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1 Executive Summary

Introduction

As the number of nuclear facilities being decommissioned in Europe increases sharply there is a need for safe and reliable methods to restore these sites and dispose of radioactive waste. The treatment and subsequent storage of radioactive material is expensive, but the financial burden can be reduced with accurate measurements and guidelines for decision-making on 'safe' disposal and contamination levels. Reliable measurements and strict criteria are important for ensuring public trust in the decommissioning process and the wider nuclear industry. Measurements of large quantities of nuclear waste also need to be fast and precise to help categorise and dispose of it safely, and in a cost-effective manner. There is also a requirement for ongoing monitoring of radioactive waste. This project addressed the needs of the decommissioning process by the development and implementation of new radioactivity measurement techniques, instruments, calibration standards and reference materials. It also ensured that knowledge was transferred to regulators, the nuclear industry, instrument makers and standardisation bodies, thus supporting the 'safe' decommissioning of European nuclear facilities.

The Problem

The world faces a major challenge of great urgency: the enormous costs of the safe decommissioning of many outdated nuclear facilities. Nuclear decommissioning covers many activities from shutdown to the environmental restoration of the site, and monitoring of the disposed radioactive waste. More than 200 power reactors are being decommissioned or will be in some phase of the process by the year 2025. The decommissioning process relies on the ability to characterise and segregate all waste material into various radioactive waste categories precisely and rapidly. This allows safe, or free, release of cleared waste into the environment; recycling or volume reduction of contaminated waste; further treatment of contaminated waste or long-term storage and monitoring in repositories. The decommissioning process is not only very costly but is currently being carried out in the context of low public confidence in the safety of nuclear technology. Therefore the restoration of public trust relies on adoption of standardised and traceable methods for measuring radioactivity, and the demonstration of compliance with the legislation

There is an urgent need to achieve a significant reduction in the enormous decommissioning costs by development and implementation of accurate, safe and economical novel measurement techniques.

The Solution

The project delivered research addressing many aspects of the decommissioning process, including the characterisation of radioactive waste materials present on the decommissioning site, pre-selection prior to free release or repository acceptance measurement, free release measurement, the measurement of radioactive waste packages thermal power prior to the repository storage, and the monitoring of radioactive stored wastes and repositories in the very long term.

The project developed new traceable measurement methods, instruments, standard sources and reference materials to improve accuracy, efficiency, reliability, throughput, modularity and mobility of measurements to improve integrity and cost effectiveness of the clearance and disposal processes and promoting safety.

We addressed the needs by development of the following novel or improved solutions:

- Alpha and gamma radiation mapping by using automated remote detection measurement devices.
- Statistically valid sampling and material characterisation strategies.
- Rapid and semi-automated radiochemical analysis for less hazardous destructive characterisation.
- Pre-selection and free release high throughput waste measurement facilities, based on standardised concept characterised by unique eco-friendly lead-free shielding,
- *In situ* and on line monitoring of radioactive gas emissions from nuclear facilities.
- Very long term temperature and integrity monitoring of radioactive waste repositories.
- Traceable and reliable calibrations and tests of the measurement systems with small uncertainties.

Impact

Improving the mapping of activation and contamination inside nuclear facilities enables worker exposure to be minimised and ensures a more effective selection of sampling points before dismantling. New sampling strategies together with rapid radiochemical analysis enables decision makers and operators to simplify the planning of the decommissioning work. The implementation of novel measurement facilities for preselection and free release enables decommissioning operators and radiation protection regulators to optimise the waste management process and leads to lower quantity of material incorrectly sentenced to repositories due

to measurement inaccuracy and thereby it will minimise disposal costs. The quantity of incorrectly released or stored material will be lowered and possible trade disputes will be less likely at national and international level. The implementation of new measurement devices for very long term monitoring of radioactive waste repositories, especially toxic and flammable gases, the thermal power of waste packages, the temperature inside the repository and the physical integrity of the repository, enables repository designers, operators and radiation protection regulators to monitor the safe operation in repositories for many years and it will improve our understanding of waste degradation and therefore the dose implications for the population.

Manufacturers of measurement devices NUVIA, CANBERRA, LabLogic Systems Ltd., Air Liquide and French National Radioactive Waste Management Agency (ANDRA) and power company Electricity of France (EDF) are utilising our research outputs and other companies as SPEX SamplePrep, Agilent Technologies and Triskem International SAS are interested in our radiochemical research outputs.

2 Project context, rationale and objectives

2.1 Context and rationale

This project addresses one of the most significant environmental challenges facing EU member states: ensuring the safe disposal of radioactive waste from decommissioning nuclear sites.

The first generation of nuclear power plants and reprocessing facilities are coming to the end of their working lives. 91 power plants are being decommissioned in the EU; most of the remaining 129 reactors plus fuel cycle facilities will also be in decommissioning by 2030.

The aim of the decommissioning process is to clear the site, while minimising the risk to the public and the environment from the hazardous waste arising. The cost of decommissioning and waste management in the EU is estimated to be in excess of 150 billion Euro.

Strong support for harmonised and consistent methodologies for managing radioactive waste is evident across the EU. There is a clear desire from European citizens that requires the EU to play an active role in managing radioactive waste.

The key to safe and cost-effective disposal of the waste is accurate characterisation – determining the physical, chemical, and radiological characteristics of the material and quantifying the radioactivity content. This enables nuclear site operators to plan the demolition process, assign the waste to the most cost effective disposal route and then to monitor that the waste is being stored safely. The metrological challenge is the characterisation of big quantities of hazardous materials for disposal that are not well known. Although there was significant progress towards developing new techniques to address this issue in EMRP JRP ENV09 MetroRWM; this project focussed on new traceable measurement methods, techniques, instruments, standards and reference materials to improve accuracy, efficiency, reliability, throughput, modularity and mobility of measurements to improve integrity and cost effectiveness of the clearance and disposal processes and promoting safety. Special attention was paid to the uptake and the commercialisation of the project outputs by the user community.

Regulatory bodies and international organisations have carried out detailed studies of technical needs in the field. The common themes that were raised are: (1) improvements in capability, (2) harmonisation and quality assurance, and (3) sharing knowledge. The improvements in capability that are required include rapid, on site measurements, improvement of the accuracy and the traceability of measurements of waste packages.

These needs are reflected in number of EU Council directives and recommendations:

The directive 2011/70/EURATOM, establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste, provides binding legal force to the main internationally endorsed principles and requirements in this field and states that radioactive waste shall be kept to the minimum by means of decommissioning practices, including the recycling and reuse of materials. Member states shall ensure that the national framework is improved where appropriate, taking into account operating experience and the development of relevant technology and research. The Commission shall encourage co-operation between the Member States in common areas of research and technological development. Minimisation of radioactive waste is ensured by accurate standardised traceable measurement methods developed in the project in cooperation of several EU states.

The directive 2013/59/EURATOM requires higher sensitivity measurements for some radionuclides, emphasising the need for improved capability. The project addresses this need by novel measurement instruments with improved parameters.

Commission Recommendation 2006/851/EURATOM on the management of financial resources for the decommissioning of nuclear installations, spent fuel and radioactive waste states that all nuclear installations should be decommissioned after permanent shutdown and the management of waste should be properly addressed. The project addresses this need by development of methods allowing accurate measurement of huge amount of waste materials typical for decommissioning.

Commission Communication COM(2005)666, A thematic strategy on the prevention and recycling of waste, states in Annex I, Article 4 Waste prevention that the potential for waste prevention depends on adopted best practice in reducing waste. Significant reduction of radioactive waste should be one of main project outcomes.

IAEA "Joint convention on the safety of spent fuel management and on the safety of radioactive waste management" and several IAEA safety standards (NS-G-2.7, GSR Part 5, SSR-5, GS-G-3.3, WS-G-2.5, TECDOC-1537) were addressed by the project: The research topic was in considerable conformity with ISO TC 85 (SC5) and IEC TC45 business plans. The results of the project have been disseminated to international and European standardisation bodies and working groups including IAEA, EC, and nuclear agencies.

2.2 Objectives

The overall objective of this project was to establish a measurement infrastructure ready for acceptance and implementation by the nuclear industry.

The aim was to create innovative metrological solutions, improved methods and measurement facilities that will have a significant technological and financial impact on the growing demands of the European decommissioning industry.

Another important aim was to minimise the environmental burden by providing means for improved sentencing and reposition of waste resulting from decommissioning and build thus public trust in nuclear technologies.

Therefore, the project focused on the following scientific and technical objectives:

- **Development of methods for the radionuclide characterisation of different types of materials present on the site being decommissioned.**

  This included improvement of remote mapping of contamination inside nuclear facilities, determination of statistically valid sampling methods and semi-automated and rapid radiochemical analysis.
- **Development of traceable methods for the pre-selection of waste materials prior to measurement for repository acceptance or possible free release.**

  This included segregation of wastes for potential free release in the environment or storage in repository, development of automated facility design, measurement and calibration procedures, and software.
- **Development and implementation of free release measurement facility (FRMF) on a decommissioning site.**

  This included implementation and testing of large scale industrial prototype eco-friendly shielded FRMF, measurement software improvement, scanning of wastes with heterogeneous density, passive neutron counting.
- **Development of methods for monitoring in radioactive waste repositories.**

  This included construction and field trial of novel gas monitoring systems including prototype of ^{14}C monitoring with mid-infrared spectroscopy, development of sensors for repository sites integrity monitoring, construction of acoustic thermometry testing facility for temperature monitoring, design of calorimeter for direct measurement of radioactive waste packages thermal power.
- **Development of reference materials and standard sources for calibration, validation and testing of devices, instruments and procedures developed in the above objectives.**

  This included development of reference materials and standard sources for pre-selection of materials and for free release measurement, radiochemical analysis, gas monitors, and surface contamination monitors.

2.3 Project structure

Project organisational structure is shown in Figure 1.

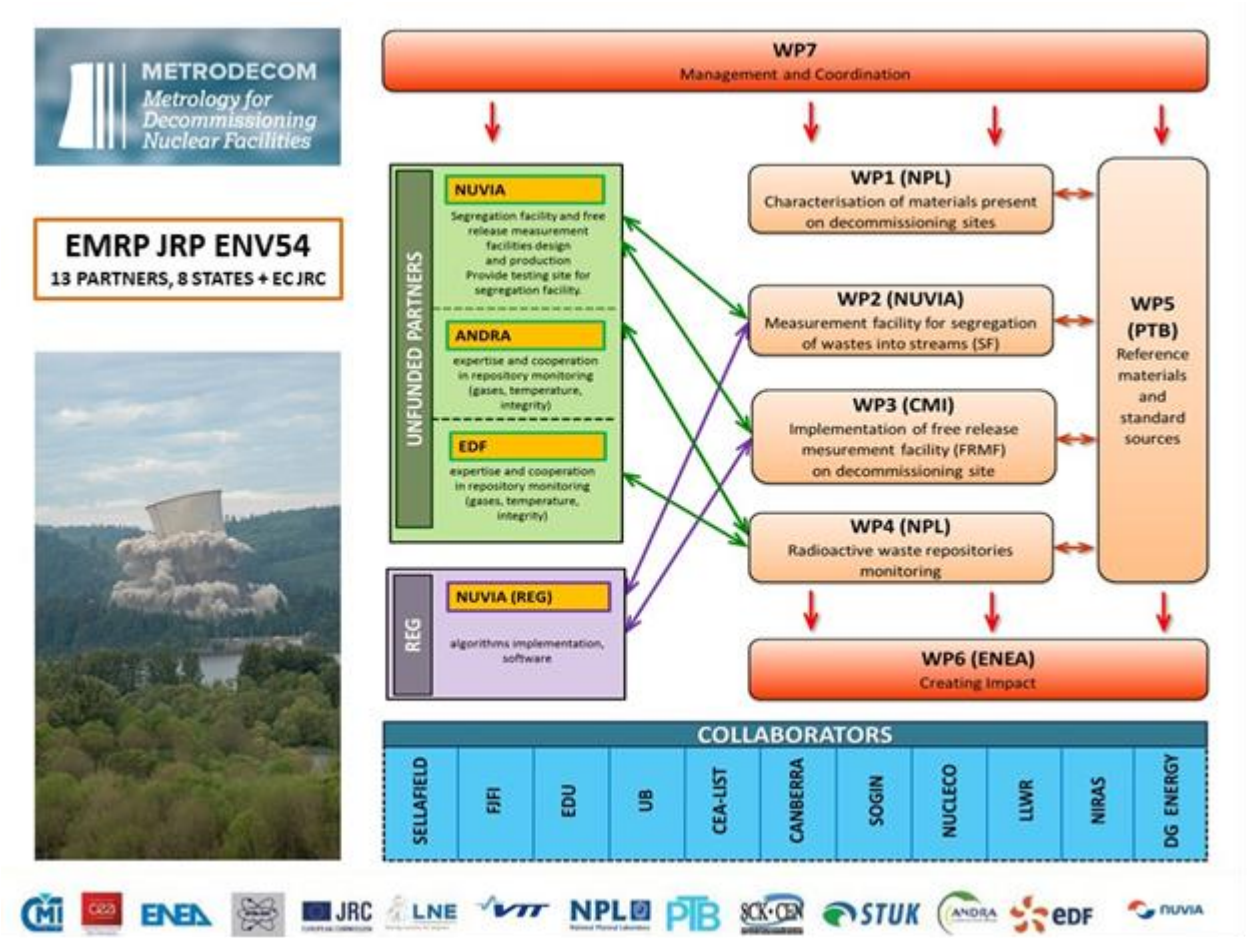


Figure 1. Organisational structure of MetroDecom project

3 Research results

3.1 Development of methods for the radionuclide characterisation of different types of materials present on the site being decommissioned

The aim of this objective was to improve the characterisation of materials and items at decommissioning sites, prior to disposal. A variety of techniques was investigated to achieve this goal, and these addressed important aspects of characterisation including:

- Improve the mapping of activation and contamination inside nuclear facilities by the development of novel measurement techniques (e.g. better targeted surface contamination determination, localising of hot-spots using stand-off alpha-particle and gamma-ray detectors, linked to in-situ gamma-spectrometry, determination of depth distribution of radionuclides and remote alpha detection methods).
- Determine statistically valid sampling methods, based on Bayesian analysis for representative samples for radiochemical analyses, with the ability to feed analytical data back into the Bayesian model, thus improving such sampling and analysis plan, with a direct improvement in efficiency resulting from this approach.
- Develop rapid radiochemical analysis procedures building on and extending the outcomes / knowledge delivered by EMRP JRP ENV09, as well as extending the range of radionuclides determined, that are capable for development of in situ analysis of materials.
- The radiochemical and instrumental measurements carried out in this work package will enable the quality of scale factors to be improved for selected radionuclides, leading to better planning for decommissioning activities, better sentencing of waste, and also a reduction in the number of

radiochemical analyses that are required. As such the quality and reliability of decommissioning is improved with a concomitant reduction in costs and time.

These outcomes provide feedback for future decommissioning work, so that each iteration of the assessment-sampling-analysis/measurement-scaling factors cycle enables the next iteration to be carried out more effectively with the ultimate aim of continually improving the process in order to better utilise the resources available to carry out such decommissioning work and to more accurately sentence the waste arising from decommissioning of nuclear facilities.

3.1.1 Mapping of activation and contamination inside nuclear facilities

Alpha radiation results

The background-free solar blind region of the radioluminescence spectra in air and other gases used in the nuclear industry was determined. Alpha radiation causes radioluminescence in these gases and the alpha radiation can thus be detected optically from a stand-off distance. Traditionally alpha radiation is detected from at most a few centimetres from the source or surface. Optical methods based on radioluminescence have been limited by the high UV background caused by, for example, our sun. Studying the solar blind spectra and the response of different photocathodes in this background-free region paves the way for optical detector designs that can be used straightforwardly on site.

An Am-241 alpha source was used for the experiments. The source was placed in a gas tight stainless steel casing with a continuous gas flow. The spectra was measured with a photomultiplier tube through a monochromator setup. Different photocathodes and optical filters were tested. The gases used were air, nitrogen and industrial nitrogen and argon.

The measured spectra confirm that there are radioluminescence emissions in air in the solar blind region (See Figure 2). Also industrial argon and nitrogen (due to NO formation) have strong emissions in the solar blind region. The photomultiplier comparison is useful for optimising the optical detector system. Solar blind photomultipliers have not been widely used for radioluminescence measurements, however, they can prove to be useful when the background light cannot be eliminated.

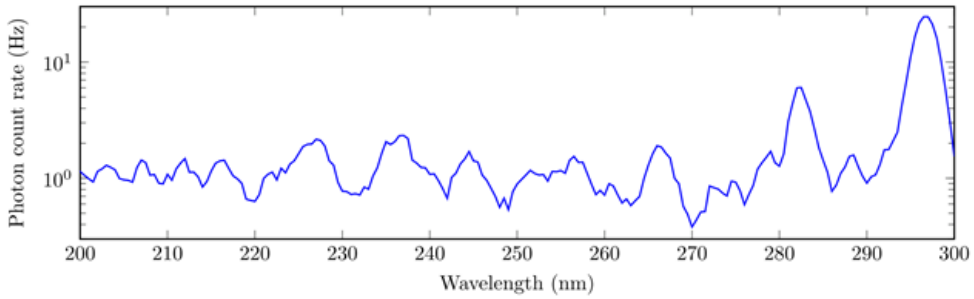


Figure 2. Measured radioluminescence spectrum of air in solar blind region.

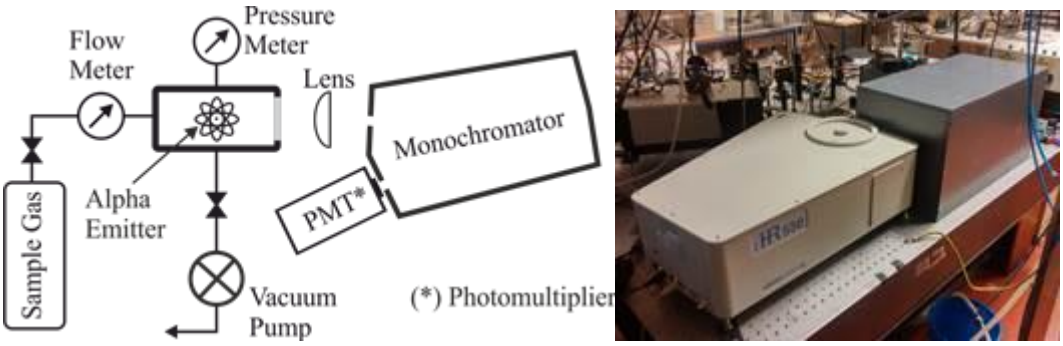


Figure 3. a) Measurement of radioluminescence spectrum of alpha particles in gases. The alpha source is enclosed to a gas tight chamber which can be flushed with the sample. A mass flow controller (5850S, Brooks Instrument) was used as flow meter. The vacuum pump (BHV-5, Leybold) facilitates gas change. A channel photomultiplier counts the spectrally separated photons.

b) Photograph of the setup. The alpha source and optics are enclosed in the grey box.

Gamma contamination mapping

A method for measuring gamma contamination using the gamma camera GAMPIX was developed. A GAMPIX is a second generation gamma camera including the Timepix pixelated detector with a coded mask in a unique portable system. It is a plug-and-play system, easy to deploy and to use.

Regarding the measurements for gamma emitters:

- Calibration sources, point and extended sources, were made by CEA and allowed to test the gamma camera GAMPIX in checking the detector response as a function of the radiation energy
- Tests were carried on with 241Am (59.5 keV), 57Co (122.1 keV ; 136.5 keV), 137Cs (661.7 keV), 60Co (1173.2 keV ; 1332.5 keV) covering the whole energy range
- Extended sources (surface of 15.5 cm x 7 cm) were made successfully with a satisfactory homogeneity of 13%. The activity of the sources were: 420 kBq for 241Am, 850 kBq and 1900 kBq for 57Co
- The measurements carried on with the GAMPIX system with point and extended sources proved that there is a very good linearity between the activity of the sources and the signal, that the reconstruction of the dose is perfectly done. That was proved using single and multiple sources in front of the field of the gamma camera
- Measurements were carried on successfully in laboratory and in-situ conditions (localisation of radioactive source in contaminated tank)

First measurements in laboratory conditions with point sources showed that the gamma camera GAMPIX has a very good response in terms of linearity. Indeed, the intensity of the hot spots in very well correlated with the activity of the sources (example of point source 57Co, figure 4 and for spread sources, figure 5). The measurements carried on with point and extended sources showed that the gamma camera reconstructed the emitted dose rate with an accurate precision.

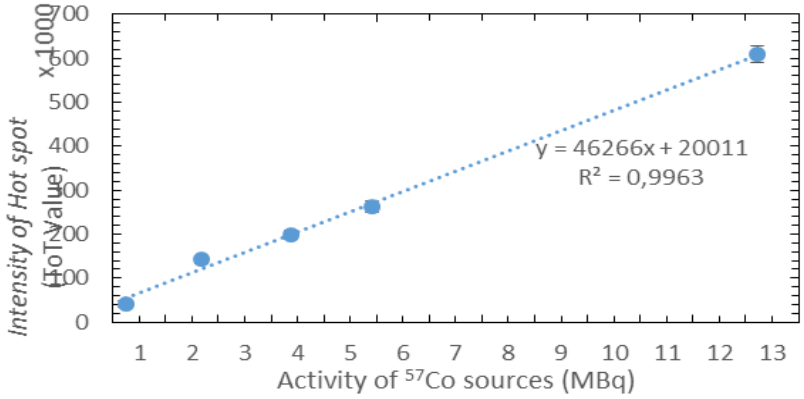


Figure 4. Hot spot intensity as a function of 57Co activities.

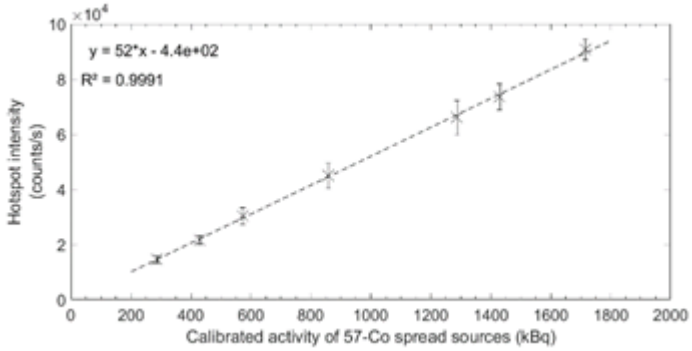


Figure 5. Evolution of hot spot intensity of 57Co spread source as a function of considered activity

Additional measurements were carried out on spread sources with different shapes with a total activity of 1.714 MBq: one measurement on a spread source with a bigger surface (10x14 cm²), and two measurements with the shape of a bent pipe from the side view and the frontal view corresponding to a more realistic situation of decommissioning (see Figure 6). Table 1 summarises the results obtained for these measurements; the hot spots intensities are similar for the different shapes. However, for a given activity, increasing the surface leads to higher relative uncertainties. This phenomenon is caused by a reduction of the surface activity density with the surface increase.

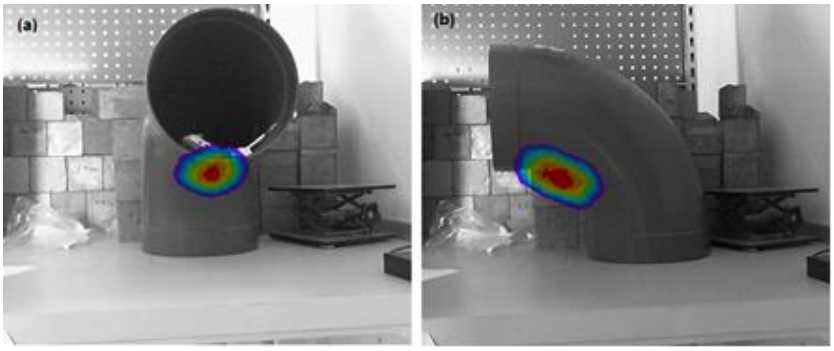


Figure 6. Superimposition of visible image and gamma image. Measurement on 57Co spread sources with an activity of 1.714 MBq placed in a bent pipe, frontal view (a) and side view (b).

	Hots spot intensity (hits/s)	Relative uncertainty (k=2)
Surface 5x7 cm²	91261	4.5 %
Surface 10x14 cm²	89769	11.9 %
Bent pipe Front view	88221	13.1 %
Bent pipe Side view	91506	14.5 %

Table 1: Comparison of the hot spot intensities for different shape of 57Co spread sources (1.716 MBq)

The final measurements were carried on in two campaigns in in-situ conditions.

The first campaign consisted in on a contaminated flower box and contaminated vegetal, mostly contaminated by 235U radionuclides family. Figure 7 shows the localisation of the radioactive source in the flowerpot, however since the contamination is mainly caused by 235U radionuclides family the dose rate cannot be reconstructed since GAMPIX has not been calibrated with those radionuclides.

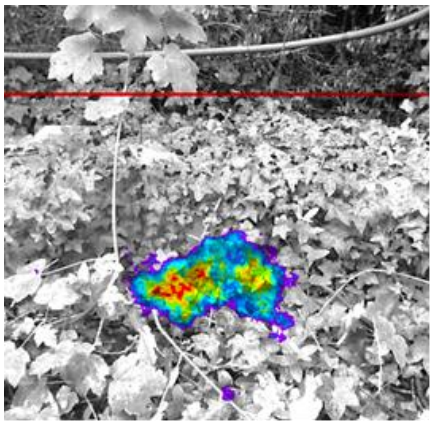


Figure 7. Decoded image from a contaminated flowerpot (235U family)

The second campaign was carried out on storage tank containing radioactive effluents from the CEA nuclear site. The tank was mostly contaminated by 137Cs, for this measurement the dose rate could be estimated. Figure 8 shows the decoded image obtained from this measurement. The estimated dose rate at contact is 7.56 mSv.h-1 which is the same order of magnitude of the real value.

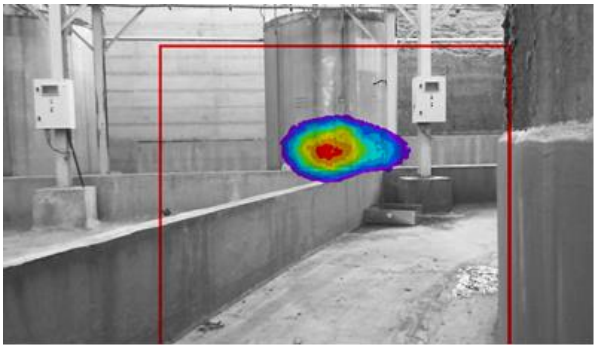


Figure 8. Localisation of a radioactive source in a contaminated fuel tank (mostly 137Cs family)

The response of the GAMPIX gamma camera to spread sources has been evaluated by CEA. The gamma camera can localise and quantify the dose rate of spread sources and its response is linear with the considered activities. The measurements on the bent pipe prove that the reconstructed hot spots intensities do not depend on the perspective, neither on the shape of the sources. Moreover in situ measurements prove the capability of GAMPIX gamma camera to reconstruct and localise radioactive source.

Mapping of the contamination in a reactor building

A method to determine the contamination depth in building structures prior to decontamination was developed and tested. The non-destructive method is based on the peak-to-peak method and uses geostatistical modelling for the results of in situ gamma-ray spectroscopy measurements. The methodology used prior to start the project was based on high resolution gamma spectrometry measurements obtained with a 100% surface coverage. Within this project, we further optimised the method by a geostatistical integration of a limited amount of primary data (in-situ gamma spectrometry measurements) and two extensive but straightforward secondary data sets (dose rate and surface beta measurements). Correlations between primary data and secondary data (Cs-137 concentration and dose rate and 32/661.6 keV ratio and beta surface contamination/dose rate ratio) were used to produce 3D contamination distribution maps including uncertainties. The method was tested on the floor of the liquid waste tanks room (75 m2) of the Belgian Reactor 3 D&D project. The uncertainties on the contamination depths were adequate taking the limitations of the decontamination techniques into account. The test case clearly showed the possibility to integrate various types of data with different spacial support and quality (e.g. dose rate, surface contamination, gamma spectroscopy) making use of advanced geostatistical analysis in order to optimise the global uncertainty on the final result as well as the costs of the pre-decontamination characterisation process (Figure 9).

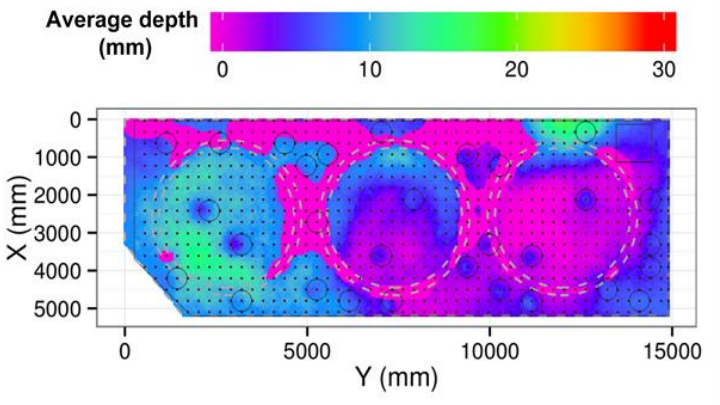


Figure 9. Contamination depth distribution map using Sequential Geostatistical Resampling of primary data (30 black circles indicating the in situ gamma spectroscopy measurements) and two secondary data sets (978 black dots indicating the dose rate and surface beta measurements). Colour bar is in mm.

Beta contamination measurements

The efficiency of contamination sources for beta radiation is the most important quantity used for evaluating the surface beta contamination by means of direct measurements using beta contamination monitors. Thus, the standard ISO 7503 requires this quantity in order to determine the activity instrument response. To address this a direct method of beta contamination measurement beyond the state of art was developed by evaluating the uncertainty of the activity per unit area and providing a new method of measurement.

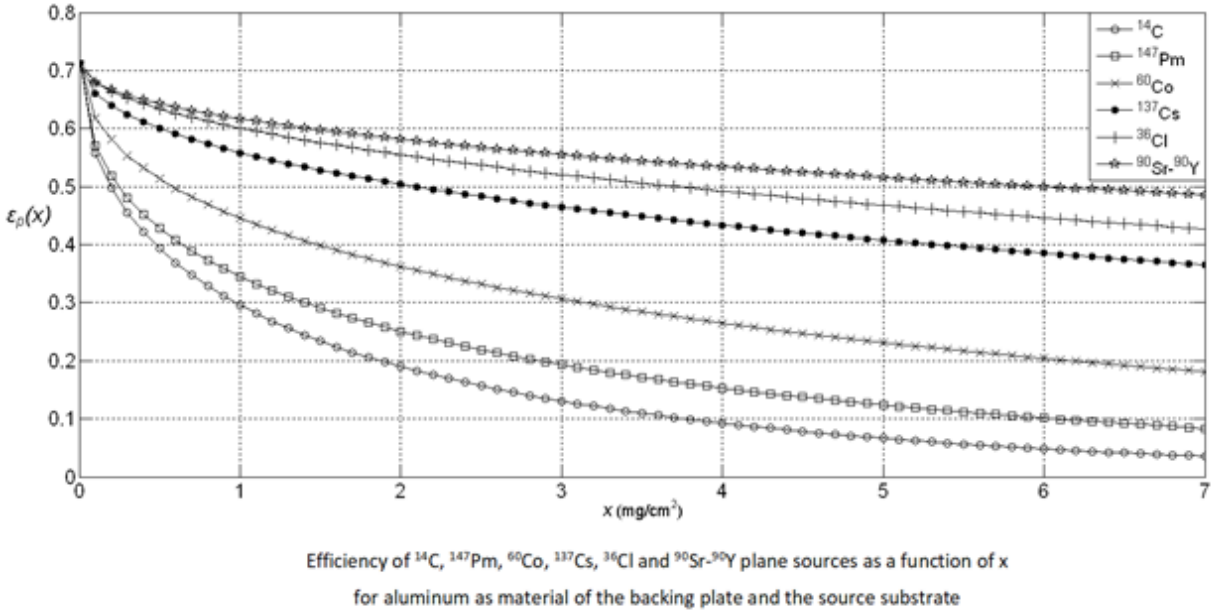


Figure 10. Efficiency of 14C,147Pm, 60Co,137Cs, 36Cl and 90Sr-90Y plane sources as a function of x for aluminium as material of the backing plate and the source substrate.

Using theoretical results, a novel method for the activity measurement of large-area beta reference sources was developed. (D. Stanga, P. DeFelice, J. Keightley, M. Capogni, M. R. Ioan, 2016. A novel method for the activity measurement of large-area beta reference sources, Appl. Radiat. Isot. 109, 358-362). By means of this method, the activity of large-area reference sources emitting beta radiation (end-point energy greater than 150 keV) can be measured with standard uncertainties smaller than 5 %.

By computing the efficiency of contamination sources, a new approaches in evaluating the uncertainty of measurement, according to ISO 7503, was developed (D. Stanga, P. De Felice, M. Capogni, 2017, A new approach in evaluating the surface beta contamination using the direct method of measurement, Applied Radiation and Isotopes 129, 135–141).

A new method for the direct measurement of the surface beta contamination was developed based on the efficiency transfer method (D. Stanga, P. De Felice, M. Capogni, Efficiency Transfer Method Applied to Surface Beta Contamination Measurements, Applied Radiation and Isotopes, https://doi.org/10.1016/j.apradiso.2017.08.009). The results show that the current practice in calibrating the beta contamination monitors must be changed by using the calibration on the basis of the activity of large-area reference sources.

In order to validate the method, direct measurements of emission rate of eight rectangular sources (12 cm × 12 cm × 0.3 cm with the size of the active area equal to 10 cm × 10 cm), constructed to mimic contamination sources, were carried out within a bilateral comparison between ENEA and IFIN-HH.

3.1.2 Sampling

Sampling is a complex matter and the aim here was to provide guidance that could minimise the numbers of samples taken, without introducing additional risk in terms of false negatives (that may result in the unplanned release of radioactive material) or false positives (which unnecessarily use up space in radioactive waste repositories.

Sampling strategies were reviewed and summarised that included approaches based on frequentist statistics, Bayesian statistics and Pierre Gy's sampling.

The outcomes from this work were the provision of general guidance based on this techniques; specific instructions or procedures were outside the scope of this project. Generalisation is not possible, due to the diversity of sites, nuclides, matrices and national priorities, but the interested user should be able to use the information presented to plan sampling campaigns in a rational and reasonable manner.

3.1.3 Rapid radiochemical procedures

The destructive analysis of nuclear decommissioning matrices is an important area as some of the radionuclides of interest are alpha and beta particle emitting radionuclides which can only be measured after radiochemical separation. Moreover. Some of these cannot be measured with radiometric techniques directly due to the decay mode and energy of the radionuclide of interest, and the interference from other radionuclides. For this reason, radioanalytical methods were developed for some difficult to measure radionuclides and some of the common decommissioning matrices selected for this project. The selected radionuclides were: Pu and U isotopes, Sm-151, Sr-90, Zr-93, in concrete, graphite and steel. The approach was to develop the separation using aqueous samples spiked with the radionuclides of interest or the stable elements to track the recovery of the elements of interest either at the end of the measurement by radiometric techniques or mass spectrometry techniques. The results of the aqueous samples for all of the nuclides of interest in aqueous solutions reached chemical recoveries of > 60% and quite consistent.

The next stage was to design strategies to dissolve solid samples and the required sample treatment and preparation to be able to follow the designed radiochemical separations. The strategies for dissolution of the different matrices were based on the different chemical properties: fusion with lithium borate followed by silicate polymerisation was the treatment for concrete samples, while steel was dissolved with aqua regia and graphite was either ashed or fussed with lithium borate or dissolved in pressurised microwave vessels.

Lastly the radiochemical separations where applied to the matrices and adjusted as necessary which impact in the chemical recoveries, however the recoveries can be measured by ICP-MS or the use of radioactive tracers and allow to correct the measurements by this parameter. The effect of matrix effect on the chemical recoveries is a higher variability among replicates as well as a reduction in the recovery values that in some cases fall to under 40%.

The Sm-151 was also measured by an ICP-QQQ-MS for the first time, and the interference of stable Eu was removed successfully within a range.

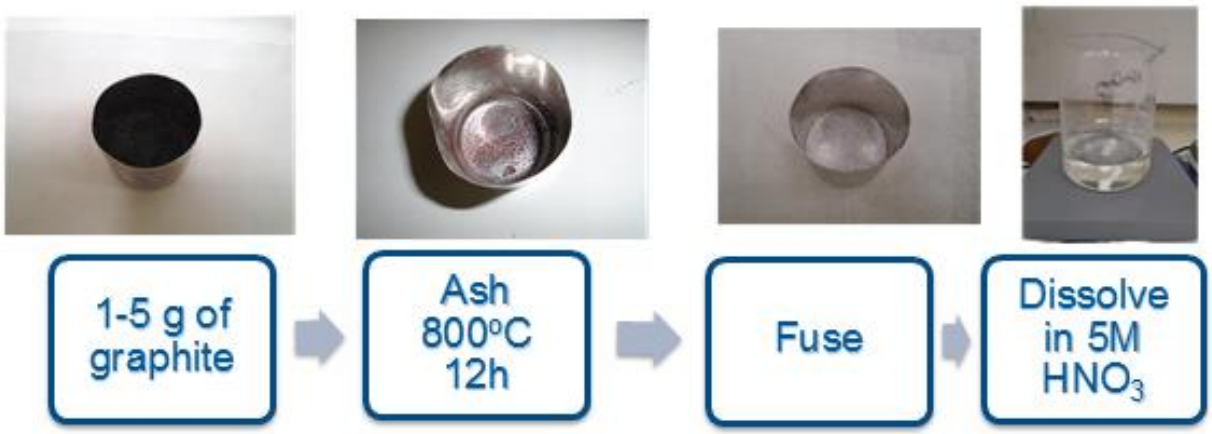


Figure 11. Graphite dissolution process by lithium borate fusion

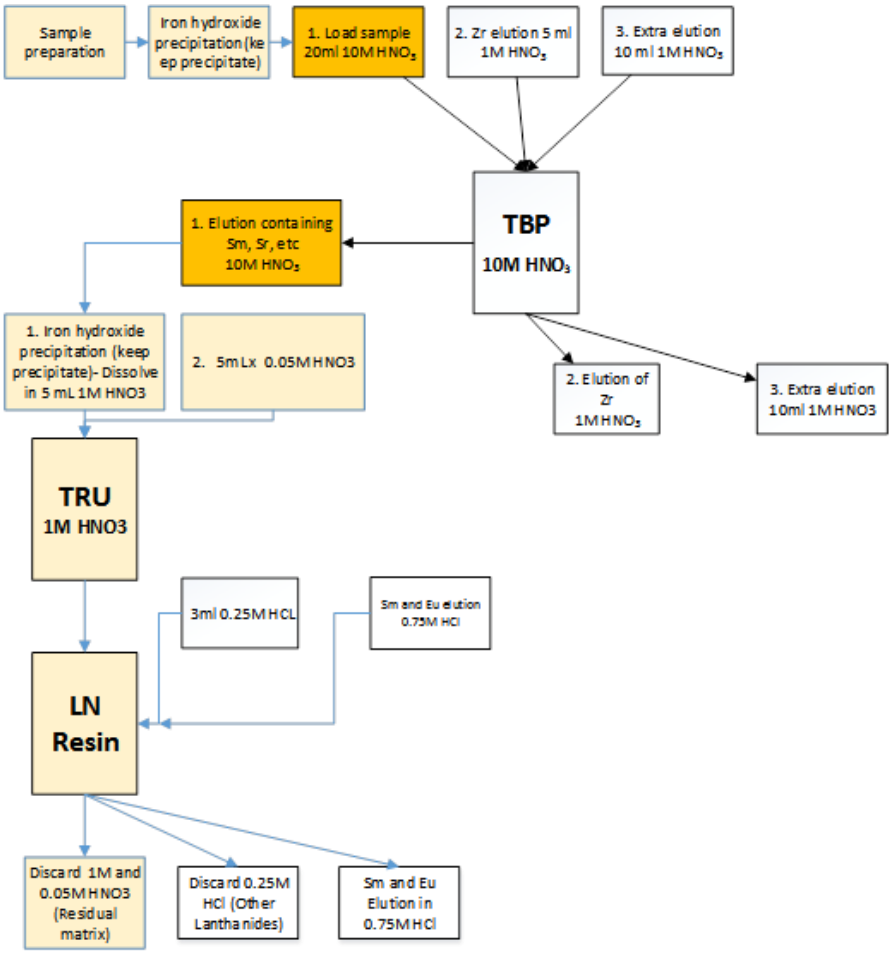


Figure 12. Radiochemical separation protocol for the separation of Zr-93 and Sm-151

3.1.4 Scaling factors

The work on scaling factors was carried out to attempt to bring together information needed to assess the utility and accuracy of existing data and to identify gaps in the knowledge base that exists worldwide. In particular, the diversity of facilities across the EU means that some areas are not well served by existing information that is necessarily driven by the predominance of PWR (and to a lesser extent BWR) reactors across the world.

Existing databases were reviewed and summarised, although many difficulties to access to the Scaling Factors databases (often with restricted access) in order to assess the current state of knowledge and to identify gaps where more information is needed. In addition, current work at existing facilities was also included in this analysis to provide examples of the day-to-day issues attendant on the use of scaling factors in decommissioning.

The outcomes from this work were the identification of gaps in the knowledge base that may impact situations encountered across the EU (particularly gas cooled, graphite moderated reactors) and 'less common' or 'difficult to measure' radionuclides that may present additional problems when long-term disposal of waste is considered. In addition guidance on the derivation, use and availability of information was reported.

3.2 Segregation measurement of solid wastes

One of the main challenges at the decommissioning of shut-down nuclear facilities is to segregate wastes for storage in a radioactive waste repository, or the potential release to the environment. The objective was that

the waste that could meet the clearance criteria wasn't sent to expensive radioactive waste repositories, and that only less than 5% of the material segregated for free release measurement is returned back to repository. This aim required accurate and traceable measurement of the waste packages and an appropriate setting regarding segregation parameters by the operator.

Within this project, a segregation measurement facility was developed and optimised. The facility was installed on the European Joint Research Centre (JRC) Ispra (Italy) decommissioning site for calibration and testing in real in-situ conditions. For the calibration, Monte Carlo calculations, reference materials and standard radionuclide sources were used. This facility is based on four plastic scintillating detectors and a low background concrete shielded measuring chamber. A complex software was developed for the segregation measurement including data acquisition, measured data evaluation, performance control, efficiency calibration, segregation criterion setting, results provision and archiving.

The segregation facility was installed in JRC in March 2015 by NUVIA CZ and the measurements were performed by JRC and CMI, with PTB support. For testing and calibration, the segregation facility was installed as a part of the free release measurement facility, however for routine industrial use the facility must be installed separately.

3.2.1 Segregation facility construction

The segregation facility consists of four plastic scintillating detectors made by NUVIA CZ, low-activity concrete shielding and material handling system. It is shown in the Figure 13.

Figure 13. Segregation facility as a part of free release measurement facility

Detectors

For accurate and effective measurement four plastic scintillating detectors of following dimensions: cross section 10 cm x 10 cm, length 120 cm were used. One detector was located under the measuring container, one above the measuring container, and two on the both sides. The upper detector could be vertically moved depending on the container height. This configuration allowed high efficiency measurements. The position of the detectors in the measuring chamber is shown in the Figure 14.

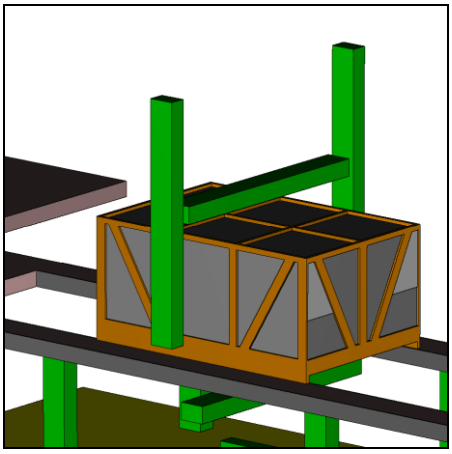


Figure 14. Position of scintillators around the container

Shielded measuring chamber

The measuring chamber was built with blocks made from low-activity concrete. This dry construction allowed shielding of the whole measuring chamber including detectors and measuring container and also fast building or dismounting of the facility within about 2 weeks. The concrete blocks are made from low-activity gravel, cement and water, or low-activity gravel and epoxy resin. The dimensions and the configuration of measuring chamber were optimised using Monte Carlo MCNP code (*D.B. Pelowitz et al., MCNPX™ 2.7.E Extensions - A General Monte Carlo N Particle Transport Code, Los Alamos National Laboratory, 2011 report LA-UR-11-01502*). The chamber's walls thickness was 40 cm, the floor thickness was 60 cm and the ceiling thickness was 40 cm. The measuring chamber drawing can be seen in Figure 15.

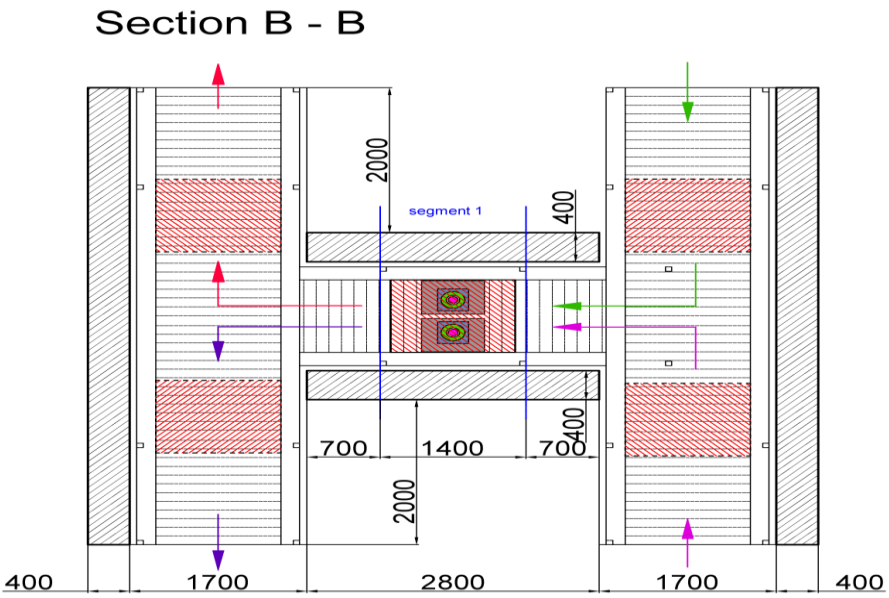


Figure 15. Ground plan of the measuring chamber

Measuring container and geometry

For these measurements, the preferred containers were IP1 or IP2 category (see IP2 container on Fig. 16) with a volume of 0.5 m^3 and dimensions 1.2 x 1.0 x 0.4 m, however the efficiency calibration can be done for different types of containers after end-user's requirements. Because of absorption of photons the container

was made from plastic plates reinforced with steel profiles which also ensures corrosive resistance and easy cleaning of the container.

Figure 16. IP2 container

3.2.2 Complex software for segregation measurement

The complex software for segregation measurements was developed by NUVIA in cooperation with CMI. The pulse count rates were acquired from each detector in four energy segments in the energy range 40 keV to 2 MeV, corrected for measuring efficiency, and the segregation coefficient calculated. The complex software also enabled the performance control, energy and efficiency calibration, and results display and archiving.

Performance control

The QA performance control was periodically assessed with a period stated by the operator. The stability of the detectors was measured using typically a point-like source with Eu-152 of activity ca. 0.1 to 1 MBq. This was done for each detector by determining the pulse count rates in 4 segments, of 256 channels each, in the energy range 40 keV to 2 MeV.

Calibration software

To calculate the segregation criterion, total efficiencies of each detector were determined in four segments in the energy range 40 keV to 2 MeV. The determination was described in chapter 3.2.3. The efficiency curves were entered into the calibration software and automatically assigned to measured spectra after the operator choose the type of measured material, its density, and a level of container filling.

Segregation criterion

The criterion for segregation of wastes is not given by nuclear regulators and each waste producer should prepare their own rules depending on the type of wastes and the possible radionuclides present. For the segregation facility testing, three basic types of waste materials were taken in account, metals, building materials and light materials, and it was also taking onto account the calculated density from the total weight and volume of measured material. The total pulse count rates were acquired in four parts of the spectra and an average measuring efficiency was calculated for each part depending on the type of material and its density. Then the total count rates were divided by the respective efficiency and the calculated values were summed. After dividing by the total weight, a segregation coefficient is obtained and compared with the segregation criterion that must not be exceeded. For materials measured within the project, segregation criterion was set to 1000.

3.2.3 Efficiency calibration

Because there is a lot of different waste materials are measured at the decommissioning sites, efficiency calibration cannot be performed for all of them using appropriate reference materials. Therefore total peak efficiency as a function of photon energy was calculated using Monte Carlo MCNP code. A precise model of the measuring geometry including detectors and measuring container was prepared and then validated using real and spiked reference materials representing three typical categories of wastes (metals, building

materials, light materials). These reference materials were standardised via inter-laboratory comparisons with participation of at least three national or designated metrology institutes and therefore traceability was assured as required by metrological legislation.

Total efficiency $\eta_t(E)$ was defined as a number of impulses registered in some energy range (here from 40 keV to 2 MeV) to a number of photons of energy E emitted by the source. The efficiency was calculated as $\eta_t(E) = T(E)/(A(X).Y(E,X).t)$, where T(E) was net number of pulses registered in the energy range (e.g. 40 keV to 2 MeV), A(X) was activity of radionuclide X in a reference material or standard source, Y(E,X) was the yield (number of photons of energy E emitted by radionuclide X per one decay), and t was live measuring time.

For the efficiency calibration, reference materials and standard sources developed in the previous EMRP project ENV09 'MetroRWM' (*https://www.euramet.org/research-innovation/emrp/emrp-calls-and-projects/emrp-call-2010-industry-and-environment/*) and new materials and sources developed with the multi-energy radionuclide Eu-152. Five sets of standard sources with radionuclides Co-57, Co-60, Cs-137, Eu-152 and Am-241of activity from 30 Bq to 1 MBq, and four reference materials (steel balls, steel tubes, gravel and clay balls) with Co-60, Ag-110m, Cs-137 and natural radionuclides were used. All calibration materials and sources were traceable to national standards of radionuclide activity.

3.2.4 Monte Carlo model of segregation facility

MCNP Monte Carlo model of the segregation facility was created including detectors and measuring container. For total efficiencies calculation, the container in the model was 'filled' by different homogeneous reference materials, or point-like standard sources. The plastic scintillation detector model is in Figure 17 and the segregation facility model in Figure 18.

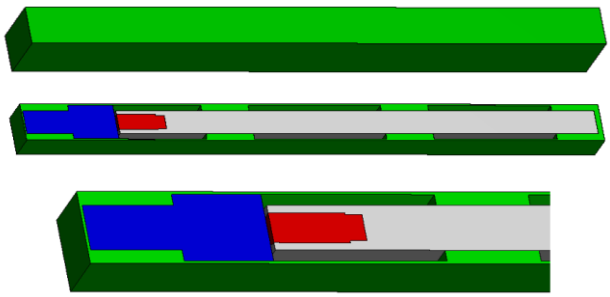


Figure 17. Plastic scintillation detector MCNP model

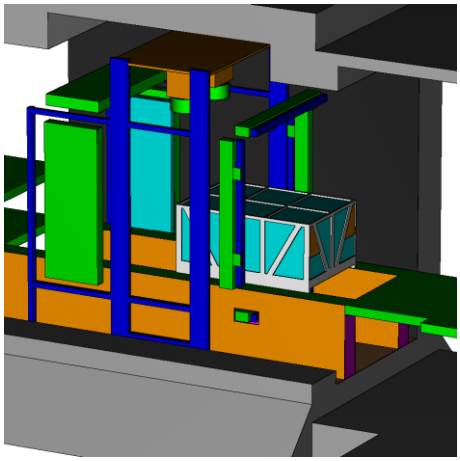


Figure 18. MCNP model of segregation facility

Model validation

Monte Carlo calculations of total efficiency η_{tcalc} were compared to experimental values η_{texp} obtained by measurement of reference materials and standard sources traceable to national standards of activity of radionuclides of several European metrology institutes. The model was considered validated, if the difference between measured and calculated values of total efficiency was less than 20%. Results for different materials are in Table 2. The calculated total efficiency as a function of photon energy for steel tubes is shown in Figure 19.

ref. material	η_{exp}	η_{calc}	$100x(\eta_{tcalc}-\eta_{texp})/\eta_{texp}$, %
steel balls	8.237×10^{-3}	6.754×10^{-3}	- 18
clay balls	17.85×10^{-3}	15.89×10^{-3}	- 11
steel tubes	13.89×10^{-3}	12.92×10^{-3}	- 7

Table 2 Comparison of measured and calculated total efficiencies

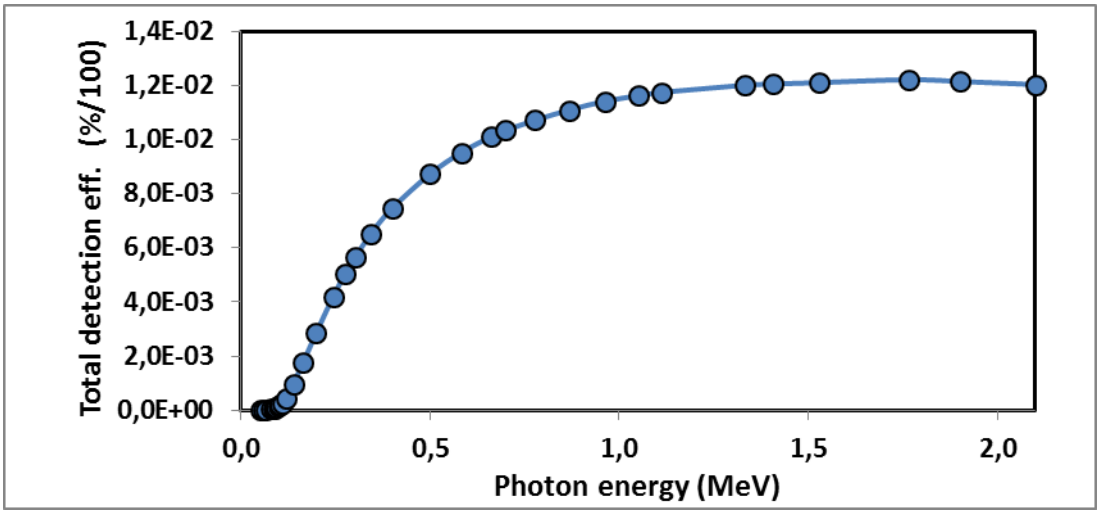


Figure 19. Calculated total efficiencies for steel tubes horizontally loaded

3.2.5 Measurement and calibration procedures

To ensure the quality of measurements as well as the derived results, JRC developed a complementary to the already existing Quality Control programme. Part of the instrument's Quality Control activities were the development of a set of experimental calibration procedures. The procedures were then applied for the performance of the measurements of the reference materials aiming at the assessment of the instrument measurement quality features.

The good practice guide was written. It focuses on the verification of the relevant performance of the segregation measurement using four plastic scintillating detectors. The measurement efficiencies were calculated for different types of waste materials using MCNP model. The validation of MC model was validated by experimental measurements using reference materials and standard sources in the MetroDecom container IP2. The standard segregation measurements were complemented with passive neutron counting in order to study the sensitivity of this method in particular for waste suspected of containing actinides. The prototyped software was tested during the background measurements, background checks and material measurements.

3.3 Free release measurement of solid wastes

For waste materials released in the environment from nuclear facilities, strict limits given by national nuclear regulators must be met for mass and surface activity of individual radionuclides. This requires very sensitive and radionuclide specific measurement based on gamma-ray spectrometry. Hard-to-measure radionuclides contribution must be included using scale factors based on laboratory measurements for different types of nuclear reactors. It is valuable to complement free release measurement with neutron detectors, and with the facility for waste scanning before the measurement. The Free Release Measurement Facility (FRMF) was

developed with optimised hardware, based on four germanium coaxial HPGe detectors and environmentally friendly shielding made from low-activity concrete. Complex software for free release measurement including data acquisition, gamma-ray spectra evaluation, performance control, efficiency calibration, homogeneity determination, scale factors calculation (*ISO 21238: 2007*), activity and minimum detectable activity calculation (*ISO 11929: 2010*), and results display and archiving was developed. Full-energy peak efficiency calibration was performed for different types of waste materials by Monte Carlo calculations based on MCNP model of FRMF and validated using traceable reference materials and standard radionuclide sources.

The FRMF facility was installed in JRC in March 2015 by NUVIA CZ and measurements were performed by JRC, CMI and NUVIA with NPL support.

3.3.1 FRMF construction

The facility consists of four germanium detectors, low-activity concrete shielding and material handling system. The shielded measuring chamber is described in Chapter 3.2.1 and shown in the Figure 13.

Detectors for gamma spectrometry

For accurate and effective measurement HPGe detectors IDM-200V (*IDM-200^{TM} HPGe Interchangeable Detector Module, Hardware Manual, ORTEC Part No. 932510*) were used with resolution FWHM ≈ 2.0 keV and relative efficiency about 50 % for photon energy of 1.33 MeV. The maintenance-free detectors had mechanical cooling and were constructed for industrial use. The detectors were accomplished by lead collimators. Two detectors were located under the measuring container and the other two above the measuring container. This configuration allowed measurement of mixtures of radionuclides with high efficiency. The detectors position in the measuring chamber is shown in Figures 20 and 21.

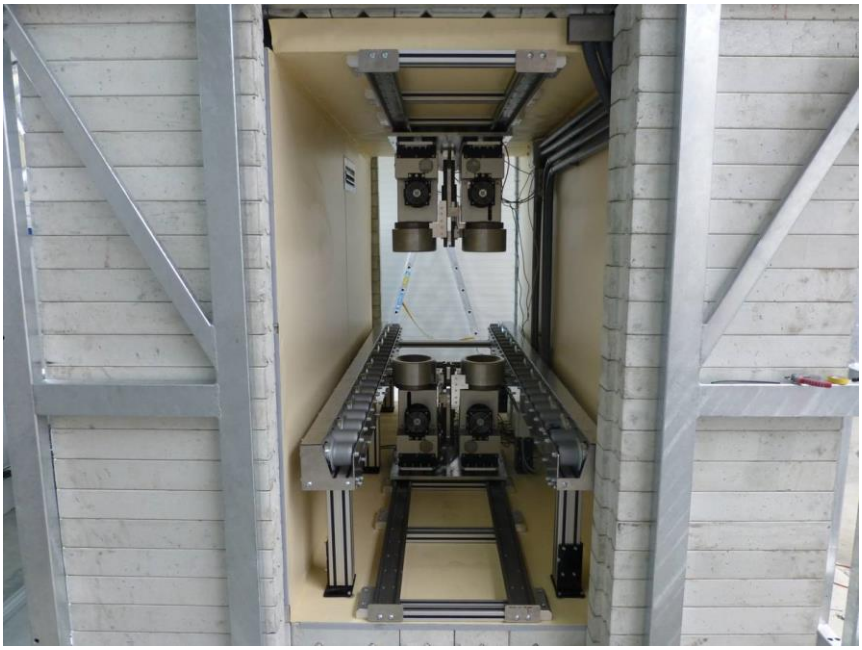


Figure 20. Position of HPGe detectors in the measuring chamber

Measuring container and geometry

The IP2 container (Figure 16) was measured in three different positions allowing measurement in twelve identical segments (3 positions and 4 detectors) and acquisition of twelve individual gamma-ray spectra. These spectra were individually evaluated, then the homogeneity was determined, or the hot-spots identified.

3.3.2 Scanning facility

Scanning was performed to refine self-absorption of gamma-rays in measured waste material. Each measured container was scanned using collimated radioactive source with Co-60 of nominal activity 95 MBq and detector NaI(Tl) 3” x 3”. The result was used to refine the measuring efficiency saved in the calibration software for the type of waste material that was measured. The scanning facility is in the Figure 20.

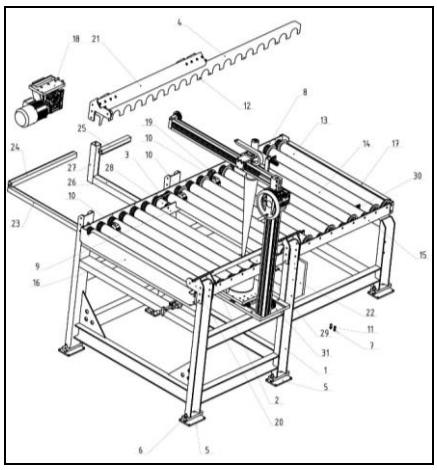


Figure 20. Scanning facility

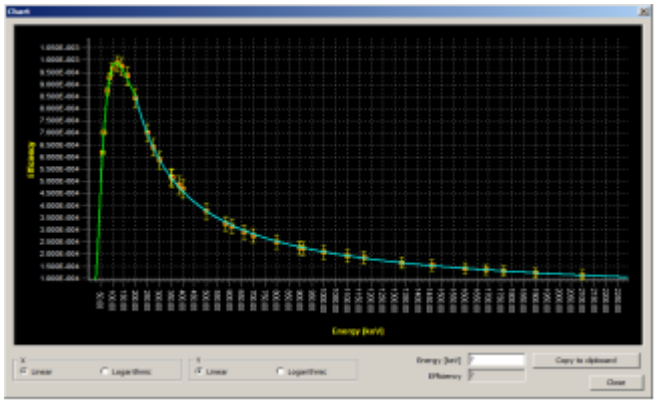


Figure 21. Full-energy peak efficiency as a function of photon energy

3.3.3 Complex software for free release measurement

Data acquisition and spectra evaluation

Each container was measured in three symmetrical positions and for each position individual spectra from four detectors were acquired, stored and evaluated. For homogeneously contaminated material, individual spectra were summed and activity was calculated from the summed spectrum. For hot-spots, activity was calculated from individual spectra. The measuring and evaluation software enables gamma-ray spectra acquisition and storage, full-energy peak identification in the spectrum and emitted photon energy determination, full-energy peak net area determination and activity and minimum detectable activity determination as described below.

Performance control for gamma spectrometry

The QA performance control was performed periodically with a period stated by the operator. The following parameters were investigated using typically a point-like source with Eu-152 of activity 0.1 to 1 MBq.

- Energy calibration was checked every day before the measurement using five full-energy peaks representing the whole measuring range (121 keV, 344 keV, 778 keV, 964 keV and 1408 keV). The deviation of the peak position should be up to 0.2 keV without recalibration.
- Detection efficiency was checked at least once a week using the same full-energy peaks and the deviation should be up to 1 %.
- Resolution FWHM was checked at least once a week using full-energy peaks 121 keV, 778 keV and 1408 keV. The relative deviation should be up to 5 % and its trend was also investigated. The ever-worsening FWHM was an indication of a detector vacuum failure.

Calibration software

To calculate the activity of radionuclides identified in the spectrum, the full-energy peak efficiency should be determined. The determination is described in the Chapter 3.3.4. The efficiency curves were entered into the calibration software and automatically assigned to the measured spectra after the operator choose type of measured material, its density and container filling. Figure 21 shows the typical calibration curve.

Homogeneity determination

Twelve individual spectra were obtained by measuring the material in the container with four detectors, two above the container and two under the container. These spectra were used for evaluation of activity distribution in the material. Energy calibration was performed for each spectrum and stability was checked periodically using QA procedures. Each spectrum was divided into 32 sections, each of 256 channels. To

include the different detection efficiencies for the different detectors into the homogeneity estimation, the efficiency was determined in the middle channel of each spectrum for each detector, in each container position (1, 2, and 3), for the type of measured material and the current filling of the container. For each spectrum the counting rate was then determined in each section and corrected to the efficiency of different detectors. The differences between counting rates were then determined between all sections and should not exceed the required limit which for FRMF developed within this project, the acceptable value is set up to 30%.

Scale factors determination

Difficult-to-measure (DTM) radionuclides are radionuclides usually found in radioactive wastes, which activity cannot be easily measured by non-destructive measurement techniques. The scaling factor (SF) method is a widely accepted technique for assessment of the concentrations of DTM radionuclides from the concentrations of the easy-to-measure (ETM) radionuclides, based on the presumption for a correlation between the ETM and DTM radionuclide concentrations. Scaling factors are determined from the long-mean average of the ratio of the DTM to the key radionuclide, from regression analysis of the paired data or from other calculation methods. The free release measurement facility (FRMF) was capable to calculate automatically the activity of the DTM radionuclides based on the activities of the measured ETM radionuclides, with the use of scaling factors, which can be entered directly by the user for each waste stream. To be able to calculate the activity of DTM radionuclides, the scaling factor should be entered for each expected DTM radionuclide. This is done via a “scale factor” dialogue in the software for scaling factors determination. The operator should enter for the key radionuclides scale factors specific for all expected DTM radionuclides and its uncertainty. The record of all the scale factors was saved into a file which can be selected for each measurement individually. The software performed the measurement and analysis of the measured spectra and at the end of the analysis, the activities of DTM radionuclides were calculated, based on measured activities of key radionuclides (usually Cs-137 and Co-60), and decay correction could also be taken into account. The activities of the DTM radionuclides were reported in the measurement protocol.

Activity and minimum detectable activity determination for gamma spectrometry

The activity of the radionuclides in homogeneously contaminated material was calculated from a spectrum created by summing of the spectra from all four detectors, for each measurement position. The activity of the hot-spots was calculated from individual spectrum, acquired in the segment, where the hot-spot was identified. Activity A(X) for radionuclide X was calculated as A(X) = P(E,X) / (η(E) . Y(E,X) . t), where P(E,X) was net full-energy peak area of photons of energy E, emitted by radionuclide X, η(E) was the full-energy peak efficiency for photons of energy E,

Y(E,X) was yield of photons of energy E, emitted by radionuclide X and t was the live measuring time. For mass activity calculation, the activity was divided by the total weight of the measured material. For the surface activity calculation, the activity was divided by the total estimated surface of the measured material.

The minimum detectable activities were determined from the background within the FRMF chamber with the measuring container containing homogeneous materials and for hot-spots according to following equation:

$$MDA(X,E) = \frac{w}{t}\frac{k^2 + 2k\sqrt{\frac{b}{2l}\left(\frac{b}{2l}+1\right)(n_a+n_b)}}{1-k^2\left(\frac{u(w)}{w}\right)^2}$$

where *w* represents the conversion factor used to obtain the measurand (multiplicative inverse of full-energy peak efficiency for photon energy E and yield of photons of energy E emitted by radionuclide X), *t* was measurement live time, *b* was number of channel defining the Region of Interest for a spectrum peak, *l* was the number of channel at the right and left side of the peak used to estimate the continuous background, n_a and n_b represent counts in the right and left side regions of the peak and *k* was appropriate quantile of the standardised normal distribution. This equation corresponds to detection limit according to the ISO 11929:2010. For hot-spots, MDAs were also measured experimentally using a set of radionuclide standard sources of activities from 30 Bq to 1 MBq. Good agreement between the calculated and measured values was found.

3.3.4 Full-energy peak efficiency calibration

Because a lot of different types of materials are measured at the decommissioning sites, the efficiency calibration cannot be performed for all these materials using appropriate reference materials. Therefore full-energy peak efficiency as a function of photon energy was calculated using Monte Carlo MCNP code. A precise model of the measuring geometry including detectors, collimators and measuring container was prepared and then validated using real and spiked reference materials representing three typical categories of wastes (metals, building materials, light materials). The reference materials were standardised via inter-laboratory comparisons with participation of at least three national or designated metrology institutes and therefore traceability was assured as required by metrological legislation. Link to reference materials and standard sources description is in the Chapter 3.2.3.

The full-energy peak efficiency was defined as a number of impulses registered in a full-energy peak to a number of photons of the given energy emitted by the source. The efficiency η(E) was calculated as $\eta(E) = P(E)/(A(X).Y(E,X).t$, where P(E) was net peak area, A(X) was activity of radionuclide X, Y(E,X) was the yield (number of photons of energy E emitted by radionuclide X per one decay), t was live measuring time.

Relative combined standard uncertainty σ_η was calculated as $\sigma_\eta = (\sigma_P^2 + \sigma_A^2 + \sigma_Y^2)^{1/2}$, where σ_P was relative combined standard uncertainty of net peak area, σ_A was relative combined standard uncertainty of activity and σ_Y was relative combined standard uncertainty of yield.

Monte Carlo model for gamma spectrometry

The MCNP Monte Carlo model of the free release measurement facility includes detectors and measuring container in the different measuring positions. For the full-energy peak efficiencies calculation, the container in the model is 'filled' by different homogeneous reference materials, or standard point-like sources in different positions in phantoms. Because for some purposes total efficiency is valuable, a model of the whole facility was also created including a measuring chamber, conveyor etc. The IDM-200V detector model is shown in Figure 22. The model of FRMF is shown in Figure 23.

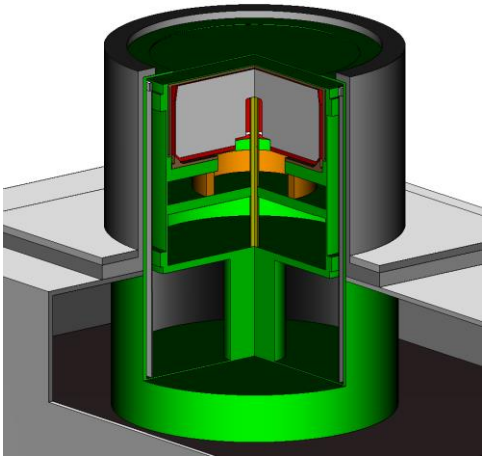


Figure 22. IDM-200V MCNP model

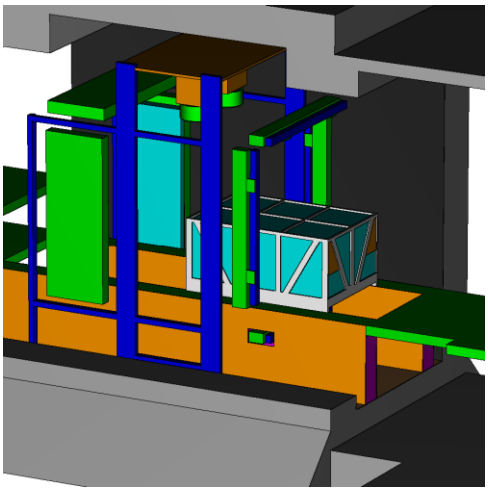


Figure 23. MCNP model of FRMF

Model validation

The Monte Carlo calculations were compared to experimental values obtained by the measurement of reference materials and standard sources traceable to national standards for the activity of radionuclides of several European metrology institutes. The model was considered validated, if the difference between measured and calculated values of full-energy peak efficiency was lower than 10 %. Results for homogeneously contaminated materials are in Table 3. The calculated full-energy peak efficiency as a function of photon energy for homogeneously contaminated gravel is shown in Figure 24.

E, keV	clay balls			gravel		
	η_{exp} ×10^4	η_{calc} ×10^4	d, %	η_{exp} ×10^4	η_{calc} ×10^4	d, %
238	4.78	5.30	-9.8	-	1.90	-
351	4.32	4.27	+1.2	1.46	1.60	-8.8
583	2.96	3.13	-5.4	-	1.28	-
609	3.25	3.10	+4.8	1.21	1.24	-2.4
1460	1.78	1.90	-6.3	1.01	0.92	+9.8
1765	1.57	1.73	-9.2	0.92	0.85	+8.2
	steel balls			* steel tubes		
661	0.728	0.784	-7.8	-	-	-
884	-	-	-	1.82	1.65	+10.3
1332	-	-	-	1.57	1.39	+12.9

* Higher value d for steel tubes is caused by different loading in the container.

$$d(\%) = 100 \times \left(\frac{\eta_{exp}}{\eta_{calc}} - 1\right)$$

Table 3 Comparison of measured and calculated full-energy peak efficiencies

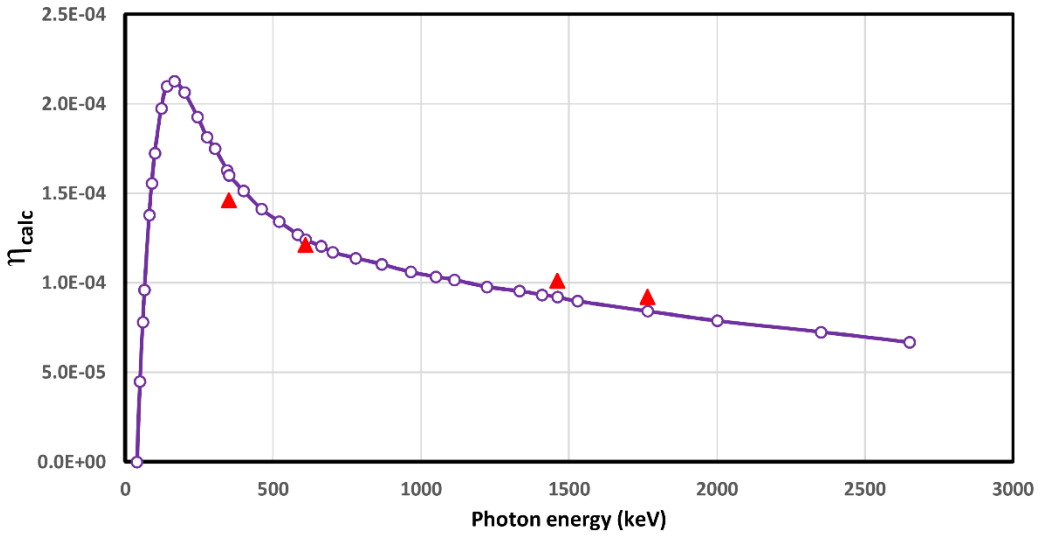


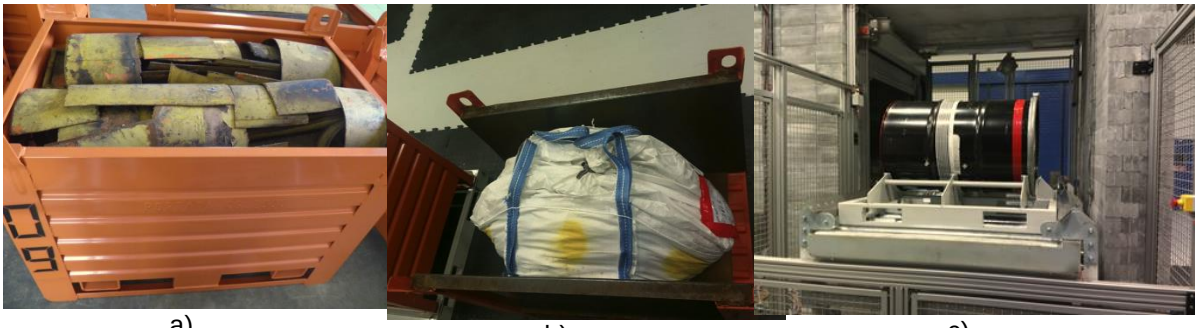
Figure 24. Full-energy peak efficiencies for gravel (red triangles are experimental values)

3.3.5 Measurement campaign

JRC performed the testing of the free release measurement system under industrial conditions on the decommissioning site at the Interim Storage Facility (ISF) in Ispra. The Decommissioning unit delivered seventy containers with clearable material according to the existing JRC license (see Figure 25).

The selected waste had not been exposed to irradiation for past 30 years, it was already measured and officially classified (according to the Italian Authorities). During the measurement campaign waste

measurements were related to the daily quality assurance measurements. Those quality assurance measurements included the background measurements and stability checks, and the measurements of simulated waste with and without reference radiation standards.



a) b) c)

Figure 25 Waste packages
a) Orange containers CPM530l 34 items (Metal, plastic, concrete) Volume (utile): ~ 0,6 m^3
b) Big bag 29 items (Concrete) Volume (utile): ~ 0.37 m^3
c) Drum 220L 7 items (Soft tissue) Volume (utile): ~ 220 l

The gamma spectroscopy measurement results under the MetroDecom measurement campaign were automatically analysed with the FRMF software. Based on these results JRC as an end-user of the FRMF concluded that:

- The measurements of JRC Ispra waste containers with the FRMF showed that the instrument was capable of providing spatial information that could be used for the localisation of non-homogeneities. The calibration factors used for determination of non-homogeneity corresponded to the actual situation and should be validated for typical measurement scenarios.
- As far as the peak search algorithm is concerned, the comparison of the results provided by the implementation of either technique showed that for short measurement times, the ROI peak search technique was more sensitive in the detection of existing peaks. However, it was noted that along with the actual energy lines the method reports also non-existing isotopes: a fact that results in incorrect estimation of the waste total activity content. Under the implemented FRMF setup and the applied activity quantification methods, reporting of results should be subject to further decay correction and the interference correction algorithm e.g. Ra-226/U-235.
- The campaign confirmed that the HPGe measurement system equivalent to the MetroDecom FRMF was capable of improving the clearance level measurements of the material packaged into the clearance container. The confidence in the clearance decision was increased by achieving MDAs much below 10% of the relevant clearance levels. Throughput of the clearance levels measurements could be increased by applying the spectra acquisition time of 3600 s. This measurement time would still allow clearing the material with sufficient margin, comparable to the overnight measurements using ISOCS.
- The instrumentation hardware of the MetroDecom FRMF required background subtraction to prevent activity overestimation. However, for heavier matrices the background reduction due to self-shielding was highly required so that suitable dummy containers (same matrix as real material but no artificial activity) should be used for background measurement.
- To ensure the quality of the measurements as well as the derived results, JRC developed a complementary to the already existing Quality Control programme. A part of the instrument Quality Control activities included the development of a set of experimental calibration procedures. The procedures were then developed and applied for the performance of measurements of reference materials aiming at the assessment of the instrument measurement quality features. The findings were reported to the instrument supplier and Monte Carlo code developer.
- The uncertainty budget reporting tailored for the possible deviations between Monte Carlo assumptions and real waste was necessary.

3.3.6 Passive neutron counting

The free release measurements with HPGe detectors were complemented with the passive neutron counting method. The both methods were used for different nuclide contents in the waste and were complementary.

The neutron slab counters were specially designed by JRC Nuclear Security Unit to have a high sensitivity to neutrons emerging from the waste package. Three slab counters were installed in a U-shaped arrangement (on top and on either side of the waste container) mounted on the inside to the FRMF shielding (Figure 26). The three neutron slab counters had fixed detector positions (unfortunately at quite some distance to the container). The conveyer system moved the container into position. Laser sensors calculated the correct location to stop the container. The dimensions of a neutron slab counter are shown in Figure 26.

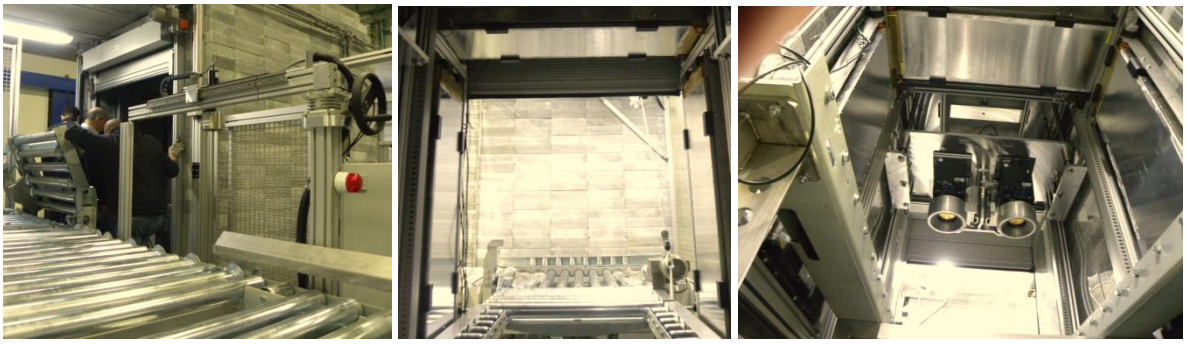


Figure 26. The JRC neutron slab counters mounted in FRMF

The point being that any observation of a net neutron count rate would indicate the presence of alpha emitters in the waste at a quantity that would disallow the waste package for free release.

Measurements were performed on both simulated waste matrices with calibrated sealed ^{252}Cf neutron sources inserted, and waste packages (orange containers) from the JRC decommissioning project. The described verification method applied for the free release verification of the sample waste containers demonstrated no presence of the neutron and alpha emitters.

Background measurements with the neutron counters were performed with a variety of waste containers. Some of the conclusions from this work are:

- The standard practice, as implemented in the FRMF operation procedures, of performing a neutron background measurement daily was of upmost importance in the effort to determine if a waste package had a net neutron count rate. This was to account for the variation of the neutron background that occurs due to changes in atmospheric conditions (pressure, humidity, cosmic-ray induced neutron background).
- The daily background measurements should be longer than the standard 300 seconds. This was necessary due to the relatively low count rate in the neutron detectors, and to establish a counting regime that allows the standard statistical treatment considering that part of the neutron background originate from relatively rare neutron bursts (of high multiplicity) produced in the atmosphere by high-energy cosmic particles.
- The background measurements should be done on a passive waste matrix similar to the objects to measure. This is important in particular for metallic waste. The reason was the changes in the neutron background level according to material type. A plausible explanation for this effect is the neutron bursts created by high energy muon created in the upper parts of the atmosphere reacting with high-atomic number (or high density) materials (spallation neutrons). A significantly higher counting rate was observed for increasing metal loading of the containers. Also, a highly hydrogenous matrix (e.g. concrete) could have the effect of lowering the neutron flux around the detectors.

A dependence of count rate due to gamma-rays was not observed. At the relatively low count rates the pile-up effect of gamma-ray events was negligible.

A potential neutron emitter that would be expected in the waste is $^{241}AmO_2$. The neutron emission was due to (α, n) reactions on oxygen. AmO_2 is a common constituent of most spent reactor fuel. In addition the low-energy gamma rays from Am-241 is easily attenuated (particularly) in a dense or large waste matrix. As an example the MDA value for an 801 kg container of metallic waste was calculated to be 139 MBq of alpha activity, or 174 Bq per gram of waste. This MDA value would be insufficient for a free release declaration but

useful in waste segregation as previously stated. In the case of difficult-to-measure isotopes (by gamma-ray measurements) such as Am-241, the neutron counting technique indeed serves a purpose.

3.4 Development of methods for monitoring in radioactive waste repositories

Radioactive waste and the facilities that store radioactive waste must be monitored for 10's to 100's of years to ensure the safety of workers, the local community and of the environment. This includes monitoring of radioactive gas emissions, thermal power of waste packages and temperature throughout the storage facility. Technology, techniques and procedures for long term monitoring have been developed through MetroDecom in fulfilment of the project aims.

3.4.1 Radioactive gas monitoring

Tritium and ^{14}C are particularly difficult to monitor as they do not emit gamma radiation. Both are produced, in varying quantities, in nuclear reactors of all designs. Radioactive waste, both operational and decommissioning, is likely to contain ^{3}H and ^{14}C. In the long term, waste is likely to degrade leading to the generation of methane and carbon dioxide gas. New technology and methods for monitoring the release of ^{14}C and ^{3}H from stored nuclear waste are required to support repository design and on-going radiation and environmental protection.

Through the project, three distinct, prototype technologies have been developed for the monitoring of ^{14}C and/or ^{3}H gas. These will be explained in turn.

Mid-infrared laser spectroscopy system for $^{14}CO_2$ monitoring

Sensitive and fast measurement of ^{14}C is needed to allow better monitoring and control of releases from nuclear power plants and radioactive waste repositories. In the previous EMRP project ENVO09 MetroRWM, a novel method for ^{14}C detection has been developed in the laboratory. It uses laser spectroscopy to detect radiocarbon in the form of carbon dioxide, and has great potential for on-line-on-site monitoring of ^{14}C emissions.

In this project, further work was carried out to develop the instrument towards a fully automatic prototype for on-site monitoring of radiocarbon gaseous emissions. Two main developments were made. First, a sampling system was implemented to extract the CO_2 from air and achieve the highest sensitivity. The system is also capable of converting methane in CO_2 thus allowing to differentiate between the different molecular forms of radiocarbon. Secondly, a new prototype of the optical detection system was built. The goal was to obtain a more suitable instrument for field measurements. Crucial improvements in terms of thermal and vibration insulation as well as size have been done, resulting in an instrument that fits within a moveable trolley.

The main component of the sampling system is a cryogenic trap where CO_2 is extracted from air. A sampling cycle consists of a period at low temperature were the CO_2 freezes in the trap and is then released by heating the trap above the freezing point of CO_2. In this way, almost pure (>90 %) is directed into the measurement cell. Once CO_2 is released, a new cycle starts as the trap is cooled down again, while the previous sample is being measured with laser spectroscopy. The developed method is only sensitivity to ^{14}C in the form of carbon dioxide, thus allowing differentiating between the different molecular forms of ^{14}C. Methane is converted into CO_2 using catalytic conversion, and by performing two measurements, with and without catalytic conversion, it is possible to determine the amount of ^{14}C in the form of CO_2 or CH_4. Figure 27 shows such spectra where one can see that the strength of the $^{14}CO_2$ absorption peak varies if the sample goes through the catalytic conversion unit or not. The absorption spectra can then be fitted by a sum of Voigt profiles, and the line area calculated. Using the known line strength, the activity concentrations can be determined. It corresponds to about 400 $Bq.m^{-3}$ of $^{14}CO_2$ and 200 $Bq.m^{-3}$ of $^{14}CH_4$. The sensitivity achieved here is thus sufficient for applications in nuclear facilities.

The optical setup was assembled on a 45 cm x 60 cm Nexus board, of which schematics is shown in the Figure 28. The components and their position were optimised to achieve a small footprint. A new design of the optical cavity was also realised with two objectives: thermal stability and damping of external vibrations. Simulations were performed to find the best design and based on the simulations, the cavity structure was finalised and is shown in Figure 28b. The cavity temperature is actively stabilised using thermoelectric modules. The optical part of the instrument sits in a 19" industrial drawer rack with wheels and has multiple options for shelf installations, as shown in Figure 28c. The components are positioned on three layers. The Nexus board with the optical setup was positioned on top to be accessible in case the optical alignments need revision. Vacuum pump, controller, power supplies and data acquisition are placed in the remaining space as illustrated in Figure 28d.

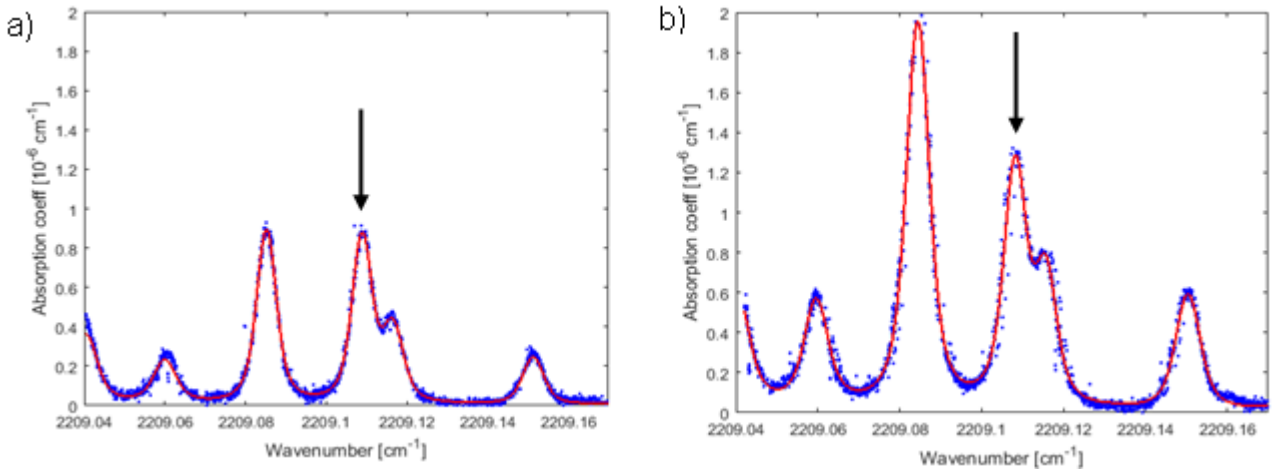


Figure 27. Spectra recorded with the laser spectroscopy setup after CO2 concentration with the sampling system. The data points are shown in blue and the fit in red. In a) no catalytic conversion was taking place, i.e. only $^{14}CO_2$ was detected. In b) methane was converted into carbon dioxide using catalytic conversion, and both $^{14}CO_2$ and $^{14}CH_4$ were detected. In both spectra, the targeted $^{14}CO_2$ absorption line is indicated by the arrow.

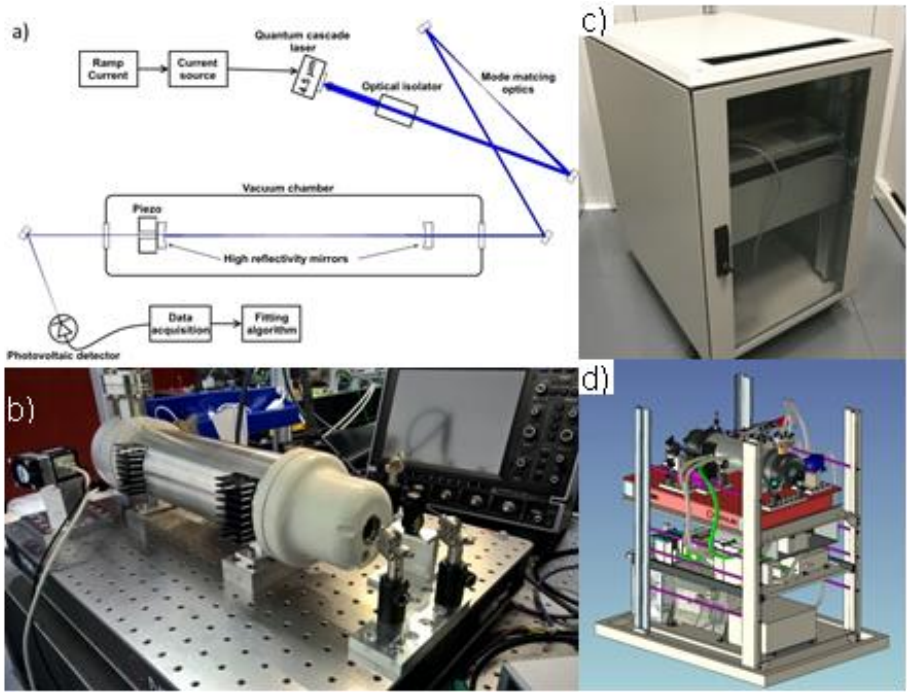


Figure 28.

a) The schematic of the cavity ring-down setup.

b) Photo of the optical cavity with its thermal insulation.

c) All the walls and the lid of the 19" Smarack can be easily removed and reassembled.

d) The component arrangement to the rack includes the components that are currently used in the spectrometer.

Real-time cryogenic separation of CO_2

A prototype radioactive $^{14}CO_2$ separation and concentration system has been developed for ^{14}C-in-air monitoring in nuclear installations or nuclear waste repositories during normal operation or decommissioning phase. The underlying idea of the method was to use a cryogenic separation process and to demonstrate the feasibility of using this generic method to isolate individual ^{14}C-labelled gases from air prior to their beta counting. The initial design was made considering the readily-available commercial instruments and devices

and combining them in an optimum way. The method investigated involved de-sublimating CO_2 at low temperature and pressure and would be accomplished using the following steps:

- aspiration of the air from the environment followed by compression (up to about 5 bar);
- cooling the air to the de-sublimation temperature of CO_2 (determined by the partial pressure of CO_2 in the gas mixture);
- pumping out and discharge back into the environment the other main gaseous components (mainly N_2, O_2 and Ar);
- warming up the frozen CO_2 and storing it in a suitable receptacle.

During the “cool-down” phase, the mixture has to be brought to a suitable temperature depending on the operating pressure: the final thermodynamic conditions in the condenser should ensure that the main components of air (N_2, O_2 and Ar) are still in gaseous phase (so they can be pumped out of the condenser and successfully separated from the CO_2), whereas the de-sublimation conditions for the CO_2 should be met, allowing the CO_2 to freeze inside the condenser (mainly on the internal walls and fins). The minimum temperature in the condenser has to be adjusted to balance the cooling conditions. This is a critical point, as the condenser walls and internal fins represent the main heat exchange interface to cool down the mixture and to freeze the CO_2. In fact, the CO_2 capture efficiency will depend mainly on the effectiveness of the heat exchange as well as on the thermodynamic conditions (temperature and pressure) that will be established and maintained for the gas mixture inside the condenser itself.

The initial conceptual design of the cryogenic system as conceived at ENEA is represented in Figure 29a. The main components required are the compressors and the cryogenic heat exchanger (referred to as the condenser). Moreover, some passive and active filters are used in the circuit to suppress the water and dust content before the mixture enters the condenser. Humidity values lower than 1% can be obtained at the exit of the filters, during the tests. The assembled system is shown in Figure 29b.

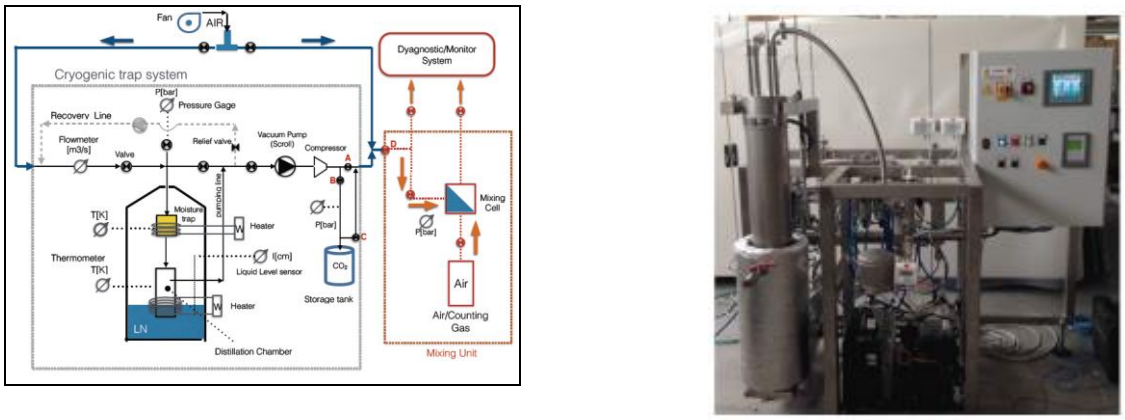


Figure 29. a) Conceptual design of the cryogenic system, b) Practical realisation of the system.

In the initial system set up, the important requirement to equip the condenser with an active tool to regulate the condenser wall temperature was not met during the system construction. This significantly limited the performance of the system, resulting in a CO_2 capture efficiency much lower than expected. To upgrade the system adequately and to maximise the amount of CO_2 captured, it was decided to use a different system to cool the air in the condenser that allows the operating temperature to be more effectively regulated. In order to maintain the condenser at higher temperature, a convective heat exchange with cold N_2 vapours has been evaluated. The “CRYOFLUID" heat exchanger, designed and realised by Air Liquide was determined to be a suitable system for this purpose and so it was coupled to the ENEA prototype for further investigation.

Inside the cylindrical trap there are two temperature sensors whose reading is acquired and monitored on the control panel: one sensor is located on the lower plate of the trap (TC_1) and the other one on the upper one (TC_2). A third temperature sensor is put in the gas out stream T_{out} at the cold-trap exit near the valve V_{20} (Figure 3.4.1.3a). An optical CO_2 detector, positioned downstream of the cold trap, enabled the CO_2 content to be monitored.

The aim of initial testing was to show the effective condensation of CO_2 in the cold trap, keeping the rest of the air in gaseous flow. Air was simultaneously pumped through the ENEA circuit (at 14 L/min), and the trap

cooled down using the CRYOFLUID exchanger. In this configuration the temperature TC_1 reached 157 K and TC_2 at 203 K, the out gas temperature T_{out} reached 253 K and the CO_2 content read on the detector downstream the trap was oscillating between 305 and 280 ppm. With the CRYOFLUID unit set to 113 K, and the air flow rate increased to 26 L/min; eventually TC_1 reached 143 K and TC2 reached 188 K. The content of the CO_2 recorded in the escaping air reduced to 20 ppm. It is reasonable to assume that temperature increases almost linearly along the height of the trap. This means that the lower base of the trap should be assessed in the temperature range 130-135 K. Since 130-140 K was the temperature range we expected to start freezing CO_2 along the inner surface of the cold trap, the settings were kept constant and the left to run for 45 min. After this time, the CO_2 content on of the discharged air stream was 0 ppm. To confirm that CO_2 had successfully been trapped, the CRYOFLUID unit was switched off and the trap allowed to warm. During the warming phase the CO_2 content reached a maximum of 3,500 ppm before returning to atmospheric levels. This test allows to claim that the cryogenic physics process to separate the CO_2 from a normal air flow is 100%.

Integrated bubbler-LSC for $^3H/^{14}C$ monitoring

A compact, automated system for the monitoring of carbon-14 and tritium-in-air has been designed by NPL and LabLogic Systems Ltd (Sheffield, UK). The system is intended for use in nuclear waste repositories as a replacement for traditional, manual methods of sample collection and measurement.

The design concept for the integrating radioactive gas monitor called for a compact, automated gas sampling and measurement instrument. It was decided to use existing technology where possible. This reduced development cost and risk. Two existing instruments were identified that, when combined, satisfy the primary functions of the proposed instrument. The first instrument is WILMA; an on-line radioactivity-in-water monitor developed by LabLogic, UK. The WILMA instrument automatically collects, prepares and measures aqueous samples. The measurement is made by liquid scintillation counting. The second instrument is HAGUE/MARC; a $^{14}C/^3H$ bubbler developed by SDEC, France. The HAGUE/MARC instrument collects $^{14}C/^3H$ by pumping air through a series of bottles containing trapping fluid. A furnace is used to convert organic-bound $^{14}C/^3H$ or methane bound ^{14}C into trappable HTO or CO_2. The bottles are then removed and liquid scintillation samples manually prepared. A photo and schematic of the integrated bubbler-LSC developed and tested through MetroDecom is shown in Figure 30 and the key performance parameters are presented in Table 4.

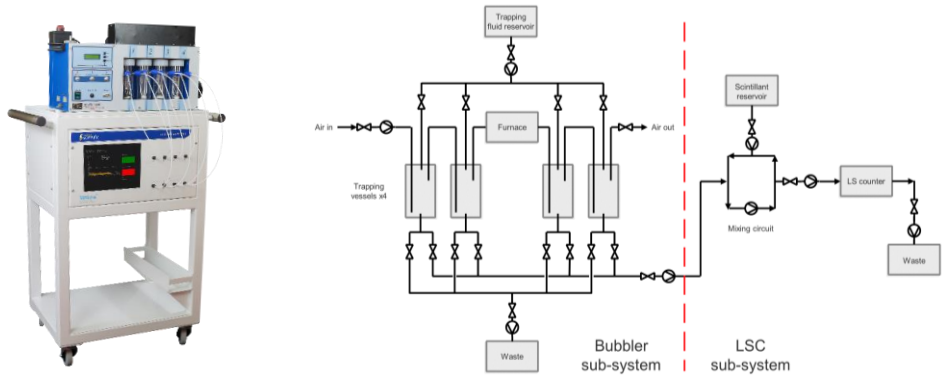


Figure 30.

a) Integrated bubbler (MARC ^3H bubbler) - LSC (WILMA on-line water monitor)

b) The schematic of the gas and fluid handling sub-systems.

Trapping yield:	96±4% (CO_2) 96±4% (HTO)
Furnace conversion yield:	93±7% (CH_4) 95±4% (HT)
Airflow accuracy:	±1%
Power:	700 Watts

Table 4 a) Performance parameters of Hague/Marc $^{14}C/^{3}H$ bubbler.

H-3 counting efficiency:	>15%
H-3 detection limit:	81.4 Bq/L (60 min count) 55.5 Bq/L (120 min count)
Sample volume:	5 ml (max)
Unattended operation:	30 days

Table 4 b) Performance parameters of WILMA on-line water monitor.

To calibrate the integrating radioactive gas monitor (and other bubbler systems), a new, two-stage flow-through dilution rig has been designed (see Figure 31 a). Each stage comprises of one high flow-rate mass flow controller (MFC) and one low flow-rate MFC. Radioactive gas is fed into the low flow MFC and diluting gas (typically air) fed into the high flow MFC. Special mixing T-pieces are used to ensure the gas is mixed before being injected into the the instrument under calibration. To confirm that this configuration produced a mixed stream of gas, a test was performed in-line with the NPL Primary Gas Counting System (PGCS). This system comprises four proportional counters of idential dimesions, except for their length, configured in series in a loop. Gas from the first stage of the flow-through dilution rig was passed directly into the PGCS. The PGCS loop was open to enable once-through counting. The flow-rates of the high and low MFC were adjusted several times over 1,500 seconds to change the active gas dilution factor (DF) and the count rate recorded for the three counters. The results are plotted in Figure 31 b). If the gas was well mixed, it would be expected that the acivity measurement made by each counter would be in good agreement. This was indeed found, with <2% difference between counters. Further the proportionality of count rate to DF varied by less than <1% for the 3 DF values investigated (regions highlighted in yellow in Figure 31 b).

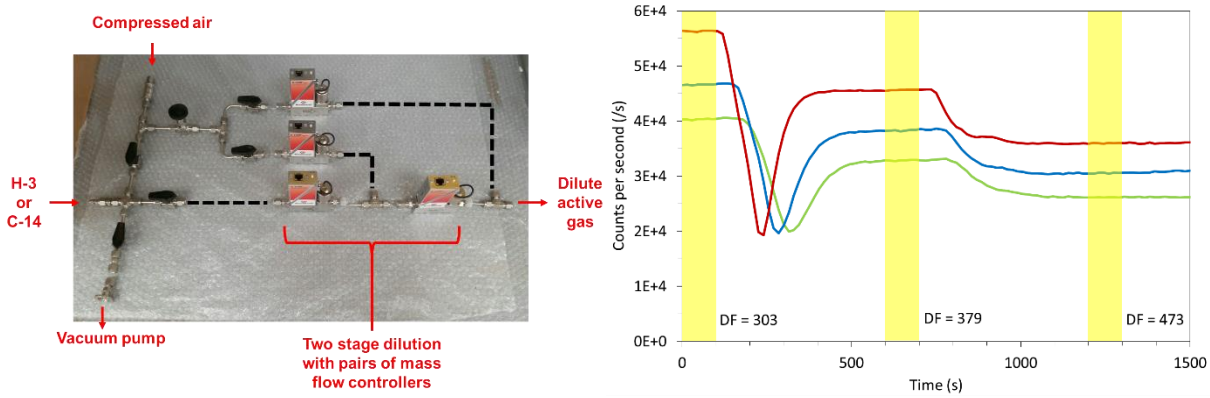


Figure 31. a) Layout of two-stage flow-through dilution rig, b) results of single stage mixing test.

3.4.2 Distributed temperature sensing

Raman distributed temperature sensing (DTS) technologies are currently under evaluation by the nuclear and hydraulic industries as it may bring promising alternatives to classical measurement techniques. The reliability of the DTS measurements, as well as the traceability to the temperature standards, must be ensured throughout the entire period of monitoring (typically over a few tens of years).
A DTS measurement is performed by the detection and the analysis of the backscattered light emitted into a silica optical fibre (OF) excited by a pump laser pulse. The OF acts as a temperature sensor along its whole length. The light scattered by an OF can be separated by three spectral emission lines which correspond to Rayleigh, Brillouin, and Raman scattering effects. As illustrated in Figure 32, some successive sections of the OF are excited and a part of the backscattered light is guided in the opposite direction by the OF towards the DTS system. Then, Stokes and Anti-Stokes radiation lines are separated by means of optical pass-band filters, and the corresponding intensities are detected by photodetectors. These elements are included in the detection path symbolised into the Figure 32. Finally the signals are processed with an electronic device, and the temperature profile along the OF is measured and displayed.

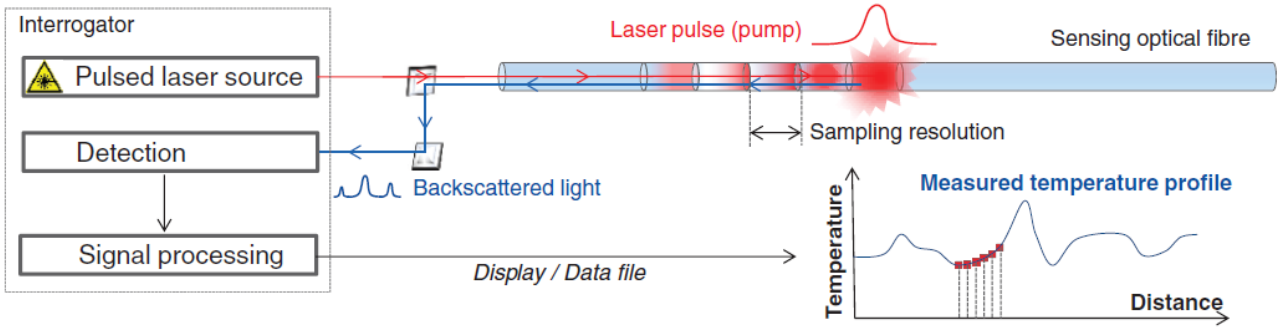


Figure 32. Principle of a Raman DTS measurement.

LNE has developed specific metrological facilities (a horizontal furnace and rectangular thermal enclosures) enabling to study the metrological performances of Raman-DTS systems equipped with an OF either deployed linearly or wounded on a spool. Such facilities will enable to separate thermal influence from parasitic mechanical effects.

As shown in Figure 33, the 25m horizontal furnace enables to perform temperature measurements with a DTS system, by controlling the stability and homogeneity of the temperature around the fibre and by avoiding any mechanical stress. The fibre is fully free to move inside the tube, and free of any mechanical constraints, besides its own weight. The furnace temperature is controlled and stabilised by a water jacket consisting in two concentric annular spaces located around the central tube.

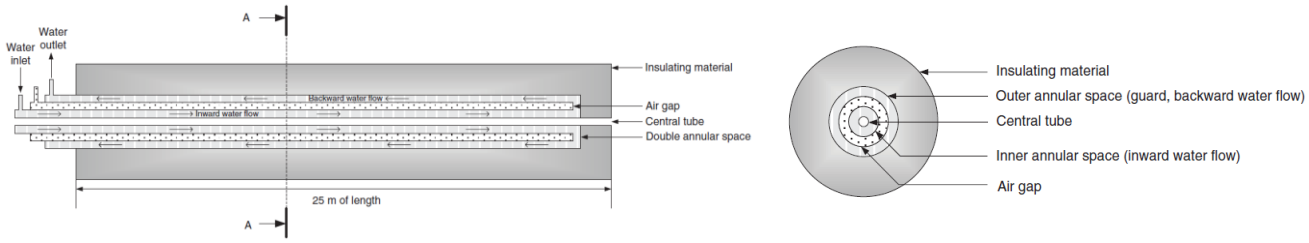


Figure 33. Schematic view of the 25 m horizontal furnace.

First, the thermal characterisation of the furnace has been performed for studying the performances of Raman DTS systems. The furnace is instrumented with two calibrated 100 Ω PRTs. The two probes are placed inside the central tube at a distance of 1 m from the front of the furnace. One probe remains at the same location while the second one can be moved into the furnace in order to study its temperature homogeneity. The temperature stability of the furnace has been evaluated over time with the two PRTs at four water temperatures of 3°C, 30°C, 60°C and 90°C. The results are shown in Table 5 a). The thermal homogeneity of the furnace is determined by comparing the temperatures measured by the two PRTs. The first probe is maintained at the same position. The second one can be shifted along the furnace length, following the axis of the central tube. The results are given in Table 5 b). The highest temperature deviation measured during the thermal homogeneity assessment is 0.004°C. As regard to the results obtained with the thermal stability measurement (up to 0.02°C), the temperature homogeneity is considered to be a negligible parameter.

Temperature (°C)	Temperature stability (°C)
3	±0.019
30	±0.011
60	±0.010
90	±0.010

Table 5 a) Temperature stability of the horizontal furnace measured at different temperatures over 5h.

Furnace temperature (°C)	Temperature homogeneity (°C)				
	Position #1	Position #2	Position #3	Position #4	Position #5
3	0.000	0.000	0.001	0.001	0.000
30	0.000	−0.001	0.000	0.000	−0.001
60	0.000	−0.003	−0.003	−0.001	0.003
90	0.000	0.002	0.003	0.004	0.003

Table 5 b) Thermal homogeneity measured in the horizontal furnace at different temperatures for five positions of the mobile PRT along the furnace length.

Thermal enclosures have been designed in order to study the influence of the OF curvature on Raman-DTS measurements. These specific facilities enable the control of the temperature for a long length of OF (up to 20 km) wound on a spool. The thermal enclosures are thermalised by means of a heat exchanger, consisting of a radiator cooled or heated by a water circulation. The homogeneity of the temperature field inside the thermal enclosures is ensured by an air flow provided by a fan positioned in the front of the radiator. Tests performed with calibrated 100 Ω PRTs inside the thermal enclosures have demonstrated that their temperature homogeneity is better than 0.1 °C and their temperature stability over duration of 20 h is better than 0.05 °C.

3.4.3 Practical acoustic thermometry

Acoustic thermometry is the method of determining temperature from the speed of sound. A simple measurement of the transit time of an acoustic wave along a waveguide, such as a pipe, is sufficient to determine temperature. Alternative technologies have limitations; thermocouples and resistance sensors that can de-calibrate and become contaminated. Thermal imaging requires a direct line-of-sight. And fibre-optics gradually darken under irradiation. In comparison, acoustic thermometry systems are intrinsically cheap and use well-established physics. The specific benefits of acoustic thermometry are:

- Microphones and speakers can removed and replaced.
- Lengths from 0.5 metres to at least 25 metres; shaped to measure average temperatures in zones.
- The tubes can be made of any material: metal, ceramic or plastic; chosen to suite the application.
- Drift free to at least 700 °C with an accuracy ~ 1 °C.

The benefits listed above mean that acoustic thermometry is well suited to nuclear waste repository monitoring. Cheap, durable, extended pipes can be custom made and installed either around individual packages or around multiple packages.

There are two configurations that enable measurements of temperature to be made; dual radius, where the transit time depends on the average temperature in the hot zone, and twin tube, where the differential transit time depends on local temperature along the length of the waveguide in the hot zone. Dual radius systems benefit from small uncertainties, low drift and high temperature tolerances. Dual radius systems tend to be simpler and can work on a larger scale. These two configurations are illustrated in Figure 34.

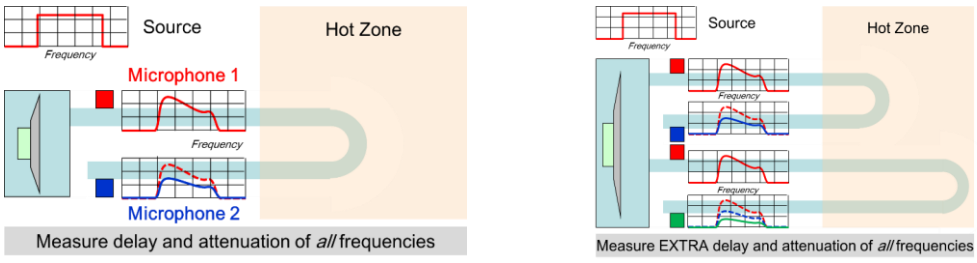


Figure 34.

a) Dual radius

b) Twin tube acoustic thermometry systems.

Both dual radius and twin tube configurations have been modelled and tested through the project. The thermal chamber built to test dual radius systems and some example results of a simple test are shown in Figure 35. The plot shows the sound intensity measured as a function of time following the emission of sound pulse. The temperature is determined from the delay and attenuation of all frequencies. The thermal chamber used to test twin tube systems is shown in Figure 36 along with calibration and drift measurements performed under three difference conditions; Test#1; 300 °C to 700 °C in steps of 100 °C, dwell 3 weeks.Test#2; 300 °C to 700 °C dwell of just 3 days dwell. And Test#3; 300 °C to 700 °C dwell of 2 days.

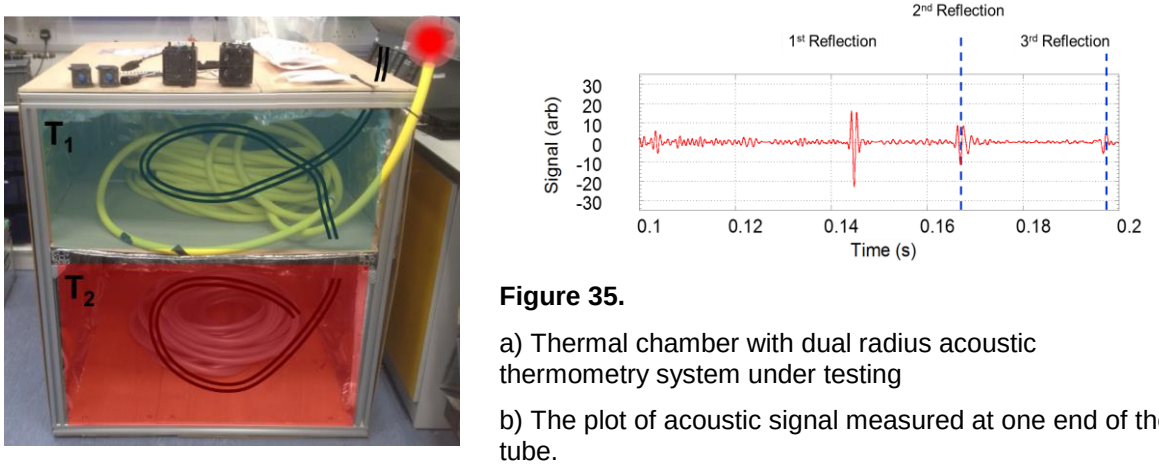


Figure 35.

a) Thermal chamber with dual radius acoustic thermometry system under testing

b) The plot of acoustic signal measured at one end of the tube.

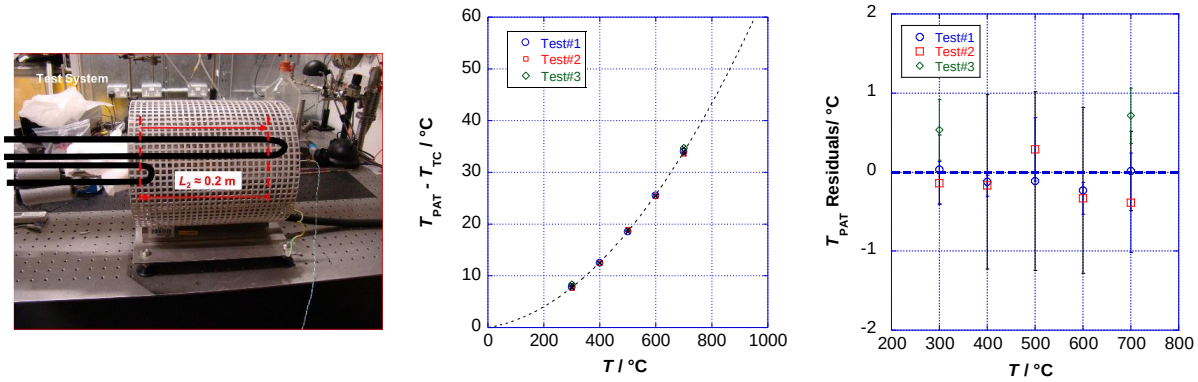


Figure 36.

a) Thermal chamber for testing twin tube acoustic thermometry systems
b) The calibration plot showing the differential temperature between the twin tubes as a function of chamber temperature
c) The drift plot

3.4.4 Measurement of thermal power of waste packages

The structural integrity of waste packages must be ensured for waste to be stored safely for extended durations. Nuclear waste may produce significant heat for many years and this thermal power may damage the waste package that it's stored in. The aim of the this work was to demonstrate the feasibility of a traceable calorimetric method for the direct measurement of thermal power (up to 500 W) of real size radioactive waste packages with an uncertainty better than 5 %. This involved the design of a prototype of large-volume calorimeter for packages of at least 0.175 m3, the establishment of calibration protocols and the evaluation of uncertainties of thermal power measurements.

The principle of the air flow calorimeter is the measurement of the temperature variation of the air circulating around an object (located inside the insulated chamber of the calorimeter) whose thermal power is needed to be measured. The determination of the thermal power is performed from the increase of air temperature with an assumption of a constant power dissipation of the object during the measurement. The design of the calorimeter with the reference package simulator is presented in Figure 37. The walls of the insulated chamber of the calorimeter are cut from thermal insulating rigid polyurethane foam whose thermal conductivity at ambient temperature is about 0.023 $W.m^{-1}.K^{-1}$.

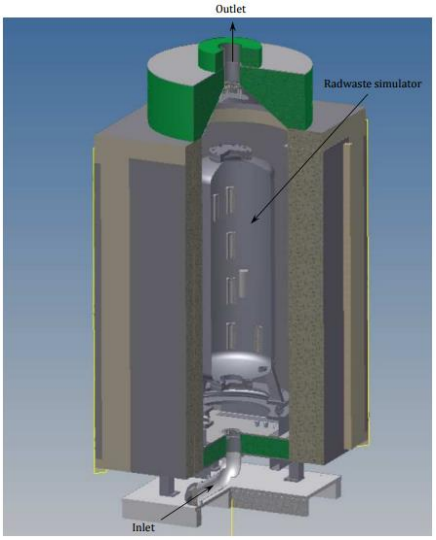


Figure 37. Design drawing of the air flow calorimeter.

In order to calibrate the calorimeter by electrical substitution (Joule effect), a radioactivity simulator was developed. This simulator consists of a metallic container from stainless steel 316L, and 8 electrical heaters placed in different locations in the sand that fills the container. This simulator tends to reproduce (or to occupy) as much as possible the volume of a real radioactive waste package inside the calorimeter. Furthermore, the simulator may mimic the heterogeneity of a radwaste package using the different heaters. The simulator is instrumented with 18 thermocouples (type K) placed on the internal face of the simulator before fill-up with sand. Power supply wires and thermocouples wires exit the simulator from two holes below the simulator. This calibration consists in the determination of the heat loss coefficient needed for the calculation of the correction to be applied, in order to get the actual thermal power released by the radwaste. The calorimeter was calibrated for two different electrical powers (around 102 W and 196 W) dissipated by the radwaste package simulator with a constant air flow rate of 25 kg/h, in order to determine the heat-loss coefficient *K*. Finally an uncertainty budget for the measurement of thermal power was calculated; as shown in Table 6, where Q_m is the air flow rate, C_p is the specific heat of air, R_{in} is the inlet resistance, R_{out} is the outlet resistance, *K* is heat-loss coefficient, T_{amb} is the ambient temperature and *P* is the measured power. An additional component for thermal power loss in the wall results in an expanded uncertainty (k=2) of 4.7 W.

Quantity X_i	Estimation x_i	Std uncertainty $u(x_i)$	Sensitivity coefficient	Contribution to the uncertainty (W)	(%)
Q_m	0.00694 kg.s^{-1}	1.76 ×10^{-5} kg.s^{-1}	$C_p \cdot (T_2 - T_1)$	0.78	13.5
C_p	1004 J. kg^{-1}.K^{-1}	2 J. kg^{-1}.K^{-1}	$Q_m \cdot (T_2 - T_1)$	0.35	2.8
R_{in}	28.95 Ω	0.01 Ω	$Q_m \cdot C_p \cdot (f(R_{in}))$	-0.38	3.2
R_{out}	33.57 Ω	0.01 Ω	$Q_m \cdot C_p \cdot (f(R_{out}))$	0.46	4.7
K	1.7937 W.K^{-1}	0.15 W.K^{-1}	$(T_{test} - T_{amb})$	1.53	51.9
T_{amb}	22.59 °C	0.58 °C	$-K$	-1.04	23.9
P	**195.4 W**			**2.1 W**	

Table 6 Uncertainty budget for the measurement of thermal power using the air flow calorimeter.

3.5 Reference Materials and standard sources

Reference materials and standard sources were developed and manufactured for calibration of measuring devices for segregation of materials and free release measurement, radiochemical analysis, measurement of gases in radioactive waste repositories and surface contamination monitors.

3.5.1 Reference materials and standard sources for segregation of materials and for free release measurements

Ball sources

1100 inactive steel balls and 1100 plastic balls with drillings for point sources as well as 1100 Eu-152 and 1100 Co-60 point sources with well-defined activities were produced by CMI and JRC. The balls together with the point sources mimic a quasi-homogenous volume activity standard for the measurement systems and allows to verify the Monte Carlo based calculation of the efficiency calibration of the systems in an energy range of photons between 121 keV and 1.4 MeV. Examples of the balls and a construction drawing is shown in Figure 38**.**

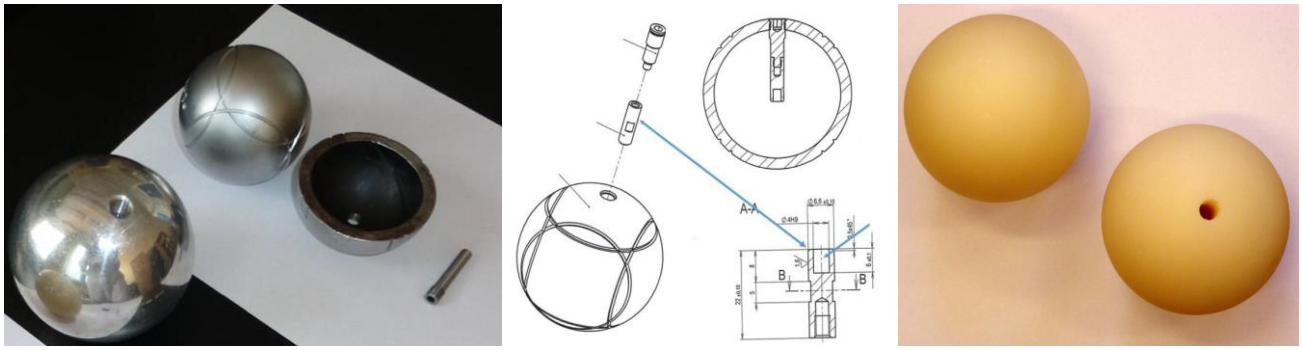


Figure 38. Steel and plastic balls with drillings for Eu-152 and Co-60 point sources

Standard sources

A set of 36 point-like standard sources of Co-57, Co-60, Cs-137, and Am-241 (9 activity levels for each nuclide in the range of 30 Bq to 1 MBq) were produced by three partnering laboratories.

A set of 9 point-like standard sources of Eu-152 with activities in the range from 100 Bq to 1 MBq were produced by CMI, JRC and PTB. The sources were used for hot-spots simulation in phantoms filled with 'inactive' steel, building or light material (steel balls, low activity gravel, plastic balls). Metal phantom and standard source are shown in Figure 39.

Figure 39. Metal phantom with Eu-152 standard source

Scanning source and scanning materials

A Co-60 source with an activity of 95 MBq and 6 different non-active materials shown in Figure 40 were selected to test the scanning facility shown in Figure 13.

Figure 40. Six non-active materials (stainless steel pipes, wood, iron tubes, glass wool, aluminium plates, copper cables) for scanning.

3.5.2 Reference materials and standard sources for radiochemical analysis

Three different blank materials (concrete, steel solution, graphite) and five activity standard solutions (Sr-85, Zr-93, Sm-151, U-236, Pu-241) were selected and characterised to test and validate the new developed radiochemical analysis methods. In the validation step, an aliquot of one of the blank materials and aliquots of one or several activity standard solutions were gravimetrically transferred into the container types, used in the first step of the radiochemical procedure.

Blank materials

Figure 41 shows the three blank materials (2.5 kg concrete powder, 2.5 L steel solution and 1 kg graphite). The residual activity of the blank material caused by natural radionuclides were determined.

Figure 41. Three inactive materials (concrete powder, steel solution, graphite) to be spiked with standardised solutions

Standardised solutions

The activity of five different single nuclide solutions (Sr-85, Zr-93, Sm-151, U-236, Pu-241) were determined by NPL, ENEA, JRC and PTB with well-established methods traceable to primary measurement methods.

3.5.3 Gaseous reference materials

NPL produced a cylinder of dilute $^{14}CO_2$ and $^{14}CH_4$ in-air to enable VTT to test the novel mid-infrared spectrometer. The cylinder was produced by a 3-stage gravimetric dilution from two “master” ^{14}C cylinders

($^{14}CO_2$ in CO_2 and $^{14}CH_4$ in CH_4). The two master cylinders were standardised by means of absolute internal-gas length compensated proportional counting. Details of the different gas cylinders produced during the project are given in the Table 6 below.

Table 6 Details of radioactive gas cylinders produced during MetroDecom for testing ^{14}C monitors.

Cylinder	Contents	How it was made	Activity determination	Activity concentration
Master $^{14}CO_2$	$^{14}CO_2$ in CO_2	Concentrated $^{14}CO_2$ transferred from break-seal ampoule to evacuated cylinder, then pressurised with pure CO_2.	Absolute gas counting	16 MBq.m^{-3}
Master $^{14}CH_4$	14CH4 in CH4	Concentrated $^{14}CH_4$ transferred from break-seal ampoule to evacuated cylinder, then pressurised with pure CH_4.	Absolute gas counting	180 MBq.m^{-3}
Dilute Mix A	$^{14}CO_2$ & $^{14}CH_4$ in zero-air	1.5g of $^{14}CH_4$ from Master $^{14}CH_4$ cylinder + 77g of $^{14}CO_2$ from Master $^{14}CO_2$ cylinder + 160g of zero-air	Gravimetric dilution	2.2 MBq.m^{-3} ($^{14}CO_2$) 3.7 MBq.m^{-3} ($^{14}CH_4$)
Dilute Mix B	$^{14}CO_2$ & $^{14}CH_4$ in zero-air	2.7g of $^{14}CH_4$ / $^{14}CO_2$ in-zero-air from Dilute Mix A cylinder + 220g of zero-air	Gravimetric dilution	2.5×10^{-2} MBq.m^{-3} ($^{14}CO_2$) 4.0×10^{-2} MBq.m^{-3} ($^{14}CH_4$)
Dilute Mix C	$^{14}CO_2$ & $^{14}CH_4$ in ambient air	0.81g of $^{14}CH_4$ / $^{14}CO_2$ in-zero-air from Dilute Mix B cylinder + 0.0086g stable CH4 + 95g of ambient air	Gravimetric dilution	2.1×10^{-4} MBq.m^{-3} ($^{14}CO_2$) 3.4×10^{-4} MBq.m^{-3} ($^{14}CH_4$)

3.5.4 Reference materials and standard sources for surface contamination monitors

IFIN-HH and ENEA-INMRI each prepared four large-area sources (eight in total) using calibrated solutions of ^{60}Co and ^{137}Cs. The surface emission rates of these sources, shown in Figure 42, were measured according to ISO 8769 by both IFIN-HH and ENEA-INMRI using similar counting systems and the results were compared.

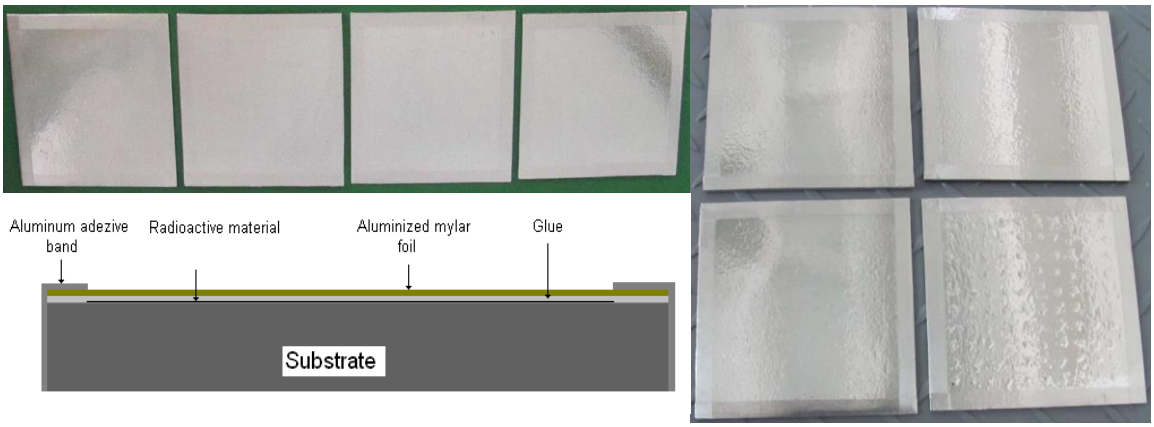


Figure 42. Eight large-area sources from calibrated solutions of Co-60 and Cs-137

3.6 Effective collaboration between JRP partners

The project gathered thirteen partners and one integral REG from eight European countries and from the EC JRC. Special attention was paid to effective collaboration, examples follows:

Collaboration between national metrology institutes CMI and NPL, decommissioning site operator JRC and measuring devices producer NUVIA produced added value that individual partners could not achieve by

themselves. NUVIA with the metrological support of CMI constructed and produced both measurement facilities and installed them on the JRC decommissioning site. All partners carried out joint on site measurements of calibration reference materials and standard sources delivered by CMI, and also real wastes delivered by JRC. The project was implemented in in relatively short time under industrial conditions of the decommissioning site. A large effort was invested in assuring compliance with industrial safety/security standards for work on the nuclear decommissioning site, to ensure the project implementation on the Ispra site is in agreement with existing Italian legislation in health, safety and environment. The safety, security and legal aspects were respected by all partners in collaborative way. The work could not be done without measurement facilities production, installation and operation, without metrological support on European level, and without access to the decommissioning site.

The realisation of metrological facilities for the calibration of Raman distributed temperature sensors by LNE relied critically on the support of ANDRA and EDF. ANDRA and EDF provided instruments for testing and expertise that guided the design and build of the facilities.

The gas samples produced to test and validate the mid-infrared laser spectroscopy ^{14}C monitor required careful planning. The precise gas mixture, level of radioactivity, pressure and volume had to be appropriate for the application. Close calibration between VTT and NPL ensured that the highly complicated samples were successfully delivered and utilised to test the ^{14}C monitor system.

3.7 Summary

Development of methods for the radionuclide characterisation of different types of materials present on the site being decommissioned.

Mapping of activation and contamination inside nuclear facilities

- Methods based on radioluminescence could be applied to detect alpha contamination from a stand-off distance in presence of normal background light by utilising the solar blind region.
- The capability of the GAMPIX to identify gamma contamination in situ has been validated in a traceable manner. The gamma camera can localise and quantify the dose rate of spread sources and its response is linear with the considered activities. Moreover in situ measurements prove the capability of GAMPIX gamma camera to reconstruct and localise radioactive source.
- A test case in a real decommissioning project dealing with contamination depth distribution mapping clearly showed the possibility to integrate various types of data with different spatial support and quality (e.g. dose rate, surface contamination, gamma spectroscopy) making use of advanced geostatistical analysis in order to optimise the global uncertainty on the final result as well as the costs of the pre-decontamination characterisation process.
- New methods were discovered to measure the activity of large-area beta emitting sources, to evaluate the uncertainty in the measurement of the surface beta contamination and to transfer efficiency for determining the activity per unit area of beta contamination.
- It was demonstrated how some of the novel mapping and measurement techniques can improve or be combined to map a contaminated area and find hotspots of different type of emissions and it some cases even quantify it, to allow the decontamination of an area in a nuclear decommissioning site.

Sampling

- The frequentist and Bayesian models applied to sampling were reviewed and compared during the work of this project, which will be able to offer guidance to different parts of the nuclear industry on what model to apply in each circumstance, to improve the efficiency of the sampling on case to case samples, which was this sub-objective aim.

Rapid radiochemical procedures

- The use of ICP-QQQ-MS to measure Sm-151 in nuclear decommissioning matrices after sample dissolution and radiochemical clean-up was developed. This is the first time ICP-QQQ-MS and reaction-cell based separation has been applied to Sm-151.
- The new dissolution method to allow destructive analysis of graphite, a common decommissioning matrix, by ashing the material and using lithium borate fusion has been developed, reducing the use of very hazardous reagents as perchloric or sulphuric acid. The method was validated by measuring the content of elemental materials at different masses of samples.
- Radioanalytical methods for the rapid and simultaneous determination of actinides and radiostrontium, and for determination of Zr-93 and Sm-151 in matrices of interest for the decommissioning field, such as concrete, graphite and steel were developed.

Scaling factors

- The progress was made in a work on scaling factors applying the principles set out in ISO 21238:2007 resulting on identifying decommissioning matrices of specific relevance to nuclear decommissioning within the EU and selecting priority hard-to-measure radionuclides for further study. Details of existing scale factors for these nuclides and matrices were specified, along with an analysis of needs for future research and scale factor measurement.

Segregation measurement of solid waste

- The developed waste pre-selection (segregation) facility allows quick and accurate measurement to avoid unnecessary storage of wastes in radioactive waste repositories, or inaccurately sending the material for free release measurement. This measurement technique is robust enough to allow long-term trouble-free in in-situ measurement.

  The facility allows direct measurement and segregation coefficient calculation via detection of gamma-ray emitting radionuclides, by using four plastic scintillating detectors in high efficiency measuring geometry. Precise calibration was performed using traceable reference materials and standard sources, and Monte Carlo method for total efficiency calculation. Each container with wastes of a volume about 0.5 m^3 was measured, and the data stored and evaluated. As a final result of the measurement of each individual container, the segregation coefficient was determined allowing the operator to make the decision about segregation of the waste package either to repository, or handover to free release measurement. The measurement parameters, current measurement status and measured data were continuously displayed during measurement. Based on this data, the measurement protocol was prepared and printed. The MC model for experiments with the MetroDecom IP2 container showed the difference between measured and calculated values of total efficiency is less than 20%.
- Neutron measurements were proposed as part of the waste segregation instrumentation in the facility. Neutrons, in contrast to gamma-rays, penetrate relatively easily high density matrices such as concrete and metal. For this purpose neutron slab counters (with a large surface area) were installed in the facility. The point being that any observation of a net neutron count rate would indicate the presence of alpha emitters in the waste at a quantity that would disallow the waste package for free release.

Free release of solid waste

- The free release measurement facility developed allows direct measurement and activity determination of gamma-ray emitting radionuclides using germanium detectors with high energy resolution and efficiency, precise calibration using traceable reference materials and standard sources, and Monte Carlo method for full-energy peak efficiency calculation. Each waste container of a volume about 0.5 m^3 was measured by four HPGe detectors in three positions and twelve individual gamma spectra were acquired, stored and evaluated for homogeneity estimation. If homogeneity was in accordance with regulators' requirements, mass and surface activity could be averaged over the total mass and surface of waste material. If If the material wasn't homogeneous, hot-spots were identified in respective segment of the container and approximate activity was calculated. For this calculation, full-energy peak efficiencies were determined for each segment using point-like standard sources and phantoms. The hot-spot were then taken out, or all material in the container was sent for repository acceptance measurement. For this decision, economic criteria were taken into account.
- For radionuclides of interest not identified in the spectrum, minimum detectable activities (MDAs) are calculated and included into final results evaluation. Difficult-to-measure radionuclides are taken into account via scale factors associated to easy-to-measure key radionuclides, generally Cs-137 and Co-60. Scale factors are calculated using developed complex free release measurement software. Measurement parameters, current measurement status and measured data are during measurement continuously displayed and based on this data, measurement protocol is prepared and printed.

Methods for monitoring in radioactive waste repositories

Monitoring of radioactive gas

- The new infrared spectroscopy system developed by VTT demonstrated for the first time detection of elevated levels of ^{14}C from an air sample matrix. The use of an advanced sampling system resulted in a sensitivity of about 40 $Bq.m^{-3}$. Such a high detection sensitivity using laser spectroscopy has never been reported before. In addition, the use of an optical technique allows for on-site measurement capabilities, which is not possible with currently used techniques.
- A prototype cryogenic system for radioactive $^{14}CO_2$ gas separation for nuclear waste monitoring has been designed and practically realised at ENEA. The system showed effective feasibility of the concept.

Accurate temperature control was the main problem encountered on the first version of the prototype. Coupling this original system with the Air Liquide CRIOFLUID heat exchanger showed that the CO_2 high efficiency separation mechanism from air by de-sublimation is feasible with very high efficiency. The next step would regard the optimisation of the $^{14}CO_2$ evaporated phase storage, possible miniaturisation of the prototype and proper selection of components in order to reduce the system price.

- An integrated bubbler-LSC has been developed and tested by LabLogic and NPL. This new system will provide time and cost savings to nuclear facilities through the automation of low-level $^{3}H/^{14}C$ in-air monitoring. The new flow-through calibration system will enable a more realistic calibration to be performed than can currently be achieved with existing closed loop gas calibration systems. It is intended to offer this as a new calibration service available to all bubbler manufacturers and operators.

Distributed temperature sensing

- New facilities and methods have been developed at LNE for the metrological characterisation of temperature sensors based on OFs. A horizontal furnace of 25 m long and several thermal enclosures have been specifically designed in order to study the metrological performances of Raman DTS systems equipped with an OF either deployed linearly or wound on a spool. The temperature homogeneity and the temperature stability over duration of 20 h of these facilities have been measured better than 0.1 °C and 0.05 °C respectively. These facilities have been used to assess a set of metrological characteristics (such as spatial resolution, temporal/spatial dispersions, and trueness error) on two Raman DTS interrogators provided by Andra and EDF. A guide describing experimental procedures applicable to the metrological characterisation of Raman DTS systems has been established, and opportunities in terms of standardisation actions have been identified in the committee IEC TC/SC 86 C/WG2

Practical acoustic thermometry

- Acoustic thermometry is a cheap, reliable and practical solution for monitoring temperature of nuclear waste packages sored in waste repositories. Equipment, methods and models have been developed through MetroDecom to enable the testing and calibration of a variety of acoustic thermometry systems that will accelerate the uptake the technology within the nuclear industry.

Measurement of thermal power of waste packages

- A new calorimeter prototype, based on the “air flow” calorimetric approach, for the measurement of thermal power of nuclear waste packages of at least 175 litres, as well as “power reference package” for its calibration by electrical substitution, have been designed. The measurements performed with this calorimeter have demonstrated the feasibility of thermal power measurements up to 500 W with a low-level of uncertainty (lower than 5 %).

Reference Materials and standard sources

- A series of activity standards and reference materials have been produced. These products were designed to compensate the lack of them in the radioactive waste characterisation field and meet the reliability and traceability requirements set. These products were used to calibrate and test the new measurement devices and methodologies which were developed within the JRP. The methods for the production of the materials developed in the project can be further used to build future calibration sources for the wide field of decommissioning measurement devices.

4 Actual and potential impact

4.1 Metrology achievements

The project addressed the needs of decommissioning process by following novel or improved solutions:

- Alpha and gamma radiation mapping automated remote detection measurement devices.
- Statistically valid sampling and material characterisation strategies.
- Rapid and semi-automated analysis procedures for *in situ* measurements.
- Pre-selection and free release high throughput waste measurement facilities, based on standardised concept characterised by unique eco-friendly lead-free shielding,
- *In situ* and on line monitoring of radioactive gas emissions from nuclear facilities.
- Very long term temperature and integrity monitoring of radioactive waste repositories.
- Traceable and reliable calibrations and tests of the measurement systems with small uncertainties.

4.2 Dissemination activities

4.2.1 Scientific publications

The project has generated 18 high impact publications submitted to peer-reviewed journals. These incorporate the significant scientific outputs of the project; a list is provided in section 6. In addition, 4 ISSN/DOI reports and 1 PHD thesis were published. A number of 7 articles/press release for publication to trade journals to disseminate beyond the specialist in nuclear decommissioning were published and are available on the project website.

4.2.2 Conferences

The results of the project were presented (oral presentations or posters) to stakeholders and wider scientific audience at national and international conferences. The project submitted 39 conference presentations in total. These presentations reached both the wider scientific audience in general conferences such as the International Conference on advancements in Nuclear Instrumentation Measurement Methods and their Applications (ANIMMA), the International Conference on Radionuclide Metrology and its Applications (ICRM), the International Nuclear Chemistry Congress (INCC), the IEEE/International Symposium on Room-Temperature Semiconductor Detectors (RTSD) as well as targeted audience in specialised conferences such as the International Conference on Nuclear Decommissioning (ICOND), the Waste Management Symposia International Conference (WM). Positive reactions were received to all these contributions, attracting discussions and comments. In most cases MetroDecom papers were included in conference proceedings published in scientific journals after external peer review.

4.2.3 Stakeholder Engagement

The project has developed metrological capabilities aimed at laboratory applications and on site measurements in the area of decommissioning of nuclear facilities for stakeholders' and end-users' needs.

A Stakeholder Committee (SC) with 12 members of high competence in all aspects of decommissioning work and organisations, including members of IAEA, EU, ISO, IEC Standards Committees was created to help steer and assess the project. The project work was oriented to maximise the impact, providing practical solutions for the end users such as the guidelines, measurement methods and tools, nuclear data and reference materials for their work. The SC and the end users were invited and actively involved in the project workshops as well as involved in training, demonstration exercises, presentations, publications and expert discussions. In addition, information exchange was maintained with 10 external collaborators in industrial and academic organisations as end-users representatives. Stakeholder database via web on-line registration was maintained. All project deliverables are available to stakeholders via web application.

During the project scientific visits of all partners to external decommissioning sites of interest for the project were arranged.

- Olkiluoto nuclear site and ONKALO deep geological repository in Finland was visited on September 2015. The Olkiluoto nuclear site consists of two running nuclear reactors and an additional one under construction. ONKALO deep geological repository for the final disposal of spent nuclear fuel is the most advanced repository site of this type in the world. Partial project results were presented to specialists of operators and discussed with them.
- Belgoprocess and Belgian Reactor 3 (BR3) decommissioning site was visited on February 2016. Visit at Belgoprocess was focused on characterisation and measurement of radwaste, flow shop for sorting and free release of waste concrete and storage of conditioned radwaste. Visit of BR3 decommissioning site at SCK•CEN was focused on decommissioning of the 10 MW PWR building, especially measurement techniques used. Partial project results were presented to executives and specialists from Belgian Federal Agency for Nuclear Control (FANC), the Belgian Agency for Radioactive Waste and Enriched Fissile Materials (ONDRAF/NIRAS), Belgoprocess, and SCK•CEN.
- French National Radioactive Waste Management Agency (ANDRA) Andra´s Aube Industrial Facilities (CSA) in Soulaines-Dhuys was visited on March 2017. This facility, with 95 hectare footprint and a capacity of 1 million m3 is dedicated to low and intermediate level short-lived waste. Discussions with CSA's specialists was focused on general concept and practical implementation of the French radioactive storage approach, characterisation, pre-selection and free release of waste and storage of conditioned waste.

The knowledge transferred during these visits was very important to understand the end-users operate, the problems they faced and discuss the scientific and technical work to better meet their needs.

More than 70 stakeholders and end-users were integrated in the project.

Special attention was paid to uptake and commercialisation of project outputs by the user community. Examples of the outputs being taken on by the relevant communities are given in section 4.4.

4.2.4 Workshops

Three workshops for wide audiences of stakeholders including nuclear decommissioning operators, nuclear regulators, standardisation bodies and measurement instruments manufacturers were held which covered the new measurement techniques, best practices, reference materials, and on-site demonstrations of new measurement systems as high-throughput clearance system.

- The first workshop was held on November 2015 in company NUVIA (JRP unfunded partner) assembly shop in town Třebíč (Czech Republic), where the new large scale industrial prototype free release measurement facility was under construction. The project outputs were presented to the wider community of stakeholders such as legal authorities, radiation protection and standardisation bodies, radioactive waste agencies, and producers of measurement devices.
- The second workshop was held on October 2016 in Italy at EC JRC, Ispra. During this workshop project outputs were presented and the free release measurement facility in ISF building was demonstrated to the wider community of stakeholders including representatives of IAEA and standardisation bodies. Demonstrations were made about the technical principles as well as practical use on decommissioning site (such as JRC Ispra). The reference materials developed were shown explaining how they can guarantee traceability and quality management of the newly developed instrumentation.
- The third MetroDecom Stakeholder workshop was held on June 2017 at The Centre at Birchwood Park, Warrington (UK). The date and venue were chosen to attract a large number of stakeholders in the Nuclear Industry sector in the UK, as the workshop was day before of the IRFM (Ionising Radiation Metrology Forum) on a strategic location. For these reasons the workshop was very successful and was attended by many UK (but not only) nuclear industry stakeholders, with 65 registered participants. The agenda included the information about the follow-up project MetroDecom II. The workshop gave the opportunity to obtain feedback on possible future technical developments.

4.2.5 Impact on standards

The results of the project have been disseminated to international and European standardisation bodies and working groups including IAEA Safety Standards and supporting publications (Safety Reports, TECDOCs), European Commission (EU Directives and Recommendations), nuclear agency (Expert Group on Fukushima Waste Management and Decommissioning R&D), ISO/IEC/CENELEC Standardisation Committees working on radioactive surface contamination monitoring, and French Standardisation Association (AFNOR).

Work from the project was incorporated in the published documentary standards: IEC 60325:2002 Radiation protection instrumentation and ISO 7503:2016, ISO 8769:2016 Measurement of radioactivity. The facility for the characterisation of Raman-DTS systems will enable LNE to contribute to the standardisation committees related to the distributed temperature measurements ie. AFNOR UF86 “photonics” and IEC TC/SC 86C/WG2 “Optical fibre sensors”.

4.2.6 Training

Training sessions were held during the second workshop with extensive participation of stakeholders and end-users. Class and practical demonstrations regarding the operation of the novel free release measurement facility were carried out including measurement of waste containers with real and simulated waste. In addition, two lectures were given at the 8th edition of the International Summer School on Nuclear Decommissioning and Waste Management on September 2016 in JRC Ispra, Italy. All training materials are available on the project web site.

4.2.7 Good practice guides

Good practice guides were developed on the following subjects: “Measurement of Surface Activity and Mapping the Contamination”, “Radionuclide Scaling Factors (based on ISO 21238:2007)”, “Segregation Measurement of Solid Waste”, and “Free Release Measurement”. All of them are available on the project web site.

4.3 Early and potential impact

The project aims to enable nuclear site operators to characterise waste materials from the beginning of the waste disposal process to the end, through providing validated measurement techniques for measuring radioactivity on site, segregating and checking wastes and monitoring the condition of wastes.

The uptake of project's outputs will build confidence in the decommissioning process, and will lead to commercialisation of the measurement techniques. Examples of uptake include:

- The partner and nuclear specialist company NUVIA made an investment in a prototype of a new measurement system for radioactive waste pre-selection and free release designed on the basis of project results. The system was constructed, implemented on decommissioning site and validated during measurement campaign.
- Developed facilities and methods for the metrological characterisation of distributed temperature sensing systems based on optical fibres are utilised by Electricity of France (EDF) and French National Radioactive Waste Management Agency (ANDRA).
- Investigated and validated performances of the prototype gamma camera GAMPIX were utilised by the multinational company CANBERRA for construction and upgrade of industrial gamma camera iPIX.
- Agreement with the company LabLogic Systems Ltd. concerning development of an integrated bubbler/liquid scintillation counter for monitoring of repositories was made. Several prototype systems have been sold for monitoring of 3H stack emissions.
- Cryogenic system for CO2 trapping was designed on the basis of cooperation with the multinational company Air Liquide.
- Air flow calorimeter prototype for the measurement of thermal power generated by large radioactive waste packages was developed in cooperation with the French National Radioactive Waste Management Agency (ANDRA).
- Negotiations about the uptake of the mid-infrared spectroscope prototype for on-line on-site measurements of radiocarbon emissions from nuclear power plants and repositories are on-going with a European company developing instrumentation for the nuclear sector.
- Radioanalytical companies SPEX SamplePrep, Agilent Technologies and Triskem International SAS are interested in the outcomes of the rapid and semi-automated radiochemical analysis procedures for less hazardous destructive measurements, as it offers them new applications and markets.

The uptake of the project results in the longer terms brings opportunities to cover initial needs:

- Opportunity for producers to launch a new generation of radioactive waste monitors with special shielding in the market.
- Opportunity for European decommissioning industry to improve the management of wastes generated during decommissioning, that can be free-released into the environment and recycled or safely consigned in repositories.
- Opportunity for nuclear regulators and investors to enhance public confidence in radioactive waste management during the decommissioning of nuclear facilities and in the long term operation of waste storage facilities.

4.4 Environmental, social and financial impact

The uptake of project results will lead to environmental, social and financial impacts:

- The implementation of improved throughput, accuracy, reliability and mobility of the measurements which are important for the decommissioning of nuclear facilities will result in more reliable decision making concerning safe release and safe disposal of radioactive wastes into the environment with reduction of costs.
- Implementation of new and more precise measurement methods and techniques will result in more effective protection of the general population and workers against undesirable additional radiation exposure and enhance public confidence in radioactive waste management during the decommissioning of nuclear facilities and in the long term operation of waste storage facilities. This will build public trust and acceptance of nuclear technologies and ultimately benefit EU citizens through an improved quality of life.

The project outcomes provide evidence that underpinned by innovative metrology, nuclear facilities decommissioning will be more safe and cost effective.

5 Website address and contact details.

A public website was open at the beginning of the project, and was kept updated with the main public deliverables and outcomes that end-users could access: http://www.decommissioning-emrp.eu/.

A stakeholder database via the web on-line registration was maintained. Most of the seventy stakeholders that were integrated into project were registered via web site. All the project deliverables were available to stakeholders via web site.

A partners' restricted area was also created, in order to give the possibility to share work documents and deliverables. A video conference system was set up and used for short meetings between JRP partners and with external collaborators.

The contact person for general questions about the project is Jiri Suran, CMI, (jsuran@cmi.cz).

The contact person for "Characterisation of materials present on decommissioning sites" is Simon Jerome NPL (simon.jerome@npl.co.uk).

The contact person for "Measurement facility for waste segregation" and "Implementation of free release measurement facility on a decommissioning site" is Petr Kovar, CMI (pkovar@cmi.cz).

The contact person for "Radioactive waste repositories monitoring" is Steven Bell, NPL (streven.bell@npl.co.uk).

The contact person for "Development of reference materials and standard sources" Dirk Arnold, PTB (dirk.arnold@ptb.de).

The contact person for "Creating impact" is Pierino De Felice, ENEA (Pierino.defelice@enea.it).

6 List of publications in the public domain

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