Characterization of Conditioned Nuclear Waste for its Safe Disposal in Europe

CHANCE project

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SYNTHESIS OF COMMONLY USED METHODOLOGY FOR CONDITIONED RADIOACTIVE WASTE CHARACTERIZATION

DELIVERABLE (D2.2)
Work Package 2

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1. Introduction

The CHANCE project aims to address the specific issue of the characterization of conditioned radioactive waste. The characterization of fully or partly conditioned radioactive waste is a specific issue because unlike for raw waste, its characterization is more complex and needs specific non-destructive techniques and methodologies. There are different and varying reasons for this: 1) conditioned waste may no longer be in its initial form (e.g., due to incineration), 2) conditioned waste is typically embedded or surrounded by a matrix, 3) conditioned waste may contain wastes that come from different primary sources and therefore the radiological spectrum might become more complex. Characterization issues within CHANCE encompass both physico-chemical characterization and radiological characterization.

The first objective of the CHANCE project is to establish at the European level a comprehensive understanding of current conditioned radioactive waste characterization and quality control schemes across the variety of different national radioactive waste management programmes, based on inputs from end-users members such as Waste Management Organizations and storage operators. CHANCE will focus on the following waste forms (IAEA classification):

- Very Low Level Waste (VLLW);
- Low Level Waste (LLW);
- Intermediate Level Waste (ILW);
- High Level Waste (HLW).

The second objective of CHANCE is to further develop, test and validate techniques already identified that will improve the characterization of conditioned radioactive waste, namely those that cannot easily be dealt with using conventional methods. Specifically, the work on conditioned radioactive waste characterization technology will focus on:

- **Calorimetry** as an innovative non-destructive technique to reduce uncertainties on the inventory of radionuclides;
- **Muon Tomography** to address the specific issue of non-destructive control of the content of large volume nuclear waste;
- **Cavity Ring-Down Spectroscopy** (CRDS) as an innovative technique to characterize outgassing of radioactive waste.

The present report focuses on activities from Work Package 2 (Task 2.1) related to the first objective of the CHANCE project, namely the establishment of an overview of current conditioned radioactive waste characterisation employed over Europe, remaining issues, etc. In order to collect information in terms of characterization, particularly links and overlaps between these requirements and waste specifications for different national disposals, a questionnaire was produced and distributed to operators of radioactive waste disposal in Europe through the CHANCE project End-User Group.

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This document presents a synthesis of the answers received from members of the End Users Group (EUG) to the questionnaire. Based on the answers collected by means of this questionnaire the requirements and methodologies for the characterization of conditioned radioactive waste used in different national contexts were evaluated. The questionnaire was also used to understand socio-ethical and technical frameworks of radioactive waste characterisation practices and policies.

The questionnaire consisted of 18 questions and its aim was to get an overview of the end-users needs for the characterisation of conditioned radioactive waste and to appreciate how the members of the End User Group (EUG) emphasise the requirement of waste characterization within each of their organizations in relation with the development of the national disposal programme and with respect to the importance of managing the whole back-end of the fuel cycle as well as the uncertainty management.

13 responses were received from the EUG and the CHANCE WP2 members, from Belgium, France, Italy, Germany, Poland, Romania, Spain, Sweden and UK. 8 of these were from national waste management organizations and 5 were from research institutes. All information received was complemented with information from the NEA general reports [1, 2, 3] and country specific reports [4] on their waste inventory and waste management programs. It was used to identify:

- key parameters required for characterization;
- technologies/methods commonly used for the characterization of conditioned waste;
- waste acceptance criteria applied and the possibilities of their harmonization in Europe;
- specific problematic issues for the characterization of conditioned radioactive waste;
- current R&D needs and the potential of on-going R&D programmes on conditioned radioactive waste characterization;
- potential applications of R&D actions included in CHANCE;
- socio-ethical and technical issued associated with the waste characterization process.

From publicly available documents, only information relevant to questions 1 (regarding types of radioactive waste) and 2 (regarding storage / disposal options) of the questionnaire have been found and utilized; however no specific information regarding radioactive waste characterisation. Only the NEA report 7373 [3] “Radiological Characterisation from a Waste and Materials End-State Perspective: Practices and Experience” provides some general information about characterisation, but it is restricted to the characterisation of waste and contaminated materials from the decommissioning of nuclear facilities and does not address the characterisation of conditioned radioactive waste.

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2. Synthesis of the questions included in the CHANCE questionnaire

2.1 What types of radioactive waste, including spent nuclear fuel (SNF), are managed by your organization?

For this question, EUG members were asked to specify the origins and the types of the waste they are managing and a table with the waste classification according to the IAEA General Safety Guide No. GSG 1 [5] was provided, but each member was advised to adapt the table provided to account for the classification scheme applied in their organization or country.

Although the IAEA Safety Guide 111-G-1.1 [5] was replaced in 2009 by IAEA Safety Guide GSG-1 [6], the older classification scheme was accepted by the European Commission (EC) for reporting, and it is still used in the current classification schemes of many European countries.

The classification scheme recommended by IAEA Safety Guide 111-G-1.1 [5] defines three main classes of radioactive waste: exempt waste (EW), low- and intermediate-level waste (LILW), with subclasses of short-lived (LILW-SL) and long-lived (LILW-LL), and high-level waste (HLW), and some values of characteristics to distinguish these classes.

In the IAEA Safety Guide GSG-1 [6], the waste classes were restructured and several new classes were introduced. This new scheme allows for the classification of RW inventorying purposes and addresses: exempt waste (EW), very short-lived waste (VSLW), very low-level waste (VLLW), low-level waste (LLW), intermediate-level waste (ILW), high-level waste (HLW).

In comparing the two IAEA schemes, the LILW-SL and LILW-LL from the IAEA Safety Guide 111-G-1.1 can be considered equivalent to the LLW and respectively ILW in GSG-1, since these categories in both schemes are distinguished by the amount of long-lived radionuclides present.

Some countries we received answers from adopted a waste classification scheme in agreement with the recent classification scheme proposed by IAEA GSG-1, while others adopted the older IAEA 111-G-1.1 classification schemes in their national recommendations.

In general, the origin of the waste is similar: low and very low level waste generally originate from medical and scientific research activities while with an increasing level of radiation the relation to nuclear (or research) power plants become more obvious: operational and decommissioning wastes and those wastes resulting from fuel cycle activities contribute significantly to the ILW and HLW categories.

For many organizations, historical wastes are a major concern for the radioactivity content and chemical characteristics.

In ***Austria***, radioactive wastes are generated mainly from medicine, industry, research, decommissioning/dismantling of nuclear research facilities. The classification scheme is in agreement with recommendation of IAEA 111-G-1.1, but a class of transition RW is foreseen between EW and LILW-SL. Austria has currently no HLW, and the SNF was generated from research reactor (RR).

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In ***Belgium***, the RW are classified in three categories: Category A: Low- and medium-level short-lived waste, Category B: Low- and medium-level long-lived waste and Category C: Long-lived high-level waste. Category A contains small amounts of mainly beta and gamma emitters with half-lives shorter than 30 years and traces of longer-lived emitters and is generated mainly from the operation of nuclear power plants, but also from reprocessing, research, production of radioisotopes and their use in nuclear medicine and industry. Category B contains mainly alpha emitters with half-lives exceeding 30 years, together, in some cases, with intermediate amounts of beta and gamma emitters. It is derived mostly from the fabrication of nuclear fuels, from nuclear research and the reprocessing of SNF. Category C contains substantial amounts of beta and gamma emitters with short and medium half-lives, together with longer-lived alpha emitters; one of its major characteristics is the considerable heat emission. It derives mostly from the reprocessing of SNF and from research.

The RW currently present in ***Croatia*** consists of waste and disused sources from medicine, industry, science and education, and are classified following the IAEA 111-G-1.1 recommendations in: exempt and cleared RW, LILW-SL, LILW-LL and HLW. Supplementary, a class of Low-level short-lived RW was introduced for the RW containing mainly radionuclides with half-lives below 100 days that decay below clearance levels within 3 years.

In ***Cyprus***, low volume of RW is generated from medicine, industry and research, and the classification scheme follows the recommendations of IAEA GSG-1. In Cyprus, currently there are no HLW and SNF.

In ***Czech Republic***, the RW are generated from nuclear power plants, medicine, industry, and research and are classified according to the IAEA 111-G-1.1 Additionally, a class of temporary or transient RW (VSLW) is included for the waste decaying rapidly to EW in 5 years.

In ***Denmark***, the RW are mainly generated from decommissioning of former RRs, but also in limited amounts from medicine, industry and research, and are classified according to the GSG-1 in three main classes: VLLW, LLW and ILW. Based on the fact that no special precautions for heat dissipation are necessary and based on the activity concentrations, the SNF is designated as ILW.

In ***Estonia***, the RW are generated mainly from decommissioning of nuclear power plant sites, medicine, industry, and research, and are classified based on recommendations of IAEA 111-G-1.1. Additionally, a class for short-lived RW are considered for the waste that decays to EW in 5 years.

In ***Finland***, the RW are generated both from institutional activities and for nuclear power production and are classified in EW, LILW, HLW (SNF), short-lived waste and long-lived waste. Finland has two classification systems: one for the purpose of processing and storage and one for the purpose of disposal.

In ***France***, based to the activity level, the classification scheme comprises of four main categories (according to the recommendations of IAEA GSG-1): VLLW, LLW, ILW and HLW. Then each waste category is subdivided according to half-lives of the radionuclides it contains: very short lifetime (half-life below 100 days), short lifetime (half-life below 31 years), and long lifetime (half-life above 31 years). The very-short-lived waste, generated mainly from medical use (half-life below 100 days) is managed through radioactive decay and then disposed of as domestic waste.

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The RW generated in ***Germany*** from medicine, industry, research, and operation and dismantling of nuclear power plants, as well as from SNF reprocessing, are ranked into three classes: EW, negligible heat generating RW and HLW (including SNF).

In ***Greece***, the RW are generated from medicine, industry, research, decommissioning of RR and the classification scheme includes all 6 classes recommended in IAEA GSG-1. Greece has no HLW, and all SNF from RR will be returned to the US.

In ***Italy***, RW are classified into three categories on the basis of the radioisotopes characteristics and concentrations, and considering the possible options for final disposal as main guiding criterion: Category I: waste that decays in a few months to radioactivity level below safety concerns (mainly hospital and research waste with half-life below 1 year); Category II: waste which decay to radioactivity level of few hundreds of Bq/g within few centuries; Category III: long lived waste not included in category I and II; high level waste from reprocessing of SNF and alpha emitting waste from the fuel cycle and R&D activities.

In ***Lithuania***, the RW is generated from past NPP operation and future decommissioning, as well as from medicine, industry, research, are is classified in: VLLW-SL (Class A), LILW-SL (Class B + C), LILW-LL (Class D + E) and Spent Sealed Sources (Class F).

There are no formalised categories for the RW generated in ***Malta*** from Medicine, industry, research.

In ***Netherlands***, the RW generated from nuclear industry, medicine, industry, and research are classified in: EW, short-lived waste, low and intermediate level waste (LMRA) and high level waste (HRA, includes SNF). Four types of LMRA are distinguished: alpha-bearing waste, waste from NPP, waste with half-life exceeding 15 years and waste with half-life below 15 years.

The RW generated in ***Norway*** from medicine, industry, research are categorized in EW (as in IAEA GSG-1), and three classes as recommended by IAEA 111-G-1.1: LILW-SL, LILW-LL and SNF.

In ***Poland***, based on the activity content, the RW are classified in three main categories: low-level activity waste, intermediate level activity waste and high-level activity waste, and the first two categories are further divided, based on activity and the half-life of the radionuclides content in 3 sub-categories: transitional, short-lived and long-lived. The spent sealed sources are treated as a distinct waste category.

In ***Romania***, both institutional RW (from research, medicine, industry) and RW from nuclear power plant are generated. The RW classification scheme follows the recommendation of IAEA 111-G-1.1 and RW are divided in four main classes: VLLW (short lived radionuclides and activity concentration above clearance levels), LILW-SL (contains radionuclides with half-life below 30 years and activity concentration above the threshold for VLLW, but whose radioactive contents and heat output are lower than those of HLW), LILW-LL (contains radionuclides with half-life above 30 years but the radioactive contents and heat output are lower than those of HLW) and HLW (contains mainly long-lived radionuclides and has a radiogenic heat output which requires special considerations for handling or final disposal; SNF belongs to this HLW class). There are also two other classes of RW in Romania: exempt waste and transition waste (very short-lived waste suitable for decay storage prior to release).

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In ***Slovak Republic***, the classification scheme for the RW generated from NPPs, medicine, industry, and research follows the recommendation of IAEA GSG-1 and comprises of seven classes: EW, transient RW, very low activity RW, low activity RW, medium activity RW, highly active RW and SNF.

In ***Slovenia***, the solid RW generated from NPP, medicine, industry, and research are categorized, based on the level and type of radioactivity, in five classes: transitional RW, VLLW (for which the competent regulatory body for nuclear and radiation safety may approve conditional clearance), LILW (with insignificant heat generation), HLW (which contains radionuclides whose decay generates such an amount of heat that it has to be considered in its management). According to the IAEA 111-G-1.1, the LILW are divided in two groups: LILW-SL (containing radionuclides with half-life below 30 years and specific activity of alpha emitters equal to or lower than 4,000 Bq/g for an individual package, but on average not higher than 400 Bq/g in the overall amount of LILW) and LILW-LL (for the waste with specific activity of alpha emitters exceeds the limitations for LILW-SL).

In ***Spain***, the RW are generated mainly from NPPs and in a lesser extend from small producers like radioactive facilities, research centres, etc. The RW are classified in VLLW (with main beta - gamma and alpha emitters below 100 Bq/g and 10 Bq/g respectively), LLW (activity is mainly due to the presence of beta or gamma radionuclides with half-life less than 30 years and very low and limited amount of long lived radionuclides), ILW and HLW (contain significant amount of long lived isotopes, with half-life more than 30 years, and cannot be disposed in the same facilities as VLLW and LLW. In order to define the requirements for characterization and conditioning level, the LLW are divided in two groups: Level 1 (low activity values) and Level 2 (higher activity values than Level 1).

In ***Sweden*** there is an established waste classification scheme used by the nuclear industry derived from existing and future repositories. Based on the radioactivity content, the RW are classified three categories: high activity (waste with decay heat > 2 kW/m^3 that needs cooling and shielding), intermediate activity (waste with surface dose rate exceeding 2 mSv/h that needs shielding but no cooling) and low activity (waste with surface dose rate below 2 mSv/h that needs no shielding or cooling). Based on the radionuclides content, the RW are ranked in short lived (dominating content of radionuclides with a half-life below 31 years) and long-lived (significant content of radionuclides with a half-life higher than 31 years). The combination of the two classification criteria results in five RW classes: VLLW-SL, LLW-SL, ILW-SL, LILW-LL and HLW.

In ***Switzerland***, the RW generated from NPP, medicine, industry, research, RR are classified in three classes: LILW (all waste except alpha-emitting waste and HLW), alpha-emitting waste (ILW-LL with a content of alpha emitters exceeding 20 kBq per gram of conditioned waste) and HLW (SNF and vitrified fission product solutions).

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In the ***United Kingdom***, the RW waste is classified under the following broad categories, according to its heat generating capacity and activity content: HLW (high-level or heat generating waste, in which the temperature may rise significantly, so that this factor has to be taken into account in designing storage or disposal facilities), ILW (with radioactivity levels exceeding the upper boundaries for LLW, but which do not require heating to be taken into account in the design of storage or disposal facilities), LLW (having a radioactive content not exceeding 4 GBq/t of alpha or 12 GBq/t of beta/gamma activity). As a sub-category of LLW, a VLLW class is defined, with two subclasses Low Volume VLLW and High Volume VLLW.

2.2. What is the option for storage / disposal of the radioactive waste and spent nuclear fuel (SNF) in your country?

For this question, EUG members were asked to specify the storage / disposal option for each waste category identified in Q1, but also the existing or planned facilities for radioactive waste storage and disposal in their country.

For VLLW and LLW, final disposal is the option for all the organizations. There are already operational disposal facilities in some European countries. Some of these are near-surface disposal facilities at ground level (i.e. Centre de l'Aube in France, El Cabril LLW and ILW disposal facility in Spain, LLW Repository at Drigg in the UK), others in caverns below ground level (SFR final repository for short-lived radioactive waste at Forsmark, Sweden – at a depth of 50m, underground repository at Olkiluoto for LLW and ILW and at Loviisa, Finland - at depth of 100 metres). In Romania, a disposal facility for institutional RW is in operation in an exhausted uranium mine, at an elevation of 840 m (Baita Bihor National Repository). In other European countries, such disposal facilities is going to be operational within few decades (LILW-SL surface disposal in Belgium, Konrad repository in Germany, LILW-SL near surface repository in Romania) and others, such as Italy, are in the stage of site search.

The disposal routes for ILW are often linked to the content of long-lived radionuclides:

- ILW-SL, that have low content of long-lived radionuclides (usually less than 4000 Bq/g in individual waste packages but less than 400 Bq/g average per waste package) usually leads to final disposal in existing (Finland, France, Spain, Sweden, Poland, Romania, in the latter two countries, only for institutional waste) or future (Belgium, Bulgaria, Germany, Italy, Romania) surface or near-surface disposal facility, or, in case of Sweden underground rock cavities at 50 m below Baltic sea level, and deep geological disposal facility in case of Germany;
- ILW-LL, due to the higher content of long-lived radionuclides, are intended to be geologically disposed of in the future geologic repositories or interim-stored and waiting for a decision.

For HLW and SNF actually interim storage, with different solutions of local or centralized interim storage facilities, is the selected option for all EU countries. Also, all European countries intend to geologically dispose of these waste categories. The geologic disposal programmes are in different stages: in some countries, geologic disposal facilities are foreseen to be operational within some decades (such as Finland, Sweden or France) and other in longer perspective, while in others the final disposal options are still in a project phase or even without final decision from the government.

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In ***Belgium***, the long-term management solution recommended by the Waste Plan concerns the geological disposal in poorly indurated clay (Boom Clay or Ypres Clay) within one underground facility constructed on a single site located on Belgian territory. There is an extensive R&D programme assessing the use of Boom Clay or Ypres Clay as potential host rock, but no decision in principle on geological disposal was taken yet by the government. Whatever the Government's decision based on the Waste Plan may be, the implementation of the chosen technical solution is bound to be a lengthy drawn-out process that will probably take a few decades before the selected solution becomes operational.

In ***France***, CIGEO disposal project is currently studied under ANDRA responsibility. It is foreseen that the geologic disposal facility will be located in Callovo-Oxfordien clay, and the repository commissioning is planned towards 2025.

In ***Italy***, SOGIN Project of a National Repository is intended for accepting VLLW and LILW with low α-content for final disposal. Moreover it shall provide an Interim Storage facility for ILW with high α-content, HLW and SNF until a decision for final disposal of HLW and SNF is taken.

In ***Germany***, the HLW and SNF are stored in nuclear power plant decay pools, dry storage in on-site facilities, and dry cask storage in transport cask storage facilities at Ahaus, Gorleben and Rubenow. Not only for heat generating waste, but also RW with negligible heat-generation that cannot be disposed of in the Konrad facility due to its nuclide inventory or its chemical composition a final disposal solution have to be decided, as well as for RW retrieved from the Asse II mine. According to the Site Selection Act, enforced on 27 July 2013, a site selection procedure has to define the site for a repository, especially for HLW. The selection procedure for the site of this disposal facility is to be concluded by the year 2031. The Federal Government plans to take the repository into operation around the year 2050.

In ***Poland***, the ILW from medical, industrial and scientific application, are disposed of in the National Radioactive Waste Repository in Rozan, while the SNF generated from research reactor Maria is stored in a wet pool storage facility (Otwock-Swierk). No decision was taken yet for the final disposal of the existing or for future SNF generated in a new NPP whenever operational.

In ***Romania***, the National Program foresees that, after an appropriate period of interim storage, the SNF from power reactors will be disposed of in a geological repository, together with other long-lived radioactive waste. The geologic disposal facility should be in operation in 2055 to accommodate the CANDU SNF and LILW-LL waste, but no decision in principle was taken on the disposal concept, neither host rock nor siting strategy (based on desk studies, some suitable rocks were identified and cost estimation was made based on Canadian geologic disposal concept).

In ***Spain***, the ILW and HLW are temporary stored in individual or centralized facilities (CTD). There is a plan for a Centralized Repository for ILW and HLW, in Villar de Cañas (Cuenca), called ATC and some individualized storages in the NPP's sites (called ATI) are building to save fuel pools capabilities. As a final option, geological disposal is considered and an R&D program is ongoing since 1985.

In ***Sweden*** the KBS3 / SFK final repository for SF, located in granite is foreseen to be in operation in 2031.

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In the ***UK***, the HLW and SNF is stored until a geological disposal will be available. The UK Government's framework for long-term managing higher activity radioactive waste (HAW) foresees geological disposal. The Welsh Government has also confirmed a policy for the geological disposal of HAW as part of a joint programme with England and Northern Ireland, while current Scottish Government policy is that of long-term storage as the primary management option. RWM has an ongoing work programme to demonstrate that geological disposal of the UK inventory is feasible in a range of geological settings, support UK management of HAW, and geologic disposal facility siting activities, and develop and maintain RWM's scientific and technological capability.

2.3. What are the waste acceptance criteria (WAC) for the storage / disposal facilities identified above in Q2 (operational and foreseen to be commenced in the future)?

For this question, EUG members were asked to specify, for each storage / disposal option, the main parameters that have to be characterized: radiological parameters, chemical parameters, mechanical parameters, but also any other type of parameters (i.e. homogeneity, types of conditioning matrices, specifications for the container, waste accepted with restriction, forbidden waste, etc.).

Depending on the presence or not of disposal facilities, WAC's variability is of very wide range for the different countries. There are *applied WAC* at the operational disposal facilities (Centre de l'Aube in France, El Cabril in Spain, Drigg LLW Repository in the UK, SFR Forsmark in Sweden, Baita Bihor national repository in Romania, Rozan national radioactive waste repository in Poland) or interim storage facilities (CEDRA facility in France, interim storage facilities in Dessel-Belgium) and *preliminary WAC* for the disposal facilities, both for LILW-SL waste (cAt-project in Belgium, near surface repository project Romania, National Repository project in Italy) and ILW-LL, HLW and SNF (Cigeo project in France, KBS3 project in Sweden), foreseen to be in operation in the future. For the Konrad repository, the WAC are legally fixed and in force. All RW that will not be disposed of in Konrad repository are characterized along predictable properties of safety-relevant parameters for a future geological repository.

For all disposal facilities, the parameters concurring to define WAC are grouped in *radiological*, *chemical*, *mechanical* and *other* parameters.

For the disposal facilities for LILW-SL waste, intended to accommodate mainly RW with short lived radionuclides and limited amount of long-lived radionuclides, *radiological parameters* limited by WAC consist of radionuclide specific activity per package and total activity (for the whole disposal facility or even per disposal zone), dose rate at surface and certain distance (mSv/h), integrated dose, surface contamination (for beta-gamma and alpha emitters). The dose rate and surface contamination are generally limited by the transport norms and for operator safety.

The list of safety relevant radionuclides differs from one facility to another, and generally consists of both short- and long-lived, beta-gamma and alpha/neutron emitting radionuclides. Also, in some cases, limits are imposed for total beta-gamma and alpha emitters. To guarantee under-critical conditions in all scenarios that could occur during the storage facility lifetime, the WAC limit the content of fissile materials both in near surface disposal facilities and in geological ones.

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While for the disposal facilities for LILW-SL the WAC impose specific and total activity limits for safety relevant radionuclides, for the geologic disposal facilities, generally the preliminary WAC do not impose a limit for specific activity nor for total activity. For KBS3 project, the WAC require that the radionuclide content in each canister to be determined and documented, and for the SNF to be in oxide form and have low dissolution rate in the repository. For Cigeo project, in order to manage site source term for radionuclides that are attempted to get out the storage facility with significantly environment & human impact activity, activity of radionuclide by radionuclide has to be known.

Limitations regarding the heating power are imposed for geologic disposal facilities for HLW and SF, in order to preserve geologic barrier functions and risk of fire. Also, for Konrad facility (for disposal of RW with negligible thermal activity), the heat generated by the waste is limited as heat per RW container and as container averaged heat constant.

In ***Belgium***, for the future near surface disposal facility, a set of radionuclides considered critical for long term safety (28 radionuclides, Ra-226, and fissile nuclides Th-232, U-235, Pu-239, Pu-241, limits for activity per waste package, per disposal zone and for the whole disposal facility are defined (for example the limit for fissile material is 15 g on 220 L waste container). In the case of interim storage, radiological parameters are referring mainly to the external dose rate, total beta/gamma activity, total alpha activity and radium content.

In ***France***, for VLLW and LLW disposal the following parameters are considered: mass activity radionuclide by radionuclide (in order to evaluate source term and therefore the radionuclide diffusion/transport within and around the storage facility); activity radionuclide by radionuclide (in order to manage site source term for radionuclides that are attempted to get out the storage facility with significantly environment & human impact); criticality (guarantee under-critical conditions in all scenarios that could occur in the storage facility life). For Cigeo project, preliminary WAC has been produced, some parameter that should be considered in the future repository are: specific heat power (in order to preserve geological barrier function and risk of fire), dose rate, alpha and beta-gamma surface contamination, nuclide vector compatibility.

In ***Germany***, the WAC for Konrad repository foreseen limits on radiation activity (total alpha and total beta activity, activity for a series of radionuclides plus U-total, Pu-total activities). Gamma & neutron dose rate at container surface and 1m distance is also considered in the WAC as well as total heat-power per container and container-averaged heat decay constant.

In ***Italy***, since the site for the National Repository was not selected yet and WAC are site dependent, only an indication on the activity concentration and specific heat power are included in the preliminary WAC.

In ***Poland***, for the Rozan national repository, as radiological parameters in WAC are specified the radionuclide composition and characteristics (very low and low level waste; short and long-lived radionuclides), maximum gamma dose rate on the surface of packaging and at the 1 m distance and unbounded contamination on the packaging surface.

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In ***Romania***, maximum limits are imposed for a list of individual radionuclides (separated in long-lived and short-lived), both for the National Repository Baita Bihor (in operation) and for the near surface disposal facility foreseen to be in operation in 2023 in the proximity of Cernavoda NPP. Radioactive waste containing fissile radionuclides are not accepted for disposal. Contact gamma dose rate and the surface removable contamination of the waste packages are also limited by the transport norms and for operator safety.

In ***Spain***, for near surface disposal facility, specific activities limits per package are imposed for lists of critical radionuclides (defined separated for VLLW disposal facility, Level 1 LLW and Level 2 LLW disposal facility) and , as well as for fissile isotopes (U-235 with enrichment between 1 and 10%, Pu-239 and Pu-241). Gamma dose rate and removable surface contamination (both for β-γ emitters and for α emitters)

In ***Sweden***, the waste acceptance criteria are defined both for SFR facility (in operation) and for future geologic disposal facility. For SFR facility, the radiologic parameters included in WAC consists of the radionuclides content, gamma dose on surface and on certain distance (mSv/h), gamma dose rate on surface and at a certain distance, surface contamination (gamma/beta, alpha), radiation impact in terms of integrated dose on waste. For future KCS 3 geologic disposal facility, the SNF must be in oxide form and have low dissolution rate in the repository, while the waste consisting in structural parts, including radioactive activation products, shall have a certain corrosion resistance. The content of radionuclides in the canister shall be determined and documented. The total decay power of the encapsulated SNF in each canister must not exceed a specified limit. Regarding the nuclear criticality, keff shall be less than 0.95 for the possibly occurring most reactive cases identified. For cases that can be deemed to be unrealistic but that would act to increase the reactivity, the criterion keff < 0.98 is applied. Also, a maximum allowed dose rate at the canister surface is set with respect to the possible impact of radiation on the functions of the engineered barriers of the final repository.

In ***UK***, there are not defined WAC for future disposal facility for. The radioactive waste management organization (RWM) produces packaging specifications as a means of providing a baseline against which the suitability of plans to package HAW for geological disposal can be assessed. In this way, RWM assists the holders of radioactive waste in the development and implementation of such plans, by defining the requirements for waste packages which would be compatible with the anticipated needs for transport to and disposal in a geological disposal facility (GDF). The packaging specifications form a hierarchy which comprises three levels:

- the Generic Waste Package Specification (GWPS) - defines the requirements for all waste packages which are destined for geological disposal;
- Generic Specifications - apply the high-level packaging requirements defined by the GWPS to waste packages containing a specific type of waste;
- Waste Package Specifications (WPS) - apply the general requirements defined by a Generic Specification to waste packages manufactured using standardised designs of waste container.

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Chemical Parameters

Regarding the *chemical parameters* considered in WAC, in all situations particular attention is paid to the chemical reactivity to ensure the stability of the waste matrices and of the engineered barriers: there are limitations on the complexing and chelating agents, accelerators of leaching processes (chlorides, fluorides, nitrates, sulphates, carbonates), organic substances, pyrophoric, flammable, explosive, corrosive or oxidizing materials. Biologically active waste (infectious or putrescible) are generally forbidden.

In ***Belgium***, the chemical parameters specified in the WAC depends on the destination option, as follows:

- for interim storage - mainly stability of the wasteform is specified;
- for the future near surface disposal facility - stability of the wasteform (no perturbations of the EBS), limited complexing agents and cement fraction are imposed;
- for geological disposal facility - no/little perturbations of the disposal system: gas production (mainly by metal corrosion), heat production, and swelling of the waste, organics (formation of complexing agents).

In ***France***, for VLLW and LLW disposal facility the following parameters are considered:

- inventory of toxic species (exhaustive list given by Andra) in order to determine the amount of such species in the repository: verification with impact studies for human activity & environment / regulatory obligation;
- chelating species: modification of radionuclides transfer behaviour - verification of impact with repository design;
- harmlessness between waste and package components on a defined time period - to assure specific function of waste package if needed (not concerned for Cires and VLLW waste storage project);
- stable waste - reactive or low kinetic reactive waste that doesn't lead to break the waste package (if package support function) or harm the facility or workers (explosion, gas generation leading to high pressure, fire, overheating…);
- list of forbidden waste and of limited waste are available;
- conditioning matrices - if needed by facility design (not the case for Cires or VLLW waste), conditioning matrices may have to contain in a significant way radionuclides diffusion or leakage.

In ***Germany***, for KONRAD disposal facility the chemical parameters considered in WAC impose that the nuclide vector and material vector (metallic, ceramics, rubble) to be declared. All RW have to be solidified: no gases, no liquids, no flammables, no swelling are admitted, corrosion has to be controlled (humidity), no chemical kinetics are acceptable. The grouting material is concrete and chemical toxicity & hazards have to be considered. No leaking and no leaching are acceptable.

In ***Italy***, the inventory of toxic species, resistance to leaching, and off-gas production are limited.

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In ***Poland***, prior to disposal the radioactive waste shall be converted to a solid form with the content of unbounded water below 1% mass, and the rate of leaching with distilled water of solidified radioactive waste, after 28 days of leaching in static conditions, cannot exceed the prescribed levels depending on waste classification. The RW have to be segregated according to its category and sub-category, and placed in a closed packaging designed for disposal, in the way that prevents the radioactive waste escape. Moreover, waste containing following substances shall be stored separately from each other and from the waste:

- organic solvents, extractants and oils
- detergents in concentration exceeding 10 mg/dm^3
- complexing agents in concentration exceeding 10 mg/dm^3
- dissolved substances and deposits with content of dry residue exceeding 10 g/dm^3

Corrosive properties of waste also have to be specified.

In ***Romania***, for **National Repository Baita Bihor,** under normal temperature and pressure condition, the radioactive waste must not react with each other or with the conditioning matrix, decompose by explosion, detonate or react explosively with water. Radioactive waste that are considered hazardous for disposal according to the in force regulations are not accepted. Pyrophoric radioactive waste is forbidden to be disposed of. Radioactive waste containing hazardous, biological (organic), pathogenic or infectious agents must be treated, conditioned and packed in such a way as to minimize the danger that would be caused by the disposal of these types of waste. The gaseous waste that contain or are capable of generating toxic gases, vapours or smoke during transport, handling or storage are not admitted for disposal. The liquid radioactive waste has to be solidified or packaged in sufficient absorbent material that is capable of absorbing a volume twice as large as the liquid volume. The solid or solidified radioactive waste has to contain as less as possible free liquid (max 1% from their volume) and this free liquid has to be noncorrosive.

In the **future near surface disposal facility,** no raw waste is accepted for disposal: homogeneous waste is to be solidified; heterogeneous waste is to be immobilised. A cement matrix is convenient for both case, other matrixes are also possible. All "substances of very high concern" according to European regulation REACH are considered as toxic chemicals and are accepted with restriction, based on a case by case decision. Other waste accepted with restriction, based on a case by case decision are: chelatants, waste which react with the matrix, waste which react with water (aluminium, zinc, uranium, magnesium), wood, absorbers, liquids into absorbers. Compounds which present – alone, or in contact with air or water - risks of explosion or ignition, such as alkali metals, metal fines, powdery magnesium, uranium, aluminium, explosives, unstable chemicals, strong reducers (hydrazine, borohydride, fuel, alcohol, phosphorus...) are forbidden to be disposed. Radioactive sources, infectious or putrescible waste, non-solidified liquids, non-solidified homogeneous waste, non-immobilized heterogeneous waste, waste containing gas are also forbidden to be disposed of. Inert homogeneous or heterogeneous waste is accepted: metallic waste that are not react with water, concrete rubble, pellets of compacted plastic and tissues (as secondary waste from operation in nuclear zone), sludge, aqueous solutions, filters, ion exchange resins. The auto-ignition temperature shall not be less than 300°C.

In ***Spain***, the main limitations and/or forbidden chemical elements for VLLW and LLW, in storage conditions, are:

- explosive, corrosive, oxidizing, easily flammable or flammable;

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- waste with animals have to be treated to prevent the generation of gas;
- pyrophoric or strongly reactive metallic materials (magnesium powder, sodium or sodium alloys);
- waste with gas (neon tubes, non-empty aerosols, etc.);
- infectious products;
- dusty materials that are not conditioned in suitable packaging to prevent their dispersion under the handling and storage conditions;
- waste whose temperature is higher than 60˚ C;
- fermentable materials are limited above 3% by total mass of the batch of packages (WAC specific for VLLW), or 10% by individual package (both types);
- an inventory of the toxic substances present in the storage is needed, and the producers must submit to ENRESA annually, the contents of the following elements: lead, copper, aluminium and asbestos.

RW producers sent yearly information on chemical analyses of the main components of the LLW homogeneous wet waste (evaporator bottoms and sludge). This information will assure the waste meets following limitations or must notification:

- accelerators of leaching processes: the producer will identify, before packaging, waste containing complexing agents above 8%, such as chlorides, fluorides, nitrates, sulphates, carbonates;
- organic substances: EDTA, NTA, DTPA, TTHA, oxalates, citrates, acetates, TBP, ethylene diamine…etc., others like sulfonates;
- organic liquids conditioned by incorporation into a solid matrix of hydraulic binder, should not contain organic liquids above 3% by volume of matrix.

In ***Sweden***, detailed requirements are expressed for each disposal alternative and even for different disposal vaults in a specific repository. For SFR/SFL facilities, description of material and typical content per waste stream has to be done. Free or contained liquid are not allowed. The fire resistance, chemical reactivity, complexing agents, leaching and content of chemical toxic material are also parameters considered in WAC.

For the future KBS3 geologic disposal facility, the WAC contain limitation regarding:

- free liquids / water content - the encapsulated water and air content shall be as low as possible and must not exceed a limit set with respect to acceptable internal corrosion and pressures in the canister and the atmosphere inside the canister shall be replaced by inert gas prior to sealing);
- toxic and corrosive materials - acceptable limits for corrosive species as well as for species that may impair the safety functions of the buffer and backfill are set for the final repository.

Mechanical parameters

Usually, through the *mechanical parameters* included in the WAC it is intending to ensure that there will not be a collapse leading to the break of a safety barrier. The mechanic parameters addressed by WAC include compression resistance, void limitation, drop resistance (also required by transport regulations), swelling, but also handling/stacking parameters, weight/package size and centre of mass as well as proper ID labelling etc. The RW have to be solidified (no free liquids are allowed) and the waste matrices have to confine in a significant way the radionuclides (limits on diffusivity and leachability).

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In ***Belgium***, for interim storage, the mainly mechanical strength of the drum (also after 75y of storage) has to be above a prescribed value. For the future surface disposal and geologic disposal facilities, the mechanical strength provided by disposal package has a minimum value.

In ***France***, regarding the mechanical parameters, the WAC include: compression resistance and void limitation (to assure that there will not be collapse leading to break a safety barrier), handling parameters (compatibility with facility equipment and time duration), drop resistance (to assure the non-dissemination of radioactivity in case of package dropping – height is dependent on facility design), integrity of the container and integrity of the matrix.

In ***Germany***, for Konrad repository, WAC require the mass, size, centre of mass, container type/category, safety class category (that includes drop-, fire-, pressure-, explosion-, collision-, resilience), handling standards (craning, stacking…).

In ***Italy***, there are limits for compression resistance, drop resistance, fire resistance, thermic cycles resistance.

In ***Poland***, design and structure of the packaging for radioactive waste disposal shall take into account physicochemical properties and categories of waste placed in this packaging, and also the repository site conditions and its structure. Packaging dimensions shall be adapted to the dimensions of the repository facilities and to the number of layers, in which the packaging will be arranged. Concrete or steel containers protected against corrosion shall be used as the packaging for radioactive waste disposal. Radioactive waste, which dimensions or shape preclude its placement in the packaging, and which should not be broken down because of radiological protection, after securing against the dispersion of radioactive contamination, can be placed in the repository without packaging. Type of canisters is relevant to the activity of the RW according to the national rules: canisters made of plastics or steel (in case of sealed sources) are kept in the lead shielding, the volume of the drums is 50 or 200L. Neither glassy nor plastics materials, which may be broken, are not qualified as material for canisters.

In ***Romania***, both for Baita Bihore repository and for future near surface disposal facility, the radioactive waste has to be treated and conditioned in stable matrices and packaged to ensure the safety during transport and handling, and the stability during disposal period, with a low leaching rate for the radionuclides contained. Free spaces inside the radioactive waste package should be reduced as much as possible. For the future near surface repository, the disposal container shall isolate the solidified waste during 300 years. This means that after 300 years underwater immersion, the container is still hermetic and that the leaching characteristics are guaranteed. There are limits on the minimum mechanical strength thickness of the disposal container. No residual water shall remain after 24 h curing. A maximum of 10 % (volume) of voids is allowed, provided that the compressive strength criterion is met, on a representative sample, with the same proportion of voids. When super absorbed liquids are solidified, the producer shall demonstrate that no sweating from the solidified block will occur during 300 years.

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In ***Spain***, the mechanical parameters depend on the disposal facility:

- for VLLW - no specific mechanical resistance values needed, but must assure the stability in disposal conditions;
- for LLW - *Level 1 RW* mechanical strength is limited:
  - compactable waste - packages are super compacted (1,200Tn pressure) and immobilized (with hydraulic binder of 30 kg/cm2 compression resistance), in concrete container called CE-2a (7 cubic meter infernal capacity);
  - non-compactable waste – the gaps in packages are filling with hydraulic binder of 30 kg/cm2 compression resistance, before to be conditioned in CE-2a concrete container;
  - cartridge filters and dispersible waste are immobilized in metallic containers (packages), with a shield above 5 cm thickness of hydraulic binder (75 kg/cm2 compression resistance), before to be conditioned in CE-2a concrete container;
  - wet homogenous waste (spent resins, evaporator bottoms, sludge) - are conditioned by incorporation into hydraulic binder matrix, which need to meet specific criteria for compression resistance and compression after 7 days water immersion.
- for *Level 2 RW*, the WAC require the assurance for activity confinement:
  - for compactable waste - packages are super compacted and immobilized CE-2a container with a shield higher than 5 cm thickness of hydraulic binder (the same shield thickness is required for cartridge filters and dispersible waste;
  - for non-compactible waste – the gaps in packages are filled with hydraulic binder of 30 kg/cm2 compression resistance, before to be conditioned in CE-2a concrete container with a shield above 5 cm thickness of hydraulic binder;
  - cartridge filters and dispersible waste are immobilized in metallic containers (packages), with a shield thicker than 5 cm of hydraulic binder (250 kg/cm2 compression resistance; flex traction or indirect traction $\geq$ 10 kg/cm2; resistance to nuclides diffusion, values are transformed at leaching rate in aqueous medium;
  - wet homogenous waste (spent resins, evaporator bottoms, sludge) - are conditioned by incorporation into matrix of hydraulic binder, which need to meet specific criteria regarding the compression resistance, compression after 7 days of water immersion and, flex traction (indirect traction).

In ***Sweden***, for SFR/SFL disposal facilities, the WAC include requirements related to the strength, inner mechanical stability and corrosion resistance. Detailed requirements are expressed for each disposal alternative and even for different disposal vaults in a specific repository.

For the future geologic disposal facility (SKB 3), the WAC include requirements regarding mechanical stability, strength and stress resistance. The canister material properties, dimensions and acceptable defects are determined so that the canister will resist (remain tight) the largest expected load occurring in the repository with a safety margin.

Other Parameters

Generally, as *other parameters* there are indicated the homogeneity of the waste, types of conditioning matrices, specification for the disposal container.

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In ***Belgium***, the non-conditioned waste must comply with the acceptance criteria of the specific waste class (combustible, compressible, non-compressible and liquid waste) as defined by NIRAS.

In ***France***, the other parameters highlighted are:

- hydrogen production by radiolysis;
- criticality: the total amount of fissile materials per WP is limited, uncertainties in fissile mass content has to be taken into account;
- size: parameter needed for operational maintenance activities.

In ***Italy***, as other parameter is indicated the conditioning requirement: cemented matrixes are foreseen for LLW and ILW level waste conditioning, while the VLLW do not need conditioning.

In ***Sweden***, for future geologic repository, the WAC include requirements related to the physical *dimensions* (dimensions of the fuel channels in the canister insert, dimensions of the insert and copper shell) and *weight* (a maximum weight is assigned to the sealed canisters based on the maximum weights of the copper shell, insert and the SNF and other components to be encapsulated, including a margin for possibly occurring weight increases). The copper shell shall be designed so it is possible to lift the canister by the lid. Transport, handling and emplacement equipments are designed with respect to the weight and dimensions of the canister. Also, a requirement regarding the *identification* is included in the WAC. The copper shell shall be provided with an id-code and a marking that shall be unique and readable after sealing, machining and deposition of the canister and the id-numbers of the SNF assemblies shall be related to the id of the canister they are encapsulated in.

2.4. Should WAC be harmonized across Europe? If so, how?

For this question, EUG members were asked to express their personal point of view regarding the opportunity for WAC harmonisation and if they consider this harmonization as opportune, to express their personal idea on how this can be done.

The WAC can be grouped in site specific criteria and general criteria. The large majority of respondents (10 out of 12) were of the opinion that harmonisation is neither possible, nor desirable in the case of WAC for safety relevant parameters. The arguments supporting the majority opinion that WAC cannot be harmonised across sites and countries were that such criteria reflect "*strategic choices*" and are "*derived from the safety assessment of a particular disposal facility"*, and therefore "*linked to the design of the facility, local constraints and also on the type of waste to be disposed*", as well as the "*design and advancement of the long-term management programme*". One participant expressed concern that harmonization could lead to "*unjustified accumulation of constraints in several cases*".

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At the same time, several respondents (10 out of 12) expressed the view that harmonization (e.g. through international guidelines from OECD-NEA, IAEA and/or technical platforms) may be appropriate and achievable in terms of the **underlying rationale** for WAC, the **basic assumptions for safety studies**, or the **potential topics** to be evaluated through WAC[1]. It was argued for instance that harmonization of general rules and guidelines would enable cooperation between countries "*having the same or similar inventory of radioactive waste*", and that harmonized HLW WAC are a pre-requisite to Europe's promoting multi-national HLW repositories.

One institution set forth the idea of a platform for information exchange between waste management organisations; this platform should focus on "*how to attain certain requirements (i.e., on how to translate requirements into practical criteria, tests to be performed by the producers,…)*".

One participant refrained from providing comments to this question.

2.5. How do you deal with the fact that the WAC as well as the final disposal concepts are in constant evolution (techno-scientific progress, experiences, stricter attitudes …)?

For this question, EUG members were asked to provide detail on how they are dealing with historical waste and if for this waste they have to characterize more (as required by the current WAC).

A number of respondents (7 out of 12) provided various arguments supporting the evolution of WAC, in order to account for new knowledge and techno-scientific progress in "*proving the safety of disposal facility and in optimizing the disposal concept*". This may include "*information accumulating during site characterisation, monitoring, or as a more adequate understanding of the important processes with impact on the disposal facility safety*", or "*new evidence demonstrating that actual WAC are not strong enough to assure safety and/or security*". Moreover, taking into consideration this "*techno-scientific progress and experience accumulated in countries with more advanced disposal programmes*", is seen by some respondents as potentially "*beneficial both as technical solution and in terms of time needed to implement the disposal programme*" for countries that are now in the planning phase.

Opposite to this, some respondents expressed concerns that this constant evolution can lead to strengthen the WAC or new criteria to fulfil, which in turn may have "*major impacts on waste acceptance"*. It was for instance argued that these changes may lead to increasing costs for the whole supply chain of waste management, due to ensuing needs to improve the characterization technics or the reconditioning of existing waste packages, "*without any demonstrated improvement of the long term safety*".

[1] For instance General Conditions of Acceptance, Physical and Chemical Properties, Waste Volume, Acceptable Waste, Reactive Metals and Materials, Radiological Properties, Packaging and Transport Requirements, Approved Disposal Containers, Non-containerized Waste, Packing, Container Weights, Disposal Container Labeling, Records, Transport Regulations , Transport of Fissile Radionuclides.

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Connected to this, another potential consequence mentioned in relation to evolving WAC or disposal concepts was linked to the technical difficulties in both achieving the more restrictive WAC or demonstrating compliance with these criteria. The example of cellulose content reduction in non-combustible waste was given. A reduction of the "*actual limit of 2 kg to nearly 0 kg in the future*", would bring about several technical challenges that may render this criterion unfeasible; such challenges are, for instance, "*the interception of such small fractions before the waste is placed in a waste drum*", or the "*verification that this low limit is respected*". Moreover, the respondent referred to the operational difficulties to adapt to these changes: "*when new rules must be implemented on the work floor, this can be a slow process even if there is a transition period*".

Currently, countries have different strategies to deal with historical waste, for instance:

- the same criteria are applied for historical waste as for well characterised "new" waste (with potential need for new characterisation of poorly documented waste, or even reconditioning or immobilization in greater packages or extra barriers, if they do not conform with disposal WAC);
- the acceptance of historical waste that has been conditioned before the establishment of WACs is done by "Historical WACs" "*that take into account the timeframe in which they were produced and the according quality, characterisation and documentation measures"*;
- no revisions of the criteria for waste already accepted for interim storage, as long as it stays in interim storage.

One participant suggested that since WAC are meant to guarantee the compliance with facility constraints for general cases, "*derogation acceptation criteria should be involved if possible*", in order to "*limit the impact of new WAC for a transition phase"*, and "*be able to accept most of already or nearly produced waste package*". It was also suggested that more flexible WAC could be allowed for specific cases, "*if there is knowledge of accurate data on specific wastes"*, especially if this is for a restricted period.

In some cases, information on historical waste is gathered through document research and interviews of senior personal and retired people.

In some countries, the WAC for the final disposal are fixed, at least for the existing final disposal options (for example in case of Konrad repository in Germany). One respondent mentioned that interim storages (e.g. at nuclear power plants or other sites, or federal collection sites) may have variable WAC; however, in the latter case, "all incoming waste must qualify for outgoing WAC, which must go into final repository".

In other countries, there are no WAC available for the final disposal, as they are directly linked to the identification of the repository site. It is however assumed that in-depth characterization will be carried out before the transfer of historical RW to the future national repository. One respondent mentioned that there exist no formal WAC for future disposal options until the Safety Analysis Report is complete according to the start of regular operation. This is posing problems for the decommissioning of NPP's, which is depending on "*preliminary WAC for disposal options that are not yet in operation*". For this situation, the respondent proposed as strategy for dealing with the absence of WAC for disposals that are not yet under planning and constructions to adopt interim WAC and also requirements that the waste should be possible to be reconditioned.

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2.6. What methods are you applying in characterisation of conditioned radioactive waste?

For this question, EUG members were asked to:

(a) specify and give details on the methods applied for radioactive waste characterisation (including characterisation of waste before its conditioning);

(b) how the difficult-to-measure radionuclides are correlated with easy-to-measure isotopes;

(c) if the chemo-toxicity is correlated with radio-toxicity of the waste;

(d) if the measured data are complemented by modeling/calculations and if yes, to specify and describe the models used and if there are any potential needs of codes' validation by National Control Authority.

a) Specify and give details on the methods applied for radioactive waste characterisation

The methods used for radioactive waste characterisation depend on the country approach, the majority apply however spectroscopic techniques (alpha, beta and gamma) as well as neutron measurements and dose rate calculation. Few institutions use more "exotic" techniques, like muon tomography and calorimetry in the process of radioactive waste characterisation.

Nuclide vector (NV) and scaling factor (SF) methods are used by a number of European countries (e.g. Belgium, France, Germany, Spain, Sweden, the UK) in the radioactive waste characterisation. These are based on acquisition of the spectrum, i.e. all the radionuclides of interest and the repartition between them if the source of radioactivity is stable in a specific field of application (part of a process, one or multiple locals that have seen the same source of contamination / activation…). For this phase, multiple measurement techniques could be used depending on the radionuclides that have or could have to be measured. Mostly it's a combination of different techniques that are used together (e.g.: alpha spectrometry, mass spectrometry and gamma spectrometry). The sample and location for sampling are very important and have to be representative for the waste stream considered.

In ***Belgium***, the characterisation process is based on the following techniques:

- gamma-spectrometry (for key nuclides determination) combined with an approved NV (base on models, calculations, and sometime on destructive measurements);
- dose rate (measured on non-conditioned waste packages) conversion with approved NVs (calculation of NV through specific conversion factors - mSv/h to MBq - determined using software such as MicroShield);
- destructive analysis on samples and approved nuclide vector;
- neutron measurements on conditioned waste;
- mass spectrometry, gamma spectrometry, alpha spectrometry,…on samples of non-conditioned waste.

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According to the WMO, the radioactive waste characterisation process, in ***France***, is split into 2 phases:

- Determination of NVs and SFs for waste streams;
- Measure the waste itself: when a practical spectrum is known, it is needed to gain access to the waste activity. In this case, the aim is to measure one or more tracer(s) that can lead – with the knowledge of spectra – to the knowledge of every significant radionuclide by means of scaling and using RN-vector concept. The measurement of this / those tracers is aimed to be easy and not hard-to-analyse.

Practically, some basic measurements (dose rate) is done, gamma spectrometry or more specific measurement (neutron fission measurement for example) but with already specified parameters to speed-up the measurement (minimum and maximum time of measurements / order of measurement) with the aim to minimize the time and cost-consuming specific analyses if not needed. These methods focus its efforts on the precise determination of RN-vectors that are representative for the particular waste stream and then use easy-to-perform mass volume and routine measurements and scaling/correlation technique to the not-measured declarable radionuclides.

As entirely part of this design, the nature of what is measured is important. If a final package has to be measured: minimize number of objects to measure but augment the difficulties / uncertainties because of shielding / mix of different composition of the waste. If a primary package has to be measured, more numerous are required but with lesser shielding and lesser complexity (especially if sorted by the nature of the waste). The specific measurement equipment may need individual adaption to the composition or specific nature of the waste and contamination. The simplest cases would need only the realization of an installation spectrum and waste cartography and finally a measurement of the package's dose rate.

If calculations are incurred, the verification of predicted results is made on both ETM radionuclides (that are good subjects to be used as tracers) and DTM radionuclides in order to be sure that there is no bias involved.

For inspection purpose, French WMO is using multiple measurement technics in order to assess radiological activity, toxic concentration and mechanical and physical parameters of the waste package.

The difficulty is that measurement are basically made on final waste package – with all biological protection (if needed) and filling materials – but since only little number of package are controlled (# 200/y) long time measuring may be required.

Depending on the waste type, different methods are used in ***France***:

- gamma spectrometry on waste package;
- X-ray inspection;

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- active neutron measurements for prompt or delayed neutrons and passive neutron measurements;
- samples of waste package may also be collected (with destruction of the waste package), dissolved, then gamma spectrometry or radiochemistry are made;
- measurement of tritium degassing with liquid scintillation;
- calorimetry.

In ***Germany***, all characterization & conditioning processes that qualify for final disposal must be regulator approved and certified by government inspectors. Either process approval or campaign approval prior to major production steps, all quality controlled and quality verified by independent inspectors, are mandatory.

Methods, whatever appropriate are:

For operational waste:

- pre-characterization consists of source/origin, radiological pre-sorting and material pre-sorting (mainly material stream separation and gamma activity / dose rate survey);
- minimize transport: on-site and public transport – pre-condition for interim transport, if necessary;
- conditioning input characterization: sampling & radio-chemical analysis (RCA), QA docs and ID/origin checks, temperature-range stability, outgassing control, temperature survey, dose rate at surface and 1m distance;
- measurement portfolio: dose rate/gamma spectroscopy, calorimetry;
- determination of heat, density, volume, mass, homogeneity, centre-of-mass, transport/handling assurance;
- swabbing and contamination tests for non-sticking volatile contamination: alpha-beta-gamma-spectroscopy;
- sampling-statistics, random sampling for RCA: applied methods: AAS, ICP-MS, OCP-MS, gamma, neutron spectroscopy, QA-process control parameters, uncertainty assessment, pH, pE / anti-corrosion/leaking / leaching assurance, LS/CRDS.

For decommissioning-waste:

- pre-characterization of radioactive wastes consists of source / origin, radiological pre-sorting and material pre-sorting (mainly material stream separation and gamma activity / dose rate survey);
- minimize transport: on-site and public transport – pre-condition for interim transport, if necessary.

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In ***Italy***, an integrated approach combining different methods (such as physical scanning, gamma spectrometry and neutron counting measurements) is used for RW characterisation. For the physical scanning, an X-ray tube is used in order to analyse the matrix and its density distribution. The gamma spectrometry is performed by using high purity germanium detectors and keeping the item (usually a drum with variable volume) in rotation to reduce the radial matrix effect. The item is investigated using several techniques: besides analysing it in open-geometry (one measurement per item), it can be "cut" in segments (reconstructing the vertical distribution of both matrix and contamination) or by means of a tomographic approach (thus with a 3-dimensional reconstruction of matrix and contamination distributions).

For the neutron counting measurements, a dedicated equipment able to detect neutrons coming from both spontaneous fissions (passive mode) and induced fissions (active mode), able to detect both fertile and fissile materials, is used.

In ***Poland***, the RW are different in terms of their isotopic composition, which generates difficulties in their characterisation. However, in order to characterise the waste intended for storage, radiation activity measurements and eventually radiation spectroscopic measurements are made. Usually, radioisotope composition of waste is determined on the basis of the end users' declaration. Characterisation of the radioactive waste is simple, limited to the gamma dose rate in a single container sent to ZUOP for the treatment and disposal. Moreover, each type of the waste is generally described: the volume, form of the waste, radionuclides present (according to the knowledge of the user) and time of creation.

Radioactive wastes are temporary stored before their transport to ZUOP for treatment and disposal in the national repository, or to the producer, in case of sealed radioactive sources.

In ***Romania***, the radiometric measurements carried out for the institutional waste packages (standard 220 l or 400 l carbon steel drums) sent for disposal in National Repository Baita Bihor consist of gamma dose rate measurement and gamma spectrometry. In some cases, the activity of alpha and beta emitters is based on the measurements carried out on row waste before conditioning or on the generators declaration. Each drums prepared for disposal in National Repository Baita Bihor (DNDR) has a compliance sheet containing the contact and 1 m gamma dose rate, activity of beta-gamma and alpha emitters, as well as its unique identification no., mass and the no. of conformity report.

In Cernavoda NPP, no conditioned radioactive waste packages were produced up to date. The content of C-14, tritium and gamma emitters is measured on raw waste that are regularly sent for incineration at a European operator (mainly solid and organic liquids absorbed on polymeric absorbent). The resulted ashes are returned to Cernavoda for storage until the conditioning method will be decided. The spent ion exchange resins (SIERs) as well as other LILW-LL categories are stored on site and no characterization campaign was carried out up to now. For the LL-SL waste, the process for developing scaling factors was initiated few years ago. The waste streams were identified and gamma measurements for ETD radionuclides were carried out. Also, the DTM radionuclides are measured on the identified waste streams by using adequate radiochemistry methods for their separation and purification and LSC and alpha spectrometry for quantification.

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In ***Spain***, SFs are used to evaluate the DTM radionuclides, while the ETM radionuclides are determined by:

- gamma dose rate measurement for waste with known gamma isotopic composition;
- segmented radiography for heterogeneous waste or for better improvement;
- samples taken before the conditioning process - for homogeneous waste, and further extrapolation as a function of the mass introduced, a final dose rate factor correction is applied if necessary;
- gamma spectrometry to the whole package/item;
- segmented /collimated gamma spectrometry for heterogeneous waste or for better improvement.

In ***Sweden***, the radiometric measurements are split in two categories: *nuclide specific measurements* of waste packages and *non-package-specific data*. The waste suppliers are requested to perform gamma spectrometry for each waste package disposed of in SFR (the final disposal for LILW-SL). The values for gamma emitters are registered in Triumf NG for each package, including information about manufacturing date and measurement date. The activation product Co-60 and the fission product Cs-137 are considered “key nuclides” and are used in most of the calculation methods to determine the activity or distribute other difficult-to-measure nuclides in the different waste vaults in SFR. The activity of interest when using the key nuclides is the original activity at the waste origin. The date of waste origin can, however, be difficult to clearly determine for a package, which is why the activity at the package’s manufacturing date is set as the original activity. The measured activity is re-calculated, with the aid of half-lives, to the activity the package had at the manufacturing date. The activity of the key nuclides at the manufacturing date is then used in most of the calculation methods.

Nuclides that cannot be detected by gamma spectrometry are generally determined for the package by correlation. For activity determination of most transuranics, Sr-90, Ni-59, Ni-63, C-14, Cl-36, Mo-93, Tc-99, I-129 and Cs-135, however, specific measurement and calculation methods have been designed. Common to these methods is that measurements and calculations of the amount of radioactivity are made on active systems in the different facilities instead of on individual packages. These data are therefore called “non-package-specific data”.

- Measurement of transuranics, strontium and nickel.
    - Every year, SKB receives information from the NPPs and Clab (interim wet storage for SNF) on the amount of radioactivity generated during the previous year regarding the alpha-emitting transuranics (Pu-238, Pu-239, Pu-240, Am-241, Am-243, Cm-242, Cm-243, Cm-244) and the beta-emitting nuclides (Sr-90, Ni-59 and Ni-63) in waste from the waste classes “Ion exchange resins” and “Trash and scrap metal”. In the different clean-up systems, water samples are taken to quantify the absorbed amount of activity in ion exchange resins in the current year.

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- For nickel nuclides, measurements are made on Ni-63, while the difficult-to-measure Ni-59 is calculated, by the waste supplier, with specific correlation factors based on the results of measurements on Ni-63. To determine the activity of "Trash and scrap metal", the waste suppliers use specific correlation factors with Co-60. Pu-239 and Pu-240, as well as Pu-238 and Am-241, have congruent alpha energies, which render them difficult to separate from each other. The activity amount of the nuclides is therefore usually reported as one.

- C-14 and Cl-36 - are mainly found in clean-up resins and therefore are determined particularly for the waste class "Ion exchange resins". The calculation methods for activity determination of these nuclides assume a production of C-14 and Cl-36 proportional to the thermal energy production in a reactor. Additional variable parameters for Cl-36 are chloride concentration in the reactor water and moisture content in the formed vapour.
- The activity of Mo-93, Tc-99, I-129 and Cs-135in the waste disposed of in SFR is determined on the basis of calculations.

The nuclides that cannot be measured or calculated with the aforementioned methods are determined by correlation factors where nuclides are correlated based on the activity of the key nuclides Co-60, Cs-137 or the sum of Pu-239 and Pu-240.

In the ***UK,*** most characterisation activities are undertaken by the waste owners, on raw waste, prior to any conditioning process. The level of characterisation is often a case by case challenge depending on the situation (e.g. wastes that are difficult to access) or the timescales under which the characterisation is required (e.g. low TRL techniques are not favourable for short timescale challenges). Routine characterisation of row radioactive waste consists of:

- radiological characterisation;
- routine sampling and analysis to confirm satisfactory operational processes;
- effluent characterisation;
- condition, monitoring and inspection (CMI) of existing storage facilities –based on British Standard inspection requirements.

Example of CMI technique is the Laser measurement of waste package geometry to detect waste container expansion, muon tomography to support an understanding of the evolution of specific waste packages, air sampling and discharge monitoring in storage facilities to confirm satisfactory operation. Gas sample collection has been undertaken for selected packages.

Dummy (inactive) packages have been produced which could be retrieved to open and see how they have behaved to provide further underpinning for expected waste package evolution.

Conditioned waste is not routinely characterised in the same way, but is part of an overarching CMI strategy on the waste owner sites.

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b) *How the difficult-to-measure radionuclides are correlated with easy-to-measure isotopes.*

Regarding the *correlation of DTM radionuclides with ETM radionuclides*, most of the countries use, as is already mentioned, the concept of nuclide vector (NV) and scaling factor (SF) derived from a combination of experimental measurements, modeling and information about the waste streams.

In ***Belgium***, the qualified vectors are based on the origin of the waste, the installation, etc. and the "key nuclides" used to correlate DTM with ETM radionuclides are Co-60 and Cs-137, and sometimes a cascade system is used for the correlation of the DTM with ETM isotopes. For instance: isotope X (or a group of isotopes) is correlated with isotope Y; if isotope Y is absent in the measured spectrum it is correlated in a next step with isotope Z (measured or calculated), and so on. Also, in ***France*** the DTM radionuclides are correlated with ETM ones based on approved NVs and SFs. In the case of calculation are used in developing NVs and SFs, verification of the predicted results are made both on ETM radionuclides and DTM ones in order to be sure that there is no bias involved.

In ***Italy***, the NVs are declared by the waste owner. If the vector is unknown or not reliable, destructive measurements (beta and gamma) are performed to derive experimental correlation factors between DTM isotopes and ^{137}Cs or ^{60}Co.

In ***Romania***, for the radioactive waste generated from TRIGA RR no SFs were developed up to now and for the waste streams generated from Cernavoda NPP operation the development of SFs is in progress.

In ***Spain***, NV and SF concepts are used to correlate the DTM with ETM radionuclides. A sample collection plan is required for the waste streams, producers, etc. to get representative SFs. For example, for operational waste (resins, sludge's…) a collection of samples taken at each campaign is a good approach to achieve in time a representative SF for that stream and producer. In dismantling projects a previous radiological campaign for whole site/facility is desirable in order to have in advance the isotopic vectors (gamma emitters plus SFs) for the different involved origins.

In ***Sweden*** the DTM radionuclides characterisation is based on direct measurements, correlation and calculations are used for different radionuclides. Those nuclides that give highest dose in the scenarios after closure and for disposal options are treated more carefully. Those nuclides that cannot be measured or calculated with the aforementioned methods are determined by correlation factors where nuclides are correlated based on the activity of the key nuclides Co-60, Cs-137 or the sum of Pu-239 and Pu-240.

c) How the chemo-toxicity is correlated with radio-toxicity of the waste.

Concerning the *correlation of the chemo-toxicity with radio-toxicity of the waste*, from the answers received, it seems that in general no correlation is made. Sometimes (Belgium, Spain and France) some chemicals and toxic elements are measured/calculated but without a correlation with radioactivity.

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d) How the measured data are complemented by modelling/calculations.

Regarding the complementation of the measurement data with modelling/calculations, generally speaking, no validation by National Control Authority is formally requested. The principal software are MCNP, Origen, MicroShield, FISPACT, but also in-house codes are used by different waste producers.

In ***Belgium,*** the following codes are used***:***

- MicroShield code - to calculate the conversion factors to convert dose rates into activities. Factors are specifically based on the - waste package, type of waste;
- Monte Carlo Software (MCNP, MCBEND,…) and LLWAA (Low Level Waste Activity Assessment - to calculate the nuclide vectors for each year of exploitation of the NPP's;
- Visiplan dose assessment program developed to assist the ALARA analyst in pre-job studies;
- Origen and FISPACT, FISPIN;
- ALEPH - in-house propriety software programs used to calculate nuclide vectors for simple activation calculations and irradiated fuel vectors, for irradiations in the BR2-reactor.

In ***Italy***, the correlation of DTM radionuclides with ETM ones is made by means of customized software: data coming from the different measurements are collected and combined in order to take into account a wide range of effects (matrix attenuation, self-absorption, inhomogeneous matrix and contamination distributions). During this step, Monte Carlo calculations are performed or reused if suitable.

In ***Spain***, to verify the theoretical models for the ratio of ETM to DTM isotopes for activated items is desirable to take samples if possible. An attempt is being made to 'validate' the theoretical neutron activation results, by performing experiments under controlled conditions, i.e. the irradiation of standard metal items of well-known chemical composition, irradiation history and neutron flux record (using a Cf source of neutron energy and distribution being quite similar to the U fission spectrum). Radiochemical analysis of irradiated items will serve to finally quantify the tolerances/uncertainties/bias of the comprised input parameters and neutron codes. It is not a validation of the codes themselves but a validation of the overall process involved (chemicals, irradiations, neutron flux, codes…). Afterwards in real cases, it is possible to predict in advance the degree of precision with the knowledge gathered from all parameters involved.

In ***Sweden***, SKB codes for fuel matrices (MCNP and Origen) for the activation of materials are used. These results are then used in different modelling codes (fuelAct, IndAct, CoolAct) for the calculation of the release into the coolant and uptake on the system surfaces and ion exchange resins.

The Swedish radiation protection agency, receives an annual report concerning the updated inventory added to the existing inventory of the final disposal facility SFR that SKB will deliver with all waste eventually comprised from the waste producers (mainly the NPPs).

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2.7. What are the uncertainties associated to the methods you are currently using in radioactive waste characterisation?

For this question, EUG members were asked to specify the levels of uncertainties for each method used for radioactive waste characterisation (a), the target level of uncertainties (b), the source of the uncertainty (c) and also the action(s) (if any) to decrease the uncertainty level in characterisation of conditioned radioactive waste (d).

a) Levels of uncertainties for each method used for radioactive waste characterisation

Most of the answers give some general values based on the methods applied. The approach is peculiar for each country but it seems that similar uncertainty levels can be found (about 30-50% on conditioned wastes and lower values for destructive assays depending on the representativeness of the sampling procedures).

In ***Belgium***, currently is undertaking of mapping all the uncertainties in the various stages of the radiological characterization process in order to find methods to decrease the level of uncertainties in the future. Actually, the uncertainty levels are ranked from LOW to HIGH depending on the methods applied:

- determination of key nuclides through gamma spectrometry: LOW;
- destructive Measurements (mass spectrometry, gamma spectrometry, alpha spectrometry,…) on samples of non-conditioned waste: LOW (if sampling is representative);
- neutron measurements on conditioned waste: MEDIUM;
- measurement of dose rate on non-conditioned waste packages and calculation of nuclide inventory through specific conversion factors (mSv/h ->MBq): HIGH.

In ***France***, due to the diversity of the techniques used and of the objects of characterisation the uncertainties are variable. Inspections of waste packages are done in order to select the more precise measurement in accordance with the optimization of measurement time of either a unique or a set of multiple assay techniques all together.

For example, in the case of measuring the Pu content in 100 liters drums, the uncertainty of characterization is about 30-50 % with the combination of gamma spectrometry, neutron measurement and X-ray imaging. This uncertainty can be reduced in case of homogeneous matrix and homogeneous repartitioning of the activity.

In ***Germany***, for technical / methodological uncertainties, Gaussian statistics propagation are applied. To comply with WAC limits, distinguish between “guaranteed parameters” (GP) and “other declarable parameters” (OP) are made:

- for GP: mean-value + 3 sigma < WAC-limit;
- for OP: mean-value + 2sigma < WAC-limit.

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In ***Italy***, the total uncertainty value is used, including all the systematic uncertainties and assuming reasonable values of statistical uncertainty. Due to the multi-techniques approach, the uncertainties arising from neutron counting and gamma spectrometry (ISOCS^{TM} or segmented/tomographic scanning mode) are integrated with destructive methods and sampling uncertainties with a considerable decrease of the overall uncertainty.

In ***Poland***, the activity measurements are complemented with the standard uncertainty of the measuring devices. All devices used for characterisation of waste are calibrated every year.

In ***Romania***, for in situ gamma measurements on conditioned waste packages, uncertainty could be between 30 and 50%. For raw wastes, gamma spectrometry measurements, total alpha and total beta activity measurements are carried out with the aim of reducing the overall uncertainty.

In ***Spain***, the degree of uncertainty is different as a function of the waste classification: if the aim is the release, a thorough uncertainty evaluation is required in order to assure that the released materials fulfil the release criteria with an established confidence level.

Also a different degree of uncertainty is allowed for VLLW and LLW because of different disposal concepts for both types. A good precision or a low degree of uncertainty is generally desirably, in particular in the area where the waste classification changes from one class to another.

In ***Sweden***, the greatest uncertainty for measured radionuclides is the representativeness of the collected samples and backtracking to different operational modes at the NPPs. For radionuclides that are correlated to ETM nuclides, the uncertainties are due to accuracy of the actual correlation factor (based on a few examples from different reactors in the western hemisphere).

b) Target level of uncertainties.

Regarding the *target level of uncertainties*, generally there are no target values imposed by the authorities.

In ***Belgium***, there is no legal limit for uncertainties and each method and its corresponding uncertainty are evaluated case by case.

In ***France***, the target level of uncertainty is either to have a result that is coherent with producer declaration or to comply with the declaration limit of the disposal facility (if possible).

In ***Italy***, even there is no target level of uncertainties required by the authority, some quality criteria are applied during analyses. For example, a measured activity value is acceptable if the statistical uncertainty is below 20%. Then, if applicable, the gamma spectrometry and the neutron counting must return compatible results (after the application of the appropriate correction factors, if needed).

In ***Spain***, in general, the confidence interval of 95% is to be achieved and this value is added to the mean value in order to check some requirement. Then the involved uncertainties have to be reduced below the 95% level unless they are independent of the target to deal with.

In ***Sweden***, the uncertainties have to be as low as reasonable achievable. More efforts are taken for radionuclides that affect most in long-term scenarios for the final repositories and the safety analysis.

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c) Source of the uncertainty.

Regarding the *source of the uncertainty*, the majority of the answers indicated uncertainties related to geometry, sampling procedures and correlation factors used.

In ***Belgium***, the uncertainties on the vectors will mostly contribute to the total uncertainty in terms of representativeness of the vector for the drum, basic assumptions of the calculation, chemical composition of materials, and chemical behaviour of the radionuclides. Other uncertainties are related to the heterogeneity of the waste and heterogeneity of the activity distribution. A factor between 5 and 100, depending on the specific nuclide (measurement/calculation), is currently taken into account as general uncertainty.

In ***Italy***, geometric efficiency (depending on geometry setup and detector characteristics), transmission (depending on the source intensity and shielding effect of the matrix) and counting statistic (depending on the radionuclide activity and the count time) are considered the main source of uncertainties.

In ***Spain***, the main source of uncertainties are considered to be intrinsic radioactivity nature, sampling process, measuring methods, devices used, calibration, laboratories, conservativeness of approaches.

In ***UK***, the heterogeneity of waste is a challenge in terms of how representative sampling is in relation to the entirety of a waste package or waste stream. Uncertainty of sampling is bounding of uncertainty related to the sampling technique. Scaling of characterisation from a small volume environment to a large volume environment introduces specific uncertainties, as well as the specific characterisation technique introduces specific systematic uncertainties.

d) Action(s) (if any) to decrease the uncertainty level in characterisation.

Each country has its own methodologies to decrease the uncertainty based on high quality calibration, statistical methods and sampling programmes, pre-characterization of unconditioned wastes, homogenization of the wastes.

In ***Belgium:*** up-stream characterisation of non-conditioned waste and use of (gamma) spectrometry as much as possible.

In ***Italy***: ISOCS™ or using a combination of certified sources and mock-up drums for calibration campaigns (simulation of a real measurement with a well-known matrix and activity distribution in order to understand and improve the system performance). A correct schematization of the measurement setup by means of a set of physical parameters for the item to be assayed helps to reduce errors and uncertainties (object-detector distance, density and material determination and distribution for ISOCS™, self-absorption contribution using software and simulating codes such as MCNP). The use of an external transmission source helps to schematize and correct the attenuation factor due to the matrix shielding effect (comparison between source net count rate with or without the item to be assayed), while the counting statistic uncertainty can be lowered increasing the measurement time (whenever is possible) and reducing the dead time (shielding, distance).

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In ***Spain***: focus on sampling, use of more appropriate methods, representativeness approaches, homogenization processes, statistical approaches when possible (number of samples, measurements…)

In ***Sweden***: statistical methods and sampling programmes to either directly measure the radionuclide or get a better parameter fit to the applied model.

2.8. Which other uncertainties (e.g. technical, conceptual, social, political, and ethical) do you anticipate with regard to waste characterisation for safe disposal? Please list the three in your opinion most important ones. How do you deal with these uncertainties? Can they be managed?

The uncertainties addressed by this question may relate to changes in the final disposal concept, regulation and/or policy, divisions of responsibilities, limits of knowledge, amounts of waste, societal incentives or pressures, financial constraints, safety/security protocols, among others. For all three of the uncertainties identified, the EUG members were asked to describe:

- *what is the impact of each of these uncertainties on waste characterization (e.g. on the relationship between actors, operational safety, costs, …);*
- *whether and how these uncertainties are dealt with at present (e.g. through various procedures, leaving the option of re-characterization, storing the waste in a certain manner, …);*
- *any suggestions they may have on whether and how these uncertainties could be managed (e.g. new equipment, a political decision, involving other disciplines, reconditioning, flexible tariffs …).*

Different types of uncertainties were mentioned in relation to waste characterisation, as illustrated in Fig. 1:

- *political* (e.g. related to policies or regulations from authorities, such as changes in the WAC or characterization strategies);
- *social* (e.g. public acceptance, social perception, stakeholders' expectations);
- *ethical* (e.g. focus on current vs. future needs?);
- *economic* (e.g. budget available);
- *technical* (e.g. conceptual, e.g. resulting from the need to include results of new R&D, or related to insufficient tests for safety demonstrations);
- *waste inventories*: this particular type of uncertainty, could be categorized as a technical uncertainty, but was mentioned as a distinct category by several respondents. Such uncertainties result for instance from lack or incomplete knowledge about waste volumes, old waste, masses and types of corrodible materials, organic inventories, but also waste generated by new nuclear technologies.

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At the same time (see blue arrows in Fig. 1, linking sources of uncertainty to types of uncertainties), there are overlaps between the different types of uncertainties: those related to the information requirements for waste inventories are of both social (focus on future vs. current need for more accurate information), technical (R&D needed to obtain the desired level of accuracy of information) and economic nature (e.g. allocation of R&D funds). Related to this, a participant to the survey mentioned that "*the amount of information asked could be excessive at a given time*" and that a balance should be kept between a precautionary approach and the "*present absolute necessity*".

Some respondents characterized the social, political and ethical uncertainties as influential, unpredictable and not addressed well (yet), while those technical are controllable. Opposite to this, one respondent argued that social, political and ethical uncertainties do not influence the uncertainty of waste characterization.

Concerning the impact of uncertainties, these were deemed critical with respect to costs, effort necessary for waste characterisation, and the time schedule for the planned disposal to become operational. As illustrated also in Fig. 1 (arrows in brown colour, hashed lines), the uncertainties or their impacts may in turn be sources of (new) uncertainties.

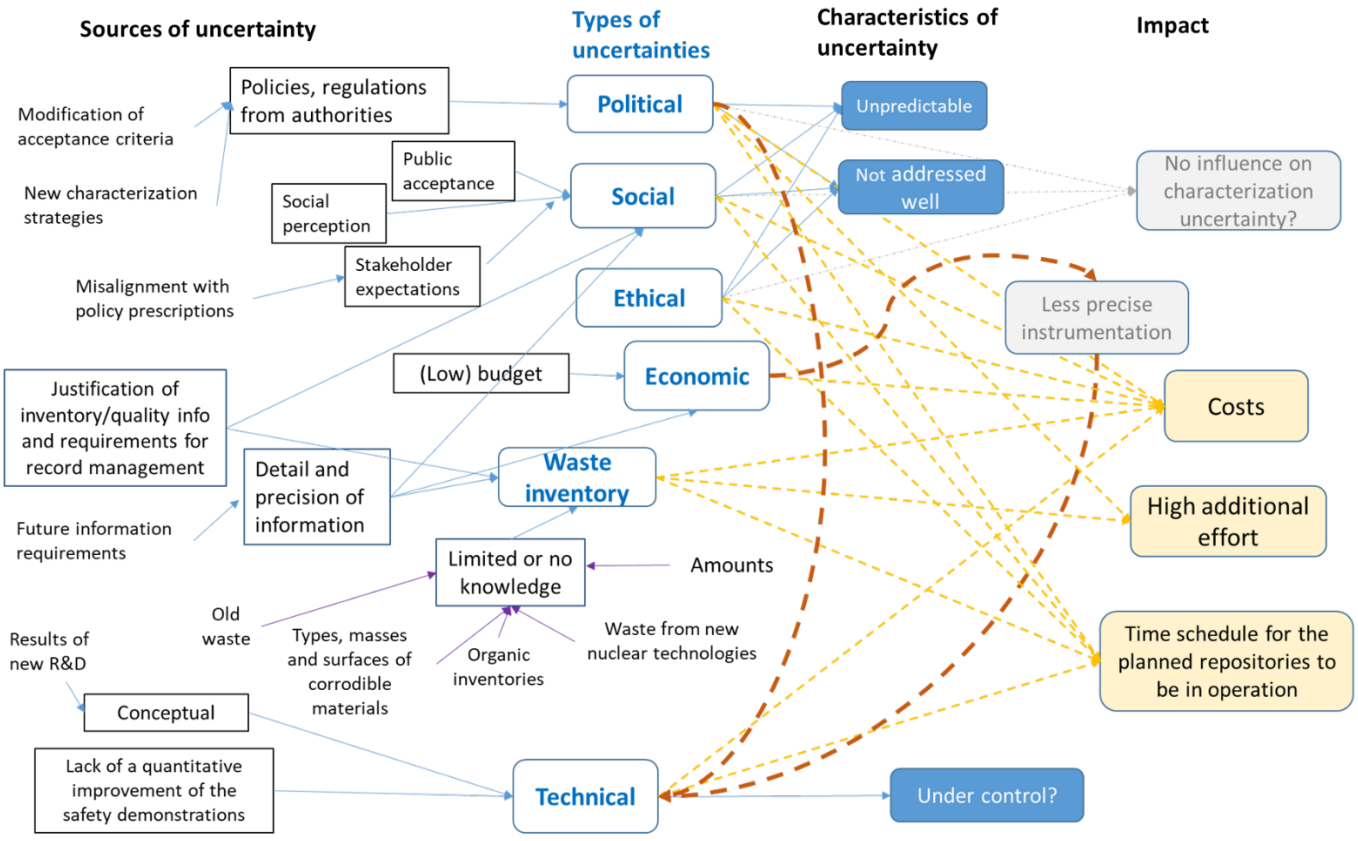

Fig. 1 Types and sources of uncertainty for waste characterisation, as well as potential impacts mentioned by survey respondents. An arrow indicates a relation between the two elements.

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For instance, one respondent pointed out that political uncertainties resulting from strengthening WAC may not only lead to higher costs, but may be difficult to implement from a technical viewpoint for already existing waste. Opposite to this, a relaxation of WAC requires attention to edge effects and the potential need for re-consideration of specific parameters/processes that did not need to be monitored under the previous, more restrictive conditions.

Several respondents highlighted current strategies to cope with uncertainties in waste characterisation and made suggestions on how these could be best dealt with. The responses are summarized in the table below. One respondent cautioned however that "*dealing with uncertainties is done case by case, country by country, and it is difficult to generalize*".

Table 1 Management of uncertainties as seen by survey respondents

Uncertainty	Strategies to deal with uncertainty in respondents' organisations / countries	Suggestions made by survey respondents
Waste inventory.	Regular revision of waste inventories. In Sweden, this includes building of final repositories, transportation, operation, maintenance, decommissioning and the final closure for the nuclear facilities. In UK, the revision covers all radioactive waste not only that destined for geological disposal.	*"Less dynamic policy decisions"*, such that "*circumstances and facts do not change too much over time*".
Waste inventory: Limited knowledge of old wastes.	Investigation of potential thermal treatment for problematic and poorly characterised wastes (organic wastes). Exploitation of available information (documentation, interviews with people involved in the production/conditioning of this waste,…).	Exploitation of available information should be done "*as early as possible in order to have the greatest effect*".
Waste inventory: Limited knowledge of types, masses and surfaces of corrodible materials.	Conservative assumptions (that might lead to higher disposal costs) and RD&D efforts to demonstrate the safety of these conservative assumptions. A first screening done to estimate as much as possible the types, masses and surfaces of metals present in the waste (based on available documentation) and to identity the potential problematic waste families. Derivation of useful information based on information delivered by producer for already produced waste, through analogies.	The producer should deliver information about types, masses and surfaces of metals present in the waste.
Waste inventory: Limited knowledge of organic inventory of wastes.	Conservative assumptions (that might lead to higher disposal costs) and RD&D efforts to demonstrate the safety of these conservative assumptions. RD&D to determine which types of organics are most impacting for long-term safety. Derivation of useful information based on information delivered by producer for already produced waste, through analogies.	The producer should deliver information about the organic inventory of wastes.

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Uncertainty	Strategies to deal with uncertainty in respondents' organisations / countries	Suggestions made by survey respondents
Ethical, social and political uncertainty: Social perception.	The regulator for radiation protection and environmental safety provides the waste management company a safety factor in terms of uncertainty, to be applied to the results of radiological characterization of radioactive waste, in particular when the characterization is finalized to the unconditional release.	"*Transparent and clean image towards the society*" of the waste management company, in order to induce positive thinking and trust in RWM.
Waste inventories: Understanding inventories: Detail and precision of information.		Balance between a precautionary approach (consider information in that future knowledge may reveal as important) and "*the present absolute necessity*".
Waste inventories: Level of justification of inventory: quality information and the requirements of waste package record management.	Periodic review of waste proposal assessments.	
Social: Stakeholder expectations: Misalignment with regulatory requirements.	Community engagement throughout the GDF siting process.	
Political: Evolving understanding of regulatory requirements	Ongoing engagement of the Radioactive Waste Management with the Regulatory body, and the Environment Agency.	There needs to be a driver for the characterisation requirement to justify the operational arrangements that would be needed to deliver it.

2.9. Do you have in your country / organisation waste categories and/or waste forms that do not have a dedicated option for disposal? If yes, are you characterising them?

For this question, the EUG members were asked to specify if they have waste with no dedicated option for storage/disposal, what are the potential limits for the acceptance of these waste categories in the existing or future disposal facilities, the plans for managing these waste and to specify what kind of measurements are done if this waste are characterized.

All the organizations seem to have to deal with some waste categories that don't have yet a dedicated option for storage.

In ***Belgium***, Ra-226 contaminated waste from the former Umicore facilities represents a waste category with no option for final disposal. Due to the Ra-226 activity levels, this waste category does not comply WAC for near surface disposal and since the volumes are too large, geological disposal

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should be excluded. Characterisation is ongoing to classify the waste according to activity levels and to optimise future management. For other waste categories, such as graphite waste, medium-active resins, low active liquid acid waste generated from the Medoc decontamination installation, silicagel, beryllium, metallic uranium, the Belgian national authority has not yet selected a conditioning technique and consequently a disposal rout was not decided.

In ***France***, mainly due to the large volume, the mine tailings generated by chemical companies (such as rare earth extraction) represents a waste categories for which it was not yet identified the final option for disposal. Other waste categories with no dedicated option for disposal are: sealed sources, waste contaminated with tritium (that needs pre-treatment or storage for activity decreasing) and waste with unknown chemical composition and/or unknown radiological activity. These waste categories need dedicated R&D actions to be disposed of, taking into account their chemical or physical properties. More details are presented in the Plan for Radioactive Materials and Waste Management (PNGMDR), on http://www.french-nuclear-safety.fr/Information/Publications/Others-ASN-reports/French-National-Plan-for-the-Management-of-Radioactive-Materials-and-Waste-for-2016-2018.

In ***Germany***, all the radioactive waste, including high level waste and SNF, which does not fulfil the WAC for the KONRAD repository are left in above ground interim storages at various sites. As the waste categories for which a disposal route was not decided yet was identified the radioactive waste retrieved from ASSE salt mine facility, the heat generative waste, historical/problematic waste, most of irradiated carbonaceous materials, pebble bed fuel.

Irradiated Graphite represents also a waste category with no decided disposal route in ***Italy***, together with materials containing NORM and TENORM. For irradiated graphite, acceptability in the future National Repository is limited to the Interim Storage section due to activity concentration values of long-lived radionuclides exceeding the limits for final disposal, but the NORM & TENORM materials are not acceptable, at this moment, in any section of the future National Repository due to legislative lack. No material containing irradiated graphite has been characterized up to now, while for NORM and TENORM materials some measurements were carried out by gamma spectrometry or by other counting systems such as gross beta-gamma/alpha or neutron counting (in case of Uranium/Plutonium presence cannot be excluded).

Since there are yet no operational NPP, in ***Poland*** there are some big volume and low activity wastes, e.g. originated from decontamination of laboratories, which cannot be transported to the repository and are stored in the place of their generation under radiological control.

In ***Romania***, irradiated graphite, beryllium, aluminium, boron carbide (that will be generated in the future from TRIGA research reactor decommissioning) were identified as waste categories for which it was not decided yet a disposal routs. For these waste categories, a R&D program was initiated to select suitable methods for treatment and conditioning in view of their disposal in one of the foreseen disposal facilities (near surface repository – for short-lived waste and geological repository – for long-lived waste and CANDU SNF). From these waste categories, only some irradiated graphite extracted from TRIGA thermal column was characterized in the frame of CARBOWASTE project (H-3, C-14, Eu-154/152, Co-60, and Cs-137). Low amount of thorium and uranium salts and powder of U3O8

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generated from research activities are stored on RATEN ICN site until a disposal route will be decided. From the radioactive waste generated from NPP operation, for the spent ion exchange resins generated from Cernavoda NPP operation it was not decided the disposal route since this waste category was not characterized yet and it is not known the level of C-14 inventory. Until a decision for SIERs treatment will be taken, these are stored under water in concrete vaults lined with epoxy, segregated as non-fuel contact resins (potentially containing high C-14 inventory) and fuel contact resins. The resins generated mainly from Moderator purification system (non-fuel contact resins) could contain high C-14 inventory that limit their disposal in the future near surface repository, but the fuel contact resins will be presumably be accepted in the near surface repository, after adequate conditioning.

Irradiated graphite was also identified in ***Spain*** as the most important waste category with no dedicated option for final disposal, mainly due to the large C-14 and H-3 inventory, but also due to the graphite interaction with other waste, engineered barriers or geosphere. The current plan for managing this waste categories comprise of selective decontamination of C-14 and/or create an impermeable coat for the graphite matrix. Other option could be the graphite crushing and its use as aggregate of a mortar in the disposal facility. In Spain, radiochemical analyses have been performed to determine the SFs and Co-60 measurements in every package, but there is no correlation of H-3, C-14 and Cl-36 with an easy to measure isotope and irradiated graphite radiological characterisation represent the most challenging process when characterizing package by package. *Calorimetry was indicated as a potential technique that could be further use in this characterisation process.*

Also in the ***UK,*** there are some minor quantities of wastes for which a difficult to define a disposal route, mainly due to the difficulty in characterisation. Due to the nature of the waste or operational challenges the traditional characterisation techniques won't work. An Integrated Project Team (IPT) on problematic radioactive waste was established in May 2016, to support the Nuclear Decommissioning Authority's (NDA) strategy (published in April 2016). The IPT is delivering a 4 year programme and it is led, on behalf of the NDA, by the Radioactive Waste Management Ltd (RWM) and LLW Repository Ltd. The objective of the Problematic Waste IPT is to develop a co-ordinated and improved approach to the industry-wide management of problematic radioactive wastes. Problematic waste, as defined by the IPT, includes radioactive wastes for which an existing waste management route is either not available or not optimised. It incorporates "orphan wastes" and "wastes requiring additional treatment". These are wastes from across the radiological spectrum, including lower and higher activity waste. The scope of the Problematic Waste IPT can be found published on the UK Government website (https://www.gov.uk/government/publications/scope-of-problematic-waste-integrated-project-team).

2.10. What are the major technical difficulties you encounter in characterising your conditioned radioactive waste?

For this question, the EUG members were asked to specify the conditioning methodologies and/or matrices used and the technical problems faced in the characterization process.

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Many different conditioning matrices are used across European countries, such as bitumen, cement, glass or polymers. Bituminisation is mainly applied for homogeneous liquids and sludges conditioning (for example in Belgium, France, Japan and Sweden) and for spent ion exchange resin (SIERs) conditioning (in Romania). Bitumen is also applied for heterogeneous solid waste conditioning. Polymers are primarily used for SIERs conditioning (for example in Belgium and in France for LLW disposal), while cement based materials are used for liquids, resins, solid waste, ashes, cartridge filters conditioning in most of the European countries.

The treatment technologies used across companies/countries that answer the questionnaire are also diverse. The following treatment technologies are applied: drying, evaporation, precipitation, incineration, melting, compaction &super-compaction, pyrolysis, vitrification of liquid residues and mechanical stabilisation.

As is apparent form the received information, matrices and conditioning methods vary on the basis of the waste types. Therefore, depending on their origin and final form, the conditioned waste can have different chemical and radiological composition and therefore specific and different characterization methods have to be used. Despite the use of various conditioning technologies and matrices, there are common difficulties related to the characterization process of the conditioned waste packages.

In some cases, the technical difficulties encountered in the radioactive waste characterisation depend on the purpose of the characterization itself, in particular because of the following reasons:

- clearance projects require devices with high precision and efficiency;
- the characterization of the waste with high dose rates can be made by low efficiency devices, but difficulties arise from the high constraints due to radiological protection issues;
- in situ characterization for classification requires rough and robust operational methods;
- a correct waste classification requires the use of operative characterization devices that optimizes the process properly and safely.

Characterization of the conditioned legacy/historical waste packages seems to represent a general problem. Radiologic characterisation of these waste packages, especially in the case of large packages, represents a big challenge either with passive or active measurements because of the difficulty of the interrogators radiation to penetrate the waste matrix and of the strong attenuation of the measurable emissions (gamma or neutron) by the waste package itself. Also the type of different materials in historical waste is difficult to identify. For example, in a Swedish project, an X-ray method was used to identify some different materials inside 200 litre drums. Moreover, the estimation of fissile materials quantities in the historical waste packages is affected by large uncertainties, which may cause an overestimation of the safety conditions to be applied both for the transport casks as well as for the disposal facility designing.

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The alpha and beta emitting radionuclides in already conditioned waste packages are difficult to be measured due to the signal attenuation caused by the waste package and the encapsulant, which could consist of irregular amount of steel (compacted drums) and concrete. These effects, in case of low or very low radioactivity or low energy emissions (such as the 59.54keV gamma line for Am-241), strongly affect the measurement efficiency, leading to relevant uncertainties on the measured activity. Consequently, determination of the correct source term in already conditioned waste represents a common problem. Further problems arise in detecting declarable difficult-to-measure isotopes (such as Sr-90) and sealed radioactive sources (SRS), specific accelerator based waste with yet undefined radionuclide-vectors and their material vector, and in the SNF for direct disposal.

Little traceability of the chemical content of waste packages, when the conditioning is done by the waste producer, accessibility of the waste for sampling (due to the limited access at the waste packages), difficulties in monitoring and periodical control of the waste drums packed deeply in a storage facility, characterization and reconditioning of the waste already stored in a repository, and those related to both the lack of standardized processes for the characterization and repackaging (or reconditioning) of spoilt drums/containers were also identified as a problems in conditioned waste characterisation.

The available characterization methods are not always sufficient due to the lack of simple and safe non-destructive methods. Therefore, more advanced non-destructive methods could be useful both for the waste management organizations (WMO) and for the operator of disposal facilities, in order to increase the accuracy of the radionuclide inventory disposed of. Thus, the general objective of WMO is to develop their expertise on advanced characterisation methods for both the radioactive waste for which the final disposal option is already available, and for those that need to be temporarily stored until a disposal option will be available.

2.11. What are the R&D needs that could solve the difficulties identified in Q10?

For this question, the EUG members were asked to identify the techniques/methods that could complement the ones already used to improve the level of radioactive waste characterisation.

Related to the characterization process, each organization has some R&D challenges aimed to solve their main difficulties.

One respondent considers that the uncertainties and therefore R&D needs are different for each case and consequently a one size fits all approach is not suitable. There are a range of potential issues that may need to be understood and therefore the key R&D challenges may be:

- Low limits of detection
- Diversity of characterisation needs
- Diversity of wastes
- Challenging environments (remote access)
- Working on a nuclear site

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- Working to underpin assumptions in post closure safety case (e.g. regarding C-14 gas)
- Representativeness of sampling (small scale, heterogeneity)

For many organizations, the priority is the research and development of new *non-destructive methods* capable to detect the radiological (including alpha and beta emitters) and fissile mass content with an acceptable uncertainty, compatible with the WAC for storage, transportation and final disposal. These methods should be able to be applied in homogeneous and heterogeneous waste in packages of different sizes, including SNF casks. One respondent proposed that the methods developed for pure beta and alpha emitters have to be applied in the raw waste characterisation to have a complete spectrum of the radionuclide content, or at least to calculate, based on measurement carried out on different waste streams, the SFs.

The development of Gamma Cameras that passively determine the activity distribution in the waste package and of methods to identify the mass distribution in the package, without the necessity of gamma sources (to avoid heavy irradiation) was suggested by one respondent as R&D need to solve the technical problems in characterising the conditioned radioactive waste as well as to be used in the dismantling process. The new equipment could also be useful at disposal facilities as good quality control devices, saving radiological protection issues of heavy sources and saving time in analysis, increasing the quality control process.

Since the major difficulty is identified as the matrix/source distribution, for error estimation, matrix effects and discrepancies between neutron counting assay and gamma scanning a set of user-defined mock-up overpacks are used in some cases. These mock-up overpacks are filled with different matrices (such as wood or steel) that are cemented, resembling typical conditioned radioactive waste. Into these mock-up overpacks point or linear certified sources can be placed.

The major improvements for the actual characterisation methods should be related to the quantitative interpretation and use of the imaging techniques output, both for the matrix distribution (e.g. using the X-ray detector or the tomographic approach) and for the contamination distribution (e.g. using tomographic scanning or imaging systems). This kind of information could be used to further improve the description of the measurement setup in the efficiency evaluation, thus improving the results while lowering the associated uncertainty.

There is a need for improvements that can lead to more sensible measurements, as well as improvements in measures that are not subject of interferences (or at least that are not subject to the same interferences in order to be used in combined measurements). Other needs are: more precise characterization of wastes at the place of origin, more specialised laboratory equipment allowing full characterisation before treatment, as well as the development of new, non-interfering methods for waste characterization.

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2.12. Do you have an active R&D programme on radioactive waste characterisation?

For this question, the EUG members were asked to specify the main topics addressed on their own R&D programme (if they have one) and if they are interested to be involved in R&D projects related to radioactive waste characterisation.

Most of the organizations are involved in one or more active R&D programs on radioactive waste characterization.

In ***Belgium***, efforts are under way to delve deeper into the uncertainties related to the radiological content of non-conditioned waste packages and to exploit as much as possible the available documentation for some historical waste.

In ***France***, R&D programmes are currently developing analyses techniques that reduce the detection limit.

The following techniques are under study and will lead to experimental validation during the coming years:

- Active Photon Interrogation: fissile mass quantification by photofission delayed gamma rays;
- High Energy Imaging – dual energy: quantification of density and mass number Z;
- Cavity Ring-Down Spectroscopy for tritium outgassing measurement;
- Active neutron interrogation for fissile mass quantification.

In ***Germany***, there is on-going material research for the safe disposal in a HLW repository.

In ***Italy***, in order to improve the company know-how and being up-to-date with the current technologies, new strategies and techniques for radioactive waste characterization are currently developed, both in terms of analysis/data processing skills and in terms of equipment/instruments. Starting from databases and calculation forms for process and data integration, and continuing to new strategies for a better efficiency calculation (based on acquired data of activity and matrix distribution) and re-analysis processes, the final data can be sensibly refined and distributed with higher accuracy. New Cadmium Zinc Telluride CZT detectors for in-situ activities and surveys to be used for gamma scanning and imaging coupling are currently tested. For on-site drum measurements the Tomographic Gamma Scanner (TGS) system is being continuously investigated for a better understanding and for determining any possible improvement (both hardware and software).

In ***Poland***, R&D projects concern the development of destructive methods of chemical analysis like ICP-MS, ion chromatography and TOC (Total Organic Carbon), namely for characterisation of organic compounds.

In ***Romania***, RATEN is managing an R&D program dedicated to SNF and radioactive waste management that has a dedicated research theme on radioactive waste characterisation, but currently only destructive methods are developed and tested. No research is carried out for developing non-destructive characterisation methods.

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There are R&D efforts in ***Spain*** focused on the development of gamma cameras and of new methods for very difficult-to-measure isotopes, for improving the detection limits by means of Accelerator Mass Spectrometry (AMS) or new radiochemical methods combined with mass spectrometry (ICP-MS) techniques.

The ***Swedish*** waste management organization (SKB) has no on-going characterisation projects, but R&D efforts are made for improvement of calculation and modelling.

Waste characterisation projects are ongoing in specific organisations in the ***UK*** to address specific challenges on a needs driven basis. This knowledge and learning is not necessarily accessible across the industry therefore it's difficult to learn from the experiences and developments that others have made. *A coordinated approach would be beneficial to understand more about the techniques available, their applicability to various situations with the ability to gauge how well certain technique could address certain challenges.* Therefore, an integrated and coordinated UK level approach to characterisation is considered to be able to bring benefit the industry in the UK.

All respondents are interested to be involved in collaborative R&D on characterisation of conditioned radioactive waste as the best way of improving the current status.

2.13. The CHANCE project will address and develop some specific techniques: Calorimetry, Muon Tomography and Cavity Ring-Down Spectroscopy (for details see www.chance-h2020.eu).

Do you foresee an application of one of these techniques in your radioactive waste characterisation?

For this question, the EUG members were asked to identify the waste type for which the methods developed in CHANCE could be used and how these methods improve the characterisation of conditioned radioactive waste.

Most of the respondents see the use of the new characterisation techniques as a means of decreasing uncertainties associated to the characterisation of different category of conditioned radioactive waste.

In ***Belgium***, no screening of the potentially interesting wastes has been done using **Calorimetry,** but for **Muon Tomography,** the large amount of waste suspected for gas (hydrogen) build-up could represent a potential application. Also, the Cavity Ring-Down Spectroscopy could be potentially applied on irradiated graphite (future waste, no fixed date).

In ***France***, **Calorimetry** is seen as useful technique in diminution the uncertainty and/or detection limit, while **Muon Tomography** is regarded a technique that could replace the X-ray high energy for package with high biological protection. The calorimetry could have a possible application for some specific waste packages, particularly for those containing tritium and plutonium, in conjunction with other nuclear measurement techniques, taking into account the industrialization feasibility. The

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Cavity Ring-Down Spectroscopy could be applied for the monitoring of all kind of waste packages, and particularly graphite waste for the measurement of H-3, C-14 and Cl-36 outgassing.

In ***Italy***, for industrial applications, techniques capable to provide, in relatively short time, quantitative results that improve and/or integrate data already obtainable by the other existing systems are required. On the basis of the type and size of items usually characterized, ***Calorimetry*** is considered to be the most suitable technique for quantitative radiological characterization, and it can be easily coupled with already operating systems. Also, the development and implementation of a **Muon Tomography** for packaged radioactive waste is considered to be precious for a better characterization of conditioned waste and heavily shielded nuclear material (like spent sealed sources).

In ***Romania***, **Calorimetry** could be applied to identify sealed spent sources in drums accepted for disposal in National Baita Bihor Repository, while the Cavity Ring-Down Spectroscopy could be used to monitor the H-3 and C-14 outgases from drums with radioactive waste in storage on Cernavoda site.

The ***Spanish*** respondent considers that **Calorimetry** combined with gamma spectrometry will bring improvements to the currently used non-destructive techniques for difficult-to-measure isotopes in waste packages, while the **Muon Tomography** combined with gamma imaging will improve the quantification of gamma emitters. The **Cavity Ring Down Spectroscopy** is considered a very useful tool to monitor in real time the emission radioisotopes (in water vapour, CO_2,…) in repositories and also for instance in the reactor caissons in case of graphite moderator stack placed inside where we know that there is radiolysis and emission of this isotopes to the controlled atmosphere of the caisson.

In ***Sweden***, there are no plans for using the methods developed in CHANCE at the moment since these methods need first to be proven and validated before their using in characterisation process. Furthermore, to be used it has to be evaluated if a particular method/concept also brings sufficient added value to the process.

The techniques developed in CHANCE are foreseen as having potential applicability to ***UK*** existing sites and to the future geologic disposal facility (GDF). The diverse range of wastes and challenges mean that the applicability of each technique would need to be assessed on a case by case basis. The techniques being developed may be useful in support other techniques that are currently deployed on site and as confirmatory measurements. From a GDF perspective, there is potential applicability for the future in terms of the waste package receipt strategy and support to waste package records.

2.14. In your country, who is in charge of characterization and who is in charge of control?

For this question, the EUG members were asked to specify the organisation(s) responsible for radioactive waste characterisation, organization is in charge with control, how the characterisation and control processes are structured and how these processes are follow up.

A first thought with this question is that the answers show a great variety of approaches to characterisation processes in the different countries. Also given the amount and character of

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interesting detail the respondents gave, this synthesis cannot and does not want to replace the collection of answers to the full extent.

All but one of the respondents specified that the responsibility for the characterisation of the radioactive waste is basically with the waste producers. One respondent included also the waste conditioners and another one the owners of the interim storage facilities. In one case, the characterisation is said to be in the hands of the waste management facility.

However, there seem to be different approaches to (or understandings of) the process of characterisation. Some respondents speak of 'characterisation' as such while others specify that producers are expected to only provide data about "*the radionuclide composition of their wastes and what radiation dose rate they generate*" while others call it "*the isotopic content*". In addition, sometimes it is mentioned that the producers are also responsible for the quality control of that data while, in other cases, quality control is said to be in the hands of the controlling organisation, expecting the producers to only provide the data. Finally, one respondent specified that in the case of dismantling, the characterisation of the waste is solely in the hands of the national waste management agency.

As organisation in charge of control, as well the (national) waste management agencies as the national control authority (or regulator) are mentioned. In one case, the distinction is made between 'control' (the waste management agency) and 'audit' (the regulator). From the answers, it is not always clear whether control is to be understood as overall surveillance of the compliance of the process with regulation or as quality control or both.

With respect to structuring of characterisation processes and their follow-up, the answers show a great variety of approaches in the different countries. In sum, specifications relate to:

- the different control methods (e.g. destructive and non-destructive; in situ or at the waste management plant, …);
- the verification of storage conditions;
- the approval of operational treatment plans;
- the need and character of additional measurements on conditioned waste packages;
- the method of collection of the waste (storage and transportation);
- the need for and nature of additional (regular or ad hoc) inspections and audits;
- whether controls should be done directly on the waste or (also) on final packages at the end of the process;
- involvement of communities in the interest of shared learning and information exchange (see also Q15).

Detailed information about the specific approaches in the different countries is presented below.

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Country	Organisation responsible for RW characterisation	Organization in charge with control	How the characterisation and control processes are structured	*how the characterisation and control are follow up*
Belgium	The waste producers and conditioners.	ONDRAF/NIRAS (WMO)	-	- Characterisation by the waste producer: proposed in qualification file; - ONDRAF/NIRAS reviews and approves qualification file; - For each waste package, the documentation is checked to verify if the package complies with the WAC; - Regular inspections by ONDRAF/NIRAS to verify declared information, used methods.
France	Producer is both in charge of characterization and to implement a quality control in order to verify that its characterization is valid.	The WMO (Andra) carries on its own controls when the WP arrives at the disposal facility. Andra, or other official quality assurance Companies, can also performs 2nd level non-destructive and destructive controls on LIWL – SL waste.		Characterization method and quality control are described in an agreement (document) that describe how is made a waste package and that is mandatory in order to send package to storage facility; Audits are made in order to check the compliance with what is really made on production site; The storage facility also implements control. Currently based on final waste package both for LILW and HLW, question is asked to know if the control still have to be at the end of process on final package (high cost, lot of induced waste generated, dilution or attenuation that make controls harder) or directly on waste or primary package (controls made inside producer installations) with different pros and cons (dependent of producer installation's authorisation, less control on experimental conditions, less attenuation or dilution, less induced waste expected, not a control on final package).
Germany	Generally, the owner of the waste or the owner/operator of a process is obliged to provide the QA data for the subsequent process' WAC.	The characterization data and underlying process parameters are verified and certified by independent inspectors on behalf of the regulator.	The process structure is part of the regulator's quality approval for the whole campaign or waste line production process.	
Italy	Producers and/or of the owner of the interim storage facilities.	Italian Control Authority (ISPRA)	ISPRA performs surveillance both for the verification of storage conditions and for the approval of operational treatment plans.	Periodic and ad hoc inspections performed on site by ISPRA, with eventual issue of technical prescription.

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Country	Organisation responsible for RW characterisation	Organization in charge with control	How the characterisation and control processes are structured	*how the characterisation and control are follow up*
Poland	Organizational units of the Radioactive Waste Management Plant (RWMP).			
Romania	The radioactive waste generators are in charge with waste characterisation, as well as for all predisposal activities.	The operator of radioactive waste disposal facility (IFIN HH in the case of Baita Bihor repository) has to control the compliance of the radioactive waste received for disposal with the WAC.		
Spain	Big producers are in charge of providing ETM isotopes in packages. Enresa is in charge of providing the SFs values by collecting samples from radiochemical analysis. In dismantling projects, Enresa/dismantling area is in charge of providing the whole isotopic content in packages/Items. Small producers from hospitals, research centres or laboratories provide the isotopic content.	Enresa is in charge of control that the activity provided by the producers is done as it is required by the specifications. Regulatory body audit the whole process, from the producers to Enresa.	Enresa performs control processes in three ways: - in situ (NPP): Enresa performs in situ gamma analysis in waste packages in NPP in order to check the producer activity determination; - by systematic analysis of gamma dose rate of the data sent by the producer over the whole production, before transport to el Cabril;. - by test at El Cabril laboratories performed over packages from different NPP, performing both non-destructive (Gamma spectrometry) and destructive test (Beta and alpha emitters).	There is an systematic annual planning of all these control processes trying to cover all waste streams and all producers every established period (1 or 2 years) always in relation to the previous year plan. As a function of the result of the mentioned processes, a tracking action is required.
Sweden	Waste producers, NPPs and other nuclear installations including SKB.	Both NPPs and SKB as WMO are performing control. In the final end it is controlled by the authority SSM.	NPPs are normally performing sampling and measuring of the waste. SKB are modelling for DTM nuclides that could also be done by the NPPs and other nuclear facilities. In the waste package specifications the normal content of radionuclides is described, this document is approved by the authority. Every single waste package is measured for gamma emitting nuclides and dose rate and is reported for each individual package. Before disposal in SFR for LILW a quality process are performed	Audits, yearly reporting, quality control during sending and receiving packages at the final repository SFR.

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Country	Organisation responsible for RW characterisation	Organization in charge with control	How the characterisation and control processes are structured	*how the characterisation and control are follow up*
			before transportation and final disposal with up to 90 different check points according to transportation, disposal and logistical rules at SFR. All of the waste packages should meet the requirements according to the license from the authority, this process are checked in an administrative tool.	

2.15. What is the role of host communities in these processes?

For this question, the EUG members were asked if in their opinion:

- *the host communities have a role in waste characterization and/or control?*
- *these communities should have a role in waste characterization and/or control if so which one?*
- *how they are dealing with the concerns of local communities, e.g. concerns about the content of the disposal facility?*
- *they are thinking that continuous improvement of waste characterization by innovative methods can improve the perception of risk associated with waste disposal?*

Host communities seem to play a role in various ways in the different countries.

Some respondents mention explicitly the involvement in the public authorisation process or licensing procedure for 'new installations with radiological risks', either through traditional environmental assessment procedures or (more active) via local community representation (e.g. in the form of a partnership) with a specific autonomy in the process of decision making and follow-up.

Other respondents specify that host communities are only informed of the waste management activities, either in a proactive way (through information channels or by inviting them to regular meetings) or upon their own request.

Answers also differ on the question whether host communities should be actively involved in decision making or not:

- Some respondents say that active involvement is the best way to create transparency and build public trust. One respondent suggests that structures such as partnerships should be put in place "*to ensure permanent, long-term involvement and participation of the local communities*". One other respondent also says that characterisation in itself can build "*confidence in our understanding of the waste and its evolution with time*". Another one acknowledges that the purpose of involvement should not be "*to convince the public and other*

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stakeholders to accept a nuclear investment, but rather to involve them in decision making process (DMP)", and calls this "*a smart way to share the responsibilities and to build sustainable decisions.*"

- Others, however, say that they have reasons to believe local citizens are aware of the various waste management research and industry activities and have trust in these whatsoever, without the need to involve them. One respondent also suggests that waste characterisation is a very technical process too difficult to grasp for the local public.

One respondent highlights the role local communities have played in various international research projects related to governance of radioactive waste and stresses that, in doing this, they 'succeeded to learn from other countries' experiences the methods and tools to be used'.

Finally, one respondent stresses that the interests and concerns of the host community may determine the characterisation process as such: "*What is important to the host community may drive requirements on characterisation in the future. It is recognised that there is a role for characterisation in terms of stakeholder engagement and building confidence in our understanding of the waste and its evolution with time.*"

2.16. Which disciplines / fields of expertise / actors are involved in the characterization of conditioned waste in your country? Are there any missing in your opinion?

For this question, the EUG members were asked to specify the disciplines, fields of expertise and actors involved in radioactive waste characterization and if there are missing ones.

For this rather general question, respondents' answers cover a wide spectrum of disciplines / fields of expertise, although mostly in the area of natural sciences and engineering:

- chemistry;
- (nuclear / radiation) physics;
- biology;
- geology;
- 'measurement';
- reactor operation.

As 'operational' fields of expertise are mentioned:

- safety, safety assessment;
- radiological characterisation.

With regard to what respondents think is missing, answers vary from general fields of expertise to specific missing actors, procedures and regulation.

One respondent is very specific stating that missing are "*actors defining appropriate and achievable (societal) targets; actors accompanying society, i.e. societal changes over the project life-time & actors communicating with the public and capable of dealing with irrationalities and anxieties, asocial requests.*"

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Others highlighted the lack of

- regulation on waste conditioning procedures in accordance with the current radioactive waste classification;
- 'advanced methods' for a more complete characterisation of the waste;
- expertise of representativeness of sample selection / quantity & number of samples collected.

One respondent says that nothing particular is missing, but that "*more integration between disciplines would be useful*". Finally, one respondent stresses that, in their situation, "*cross disciplinary teams are deployed on site to address specific characterisation challenges*".

2.17. Why is waste characterization important for your organisation?

This question inquired about the importance attributed by respondents to various aspects, with respect to waste characterisation. Some respondents reflected the views of waste producers, thereby focusing on non-conditioned waste; others reflected the views of waste management organisations.

Overall, results seem to suggest that waste characterisation is seen by most respondents as a problem mainly driven by technical considerations (Figure 2). The respondents' answers highlight *verification of the declared inventory* as the key and most important reason for waste characterisation, followed by the *operational safety*, and the *waste classification in view of disposal choices*. All respondents consider the regulatory requirements as a very high or highly important driver for waste characterisation.

Social and political uncertainties were mentioned by several respondents as important uncertainties with potentially high impact. Conversely, half of the respondents attributed a high or very high importance to stakeholder involvement as a reason for waste characterisation, and 9 out of 12 to communication.

One respondent suggested that quality management and incident management bring in an important aspect, "*linked to communication but separate from it*", which had not been specified as such in the initial version of the questionnaire. Consequently, some of the survey responses took into account also this aspect, which was deemed to have a very high or high importance as a justification for waste characterisation by 5 out of 12 respondents.

Only one respondent justified the importance attributed to the different aspects. For instance, it was argued that verification of the declared inventory has a high importance because "*most characterisation is done prior to conditioning, and the characterisation of conditioned waste is a smaller part of that*". In the same way, regulatory requirements were deemed to have in general a high importance, but since "*characterisation requirements are needs driven*", "*the importance [of regulatory requirements] will be defined on a case by case basis*" and "*characterisation may only be necessary if confidence is needed in the pre- conditioning characterisation*".

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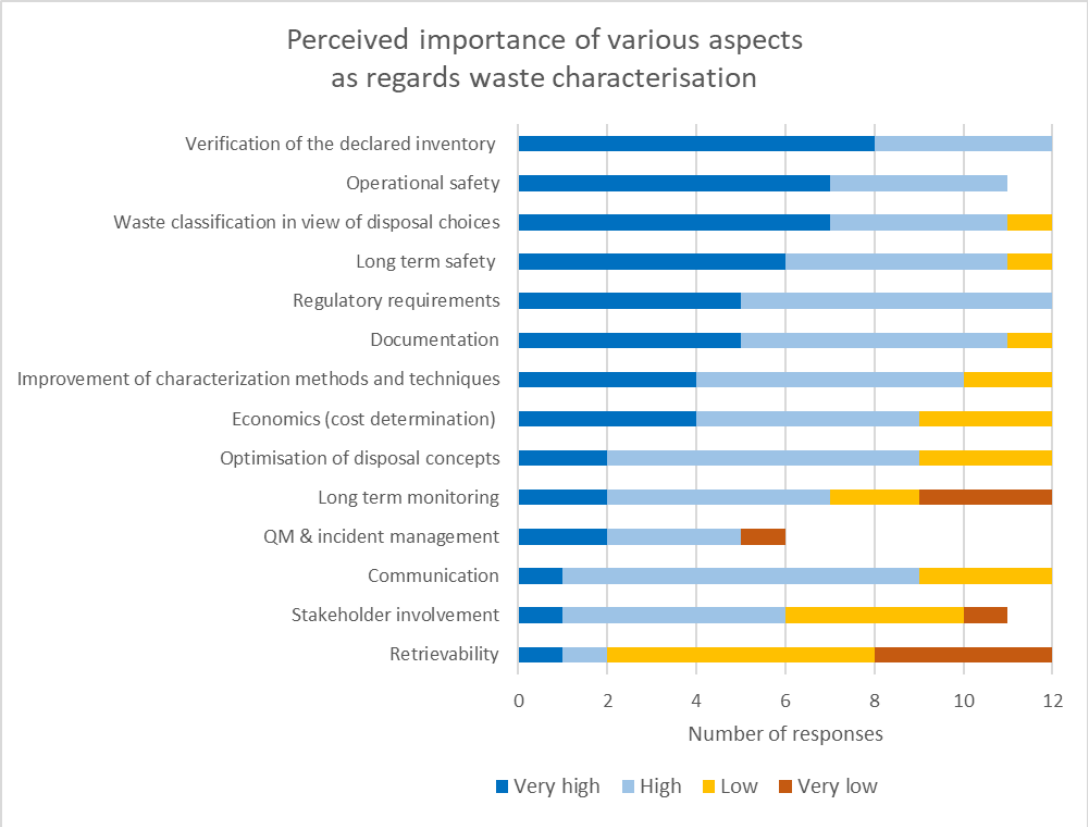

Fig. 2 Perceived importance of different aspects for waste characterisation

Another respondent commented that "*the entire measurement chain deployed for the waste by the [..] producer, up to the one used in the manufacture of the waste package, must be proof for [the] waste management agency*".

It should be noted that the formulation of the question contained some semantic ambiguity in whether the scoring of importance referred to the justification of waste characterisation (why it is being done?) or to the aspects that are of importance in the process of waste characterisation (how it has to be done?). The analysis was done mostly taking the former perspective.

2.18. What lessons can be drawn from the waste characterization processes in your country?

Various lessons learned were highlighted by the respondents. Similar to the case of Question 14, a synthesis summary would do injustice to the variety of useful information given in the respondents' answers. Therefore, this synthesis cannot be more than a rough indication and cannot and does not want to replace the specific in-depth answers of the respondents.

Most answers are in line with the observation made by one respondent, namely that waste characterisation is in constant evolution, bringing along the need for advanced methods and techniques and new ('better') parameters and criteria:

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- *"WAC [evolves] constantly, making new parameters important. The question then is how to fulfill [the] criteria for historical waste when in the past these criteria were not yet asked or important*".
- "*The system for waste characterization must be upgraded to deal with the new radioactive waste classification and this upgrade could in the future strongly reduce the amount of material for free release. At the same time the characterization system must be harmonized with the new standard for waste treatment in terms of conditioning process*";
- "*The creation of new standards, regulations [is] necessary in order to harmonise the approach of different institutions (national and from […] abroad) to the waste characterisation. […] There are also social aspects of waste characterization, very important for new investments in [the] nuclear sector*";
- "*Skills and techniques applied in radioactive waste characterisation have to be constantly improved to ensure the compliance of the waste packages with the WAC defined for the disposal facility, and to have a characterisation as complete as possible of the radioactive waste generated […]*";
- "*More improvements have to be done concerning old legacy waste*".

Other lessons learned refer more to the improvement of the characterisation process itself. Some quotes:

- "*Characterization of nuclear waste has to be done at very early stage, during waste production, by taking into account (if existing) all the WAC for handling, storage, transportation and final disposal*";
- "*Considering […] the various streams of waste in terms of radionuclides and materials the characterisation must be adapted to the process and nature of waste. Consequently, characterisation methods have to take into account all the production parameters – in order to determine the best choice for a specific case. […] There is no generic way of characterisation*";
- "*Interaction and a good understanding between WMO and waste producer are necessary to obtain reliable characterisation, especially chemically*".
- "*The high amount of legacy waste that needs further characterisation, and the difficulties associated to this, proves that it is useful to characterise and document waste as good as possible at the moment it is produced, even if the information is not deemed useful at that time since it might be required later*".
- "*Regular inspections on site are essential to guarantee the effectivity of the characterisation process*".

2.19. Open comments?

Two respondents made comments in the 'open comments' section.

One respondent noted that, not being a producer, it remained difficult to answer some of the questions.

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The other respondent referred to specific challenges related to the planned construction of a new repository in the home country, noting that "*[…] the development of criteria for safe storage in the new repository will require substantial research on the characterization of conditioned waste and advanced methods with the use of specialized equipment. A qualified and trained staff is necessary to achieve these goals*".

3. Conclusions

This report comprises the answers and results extracted from the EUG questionnaire of the CHANCE work package 2 addressing specific but also cross-European requirements for the characterization of conditioned and disposable RW. Identifying common practice and potential for harmonizing waste acceptance procedures or, in turn, identifying metrology gaps and currently remaining R&D needs is also an anticipated outcome of this report, which may also feed into recommendations for further improvement and optimization.

1. Key parameters required for characterisation

No matter what categories of RW managed, the requirements for characterisation are driven by the WAC, either for storage facilities or repositories. The following parameters are usually required to be assessed through RW characterisation:

- specific activities for ETM and DTM radionuclides;
- the fissile nuclides (Th-232, U-235, Pu-239 and Pu-241) – are usually required to be measured to ensure sub-criticality in the disposal facility;
- specific heat power – it has to be specified particularly for geological disposal facilities;
- complexing and chelating agents, accelerators of leaching processes, organic substances, pyrophoric, flammable, explosive, corrosive or oxidizing materials;
- nuclide vector compatibility and material vector compatibility has to be ensured;
- material vector (metallic, ceramics, rubble) has to be declared for some storage/disposal facilities;

2. Technologies/methods commonly used for characterisation of conditioned RW

- NV and SF methods are widely used in many European countries (e.g. Belgium, France, Germany, Spain, Sweden) for the characterisation of standardized RW streams;
- direct information about the origin of waste streams under consideration is also used in many cases;
- gamma spectrometry (open-geometry for the whole RW package, or segmented /collimated) is the widely non-destructive techniques used in the characterisation of conditioning RW;
- dose rate conversion with approved NVs;

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- spectroscopic techniques (alpha, beta and gamma spectrometry) - are applied in the majority of the institutions involved in RW characterisation either for inspection or in the process of SF development;
- neutron measurements (active and passive neutron measurements) are used in some specific cases (e.g. Belgium, France, Italy) to identify the fertile and fissile materials;
- calorimetry, applied mainly for ETM isotopes, is used by a limited number of institutions (e.g. France, Germany);
- X-ray inspection – used mostly to analyse the matrix and its density distribution.

3. WAC and the possibilities of their harmonization in Europe

There are *applied WAC* at the operational disposal facilities or interim storage facilities and *preliminary WAC* for the disposal facilities, both for LILW-SL waste and ILW-LL, HLW and SF foreseen to be in operation in the future. For all disposal facilities, the parameters concurring to define WAC are grouped in *radiological*, *chemical*, *mechanical* and *other* parameters.

Radiological parameters:

- For the disposal facilities for LIL-SL waste:
    - radionuclide specific activity per waste package – for safety relevant RNs;
    - total activity: for the whole disposal facility or even per disposal zone;
    - dose rate at surface and certain distance and integrated dose;
    - surface contamination (for beta-gamma and alpha emitters);
    - limits for total β-γ and α emitters are imposed in some cases;
    - content of fissile materials.
- For the *geologic disposal facilities:*
    - generally the preliminary WAC do not impose limits for specific activity nor for the total activity;
    - activity of radionuclide by radionuclide has to be known and documented;
    - content of fissile materials;
    - heating power.

Chemical parameters:

- inventory of toxic species;
- limitations on the complexing and chelating agents;
- limitations on the accelerators of leaching processes;
- limitations on the organic substances, pyrophoric, flammable, explosive, corrosive or oxidizing materials;
- waste in liquid form or waste containing liquid are not allowed;
- biologically active waste (infectious or putrescible) are generally forbidden.

Mechanical parameters:

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- compression resistance;
- void limitation;
- drop resistance (also required by transport regulations);
- swelling;
- handling/stacking parameters (weight/package size and centre of mass);
- the waste matrices have to confine in a significant way the radionuclides (limits on diffusivity and leachability).

Other parameters:

- homogeneity of the waste;
- types of conditioning matrices;
- specification for the disposal container (physical dimensions and weight, labelling/identification).

Opinions on WAC harmonisation

- for ***safety relevant parameters,*** harmonisation is considered by a large majority of respondents neither possible, nor desirable;
- in terms of the ***underlying rationale for WAC***, the ***basic assumptions for safety studies***, or the ***potential parameters to be evaluated through WAC***, harmonization may be appropriate and achievable.

4. Specific problematic issues for the characterisation of conditioned RW

- proper characterization of the ***conditioned legacy/ historical waste packages*** – a general problem, especially for countries with long-term utilization of nuclear technology (Be, Es, Fr, De, It, the UK);
  - radiologic characterisation - a major challenge to either passive or active measurements;
  - the type of different materials – often difficult to identify.
- determination of a viable source term in already conditioned RW;
- detection of declarable DTM isotopes (such as Sr-90) and sealed radioactive sources (SRS);
- other waste category considered difficult to be characterised:
  - accelerator based waste with radionuclide-vectors and material vectors not yet defined;
  - SNF for direct disposal.
- little traceability of the chemical content of waste packages, when the conditioning is done by the waste producer;
- accessibility of the waste for sampling (due to the limited access at the waste packages);
- difficulties in monitoring and periodical control of the waste drums packed deeply in a storage facility;
- characterization and reconditioning of the waste already stored in a repository;

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- the lack of standardized processes for the characterization and repackaging (or reconditioning) of spoilt drums/containers.

5. R&D needs and potential on-going R&D programmes

- Developing of ***non-destructive methods*** **capable to detect the radiological (including α and β emitters) and fissile mass** content with an acceptable uncertainty, compatible with the WAC for storage, transportation and final disposal.
- The new developed methods should be **able to be applied:**
  - for **homogeneous and heterogeneous waste;**
  - on waste packages of **different sizes**, including **SNF** casks.

The R&D needs will be further developed in CHANCE Task 2.2 and reported in Deliverable D2.3.

6. Potential application of the techniques addressed in CHANCE

Calorimetry is seen as a useful technique in:

- ***diminution the uncertainty and/or detection limit*** for conditioned RW such as waste containing H-3 and Pu (in conjunction with other nuclear measurement techniques);
- ***to identify sealed spent sources or alpha/beta emitters*** with considerable heat load hidden in packages to be disposed of;
- Combined with gamma spectrometry, ***Calorimetry*** is considered to ***bring improvements to the currently used non-destructive techniques*** for DTM isotopes in conditioned waste packages.

CHANCE attempts to make calorimetry applicable for large volume, heterogeneous and multi-nuclide waste packages.

Muon Tomography is regarded as a technique that:

- ***could replace the X-ray*** high energy for package with high biological protection;
- stand-alone or combined with gamma imaging, ***Muon Tomography*** is considered valuable ***to identify the location and amount of high-Z, potentially fissile material, position and integrity of SNF assemblies*** in very large storage casks, unwanted density depletion from voids or bubble formation in cemented drums;
- possibly to be used for ***monitoring interim storage halls***, maybe even in ***filled and sealed repository galleries***.

Cavity Ring-Down Spectroscopy is considered a very useful tool for:

- the ***extremely low limit outgassing monitoring*** of waste packages containing ***graphite waste*** exuding very small amounts of H-3 or C-14 or for the detection of Cl-36 outgassing;
- ***measuring in real time the emission of radioisotopes*** (in water vapour, CO_2,…) ***in repositories*** and even in the ***reactor caissons***, in case of graphite moderated reactors.

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A coordinated approach is considered to be beneficial in order to understand more about the techniques available, their applicability range for various situations and with the ability to judge how well certain technique could address certain challenges.

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Glossary

AAS: atomic absorption spectrometry

CMI: condition, monitoring and inspection

CRW: conditioned radioactive waste

CZT detector: Cadmium Zinc Telluride detector

DA: destructive analysis

DTM: difficult-to-measure

EUG: End-User Group

ETM: easy-to-measure

EW: exempt waste

GDF: geologic disposal facility

GP: guaranteed parameters

HAW: higher activity radioactive waste

HLW: high-level waste

ICP-MS: Inductively Coupled Plasma Mass Spectrometry

ILW: intermediate-level waste

LILW: low- and intermediate-level waste

LILW-LL: low- and intermediate-level waste, long lived

LILW-SL: low- and intermediate-level waste, short lived

LLW: low-level waste

LS/CRDS: liquid scintillation/ cavity ring-down spectrometry

NDA: non-destructive analysis

NV: nuclide vector

OCP-MS: optically coupled plasma mass spectroscopy

QA: quality assurance

RCA: radio-chemical analysis

RR: research reactor

RW: radioactive waste

RWM: radioactive waste management

SF: scaling factor

SNF: spent nuclear fuel

SRS: sealed radioactive sources

TOC: total organic carbon

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VLLW: very low-level waste

VSLW: very short-lived waste

Reference

1. NEA Report 7323 “National Inventories and Management Strategies for Spent Nuclear Fuel and Radioactive Waste: Methodology for Common Presentation of Data”. Radioactive Waste Management, 2016
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4. https://eur-lex.europa.eu/legal-content/EN/TXT/?qid=1528108752221&uri=CELEX:52017DC0236
5. IAEA Safety Guides. Classification of Radioactive Waste. Safety Series No. 111-G-1.1. 1994
6. IAEA Safety Standards. Classification of Radioactive Waste. General Safety Guide No. GSG-1. 2009

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Appendix A

IAEA 111-G-1.1 classification scheme		
Waste class	**Characteristics**	**Disposal**
1. Exempt waste (EW)	Activity level at or below clearance levels given in IAEA III-G-1.5, which are based on an annual dose to the public of < 0.01 mSv	No restrictions
2. Low and intermediate level waste (LILW)	Activity levels > clearance levels and thermal power < 2 kW/m^3;	
2.1. Short-lived waste (LILW-SL)	Restricted long-lived radionuclide concentrations (LL α emitting radionuclides < 4 000 Bq/g in individual waste packages and < 400 Bq/g per waste package)	Near-surface or geological disposal facility
2.2. Long-lived waste (LILW-LL)	Long-lived radionuclide concentrations > LILW-SL class	Geological disposal facility
3. High level waste (HLW)	Long-lived radionuclide concentrations > 2.1 and thermal power > 2 kW/m^3	Geological disposal facility

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IAEA GSG-1 classification scheme		
Waste class	**Characteristics**	**Disposal**
Exempt waste (EW)	Meets criteria for clearance (IAEA RS-G-1.7), based on an effective dose to individual Waste Package < 10 μSv for artificial radionuclides and natural concentration for natural radionuclides Exemption or exclusion from regulatory control (EW is no radioactive waste after clearance)	
Very short-lived waste (VSLW)	Storage for a few years and then cleared $T_{1/2}$ < 100 days	
Very low level waste (VLLW)	No need for high level of containment and isolation Proposed: < 100 * EW levels (for SL nuclides)	Near-surface landfill facilities - limited regulatory control
Low level waste (LLW)	Limited amounts of LL isotopes; Robust isolation and containment for up to a few hundred years; LL α-nuclides: < 400 Bq/g average per waste package and < 4 000 Bq/g for individual waste packages LL γ- and β-radionuclides - up to tens of kBq/g; Contact dose rate < 2 mSv/h for.	Engineered near-surface facilities (surface to - 30 m)
Intermediate level waste (ILW)	Greater degree of containment and isolation; No or limited provision for heat dissipation during storage and disposal; LL radionuclides, especially α-radionuclides; Heat < 2 - 20 kW/m^3 (total activity 10^4-10^6 TBq/m^3 or 10^9-10^{11} Bq/g).	Disposal at a depth of tens to a few hundred meters
High level waste (HLW)	Significant heat generation or large amounts of LL radionuclides	Disposal in deep, stable geological formations, several hundred meters below surface (after a few decades of cooling)

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NEA report 7323 gives a matrix to transpose the IAEA 111-G-1.1 classification to IAEA GSG-1, as well as a translation scheme of national RW inventories into the IAEA GSG-1 scheme.

Note: NEA 7323 states that there is a general consensus on the need to develop a method to transpose national classifications to a common scheme, but that this transposal will have no influence on the countries' existing schemes.

Matrix of transposition: IAEA 111-G-1.1 to IAEA GSG-1

	LILW-SL	**LILW-LL**	**HLW**
VLLW	X%	0%	0%
LLW	(100-X)%	0%	0%
ILW	0%	100%	0%
HLW	0%	0%	100%

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Translation of national RW inventory into GSG-1 scheme

	Minimal disposal	**Lower limit**	**Upper limit**
HLW	Geological repository	Total activity > 10^8 Bq/g. Heat generation > 2 kW/m^3.	
ILW	Intermediate depth repository	Long-lived alpha emitters > 4 000 Bq/g for single package (maximum) or > 400 Bq/g average over packages.	Total activity < 10^8 to 10^9 Bq/g. Heat generation < 2 kW/m^3.
LLW	Near-surface	> 100 times exemption levels (if no alpha activity present: > 100 Bq/g average activity may be used).	Long-lived alpha emitters < 4 000 Bq/g for single package (maximum) or < 400 Bq/g average over packages.
VLLW	Landfill		Total activity < 100 Bq/g.

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