

IN-SITU NUCLEAR MEASUREMENTS FOR A FACILITY PRIOR TO DECOMMISSIONING

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INTRODUCTION

Within the framework of nuclear facility decommissioning, a radiological description of the premises and of the process elements that it contains remains essential.

It enables us well in advance to:

- guide and provide assistance in the definition of dismantling scenarios
- optimise the worksite with regard to radioprotection
- know ahead of time the volumes and conditioning of the waste according to the possible or specifically envisioned disposal methods (HLW, LLW, ...).

This approach enables us to optimise dismantling procedures from both the technical and economic standpoints.

The CEA has a methodology for carrying out the radiological characterisation of a nuclear facility undergoing dismantling. This method, based on in-situ measurements was perfected thanks to feedback obtained from many measurement campaigns carried out over the past few years on different types of facilities (reactors, hot cells, ventilation ducts,...).

Such a procedure can establish all source terms of a facility. It therefore provides the operator with complete data concerning the distribution of activities (contamination and activation) in his facility.

DESCRIPTION OF THE METHOD

The method is divided into 6 major phases that correspond to the various stages arranged in a chronological manner:

- Study of the facility,
- Definition of the input data,
- Establishment a plan of measurements,
- Realisation of in-situ measurements,
- Interpretation of the measurements,
- Publication of a cartography or «mapping» of the activity.

Phase 1: STUDY OF THE FACILITY

This first phase is essential in order to know in the most accurate manner possible the different aspects of the facility (operation, the role of the circuits, incidents, radioelements that are either present or absent, chemical treatments, ...).

This phase therefore begins with the collecting of floor and circuits plans, process descriptions and other documents allowing us to become familiar with the facility. Its study is afterwards carried out with the operator, based on the documents that have been obtained previously.

This phase must yield a physico-chemical description of the process concerned, its history and its radiological characteristics. It is necessary for the following step: the definition of input data.

Phase 2: DEFINITION OF THE INPUT DATA

These data include the list of radionuclides potentially present and their formation processes, (activation – contamination) as well as a «zoning» of the facility according to the physico-chemical character of the process.

Each zone is characterised by ratios between measurable and non-measurable radionuclides. The radionuclides are termed «measurable» according to the used measurement technique. Concerning an overall characterisation of a facility, the measurement of gamma radiation enables us to use non-destructive methods that are easy to implement. (spectrometry, dose rates) These measurable radioelements will therefore be gamma emitters with an emission rate and energies such as their measurement through structure elements of the facility is both technically and economically feasible.

In certain conditions, (for transuranics measurements) we may also use neutron methods. For this type of radiation which is less attenuated by the matrix than the gamma radiation is. These techniques (passive counting or active interrogation) are more difficult to implement and to interpret.

Furthermore, they require complementary data such as the isotopic composition or the chemical nature of the deposits in order to be able to determine in a reliable manner the activity or the mass of isotopes that are sought.

Therefore neutron methods are reserved for configurations in which the use of gamma measurements does not allow us to quantify the radio tracer characteristics of the isotopes present in the facility.

The ratios existing between the measurable and non-measurable radionuclides can be obtained with several tools which are used either alone or in combinations :

- Empirical correlation laws,
- Calculation codes,
- Specific sampling and analysis programme.

Empirical correlation laws

These correlation laws come from experimental studies carried out on a large number of facilities or processes rather similar. In many cases, these ratios are not available. Another solution must then be envisioned in order to be able to obtain quantified correlations between the gamma emitters and other radionuclides on the whole facility.

Calculation Codes

Based on characteristics of the design and operation of a plant, certain calculation codes can predict the activity of some non-measurable nuclides (Example: Evolution codes: Darwin^{©CEA}, contamination transfer codes in PWRs: Pactole^{©CEA} and Profip^{©CEA} ..). Nevertheless, it remains isotopes for which neither of the two preceding approaches can provide a correlation with a measurable radioelement.

Specific Sampling and Analysis Programme

The only means left is to carry out a programme of samplings, as reduced as possible, and afterwards to perform radio-chemical analyses on the isotopes potentially present in the portion of the sampled circuit

This technique is complicated and particularly laborious to implement. Phase 1 (Study of the facility) enables us to optimise the sampling programme so that we can choose the portions of the circuits that are the most representative and of the easiest access.

Phase 3: ESTABLISHING A PLAN OF MEASUREMENTS

Based on the data determined during the two preceding phases, a plan can be established. It defines the number and localisation of the non-destructive measurement points as well as the possible sampling points.

The following are defined by zone:

- A measurement point by gamma spectrometry in order to determine the representative spectrum of the activity in the zone,
- The points of dose rate mapping that will enable us to assess the mass or surface activities in different points of the zone.

The radiological singularities that certain components present (exchanger, tank, evaporator, ...) necessitate a specific treatment that requests, depending on the geometric and radiological complexity of the object, a more or less greater number of measurements including those performed using gamma spectrometry.

At the time of a comprehensive radiological characterisation of the entire facility, we must remain careful of certain «hot points» that can generate high dose rates due to a geometric concentration of the activity, but which, in reality, only represent low activity in comparison to the entire circuit.

On the other hand, certain «large components» such as exchangers, owing to their considerable surface, may concentrate a significant fraction of the total activity of the circuit and yet without generating a high dose rate.

Phase 4: CARRYING OUT THE IN-SITU MEASUREMENTS

Generalities

The realisation of in-situ measurements is done using measurement devices easily transportable and having sufficient autonomy. In the majority of configurations, the use of a standard measurement system is possible (Ex: EMECC^{©CEA}, ...). Nevertheless, the adaptation or the creation of specific measurement systems is often better in considering accuracy and manipulation.

It is difficult to use gamma spectrometry, a very precise but complicated technique to implement in order to establish a total activity balance. This procedure is usually reserved in

the determination of gamma-type spectra and for the fine characterisation of singularities (components, hot points,...).

The facility as a whole is afterwards mapped by means of global measurements (or dose rates) on contact or at a distance. Certain singularities, the behaviour of which has been perfectly mastered with regard to radioelements, can also be partially or entirely characterised in this way.

In the case of an absence of diagrams, allowing us to know the lengths, surfaces and volumes of the different contaminated circuits, we carry out a dimension survey which is done at the same time as the mapping. These dimensions then enable us to establish numerical modelling of each measurement point and to determine the surface or volume value characterised by each point.

The neutron measurements are applied occasionally at specific points of the process for the isotopes that are not measurable when using gamma spectrometry (Example : heavy shielding or great self shielding of the source).

Gamma-Spectrometry Measurements

Generally, Germanium detectors are used. The range of detectors available allows us to satisfy most measurement needs.

Alternative solutions may be applied such as the use of CdTe spectrometers in the case of very high radiation or a lack of space to set up a Germanium detector that might prove to be too cumbersome.

Furthermore, the use of a high volume scintillator (NaI, ...) can reach lower contamination thresholds than those obtained using a Germanium detector.

Nevertheless it is impossible to reach the precision and resolution of a Germanium detector with other types of detectors.

The detectors must be equipped with a protection against the irradiating environment. The acquisition of a «background spectrum» obtained by blocking the collimator aperture with a «plug» allows us to ensure adequate rejection of all noising sources.

Dose rate measurements / Global measurements

These measurements are faster and easier to perform than the gamma spectrometries. The care taken in their realisation, guarantees accuracy in the calculation of the total activity of the facility. Then the calibration of the measurement instruments is regularly performed.

Disposing the probes in a hemispheric protection, which will ensure, along with the measured component on contact, efficient shielding, must reduce noise sources.

For certain types of contaminant, we prefer to carry out global measurements using a low resolution spectrometer. A coarse energy histogram is better in this case than a fine spectrum, which necessitates a complex post treatment thereby forcing a great number of measurements to be used.

Neutron Measurements

The use of neutron measurements is limited to the characterisation of components contaminated by the transuranics. They are difficult to perform in-situ and involve generally the design of a specific system of measurement for each configuration.

Measurement obtained by passive counting of neutronic coincidences can seldom be used due to the low efficiency of this measurement. Even though the total counting has a better sensitivity, its interpretation is conditioned by an excellent knowledge of the chemical nature of the source, which is seldom possible.

Active neutron measurements necessitate the setting up of additional radioprotection rules. For large quantities of nuclear material, the self-shielding phenomena can lead to very large uncertainties.

The use of these measurement techniques is therefore limited to configurations in which the gamma measurements are not applicable.

Phase 5: INTERPRETATION OF THE MEASUREMENTS

The purpose of the interpretation is to provide knowledge of the activities based on all the measurements carried out. Generally, surface and mass activities are obtained. The transformation in total activities is done in the following phase.

The interpretation of the measurements relies heavily on numerical simulation. For geometries as complex and varied as those encountered in a facility, it is impossible to envision experimental calibration (with sources).

Numerical simulation intervenes in each measurement technique for:

- Gamma spectrometries: the use of tools directly integrating numerical modelling for commonplace geometries (Ex: EMECC^{©CEA}), or manual modelling for complex components with calculation codes (MERCURE^{©CEA}) or Monte-Carlo (MCNP^{©LANL}).
- Global measurements or dose rates: given the number of measurements to be performed, it has been necessary to create tools, which enable us to handle a great number of cases. These tools were developed around MERCURE^{©CEA} due to the fact that its operating procedure (blind mode, inlet/outlet through a text file, straight line calculation) facilitates automatic processing. It enables us to calculate, for each point, a coefficient that obtains the activity based on the measured value. This coefficient integrates a numerical transfer function determined for the dimensions of the measured portion of the facility and the typical spectrum of the gamma emitters.
- Neutron measurements: the use of a three dimensional Monte-Carlo code (TRIPOLI^{©CEA}, MCNP^{©LANL}) is necessary to perform the interpretation of such measurements.

Phase 6: ESTABLISHING A «CARTOGRAPHY» OF THE ACTIVITIES

All the numerical data (activities, dimensions of the circuits, surfaces/volumes associated with the measurement points, ratios, ...) is recorded in a spreadsheet which calculates the products of the activities through values of the surfaces or volumes and carries out the sum.

The cartography of the activities is then established zone by zone, outside singularities, by means of surface or mass activities obtained and from the surfaces and masses coming from the study of the facility. It is completed by the activity of the singularities which have been treated independently.

FEEDBACK: EXAMPLE OF USE

This method has been in use for the past ten years or so on several types of facilities. Three examples of application are presented hereafter.

Hot Cells

In order to obtain the source term from a hot cell, we combined:

- A dosimetric cartography carried out using a probe suspended from a 2D tower which allows us to know the spatial distribution of the sources in the cell. The activity of each targeted zone was obtained thanks to numerical modelling.
- The gamma spectrometry carried out using a Germanium detector of slight volume, associated with a modelling of the cell provided us with the total activity of the gamma emitters. The preceding dosimetric cartography used as a basis in the establishment of the model
- Samplings carried out using a remote-controlled manipulator arm (TAM) underwent radio-chemical analyses that subsequently enabled us to establish a typical spectrum of the cell.

This method has allowed us to characterise about ten different high-activity hot cells.

Ventilation ducts of hot cells

The objective of this action was to determine, in a non-destructive manner, and without shutting down the ventilation, the deposited activity in the various circuits upstream from the filters.

The following were carried out:

- The determination of the most unfavourable typical spectrum in relation to the characteristics of the different treated fuels
- Gamma-spectrometries on each duct of the cell and on each plenum in order to determine the typical gamma spectrum at different points.
- Gamma spectrometries on hot points
- A dosimetric cartography of all the ducts and plenums (a measurement taken on each side to take into account the dust sedimentation effects).

The total ventilation system of a facility having about ten cells was thus characterised.

Activity balance of a reactor

Within the framework of the study of dismantling scenarios, it is crucial to know the contamination level of all the circuits of a nuclear reactor. This was carried out on a

pressurised water reactor in order to determine the deposited contamination outside the neutronic flux.

The following phases were carried out:

- Examination of the different circuits with the plant operator and selection of the contaminated circuits.
- Choice of the measurement points on the circuits and the components (tanks, exchangers, pressuriser).
- Gamma spectrometry measurements (EMECC^{©CEA}) on the different points chosen beforehand (~ 10 points).
- Cartography in dose rates (using a shielded probe) of all the contaminated circuits and of certain components (~ 100 points).
- A manual survey of the lengths, diameters, thickness of the insulation in the contaminated circuits.
- Sampling of a circuit portion and radio-chemical analyses.
- Numerical modelling of the gamma spectrometries and calculation of the surface
- Numerical modelling of the dose rate measurements and calculations of the activities corresponding to the typical gamma spectrum
- Calculation of the internal surfaces of the circuits and components
- Global activity calculation based on the partial results of the preceding stages

Finally, we obtain the source term for all the reactor circuits for all the isotopes present.

CONCLUSION

This methodology is based on well-identified stages, to be carried out chronologically.

The instrumentation and the calculation codes used are based on known principles but necessitate in certain cases a specific adaptation.

Its originality resides in the combination of the different tools and techniques (instrumentation, calculation codes, radio-chemical analyses).

Based on this structured procedure, we may examine a piece of data as complex as the global source term of the facility.

Furthermore, it enables us to assist, step by step the work master in the complex process of dismantling a nuclear facility.

This global characterisation method has been applied successfully on very different types of facilities.



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In-situ nuclear measurements for a facility prior to decommissioning



Radiological characterization for nuclear facilities

- **Methodology**
- **Tools**
- **Waste and fuel element characterization**
- **Feedback**

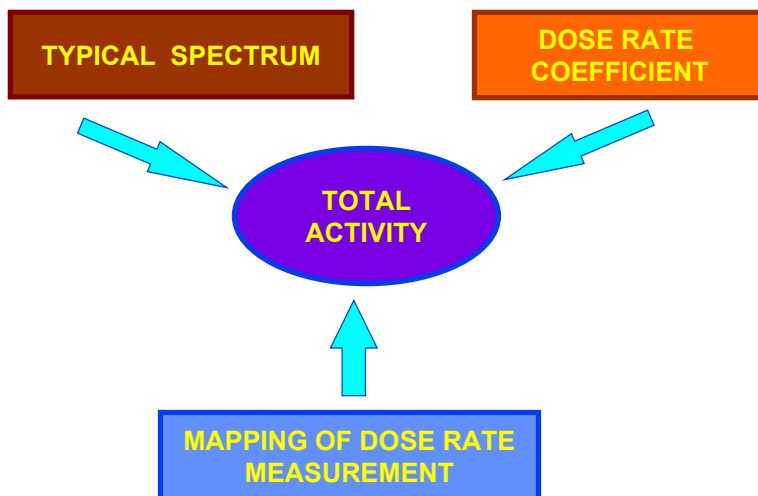


Objectives :

- Assistance in the definition of dismantling scenarios
- Optimisation of the work with regard to radioprotection and costs
- Optimisation of the waste management
 - Classification of the primary waste according to activities and radioactive half lives
 - Reduction of volume
 - Adaptation of the conditioning

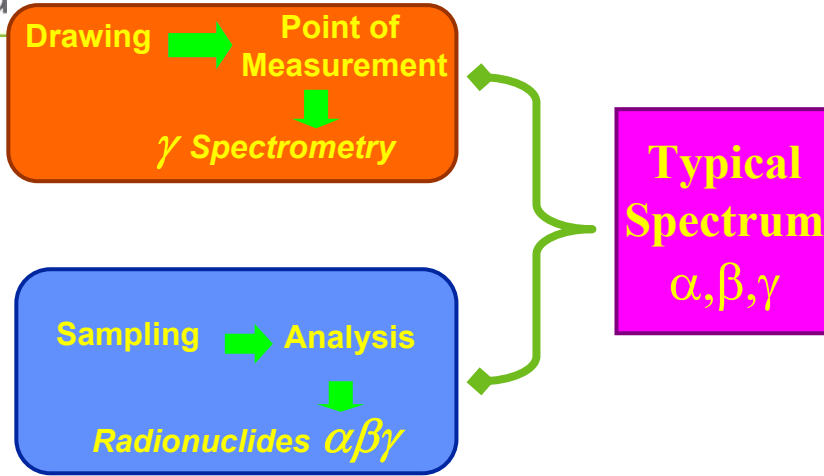


Presentation of the method



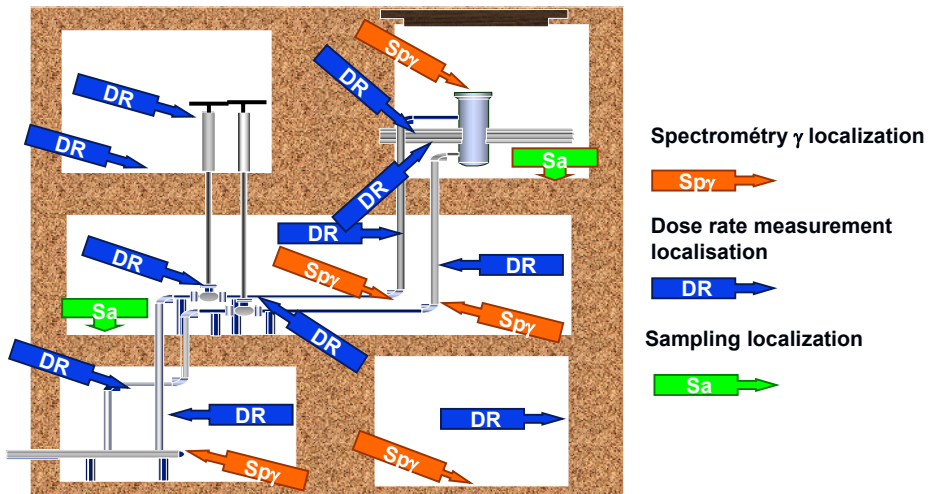
Radiological characterization for nuclear facilities

Presentation of the method : Determination of the typical spectrum

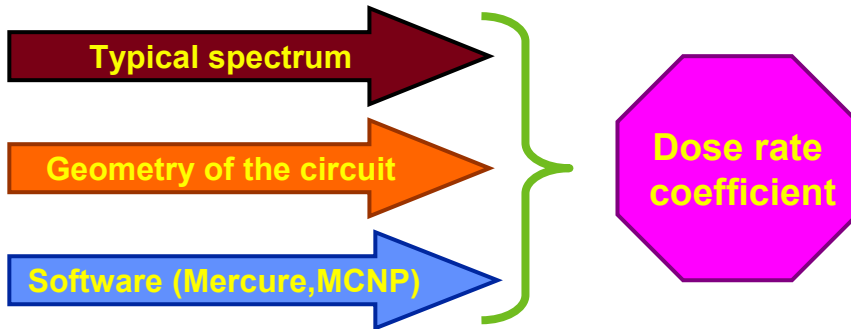


Radiological characterization for nuclear facilities

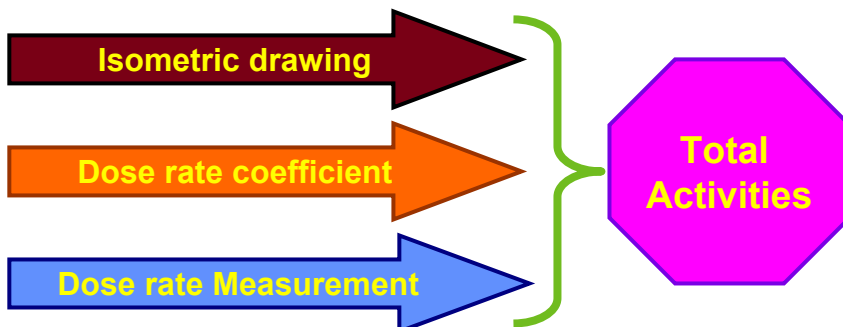
Localization of the measurements



Presentation of the method : Calculation of the dose rate coefficient

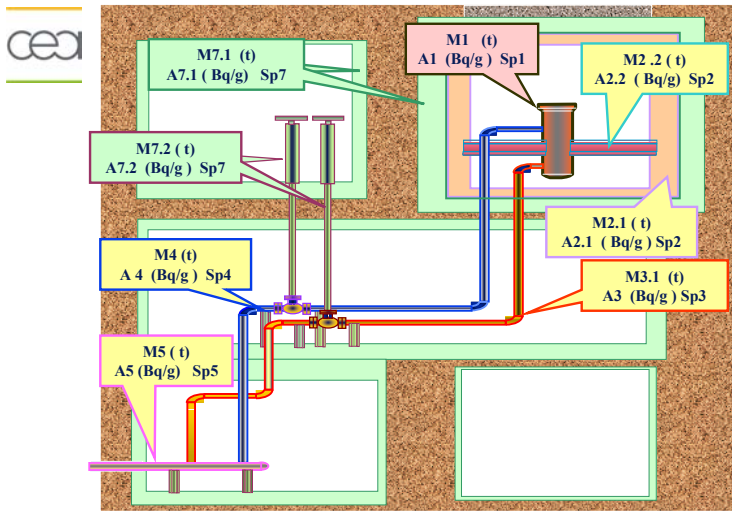


Presentation of the method : Calculation of the total activities



Radiological characterization for nuclear facilities

Results



M x.y : Mass of waste (t)
A x.y : Specific activity of the waste (Bq/g)
Sp y : Typical spectrum
X : spectrum number
Y : kind of material

Tools for radiological characterization

Activity contamination measurement device (EMECC)

Non destructive measurement of the contamination inside nuclear facilities

Range of measurement : 0,001 à 100 GBq/m²

Numerical calibration

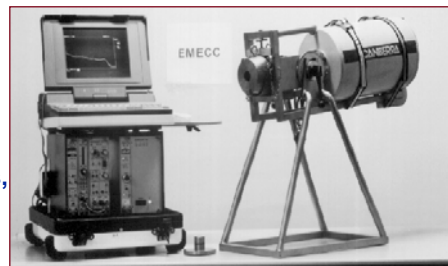
Field of application :

- Activation and fission products
- Water reactor circuit
- Hot cells

Feedback :

More than 300 measurement campaigns in French or foreign PWR's :

Germany, Spain, Switzerland, Sweden,...
Dismantling of reactors : Sena, Brennilis, G2, G3, Rapsodie, Saint-Laurent A, ChinonA, submarines, reprocessing plants



Gamma spectrometry measurement at room temperature



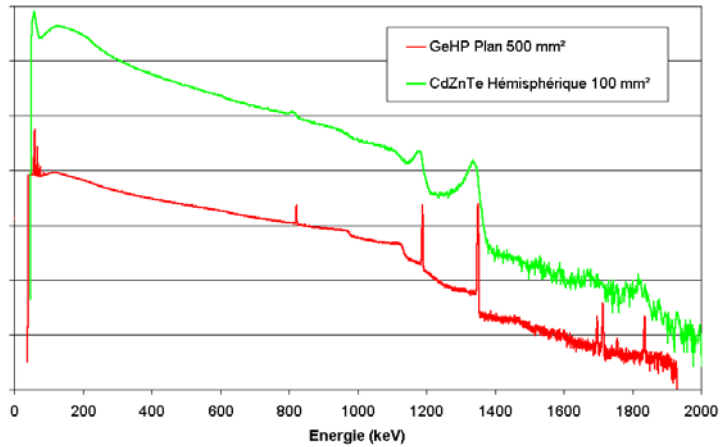
Measurement on a PWR's steam generator



GeHP



CdZnTe



Collimated dose rate measurements



- Discrimination between source and background radiation field
- Shielded device :
 - Geiger Muller detector
- 2 detectors :
 - AD15 Measurement range :
0,01 mSv/h -> 10 Sv/h
 - AD18 Measurement range :
0,01 μ Sv/h -> 10 mSv/h

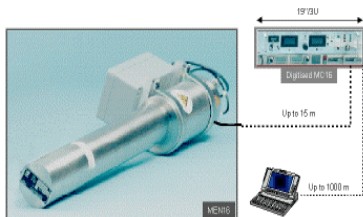


Tools for radiological characterization



DINAMO

- Characterization of fissile materials by passive and active neutron measurements
- Portable pulsed neutron generator : $2 \cdot 10^8$ n/s
- Detector : He3
- Numerical calibration by neutron transport code



CEG Workshop

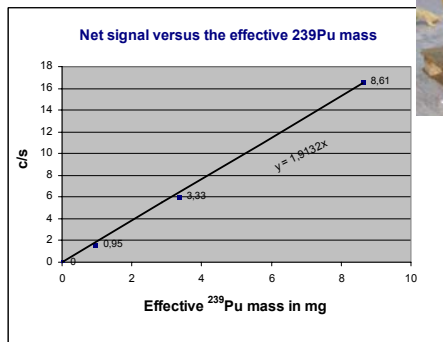
Petten 26-28 May 2004

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Tools for radiological characterization



COQUINA : Transportable neutron cell for fissile mass measurement in small size samples by Active & Passive Neutron Analysis



**DL less than
1mg Pu**



CEG Workshop

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Tools for radiological characterization

Software



- **MERCURE :**
 - Straight line attenuation for gamma
 - Monte-Carlo method
 - Used for :
 - Numerical calibration for the gamma spectrometry measurements
 - Dose rate coefficient calculation
- **MCNP :**
 - N particles transport software based on Monte- Carlo method
 - Used for :
 - numerical calibration of the active and passive neutron measurements systems.
 - numerical calibration for the gamma spectrometry measurements of complex geometry

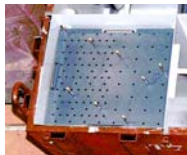
Waste and fuel element characterization

Design of measurement device and software for waste package measurement



CARACO, EMEF, DECALOG : for EDF and CEA waste package measurements

- Concrete canisters
- 100 and 200 liter steel drums
- 5 and 10 m³ boxes
- Numerical calibration
- Experimental validation

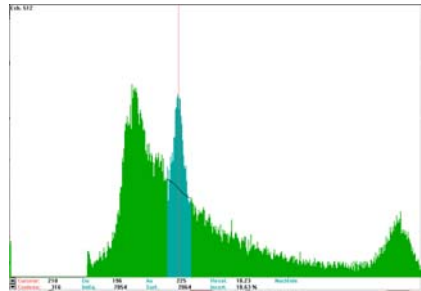


The screenshot shows the EMEF-Resultat software interface. It features a menu bar (Fichier, Edition, Calcul, Options ?) and a toolbar. A table displays measurement data for various sources (88, 131, 194, 287, 424, 679, 1000, 1500, 2000, 2500) with columns for source number, activity, and another activity value. Below the table, the EMEF configuration parameters are listed, including the number of sources (95), distance of measurement (95 cm), and various screen and detector settings. A dialog box titled 'Coque avec déchets du type hétérogènes en panier' is open, showing configuration options for the measurement device, such as screen type (Acier, Plomb, Béton), screen thickness, and detector type (Enrobage).

Measurement tunnel for bulk wastes or earth (Very low level activity)



Big Area
detector
NaI(Tl)



Measurement for 1 Bq/g ^{235}U

Neutron probe

- with 1 high efficiency fission chamber (wide range of burn up and cooling time)

Room temperature gamma spectrometry

- Ritec SDP probe with 5 to 60mm³ crystal (CdTe detector)

Compact cylinder

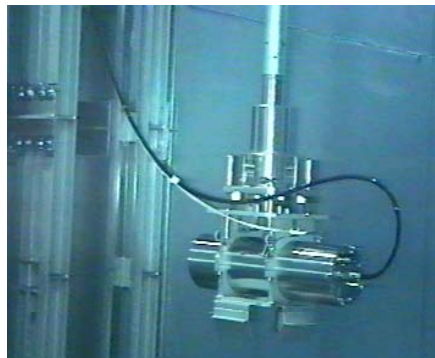
- (F160 mm, L 600 mm, weight 40kg)

Carrier fitted to NPP geometry

- Upper part fits fuel handling tool
- Lower part fits storage racks (positioning accurate 0.5cm)

Successful SMOPY tests for :

- Operation in industrial environment
- Qualification for use of SMOPY with regards to safety and operability
- BU confirmed within 5% on 3 cycles LEU fuels



Feedback



Field of application

- **Nuclear reactor :**
 - **Natural uranium and graphite or heavy water reactors**
 - **Pressurized water reactor**
 - **Submarine reactor**
- **Hot cells**
- **Reprocessing plants**
- **Uranium enrichment facilities**

Feedback



Conclusion :

- **General method to perform before dismantling of nuclear facilities**
- **Successful application on very different kind of facilities**
- **Adapted tools for radiological characterization**
- **Flexible and industrial methodology**