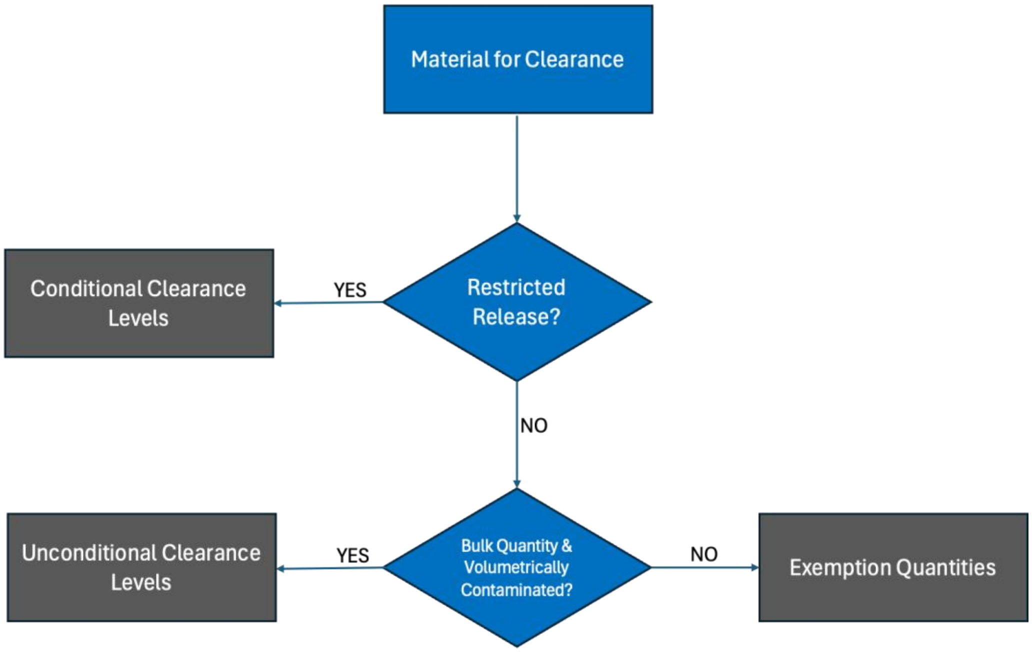


Figure 5-1: Selection of Clearance Criteria

5.1 Exemption Quantities

Materials are considered exempt if their radionuclide inventory is sufficiently low as to not warrant regulatory control. That is, their activity or activity concentration is below the threshold that requires a licence. Schedule 1 of the NSRDR [4], lists exemption quantities specified in terms of the total activity (Bq) or activity concentration (Bq/g) per radionuclide.

For clearance purposes, exemption quantities can be applied to moderate, non-bulk, quantities of material (i.e. <1000 kg per year per facility) that are being released from regulatory control. Furthermore, it is recommended that activity concentration (Bq/g) values be used when the radioactive material is uniformly distributed within a material, while total activity (Bq) values can be used for both non-uniformly and uniformly distributed radioactive material.

5.2 Unconditional Clearance Levels

Unconditional clearance levels, listed in Schedule 2 of the NSRDR [4], are specified only in terms of an activity concentration (Bq/g). They are generally more restrictive than exemption quantities and are used when releasing bulk quantities of material (i.e., >1000 kg or > 1000 kg per year per facility) with uniformly distributed radionuclides. The unconditional clearance levels may also be used for moderate (i.e., non-bulk) quantities since that would apply a greater degree of conservatism, however they do not apply to surface contaminated objects (whose release criteria are typically defined in the CNSC issued license or based on CNSC REGDOC 1.6.1 [17]).

5.3 Clearance Criteria for Mixtures

In practical terms, materials being considered for clearance from Class 1 Nuclear facilities will contain more than one nuclear substance. In such cases, a clearance criterion for the mixture should be determined and compared against for compliance. The rules for dealing with mixtures applies to the use of both exemption quantities and unconditional clearance levels. However, it is important to make the distinction between radionuclides of artificial origin versus those of natural origin since their respective clearance criteria are determined slightly differently, as described in the following sections. To that end, a mixture containing both artificial and naturally occurring radionuclides would need to be evaluated independently against both criteria for compliance.

5.3.1 Clearance Criteria for Mixtures of Artificial Radionuclides

The clearance criterion for a mixture of artificial radionuclides is calculated as shown below. Note that for radionuclides with progeny in secular equilibrium, only the parent radionuclide should be included in the calculation:

$$CC_{mix} = \frac{\sum_{i=1}^{n} A_i}{\sum_{i=1}^{n} \frac{A_i}{CC_i}} \qquad \textbf{Eq. (4)}$$

Where, for a mixture containing n radionuclides of artificial origin

CC_{mix} = clearance criterion for the mixture. Note that this can be in terms of an exemption quantity or an unconditional clearance level, as applicable

CC_i = clearance criterion of the i^{th} radionuclide. Note that this can be either an exemption quantity (Sch 1 of NSRDR) or an unconditional clearance level (Sch 2 of NSRDR)

A_i = total activity (Bq) or activity concentration (Bq/g) of the i^{th} radionuclide within the mixture

It is important to note that the units of A_i and CC_i must be the same.

Compliance with the clearance criterion is calculated by comparing the sum of the mixtures with CC_{mix} [5] as shown below:

$$\sum_{i=1}^{n} A_i \leq CC_{mix} \qquad \textbf{Eq. (5)}$$

Alternatively, the sum of fractions may be used, as shown below, without having to explicitly compute CC_{mix}:

$$\sum_{i=1}^{n} \frac{A_i}{CC_i} \leq 1 \qquad \textbf{Eq. (6)}$$

5.3.2 Clearance Criteria for Materials Containing Natural Radionuclides

Clearance criteria for material containing naturally occurring radionuclides are treated differently and the rules of mixtures described above do not apply. For substances described in Section 3.1 the activity or activity concentration of each nuclear substance is compared against their respective exemption quantity or unconditional clearance level as applicable [5]. Compliance with clearance criteria is therefore given by the following:

$$\frac{A_i}{CC_i} < 1 \qquad \textbf{Eq. (7)}$$

It is important to note that while the CSA N292.5 [5] makes no mention of ^{40}K and only includes ^{232}Th, ^{235}U, ^{238}U, and their radioactive progeny, IAEA GSR part 3 [18] (Table I.3) does include ^{40}K as a radionuclide of natural origin. Since the production of ^{40}K is not associated with the operation of a nuclear facility, it should be treated as a radionuclide of natural origin.

5.4 Conditional Clearance Levels

While unconditional and exempt clearance correspond to unrestricted release of material, for conditional clearance details related to material type and final disposition must be known and defined. This can be thought of as a restricted release where only specified configurations and conditions would apply. Thus, conditional clearance criteria are based on only a limited number of reasonably possible exposure pathways. A common misconception is that the storage of conditionally cleared material still requires a license. However, the NSRDR [4] (section 5 and 5.1) include material that meets conditional clearance levels as either exempt from a license or for disposal or abandonment without a license.

Explicit values do not exist for conditional clearance since each scenario would be different. Clearance criteria must therefore be derived using exposure pathways analyses based on the dose limits previously mentioned, namely a whole body effective dose of 10 uSv/year for realistic scenarios and 1 mSv/y for low-probability scenarios. An example of such pathways analysis to establish conditional clearance levels for building materials is outlined in U.S. Nuclear Regulatory Commission Report NUREG-1640 [19]. Typically, derived conditional clearance levels are reviewed and accepted by the CNSC. An example found on the CNSC website, and shown in Annex D, is “Conditional clearance levels for the disposal, recycling and reuse of activated medical accelerator components”, in which the conditional clearance level was defined in terms of a dose rate at a specified distance.

6 SELECTION OF ANALYSIS TECHNIQUES

Radiation levels present in the environment are generally low and therefore require detection methods with a high degree of sensitivity and low detection limits for accurate identification and quantification. The goal of the analysis to determine the radioactive inventory of a given sample or sample area including the individual radionuclides that are present. This may be in terms of a count rate (e.g., CPM), an activity (Bq), or a concentration of either (e.g., CPM/cm^2 or Bq/g or Bq/cm^2) per radionuclide. The methods of detection are divided into two categories:

- Radiometric Techniques: These techniques utilize the radiological properties of the material for detection and quantification. These include the type of radiation emitted (α, β, ɣ), energy of radiation, and any secondary radiation (ionizing or non-ionizing) from interaction with matter; and
- Non-Radiometric Techniques: These techniques rely on the physical properties of the material, such as mass, for detection and quantification.

This section presents a summary of common analysis techniques. Annex B provides a high-level overview of these techniques and discusses their underlying principles.

The selection of the appropriate analysis method(s) depends on a number of factors including:

- Cost;
- Timeline;
- Ease of implementation;
- Availability of equipment;
- Availability of experienced personnel; and
- Availability of an accredited lab.

The various analysis methods, their application, usage, speed of results, and cost category are summarized in Table 6-1. In the most ideal case, having no information regarding the radioactive inventory of the material, the most accurate identification and quantification of radionuclides would be obtained by performing gamma spectrometry, alpha spectrometry, and liquid scintillation counting on the sample(s) collected. In practical terms, however, this is neither cost effective nor timely.

Having some information about the source and type material as well as performing a cost benefit plan and defining clear data quality objectives, selection of analysis methods can be narrowed down. For common building materials such as concrete and steel, in most practical cases, the method of choice will likely be gamma spectrometry. With its ability to identify and quantify both NORM and artificial sources of radiation, relative ease of use either in the field or lab, and non-destructive nature, gamma spectrometry can provide near-seamless integration into existing testing requirements.

Table 6-1: Summary of Analysis Methods and Applications

Analysis Method	Ideal Application	Usage	Speed of Results	Cost
Gamma Spectrometry	Detection of gamma emitting isotopes including NORM and artificial radionuclides. U and Th Series ^{40}K, ^{60}Co, ^{137}Cs, ^{226}Ra, ^{210}Pb	Field or Lab	Several hours for field	Inexpensive
Alpha Spectrometry	Detection of Transuranic Alpha Emitters and NORM.	Lab	Several Weeks	Expensive
Liquid Scintillation Counting	Detection of pure beta emitters. ^{3}H, ^{14}C	Lab	Hours or Days	Moderate
Gross Alpha/Beta Counting	Detection of alpha/beta radiation, no radionuclide identification capability	Field or Lab	Immediate	Inexpensive
X-ray Fluorescence (XRF)	Detect elemental composition down to several ppm. ^{238}U and ^{232}Th	Field or Lab	Immediate	Inexpensive
Mass Spectrometry	Detect elemental composition. ^{238}U and ^{232}Th	Field or Lab	Immediate	Moderate
Note: Lab analyses can take several days to weeks depending on sample transportation, sample preparation, and lab workloads.				

7 SAMPLING

7.1 Background Activity

CSA N292.5 [5] defines *radioactive material* as any material, equipment, or system component determined or suspected to contain a radioactive nuclear substance in excess of radioactivity that occurs naturally in the material. The clearance criteria presented in the NSRDR [4] are specified for the total activity of radioactive substances irrespective of their natural or artificial origin. Section 5.3 describes how mixtures of artificial and natural radionuclides are treated with respect to material clearance.

In practice, the gross activity (total) and the net activity (incremental) have been compared with the clearance criteria in Canadian regulations. CSA N292.5 clause 7.2.1 [5] allows both. There is some operating experience that for NORM, it is simpler to directly compare the measured activity concentration to the release criteria without attempting to subtract a background. In cases where activity levels are close to release criteria, regulatory consultation and concurrence on a particular approach may be required.

7.2 MARSSIM Data Requirements

This section summarizes the guidance in the MARSSIM methodology [9] and quotes the relevant sections of the MARSSIM document.

NUREG-1505 [23] presents two methods of survey design, depending on the type of measurements available:

1. For radionuclide specific measurements of radionuclides that do not appear in natural background, it is clear that no background subtraction is needed.
2. If the radionuclide appears in the natural background, or even if the radionuclide does not appear in background but gross activity or exposure rate measurements are used, then background subtraction is needed.

7.2.1 No Background Subtraction

When individual samples do not require background subtraction, the results of the survey unit measurements can be compared directly to the clearance criteria, since each is a measurement of the residual radioactivity alone. Because only one set of measurements is involved in this comparison, the statistical test is called a one-sample test.

The MARSSIM [9] Sign test is a one-sample, nonparametric statistical test used to determine compliance with a release criterion (Weighted Derived Concentration Guidance Level, or $DCGL_W$)

when the radionuclide of concern is not present in background. The data needed for this test is a set of measurements in a survey unit that will be compared to a fixed value ($DCGL_w$).

Table 7-1: Radionuclide not in background

Survey Result	Conclusion
All measurements below $DCGL_w$	Survey unit meets release criterion
Average above $DCGL_w$	Survey unit does not meet release criterion
Otherwise (some measurements above $DCGL_w$, but average below $DCGL_w$)	Conduct Sign Test and elevated measurement comparison

Examples of measurements where this would apply include:

- Tritium measurements since natural tritium levels in construction materials are negligible;
- Gamma spectroscopy measurements of radionuclides that are not naturally present in construction materials; and
- Alpha spectroscopy for alpha emitters that are not naturally present in construction materials.

7.2.2 Background Subtraction

When individual samples require a background subtraction, the survey unit measurements cannot be directly compared to the DCGL, since each is a measurement of the total of any residual radioactivity plus the survey unit background. In this case, the measurements in a survey unit must be compared to similar measurements in local reference areas that have been matched to the survey unit in terms of geological, chemical, and biological attributes, but which have not been affected by site operations. The distribution of the measurements in a survey unit is compared to the distribution of background measurements in a reference area. Because two sets of measurements are used in making this comparison, the statistical test is called a two-sample test.

The MARSSIM Wilcoxon Rank Sum (WRS) test is a two-sample test that compares the distribution of a set of measurements in a survey unit to that of a set of measurements in a reference area (i.e., background area). The MARSSIM WRS test is used to test whether the true median in a survey unit population is greater than the true median in a reference area population. The data needed for this test is a set of measurements in the reference area (collected before commissioning) and a set of measurements in the survey unit.

Table 7-2: Radionuclide in background and gross measurements

Survey Result	Conclusion
Difference between maximum survey unit measurement and minimum reference area measurement is below $DCGL_w$	Survey unit meets release criterion
Difference of survey unit average and reference area average is above $DCGL_w$	Survey unit does not meet release criterion
Otherwise (maximum difference above $DCGL_w$ but average difference below $DCGL_w$)	Conduct WRS Test and elevated measurement comparison

Examples of measurements where this would apply include:

- Gross alpha, beta, gamma count rate;
- Dose rate measurements;
- Carbon-14 measurements (particularly in organic materials);
- Gamma spectroscopy measurements for radionuclides that are naturally present in construction materials (^{40}K, ^{137}Cs); and
- Alpha spectroscopy for alpha emitters that are naturally present in construction materials (^{238}U, ^{235}U).

The reference area is where background is measured and defined for the purpose of decommissioning. To minimize systematic biases in the comparison, the same sampling procedure, measurement techniques, and type of instrumentation should be used at both the survey unit and the reference area. This creates a challenge for a decommissioning survey that may take place 60 to 100 years in the future.

The distribution of background measurements in the reference area should be the same as that which would be expected in the survey unit if that survey unit had never been contaminated.

7.2.3 Data Quality Objectives and Reporting Requirements

In MARSSIM [9] statistics, all quantities are calculated relative to the variability of the measurements. The greater the variability, the more survey measurements are required to make a decision. There is therefore an incentive to reduce the variability.

The variability in the measurement data is a combination of the inherent variability of the measurement process (for example Poisson statistics), and the real spatial variability of the quantity being measured in the survey unit. Variability can be reduced by using more precise measurement methods, such as increasing the count time, but the spatial variability remains. The mechanism by which spatial variability can be reduced is by choosing survey units that are as homogeneous as possible with respect to the expected level of residual radioactivity. This means that survey units should generally be formed from areas with similar construction, use, contamination potential, and remediation history.

MARSSIM [9] recommends that the detection limit has to be below the $DCGL_w$ since the variability of the measurements increases near the detection limit. If the detection limit of the measurement method is close to the $DCGL_w$, the number of samples needed to make a decision may be prohibitively large.

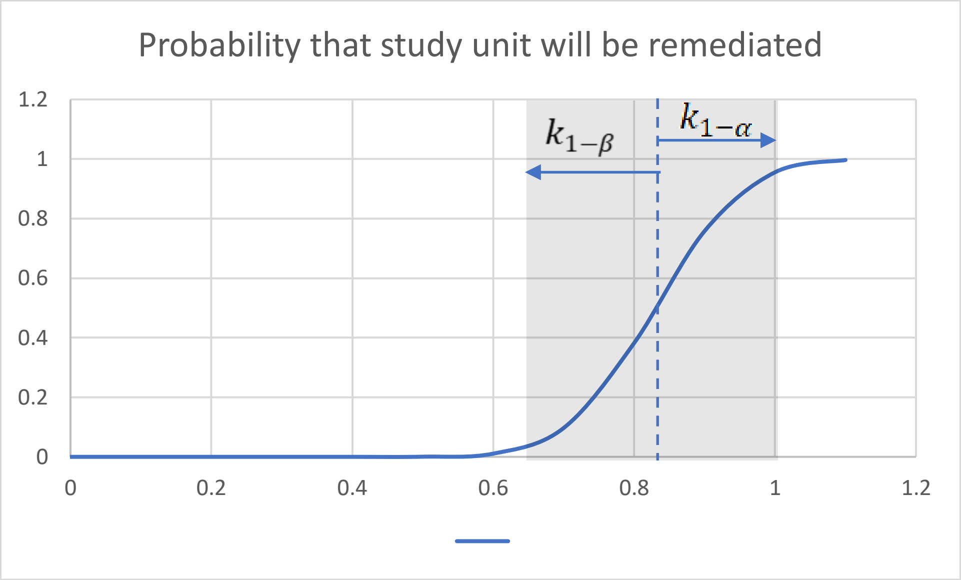
For survey measurements at or near background, there may be some measurement data which are at or below instrumental detection limits. These data points are sometimes reported as "less than" or "non-detects". Such data are not easily treated using nonparametric methods. It is recommended that the actual numerical results of measurements always be reported, even if these are negative or below calculated detection limits.

7.2.4 Number of Samples Based on MARSSIM Methodology

Experienced decommissioning practitioners who use MARSSIM [9] know that the number of samples required in a survey unit is usually around twenty. There is a logic to this number since a false positive rate of 5% corresponds to a one in twenty chance, given that no other assumption is made regarding the parametric distribution of the measurements.

Nevertheless, it is useful to review how MARSSIM [9] determines the number of samples required to make a decision under Scenario A. This scenario tests the null hypothesis that the survey unit does not meet the release criterion, i.e., it assumes that the survey unit is comparable to a reference area with uniform contamination at the $DCGL_w$ level.

1. Conduct surveys to determine the variability of the measurements in the reference area and the survey unit. The purpose of these pre-surveys is to calculate the standard deviation of the measurements;
2. Determine $DCGL_w$ in the correct units for the measurement method;
3. Set the Lower Boundary of the Gray Region (LBGR) to a concentration value between zero and the $DCGL_w$. The logic here is that for all levels between LBGR and $DCGL_w$, the probability that remediation will be required increases from β at LBGR until it reaches $1 - \alpha$ at $DCGL_w$. There is considerable freedom to set this value, however the closer it is to $DCGL_w$, the more samples will be needed, so in theory, there is an incentive to select a value closer to zero. However, MARSSIM recommends that LBGR be set at a concentration level such that remediation above the LBGR level is considered reasonably achievable;



4. Specify the false negative probability β at the LBGR activity concentration. This is the survey unit incorrectly fails the statistical test when the true condition is that it meets the release criterion. This is conventionally set as β=0.05;
5. Set the false positive rate α at the DCGL$_w$ activity concentration. This is the probability that the survey unit passes the statistical test when the true condition is that it does not meet the release criterion. This is conventionally set as α=0.05; and
6. Calculate the number of samples required to achieve the specified confidence level.

The difference the mean of the reference measurements and the study area measurements is calculated in two different ways:

First using the observed rank of the Wilcoxon distribution:

$$\left(\frac{\bar{x}-\bar{x}_0}{\sigma_{x0}}\right)^2 \approx 3N(p-0.5)^2 \qquad \textbf{Eq. (8)}$$

Where,

N = number of measurements
p = probability that the difference between the means is less DCGLw

Second, using the Gaussian approximation:

$$\left(\frac{\bar{x}_0-\bar{x}}{\sigma_{x0}}\right)^2 = \left(z_\alpha+\rho z_\beta\right)^2 \qquad \textbf{Eq. (9)}$$

$$1-\alpha = P(x < x_c|H_0) \qquad \textbf{Eq. (10)}$$

$$1-\beta = P(x > x_c|H_a) \qquad \textbf{Eq. (11)}$$

Where,

z_α = coverage factor for the probability α

H_0 = hypothesis is that the survey unit is comparable to a reference area with uniform contamination at the DCGL$_\mathrm{w}$ level

z_β = coverage factor for the probability β

H_a = alternative hypothesis that the that the survey unit is comparable to a reference area with uniform contamination at the LBGR level

It is also assumed that the standard deviation of the reference area and the survey unit are the same:

$$\rho = \frac{\sigma_x}{\sigma_{x0}} = 1 \tag*{Eq. (12)}$$

Using the fact that these two equations for the difference of the means are equal, then:

$$N = \frac{\left(z_{1-\alpha} + z_{1-\beta}\ \right)^2}{3(p_r - 0.5)^2} \tag*{Eq. (13)}$$

The subsequent step consists of taking the specified number of measurements in the survey unit and performing the statistical test to determine if the survey unit meets the release criteria.

One disadvantage of the Neymann-Pearson decision theory used by MARSSIM is that previous measurements cannot be used. In Classic statistics, once a decision rule has been selected, each new set of measurements is a sample from an imaginary population. If more measurements are taken, they represent another sample from this population. If the variability of the reference unit or the survey unit is higher than was previously estimated, the number of samples might not be sufficient to make a decision at the stated confidence level and the whole process must be repeated with another set of measurements.

7.2.5 Number of Samples Using Bayesian Methodology

In contrast to classic statistics, the Bayesian theorem offers a methodology for refining a decision as more data accumulates. This means that previous measurements are not discarded, but are included in the data set each time new measurements are taken. This is a major advantage when acquiring and analysing samples is expensive.

In addition, it is not necessary to pre-determine the number of samples. More samples are accumulated until the data supports a decision, one way or the other.

There are several Bayesian non-parametric methods that can be used instead of the Wilcoxon Rank Sum (WRS) test or the sign-test, while keeping most of the excellent features of the MARSSIM [9] process.

The book by Chechile [24] describes Bayesian non-parametric statistical methods that are equivalent to the MARSSIM WRS or sign-test. The following example shows how the Bayesian Mann-Whitney statistics can be used to make the same type of decisions as the WRS method.

The Bayesian methodology requires about the same number of samples as the Classic statistics methodology since it has roughly the same statistical power. Its advantage is related to the efficient use of the data. Chechille describes the number of samples needed to detect an effect size $\Delta = |\Omega_s - 0.5|$ with a probability $1 - \alpha$. Assuming that the binomial distribution can be approximated with the gaussian approximation, the cumulative probability distribution corresponding $1 - \alpha$ is given by:

$$\Delta - k_{1-\alpha} \sqrt{\frac{(0.5 - \Delta)^2}{n + 3}} > 0 \qquad \textbf{Eq. (14)}$$

Which gives the approximate number of samples:

$$n > \frac{k_{1-\alpha}^2 (0.5 - \Delta)^2}{\Delta^2} - 3 \qquad \textbf{Eq. (15)}$$

The effect size is related to the difference between LBGR and DCGL$_w$ and the dispersion of the measurements, as was the case with the WRS method. A few tests show that a value of $0.1 < \Delta < 0.15$ is generally adequate.

The null hypothesis will be that the reference area, augmented by DCGLw, is the same as the survey unit. The null hypothesis is rejected if the probability that the two survey units are the same is less than $\alpha = 0.05$.

Let's assume that the measurements in the reference area are described by the variable X_R and those in the survey area by X_S. The probability density at point x is given by $f_R(x)$ and $f_S(x)$, while the cumulative probability distribution is given by $F_R(x)$ and $F_S(x)$.

The probability that the measurements in X_S are greater than X_R is given by:

$$P(X_S > X_R) = \Omega_S = \int f_R(x)\big(1 - F_S(x)\big)dx = \int f_S(x) F_R(x) dx \qquad \textbf{Eq. (16)}$$

When $X_S = X_R$, Ω_S=0.5.

Using the statistics of Mann-Whitney, Chechile starts by calculating the number of times a measurement in the survey unit is greater than a measurement in the reference unit, augmented by DCGCLw (the measurements don't need to be sorted since each reference area measurement is compared to all survey unit measurement).

$$U_S = \sum_{i=1}^{n_S} \sum_{j=1}^{n_R} \delta_E(i,j)$$ **Eq. (17)**

Where,

$\delta_E(i,j)$ = 1 when $x_{Rj} < x_{Si}$

$\delta_E(i,j)$ = 0 when $x_{Rj} \geq x_{Si}$

Chechile devised a method to calculate the posterior probability distribution of Ω_S, given the measurements $x_{Ri} + DGCLw$, x_{Si}. He obtains the cumulative probability distribution for Ω_S and verifies that the probability $P(\Omega_S < 0.5) > 1 - \alpha$.

For this example, the set of measurements (in relative units) for the reference area and the survey unit is presented in Table 7-3. The value of DCGLw is set to 1000 above background. Initially, 5 measurements were taken in both areas.

Table 7-3: Set of measurements for the reference area and the survey unit

Reference Area	Reference Area + DCGL$_w$	Survey Unit
1793	2793	1984
1102	2102	2053
2565	3565	11
900	1900	400
2697	3697	3026
		1550

The value of U_S=5 and the maximum value is $U_S = n_S \times n_R = 25$. The probability that $\Omega_s < 0.5$ is 0.93, read off the vertical axis for a value of $\Omega_S = 0.5$ on the horizontal axis, therefore the null hypothesis is not rejected.

The surveyor takes another measurement in the survey unit (x_{S6} =1550) and adds it to the Mann-Whitney statistics. The value of UE=5 and now the maximum value of $U_S = 30$. The probability $P(\Omega_S < 0.5) = 0.958$, therefore the null hypothesis is rejected. The survey unit can be cleared.

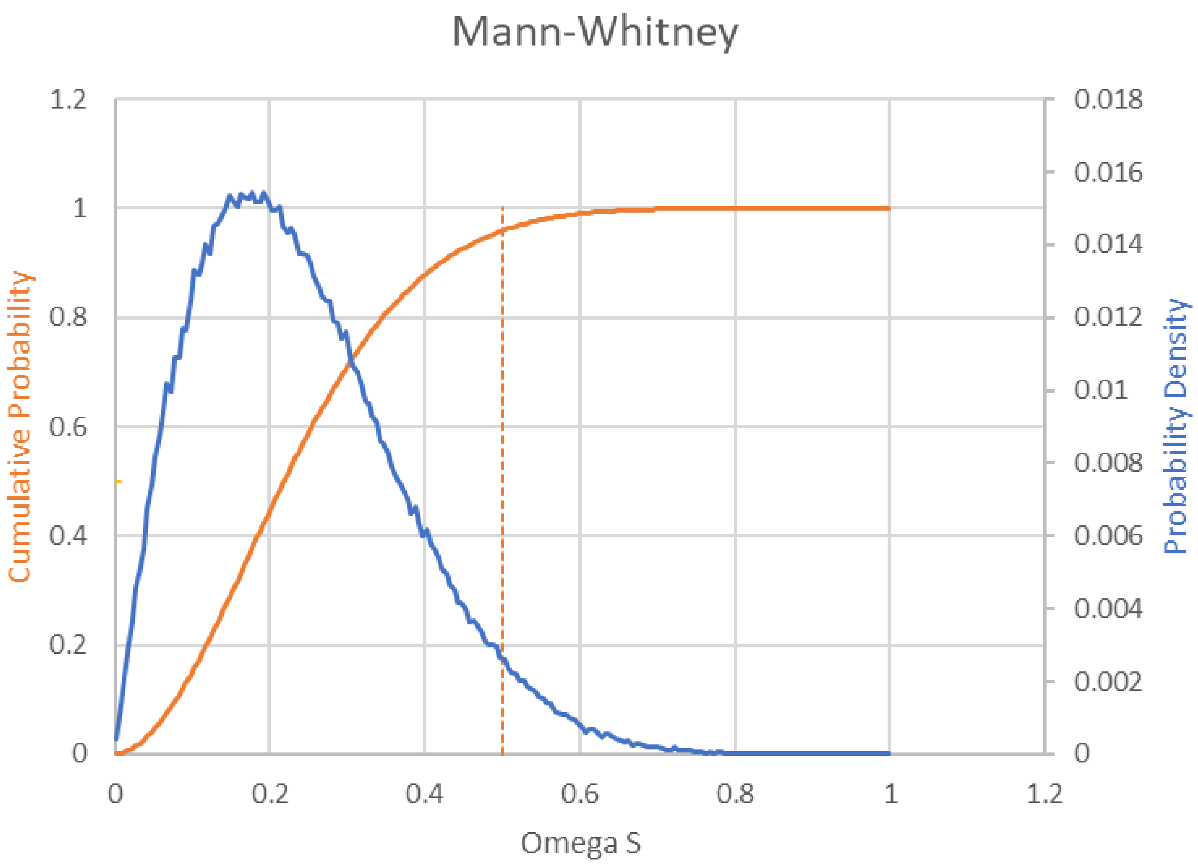


Figure 7-1: Probability distribution for Omega-S

The VBA code for Excel is presented in Annex E.

8 BUILDING MATERIALS (CONCRETE AND STEEL)

Concrete and steel are the two primary building materials in nuclear construction. Applications for concrete range from foundations and footings to containment structures and shielding walls. Similarly, applications for steel range from rebar and structural members to piping and equipment. Combined, the two materials account for about 90% of the materials used in nuclear construction. This section presents and overview of these two critical building materials. The discussion provides a high-level overview on the supply chain, main inputs and recommended sampling points for each.

The aim of sampling for baseline radionuclide inventory is to not produce a significant analysis burden on the constructors through the introduction of numerous radiological tests. Rather, this guidance aims to integrate radiation sampling and monitoring into the various tests that are already conducted to meet existing codes standards. In this sense, the objectives of the baseline characterization surveys diverge from those of facility decommissioning.

8.1 Concrete

8.1.1 General

Globally concrete is the most widely used building construction material. It’s usage dates back to several hundred years BC and it has been the foundation of civilization. Concrete is made up of three primary ingredients, these are:

- Aggregate – which consists of sand, gravel and crushed stone;
- Cement – a binding agent that hardens and adheres to other material; and
- Water.

When these three ingredients are mixed, the result is a fluid paste like material that is referred to as the plastic state of concrete. It is in this state that concrete can be formed or molded into essentially any shape. This is the key characteristic of concrete that has resulted in its extensive use as a building material. Once cured, the fluid paste is transformed into a durable stone like material. The key to achieving a durable structure is to ensure that the proportions of the primary ingredients are correct for the desired application. By itself, concrete has exceptional compressive strength but weak tensile strength. It is therefore often used in conjunction with steel rebar to enhance its tensile properties and make it more resistant to cracking or breaking.

8.1.2 Supply Chain

The supply chain for concrete is built around the acquisition and mixing of the three primary ingredients. An overview of the supply chain as well as recommended points for radionuclide sampling is shown in Figure 8-1 and described below.

The cement supply begins with the extraction of raw materials including limestone, clay and sand. These materials are transported to the manufacturing plant for cement production. At the plant, the raw materials are crushed, ground and blended with additional minerals in defined proportions depending on the type of cement being manufactured. The materials are then heated in a kiln at temperatures up to 1470 °C, which transforms the materials into a molten product called clinker. The clinker is rapidly cooled and finely ground. Gypsum along with other cementing products such as fly ash and slag are added to produce the final cement product with the desired properties. The addition of these supplementary products, particularly slag and fly ash, tends to increase NORM concentrations in cement. The finished product is packaged and shipped to distributors across the country or overseas. Cement plants are typically located near raw material deposits, such as limestone quarries, to minimize the transportation cost associated with cement production. In Canada there are only a handful of manufacturing companies, and cement production is concentrated in the central region with Ontario and Quebec responsible for 50% and 17% of the Canadian industry respectively [25]. In addition, Cement is also imported into Canada from the United States, Turkey and Vietnam.

Unlike the cement supply chain, the aggregate supply, which consists of sand, gravel and crushed stone, is typically more local to the construction site. The extracted raw materials undergo crushing and screening to achieve the desired grade (size distribution) of the aggregate. The final step is washing and sorting. NORM concentrations in aggregate therefore depend on the local geology of a given quarry. Several aggregate plants also receive material from concrete recycling facilities that has been crushed and screened. This stream is blended with the extracted aggregate at the screening stage and may be a part of the final aggregate mix. Aggregate can be sourced from several quarries in relatively close proximity to the construction site based on supply and demand.

The final step in concrete production takes place as the batching plant where the three primary ingredients are mixed along with chemical additives to achieve the desired properties of the concrete. There are several types of batching plants, ranging from stationary long-term plants to compact mobile plants, that are deployed based on the needs of a given project. Stationary plants are high-capacity and high-efficiency facilities that are located at the construction site for large scale projects such as dams, bridges, and nuclear power plants. The concrete produced is transported to the job site via mixers or mixing trucks.

8.1.3 Sampling Points

Since concrete is a manufactured product, there are a number of quality assurance tests and evaluations that are required to be carried out to produce a uniform, high-quality product. These sampling and testing requirements are outlined in CSA A23.2 [21]. Each type of test is used to assess a different characteristic of the concrete batch sample and to determine its compliance with the standards. For nuclear safety related structures, CSA N291 [26] specifies higher frequencies of the tests described in CSA A23.2 [21]. For applications specific to containment structures of nuclear power plants, CSA N287.1 [22] provides further requirements and higher frequencies of tests. To leverage existing experience with sample collection at the various stages, two sampling points within the concrete supply chain are recommended. These are at the aggregate and the final concrete mix stages. A third in-field sampling point is also suggested after the final structure has been poured and cured. The sampling points are shown in Figure 8-1. While it is the radionuclide inventory of the final concrete mix that will be of use during decommissioning, sampling the aggregate may allow for the collection of information and potential screening of quarries with higher concentrations of NORM for SMR construction. On the other hand, radionuclide sampling within the cement supply chain will likely not provide beneficial data for decommissioning purposes due to the limited number of domestic and international suppliers as discussed in the previous section.

8.1.3.1 Sampling Aggregate

Aggregate is one component of the final concrete mix. Thus, concentrations of NORM measured in aggregate samples, which are dependent on the local quarry geology, may not be representative of NORM concentrations in the final concrete pour. Typical concrete mixtures consist of 1:2 or 1:3 proportions of cement and aggregate. Therefore, the purpose of radionuclide sampling at the aggregate stage is to establish NORM concentrations at a given quarry and potentially identify quarries with high levels of NORM, particularly above or near clearance levels, for the construction of buildings in future radiological (potentially impacted) areas. Radionuclide inventory testing in aggregates is therefore recommended at the pre-construction phase. When a potential construction site is selected, aggregate from local quarries may be sampled to assess NORM levels. The resulting data would factor into the decision to source aggregate from a given quarry. This may result in quarries with inherently low NORM concentrations catering to solely service the nuclear industry and potentially inflating costs. However, the result would ensure NORM clearance levels would be met thereby reducing the low-level waste footprint during decommissioning.

As outlined in CSA A23.2 [21], sampling at the aggregate stage is intended for the preliminary investigation of the supply source. A series of aggregate tests are outlined in the standard and are typically conducted on an annual basis. For the purposes of radionuclide analysis, samples collected must reflect the nature and condition of the materials they represent, that is they shall not be modified (e.g. by heating or by the addition of any foreign substances). From the list of aggregate tests, two test samples are considered viable for radionuclide analysis. These are tests A23.2-2A: Gradation of Aggregate and A23.2-10A: Bulk Density of Aggregate.

Test A23.2-2A involves assessing the grade, or particle size distribution, of the aggregate. The test is conducted with a series of sieves with progressively smaller openings. Its purpose is to separate the aggregate sample into its component size fractions. The mass of material contained in each individual sieve is used to determine the sample grade. Once the test has been completed and the mass of the sample known, it may be sent for radionuclide analysis.

Test A23.2-10A involves determining the bulk density of the aggregate. A calibrated m0065atal measure of known volume is filled with the test aggregate. The bulk density is determined by measuring the mass of the sample contained in the known volume. As with the previous test, once measurement is complete, the sample may be sent for radionuclide analysis.

8.1.3.2 Sampling The Concrete Mix

Sampling the final concrete mix provides a direct indication of the radionuclide concentrations in the construction of a given structure. As with aggregate, CSA A23.2 [21] defines a series of tests and minimum frequencies at which they are required for the final concrete mix. For nuclear safety-related structures and containment structures, CSA N291 [26] and CSA N287.5 [22] outline additional testing requirements respectively.

During concrete placement (pouring), test A23.2-3C describes the process for casting and curing specimens for subsequent compressive and flexural strength testing. Creating the cast specimen entails pouring a sample of the concrete batch into a standardized cylindrical mold (typically 100 mm in diameter and 200 mm in height). The collected samples are then cured for an initial period at site before being sent to a lab for subsequent curing and testing. Typically, three such cylinders are cast for each test with additional cylinders often cast for contingency purposes. The minimum testing frequency as per CSA A23.2 [21] is 100 m^3 or one set per day. Testing of the cylinders are then conducted at cure times of 7, 14, and 28 days. While significant efforts are made during sample collection to ensure a uniform and representative specimen, the compressive tests are ultimately destructive. This guide therefore makes the following sampling recommendations:

1. An additional cylinder be cast for radionuclide analysis. This minimizes the potential for cross-contamination and also simplifies the chain-of-custody of the samples. The additional specimen may be smaller if an appropriate mold is available, but sample collection and preparation should be the same as defined in test A23.2-3C; and
2. The location and use of the pour from which the sample is collected be tracked. For example, if the batch was utilized to form the foundation of a given building, or its walls, or footings. This allows for a direct comparison during decommissioning.

8.1.3.3 Measuring the Finished Structure

Final sampling of the completed structure may be conducted through the use of in-field techniques including portable gamma spectrometry and XRF analysis. This type of measurement is essentially akin to performing decommissioning-type surveys prior to the commencement of plant operations. Data gathered can be building or room specific and can serve as a direct comparison during decommissioning. The acquisition of samples (such as core drilling) at this stage of construction is not recommended since that would inherently be destructive to the newly built structure. Such sampling would likely require subsequent repairs and may risk meeting the required standards for commissioning and operation.

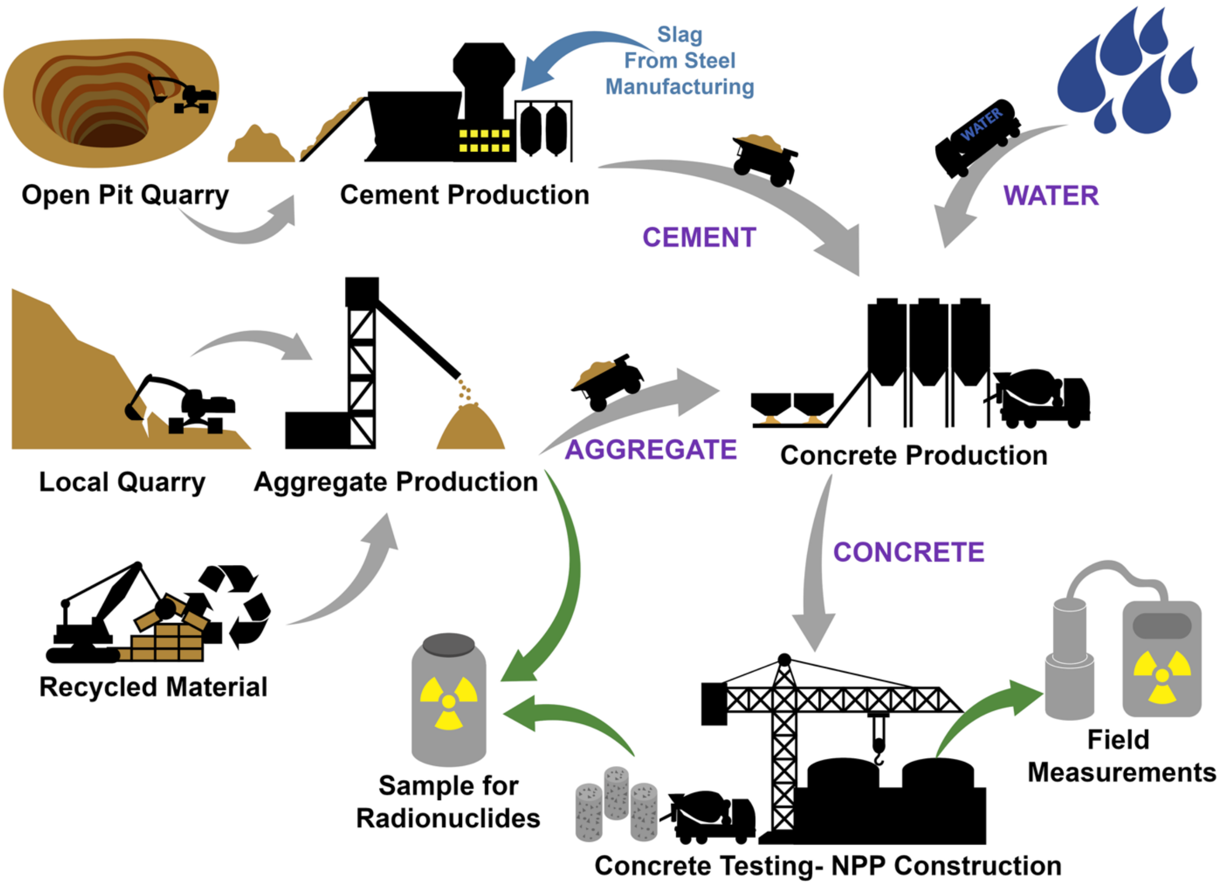


Figure 8-1: Concrete Supply Chain and Radionuclide Sampling Recommendation

8.1.4 Radionuclides of Interest

The main radionuclides of interest for raw materials found in cement and aggregate are NORM sources as described in Section 3.1, which consist of the ^{238}U and ^{232}Th decays series and ^{40}K. Concentrations of these radionuclides depend on the local geology of the quarry and can vary considerably from one location to another. Also, additives such as fly ash (a byproduct of coal fired plants) can significantly increase the NORM concentration. In addition to NORM, for recycled material, depending on the source, a number of artificially created radionuclides may be present. Therefore, having some information pertaining to the origin of recycled materials may necessitate the inclusion or allow for the exclusion of certain type of testing.

8.1.5 Analysis Methods

8.1.5.1 Gamma Spectrometry

With the primary aim of measuring NORM sources, a practical analysis method needs to be capable of detecting and quantifying ^{238}U and ^{232}Th (and corresponding daughter products) and ^{40}K. Gamma spectrometry is a suitable technique since it can readily detect the gamma emitting radionuclides within the two decay series as well as ^{40}K. The key gamma emitting radionuclides along with their dominant (highest intensity) gamma energies listed in Table 8-1 and Figure C-2, which is an example of NORM material (granite) with these gamma peaks present. For optimal resolution and accurate quantification a Hyper Pure Germanium (HPGe) gamma spectrometer is recommended.

Table 8-1: Gamma Emitting Radionuclides in Terrestrial NORM

Radionuclide	Decay Series	Main Gamma Energy (keV)	Half Life
^{234m}Pa	^{238}U	1001	1.2 min
^{226}Ra	^{238}U	186	1.6 years
^{214}Pb	^{238}U	352	27 min
^{214}Bi	^{238}U	609	20 min
^{210}Pb	^{238}U	47	22 years
^{228}Ac	^{232}Th	463, 911	5.7 years
^{212}Pb	^{232}Th	239	11 hours
^{208}Tl	^{232}Th	510.8, 583, 2614	3 min
^{40}K	N/A	1461	1.25E9 years

The unique feature of the uranium and thorium decay series is the gaseous phase radon isotope that is present in both series and can escape from the sample resulting in an underestimation of activity. Therefore, caution must be taken to ensure proper sample preparation techniques are employed when counting and quantifying these decay series. Sample preparation begins with the collection of a few hundred grams from the concrete or aggregate from the specimen provided. The sample is then ground to a fine powder and dried by heating to above 100 °C to remove the moisture content. At this point the sample is placed in a plastic container and sealed. This ensures that the radon progeny is retained within the container. Secular equilibrium with parent radionuclides is typically established within about 4 weeks after which point the sample can be counted with the gamma spectrometer. Software analysis is used to identify the gamma peaks and quantify the sample activity based on an established efficiency curve. The use of an accredited lab or university is recommended for accurate sample preparation and quantification.

Gamma spectrometry will also identify non-NORM gamma emitting species that may be present in the sample. Studies have shown that trace quantities of fallout radiation in the form of ^{137}Cs may be present in some concrete constituents, particularly fly ash.

8.1.5.2 X-Ray Fluorescence Analysis

A complementary technique that can be used in conjunction with gamma spectrometry is XRF analysis. The primary advantage of XRF analysis is its ability to quantify the uranium and thorium content directly. This eliminates the need for sample preparation to contain radon progeny, and the additional wait time for secular equilibrium to be established. These methods can be used with both powders or solids.

The results of the XRF analysis will likely be in ppm (parts per million) of uranium or thorium, which can then be converted to an activity concentration based on the following:

The specific activity, in terms of Bq/g, of a substance is given by

$$SA = F\frac{6.02 \times 10^{23}}{A}\lambda \qquad \textbf{Eq. (18)}$$

Where,

F = natural abundance fraction of a given radionuclide
A = atomic mass in g/mol
N_A = Avogadro's constant, 6.02 x 10^{23} in atoms/mol
λ = decay constat for the radionuclide in s^{-1}

The ppm conversion is then given by dividing the specific activity by $1x10^6$. The parameters for ^{238}U and ^{232}Th and corresponding ppm conversions are shown below in Table 8-2.

Table 8-2: Parts Per Million (PPM) to Specific Activity Conversion

Radionuclide	Abundance [F]	l (s^{-1})	SA (Bq/g)	Bq/g per ppm
^{238}U	99.3	$4.88x10^{-18}$	$1.2x10^4$	0.012
^{232}Th	100	$1.57x10^{-18}$	$4.1x10^3$	0.0041

The use of portable XRF, for the detection of U and Th, is in conjunction with portable spectrometry, for the detection of ^{40}K, can result in a completely in-the field NORM detection system that can be deployed after a structure has been poured. While this may not eliminate the need for sample collection and lab analysis, it can offer a more cost-effective option for the collection of additional data points. It is important to note that in-field measurements may not be able to achieve the necessary detection limits based on background levels. In such cases, lab analysis would be the only means of achieving data relevant to decommissioning. A potential application for such a system could be in cases where lab analyses show elevated concentrations of NORM, and greater spatial resolution within a building is desired. This would be more akin to a decommissioning survey that is carried out prior to the commencement of facility operations.

8.2 Steel

8.2.1 General

Steel is the backbone of modern society and its usage in the nuclear construction industry as a building material is second only to concrete. However, it's versatility in terms of composition, properties and strength to weight ratio sets it apart from all other building materials. In addition, steel's ability to be infinitely recycled and reused allows for the development of circular economies making it the ultimate construction material of the present and future.

Structural steel is an alloy of iron and other metals with essentially all its carbon content removed to result in a ductile yet durable material. There are many grades of steel, each with its own chemical composition and physical properties depending on the wide array of applications. To verify that a given quantity of steel meets the defined parameters of a grade or class, a mill test report (MTR) is generated. The MTR is a quality assurance document that shows a given product's compliance with the required standards for its grade. The MTR provides valuable information including the steel's mechanical properties, chemical composition, dimensions, and where the individual piece originated for traceability. MTRs are required for all structural steel, rebar, and welding consumables. For safety structures, a certified MTR is required where third party analysis has been conducted for independent verification of composition and properties.

8.2.2 Supply Chain

The steel supply chain, shown in Figure 8-2, begins with the mining and extraction of raw materials. There are three main raw material required in the manufacturing of steel, these are:

- Iron ore – In nature, most metals are found in ores which consist of the desired metal in oxidized states and mixed with other metals and impurities;
- Coal – Which will be converted to coke to serve as a reducing agent to remove the oxygen from the ore; and
- Limestone – Which will be used to remove impurities such as silica, phosphorus, and sulfur.

Extracted coal is transported to the coking plant where it is heated to 1200 °C in the absence of oxygen. This results in a purified porous rock-like material consisting almost entirely of carbon (about 92%) known as coke. In parallel, the extracted iron ore is sent to a sinter plant where the ore is heated to transform fine grain ore into a porous uniform substance.

The coke, iron ore, and limestone are then sent to a blast furnace. The purpose of the blast furnace is to purify the iron ore by melting it at a temperature of about 1300 °C. The coke has two purposes in the process. It provides the heat necessary to melt the ore and also scavenges the oxygen from the iron oxide to produce pure iron. The limestone serves to remove other impurities. The output of the blast furnace is liquid iron with a high carbon content (around 4%) referred to as pig iron. This form of iron is highly brittle and not particularly useful for construction applications. Two additional by-products are produced from the blast furnace; these are CO_2 and slag (left over impurities). Slag, which can consist of high concentrations of NORM, is diverted for use in road-fill or cement production while CO_2 is released as an effluent.

The next step in the process reduces the carbon content of the pig iron from about 4% down to about 0.5%. This is achieved in the basic oxygen furnace where the molten pig iron is poured into the furnace and high-purity oxygen is blown into the molten iron to combine with the carbon. This low carbon iron is referred to as steel. Depending on the grade required, further refinement of the chemical composition takes place. The by-products of this stage are CO, CO_2 and slag.

The molten steel is brought to a continuous caster where it is formed into standard intermediate shapes. During this stage, molten steel is continuously fed into a casting mold that starts vertically but curves into a horizontal position. Solid steel is discharged and cut into required lengths.

Another input to steel manufacturing process is the addition of scrap steel from recycling plants. Scrap steel can be added at various stages of process including the blast furnace and the basic oxygen furnace for temperature control. In addition, the scrap steel can also melted down in an electric arc furnace where high power electricity that produces an arc is used to melt the scrap steel. It can them be fed directly into the continuous caster and mixed with steel produced from ore (provided it’s the same grade).

The final step in the steel manufacturing process is hot and cold rolling to produce the various sizes and profiles of steel. Typical shapes include rolls, beams, plates, tubes, pipes and wire. Finishing steps such as galvanizing, annealing, and coating are applied as required prior to shipping.

The slag by-product that results from the various stages of steel production is a concentration of iron ore impurities including NORM. The slag produced in the blast furnace from smelting iron ore has almost 100% utilization due to its cementation properties. Once cooled, the blast furnace slag is ground and often mixed with cement and aggregates. On the other hand, slag produced in the basic oxygen furnace or electric arc furnace has lower utilization and often ends up as landfill waste or is used to a lesser degree in applications including aggregate base and asphalt. Slag therefore forms a crucial link between the two most widely used construction materials in the world.

8.2.3 Sampling Points

Based on the discussion above, much of the steel manufacturing process is conducted in high temperature environments that is not conducive to intermediate sample collection for the monitoring for radionuclides. Therefore, the most practical time for radionuclide monitoring would be at the end of the manufacturing process. Testing and verification for the MTR, which includes the chemical composition and physical properties, is also conducted at final stage of manufacturing. During construction, standard steel shapes are received and modified by cutting, drilling and welding to fabricate the final structure. Thus, monitoring for radionuclides is recommended at the material receipt stage when the steel shipment arrives at the construction site.

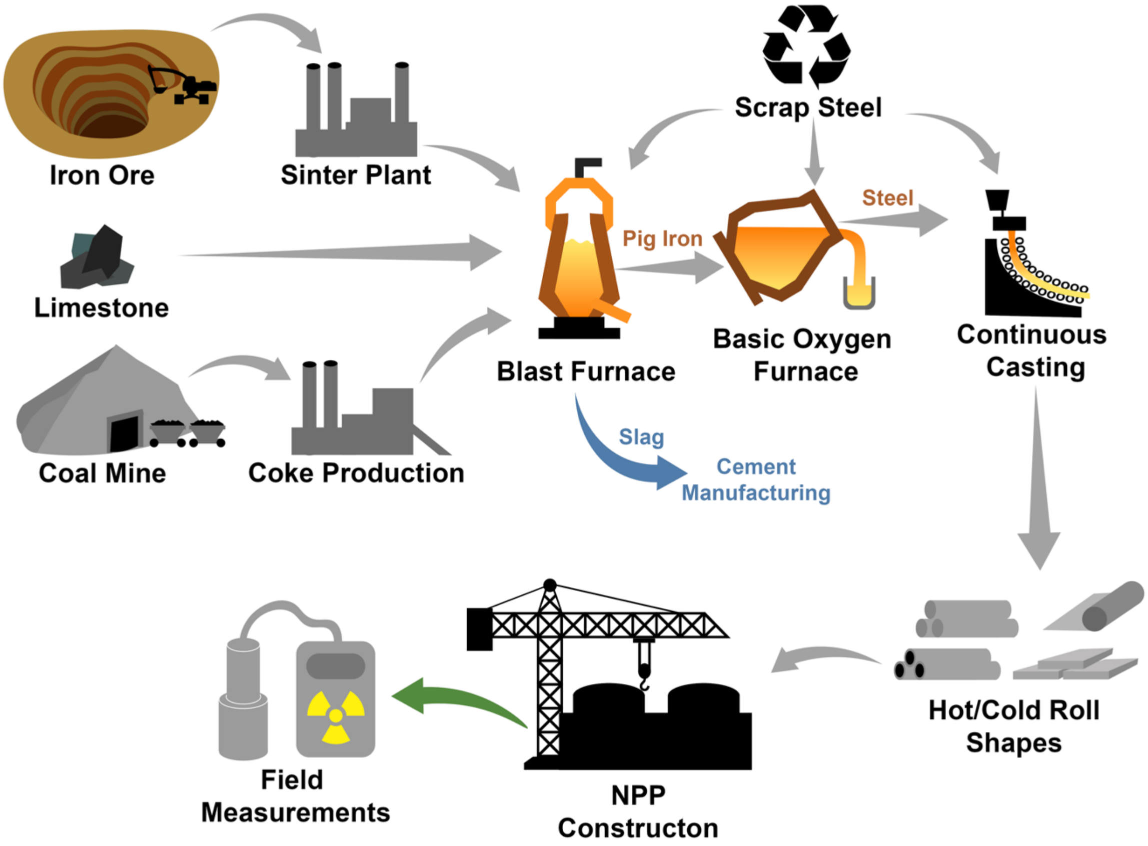


Figure 8-2: Steel Supply Chain and Monitoring Recommendation

8.2.4 Radionuclides of Interest

The steel manufacturing process removes most of the non-metallic impurities from the raw iron ore, which in turn removes sources of NORM by concentrating them in the slag by-product. Slag usage for the manufacturing of cement and concrete production can be thought of as essentially transporting NORM from the steel supply chain to the concrete supply chain. The resulting steel that is produced can be considered to be free of NORM.

Metallic impurities on the other hand are not entirely removed, and steel often contains trace quantities of cobalt, nickel, manganese, and other metals. Radioactive sources that that enter the recycling stream can therefore be incorporated into the final steel product. The primary radionuclide of concern is ^{60}Co. As discussed earlier in this report, there have been cases in the 1980s where significant quantities ^{60}Co have made their way into the steel manufacturing stream and subsequently into the construction of buildings.

8.2.5 Analysis Techniques

With ^{60}Co being the main radionuclide of concern, in-field gamma spectrometry is the recommended analysis technique since it can readily detect and quantify the two characteristic decay gammas. Depending on the activity present, typical in-field gamma spectrometry may require an acquisition time in the range of hours and performing such measurements on each steel component is likely not practical. Therefore, as a screening mechanism, it is recommended that incoming steel be surveyed with an environmental gamma meter (typically NaI) to assess the presence of any gamma emitting radionuclides. Readings above background may then be analyzed with a gamma spectrometer. In cases where radionuclides are detected, a record of the item(s) fabricated and location in the building be kept for direct comparison during decommissioning.

9 SURVEY PLANNING AND RECORDS

9.1 Data Quality Objectives & Survey Planning

This document recommends the use of the Data Quality Objective (DQO) process, outlined in Reference [27], to systematically plan and acquire the baseline radiological data in SMR building materials. The aim of the process is to ensure that the level of effort expended in the collection of data is commensurate with the resources available and ultimate use of the data. The output of the DQO process will be a survey plan(s) that balances project resources and the quality and quantity of data that will be collected with the ultimate goal of making a decision on compliance with clearance criteria during decommissioning. To that end, the survey plan may serve as an input to a cost benefit analysis. While the ultimate decision will be made 50 to 60 years after the data is collected, and may be impacted by many unanticipated factors not initially considered, the development of a systematic survey plan using the DQO process will clearly outline the validity and basis of the data acquired.

The DQO process consists of the seven iterative steps described below in Figure 9-1. If it is determined that the recommended plan will not meet the objectives, then input constraints (such as resource allocation or schedule) may need to be revised. An example of a survey plan developed with the DQO process is presented in Annex C.

Step 1 – State the Problem
Define the problem and the purpose of study. Include the following:
- Project team;
- Budget; and
- Schedule.

Step 2 – Identify the Goal
Define how data will be used to meet an objective. Define the decision that will be solved. Identify actions from each possible resolution outcome.

Step 3 –Identify Information Inputs
Identify the data and information needed to meet the goal of the study.

Step 4 – Define the Boundaries of the Study
Define spatial and temporal limits of the study so the data can be clearly interpreted.

Step 5 – Develop the Analytic Approach (Decision Rules)
Define the parameter of interest, determine action levels, develop logic for drawing conclusions from findings.

Step 6 – Specify Performance or Acceptance Criteria
Specify limits on acceptable limits on decision errors.

Step 7 –Develop the Plan for Obtaining Data
Identify the most resource-effective method for collecting data to meet the acceptance criteria.

Figure 9-1: Data Quality Objectives Process [27]

As the survey plan is executed and results become available, a preliminary review of the data is recommended to ensure that the DQOs as defined are being met. While there may not be any release related decisions that are made during pre-construction characterization, the review should include whether the required MDAs are being achieved and a comparison against clearance criteria. At this stage, changes to the plan may be required to ensure DQOs are being met.

Once all the data is collected and analysed, results and conclusions should be complied and documented in a baseline characterization report. This report should include the DQOs, describe the analysis techniques used, the applicable clearance criteria, and how the DQOs were met. It is recommended that the report also include a comparison against the applicable clearance criteria to unambiguously demonstrate the pre-operational condition of the building material being tested. Sufficient information should be included in the report to enable independent evaluation of the data and repeating measurements during decommissioning. Specifically, the characterization report should include the following:

- Identifying information of materials that were surveyed such as:
    - Maps, diagrams, photographs;
    - Sample identification and location numbers; and
    - Material identification numbers or batch numbers.
- Instrumentation utilized including:
    - Model and serial numbers;
    - Calibration certificates; and
    - MDAs.
- Tabulated results including backgrounds, gross and net values
- Analyses conducted and methods of data manipulation; and
- Original copies of lab analysis reports.

9.2 Records Retention

It is important, and a recognized best practice, that a data collection and Records Management System (RMS) be established early in the SMR development process to ensure appropriate data retention. The RMS ensures that records are legible, retrievable, accessible, secure and traceable during the lifetime of the facility. However, practical experience has been that records that are required at the decommissioning stage, especially those generated at the time of construction and commissioning, often can’t be located, or may be degraded and not legible when located. This challenge can be mitigated with a well-established RMS that includes choosing an appropriate record medium along with regular maintenance and assessments of the RMS focusing on data preservation and accessibility.

Records generated as part of the construction and commissioning of SMRs, including construction material testing procedures, reports, results and data, are considered permanent records and should be retained for the life of the facility (including decommissioning). Types of records that should be kept for decommissioning stage include:

- As built construction drawings;

- Photographs and video recordings (during construction and installation);
- Construction sequences and methods;
- Scale model of nuclear installation;
- Relevant details of calculations and material test certificates;
- Logbooks;
- Testing procedures (and photographs);
- Survey equipment used;
- Photo documentation logs;
- Survey plans and results;
- Quality control standards, procedures, & results; and
- Data spreadsheets.

Records generated during construction and commissioning may include material testing and results. To make sense of the information in these testing and results records over time, it's important to include details about how samples were taken, what equipment was used (including model and serial numbers), and how the analysis was conducted. Pictures (e.g., of equipment used) should be retained. In addition to spreadsheets of data, records of data in a simpler form (e.g., text) should be considered as a duplicate. This can mitigate readability issues with ever evolving software applications.

Records must be properly maintained (including the proper medium) to protect them from data corruption, deterioration, or inability to open encrypted and password protected files. They must also be protected from possible loss or destruction from disasters, changes in software and Information Technology (IT) infrastructure.

Records should be periodically checked to ensure their preservation and protection from loss, deterioration and destruction. Technology used to generate, view, and store a record can change significantly over time. Information is often saved in different electronic file formats throughout the lifetime of a facility which can lead to access and readability issues. This is mitigated by having well established and tested RMS.

Additionally, knowledge of the facility may diminish over time and information may not be available with a changing workforce. Therefore, it is important that permanent records be available to future users [28]. According to IAEA TecDoc 1097 [29], retrievability and usability of records is dependent on the continual migration or conversion to new technologies. However, this can be costly, time consuming, and potentially error likely. For example, some text converters can interpret the Greek letter “μ” as the Latin letter “m”. This can have significant consequences following conversion since the “μ” prefix designating micro (10^{-6}) would be translated to “m” prefix designating milli (10^{-3}) and all affected data would be presented as being a factor of 1000 higher (i.e. μSv become mSv). Therefore, early discussions with experts (such as engineers, data analysts and designers) to establish the medium of data storage could help with future proofing and potentially avoiding costly conversions to new technologies.

As outlined in IAEA TecDoc 1097 [29], the desired attributes of a robust data recording medium are listed below:

(a) Capable of capturing and storing the required information;
(b) Physically and chemically stable, so that the legibility is preserved for a long period of time;
(c) Capable of being easily copied or transferred to another medium, without loss of information;
(d) Retrievable over very long periods of time (i.e., >60 years);
(e) Readable and understandable; and
(f) Resistant to tampering, i.e., to alteration by unauthorized individuals.

Redundancy and diversity in an RMS are necessary for the effective management of the records [28]; the media used for records needs to be selected to ensure the durability, readability and retrievability of the information they contain. Overall, most countries have, or are, migrating to digital storage solutions with at least two backups; paper copies are getting converted to digital media [28].

Table 9-1, reproduced from Reference [28], provides typical lifetime and advantages/disadvantages of various media types. Microfilm and microfiche, and Optical disk, compact disk and digital video disks offer the longest storage lifetime. Microfilm and microfiche can contain large amounts of information (e.g., drawings) on very limited physical space, however, require a controlled environment for permanent storage. Digital media is compact and difficult to alter (more secure), however requires periodic updating and transfer to another digital media because of changes to computer hardware and software.

Table 9-1: Lifetime and Advantages/Disadvantages of Media Types

Storage Media	Typical Lifetime (years)	Advantage	Disadvantage
Hard copy	10+	Original masters held Difficult to alter or modify Paper and copying mechanisms are available for the archive period Legally acceptable	Controlled environment required Large volume Damage due to handling Cumbersome copying Easily lost or misfiled
Microfilms and microfiches	100+	Difficult to alter or modify Easily duplicated Compact storage Legally acceptable in some countries	Controlled environment required Large volume Damage due to handling Cumbersome procedures for producing hard copies Easily lost or misfiled
Magnetic tapes and disks	5-10	Compact, easy storage	stored images can easily be altered

Storage Media	Typical Lifetime (years)	Advantage	Disadvantage
		Tape or disk easily duplicated with no deterioration of data Accessible Can be updated	Controlled environment required Damage due to handling and magnetic erasure Hard copies required to be scanned to tape or disk Legality unclear Hardware and software require periodic updating (every 5–10 years) to ensure accessibility
Optical disks, compact disks and digital video disks	100+	Difficult to erase or alter Compact storage Easy remote access to data Few environmental controls Duplication with no deterioration of data	Hard copies required to be scanned to tape then disk Legality unclear Hardware and software require periodic updating (every 5–10 years) to ensure accessibility

10 CONCLUSION

The ARMSC aims to address the regulatory expectation of assessing the baseline radioactivity in construction materials prior to commissioning is described CNSC REGDOC-2.11.2 [2]. With the overall objective of minimizing radioactive waste during facility decommissioning, the document provides guidance on a range of tests and measurements that can be implemented to assess the initial radioactivity inventory of construction materials. By establishing this baseline inventory, and by referencing the operational history of the facility, future stakeholders will be better equipped to make informed, evidence-based decisions regarding the final handling and disposition of these materials at the end of an SMR facility's life cycle. The current version of the ARMSC focuses on the two most commonly used building materials, namely concrete and steel. However, the principles and methodologies described can be adapted and applied to other building materials as needed.

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ANNEX A. RADIATION DETECTION LIMITS

A.1 MEASUREMENT PROCESS

The ISO Guide to the Expression of Uncertainty in Measurement [30] defines two quantities: the measurand and the measurement.

- **Measurand:** The measurand, or true quantity (mass, activity, concentration, etc.), of an analyte in a media is denoted as L; and
- **Measurement:** The measurement is an estimate of the measurand and can be obtained by collecting a sample of the media, analyzing that sample, and interpreting the result of the analysis. It is denoted by (L˜) and is also referred to as the "measured value".

The objective of the measurement is to determine the value of the measurand. The uncertainty of a measurement is defined as the parameter associated with the result of a measurement that characterizes the dispersion of the values that could reasonably be attributed to the measurand. The uncertainty may be a standard deviation or an interval having a stated degree of probability to contain the value of the measurand.

A.2 DETECTION LIMIT EQUATIONS OF CURRIE

In its original publication in 1968 [31], Lloyd Currie set up the problem of defining the detection limit using two bins. One when the measurand is equal to the background and the other when the measurand is equal to the detection limit and above. The boundary between the two bins is the decision threshold. If a measurement is above the decision threshold, it falls in the bin of the detection limit, while if it is below the decision threshold, it falls in the bin of the background. The spread of the measured values around these two measurands is used to define two quantities:

- The decision threshold L_c, (critical level in Currie's article), which is defined as the net signal level (in terms of counts) above which an observed signal can be reliably considered detected (i.e. considered above background); and
- The detection limit, L_d, which is defined as the net signal level (in terms of counts) that can be expected to be observed with a defined level of certainty.

Currie used the statistical decision theory of Neyman-Pearson to determine the presence of net radioactivity as a choice between two hypotheses. The first is the null hypothesis, H_0, that the measurement contains no net radioactivity. The second is the alternative hypothesis, H_1, that the measurement contains the presence of net radioactivity. In any hypothesis test there are two types of errors:

- A false positive, or Type I error, indicates the presence of radioactivity when there isn't any. The probability of a Type I error is denoted by α; and
- A false negative, Type II error, indicates the absence of radioactivity when in fact there is radioactivity present. The probability of a Type II error is denoted by β.

For most applications, values for α & β are selected to be 0.05 which represent the upper and lower bounds of Type I and Type II errors respectively, and results in a 95% confidence level. The relationship between α & β is shown in **Error! Reference source not found.** based on the normal distribution.

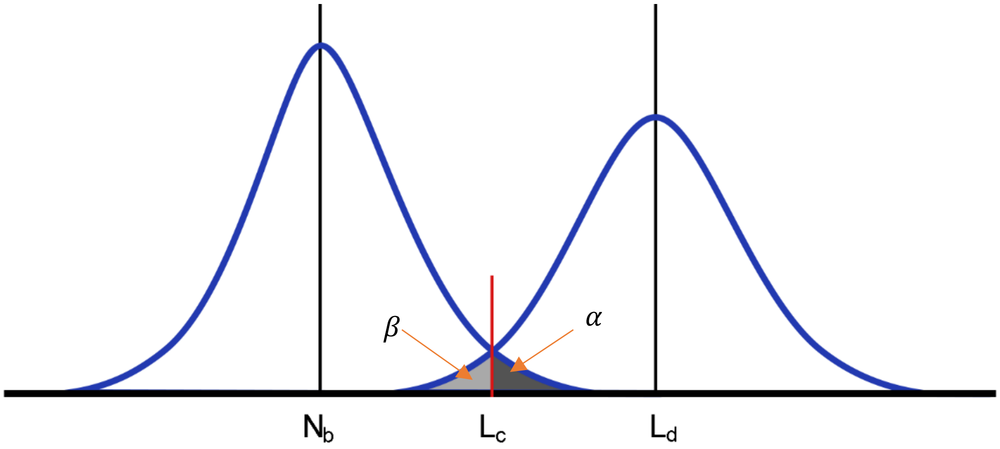


Figure A-1: Relationship Between Type I and Type II Errors for the net counts

Radiation measurements involve counting discrete interactions that follow the Poisson distribution. While counting data doesn't exactly follow the normal distribution, at higher count levels (say >50 counts), the normal distribution is a good approximation of the difference between two Poisson distributions. Therefore, features from both models are utilized for the definition of L_c and L_d.

In the most general case, the count time for the background t_0 is not necessarily the same as the count time for the sample t_g. The activity in the sample is proportional to the net count rate:

$$R = \frac{N_g}{t_g} - \frac{N_0}{t_0} \qquad \textbf{Eq. (19)}$$

Based on this, the decision threshold, expressed as a count rate, can be defined as:

$$R_c = k_\alpha \sigma_s(0) \qquad \textbf{Eq. (20)}$$

In this case, k_α is the one-sided confidence factor (z-score) of a normal distribution and $\sigma_s(0)$ is the standard deviation of the net count rate evaluated for a net count rate of zero.

$$R = 0 = \frac{N_g}{t_g} - \frac{N_0}{t_0} \qquad \textbf{Eq. (21)}$$

Such that

$$N_g = N_0 \frac{t_g}{t_0}$$

Eq. (22)

For a 95% confidence level, $k_\alpha = 1.645$.

$$\sigma_s = \sqrt{(\sigma_b)^2 + (\sigma_t)^2}$$

Eq. (23)

From the Poisson model, the variance (σ^2) is equal to the number of counts N.

$$\sigma_s^2 = \frac{N_0}{t_0^2} + \frac{N_0 t_g}{t_0 t_g^2}$$

Eq. (24)

Note that the definition of R_c accounts only for Type I errors.

$$R_C = k_\alpha \sqrt{\left(\frac{\sqrt{N_0}}{t_0}\right)^2 + \left(\frac{\sqrt{N_0\, t_g / t_0}}{t_g}\right)^2} = k_\alpha \sqrt{\frac{N_0}{t_0^2} + \frac{N_0 t_g}{t_0 t_g^2}}$$

Eq. (25)

$$R_C = k_\alpha \sqrt{\frac{N_0}{t_0}\left(\frac{1}{t_0} + \frac{1}{t_g}\right)}$$

Eq. (26)

When the decision threshold is expressed as a net count for a count time t_g.

$$L_c = k_\alpha \sqrt{\frac{N_0 t_g^2}{t_0}\left(\frac{1}{t_g} + \frac{1}{t_0}\right)}$$

Eq. (27)

For equal count times, this simplifies to:

$$L_c = 2.326\sqrt{N_b}$$

Eq. (28)

If there is radioactivity present, then the net counts R will have a positive value. The value of R must then be determined to keep false negatives (Type II errors) acceptable. This will define the detection limit, R_d, of the system. For example, if R_d were set to R_c then the probability of a Type II error (β) would be 50% since half the distribution would fall below R_c. Thus, R_d must be set at a value greater than the decision threshold to achieve greater detection reliability.

By definition, R_d is a net count rate above the background count rate:

$$R_d = \frac{N_g}{t_g} - \frac{N_0}{t_0}$$

Eq. (29)

$$N_g = R_d t_g + N_0 \frac{t_g}{t_0}$$

Eq. (30)

Based on this, the detection limit is then given by:

$$R_d = R_c + k_\beta \sigma_s(L_d)$$ **Eq. (31)**

$$R_D = R_C + k_\beta \sqrt{\frac{N_0}{t_0^2} + \frac{R_d}{t_g} + \frac{N_0}{t_g t_0}} = R_C + k_\beta \sqrt{\frac{R_d}{t_g} + \frac{N_0}{t_0}\left(\frac{1}{t_0} + \frac{1}{t_g}\right)}$$ **Eq. (32)**

Since

$$R_C = k_\alpha \sqrt{\frac{N_0}{t_0}\left(\frac{1}{t_0} + \frac{1}{t_g}\right)}$$ **Eq. (33)**

Then

$$R_D = R_C + k_\beta \sqrt{\frac{R_d}{t_g} + \frac{R_{\mathrm{C}}^2}{k_\alpha^2}}$$ **Eq. (34)**

$$(R_D - R_C)^2 = k_\beta^2 \left[\frac{R_d}{t_g} + \frac{R_{\mathrm{C}}^2}{k_\alpha^2}\right]$$ **Eq. (35)**

$$R_D = R_C + \frac{k_\beta^2}{2t_g}\left[1 + \sqrt{1 + \frac{4R_C^2 t_g^2}{k_\alpha^2 k_\beta^2}}\right]$$ **Eq. (36)**

It should be noted that R_d is defined once R_c has been set. As stated above, a value of $\beta = 0.05$ is selected to ensure a 95% confidence level. Similarly, k_α , $k_\beta = 1.645$.

When the count time of the sample is the same as the count time of the background, the expression becomes

$$L_D = L_C + \frac{k_\beta^2}{2}\left[1 + \sqrt{1 + \frac{4L_C^2}{k_\alpha^2 k_\beta^2}}\right]$$ **Eq. (37)**

The final expression for L_d is shown below in terms of background counts.

$$L_d = 3 + 4.653\sqrt{N_b}$$ **Eq. (38)**

The detection limit, therefore, takes both Type I and Type II errors in consideration to achieve a 95% confidence level when the sample and background count times are equal. The equations of Currie are implemented in most radiation detection software, including gamma spectroscopy, alpha-beta counters and liquid scintillation counters. They can be found in Chapter 20 of the Multi-Agency Radiological Laboratory Analytical Protocols (MARLAP) Manual. [32]

In 2001, Strom and McLellan [8] published an article where they tested the frequency properties of the Currie equation. As presented in **Error! Reference source not found.**, they showed that the actual false positive rate is much higher than the claimed rate α, for a wide range of background counts below 100. For this reason, it recommended to avoid using the equations of Currie, unless the number of background counts is expected to be higher than 100.

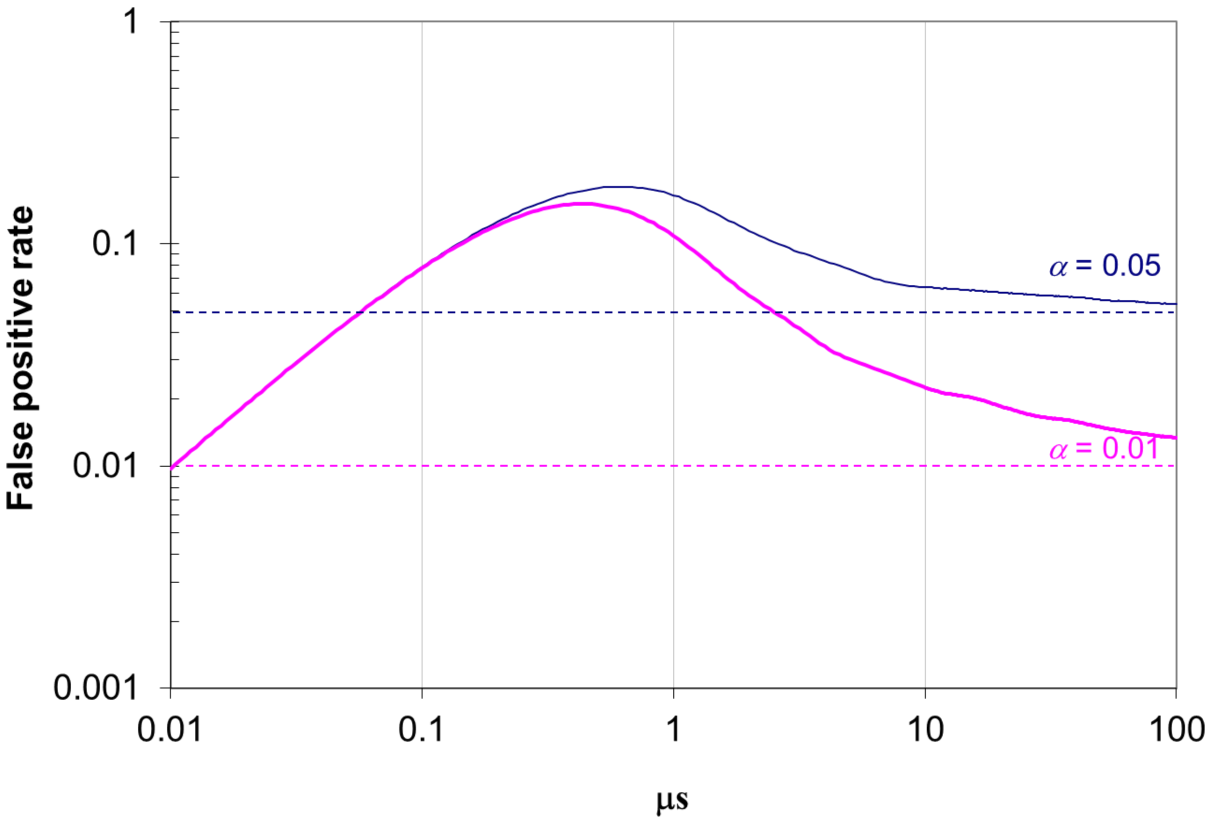


Figure A-2: False positive rate as a function of the background counts for Currie's equation

A.3 BAYESIAN DETECTION LIMIT

Bayes' theorem describes how to combine prior information with the likelihood of observing a measurement to obtain a posterior probability distribution for the measurand. This is different from Classic statistics where the measurand is assumed to have a single value.

The decision threshold corresponds to the probability $1 - \alpha$ that the sample has greater activity than the background:

$$P(\omega > \mu) = \int_0^\infty d\mu \int_\mu^\infty P(\mu | N_0, t_0) P(\omega | N_g, t_g) d\omega \qquad \textbf{Eq. (39)}$$

Where,

μ = value of the measurand for the background
$P(\mu|N_0, t_0)$ = posterior probability distribution of the background measurand
ω = value of the measurand for the sample
$P(\omega|N_g, t_g)$ = posterior probability distribution of the sample measurand

If the sample measurand is defined as $\omega = \mu + \Delta$, where $-\infty < \Delta < +\infty$, the equation for the decision threshold can be recast as follows by integrating over μ:

$$P(\omega - \mu > 0) = \int_0^\infty P(\Delta|N_0, t_0, N_g, t_g) d\Delta \qquad \textbf{Eq. (40)}$$

The shaded area in **Error! Reference source not found.** corresponds to $1 - \alpha$.

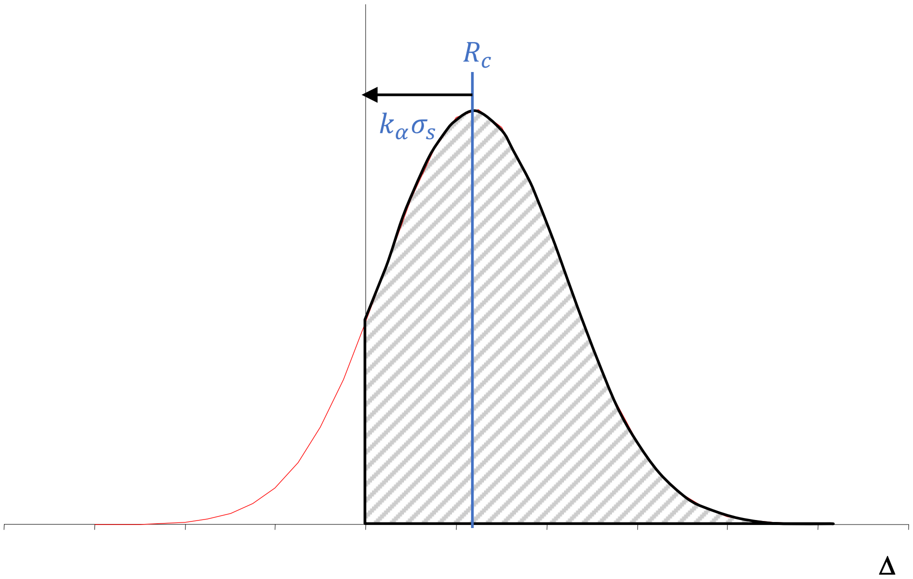


Figure A-3: Bayesian Decision Threshold

Assuming that the posterior probability distribution for the net count rate can be approximated with a normal distribution:

$$P(\Delta|N_0, t_0, N_g, t_g) = \frac{1}{\sqrt{2\pi}\sigma_s} e^{-\frac{(\Delta - r)^2}{2\sigma_s^2}} \qquad \textbf{Eq. (41)}$$

Where

$$r = \frac{N_g}{t_g} - \frac{N_0}{t_0} \qquad \textbf{Eq. (42)}$$

$$\sigma_s^2 = \frac{N_g}{t_g^2} - \frac{N_0}{t_0^2} \qquad \textbf{Eq. (43)}$$

Based on this, the decision threshold, expressed as a count rate, can be defined as:

$$R_c = k_\alpha \sigma(R_c) \qquad \textbf{Eq. (44)}$$

Solving for R_c

$$R_C = \frac{k_\alpha^2}{2t_0}\left[1 + \sqrt{1 + \frac{4N_0 t_g^2}{k_\alpha^2 t_0}\left(\frac{1}{t_0} + \frac{1}{t_g}\right)}\right] \qquad \textbf{Eq. (45)}$$

$$L_C = \frac{k_\alpha^2}{2}\left[1 + \sqrt{1 + \frac{4N_0 t_g^2}{k_\alpha^2 t_0}\left(\frac{1}{t_0} + \frac{1}{t_g}\right)}\right] \qquad \textbf{Eq. (46)}$$

Since the detection limit is a measurand, and it is expressed in terms of L_c, the equation for R_d and L_d remain the same. These equations are presented in CSA N288.0 Appendix D [7].

Checking the frequency properties of the Bayesian decision threshold with the method used by Storm and McLellan gives the false positive curve shown in **Error! Reference source not found.**. As long as the count time of the background and the sample are the same, the false positive rate is the claimed rate when the number of background counts is greater than four. If the count time are significantly different, the false positive rate will be slightly over or under the claimed rate.

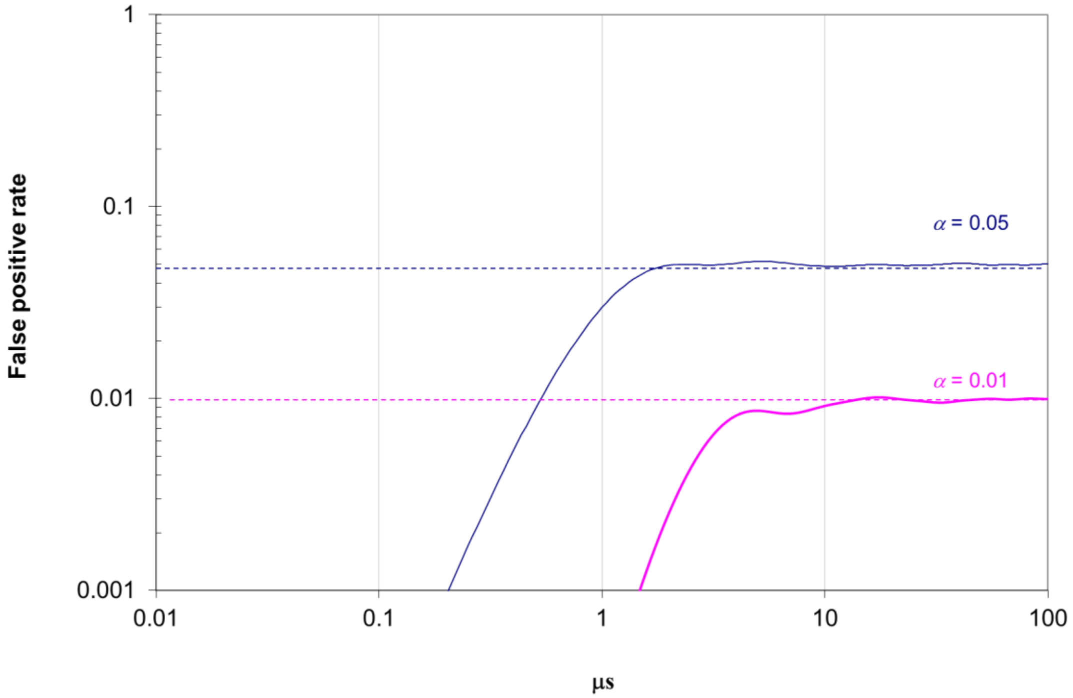


Figure A-4: False positive rate as a function of the background counts for the Bayesian equation

ANNEX B. ANALYSIS METHODS

This section provides a high-level overview of some of the more common analysis techniques. The methods of detection are divided into two categories:

- Radiometric Techniques: These techniques utilize the radiological properties of the material for detection and quantification. These include the type of radiation emitted (α, β, ɣ), energy of radiation, and any secondary radiation (ionizing or non-ionizing) from interaction with matter; and
- Non-Radiometric Techniques: These techniques rely on the physical properties of the material, such as mass, for detection and quantification.

B.1 RADIOMETRIC TECHNIQUES

Radiometric techniques are based on the principle of emitted radiation interacting with the detector media producing a physical response, the magnitude of which is proportional to the incident radiation, that can then be measured. Typical responses include the production of ion pairs that are collected to give an electrical signal (gas filled or semiconductor detectors) or the generation of pulses of light that can be counted (scintillation detectors). A simplified diagram of a radiation detection system is shown in **Error! Reference source not found.**.

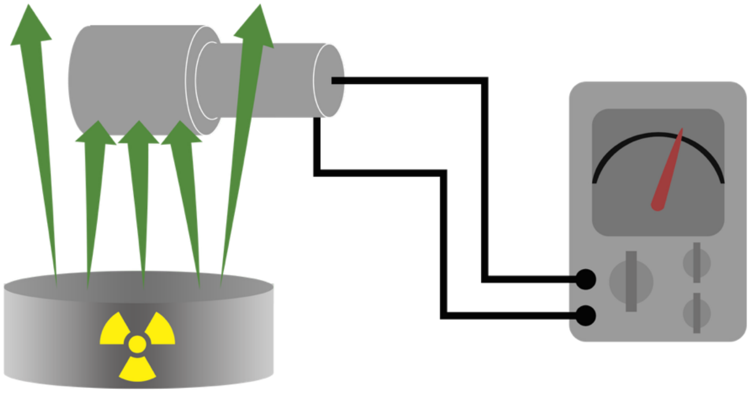


Figure B-1: Simplified Radiation Detection System

Detectors are designed to indicate either the integrated radiation received over a period of time or the rate at which radiation is being detected. While a detailed discussion of detection principles is beyond the scope of this document, general principles along with the main advantages and disadvantages of each are discussed below.

B.1.1 Gamma Spectrometry

Gamma spectrometry is one of the most widely used techniques for radionuclide identification with the only stipulation being that the radionuclides, their parents, or daughter products must be gamma emitters. This is very convenient when it comes to the detection of NORM and other relevant sources of radiation in the environment since the radionuclides of interest emit distinct characteristic gammas. These gammas are akin to a fingerprint of a given radionuclide and can be detected for positive identification and subsequent quantification. For radionuclides that emit few characteristic gammas at high intensities, such as ^{60}Co and ^{40}K, identification is straight forward. However, for radionuclides with multiple gamma emissions at varying intensities or materials of mixed radioisotopes, the resulting gamma spectrum may require deconvolution (manually or using software) to identify the constituents. In addition, there are several other features that result from the interaction of gammas in the detector which produce distinct features in the spectrum and can sometimes add to its complexity.

The process of gamma spectrometry involves separating output pulses from an energy sensitive detector according to their size which is proportional to the energy deposited in the detector. Common detector media for gamma spectrometry are either scintillation detectors, such as sodium iodide (NaI) sodium iodide, or semiconductor detectors, such as high-purity germanium (HPGe). Accurate detection relies on all the gamma energy being deposited within the detector media (i.e. photoelectric interaction). However, Compton scatter and pair production interactions may result in partial energy deposition within the detector and produce other (often identifiable) features in the output. This is where concept of detector resolution has an impact on the type (and complexity) of detector selected. Typically, semiconductor detectors offer far superior resolution to that of scintillation detectors. The trade-off however is that HPGe systems require cryogenic operating temperatures and cooling (by liquid nitrogen typically), making them less portable and more suited to lab environments. However, recent advances in cooling technology have resulted in high-resolution portable HPGe systems that utilize lower power mechanical cooling to reach liquid nitrogen temperatures in the field. Some even feature an onboard battery to maintain cooling without power. Scintillation type detectors on the other hand have been portable for decades and deployed as field instruments.

The output of a gamma spectrometry measurement is a histogram showing the energy (channel) on the x-axis the counts per channel in the y-axis. Characteristic gammas appear as distinct peaks (referred to as photopeaks) on the histogram the amplitude of which is proportional to the sample activity. There are several other features on the histogram that result from the different types of gamma interactions with matter. The typical difference in resolution between the two types is shown in **Error! Reference source not found.**below.

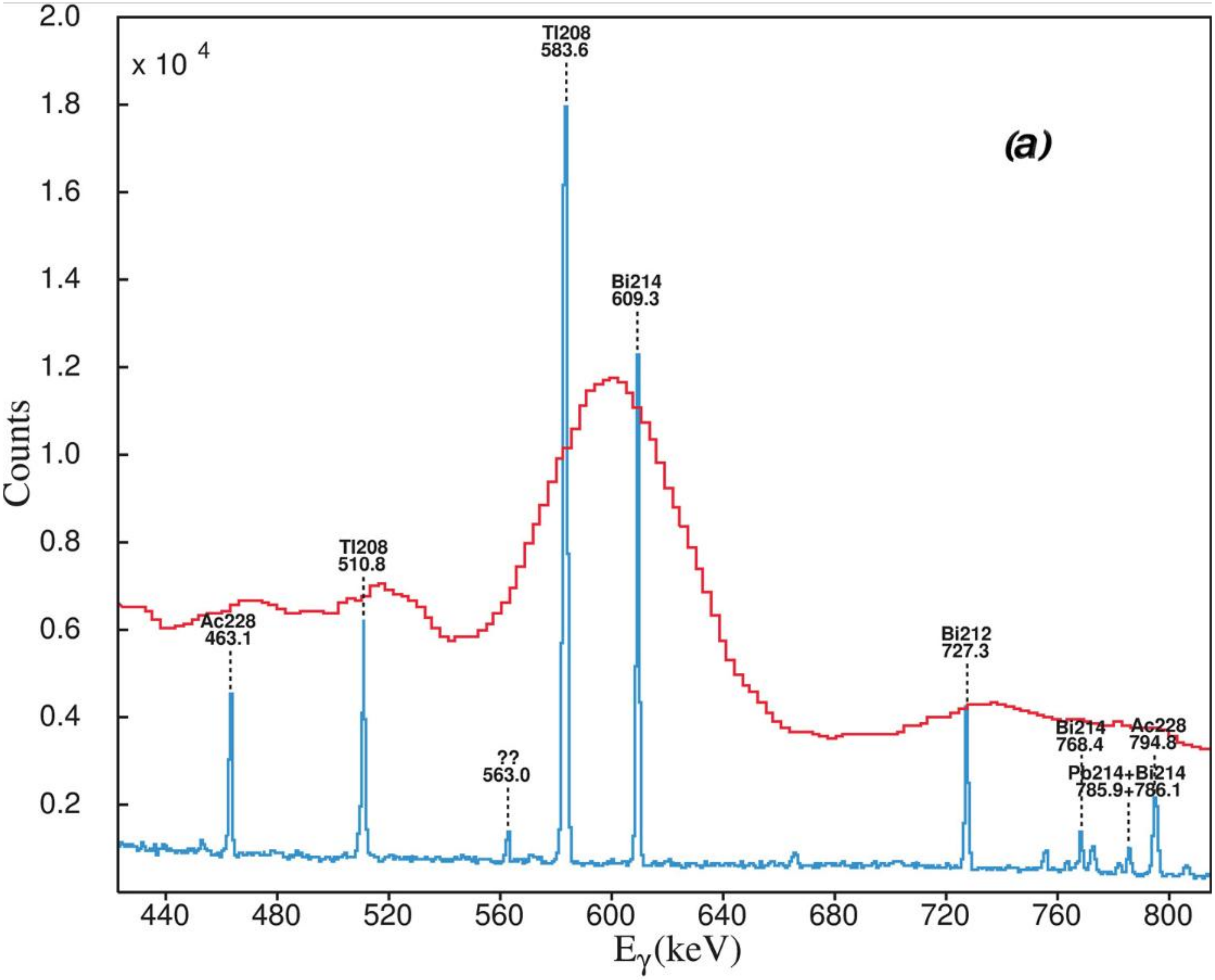


Figure B-2: Gamma Spectra of granite (NORM) from an HPGe (resolved peaks in blue) and NaI (broad peaks in red) detector. Sample contains background NORM with several characteristic gamma peaks as listed in Table 8-1

The main advantages and disadvantages of gamma spectrometry are listed below:

Advantages:

- Ability to identify individual radionuclides;
- Distinguish between NORM and activation or fission products produced from facility operation;
- Direct correlation and activity quantification;
- May be used as a field instrument;
- Measurement may be performed remotely, or detector may be left in place to acquire a spectrum;
- May be used in a lab environment for better resolution and offline sample counting;
- Simple sample preparation;

- Relatively short counting times (in the order of a few hours);
- Relatively inexpensive; and
- Non-destructive analysis (i.e. samples are not destroyed during the measurement and can be re-counted if needed).

Disadvantages:

- Requires qualified expertise to correctly interpret the results and minimize the risk of misidentification of radionuclides;
- Can only detect gamma emitters, which may be used as surrogates for hard to detect radionuclides;
- For identification using parent or daughter products that are gamma emitters, the assumption of equilibrium must be made (or validated) to compute the activity;
- Some HPGe applications, would require liquid nitrogen cooling making the system less portable for field deployment; and
- Sensitive to background radiation and may require the use of heavily shielded collimators to reduce background effects. Particularly for low activity measurements in the field.

B.1.2 Alpha Spectrometry

The ideal use for alpha spectrometry is for the identification and quantification of hard to detect alpha emitting sources including naturally occurring and transuranic species. Similar to gamma radiation, the alpha radiation emitted from decay is mono-energetic and can be used to identify the originating radionuclide. The main challenge with alpha spectrometry however is related to sample preparation. Due to the limited range of alpha particles, direct measurement of the sample does not yield accurate results. The majority of alpha particles generated within the sample will be absorbed by the sample media and not reach the detector. This self-shielding property makes it necessary to undertake a labour-intensive process that includes the following 4 steps, which are briefly discussed below:

1. Sample preparation;
2. Chemical separation;
3. Sample mounting; and
4. Counting.

The first step of sample preparation is to remove as many of the impurities as possible leaving behind only the alpha emitting sources. Different techniques (such as dissolving in various acids or melting at high temperatures) are employed depending on the sample media. Typically, at the end of this first step, the sample is in the form of acidified liquid. The second step of chemical separation targets a specific element of interest removing as many other elements as possible that may interfere with the detection process. Typical alpha detection systems have a fairly low resolution (compared to gamma spectrometry), making it difficult to discern between peaks that are less than say 20 keV apart. For example, if one is interested in quantifying isotopes of Pu the Am species would interfere since both have isotopes that emit alpha particles in the 5.5 MeV range. For this reason, the elements of interest must be targeted and separated before they can be counted. After the sample has been prepared and purified, the final step before the sample can be counted is sample mounting. This goal of this step is to maximize spectral resolution. Any interaction of the alpha particle on its trajectory from the source to the detector will result in energy loss thereby producing a poor (widened) peak shape. The ideal sample would therefore be a uniform monomolecular layer of the target element. While there are many methods for sample mounting, electrodeposition on a stainless-steel disk produces close to ideal geometries. The final step, much like gamma spectrometry, is counting the prepared sample using a solid-state detector with spectrum acquisition system.

Based on the discussion above, more than any other factor, sample preparation dictates the quality of the data. Adequate sample preparation is the result of a multitude of individual steps performed by numerous technicians. The resulting data undergoes a high level of scrutiny to ensure its accuracy. Therefore, alpha spectrometry is only conducted by an accredited laboratory with strict adherence to quality assurance standards. It is this labour-intensive cumbersome process of sample preparation that makes alpha spectrometry one of the most expensive types of radiometric analysis. Thus, only a subset of the samples acquired are sent for alpha spectrometry. Typically samples from a representative area or material are analysed to identify and quantify alpha emitters and used to establish radionuclide fingerprints. These fingerprints are then used in conjunction with alpha/beta counting of samples to quantify the activity of the constituents.

The main advantages and disadvantages of alpha spectrometry are listed below:

Advantages:

- Ability to identify the different hard to detect alpha emitting radionuclides;
- Distinguish between NORM and transuranic species;
- Ability to measure very low concentrations;
- High sensitivity; and
- Low background.

Disadvantages:

- Expensive;
- Complex sample preparation;
- Significant time between sample acquisition and results;
- Can only be performed by an accredited lab;
- Analysis is destructive (i.e., sample is destroyed during analysis process); and
- Multiple samples may be required.

B.1.3 Liquid Scintillation Counting

The concept of Liquid Scintillation Counting (LSC) exploits the property of some organic compounds to convert the energy absorbed into UV or visible light. The intensity of the light (number of photons) produced is proportional to the energy absorbed can subsequently be detected. These organic compounds are referred to as “fluors”. The basic process involves mixing the radioactive sample with a scintillation cocktail, which consists of a solvent and one or more fluors, in a transparent glass or plastic vial. Alpha or beta radiation from the sample interacts with the cocktail causing a scintillation event, known as a flash, resulting in the emission of photons in all directions. The prepared vials are placed in a light-tight enclosure that typically features two Photo Multiplier Tubes (PMT) that detect the flashes of light. The PMTs convert the photons to an electric pules that are counted.

The primary use of LSC is to quantify the activity of a known radionuclide. The cocktail is the medium for radiation detection into which the sample is fully immersed. Essentially all beta and alpha radiation from the sample interacts within the detector resulting in very high counting efficiencies (4π geometry). For this reason, LSC is typically used to quantify hard to detect pure beta emitters such as ^{3}H and ^{14}C. The counting efficiency for tritium can be as high as 70% and close to 100% for many other radionuclides. Energy ranges referred to as “windows” are established distinguish between known beta energies. In addition, since the pulse duration that results from an alpha particle is longer than that of a beta particle, pulse shape analysers and pulse decay discriminators allow for the separation beta emitters and alpha emitters.

Liquid scintillation counters are typically large pieces of equipment capable of counting numerous vails at a time. While not in-the-field instruments, they are commonly found in radiation labs in most operating nuclear stations.

The main advantages and disadvantages of LSC are listed below:

Advantages:

- Used for hard to detect pure beta emitters such as ^{3}H and ^{14}C;
- Can be used for alpha counting;
- High counting efficiencies;
- Low backgrounds;
- High sensitivity;
- Short count times;
- Relatively inexpensive; and
- Relatively simple sample preparation (compared to alpha spectrometry).

Disadvantages:

- Sample must be dissolved or uniformly dispersed through the cocktail;

- Opacity and colour can impact results;
- Foreign material (e.g. fingerprints) and condensation on vials can impact results;
- Destructive as the sample is typically dissolved in the cocktail;
- Not portable ;
- Not be used for radionuclide identification, but can determine the activity of known radionuclides; and
- Used scintillation cocktail must be disposed and is a consumable.

B.1.4 .Gross Alpha/Beta Counting

The technique of gross alpha/beta counting is used to rapidly detect the presence of radioactivity my measuring the total alpha and beta count rate of a given sample or location. The total number of alpha and beta interactions within the detector are counted for a specified duration and knowing the detector efficiency, the alpha and beta disintegration rate within the sample can be determined. While this method cannot be used to identify individual radionuclides, with a reliable and representative radionuclide fingerprint, the activity of the individual radionuclides can be computed, fairly accurately, by considering their relative contributions to the total alpha and beta count rates measured.

There are many detector types suitable for alpha/beta gross counting. The two most common types are proportional counters and dual-phosphor scintillation detectors. Proportional counters a type of gas filled detector in which the voltage applied is such that gas amplification is linear. Since incident alpha particles will produce much larger ionization signals than beta particles, the two can be easily distinguished. Dual-phosphor detectors employ two different scintillation media, once sensitive to alpha radiation (such as zinc sulfide) and another sensitive to beta radiation (such as plastic scintillator). As with liquid scintillation, a PMT is used to convert the light generated into an electrical pulse. Both types are used to simultaneously detect alpha and beta radiation in a given sample.

During the initial stages of a project, gross counting may be utilized as a screening tool to identify samples or sampling locations that may be good candidates for more detailed spectrometry analyses to establish radionuclide fingerprints. Gross counting may also be used to identify and verify impacted areas quickly. Once the radionuclide fingerprint is established, a broad array of samples may then be collected. The individual samples can be counted and, along with the established fingerprints, used to assess compliance with clearance criteria. This provides a practical and cost optimized method of using the more expensive analyses techniques to determine the radionuclide inventory and more cost effective gross counting techniques to assess compliance to clearance criteria.

The main advantages and disadvantages of Gross Alpha/Beta counting are listed below:

Advantages:

- Quick results, immediately detect the presence of radioactivity;
- In-the-field instrument;
- Inexpensive;
- May be used as a screening technique for the identification and verification of impacted areas and additional detailed analyses; and
- Minimal sample preparation.

Disadvantages:

- Results only show the alpha and beta count rate. No radionuclide information is obtained;
- Require a detailed radionuclide fingerprint based on gamma spectrometry and or alpha spectrometry to determine activities;
- Impacted by background radiation levels; and
- Can be impacted by self-shielding, particularly for alpha detection.

B.2 NON-RADIOMETRIC TECHNIQUES

While the majority of the techniques discussed in this section focus on radiometric analysis, since the activity is what will ultimately be ascertained, there are two non-radiometric techniques that can offer quick and reliable identification of elemental species. These may be used as stand-alone detection systems or conjunction with the radiometric techniques.

B.2.1 X-Ray Fluorescence Spectrometry

XRF spectroscopy is based on the principle that each element produces a unique set of characteristic fluorescent (secondary) x-rays after its atoms have been excited by high-energy x-rays or gamma rays. The process of generating fluorescent x-rays is as follows:

1. A solid or liquid sample is exposed to high-energy x-rays;
2. Ionization occurs and incident x-rays of sufficient energy can eject an electron from the inner orbitals of an atom;
3. The removal of an inner orbital electron results in electronic instability and a “hole”;
4. To restore stability, an electron from a higher orbital “falls” into hole;
5. Since this is a lower energy position, the excess energy (equal to the difference in energy between the two orbits) is released in the form of a characteristic fluorescent x-ray. The hole may be filled from a number of external orbitals and thus a given element can have a number of fluorescence x-rays; and
6. The resulting fluorescence x-rays are detected using a solid-state detector with multi-channel analyzer to generate an XRF spectrum (much like gamma spectrometry).

The source of incident high-energy x-rays or gamma rays may be an x-ray generator (based on bremsstrahlung interaction with a heavy metal) or gamma emitting radioisotope. In both cases, specific radiation safety training may be required to use the equipment. In addition, ALARA considerations may require the use of exclusion areas during XRF operation. Typical handled XRF units can deliver exposures similar to a medical x-ray if not handled correctly.

The XRF technique is commonly used to determine the elemental composition of materials. However, this method can specifically be used to quantify concentrations uranium and thorium down to the level of several ppm. An additional advantage of XRF is the ability to detect other non-radioactive impurities such as heavy metals whose concentrations may impact future decommissioning and disposition pathways.

The main advantages and disadvantages of XRF spectroscopy are listed below:

Advantages:

- Rapid identification of elements;
- Used to detect NORM down to concentrations of several ppm;
- Can be used to detect other non-radioactive elements that may also have restrictions for clearance; and
- Portable and can be used in-the-field.

Disadvantages:

- X-ray or gamma ray sources;
- Additional training and protective measures may be required;
- Cannot detect individual isotopes; and
- Must be calibrated to the type of material being analysed.

B.2.2 Mass Spectroscopy

Mass Spectrometry (MS) is an analytical technique used to measure the mass-to-charge ratio of ions. It helps identify the amount and type of chemicals present in a sample by generating a mass spectrum, which is a plot of ion signal as a function of the mass-to-charge ratio.

The main advantages and disadvantages of MS are listed below:

Advantages:

- Can detect very low levels of radionuclides;
- Can analyze a wide range of samples, including solids, liquids, and gases; and
- Rapid identification of radionuclides and concentrations.

Disadvantages:

- High cost of equipment and maintenance;
- Skilled personnel to operate equipment;
- Sample preparation can be extensive and time-consuming; and
- The presence of other substances in the sample can interfere with the analysis.

ANNEX C. SAMPLE SURVEY PLAN

This section presents an example of a survey plan for a relatively simple scenario. Its purpose is to highlight the major steps of DQO process and suggested format for preparing the survey plan. Note that the scenario outlined below is a purely hypothetical and simplified example. No engineering or design analysis has been conducted to define any of the parameters mentioned, and any construction methods and durations discussed are assumed to be under ideal conditions.

Sample Scenario:

A training and storage facility is being designed and constructed within the protected area (site license boundary) of an SMR site. The building will be part of a support service complex and have the following three primary functions:

1. *Field Staff Training area (non-radiological area)*
2. *Office space on a mezzanine (non-radiological area)*
3. *Storage area for contaminated material and equipment (radiological area).*

The current intent of the building is to not process any radioactive material, (i.e. any radioactive material that is stored will be in sealed containers), however that may change over the 60-year operation of the facility, but would require a licence amendment. Sketches of the building are shown in Figure C-1 and Figure C-2.

Based on the geotechnical study of the site, a pile and cap foundation is required to support the structure as shown in Figure C-1. The soil test report indicates that a pile depth of 33′ (10 m) is required. Piles will be cast in-situ, which involves drilling bore holes and filling with reinforced concrete. The building will consist of a total of 50 piles, each with a diameter of 12″ (30.5 cm) spaced 3.2′ (1 m) apart. The pile cap will be ordinary concrete that is 1.64′ (0.5 m) thick. The pile cap will be poured above grade level and will be supported by the piles. The above ground structure will consist of steel columns, beams, and trusses with external corrugated steel cladding.

The radioactive material storage area will be enclosed by a 12″ (30.5 cm) thick concrete block shielding wall that is 10′ (3 m) high. The wall will be constructed prefabricated blocks that are 12″ (30.4 cm) in length by 24″ (61 cm) wide by 12″ (30.5 cm) in height held by mortar. An external epoxy finish will be applied on any exposed surfaces of the block wall. The dimensions of the radioactive material storage area are 48′ (15 m) by 20′ (6.1 m). The shielding wall will be constructed after building construction is completed (including electrical, plumbing and HVAC).

No recycled materials will be utilized for this construction, that is the steel will be free of any radioactive material. The only source of radioactivity will be NORM present in the concrete. Aggregate for concrete may be sourced from three local quarries in relatively close proximity to the SMR construction site.

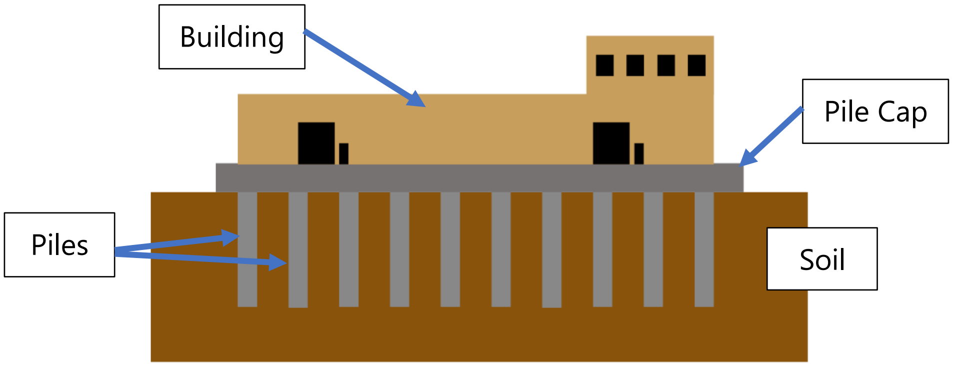


Figure C-1: General Arrangement of Building & Foundation

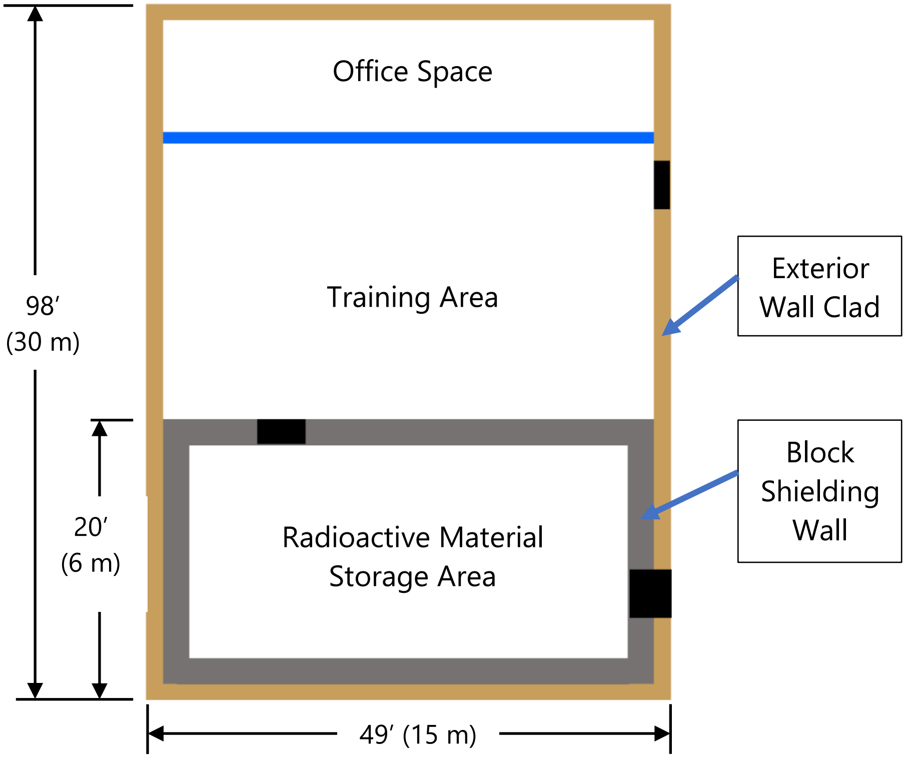


Figure C-2: Building Layout and Dimensions

Sample Survey Plan for Training & Storage Facility

Date:	Record No:	
Prepared By:		
Reviewed By:		
Approved By:		

Section 1: State the Problem

The purpose of this survey plan is to determine the radioactive material inventory of the concrete being utilized in the construction of the training and storage facility. Concrete will be primarily utilized in the three locations listed below:

- Piles
- Pile Cap
- Block Shielding Wall

Testing will be conducted in the following three phases:

1. Pre-Construction testing of aggregate at the quarries
2. During the construction in parallel with concrete pours
3. In-Situ measurements following construction

Since there will be no recycled material being used, testing of the concrete will consist only of NORM identification and quantification. Testing will not impact the construction critical path schedule and will therefore consist of utilizing existing concrete sampling as well as in-situ field measurements after the structure has been completed.

Section 2: Identify the Goal

The goal of the analysis is to quantify the NORM (^{238}U ^{232}Th and ^{40}K) concentration in (Bq/g) and compare against current unconditional clearance levels (UCL).

Phase 1:
Results from the pre-construction aggregate testing phase will be used to screed quarries and appropriately source aggregate to ensure UCLs are not exceeded. A screening report will be produced and suppliers meeting the UCL screening requirements will be identified and recommended.

Phase 2/3:
During the construction phase, areas where the NORM concentration is found to exceed the UCL will be identified and flagged in the characterization report. The data collected and analysis results will be stored in the construction history dockets for the lifetime of the facility.

Section 3: Information Inputs

Phase 1:
Confirm the three local aggregate suppliers
Confirm sampling can be performed

Phase 2:
Confirm the number of existing tests as per CSA A23.2-3C [21] that will be performed on the concrete for the piles and pile cap.

Phase 3:
Confirm that the shielding blocks are procured from a single supplier to ensure uniformity of concrete mixture.

Section 4: Boundaries of the Study

The analysis is limited to the concrete used in the construction of the training and storage facility.

Phase 1:
Aggregate samples will be acquired from the three local quarries in the vicinity of the construction site. Samples will be acquired at least 6 months prior to the start of construction to ensure adequate time is provided for the interpretation of results and subsequent screening if required. Samples will be acquired between late spring and early fall due to weather constraints.

Phase 2:
The acquisition of samples will take place at the construction site. Samples for NORM analysis will be taken in addition to the typical concrete testing as required by construction standards during the pour process. Samples will be taken of the piles followed by the pile cap.

Phase 3:
Field measurements of the completed building can only be conducted for the pile cap and the shield wall since the piles themselves will be covered by the pile cap.

Once the structure is complete (including electrical, plumbing, and HVAC), field measurements of the pile cap will be conducted over a period of 2 weeks.

The shielding wall will be installed following the completion of the building (including roof and services). Field measurements of the shielding wall will be conducted during construction prior to the epoxy finish being applied to the wall.

Section 5: Analytic Approach (Decision Rule)

Phase 1:

1. If the average NORM concentrations in the aggregate sourced from a given quarry exceed 60% of their respective UCLs, then aggregate from that quarry(s) will not be utilized for the project. This is based on a 1:3 cement to aggregate mixing ratio.

$$\frac{A_i}{CC_i} < 0.6$$

2. If the average NORM concentrations from all sampled quarries exceed 60% of their respective UCLs, then the quarry(s) where NORM concentrations are less than UCLs will be utilized.

$$\frac{A_i}{CC_i} < 1$$

3. If the average NORM concentrations from all sampled quarries exceed their respective UCLs, then a high-level project decision will be made to determine a path forward on utilizing one or more of the three local quarries or obtaining aggregate from other sources. If the latter is selected, then this survey plan will be revised to include any additional quarry(s).

Phase 2:
No decision rule exists for Phase 2 sine once constructed, all or portions of the building will not be demolished. The NORM radionuclides quantified during in-situ measurements will be compared to their respective NORM UCLs. Any locations where the measured activity concentration exceeds the UCL will be identified in a report.

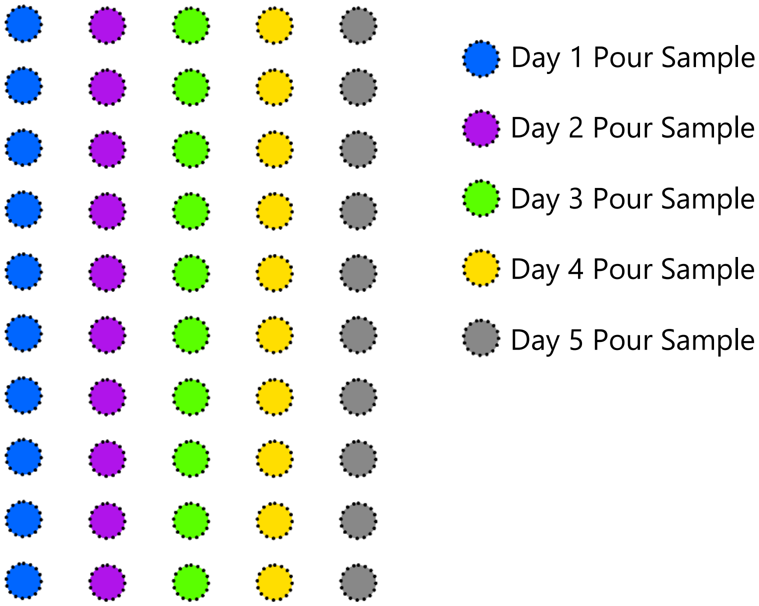
Section 6: Performance or Acceptance Criteria

Ensure the minimum detectable activity (MDA) of the detection system is at least 50% of the UCL value.

Section 7: Plan for Obtaining Data

Phase 1: Aggregate Sampling
A total of 10 aggregate samples will be collected from each of the three local quarries identified. Each sample will be collected from a different stockpile of aggregate. Sampling will be performed in accordance with CSA A23.2 -A1.2 [21], Sampling from Stockpiles, to ensure a representative sample from each stockpile is obtained. The final mass will be 1 kg of aggregate per sample. Each sample package consisting of 10 samples from a given quarry will be sent to an accredited lab for gamma spectrometry analysis. The lab shall adhere to approved sample preparation methods to ensure secular equilibrium is established.

Phase 2: During Construction - Concrete Pour
Piles:
The building contains a total of 50 piles. As per the dimensions provided, each pile has a volume of 26 ft^3 (0.73 m^3) resulting in a total reinforced concrete volume of 1296 ft^3 (36.4 m^3). As per CSA A23.2-3C [21] the minimum compression testing frequency using casted cylinders is once per day or every 100m^3. In addition to the required three sample cylinders, a fourth cylinder will be poured for NORM analysis purposes. Assuming 10 piles are poured per day, a total of 5 test sample cylinders will be acquired. These will be sent to an accredited lab for gamma spectrometry following approved methods of sample preparation to ensure secular equilibrium is established. Records will indicate which sets of piles were poured on a given day similar to the figure below.



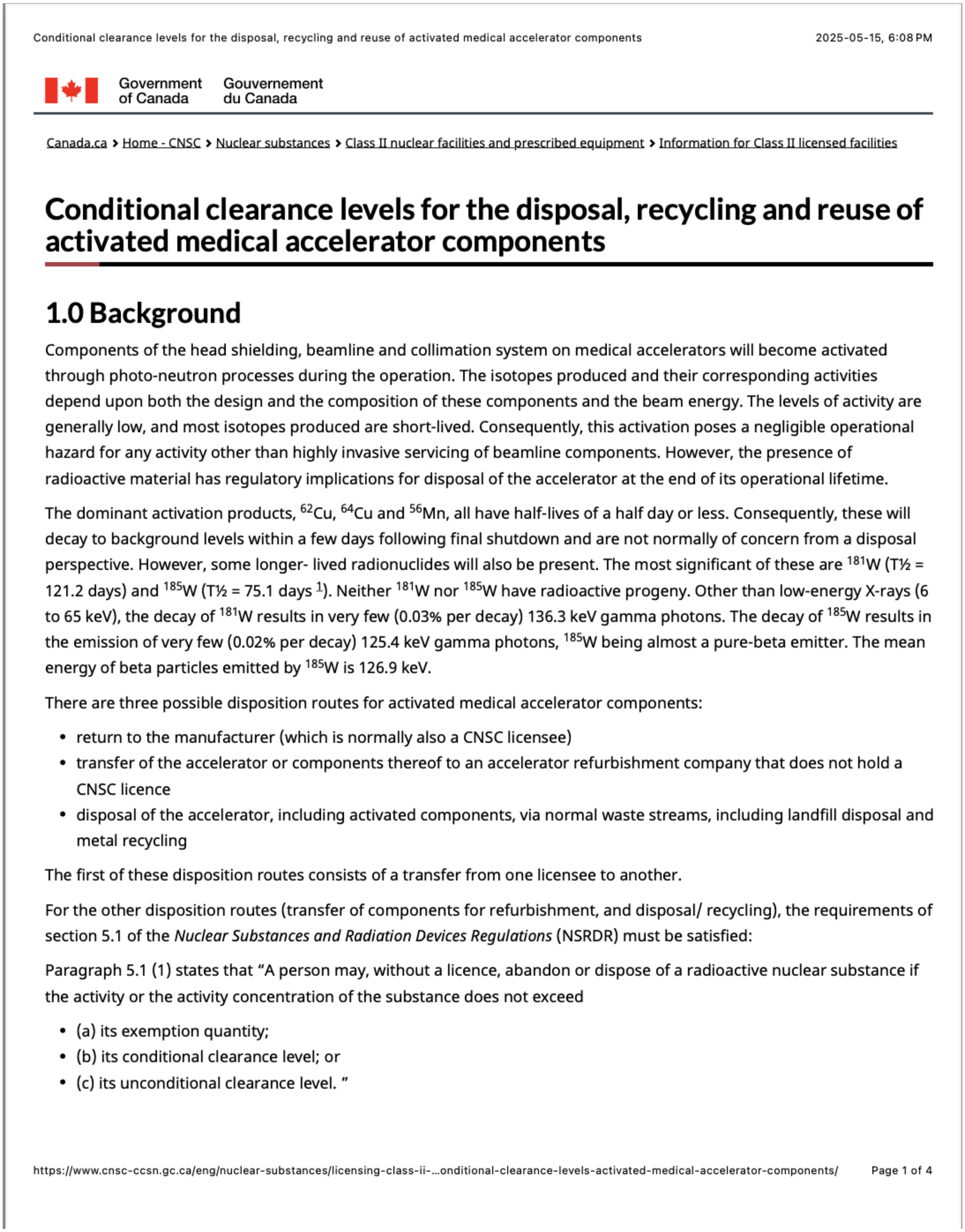
Piles Poured per day

Pile Cap:
Based on dimensions, the pile cap has a total volume of 225 m^3 of concrete (not including any rebar). As per the requirements of CSA A23.2-3C [21], 3 test cylinder samples would be prepared. Similar to the piles, an additional 3 test cylinders will be poured for radiological analysis at an accredited lab.

Phase 3: In-Situ Measurements
Pile Cap:
Once the building is complete, the 16′ (5 m) by 16′ (5 m) grid will be projected onto the pile cap top surface and a total of 18 XRF measurements will be conducted at the centre of each grid point.

Shielding Block Wall:
Based on the shielding block dimensions, a total of 680 shielding blocks will be required to cover the perimeter of the radiological material storage area. At a minimum 10% of the blocks used, 68, will be scanned with the XRF. In addition, any damaged or cut off sections of the wall may be sampled and sent to an accredited lab for analysis, depending on time and budget.

ANNEX D. CONDITIONAL CLEARANCE EXAMPLE



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Conditional clearance levels for the disposal, recycling and reuse of activated medical accelerator components

1.0 Background

Components of the head shielding, beamline and collimation system on medical accelerators will become activated through photo-neutron processes during the operation. The isotopes produced and their corresponding activities depend upon both the design and the composition of these components and the beam energy. The levels of activity are generally low, and most isotopes produced are short-lived. Consequently, this activation poses a negligible operational hazard for any activity other than highly invasive servicing of beamline components. However, the presence of radioactive material has regulatory implications for disposal of the accelerator at the end of its operational lifetime.

The dominant activation products, ^{62}Cu, ^{64}Cu and ^{56}Mn, all have half-lives of a half day or less. Consequently, these will decay to background levels within a few days following final shutdown and are not normally of concern from a disposal perspective. However, some longer- lived radionuclides will also be present. The most significant of these are ^{181}W (T½ = 121.2 days) and ^{185}W (T½ = 75.1 days [1]). Neither ^{181}W nor ^{185}W have radioactive progeny. Other than low-energy X-rays (6 to 65 keV), the decay of ^{181}W results in very few (0.03% per decay) 136.3 keV gamma photons. The decay of ^{185}W results in the emission of very few (0.02% per decay) 125.4 keV gamma photons, ^{185}W being almost a pure-beta emitter. The mean energy of beta particles emitted by ^{185}W is 126.9 keV.

There are three possible disposition routes for activated medical accelerator components:

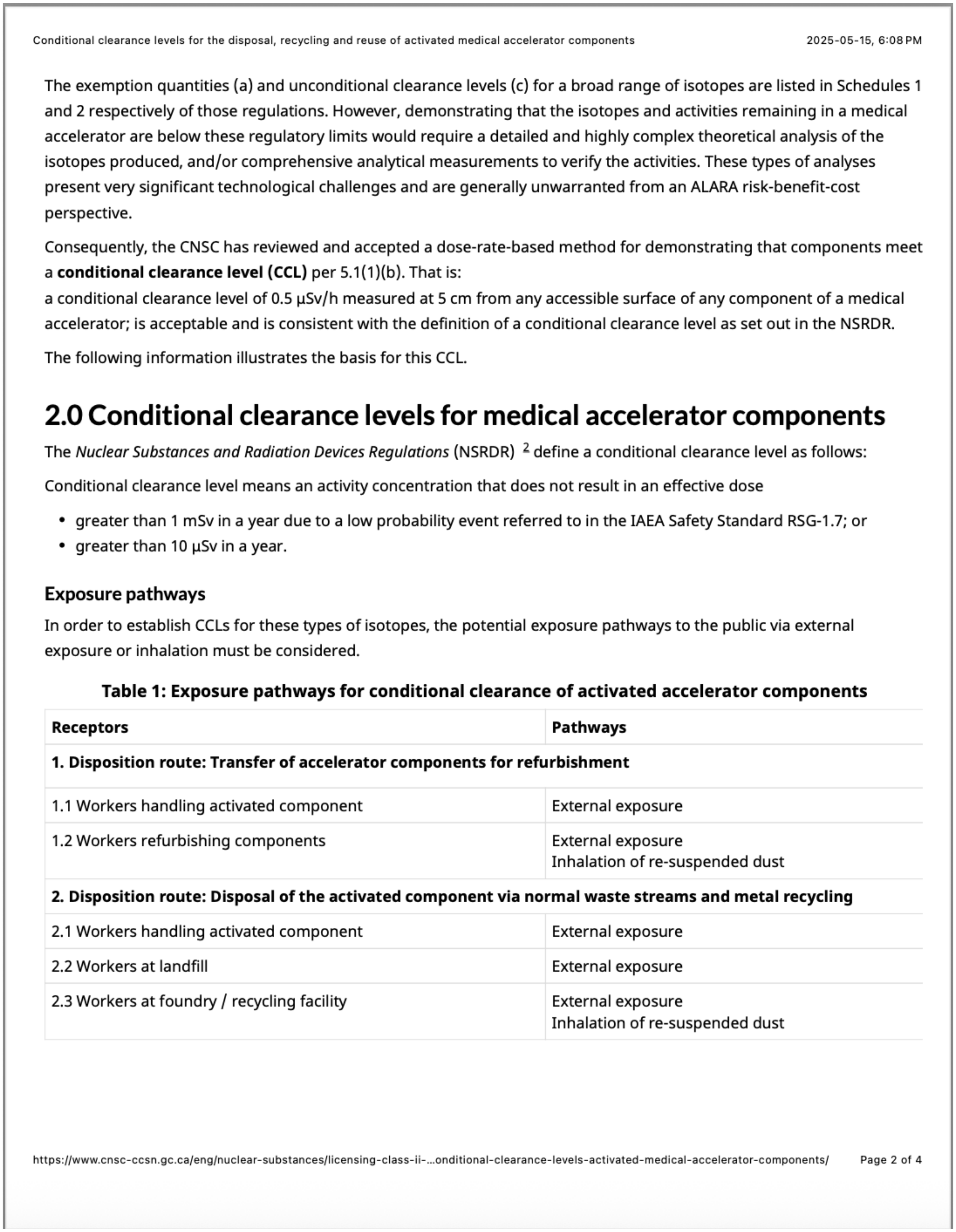
- return to the manufacturer (which is normally also a CNSC licensee)
- transfer of the accelerator or components thereof to an accelerator refurbishment company that does not hold a CNSC licence
- disposal of the accelerator, including activated components, via normal waste streams, including landfill disposal and metal recycling

The first of these disposition routes consists of a transfer from one licensee to another.

For the other disposition routes (transfer of components for refurbishment, and disposal/ recycling), the requirements of section 5.1 of the *Nuclear Substances and Radiation Devices Regulations* (NSRDR) must be satisfied:

Paragraph 5.1 (1) states that “A person may, without a licence, abandon or dispose of a radioactive nuclear substance if the activity or the activity concentration of the substance does not exceed

- (a) its exemption quantity;
- (b) its conditional clearance level; or
- (c) its unconditional clearance level. ”



The exemption quantities (a) and unconditional clearance levels (c) for a broad range of isotopes are listed in Schedules 1 and 2 respectively of those regulations. However, demonstrating that the isotopes and activities remaining in a medical accelerator are below these regulatory limits would require a detailed and highly complex theoretical analysis of the isotopes produced, and/or comprehensive analytical measurements to verify the activities. These types of analyses present very significant technological challenges and are generally unwarranted from an ALARA risk-benefit-cost perspective.

Consequently, the CNSC has reviewed and accepted a dose-rate-based method for demonstrating that components meet a **conditional clearance level (CCL)** per 5.1(1)(b). That is:
a conditional clearance level of 0.5 µSv/h measured at 5 cm from any accessible surface of any component of a medical accelerator; is acceptable and is consistent with the definition of a conditional clearance level as set out in the NSRDR.

The following information illustrates the basis for this CCL.

2.0 Conditional clearance levels for medical accelerator components

The *Nuclear Substances and Radiation Devices Regulations* (NSRDR) [2] define a conditional clearance level as follows:

Conditional clearance level means an activity concentration that does not result in an effective dose

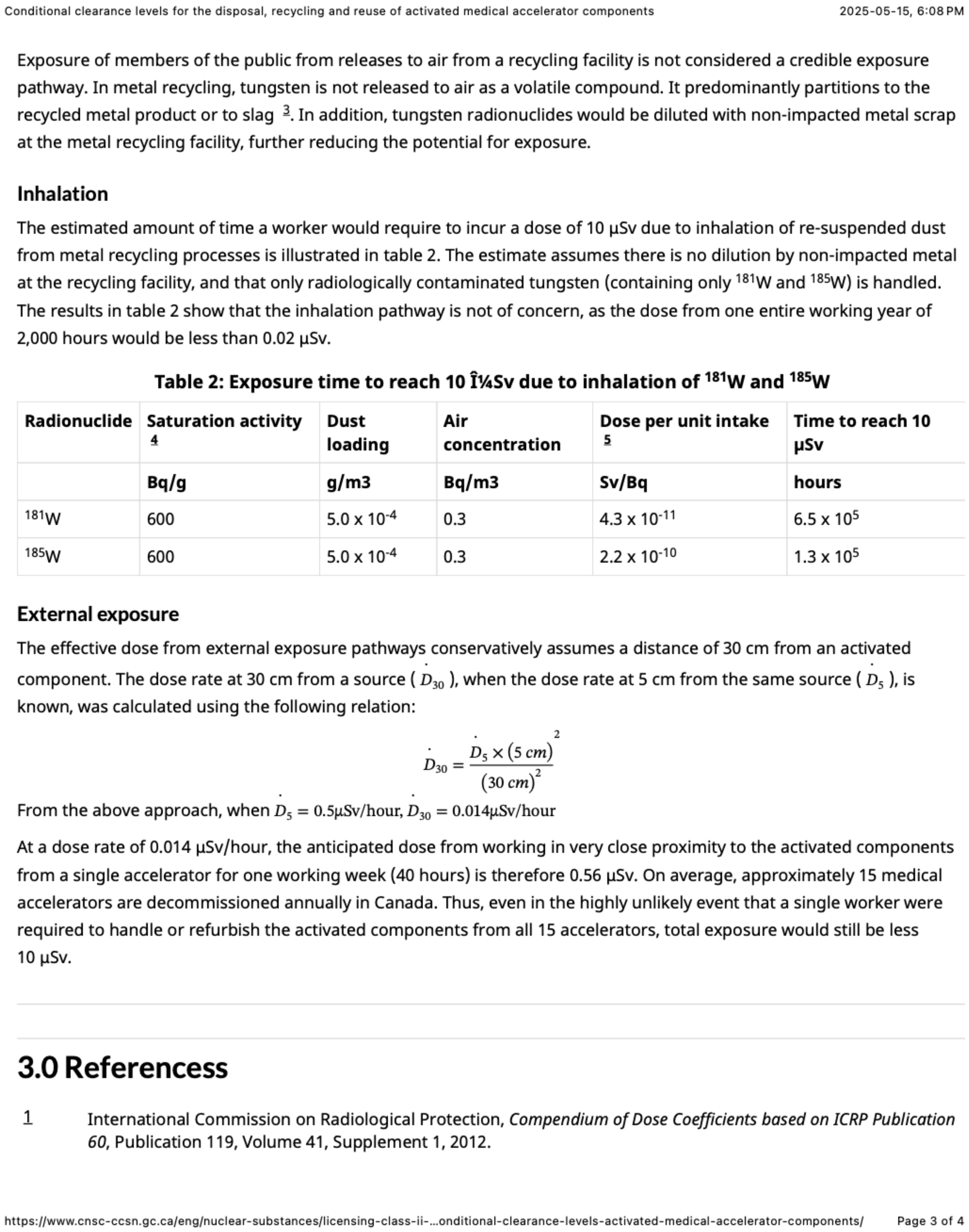
- greater than 1 mSv in a year due to a low probability event referred to in the IAEA Safety Standard RSG-1.7; or
- greater than 10 µSv in a year.

Exposure pathways

In order to establish CCLs for these types of isotopes, the potential exposure pathways to the public via external exposure or inhalation must be considered.

Table 1: Exposure pathways for conditional clearance of activated accelerator components

Receptors	Pathways
1. Disposition route: Transfer of accelerator components for refurbishment	
1.1 Workers handling activated component	External exposure
1.2 Workers refurbishing components	External exposure Inhalation of re-suspended dust
2. Disposition route: Disposal of the activated component via normal waste streams and metal recycling	
2.1 Workers handling activated component	External exposure
2.2 Workers at landfill	External exposure
2.3 Workers at foundry / recycling facility	External exposure Inhalation of re-suspended dust



Exposure of members of the public from releases to air from a recycling facility is not considered a credible exposure pathway. In metal recycling, tungsten is not released to air as a volatile compound. It predominantly partitions to the recycled metal product or to slag [3]. In addition, tungsten radionuclides would be diluted with non-impacted metal scrap at the metal recycling facility, further reducing the potential for exposure.

Inhalation

The estimated amount of time a worker would require to incur a dose of 10 µSv due to inhalation of re-suspended dust from metal recycling processes is illustrated in table 2. The estimate assumes there is no dilution by non-impacted metal at the recycling facility, and that only radiologically contaminated tungsten (containing only ^{181}W and ^{185}W) is handled. The results in table 2 show that the inhalation pathway is not of concern, as the dose from one entire working year of 2,000 hours would be less than 0.02 µSv.

Table 2: Exposure time to reach 10 Î¼Sv due to inhalation of ^{181}W and ^{185}W

Radionuclide	Saturation activity [4]	Dust loading	Air concentration	Dose per unit intake [5]	Time to reach 10 µSv
	Bq/g	**g/m3**	**Bq/m3**	**Sv/Bq**	**hours**
^{181}W	600	5.0×10^{-4}	0.3	4.3×10^{-11}	6.5×10^{5}
^{185}W	600	5.0×10^{-4}	0.3	2.2×10^{-10}	1.3×10^{5}

External exposure

The effective dose from external exposure pathways conservatively assumes a distance of 30 cm from an activated component. The dose rate at 30 cm from a source ($\dot{D}_{30}$), when the dose rate at 5 cm from the same source ($\dot{D}_5$), is known, was calculated using the following relation:

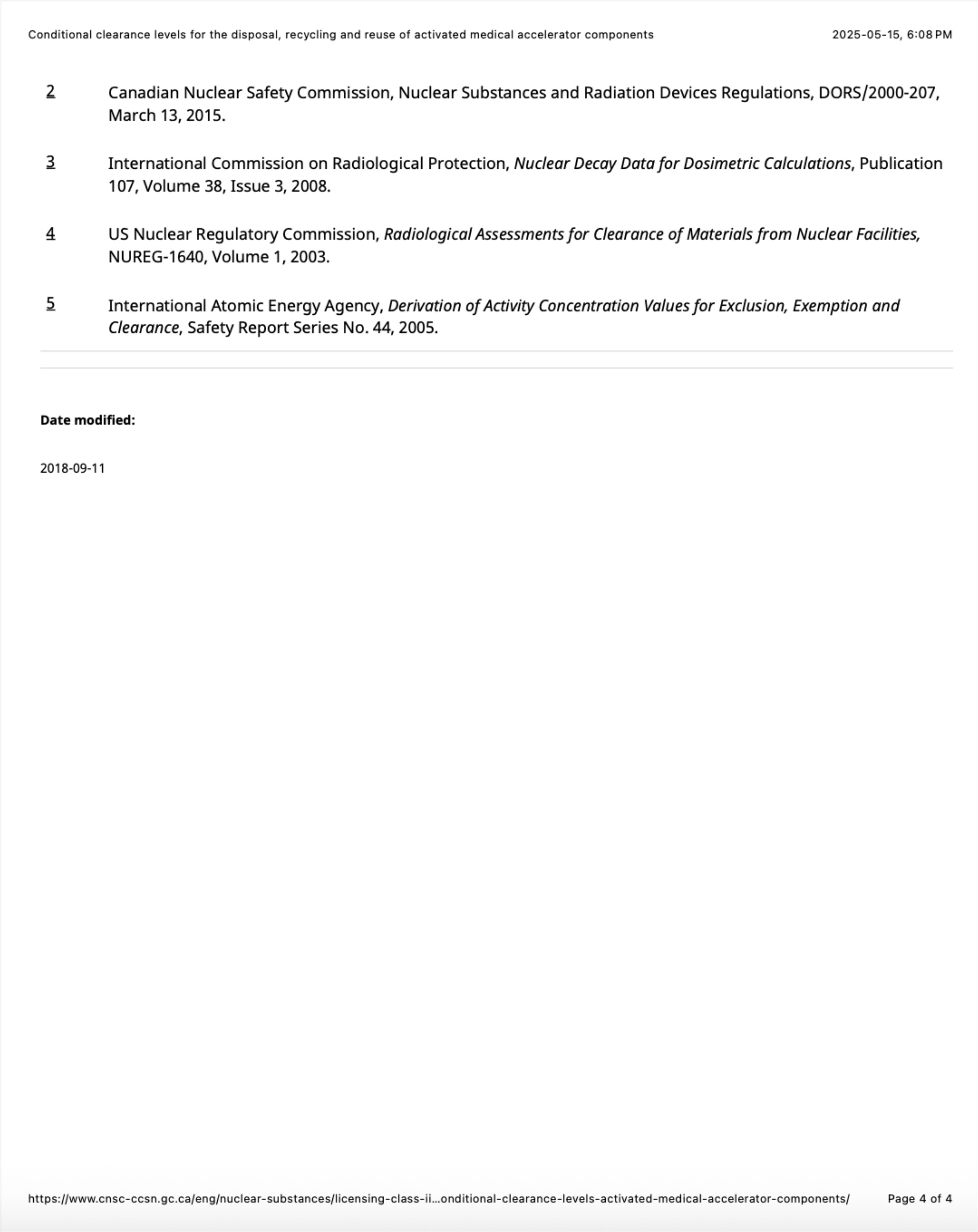
$$\dot{D}_{30} = \frac{\dot{D}_5 \times (5\,cm)^2}{(30\,cm)^2}$$

From the above approach, when $\dot{D}_5 = 0.5\mu\text{Sv/hour}, \dot{D}_{30} = 0.014\mu\text{Sv/hour}$

At a dose rate of 0.014 µSv/hour, the anticipated dose from working in very close proximity to the activated components from a single accelerator for one working week (40 hours) is therefore 0.56 µSv. On average, approximately 15 medical accelerators are decommissioned annually in Canada. Thus, even in the highly unlikely event that a single worker were required to handle or refurbish the activated components from all 15 accelerators, total exposure would still be less 10 µSv.

3.0 Referencess

1 International Commission on Radiological Protection, *Compendium of Dose Coefficients based on ICRP Publication 60*, Publication 119, Volume 41, Supplement 1, 2012.



Conditional clearance levels for the disposal, recycling and reuse of activated medical accelerator components 2025-05-15, 6:08 PM

2 Canadian Nuclear Safety Commission, Nuclear Substances and Radiation Devices Regulations, DORS/2000-207, March 13, 2015.

3 International Commission on Radiological Protection, *Nuclear Decay Data for Dosimetric Calculations*, Publication 107, Volume 38, Issue 3, 2008.

4 US Nuclear Regulatory Commission, *Radiological Assessments for Clearance of Materials from Nuclear Facilities,* NUREG-1640, Volume 1, 2003.

5 International Atomic Energy Agency, *Derivation of Activity Concentration Values for Exclusion, Exemption and Clearance*, Safety Report Series No. 44, 2005.

Date modified:

2018-09-11

https://www.cnsc-ccsn.gc.ca/eng/nuclear-substances/licensing-class-ii...onditional-clearance-levels-activated-medical-accelerator-components/ Page 4 of 4

ANNEX E. VBA CODE FOR MANN-WHITNEY STATISTICS

```
Const NZ As Integer = 200
Public vOmeg(1 To NZ) As Double
Public fOmeg(1 To NZ) As Double
Public cOmeg(1 To NZ) As Double

Sub Macro1()
'
‘ Output the probability distribution for plots
With ThisWorkbook.Worksheets("Sheet2")
    For i = 1 To 200
       .Cells(4 + i, 5).Value = vOmeg(i)
       .Cells(4 + i, 6).Value = fOmeg(i)
       .Cells(4 + i, 7).Value = cOmeg(i)
    Next i
End With

End Sub

Function Mann_Whitney(RE As Variant, RC As Variant) As Double

Dim vE() As Double
Dim vC() As Double
Dim vt() As Variant
Dim XE() As Double
Dim XC() As Double

Dim Omeg As Double
Dim kOmeg As Double
Dim barOmeg As Double
Dim Ombar As Double
Dim UE As Double
Dim UC As Double
Dim Uz As Double
Dim tot As Double
Dim so As Double
Dim sm As Double
Dim nE As Double
Dim nC As Double
Dim nH As Double
Dim xs As Double
Dim x As Double
Dim absum As Double
Dim a As Double
Dim b As Double
Dim na As Double
Dim nb As Double
Dim i As Long
Dim j As Long
Dim k As Long
Dim l As Long
Dim iE As Long
Dim jC As Long

' Normalize the data

nE = RE.Count     ' Sort vE
ReDim vE(1 To nE)
ReDim vt(1 To nE)
ReDim XE(1 To nE) As Double
For i = 1 To nE
    vt(i) = RE(i)
Next i
'vt = Application.WorksheetFunction.Sort(vt, 1, 1, True)
For i = 1 To nE
    vE(i) = vt(i)
Next i
```

```
nC = RC.Count     ' Sort vC
ReDim vC(1 To nC)
ReDim vt(1 To nC)
ReDim XC(1 To nC) As Double
For i = 1 To nC
    vt(i) = RC(i)
Next i
'vt = Application.WorksheetFunction.Sort(vt, 1, 1, True)
For i = 1 To nC
    vC(i) = vt(i)
Next i

nT = nE + nC       ' Sort vE & vC
ReDim vt(1 To nT)
For i = 1 To nE
    vt(i) = RE(i)
Next i
For i = 1 To nC
    vt(i + nE) = RC(i)
Next i
'vt = Application.WorksheetFunction.Sort(vt, 1, 1, True)
median = Application.WorksheetFunction.median(vt)
qmr = Application.WorksheetFunction.Quartile(vt, 3) -
Application.WorksheetFunction.Quartile(vt, 1)

' Calculate UE, UC
UE = 0
UC = 0
For iE = 1 To nE
    For jC = 1 To nC
        If vE(iE) > vC(jC) Then
           UE = UE + 1
        Else
           UC = UC + 1
        End If
    Next jC
Next iE
' For nE and nC < 15, use exact calculation - else use Beta model
If nE + nC <= 30 Then
   so = 1# / (2# * CDbl(NZ))
   sm = 1# / CDbl(NZ)
   For j = 1 To NZ
       fOmeg(j) = 0#
       Omeg = so + (j - 1) * sm
       vOmeg(j) = Omeg
       kOmeg = (1 - Omeg) / Omeg
       For k = 1 To 50000
           For l = 1 To nE
               XE(l) = rexp(kOmeg)
           Next l
           For l = 1 To nC
               XC(l) = rexp(1)
           Next l

           Uz = 0
           For iE = 1 To nE
               For jC = 1 To nC
                   If XE(iE) > XC(jC) Then
                      Uz = Uz + 1
                   End If
               Next jC
           Next iE
           If Uz = UE Then
              fOmeg(j) = fOmeg(j) + 1
           End If
       Next k
   Next j
   ' Normalize
   tot = 0
   For i = 1 To NZ
       tot = tot + fOmeg(i)
```

```
   Next i
   For i = 1 To NZ
       fOmeg(i) = fOmeg(i) / tot
   Next i
   ' Calculate the mean and cumulative distribution
   barOmeg = 0
   For i = 1 To NZ
       barOmeg = barOmeg + fOmeg(i) * (so + (i - 1) * sm)
   Next i
   cOmeg(1) = fOmeg(1)
   For i = 2 To NZ
       cOmeg(i) = cOmeg(i - 1) + fOmeg(i)
   Next i
Else
' Large measurement model using binomial distribution
   xs = UE / (UE + UC)
   If xs >= 0.5 Then
      x = xs
   Else
      x = 1 - xs
   End If

   Ombar = LagrangeOmeg(x, nC, nE, UE, UC)

   nH = (2 * nE * nC) / (nE + nC)
   absum = nH * (1.028 + (0.75 * x)) + 2
   a = Ombar * absum
   b = (1 - Ombar) * absum
   omegabar = Ombar
   If xs < 0.5 Then
      a = (1 - Ombar) * absum
      b = Ombar * absum
      omegabar = 1 - Ombar
   End If

   'Prior
   a0 = 0
   b0 = 0
   na = a - 1
   nb = b - 1
   a = a0 + na
   b = b0 + nb

   barOmeg = a / (a + b)
   medianOmeg = qbeta(0.5, a, b)

   so = 1# / (2# * CDbl(NZ))
   sm = 1# / CDbl(NZ)
   tot = 0
   For j = 1 To NZ
       Omeg = so + (j - 1) * sm
       vOmeg(j) = Omeg
       fOmeg(j) = qbeta(Omeg, a, b)
       tot = tot + fOmeg(j)
   Next j

   For j = 1 To NZ
       fOmeg(j) = fOmeg(j) / tot
   Next j

   cOmeg(1) = fOmeg(1)
   For i = 2 To NZ
       cOmeg(i) = cOmeg(i - 1) + fOmeg(i)
   Next i

End If

'Probability H0 that Omeg-z < 0.5
PrH0 = cOmeg(NZ / 2)
prH1 = 1 - PrH0
```

```
Mann_Whitney = PrH0

End Function

Function rexp(rate As Double) As Double

' generates a random sample from the exponential distribution
' lambda * exp(-rate * x)
' The sample x is given by  x = (-1/rate) * ln(1 - u) = (-1/rate) * ln(u)
tmp = Rnd
If tmp > 0 Then
   rexp = -Log(tmp) / rate
End If

End Function

Function LagrangeOmeg(x As Double, nC As Double, nE As Double, UE As Double, UC As
Double) As Double
'Lagrange interpolation method to find posterior a and b
'shape parameters for the beta approximation to the omega_E distribution

Dim xs As Double
Dim nH As Double
Dim Y(0 To 5) As Double
Dim La(0 To 5) As Double

nH = (2 * nE * nC) / (nE + nC)
w1 = 0.2 - (1 / (1 + (4.813 * nH)))
w2 = 0.4 - (1 / (1 + (2.52 * nH)))
w3 = 0.6 - (1 / (1 + (2.111 * nH)))
w4 = 0.8 - (1 / (1 + (1.833 * nH)))
Y(5) = (nH ^ 1.1489) / (0.4972 + (nH ^ 1.1489))
Y(4) = (Y(5) * w4) + (1 - w4) * 0.5
Y(3) = (Y(5) * w3) + (1 - w3) * 0.5
Y(2) = (Y(5) * w2) + (1 - w2) * 0.5
Y(1) = (Y(5) * w1) + (1 - w1) * 0.5
Y(0) = 0.5

La(0) = 252 - (1627 * x) + ((12500 * x ^ 2) - (15875 * x ^ 3) _
      + (10000 * x ^ 4) - (2500 * x ^ 5)) / 3
La(1) = -1050 + ((42775 * x) / 6) - (38075 * 0.5 * x ^ 2) _
      + ((75125 * x ^ 3) - (48750 * x ^ 4) + (12500 * x ^ 5)) / 3
La(2) = 1800 - (12650 * x) + ((104800 * x ^ 2) - (142250 * x ^ 3) _
      + (95000 * x ^ 4) - (25000 * x ^ 5)) / 3
La(3) = -1575 + (11350 * x) + ((-96575 * x ^ 2) + (134750 * x ^ 3) _
      - (92500 * x ^ 4) + (25000 * x ^ 5)) / 3
La(4) = 700 + (14900 * x ^ 2) + (15000 * x ^ 4) - ((15425 * x) _
      + (63875 * x ^ 3) + (12500 * x ^ 5)) / 3
La(5) = -126 + (1879 * 0.5 * x) + ((-16625 * 0.5 * x ^ 2) _
      + (12125 * x ^ 3) - (8750 * x ^ 4) + (2500 * x ^ 5)) / 3

Ombar = 0
For i = 0 To 5
    Ombar = Ombar + Y(i) * La(i)
Next i

LagrangeOmeg = Ombar

End Function

Function qbeta(p As Double, na As Double, nb As Double) As Double

qbeta = Application.WorksheetFunction.Gamma(na + nb + 1) * (p ^ na) * ((1 - p) ^ nb) _
      / (Application.WorksheetFunction.Gamma(na + 1) *
Application.WorksheetFunction.Gamma(nb + 1))

End Function

Function pbeta(p As Double, na As Double, nb As Double) As Double

pbeta = 0
```

```
For i = 1 To na
    pbeta = pbeta + qbeta(p, i, nb)
Next i

End Function
```

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