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Nuclear energy — Nuclear fuel technology — Methodologies for radioactivity characterization of very low-level waste (VLLW) generated by nuclear facilities

Énergie nucléaire — Technologie du combustible nucléaire — Méthodologies pour l'évaluation de la radioactivité des déchets de Très Faibles Activité (TFA) produits par les installations nucléaires

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ISO copyright office
CP 401 • Ch. de Blandonnet 8
CH-1214 Vernier, Geneva
Phone: +41 22 749 01 11
Email: copyright@iso.org
Website: www.iso.org

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Foreword

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This document was prepared by Technical Committee ISO/TC 85, *Nuclear energy, nuclear technologies, and radiological protection*, Subcommittee SC 5, *Nuclear installations, processes and technologies*.

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Introduction

The activity concentration of very low-level waste (VLLW) is generally below a few becquerels per gram (Bq/g), which is still greater than the allowable limits for clearance waste (often 10 times to 100 times greater). It is generally accepted that due to the low levels of activity associated with this type of waste, VLLW does not require a high level of containment and isolation, as is the case for low and intermediate level waste.

To take full advantage of opportunities for directing waste to alternative waste management routes that are more advantageous, the waste should be appropriately characterized and classified. Accurate waste characterization is also crucial for the protection of people and the environment, given the lower levels of isolation or containment barriers at VLLW disposal sites (generally in ordinary landfills). Additionally, proper characterization may allow waste classification for reuse or recycling.

Although the process for radioactively characterizing waste as low-level waste (LLW), VLLW and clearance generally follows common principles, it is appropriate to establish a specific document to assist in identifying low-level waste against waste acceptance criteria on VLLW.

This document describes the methodologies and procedures for the identification of waste that can be categorized as VLLW.

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Nuclear energy — Nuclear fuel technology — Methodologies for radioactivity characterization of very low-level waste (VLLW) generated by nuclear facilities

1 Scope

This document describes methodologies for radioactivity characterization of very low-level waste (VLLW) generated from the operation or decommissioning of nuclear facilities. The purpose is to differentiate VLLW from low-level radioactive solid waste and waste below clearance levels. The aim is to effectively characterize and to demonstrate that it satisfies the criteria for VLLW.

This document focuses specifically on characterization methods of radioactive solid waste. Clearance and exemption monitoring are not covered within this document. Additionally, the characterization of liquid and gaseous wastes is also excluded from this document.

2 Normative references

The following documents are referred to in the text in such a way that some or all of their content constitutes requirements of this document. For dated references, only the edition cited applies. For undated references, the latest edition of the referenced document (including any amendments) applies.

ISO 12749-3, *Nuclear energy, nuclear technologies, and radiological protection — Vocabulary — Part 3: Nuclear fuel cycle*

3 Terms and definitions

For the purposes of this document, the terms and definitions given in ISO 12749-3 and the following apply.

ISO and IEC maintain terminology databases for use in standardization at the following addresses:

— ISO Online browsing platform: available at https://www.iso.org/obp

— IEC Electropedia: available at https://www.electropedia.org/

3.1
very low-level waste
VLLW
radioactive waste that does not necessarily meet the criteria of exempt waste, but that does not need a high level of containment and isolation and, therefore, is suitable for disposal in landfill type near surface repositories with limited regulatory control

Note 1 to entry: Such landfill type near surface repositories may also contain other hazardous waste. Typical waste in this class includes soil and rubble with low levels of activity concentration. Concentrations of longer-lived radionuclides in VLLW are generally very limited.

[SOURCE: IAEA Safety Glossary: 2022 edition]

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3.2
waste acceptance criteria
WAC
quantitative or qualitative criteria specified for the waste form and waste package to be accepted by the operator of a waste management facility

[SOURCE: IAEA Safety Glossary: 2022 edition, modified — Definition revised.]

3.3
data quality objective
DQO
process used to establish performance or acceptance criteria, which serve as the basis for designing a plan for collecting data of sufficient quality and quantity to support the goals of a study

[SOURCE: ISO 18557:2017, 3.8]

3.4
difficult-to-measure radionuclide
DTM radionuclide
radionuclide whose radioactivity is difficult to measure directly from the outside of the waste packages by non-destructive assay means

[SOURCE: ISO 21238:2007, 2.1, modified — Examples removed.]

3.5
key radionuclide
gamma-emitting radionuclide whose radioactivity is correlated with that of *difficult-to-measure radionuclides* (3.4) and can be readily measured directly by non-destructive assay means

Note 1 to entry: Also called "easy-to-measure radionuclide" or "marker radionuclide".

[SOURCE: ISO 21238:2007, 2.2, modified — Example removed.]

3.6
scaling factor
factor or parameter derived from the mathematical relationship used in calculating the radioactivity of *difficult-to-measure radionuclides* (3.4) from that of *key radionuclide* (3.5) determined from sampling and analysis data

[SOURCE: ISO 21238:2007, 2.3]

3.7
nuclide vector
fingerprint
used to infer and quantify the presence of other key nuclides

Note 1 to entry: Applying correlation factors enables estimations of *difficult-to-measure radionuclides* (3.4).

Note 2 to entry: It is a method which involves measurements of *key radionuclides* (3.5) (usually gamma emitters, e.g. ^{137}Cs, ^{60}Co) to quantify difficult-to-measure nuclides.

[SOURCE: ISO 18557:2017, 3.12]

3.8
heterogeneous waste
radioactive waste that does not meet the definition of *homogeneous waste* (3.9), including solid components and mixtures of solid components

EXAMPLE Cartridge filters, contaminated tools or instruments.

[SOURCE: ISO 21238:2007, 2.13, modified — Part of definition used to create EXAMPLES.]

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3.9
homogeneous waste
radioactive waste that shows an essentially uniform distribution of activity and physical contents

EXAMPLE Flowable wastes such as concentrates, solidified liquids and spent resins.

[SOURCE: ISO 21238:2007, 2.12, modified — EXAMPLES revised.]

3.10
destructive analysis
analytical techniques of radioactive and chemical materials using methods which involve the destruction of a sample, for example chemical and radiochemical analysis, ICP-MS or alpha spectrometry

[SOURCE: ISO 18557:2017, 3.9, modified — Definition revised.]

3.11
non-destructive analysis
NDA
analytical techniques that allow measurement of specific properties without physical destruction of the media or item

Note 1 to entry: Generally used for in situ measurements.

[SOURCE: ISO 18557:2017, 3.20]

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4 Waste acceptance criteria (WAC) for VLLW

Waste acceptance criteria (WAC) are quantitative or qualitative criteria which state the conditions by which waste can be accepted by the operator of facilities that process, store or dispose of VLLW.

WAC specify the radiological, mechanical, physical, chemical and biological characteristics of the waste packages or unpackaged waste which may be accepted into the facility.

WAC are important because they:

— ensure compliance with safety and environmental requirements;

— are designed to assist with the selection of appropriate processing and packaging options;

— prevent technological problems during processing;

— standardize waste management operations;

— ensure waste tracking.

WAC are developed so as to be relevant, concise, measurable and verifiable, provide some flexibility, and be appropriate to each waste stream. WAC ensure that the interfaces between all parties and facilities associated with the management and disposal of VLLW are clearly understood.

Waste characterization requirements are typically developed from disposal safety and/or performance assessment, and the waste acceptance criteria for disposal are derived at the same time.

The radioactivity characterization of VLLW should address the requirements of WAC and should ensure that the requirements for each stage associated with waste management and disposal are considered. It is good practice to develop and justify the requirements of WAC using a robust process, such as data quality objectives (DQO).

The requirements for radioactivity characterization should be interpreted and confirmed, and sufficient characterization should be accomplished to satisfy the requirements of the WAC.

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5 Radioactivity characterization

5.1 Principle of radioactivity characterization of VLLW

5.1.1 Requirements and limits

The main purpose of VLLW characterization is to identify conveniently this waste stream from higher-level radioactive waste (LLW) and lower-level radioactive waste (clearance waste). The general measurement methods used for the characterization of LLW and clearance waste are also applicable to VLLW. The selection of characterization methods for VLLW mainly depends on:

— regulatory requirements, including activity limits or dose rate limits;

— monitoring purpose, such as reuse, recycling or landfill disposal;

— limitations on measurement possibilities.

Activity limits can be expressed in terms of surface activity or mass activity and can be fixed for a single radionuclide or a group of radionuclides (e.g. alpha emitters, beta-gamma emitters, pure beta emitters).

The limits of dose rate of gamma emitters can be derived from activity limits and are recommended to identify and determine the classification of radioactive waste (as seen in 5.3).

5.1.2 Measurement methodology

During the radioactivity measurement of VLLW, the following considerations should be taken into account:

— surface activity measurement, i.e. in situ direct measurement, consists mostly of beta and gamma measurement;

— results of in situ direct measurement can be used to show the preliminary distribution of the contamination and confirm “active spots”;

— surface activity measurements can guide targeted sampling and associated gamma spectrometry or destructive analysis;

— alpha particle, soft beta radiation as well as low energy gamma radiation are difficult to detect by in situ direct measurement;

— measurement of alpha particles and soft beta radiation typically requires radio-chemical analysis or spectroscopic analysis to define the composition of mixed radionuclides;

— radio-chemical analysis of low activity requires sufficient samples to facilitate easier measurement and to improve the accuracy of specific activities;

— alpha contamination measurement of VLLW from reactors is usually unnecessary unless cladding ruptures have occurred;

— for mixtures of radionuclides, easily detectable radionuclides (e.g. ^{60}Co) can be used as contamination indicators to determine quickly the activity level of radioactive waste;

— selection of measurement apparatus should be based on the activity limits and characteristics of the apparatus;

— the apparatus should be calibrated following the standards of various energies; the detection thresholds and background levels should be regularly checked to prevent any major error;

— the uncertainty should be carefully considered during the sampling and measurement.

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5.2 Process for radioactivity characterization of VLLW

5.2.1 General

The following steps are considered good practice for identification of waste that may be categorized as VLLW:

— investigation of waste properties;

— surface scanning, dose rate assessment;

— theoretical calculation and measurement of waste activity;

— activity measurement by means of destructive analysis.

The process is shown in Figure 1.

The steps in the process are further described in 5.2.2 to 5.2.5.

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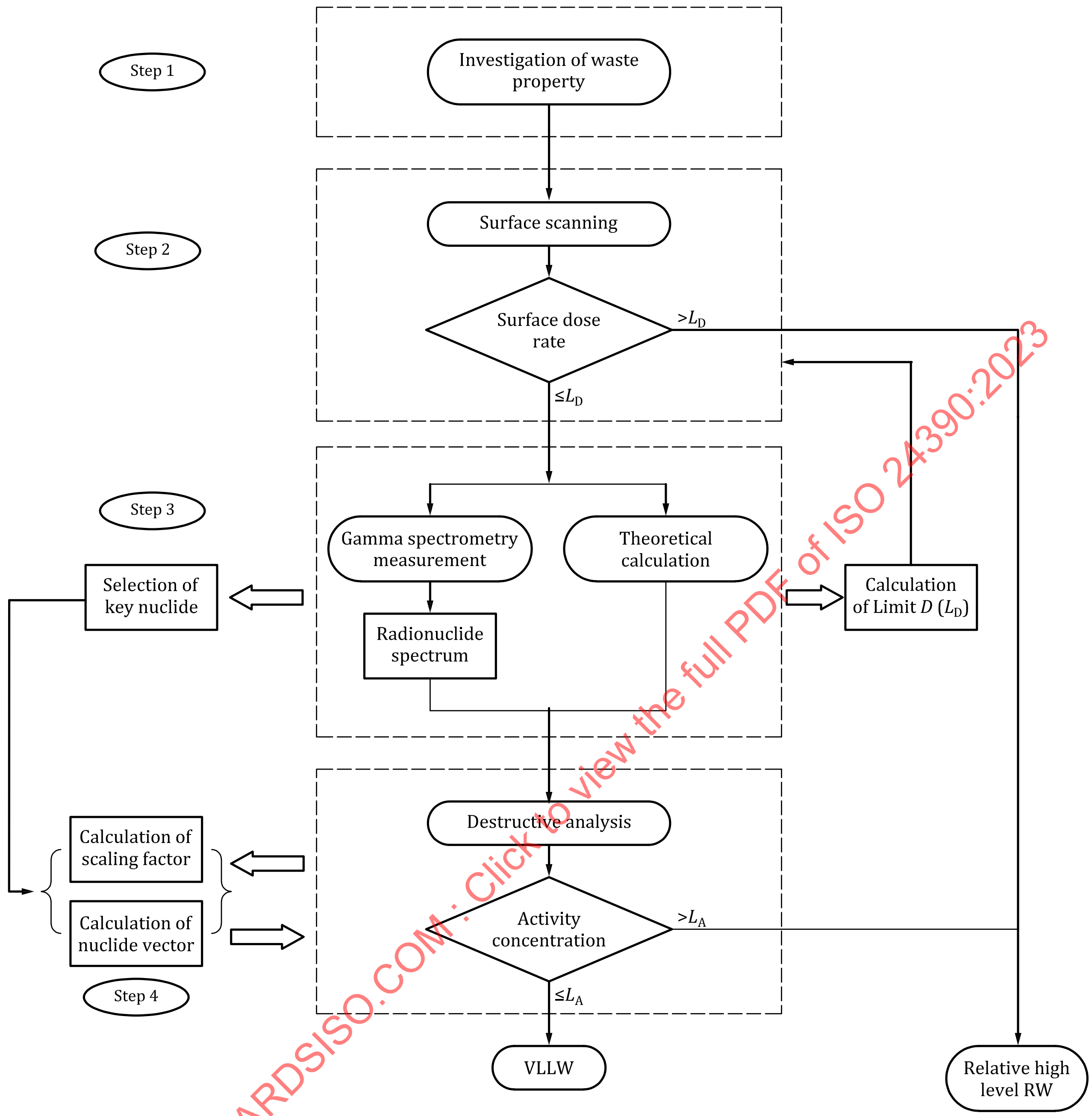

Key

L_A activity limit, in becquerels

L_D dose rate limit, in grays per hour

Figure 1 — VLLW radioactivity characterization process

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5.2.2 Step 1: Investigation of waste characteristics

The objective of the first step is to obtain as much information as possible about the properties of the waste. This involves collecting, reviewing and analysis of all available data, guided by the DQO process. The aim is to define the waste characterization objectives and associated sampling and analysis

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strategy against the appropriate waste acceptance criteria. Typically, the following information, at a minimum, should be considered based on the specific waste stream:

- basic information, including:
  - the process by which the waste was generated;
  - the location of the waste;
  - the radiological history of the waste;
  - the processing technology and storage conditions;
- physical properties, including:
  - waste volume or weight;
  - quantity of waste;
  - form of waste;
  - material composition;
  - geometry;
  - moisture content;
  - containment, if any;
- chemical properties, including:
  - pH;
  - oxidation reducibility;
  - thermal stability;
  - presence of organic and inorganic compounds;
- radiation characteristics, including:
  - distribution of radioactivity (i.e. homogenous or heterogeneous);
  - surface activity;
  - radionuclides present;
  - radionuclide concentrations;
  - surface dose rates;
  - results from any previous measurements or characterization campaigns, including surface scans.

VLLW can be generated from various processes at nuclear facilities, and the radiation characteristics of waste from different sources can vary significantly. The characteristics of the waste are likely to be different from those of other radioactive sources, depending on the source of contamination.

The radiation properties of waste generated from uranium enrichment, conversion and fuel fabrication are relatively well-defined, simple and stable. The radioactive contamination in such waste is mainly uranium and its daughter nuclides. Waste generated from nuclear power plants (NPPs) has relatively clear and stable radiation properties, but it is more complex. This waste could be contaminated with alpha, beta and gamma-emitting radionuclides. A large number of different radionuclides are generated as a result of fission, activation and decay. The radiation properties of VLLW generated

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from reprocessing of spent fuel are very complex. Radiation sources in such waste include fission and activation products.

The application example for the characterization process of the three different waste streams is presented in Annex A.

During the investigation of waste properties, the integrity and reliability of the waste characterization information should be assessed and updated regularly. This information will be used to establish the scaling factors or ratios between different radionuclides. Depending on the specific radionuclides present in the waste, verification can also be achieved through destructive sampling analysis.

5.2.3 Step 2: Surface scanning

5.2.3.1 General

Surface scanning is typically performed using portable instruments to detect surface dose rates and measure surface contamination. Portable instruments are commonly used for this purpose. During the operation of these instruments, several factors should be considered, including the environmental background level, instrument resolution, detection limit and range and alarm threshold.

5.2.3.2 Surface dose rate detection

The surface dose rate detection is used to:

— prevent unnecessary radiation exposure during radioactivity measurement;

— identify hot spots inside the waste;

— measure the surface dose rate and determine the radioactivity level of waste following Formula (1) and Formula (2) as specified in 5.3. As shown in Figure 1, if the measured result exceeds L_D or L_A, the radioactive waste cannot be classified as VLLW.

The uncertainty of surface scanning results from the uncertainty in instrument calibration and the measurement process. For the dose rate detector, the uncertainty of calibration includes:

— the choice of primary or secondary standard instrument;

— the distance of the instrument from radiation source;

— the uniformity of the radiation beam;

— the variation of background dose rate;

— variation of environmental conditions.

The uncertainty of dose rate detection includes:

— variations of environmental conditions during detection (such as background, temperature, humidity, etc.);

— the characteristics of apparatus (such as background, efficiency, detection limit, stability, etc.);

— method of measurement, including speed of probe movement, distance to the surface of contamination and position of counts reading.

5.2.3.3 Surface contamination measurement

Before conditioning and treatment of radioactive waste, surface contamination measurement is usually used to initially estimate the characteristics of radioactive contaminants. The purpose of surface contamination measurement is to prevent unnecessary contamination during radioactivity measurement and to decrease the radioactive level through decontamination.

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The principles, wipe test method and apparatus calibration of surface contamination measurement are given in ISO 7503-1, ISO 7503-2, ISO 7503-3, respectively.

5.2.4 Step 3: Gamma activity measurement

Given the potential waste stream generators identified in 5.2.2, gamma activity measurement can usually be achieved through direct measurement. The objectives of gamma measurement include:

— obtain the γ spectra;

— identify possible γ radionuclides in the waste;

— identify easily measurable key radionuclides;

— provide input for the calculation of scaling factors between difficult-to-measure radionuclides and key radionuclides;

— provide input for the calculation of ratios between dose rate and activity concentration;

— verify the relevant parameters of radiation characteristics obtained in the first step, such as radiation contamination type, possible radionuclide type and previous surface scans.

It is good practice to measure gamma activity through non-destructive analysis (NDA) means, ideally through use of a system capable of acquiring a gamma spectrum and suitable for the measurement of waste containing high energy γ radionuclides (e.g. ^{60}Co, ^{137}Cs, ^{51}Cr, ^{54}Mn).

The measurement results may be given in units of total activity or activity concentration. For VLLW, it is recommended that the minimum detectable activity (MDA) is selected based on the activity measurement that is most advantageous in differentiating VLLW from other categories of waste.

Gamma activity measurement can be performed using different techniques, such as segment gamma spectrum, rotating gamma scanning or tomographic gamma scanning. The implementation, calibration and quality control of γ spectrometry should be carried out following the requirements given in ISO 19017. Alternatively, the theoretical activation calculation method can be used following the range method given in ISO 16966.

As shown in Figure 1, the results of the activity measurement, together with the analysis performed, can be used to identify the key radionuclides and calculate the judgment limits of the dose rate. L_D represents the dose rate limit, which can be calculated following Formula (1) and Formula (2) in 5.3. The identified key radionuclides will be used to establish scaling factors in step 4. L_D is used to judge the measurement results of surface scan in the next step.

The uncertainty in the gamma activity measurement arises from various sources, including:

— counting;

— measurement geometry;

— activity;

— matrix distribution within the measured container;

— container size and wall thickness;

— self-attenuation of source;

— self-attenuation of matrix;

— calibration and correction of measuring equipment;

— operator error;

— reading and processing of measurement data.

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The quantified uncertainty of the γ measurement should include the coverage interval (the range in which the true value lies) and the confidence level. The uncertainty calculation of γ spectrometry can be found in ISO 19017:2015, 6.4.

5.2.5 Step 4: Destructive analysis

Destructive analysis, also known as radiochemical analysis, is an essential process that typically provides the most accurate activity concentration of radionuclide, especially for DTM radionuclides. Samples for destructive analysis are usually taken under the following conditions:

— when it is a new waste stream;

— when the radionuclide in the waste stream cannot be confirmed through direct measurements;

— when activity concentration measurement of key radionuclides and DTM radionuclides data are not available for establishing correlation factors or coefficients.

It is good practice to periodically verify the validity of defined radionuclide vectors for existing waste streams through sample and destructive analysis. This should also be done when changes in the nuclide vector can be expected due to events or changes in the operation.

The destructive analysis process involves sampling, sample preparation and, often, chemical separation, prior to any measurement being made. Sampling is a critical step in the destructive analysis. The design of the sampling process and tool, the homogeneity of the waste and the representativeness of the sample should be considered.

Information for sampling is provided in Clause 6.

Sample preparation may include techniques such as:

— dissolution;

— mineralization;

— chemical separation.

Additional information can be found in Tables X and XI of IAEA TECDOC 1537:2007.

The activity concentration of the relevant radionuclides, obtained from destructive analysis, is used to establish the scaling factor and the radionuclide vector, from which the waste is classified. If the measured activity concentration exceeds L_A (as defined in 5.3), the waste cannot be classed as VLLW. If the measured activity concentration does not exceed L_A, the waste should be classified as VLLW.

Background levels should be considered, as they can affect the ability to identify waste as VLLW. It can be necessary to reduce background levels to achieve a sufficiently low limit of detection.

5.3 Decision thresholds

The decision thresholds identified in Figure 1 consist of the dose rate limit (L_D) and the activity limit (L_A). L_A represents the activity limit for a given radionuclide or group of radionuclides in the waste stream, expressed as an activity concentration (per unit value). The activity concentration limit should be in accordance with the national standards on classification of radioactive waste and WAC.

For a single radionuclide and point source, the dose rate limit can be calculated following Formula (1).

$$L_D = L_A \times \tau / r^2 \quad (1)$$

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where

L_D is the dose rate limit, expressed in grays per hour;

L_A is the activity limit, expressed in becquerels;

r is the distance from the point source, expressed in metres;

τ is the conversion coefficient of radionuclide, expressed in grays times square metres times hours per becquerel.

For radionuclide mixtures, the dose rate limit can be calculated following Formula (2).

$$L_D = \sum_{i=1}^{i=n} L_{A,i} \times \tau_i \times V_i / r^2 \quad (2)$$

where

L_D is the dose rate limit, expressed in grays per hour;

$L_{A,i}$ is the activity limit of radionuclide i, expressed in becquerels;

r is the distance from the point source, expressed in metres;

τ_i is the conversion coefficient of radionuclides i, expressed in grays times square metres times hours per becquerel

V_i is the weighting factor of single radionuclide, which is calculated by activity based on the gamma spectrometry by using Formula (4).

5.4 Correlation of measurement methods

Figure 2 shows the method by which the results obtained from the various measurements discussed in 5.2.3 to 5.2.5 (surface scanning, gamma spectrometry and destructive analysis) are used to enable identification and classification of VLLW.

The measurement techniques can be applied independently or in combination, as appropriate for a particular waste stream.

Dose rate is often measured by surface scanning, activity concentration of key radionuclides by gamma spectrometry and activity concentration of DTM radionuclides by destructive analysis.

This enables a correlation between dose rate and activity concentration (for relevant radionuclides) to be defined, which facilitates identification and classification of VLLW based on a dose rate measurement.

Scaling factor and radionuclide vector are described in 5.5 and 5.6, respectively.

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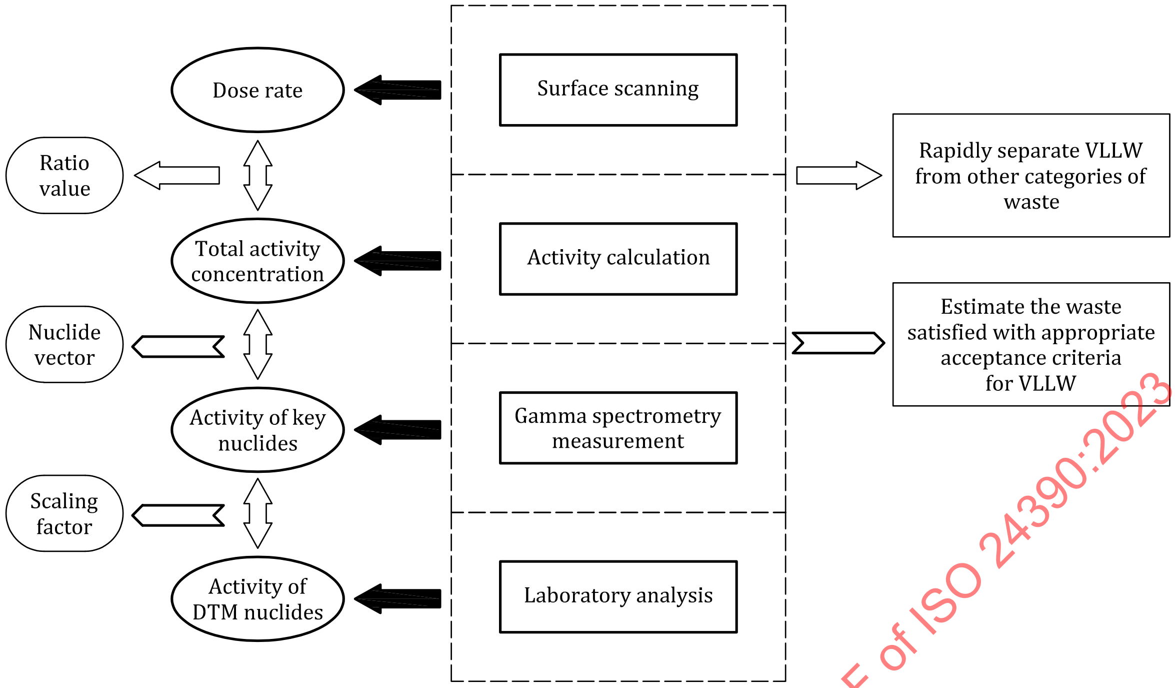

Figure 2 — Correlation of measurement methods for radioactivity characterization of VLLW

5.5 Scaling factor method

It is good practice to use the scaling factor method to evaluate the activity concentration of DTM radionuclides. This is done by applying the scaling factor (SF) that has been established based on the relationship that exists between DTM radionuclides and key radionuclides (KN), such as ^{60}Co and/or ^{137}Cs for the waste stream.

The relationship between the DTM nuclide and the key nuclide can be calculated using Formula (3).

$$f_{SF} = \sqrt[n]{\left(a_{d,1}/a_{k,1} \times \cdots \times a_{d,i}/a_{k,i} \times \cdots \times a_{d,n}/a_{k,n}\right)} \quad (3)$$

where

$a_{k,i}$ is the activity concentration of the key radionuclide in the sample i (i = 1...n), typically expressed in becquerels per kilogram or in becquerels per cubic metre;

$a_{d,i}$ is the activity concentration of the DTM radionuclide in the sample i (i = 1...n), typically expressed in becquerels per kilogram or in becquerels per cubic metre;

n is the number of samples.

Depending on the nonlinear relationship between the DTM radionuclide and the key radionuclide, the scaling factor (SF) between the DTM radionuclide and the key radionuclide can be evaluated by linear regression of logarithm or other statistical methods. Details of scaling factor methods can be found in ISO 21238.

The main sources of uncertainty of the scaling factor method are representative sampling and laboratory analysis. The uncertainty can be calculated based on the number of samples and standard deviation. Further information for evaluating the uncertainty on scaling factors can be found in ISO 21238:2007, 5.6. The practices and examples of application of the scaling-factor method can be found in ISO 21238:2007, Annex A.

For further information on radionuclide vectors, see ISO 21238.

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5.6 Radionuclide vector method

The radionuclide vector is the relative proportion of the activity concentration of the individual radionuclides in relation to the total activity concentration and can be calculated using Formula (4).

$$V_i = a_i / \sum a_i \quad (4)$$

where

a_i is the activity concentration of radionuclide *i*, usually expressed in becquerels per kilogram or in becquerels per cubic metre;

V_i is the relative proportion of the radionuclide *i* in all involved radionuclides of the waste stream.

Establishing the radionuclide vector involves determination of the waste stream, selection of radionuclides to be analysed (including key radionuclide), creation of a sampling plan, sampling, radiochemical characterization of samples, calculation of the radionuclide vector and update of the radionuclide vector.

For further information on radionuclide vectors, see ISO 18557.

6 Sampling

6.1 General

There is typically a requirement to collect adequate and representative samples for developing methods for identification and characterization of VLLW for a given waste stream, and for confirming that these methods remain valid.

Waste will arise from different nuclear facilities and will have different activity signatures. The processes which generated the waste, the constituents of the waste stream, storage locations and conditions, physical properties and chemical properties of the waste will be different. It is, therefore, necessary to establish and justify a robust sampling plan in order to ensure the accuracy of radiochemical characterization.

It is good practice to employ the DQO process to design the sampling and analysis plan. This ensures that the plan is demonstrably aligned with the characterization objectives, as established by the DQO review.

Additional guidance for planning waste sampling can be found in ASTM D4687, which includes the development of a sampling plan, the selection of sampling points and the design of sample numbers.

Statistical sampling methods are preferred for radioactivity characterization for VLLW. Guidance for sampling of homogeneous waste and heterogeneous waste can be found in 6.2 and 6.3, respectively.

Judgemental sampling may be considered to identify the upper limits of potential contamination. In some cases, it may also be considered as a screening method to reduce the amount of sampling required, such as when a hot spot is below the threshold of VLLW. However, justification should always be provided for using judgemental sampling.

To develop the sampling strategy, it is necessary to collect detailed information about the waste stream, as discussed within step 1 of the characterization process (see 5.2). Having more available information will increase the effectiveness and efficiency of sampling, yielding associated benefits in terms of safety, cost and time.

It is important to note that unknown materials within the drum or other container from which the sample is taken can pose potential health risks to the operator performing the sampling.

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Therefore, appropriate safety precautions should be taken during the sampling. Further guidance can be found in ASTM D4687.

6.2 Homogeneous waste

Homogeneous waste is relatively uniform in terms of physical, chemical and radiological properties. The following steps should be considered to determine the homogeneity of waste:

— The homogeneity of the waste should be initially determined according to the basic information of the waste properties collected during the first step of characterization (see Clause 5).

— The homogeneity can be confirmed based on reviewing the operation history, the waste generation processes, waste management records and interviews with operational personnel.

— During the sampling process, the homogeneity of the waste should be rechecked to ensure consistency.

— Direct scanning methods, such as γ-camera or dose rate measurement, can be employed to confirm homogeneity of the waste. These methods are useful for identifying and locating any hot spots within the waste.

— Alternatively, direct measurement of key radionuclides within the waste can be used to confirm the homogeneity of waste.

— When possible, mixing is recommended before sampling.

Radiography can also be considered as a method to determine the homogeneity of waste.

6.3 Heterogeneous waste

For heterogeneous waste streams, which may include cartridge filters, contaminated tools or instruments from a nuclear facility, it is necessary to use suitable sampling tools to collect samples.

Guidance for the selection and fabrication of sampling equipment can be found in sections 7.1 and 7.2 of ASTM D4687.

In addition, ASTM D5956 offers guidance on developing sampling strategies for heterogeneous waste streams. The resource addresses the challenges of sampling heterogeneous waste and provides a stratification method that can be employed.

6.4 Sampling uncertainty

A key term in sampling is “representative”. The term refers to the degree of uncertainty regarding whether the sample accurately represents the content of the waste stream. Sampling uncertainty is often a major component of the uncertainty in waste characterization.

Whether a sample can be considered representative or not is determined by the amount of uncertainty that can be tolerated. In the case of waste sentencing, higher uncertainty typically results in waste that could be sentenced down to the VLLW route being sentenced to other, more expensive, routes. Hence, there are benefits to be derived from reducing the sampling uncertainty as much as possible.

The definition of the representativeness of a sample and how it will be achieved are important considerations.

General guidance for representative sampling can be found in ASTM D6044.

The number of samples is a means of improving representativeness. Guidance on establishing the minimum number of samples required for representative sampling is provided in ASTM D4687.

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7 Data quality objectives (DQO)

It is considered good practice to utilize a robust process, such as the data quality objectives (DQO) process, to establish and justify the requirements for collecting data and information to support the management and disposal of radioactive waste and for confirming that those requirements have been met.

The DQO process helps identify the data and information acquisition objectives, the appropriate boundary conditions for collecting data, the confidence level (the maximum acceptable levels of decision errors) and the optimization of data collection activities.

General guidance on the DQO process can be found in Reference [16]. The usage of the DQO process for the sampling design of heterogeneous materials is described in ASTM D5956. Usage of the DQO process for environmental data related to waste management activities is described in ASTM D5792.

A typical DQO process includes seven steps: stating the problem, identifying possible decisions, identifying inputs to decisions, defining boundaries, developing decision rules, specifying limits on decision errors and optimizing data collection design.

Although the DQO process is used most often for the development of sample campaigns, it is equally applicable to data and information that is acquired through any mechanism. This includes measurement.

8 Quality assurance

8.1 General

Quality assurance is essential to improve the accuracy and reduce the uncertainty of radioactivity measurement. A quality assurance programme for radioactivity characterization should be established, encompassing the methods for collecting data and information. It includes the testing laboratory, measurement instruments, documents and the personnel involved, which should be clearly described.

Implementing a quality assurance programme is considered good practice in general and is also a requirement of the DQO process.

8.2 Laboratory

The laboratory is responsible for the impartiality and confidentiality of the sample analysis in the laboratory.

ISO/IEC 17025 specifies the general requirements for laboratory competence, impartiality and consistent operation, including analysis methods, operational procedure and management systems.

The requirements of quality control samples can be found in ASTM D4687. The detailed quality assurance requirements of gamma spectrometry measurements are provided in ISO 19017:2015, Clause 7.

8.3 Measuring instruments

The available instruments and equipment to be used for the characterization activities should be documented. This documentation includes specification, design, build, deployment, calibration and use, as appropriate. The calibration, transport, storage, use and maintenance of instruments should be performed following established procedures.

Measuring (pure) alpha-emitters by dose-rate measurement devices will give completely wrong results or other scenarios. Therefore, it is crucial to ensure that the measurement device is sensitive to the type of radiation being measured.

A calibration programme for the measuring instruments should be established. The measuring instruments shall be calibrated by the accredited and specialist organizations (as appropriate), with

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the periodicity that is defined within the programme. The instruments should also be subjected to an appropriate quality control programme.

8.4 Personnel

Personnel involved in characterization activities must meet the required competence standards, which encompass various areas such as sampling, testing, operating, data analysis, calibration and quality management.

An assessment of personnel competence should be made based on education, qualification, training, technical knowledge, skills and experience.

8.5 Documentation and procedures

All of the activities involving measurement, sampling and calculation should be documented and recorded. The important documents and procedures should be reviewed and approved before becoming effective, including the characterization plan, sampling plan, personnel training plan, radionuclide analysis procedure, calibration procedure and management procedure of measurement instruments.

This is good practice in general and is a requirement of the DQO process.

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