Instrumentation for decommissioning and dismantling of nuclear installation

Philippe Girones

Abstract: Today, facility cleanup, dismantling and decommissioning activities (D&D²) represent a mature industry. For CEA facilities in France, the results obtained on the Grenoble site[1] are proof of this, and technological advances such as the implementation of the Maestro system[2] confirm it. Process efficiency is based on mastering the three generic phases (high level functions) of any manufacturing industry: storage management, treatment (physical and radiological reduction) and then waste conditioning (product). The procedures have been deployed in complex situations and are adapted to meet the challenges of different technical configurations: reactor, research laboratories, spent fuel treatment plants, etc. The point in common is the management of a source term (Bq). Each step in a D&D² procedure is controlled depending on the evaluation of the radiological levels involved. Nuclear instrumentation is therefore a major function in the deployment and the mastery of D&D² processes.

During facility operation, most of the radiological and physical-chemical characterization equipment for materials at the end of the life cycle is concentrated on the waste packages. These devices are maintained during the D&D² phase. Further measurement and checking apparatus are distributed throughout the facility, from the components in their original positions to the storage of final packages. The functional and logical architecture of this instrumentation is in the form of an instrumentation diagram where the first apparatuses are adapted to characterizing the scenes to dismantle, then on to control cleanup and dismantling procedures, and at the end of the line, to demonstrate that the packages or the facility are free of radiological constraints.

Measurements of gamma and neutron rays dominate the non destructive analysis techniques. Gamma spectrometry is the reference technique in D& D². Its deployment has been facilitated by the arrival of detectors which function at ambient temperature (CdZnTe, LaBr3, etc.). For complex scenes, the gamma spectrum processing result is combined with techniques for the localization of concentration points (hot spots). The combined processing of measurement results is carried out by the use of digital methods for drawing up curves for the yield and the estimation of the uncertainty associated with the quantity of interest. Overall, the characterization of D&D² waste has reinforced the use of coupled methods: physical and chemical analysis, gamma spectrometry, gamma camera and neutron counting.

Key words: cleanup, dismantling, decommissioning, instrumentation, gamma spectrometry, absorbed dose, gamma camera

[1] http://www.cea.fr/Pages/domaines-recherche/energies/energie-nucleaire/demantelement-nucleaire-CEA.aspx
[2] http://www.cea.fr/presse/Documents/DP/2016/DP_Maestro.pdf

Introduction

The nuclear facility cleanup and dismantling industry is a manufacturing-type activity. Its two **products** are the packages and buildings “freed” from radioactivity. They are obtained based on the definition of a **process** which is then deployed as a **procedure**. Piloting the procedure is guaranteed by the use of measureable indicators.

The control model is expressed in the form of an **instrumentation diagram**. This diagram is structured based on the final result required by the dismantling industry: the reduction of the source term contained in the facility. It is divided into four functions: raw material inventory, control of the treatment processes, waste package characterization, and checks prior to decommissioning. This breakdown enables the necessary radiological analysis methods to be defined so that the D&D operations can be carried out in safe, efficient conditions.

A simplified model for the nuclear facility cleanup & dismantling industry based on a facility life cycle and stakeholder objectives

Each stage in the life cycle of a nuclear facility (ASN, Guide de l'ASN n°6 : Mise à l’arrêt définitif, démantèlement et déclassement des INB en France, 2015) is supported [operated] by players ensuring production under certain constraints. These concern safety, the management of wastes and their transport, and implicitly process control and control of their impact on the environment (Figure 1).

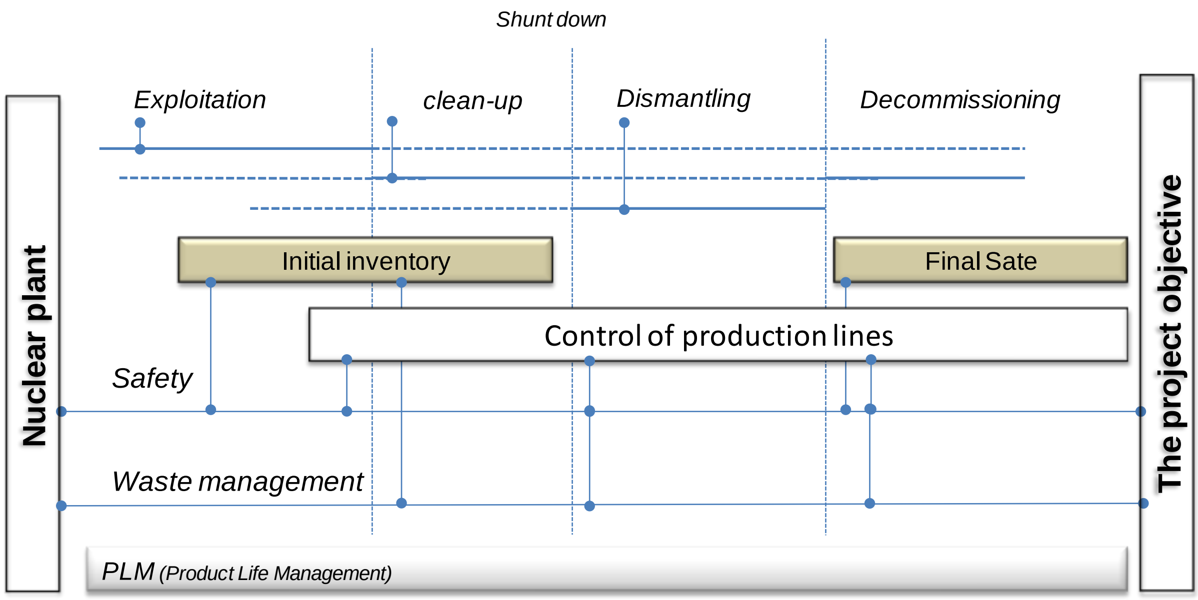

Figure 1: Simplified nuclear facility life cycle

The representative model for D&D which is *a priori* the most commonly admitted or explicit, represents nuclear facility cleanup and dismantling by a waste package production line. To quote from the paper “L’usine à colis de déchets nucléaires” (GIRONES P. , 2013):

> *The “nuclear waste package factory” is a procedure which transforms in piece of industrial equipment with no further use and recognized as nuclear waste into a waste package compatible with the requirements for storage under safe, secure conditions.*

The products of this factory are a set of waste packages and a facility or a site where, in the best case, the constraints due to radioactivity have been withdrawn (ASN, Guide de l'ASN N°14 : Mise à l'arrêt définitif, démantèlement et déclassement des installations nucléaires de base en France, 2015). This model has the advantage of respecting the scheduling for the instruction of files sent to the Nuclear Safety Authority and meets the recommendations for the preparation of a cleanup-dismantling scenario (NUCADVISOR, 2015). Then during the operational phase it also meets the need to control waste management processes (Andra, 2013).

D&D² : A final conclusion expressed by an equation for process checking and control

Cleanup and dismantling process control unites numerous professions and actions. There are many performance indicators which depend on the stakeholders, waste management, safety and security, production, etc. IAEA (AIEA, 2011). However, instrumentation is used to meet the needs of process control, i.e. to ensure the production of quality packages under safe, secure conditions.
Nuclear instrumentation for processes and the associated analysis techniques rely on an evaluation of physical and physic-chemical quantities. The act of transforming a factory or a facility into waste packages, followed by its decommissioning, is thus represented by a function called Objectives (Equation 1) which represents the evolution of constraints as time passes: $f(t, \dot{D}, A, m_f)$

where :

t : time (project: milestones)
A : source of contamination which imposes the wearing of protective clothing and equipment,
$\dot{D}$ *: term in dose rate, the consequence of accumulated radioactive sources (gamma, neutron, secondary source, X),*
m_f*: term grouping radionuclides emitting neutrons by spontaneous fission or by neutron production through nuclear reaction, e.g. : (α, n) and the fissile radionuclides.*

In practice, operational behavior (trades/professions) can be separated into three risks: *contamination, irradiation and criticality risk (decision variables).* The equation can however be reduced to a single term responsible for all the constraints (dose rate, contamination, criticality risk), i.e. **the activity expressed in Bq.** From this comes a time function: $f(t) = A \; with \; A \in [A_0, A_c]$. A_0 is the activity evaluated in the inventory phase and A_c the activity targeted. This is the basic information when writing the scenario (definition of the sequences for term source reduction). The dose is the consequence of radionuclide activity which, for example, undergoes gamma transitions; the fissile nuclei are radioactive and the neutron source is the consequence of spontaneous fission or of reaction (α, n). Managing sub-criticality during dismantling operations is mainly ensured by controlling the fissile mass ($A = \lambda.N$, with A the activity in Bq, N the number of nuclei, λ radioactive constant in s^{-1}). It is therefore a matter of observing the term source evolution: A, depending on time, in order to minimize the activity level (Bq), by determining the absolute minimum.

$$f(t_{k+1}) < f(t_k) \; or \; ever \; A_{k+1} < A_k \text{ , Equation 1}$$

There are two broad strategies arising from this principle, depending on the quality of the radionuclides. The first of them consists in proposing deferred dismantling operations for facilities where the radionuclides have a short radioactive half-life (LECOCQ, 1999). This principle is not applied when dealing with radionuclides whose radioactive half-life is longer than 30 years, as is for example the case for the equipment in spent fuel reprocessing facilities (TALLEC & KUS, 2010). There the second strategy, involving immediate dismantling, is often decided on in order to reduce the loss of valuable knowledge and know-how the personnel has accumulated during the facility's operation phase.

Whichever strategy is retained, the deployment of an instrumentation diagram is imperative to monitor and control the term source reduction. Appropriate instrumentation is therefore indispensable for each phase of D&D operations. Thus the quality of each apparatus and the structuring, or instrumentation diagram, must be defined.

An ideal functional architecture for a nuclear facility activity reduction process

The flow sheet for the reduction of the three constraints (*contamination, irradiation and criticality risk*) is deployed in order to control the term source (Bq) and is concluded in the best case by the total removal of the radiological constraint (**Figure 2**).

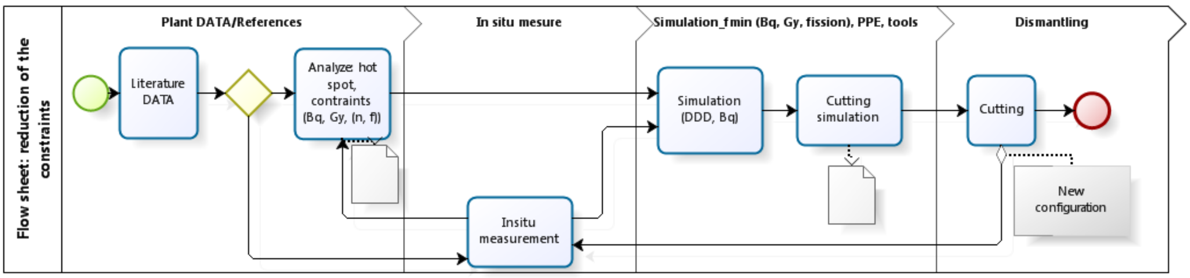

Figure 2: Simplified flow sheet for the minimization of constraint parameters, or operational functional architecture of the reduction function

The optimization procedure is first based on a radiological simulation (THEVENON, 2009) which leads, if the level of data is sufficient, to the identification of a plan for deconstruction and cleanup (scenario). If this is not the case, starting from an inventory of the deficits in radiological and physical-chemical data, a campaign of *in situ* measurements is undertaken (GIRONES P. , 2013) (GIRONES & al., Underwater Radiological Characterization of a Reactor Vessel, 2005). The consolidated data is then placed in an iterative loop (**Figure 2**) where the measurement results are the performance indicators for the source term reduction function (GHIBAN & Al., 2016).

Simplified functional block flow sheet for a D&D process, and instrumentation functions

Source term reduction is the "driving force" of cleanup and dismantling. It can include a division of the operations into phases formalized by a functional block flow sheet, useful as a base for a general model: "The waste package factory". The focus of waste packaging, or conditioning, involves the physical size and radiological reduction, and the sorting of different contaminated materials (treatment). This process core is receives retrieved legacy waste packages (Waste Retrieval and Conditioning), as well as fluids or contaminated facility equipment (storage). Waste conditioning in a final container (**Figure 3**) is the objective of package production and the decommissioning or release of the site is the second product.

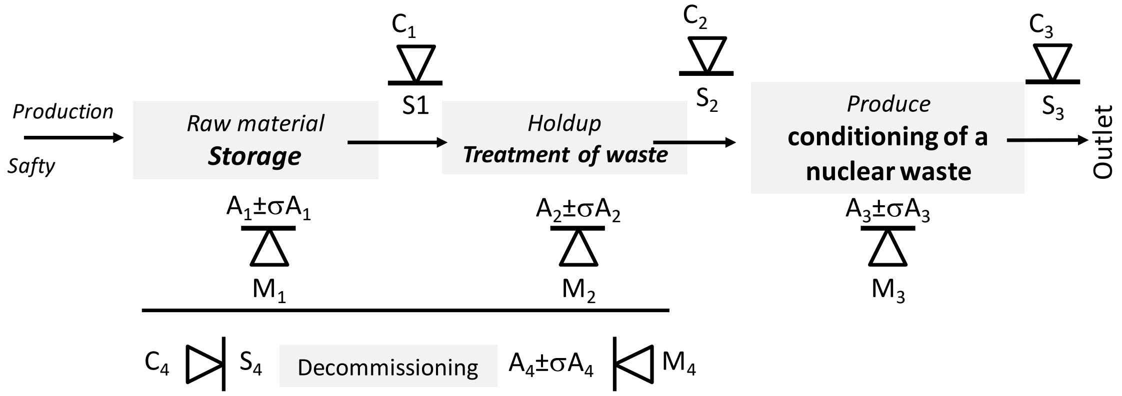

Figure 3: Functional block flow chart for waste treatment in three phases - Ax: activity in Bq, Mx: measurement point, C1: check/control point, Sx: threshold in Bq)

There are two types of instrumented points: control (monitoring) (Cx) or measurement (Mx). They meet two types of requirements: **product quality or safety/security**. Control is a means which can be distinguished from measurement by the quality or the use made of the instrumentation. A threshold (S_x) is the indicator for control. For the measurement, a value resulting from a measurement act is necessary. This value has an uncertainty associated with the quantity of interest ($A_x \pm \sigma A_x$).

From the models and principles (**Erreur ! Source du renvoi introuvable.**, **Figure 3**), a functional categorization of the measurement methods and technologies for the controls and measurements is obtained as necessary for cleanup, dismantling and decommissioning (**Table 1**). This structures the document, to prepare for the future actions.

N°	*Objective*	*Result*
F1	Carry out the spare parts inventory	Initial state characterization
F2	Instrument a process (package production, cleanup)	On line follow-up: from the spare part to the waste package
F3	Characterize the waste packages	Supply data to guarantee product quality (for shipping, waste route destination)
F4	Release a nuclear site (building, land)	Demonstrate the absence of impact ; final state characterization

Table 1: Example of categorization of techniques and analysis methods for cleanup, dismantling and decommissioning

Preparation for a spare parts inventory, or initial state of the facility covered by function F1

In this phase, the source term evaluation is carried out based on analyses of bibliographical and technical data, digital simulation, *in situ* analysis methods (TOUBON, CORDIER, & FERET, 2007) and laboratory work on samples collected. The work begins with the collection and analysis of bibliographical data (operation, Figure 1) (GIRONES P. , First report from an advanced radiological inventory for a spent fuel reprocessing plant, 2013) (DAVIKDO, 1999). It then continues with the addition of results to a data base (DEVAUX, 2010). This is the inventory stage for the radionuclides. More specifically, the radio-tracers are identified.

There are many methodological and technological initiatives in this field. They "surf" on the growing digitalization of the industry, but no universal solution has been found so far. However tools are becoming more specialized while respecting information management domains, distinguishing the means of management IAEA (AIEA, Information Technology for Nuclear Power Plant Configuration Management IAEA - TECDOC - 1651 , 2010) (GIRONES, et al., 2006), from the means of data processing [augmented reality] (CHABAL, 2011) (HIRON, DIOP, & SUTEAU, 2003).

Bibliographical data processing and structuring enables the next step, that of detecting data deficits (GIRONES P. , Perspective d'une solution intégrée pour une maîtrise d'ouvrage, 2012). *In situ* radiological and physical-chemical mapping data acquisition is then undertaken. Mapping gives a model of the facility with information about where the concentration points (hot spots) have been found. To carry out these operations, source location techniques are deployed.

An inventory of imagers to locate the source (function F1)

There are basically two types of source location techniques: absorbed dose or energy fluence rate, and the use of imagers. In this field, gamma imagers first equipped with scintillation detectors and then with an optical pinhole (CARCREFF, 1995) have been used (LEGOALLER & IMBARD, 1998). Over the last 10 years, there have been considerable technical evolutions, the first including the use of coded masks (GAL, 2006). The gains in yield and spatial resolution (**Figure 4**) are significant compared to the pinhole.

Figure 4: Example of gamma images (Aladin imager equipped with a CSi detector) – on the left, pinhole optics, on the right coded optical mask. The situation shown is 241Am contamination in a glove box containing a nuclear fuel pellet press.

The scintillation detector has now been superseded by CdTe type solid detectors (LEMAIRE, 2014). The contribution from these new detectors in fundamental, as it is now possible to collect energetic data (spectra). Recently, new-generation cameras using the Compton scattering (WAHL, 2014) have been tested in nuclear facilities (Table 2). The first results seem to be convincing.

Equipment reference	Technology	References
Coll@gam	Dose (Si) or energetic fluence rate (CdZnTe) equipped with a collimator	(GIRONES P. , 2010)

Equipment reference	**Technology**	**References**
Maunela	Dose rate, collimation, GPS	
Radscan 600 and 800	NaI[Tl], scintillator variable collimation	(SANTO & RANDY, 2006)
Aladin	CsI, BGo scintillator with pinhole then coded mask	(LEGOALLER & IMBARD, 1998)
Cartogam	CsI, BGo scintillator with pinhole	(CHIRON, 2008)
IpIX	CdTe, coded mask	(CANBERRA, 2015)
Polaris _ Compton	CdTe, Compton	(WAHL, 2014)

Table 2: An inventory of instruments for the location of hot spots contaminated by gamma emitters

Imagers which reveal the distribution of alpha contamination (LAMADIE, 2005) are also currently being industrialized, and neutron imager technologies which may have other prospects are under study (GAMAGE, 2015).

The deployment of this type of equipment requires coupling with instruments or additional acquisition methods: vision, telemetry, gamma spectrometry, etc. These additional systems should collect at least dimensional data (position of the probe in space), an energetic fluence rate, an absorbed dose, and/or a visible image. Together with the cost, this is probably what limits the use of this type of equipment. Today, modules grouping all the instruments (**Figure 5**) are deployed in highly contaminated environments (GIRONES, 2006). They are often associated with a robotized platform (DUCROS, HAUSER, MAHJOUBI, & GIRONES, 2016).

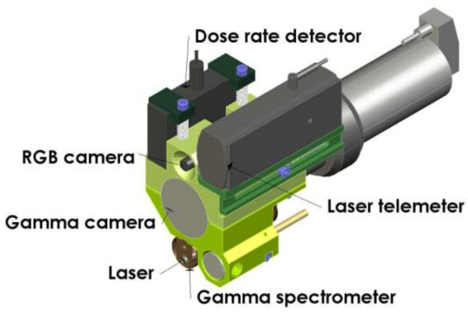

Figure 5: Example of a module grouping instruments for the location and radiological characterization of hot spots (DUCROS, HAUSER, Majhoubi, & GIRONES, 2016)

Source spatialization is the basic information. This is competed by an evaluation of the activity for each concentration point (hot spot).

***In situ* gamma spectrometry for hot spot radiological characterization (function F1)**

Most *in situ* non destructive radiological characterizations are carried out based on gamma spectrum processing. The use *in situ* of the gamma spectrometry analysis technique has had a considerable rise in popularity due to the development of detectors equipped with CdZnTe crystals, as have $LaBr_3$ scintillators. These detectors function at ambient temperature, and their small size is compatible with the constraints imposed by *in situ* measurement. Their resolution quality is sufficient for the acquisition and a quantitative processing in the majority of cases.

Detector type	Crystal type, technical data	Observations/implementation
Scintillator	NaI[Tl], LaBr3, Volumes: 3x3 inches, 5x5 inches	Ambiant temperature
Semi-conductor: CdZnte	Volume: 0.1 mm^3 to 4000 mm^3	Ambiant temperature, small dimensions
Semi-conductor: GeHp	Volumes and shapes depending on the use required, favors resolution or yield or even counting rate	Liquid nitrogen or electrical cooling

Table 3: An inventory of gamma spectrometry detectors used for in situ radiological characterization during the inventory phase _ F1

The detectors are placed in collimators which favor the signal/noise ratio by reducing the sold angle at the zone of interest containing the source to be characterized, thus decreasing the effect of parasite sources (outside the solid angle). The detection system is therefore a couple: detector, plus collimation. There are detector type selection methods (technology). The criteria include: counting rate, scene accessibility, and energy resolution. The measurement of activation products for a component under flux: reactor vessel, control rods, etc., leads to the selection of a small-sized, heavily collimated detector, (LAMADIE & GIRONES, 2007). The CdZNte technology has been retained for the characterization of components under flux, i.e. the vessel, and GeHp for equipment characterized in a special dedicated cell. However, when complex spectra need to be characterized, for example industrial components containing actinides (uranium, plutonium, etc.), the energy resolution of the detectors is the de selection criterion (**Table 4**).

Radioelements in majority (energy dynamics)	Detector types	Comments
E > 500 keV	Coaxial Ge At least 20% efficiency	Case when measuring FP or activation products (AP)
60 keV < E < 600 keV	Low Energy Ge Width at mid-height lower than 700 eV at 121 keV	Case when measuring plutonium
60 keV < E < 3000 keV	Hybrid, Broad Energy Ge Width at mid-height lower than 700 eV at 121 keV	Case when measuring uranium

Table 4: Inventory of detector technologies – Energy dynamics

Gamma spectrum processing is carried out using either a relative or absolute method. There are many relative processing possibilities (Table 5), and relative processing is systematically applied for complex spectra where the presence of actinides is known. The result is an isotopy expression for plutonium or of the enrichment for uranium, completed by the content in certain radionuclides resulting from radioactive decay like ^{241}Am or in fission products like ^{137}Cs.

Software package	Reference	Comments
MGA Pu	(GUNNINK R. , 1990)	Sensitive to the presence of fission and activation products (Compton diffusion)
MGA U	(GUNNINK, 1994) (BERLIZOV, 2007)	Functioning range: enrichment from 3% to 20%
FRAM Isotopic	(SAMPSON, 1995)	Fixed-energy Response-function Analysis with Multiple efficiencies
IGA	(SIMON, 2011)	Rich interface
TRIFID	(BONNER, 1994)	No feedback to date

Table 5: An inventory of gamma spectrum processing software packages for the expression of plutonium isotopy and of uranium enrichment

These software packages do not need yield calibration. The relative processing of significant peaks, after self-calibration, is carried out by extrapolation of correction functions for the detector response, for self-absorption in the source and the attenuation of the screens. Absolute processing requires a yield calibration. Two calibration methods are used, the first of which is called relative. It works with samples whose shape and radionuclide quality are identical to the unknown source. In the case of *in situ* measurements, the quality of the sources and of the industrial equipment containing the source can vary widely. For the second method, called absolute, most calibrations are carried out by digital methods.

The principle of these digital calibration methods is simple. It consists in dividing the yield $\varepsilon(E)$ into two terms:

- A term dependent on the detector, K(E), the detector's intrinsic yield considering the detector volume reduced to a point placed in the center (cm^2),
- A term dependent on the geometry and the quality of the component to measure, and on the shape of the radioactive source: TF(E), the transfer function (cm^{-2}).

$$\varepsilon(E)=FT(E).K(E)$$, Equation 2

The term dependent on the detector is determined experimentally using standard point sources positioned at a distance from the detector, or by digital modeling then simulation. These drawing methods for the yield function are integrated in the gamma spectrometry software (CANBERA, 2010) or autonomous methods can be used based on a deterministic code or Monte-Carlo.

Conclusion on analysis methods to obtain the initial radiological state information (function F1)

The form of the result from this phase is not standardized, and work on standardizing this representation is currently running. Software packages dedicated to scenario preparation may well meet the need (OREKASOLUTIONS).

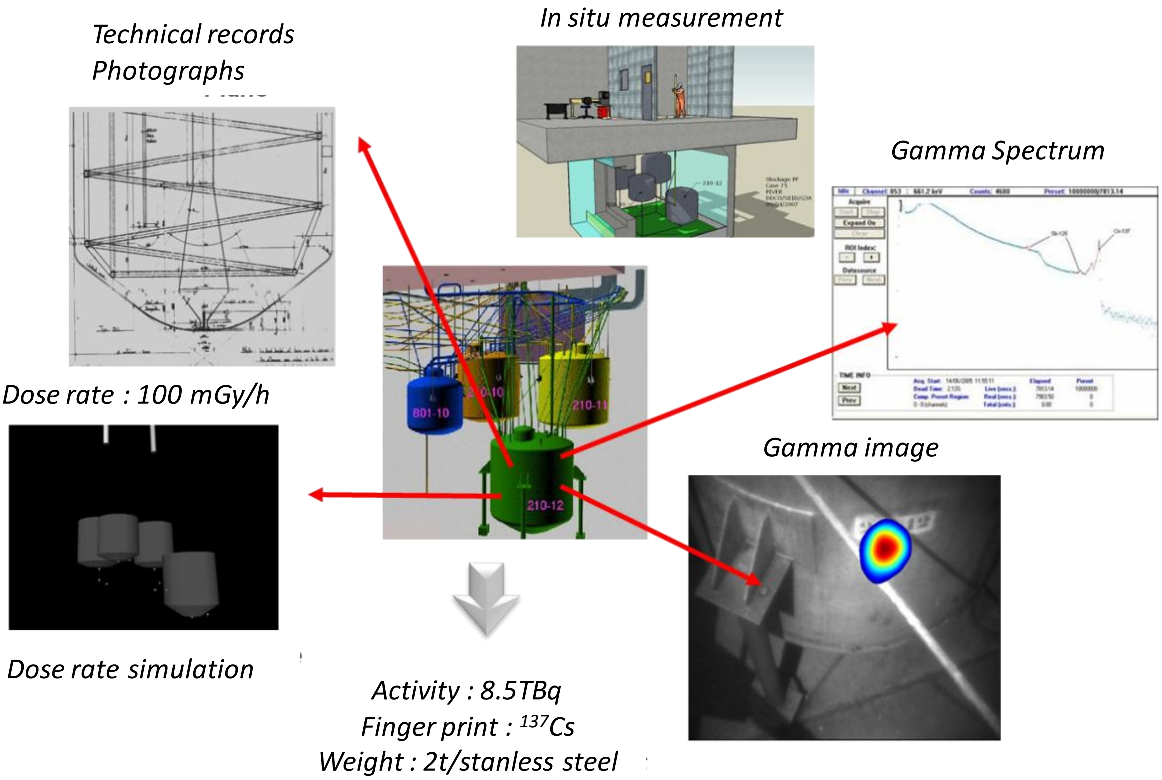

Figure 6: An example representing the inventory collection and processing of radiological data

Instrumentation in a process for cleanup, deconstruction, resizing or conditioning (function F2)

At this stage the radiological quality of the raw material is understood and controlled. The process implemented meets of source term reduction objectives for the facility.

The cleanup operations are monitored by destructive analysis methods. To indicate the changes to the quality of the location where the component being cleaned up is situated, these analyses are often completed by *in situ* collection methods. Among these, dose rate measurements are the most common. A typical case consists in ensuring the monitoring of a cleanup operation involving fission products. A modular collimator has been developed to ensure a spatial selectivity of the source. There is homogeneous radial protection, composed of rings in dense material with the detector placed in the center. Axial protection comes from a combination of the ring stacking, organized depending on the distribution of sources within the space. It may be asymmetric if most of the parasite source term is highly localized, i.e. concentrated in the space.

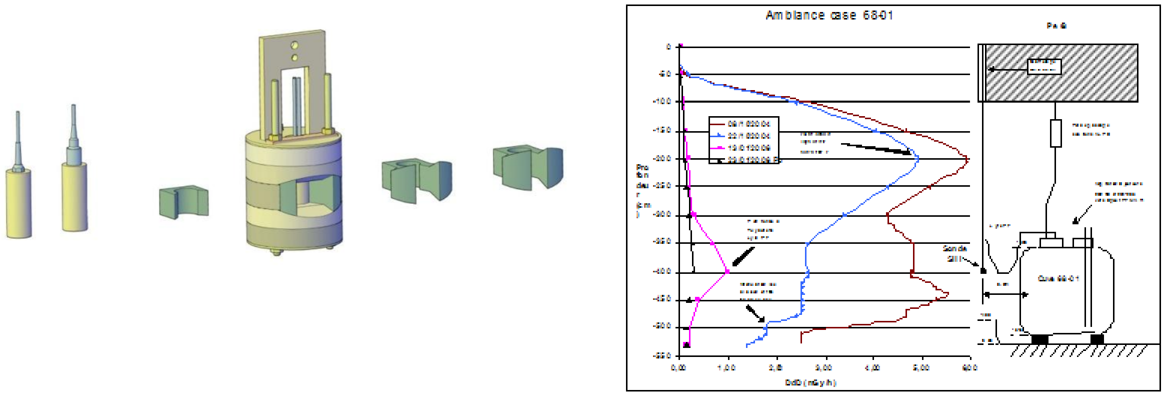

Figure 7: Example of a modular collimation used to monitor component cleanup within multi-source scenes (GIRONES P. , 2009)

This equipment has, for example, enabled the movement of a major hot spot from the bottom of a tank towards an external concentration point to be seen, and to follow the source term evolution in the scene (GIRONES, 2006). The cleanup objective is to allow human intervention while respecting the ALARA principle. These direct reading techniques (without interpretation) are simple and safe, requiring no complex means other than the systems (carrier) needed to set the heavy equipment up within the scene.

Configurations which are more complex to monitor can be encountered. These are situations where measurement must meet control objectives for processes involving nuclear matter handling (LE BARS & GALET, 2010). The procedure must meet the need to control sub-criticality by fissile matter mass, monitor the product quality (the waste package) and control the accumulation or decontamination factor for the equipment which had contained fissile matter. To evaluate the Decontamination Factor (DF) or the Accumulation Factor (AF), the gamma spectrometry analysis technique is used. Processing gamma spectrometry results is based on a relative processing of the line surfaces which are significant and characteristic of the radionuclides present in the source. From this comes a source term evolution indicator (Decontamination Factor DF, or Accumulation Factor AF) obtained by the sequential processing of the significant line surfaces.

$$DF\_or\_AF_{(1,2)E_i} = \frac{N_{1E_i}}{N_{2E_i}}$$, Equation 3

N_{1Ei} is the result of counting at time t_1, and N_{2Ei}, is the result of counting at $t_2 > t_1$, considering a source term evolution between t_1 and t_2.

A CdZnTe detector is placed in a protection (collimation) which limits the solid angle and favors the signal. The unit with the detector in the collimator is placed into the glove box with an apparatus which does not modify the statistical containment: glove box and gloves or sleeves.

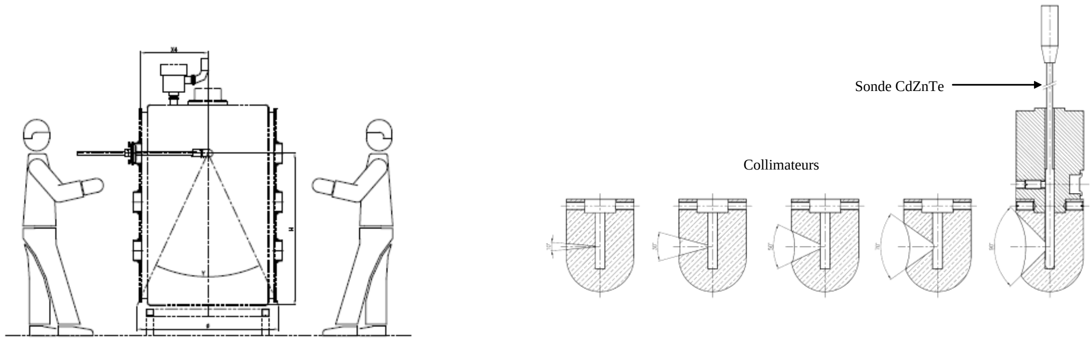

Figure 8: Positioning system for the detector in a glove box - Collimators with solid angles from 10° to 90° (BRENNEIS & GIRONES, 2011)

Trials have shown the ability of this type of equipment to meet the needs of production line control, health control (radiation protection) and sub-criticality control. These first two examples show that controlling the gamma energy fluence or measuring absorbed dose coupled with collimation are techniques which are simple to implement for process monitoring (Figure 9).

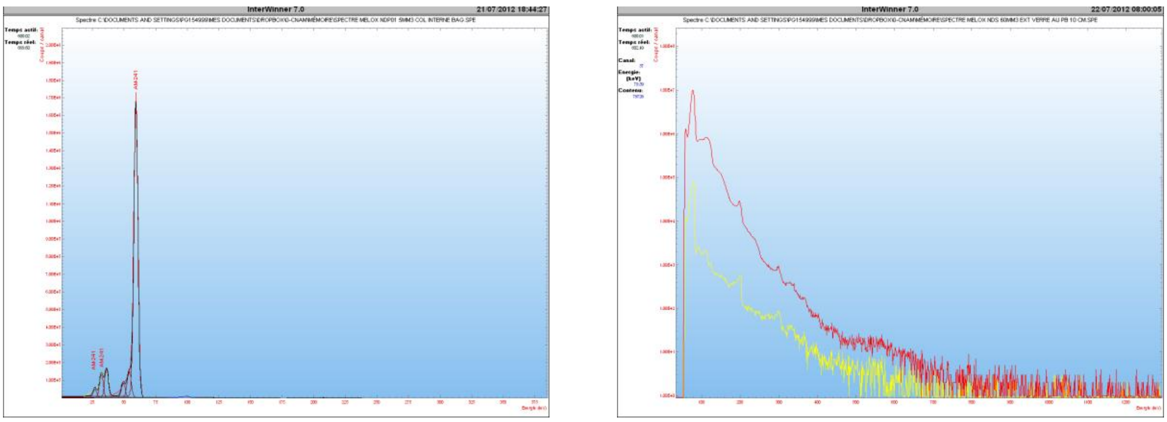

Figure 9: (left) 60 keV zone of the gamma spectrum, Zone: 90-105 keV _ (right) Red spectrum from inside the glove box and yellow, from the exterior protected by lead glass

The qualities of products or the attitudes related to safety also depend on the physical and chemical quality of the matter involved. In addition to destructive characterization methods, it is therefore essential to deploy *in situ* non destructive technologies. Analysis methods such as LIBS (Laser Induced Breakdown Spectroscopy) are mature (HAHN & OMENETTO, 2012) and some already meet D&D objectives (MAURY, SIRVEN, & TABARANT, 2013), while others still require nuclearization work, like FXL.

Conclusion: Instrumentation in a cleanup-dismantling process _F2

To carry out a D&D process safely and efficiently, it is necessary to run it based on measurable indicators: the activity (Bq) and the physical quality of the raw material. The optimization loop for constraint reduction starts with product control, based on measurement results. This is followed by modeling to estimate the radiological evolution of the scene. The plan for the deconstruction and removal of sources is governed by risk reduction, for which dose rate is the most direct indicator. Processing with radiation protection calculation codes enables the deconstruction plan to be validated. Deconstruction simulations in a virtual reality immersive room, for example, can validate the operation's feasibility by viewing the accessibility of the equipment to be removed. During the operation and then the total removal of the sources targeted, monitoring guarantees operation efficiency follow-up. Instrumentation plays an essential role for the control of this loop, with the activity value as the pivot (**Figure 2**).

Inventory of methods and instruments for the radiological characterization of waste packages (function F3)

The cleanup and dismantling of nuclear facilities represents a special case for nuclear waste management. The volume, i.e. quantity, and the diversity are two parameters which make the production of waste packages in the D&D phase a particular challenge. Characterization instruments and methods are based on the practices and means implemented during the facility's operation phase (**Figure 1**). However, different issues have encouraged the development of industrial systems which integrate all the means to ensure the acquisition, processing and traceability of the data.

Characterization methods for waste packages are often structured base on a combination of results obtained from several techniques of destructive and non destructive analyses, data and calculations (Figure 10).

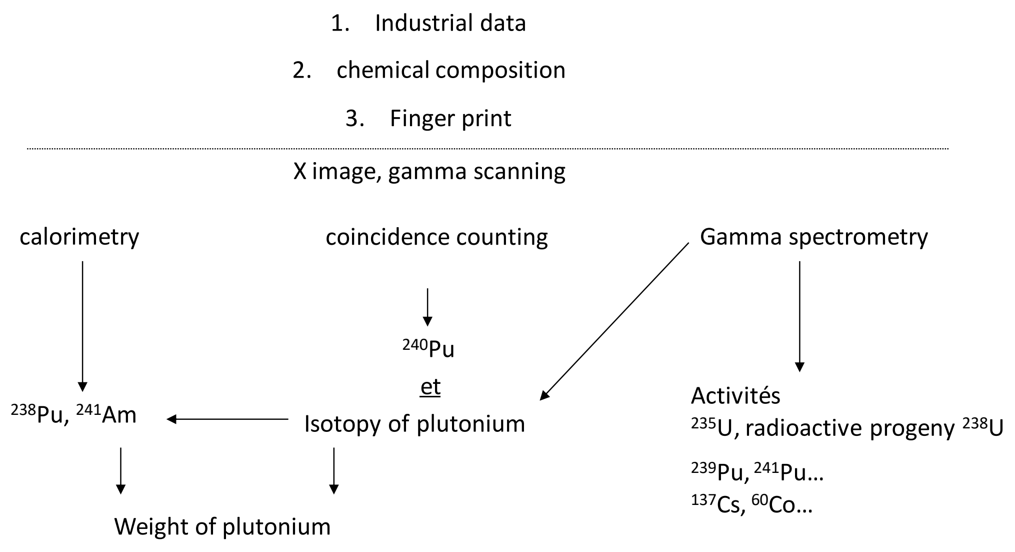

Figure 10: Flow sheet of non destructive analysis method combinations (process control: package)

Different results are used, even though identifications of the radionuclides and the activity are enough to meet the requirements for radioactive waste management. The analysis techniques do however have performances in common, expressed from:

- The measurement dynamics (energy, energy fluence rate),
- The expression of the detection limit,
- The expression of an uncertainty associated with the quantity of interest.

In this field, as in the inventory phase (F1), gamma spectrometry is the main technique used. However the configuration of the stations is different from that of the inventory phase (F1). The source, now the waste package, is placed in an environment where the background noise is controlled, and the content is, *a priori,* known. The operation therefore consists in characterizing a radio-tracer which will be attached to or associated with a typical spectrum, divided into three groups (**Table 6**).

Radionuclide group	Descriptions
Radio-tracer	Detectable: statistical accuracy compatible with the process – counting time, Uncertainty associated with the activity is controlled, Radio-tracer penalizing elements ratio and fissile material are reliable – variability is controlled
Penalizing radionuclides	High radio-toxicity, difficult to detect by non-destructive methods or undetectable: for example, ^{36}Cl
Fissile radionuclides	All the fissile radionuclides involved

Table 6: Division of the typical spectrum into three radionuclide groups

The radio-tracer is the target for non-destructive methods, and its characterization is carried out based on a system. Such systems integrate the three basic functions for waste package characterization by meeting the needs of an industrial approach: acquisition, processing and traceability. The industrial aspect means the ability to be maintained in operational conditions and to guarantee traceability for the data and the processing actions (Figure 11). There is therefore a digital "spine" in the form of a unique data base and of acquisition modules and processing (BUTEZ, 2011).

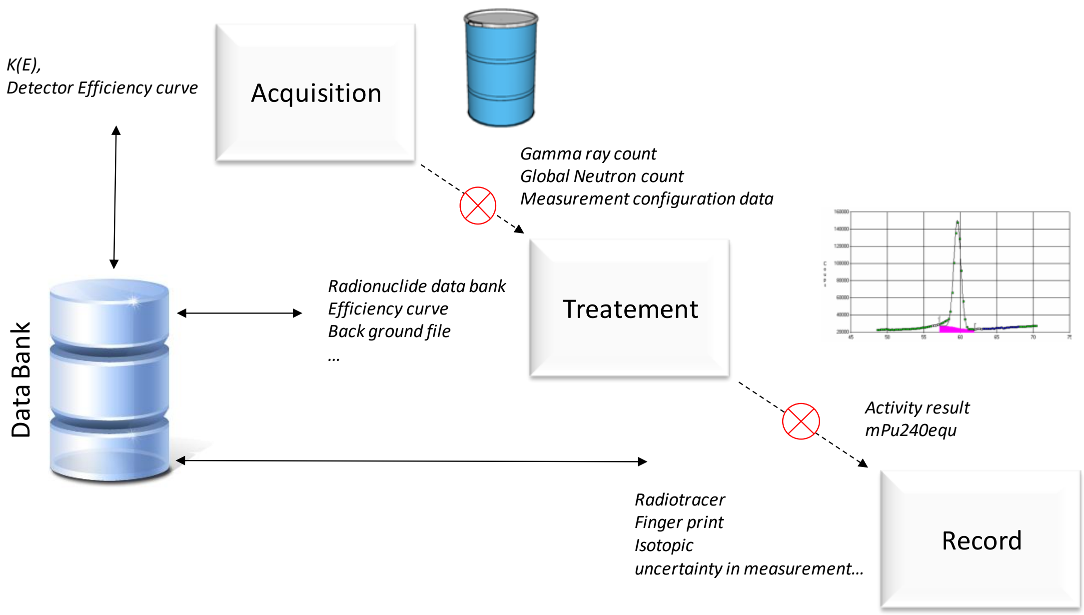

Figure 11: Functional flow sheet of an Information System (IS) for nuclear waste package management in the CEA

Based on this functional model, the standardization of acquisition, processing and declaration practices has been facilitated. Going further, characterization systems related to requirements set up by the facility and waste route operators have been standardized in order to simplify connections with the Information System (IS). More complex characterization systems can be found, combining and linking together radiological characterization and imager techniques (GOSSEN, DEYGLUN, & GIRONES, 2015) (JALLU, 2011): gamma spectrometry, X-ray imagery, passive neutron counting, and calorimetry.

The combination of analysis techniques is motivated by the ability to formalize and reduce the uncertainty associated with a quantity of interest. In most cases, this objective has been reached. (LAMADIE & DEVENELLE, 2003).

This performance is all the more complex to formalize for cleanup and dismantling operations where it is necessary to characterize waste packages *in situ*. In this case, two issues arise. The first concerns the radio-tracer validity, and the second the expression of the uncertainty associated with the quantity of interest (Activity or mass of fissile matter). The methods are therefore complex, and require the development of special means for the acquisition, the data processing and the evaluation of the associated uncertainties. The method is then comparable to that deployed in the inventory phase (F1).

Conclusion: Characterizing a nuclear waste package (function F3)

The development of characterization techniques or systems for nuclear waste packages is carried forward by the cleanup and dismantling industry. New technical requirements are continually appearing, and support the deployment of systems at work stations instead of only one measurement method. This development has been facilitated by data structuring imposed by information systems, among which CARAIBES is the most advanced in France. This structuring has also enabled support for a standardization approach for acquisition and processing systems, which ensures a mature industrial dimension. Finally, this structuring imposed by the IS is also the vector or the support for the integration of new characterization systems, among which calorimetry is a representative.

Low level measurements applied to large surfaces for site decommissioning or release (Function F4)

The second deliverable for the cleanup and dismantling industry is a building or a site where the radiological constraints have been reduced to the lowest possible level, ideally the total removal of radiological constraints. In this case, the facility or the site can be decommissioned. The French Nuclear Safety Authority Guide 14 (ASN, Guide de l'ASN N°14 : Mise à l'arrêt définitif, démantèlement et déclassement des installations nucléaires de base en France, 2015) specifies the regulatory context and the principles for a complete cleanup, following the use of independent successive lines of defense. To meet the requirements of these lines of defense, which are used systematically whatever the radiological level targeted for the cleanup site, instrumentation must be implemented for the structure cleanup steps (floors, walls).

Preparing the lines of defense begins with the definition of a diffusion model for the contaminant in the substrate (concrete). The model definition is based on the historical and functional analysis of the facility described in the first part of this document. The second line of defense involves creating a surface mapping, from which concentration points (hot spots) can be identified. Finally, results of the two lines of defense are combined to make three-dimensional mappings. Destructive analyses are often added to this final step to validate the migration within the substrate by measurements. The destructive characterization campaign works to a sampling plan which is based on optimization methods like geostatistics (JEANNEE & FAUCHEUX, 2013).

Instrumentation for cleanup monitoring on building structures based on surface mapping

Classical instrument techniques are used for structure characterization: gamma spectrometry and charged particle counting. The zone probing technique limits when using radiation protection equipment are also classical, the first limit being the quality of the support. It is tricky to measure alpha contamination deposited on a concrete floor, for example. The second limit is of a practical nature, as the supports must be accessible (Figure 12).

Figure 12: (left) Example of a CoMo measurement apparatus for contamination measurement, equipped with a plastic scintillator and Zns for the detection of □□and□□ particles and gamma rays, (right) Example of a duct to be checked, containing debris and cables.

With reserves as to measurement feasibility, the parameters for the detector selection techniques include the quality of the radiation detected, the yield, the movement itself, and the decision threshold. Then, depending on the implementation method, a detection limit is calculated.

Equipment	Probe	Probe surface (cm²)		Background level (imp/s)	decision threshold	Detection limit	
						Probe Bq/cm²	smear Bq/cm²
	SA 70.2/SMIA	30	α	<0.2	0.5	0.2	0.4
	SB70.2/SMIB	30	β	<2	1.2	0.4	0.4
	SAP 400-2	386	α	<0.2	0.5	0.8	0.8
	SBM/SIBM	6	α	<2	1.2	3.0	1.2
	SBG/SMIBG	50	β	<3	1.2	1.6	2.7
	SBS	75	β	<7	1.5	0.4	2.1
	SG2/SMIG	8	γ	<40	9	36	9.5

Equipment	Probe	Probe		Background	decision	Detection limit	
	SBM2D	30	β	<2	1.2	0.8	1.6
	SMIX/SX2	8	X	<20	7	31	8.1

Table 7: Extract from a technical indicator synthesis table for contamination measurement probes (MONTREUIL, 2010)

For gamma spectrometry, the choice of the detector is important when the background noise is controlled. Two technologies meet the implementation requirements for the radiological characterization of large surfaces: scintillators, with their high yield, and detectors equipped with GeHp crystals (GIRONES P. , 2016), (Figure 13).

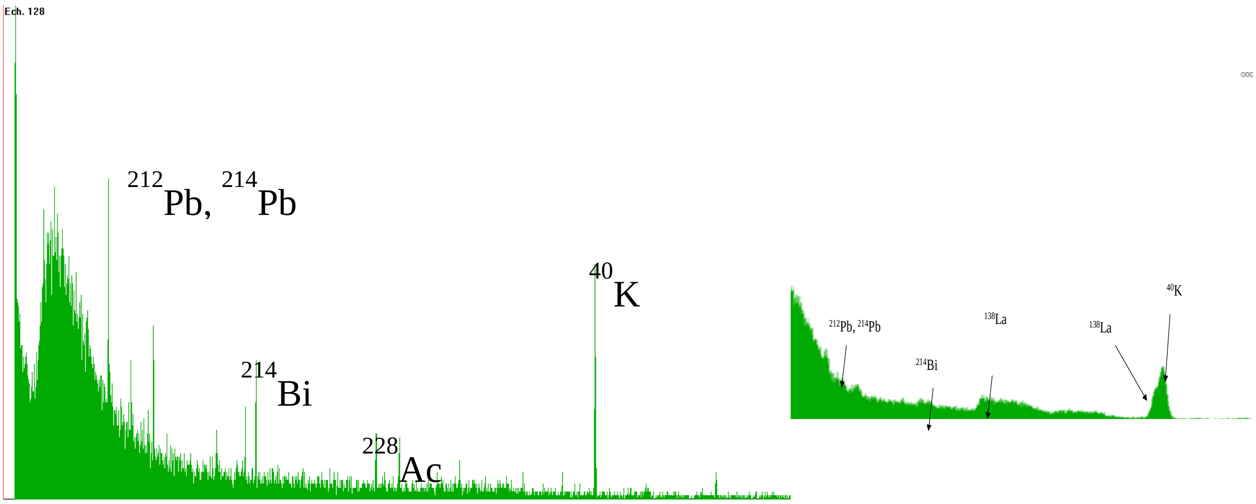

Figure 13: (left) gamma spectrum of a concrete surface - GeHp detector, 40% relative yield; (right) gamma spectrum of a concrete surface with a LaBr3 3x3 inch detector.

The implementation advantages of scintillator detectors (NaI[Tl], $LaBr_3$) – ambient temperature functioning, apparatus mass, price, processing electronic simplicity – have encouraged operators to use this type of equipment. Results show that the detection limits obtained under identical acquisition conditions favor the use of a detector equipped with a GeHp crystal (Table 8).

Energy (keV)	LD $Labr_3$ ($Bq.g^{-1}$)	LD GeHP($Bq.g^{-1}$)	Difference
121.8	0.162	0.070	57%
244.7	0.471	0.200	57%
344.3	0.112	0.044	61%
661.6	0.039	0.014	64%
778.9	0.338	0.083	75%
964.1	0.341	0.103	70%
1408	0.247	0.063	75%

Table 8: Expression of the detection limit in a concrete support measurement situation with two detectors, GeHp and LaBr3, with a 600 s counting time - identical measurement configuration.

For a decommissioning approval threshold of 0.1 Bq.g^{-1} for the radionuclides ^{60}Co, ^{152}Eu, ^{137}Cs, 70% of the surfaces are checked with GeHP detectors, compared to 15% for $Labr_3$. The expression of the detection limit enables the identification of the two important parameters when choosing the detector type (LEPY, 2008), i.e. noise level and resolution.

$$DL = \left(\frac{8{,}76}{t \, . \, TF(E) \, . \, K(E) \, . \, \Gamma(E)} \right) . \sqrt{FWHM \, . \, B(E)}$$ Equation 4

With: t counting time in seconds, TF(E) transfer function (cm^{-2}), K(E) intrinsic yield of the detector, Γ(E) branching ratio (%), FWHM (Full Width at Half Maximum, or width at half-height) is the resolution (in channels), B the noise (content of a channel in the region corresponding to the energy E).

As noise and yield are not differentiating parameters, resolution is therefore the technical indicator when choosing a detector. In practice, fast probing is carried out using light techniques, for example beta counting with radiation protection probes. Then gamma spectrometry characterization enables conclusions as to the level and quality of the contamination in the structures (GIRONES P. , Développement d'un système de mesure destiné à l'analyse non destructive d'élément d'installation nucléaire de base en vue de leur démantèlement ou de leur déclassement, 1997).

Feedback has shown the industrial limits of implementing such techniques which consist in “probing” large surfaces. There are often considerable numbers of measurement points (no 60 cm probing) and the devices are tricky to set up and use. To meet the need for operational optimization, geostatistics are deployed.

Geostatistics: a method serving nuclear facility decommissioning

Geostatistics is a method for processing digital data with a spatial and/or temporal support. The method takes into account the spatial structure of the data for a space with any given dimensions for irregular, incomplete sampling campaigns, integrating the external data. In D&D, the objectives are to optimize a sampling mesh, prepare contamination mappings or sometimes to calculate the likelihood of exceeding regulatory thresholds.

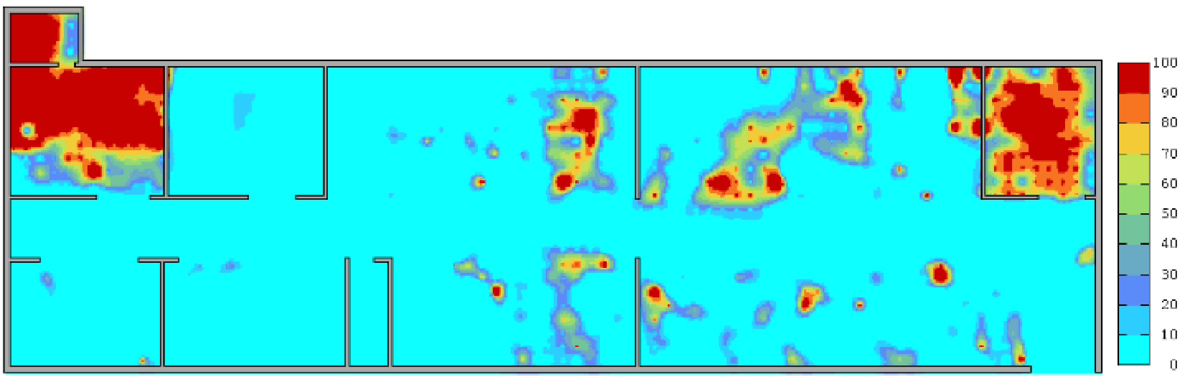

Figure 14: Risk of exceeding (in %) a threshold of 150 cps, for the signal □□□, uranium contamination (DESNOYER, 2010)

Application of the technique begins with analysis of the variogramme, then continues with a spatial estimation and concludes with a simulation of functions and of random fields. It should be remembered that using these methods based on measurement points enables not just control of data quality, but also an understanding of the pollution which still exists, and an evaluation of the uncertainties associated with the quantities of interest, for example when seeking to release a surface after cleanup.

Conclusion: Instrumentation of the lines of defense

The first line of defense is based on the definition of a simplified model of contaminant migration. There has been a great deal of work in this field, and it must be remembered that the support is often in concrete, which is a porous medium and a solid material. There is therefore a combination of diffusion phenomena governed by Fick's Law and by the behavior of the chemical elements making up the concrete in contact with the contaminant which enable a migration model for the radionuclides in the concrete to be established. Surface mapping adds to and completes the model. The characterization methods to carry out the surface mapping are mainly probe measurements with the aid of a contamination meter. The location of concentration points (hot spots) means the contaminant can then be characterized by gamma spectrometry.

These results, from the migration model and surface mapping are combined to provide a 3D mapping. This is useful when undertaking and following up cleanup operations, to keep on target and move towards decommissioning. Geostatistics is a method which is used to define the sampling plan, which will then conclude the process with samplings and destructive analyses.

New techniques like autoradiography have been applied for drawing up surface mapping, especially for beta emitting contaminants.

General conclusion

A methodological approach for process instrumentation has been prepared based on feedback and lessons learned from numerous CEA operations. This approach is based on four functions to which instrumentation can be applied: the initial inventory, waste treatment monitoring, waste package characterization, and the characterization of structures to assess their readiness for decommissioning. For all of these functions, measurements involving gamma spectrometry dominate. However, the current trend is to make use of combined methods. During the inventory phase the gamma imager is an indispensable tool for the localization of contamination concentration points. In the treatment phase, physical and chemical analysis techniques, for which nuclearization work remains to be done (PICARD, 2015), have proved to be indispensable, in particular to meet the challenge of adapting to the containment matrices. Finally, low level measurement needed during facility decommissioning has seen the convergence of data processing techniques such as geostatistics, and innovative mapping techniques like autoradiography (FICHET, 2016).

To conclude, the cleanup-dismantling industry is a challenging field experimenting with innovative instrumentation coupling full of promise for the future.

References

IAEA. (2010). *Information Technology for Nuclear Power Plant Configuration Management IAEA - TECDOC - 1651 .* Consulted July 24, 2016, at http://www-pub.iaea.org/MTCD/publications/PDF/te_1651_web.pdf

IAEA. (2011). *Selection and Use of Performance Indicators in Decommissioning.* Consulté le juillet 16, 2016, sur http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1504_web.pdf

Andra. (2013, Mars). *ANDRA.* Consulted July 15, 2016, at Andra: https://www.andra.fr/download/site-principal/document/synthese-pngmdr-2013-2015.pdf

ASN. (2015, July 16). *Guide de l'ASN N°14 : Mise à l'arrêt définitif, démantèlement et déclassement des installations nucléaires de base en France.* Consulted July 15, 2016, at ASN: http://professionnels.asn.fr/Les-Guides-de-l-ASN/Guide-de-l-ASN-n-6-Mise-a-l-arret-definitif-demantelement-et-declassement-des-INB-en-France

ASN. (2015). Guide de l'ASN n°6 : Mise à l'arrêt définitif, démantèlement et déclassement des INB en France.

BERLIZOV, A. (2007). Performance testing of the upgraded uranium isotopics multi-group analysis code MGAU. *NIM Section A* , 498-506.

BONNER, C. (1994). Comparison of three gamma-ray isotopic determination codes: FRAM, MGA, and TRIFID. *Conference: International symposium on nuclear material safeguards.* VIENNE.

BRENNEIS, C., & GIRONES, P. (2011). *Patent n° FR 2961606.* France.

BUTEZ, M. I. (2011). Radioactive Waste Management Optimization. *WM2011.* Phoenix, USA.

CANBERA. (2010). *CANBERRA.COM.* Consulte in July 2016, at Technical Advantages of ISOCS™/LabSOCS™: http://www.canberra.com/literature/isocs/application_notes/ISOCS-LabSOCS-App-Note-C39530.pdf

CANBERRA. (2015). *iPIX - Système d'imagerie gamma ultra portable.* Consulted July 24, 2016, at CANBERRA.COM: http://www.canberra.com/fr/produits/insitu_systems/pdf/C47248_iPIX-SS_FR.pdf

CARCREFF, H. (1995). *Patent n° 2717587 (A1).* france.

CHABAL, C. (2011). Virtual Reality Technologies: a Way to Verify Dismantling Operations : First application case in a highly radioactive cell.

CHIRON, M. (2008). *THE USE OF THREE DEDICATED NUCLEAR MEASUREMENT TOOLS FOR D & D OPERATIONS IN NUCLEAR FACILITIES.* Consulte July 2016, at CANBERRA.COM: http://www.canberra.com/literature/waste_special_systems/tech_papers/3dedinuctools-paper.pdf

DAVIKDO, A. M. (1999). *A RADIOLOGICAL SURVEY TO DECOMMISSIONING: RESULTS FROM A TECHNOLOGY PROJECT FOCUSED ON THE CHORNOBYL NPP.* Pacific Northwest National Laboratory.

DESNOYER, Y. (2010). Approche méthodologique pour la caractérisation géostatistique des contaminations radiologiques dans les installations nucléaires. *Thesis*. France.

DEVAUX, P. (2010). *Patent n° FR 2936611 (A1).* France.

DUCROS, C., HAUSER, G., MAHJOUBI, N., & GIRONES, P. (2016). RICA: A Tracked Robot for Sampling and Radiological Characterization in the Nuclear Field. *Journal of Field Robotics* .

FICHET, P. (2016, février 17). *How Digital Autoradiography technique can be useful for D&D projects.* Consulted June 12, 2016, at www.oecd-nea.org: https://www.oecd-nea.org/rwm/wpdd/predec2016/docs/S-7-1___ppt__FICHET__.pdf

GAL, O. (2006). Development of a portable gamma camera with coded aperture. *Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment* , 233–237.

GAMAGE, K. (2015). Neutron gamma fraction imaging: Detection, location and identification of neutron sources. *Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment* , 9-12.

GHIBAN, A., & Al., A. (September 2016). L'usine du futur au service de l'assainissement et du démantèlement.

GIRONES. (2006). Methodology for determining the radiological status of a process: Application to decommissioning of fission product storage tanks. Athens: IAEA.

GIRONES, P. (March 2016). Course for Master ITDD University of Grenoble. *Course INSTN* . Valence, France.

GIRONES, P. (1997). *Développement d'un système de mesure destiné à l'analyse non destructive d'élément d'installation nucléaire de base en vue de leur démantèlement ou de leur déclassement.* PARIS: CNAM.

GIRONES, P. (2013). First report from an advanced radiological inventory for a spent fuel reprocessing plant. *SFEN.* Avignon.

GIRONES, P. (2010). *Patent n° FR2943143.* France.

GIRONES, P. (2012). Perspective d'une solution intégrée pour une maîtrise d'ouvrage. *Inventaire radiologique, caractérisation et échantillonnage en amont et en aval des opérations de démantèlement, SFEN.* PARIS: SFEN.

GIRONES, P. (2009). *Patent n° WO2010103071.* France.

GIRONES, P., & al., A. (2005). Underwater Radiological Characterization of a Reactor Vessel. *ICEM'05.* Glasgow (UK).

GIRONES, P., DUCROS, C., LEGOALLER, C., LAMADIE, F., FULCONIS, J., THIEBAUT, V., et al. (2006). Methodology for Determining the Radiological Status of a Process: Application to Decommissioning of a Fuel Reprocessing Plant. *American Society of Mechanical Engineers - ASME; New York (United States); 14. International conference on nuclear engineering (ICONE 14*, (p. 8 p). MIAMI.

GOSSEN, G., DEYGLUN, C., & GIRONES, P. (2015). NDA quantification of Nuclear Waste Containers. *INMM.*

GUNNINK, R. (1994). A new analysis code for measuring U-235 enrichments in arbitrary samples. *IAEA Symposium on International Safeguards.* Vienna.

GUNNINK, R. (1990). *Nondestructive measurements of x-ray and gamma-ray emissions can be used to analyze a sample for plutonium.* Lawrence Livermore National Lab.: DOE/DP; NRC.

HAHN, D. W., & OMENETTO, N. (2012). Laser-Induced Breakdown Spectroscopy (LIBS), Part II: Review of Instrumental and Methodological Approaches to Material Analysis and Applications to Different Fields. *Appl Spectrosc* , vol. 66 no. 4 347-419.

HIRON, M., DIOP, C. M., & SUTEAU, C. (2003). The MERCURAD dose rate modeling software for D and D operations in nuclear facilities. *international conference on dismantling nuclear facilities. Decommissioning Challenges: An Industrial Reality?; Avignon (France).* Avignon: SFEN.

JALLU, F. A.-G. (2011). The use of non-destructive passive neutron measurement methods in dismantling and radioactive waste characterization. *ANIMMA*, (pp. 1-8).

JEANNEE, N., & FAUCHEUX, C. (2013). *RETOUR D'EXPERIENCE CRITIQUE SUR L'UTILISATION DE METHODES GEOSTATISTIQUES POUR LA CARACTERISATION DES SITES ET SOLS POLLUES.* RECORD.

LAMADIE, F. a. (2005). Remote alpha imaging in nuclear installations: new results and prospects. *52* (6).

LAMADIE, F., & DEVENELLE, P. (2003). Chaîne Automatique de Mesures des Déchets de l'Installation Chargée de l'Entreposage des Solides = Automatic chain of measurements of waste of the nuclear facilities in charge of the storage of solids. *Bulletin du Bureau national de métrologie* , pp. 42-48.

LAMADIE, F., & GIRONES, P. (2007). Recent Achievements for In-Situ Mesaurement: Applications to an Actual Decommissionning Project. *WM'7 Conference.* Tuscon.

LE BARS, I., & GALET, C. (2010). *LES RISQUES DE CRITICITE ET LEUR PREVENTION DANS LES USINES ET LES LABORATOIRES.* PARIS, IRSN: IRSN.

LECOCQ, P. (1999). *Démantèlement et déconstruction des centrales nucléaires, la stratégie d'EDF.* Consulted July 2016, at Annales des mines: http://www.annales.org/re/1999/re04-14-1999/024-034%20Lecoq.pdf

LEGOALLER, C., & IMBARD, G. (1998). *The development and improvement of the Aladin gamma camera.* European Commission, Nuclear Science and Technology.

LEMAIRE, H. (2014). Implementation of an imaging spectrometer for localization and identification of radioactive sources. *Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment* , 97–103.

LEPY, M.-C. (2008, janvier 8). *nucleide.org.* Consulted August 2, 2016, on the LNHB site: http://www.nucleide.org/Gamma_spectro/NT_08-05_principe_spectro_gamma.pdf

MAURY, C., SIRVEN, J.-B., & TABARANT, M. (2013). Analysis of liquid sodium purity by laser-induced breakdown spectroscopy. Modeling and correction of signal fluctuation prior to quantitation of trace elements. *Spectrochimica Acta Part B: Atomic Spectroscopy* , 28-35.

MONTREUIL, F. (2010, septembre 15). *ASTR.* Consulted July 10, 2016, at http://www.atsr-manifestations.fr/: http://www.atsr-manifestations.fr/cariboost_files/s8-Pr_C3_A9sentation_20forum_20ATSR_20MONTREUIL.pdf

NUCADVISOR. (2015, aout). Consulted July 15, 2016, at http://www.developpement-durable.gouv.fr/IMG/pdf/061015_Resume_non_technique_diffusable_Version_a_publier.pdf

OREKASOLUTIONS. (s.d.). *http://www.orekasolutions.com/demplusfornuclear/*. Consulted July 25, 2016, at OREKAsolutions: http://www.orekasolutions.com/demplusfornuclear/

PICARD, J. (2015, janvier 22). Développement de la LIBS pour l'analyse en ligne de produits uranifères et plutonifères solides. *Thèse pour obtenir le grade de docteur de l'université de Bourgogne* .

SAMPSON, T. (1995). PC/FRAM: New capabilities for the gamma-ray spectrometry measurement of plutonium isotopic composition. *Conference: 5. International conference on facility-safeguards interface.* Jackson Hole.

SANTO, J., & RANDY, M. P. (2006). *Application of Remote Gamma Imaging Surveys at the Turkey Point PWR Reactor Facility.* Consulté le 2016, sur http://www.pajaritoscientific.com/pdf/TECH_APP_REMOTE_GAMMA_IMAGING_TURKEY_POINT_FACILITY.pdf

SIMON, A. (2011). Determination of Actinide Isotopic Composition: Performances of the IGA Code on Plutonium Spectra According to the Experimental Setup. *58* (2).

TALLEC, M., & KUS, J.-P. (2010). Démantèlement des Installations Nucleaires. *Techniques de l'ingénieur* .

THEVENON, J.-B. a. (2009). Using simulation for intervention design in radiating environment: first evaluation of NARVEOS. *GLOBAL 2009 congress - The Nuclear Fuel Cycle: Sustainable Options and Industrial Perspectives* (p. 504). Paris: SFEN.

TOUBON, H., CORDIER, C., & FERET, B. (2007). CANBERRA solutions for source location and activity determination for investigations of ORCADE dismantling project at AREVA La Haque site. *ENC.* Brussels.

WAHL, C. (2014). Polaris-H measurements and performance. *IEEE Nuclear Science Symposium and Medical Imaging Conference (NSS/MIC)* (pp. 1-4). Seattle: IEEE.