INNOVATIVE NUCLEAR MEASUREMENT TECHNIQUES USED TO CHARACTERIZE WASTE PRODUCED BY COGEMA'S NEW COMPACTION FACILITY

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Spent nuclear fuel from commercial power reactors throughout the world is reprocessed at the COGEMA plant in La Hague. After shearing and dissolution of the fuel assemblies in nitric acid, the hulls and end pieces are sent to COGEMA's new compaction facility to reduce the final volume of waste. Technological waste produced in the reprocessing plant is also sent to this facility.

The compacted waste is characterized by two measurement stations:

- a gamma spectrometry station,
- an active and passive neutron measurement station.

The main purpose of these measurement stations is to determine a set of guaranteed nuclear parameters and their associated uncertainties:

- Total uranium and plutonium masses
- Alpha activity (Pu, Cm, total)
- Beta activity (^{137}Cs, ^{90}Sr-^{90}Y, ^{241}Pu)
- Thermal power.

This paper will describe the two measurement stations in detail. It will focus more specifically on their most innovative features:

- Optimization of the neutron cell station design (249 counters and two high-flux neutron generators) to measure simultaneously prompt and delayed neutrons from fission reactions generated by the neutron generator;
- Use of a new high flux neutron generator (2×10^9 n/sec);
- Development of an algorithm combining neutron and gamma measurements as well as correlation laws based on a simplified evolution code (CESAR);
- Use of a Monte-Carlo method for the propagation of uncertainties.

The performance characteristics of these measurement stations, as well as results of tests performed prior to the commissioning of the new compaction facility, will also be presented.

1. BACKGROUND

COGEMA's facilities in La Hague, France reprocess irradiated fuel assemblies from approximately 100 nuclear reactors around the world. After shearing and dissolution of the fuel assemblies, hulls and end pieces are compacted in a COGEMA's new compaction facility which will be commissioned during the year 2001. The compacted waste is conditioned in standard-design waste drums, known as CSD-C containers, for permanent disposal. Radiological characteristics of the CSD-Cs are determined through extensive assessment with two non-destructive measurement stations : a gamma spectrometry unit with five germanium diode detectors and an active and passive neutron measurement station with 249 neutron counters and two new-generation neutron generators.

2. COMPACTING PROCESS DESCRIPTION

After shearing and dissolution, the fuel assembly hulls and end pieces fall into drums measuring approximately 1 m in diameter and 1.5 m high. The mean bulk specific density of the structural waste in the drums is about 0.8 kg/l. The drums are brought through the compacting facility entrance and checked by gamma spectrometry and by active and passive neutron measurement identical to that performed at the exit of the ACC facility. The hulls and end pieces are then emptied into a metering separator unit that fills 80 l metal containers. After drying, the waste is compacted in a 2500 ton press and the compacted containers, called disks, are inserted into standard-design CSD-C containers which are closed with weld-sealed lids. The CSD-Cs may also contain disks of technological waste that are not suitable for surface storage. Radiological characteristics of the CSD-Cs are checked at the facility exit by gamma spectrometry and by active and passive neutron measurement identical to that performed at the entrance of the ACC facility.

3. MEASUREMENT OBJECTIVES

3.1 INCOMING MEASUREMENT STATION

This measurement station located at the compacting facility entrance is used to:

- determine the fissile mass in the drums and compare it with a threshold to ensure that all areas remain subcritical up to the intermediate storage area at the facility exit ;

- ensure similar determination of characteristics for all types of drums produced in the reprocessing facilities ;

- determine the structural waste residue unit (URSD) inventory for each COGEMA La Hague customer.

The URSD inventory, which depends on the alpha and beta activity of the waste package, enables a balanced distribution of CSD-Cs for return of waste to each customer country.

3.2 OUTGOING CSD-C MEASUREMENT STATION

This measurement station located at the compacting facility exit is used to :

- determine the fissile mass in the CSD-Cs and compare it with a threshold to ensure that subcritical conditions are maintained in the downstream facility intermediate storage area ;

- characterize the finished product CSD-Cs by determining ensured variables (activity and mass with associated threshold for each variable) and complementary variables expressing the specifications for the waste repository contractor (ANDRA in France) ;

- determine the URSD rate, which enables a balanced distribution of CSD-Cs for each COGEMA La Hague customer.

4. MEASUREMENT STATIONS DESCRIPTION

4.1 INCOMING MEASUREMENT STATION

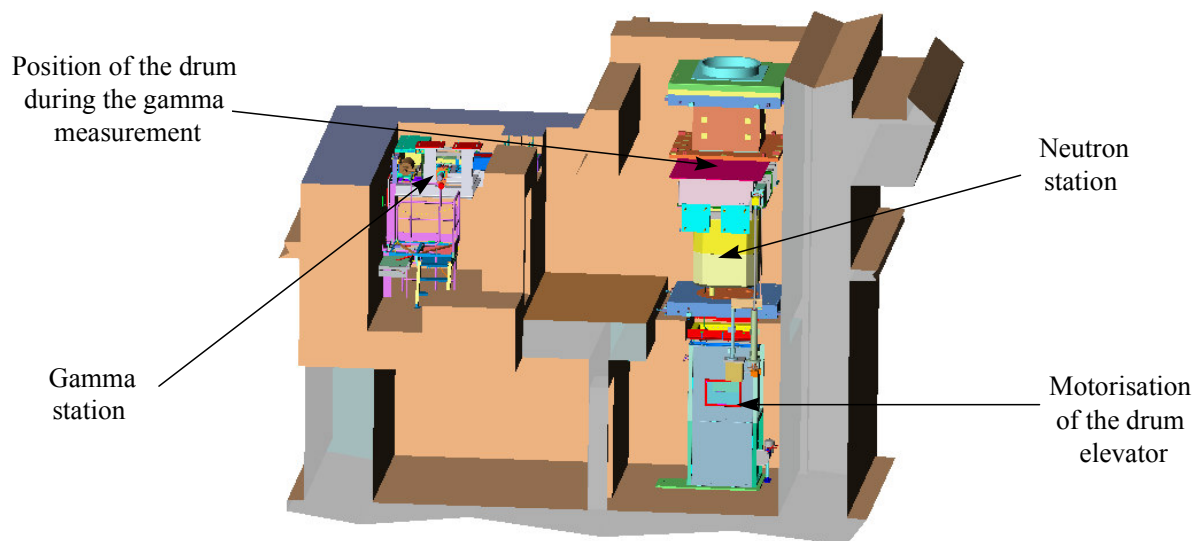Figure 1 shows the general layout of the incoming measurement station

Figure 1

4.1.1 Gamma Spectrometry

Gamma spectrometry is carried out in five steps with the drum in rotation for each measurement. A fixed collimator and an auxiliary mobile collimator placed in front of the detector ensure an acceptable dose rate that limits dead time in the signal acquisition channel.

Gamma spectrometry is performed with an EGP 500-13-TR hyper-pure, semi-planar germanium detector manufactured by EURISYS MESURES. This type of semi-planar detector, which is 500 mm^2 and 13 mm thick, features relatively low sensitivity to high energy gamma rays (e.g., ^{60}Co at 1173 and 13,332 keV) and reduces the Compton front at low energy levels, thereby lowering the detection limits for lower energy peaks such as for ^{137}Cs (662 keV) and ^{134}Cs (796-802 keV).

Figure 2 shows details of the gamma spectrometry station layout.

Collimator with 4 slits

Dewar (N_2)

Ge detector
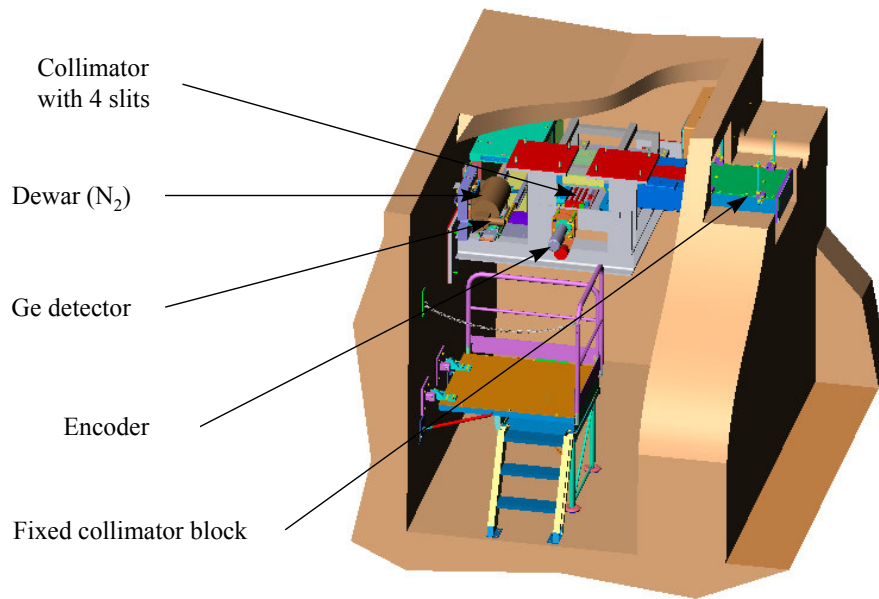
Encoder

Fixed collimator block

Figure 2

4.1.2 Neutron Measurement

The neutron measurement cell includes:

- two GENIE 36 neutron generators developed by SODERN for the ACC compacting facility project. The innovative features of these generators are detailed in 6.1 ;

- three blocks of 83 one-meter-long, 1 inch, 4 bar ^{3}He counters (type 150NH100 manufactured by EURISYS MESURES) covering three sides of the cell.

The cell has a measurement efficiency of around 8%. The drums rotate during the measurements. Figure 3 details the layout of the cell.
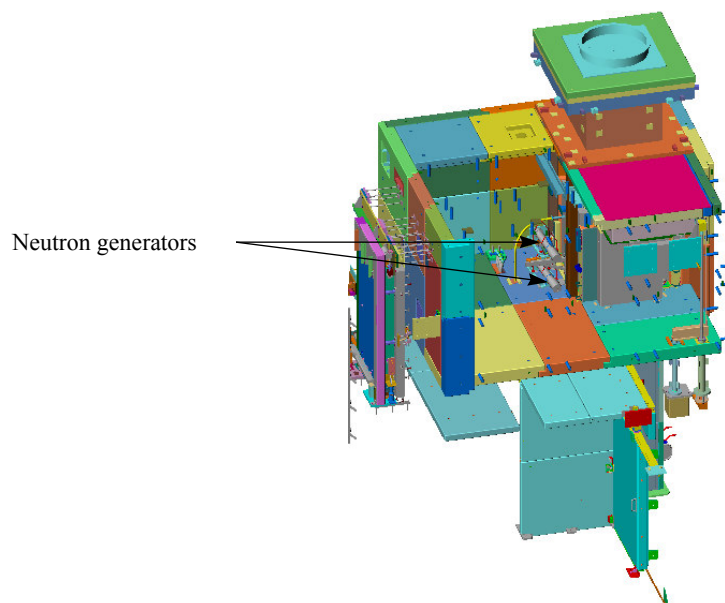

Neutron generators

Figure 3

4.2   CSD-C OUTGOING MEASUREMENT STATION

Figure 4 shows the general layout of the measurement station.

Neutron cell

Drum in gamma
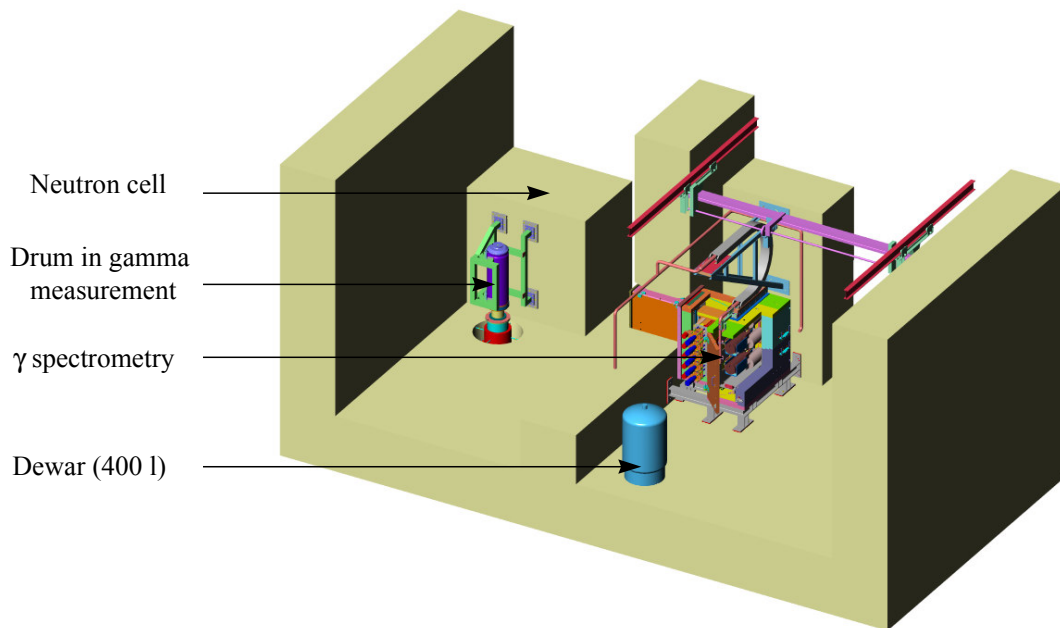measurement

γ spectrometry

Dewar (400 l)

Figure 4

4.2.1   Gamma Spectrometry

The gamma spectrometry station features the same design as the incoming measurement station described in 4.1.1. However, because of the CSD-C flows to be handled, five detectors have been installed to perform all CSD-C measurements in a single operation. A 2-min. preliminary measurement determines the total duration of the drum measurement and the detection limits for the specified peaks. Total measurement time ranges from 15 to 45 minutes.

4.2.2   Neutron Measurement

The neutron measurement station uses the same instrumentation as the station at the facility entrance. Since the CSD-Cs have a smaller diameter than the incoming containers, the detection blocks have been installed closer together, which provides a cell detection efficiency of 15%.

Figures 5 and 6 show the vertical and horizontal views of the measurement station.
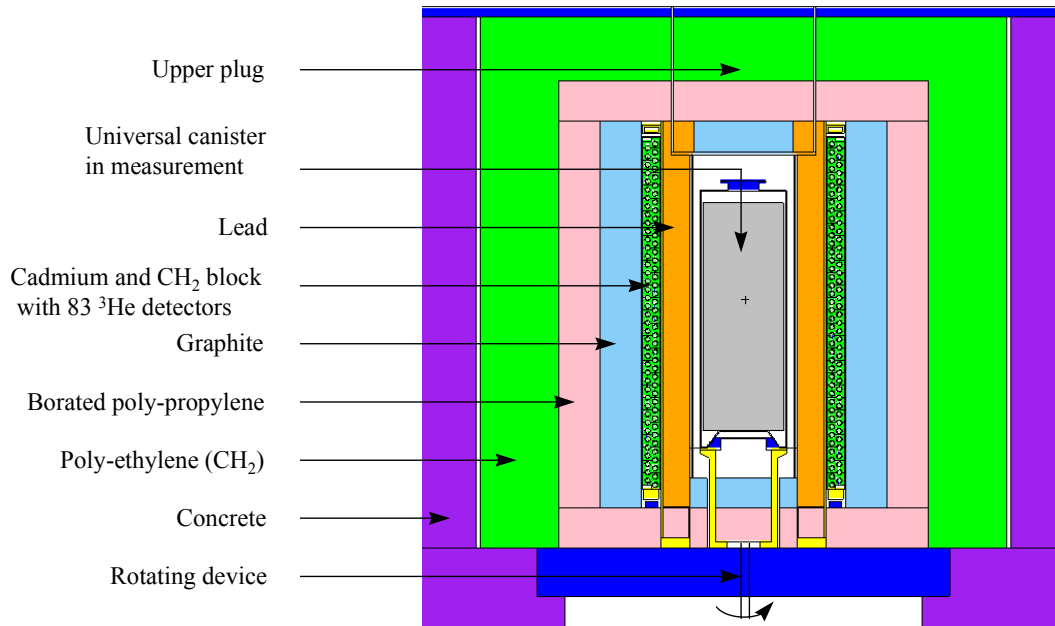
Upper plug

Universal canister
in measurement

Lead

Cadmium and CH_2 block
with 83 ^{3}He detectors

Graphite

Borated poly-propylene

Poly-ethylene (CH_2)

Concrete

Rotating device

Figure 5

Drum in measurement

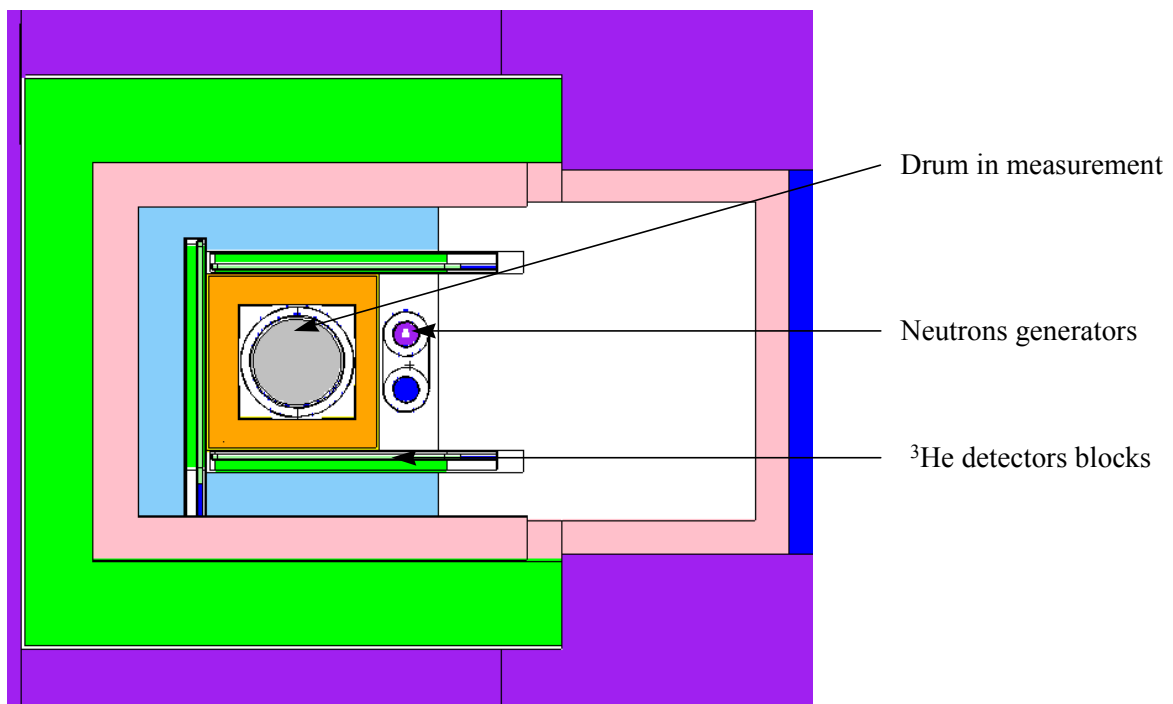Neutrons generators

^{3}He detectors blocks

Figure 6

The cell outer layers protect the neutron measurement section from gamma rays and neutrons emitted by CSD-Cs outside the measurement cell. A layer of concrete attenuates the gamma rays, a polyethylene layer slows down the neutrons and a borated polypropylene layer absorbs the slowed down neutrons. The measurement cell is therefore insensitive to outside radiation after the borated polypropylene layer.

The cell inner layers perform the following functions. The lead shielding was dimensioned to maintain the gamma dose rate at the ^{3}He neutron detectors below 2 rad/h. The graphite layer slows down the neutrons and was selected because it has a lower capture capacity than polyethylene. Once the epithermal neutrons have been slowed down, after going through a thin layer of cadmium, they enter the polyethylene detection blocks. Then the neutron produces the following reaction :

$$^3He + n => {}^1H + p + Q \qquad \text{where } Q = 765keV$$

The proton is detected. An electronic circuit with current amplifiers and a very fast acquisition card developed by EURISYS MESURES enable to count extremely rapidly each signal coming from one neutron. The detail of the signal interpretation is given in paragraph 5.2.2.2.

5. GENERAL MEASUREMENT INTERPRETATION METHODOLOGY

5.1 GAMMA MEASUREMENT INTERPRETATION

5.1.1 Gamma Measurement Interpretation Levels

Gamma spectrometry interpretation consists of two parts, referred to as Level 1 and Level 2.

Level 1 interpretation includes the following functions:

1. preliminary measurement to estimate the total measurement time required to achieve a specified statistical accuracy ;

2. signal acquisition and determination of an energy spectrum ;

3. extraction of the net areas (effective signal) of the specified gamma may ;

4. determination of the final statistical accuracy for each gamma may.

The calculations required for the above functions are performed on a dedicated signal acquisition PC. As a result, for each gamma may, the Level 1 interpretation produces a signal in counts per second and an uncertainty expressed as a percentage. These data are then input to a computer that performs the Level 2 interpretation.

The Level 2 interpretation includes:

1. evaluating the beta activity of the measurable radionuclides and the related uncertainties ;

2. evaluating, at the facility outgoing station, the average burn-up and cooling time of the irradiated fuel from which the compacted hulls and end pieces originate.

These operations are described in the next two sub-sections.

5.1.2 Evaluation of activities

For each of the five spectrometry areas, the activity of each specified gamma may is determined with the following equation :

$$A_i(X) = \frac{S_i(E)}{\varepsilon_i(E).TF_i(E).I(E)}$$

where the following notations are adopted :

- A_i is the activity of radionuclide X in area i (i ([1 ; 5]) ;

- E is the energy of the radionuclide X gamma may ;

- $S_i(E)$ is the net area of the peak energy E in counts/s, in area i, determined by Level-1 gamma spectrum processing ;

- $\varepsilon_i(E)$ is the detector intrinsic efficiency in area i at energy E, representing the ratio of the number of photons detected in the total absorption peak at the energy E and the number of photons arriving at the detector at this energy ;

- TFi(E) is the transfer function of area i, also called the geometric efficiency, representing the number of energy photons E arriving at the detector relative to the number of photons emitted in the waste package at this energy ;

- I(E) is the intensity of energy line E (number of gamma rays emitted at this energy through decay of a radionuclide X).

Total activity of a drum at the facility entrance or a CSD-C at the facility exit is determined by :

$$A(X) = \sum_{i=1}^{5} A_i(X)$$

The 95% relative uncertainty (IR(X)) for activity A(X) is determined as follows :

$$IR(X) = 2.\sqrt{\sum_{i=1}^{5} IR_i(X)}$$

where $IR_i(X)$ is the uncertainty for $A_i(X)$:

$$IR_i(X) = \sqrt{IR[S_i(E)]^2 + IR[\varepsilon_i(E)]^2 + IR[TF_i(E)]^2 + IR[I(E)]^2}$$

The IR values are the relative uncertainties for each variable $S_i(E)$, $\varepsilon_i(E)$, $TF_i(E)$, and I(E).

5.1.3 Burn-up and Cooling Time Evaluation

Burn-up (BU) and cooling time (CT) are two key variables that characterize the residual irradiated fuel in the hulls and end pieces. They are not known individually for each waste package, since waste in the process is not traceable. This information is, however, vital for determining the contaminant's isotopic composition to enable more effective interpretation of the neutron measurements (see 5.3 on combined interpretation measurements).

5.1.3.1 Cooling Time Evaluation

Using the CESAR code [1] , for all light water reactor fuel covered by the La Hague site reprocessing licenses, a correlation was established between the cooling time and the ratio of the ^{154}Eu and ^{134}Cs activities. This correlation is especially advantageous because the ^{154}Eu/^{134}Cs ratio is independent of burn-up. Because of there formation, ^{134}Cs and ^{154}Eu are two radionuclides which are proportional to the square of the burn-up. Their ratio is therefore virtually independent of burn-up. In addition, because the two radionuclides have different half-lives ($T_{1/2}$ = 8.6 years for ^{154}Eu and $T_{1/2}$ = 2.1 years for ^{134}Cs), their ratio is especially sensitive to cooling time.

A minimum CT correlation and a maximum CT correlation were determined for the entire range of fuel covered, as follows:

$$CT_{min} = c_{1\,min}.\ln\left(\frac{A(^{154}Eu)}{A(^{134}Cs)}\right) + c_{2min}$$

$$CT_{max} = c_{1\,max}.\ln\left(\frac{A(^{154}Eu)}{A(^{134}Cs)}\right) + c_{2max}$$

5.1.3.2 Burn-up Evaluation

In a manner similar to the method used for cooling time, burn-up was correlated for the entire light water reactor fuel as a function of the ^{154}Eu and ^{137}Cs activities ratio. Since ^{137}Cs is a direct fission product, its formation is proportional to burnup, while the ^{154}Eu formation from ^{152}Sm and ^{153}Sm is more proportional to $(BU)^2$. Hence, the following correlation is used for CT = 0 :

$$BU = c_3.\left(\frac{A(^{154}Eu)}{A(^{137}Cs)}\right)^{c_4} \quad \text{with } c_4 \approx 1$$

Since ^{154}Eu has a shorter half-life than ^{137}Cs, which has a half-life of 30 years, the above correlation is only valid for CT = 0. A cooling time correction determined from the equation in 5.1.3.1 is therefore necessary. This correction leads to the following equation :

$$BU = c_3.\left(\frac{A(^{154}Eu).2^{\frac{CT}{T_{Eu154}}}}{A(^{137}Cs).2^{\frac{CT}{T_{Cs137}}}}\right)^{c_4}$$

where T_{Cs137} is the half-life of ^{137}Cs and T_{Eu154} is the half-life of ^{154}Eu.

As for the CT correlation, the above correlation was established for the covered fuel range, which enabled defining a minimum and a maximum for each of the c_3 and c_4 values.

5.2 Neutron Measurement Interpretation

5.2.1 Interpretation Levels

As for the gamma measurements, the neutron measurements interpretation consists of two parts, referred to as Level 1 and Level 2. Level 1 interpretation includes the following functions:

1. signal acquisition for the passive and active neutron measurements ;

2. extraction of signals after preliminary measurements and estimate of required total measurement times ;

3. extraction of effective signals after complete measurements ;

4. determination of statistical accuracy with the effective signals ;

5. determination of the fissile mass ensured for criticality safety.

Level 2 interpretation includes the following functions :

1. interpretation of passive neutron measurements to determine drum or CSD-C neutron emission ;

2. interpretation of active neutron measurements to determine the ^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu masses.

The next two sub-sections detail each of these interpretation levels.

5.2.2 Level 1 Interpretation

5.2.2.1 Passive Neutron Measurement Interpretation

Passive neutron measurement interpretation is used to determine drum or CSD-C Neutron Emission (NE in n/s) with the following equation :

$$\text{NE} = \frac{PM}{\varepsilon}$$

where the following notations are adopted :

- PM is the Passive neutron Measurement in counts/s ;

- ε is the measurement cell passive efficiency in counts/neutron emitted by the drum.

Efficiency is determined during measurement cell calibration.

5.2.2.2 Active Neutron Measurement Interpretation

Active neutron measurement is performed with neutron pulses emitted by a generator. The innovative technology used for the neutron generator is described in 6.2. The following paragraphs describe the active neutron interrogation technique employed.

The neutron generator emits pulses of around 200 μs every 10 ms (Figure 7). The emitted neutrons create fission reactions in the drum and the resulting neutron emissions are sensed by ^{3}He detectors.
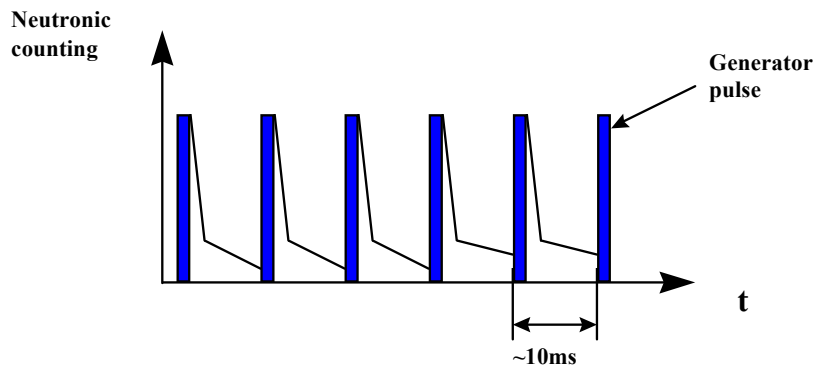
Figure 7

For each 10 ms cycle, the neutron count versus time is stored in a register on a fast counting card. Total measurement time ranges from 15 to 45 minutes As a result, more than 100,000 measurement cycles are added together for signal acquisition, as shown in Figure 8.
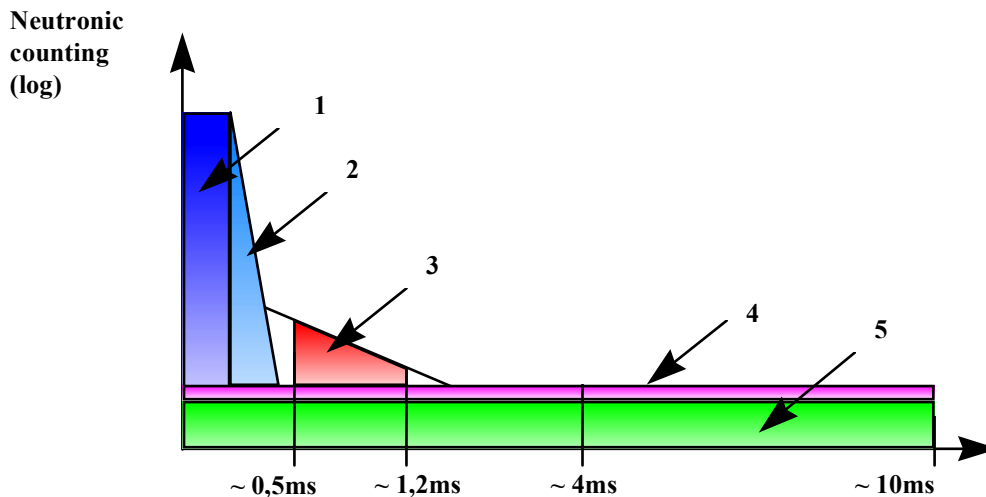
Figure 8

Each of the five neutron types in Figure 8 corresponds to :

1. Neutrons from the generator which overload the detectors because of their high intensity.

2. Neutrons from the generator as they are slowed down and prompt neutrons emitted by fast ^{238}U fissions. This signal area cannot be exploited because of the neutron background noise from the generator.

3. Prompt neutrons produced by ^{235}U , ^{239}Pu, and ^{241}Pu thermal fissions.

4. Delayed neutrons from thermal and epithermal ^{235}U , ^{239}Pu, and ^{241}Pu fissions, and fast ^{238}U fissions.

5. Passive neutron emission and active background.

Signals are extracted as described in the following paragraphs.

A specified target range of 4 ms to 10 ms enables determining the number of delayed neutrons from the passive emission neutron determination, as described in 5.2.2.1. A target range of 0.5 ms to 1.2 ms then permits determining the number of prompt neutrons.

Since the prompt neutron counting is more sensitive to Pu, the measurement cell was optimized so the delayed neutron counting would be sensitive, as a maximum, to ^{238}U. As a result, the two measurements can be used to separate the U and Pu contributions. This optimization enables prompt and delayed neutron counting to separate the U and Pu contributions not only with one measurement cell but also in a single operation. The patented optimization method [3] employed is detailed in 6.1.

The prompt neutron signal also enables determining the fissile mass in a drum. A number of safety coefficients are applied to this fissile mass to ensure compliance with thresholds specified in the safety studies conducted for each measurement station.

5.2.3 Level 2 Interpretation

The prompt and delayed neutron measurements enable writing a linear system comprising two equations with four unknowns :

$$PNS = CP(U5). MU5 + CP(Pu9). MPu9 + CP(Pu1). MPu1$$

$$PDS = CD(U5). MU5 + CD(U8). MU8 + CD(Pu9). MPu9 + CD(Pu1). MPu1$$

where the following notations are adopted :

- PNS is the prompt neutron signal and PDS is the delayed neutron signal in counts/s ;

- The CP coefficients are the prompt neutron calibration coefficients in counts/s/g of each ^{235}U , ^{239}Pu and ^{241}Pu isotope ;

- The CD coefficients are the delayed neutron calibration coefficients in counts/s/g of each ^{235}U , ^{238}U , ^{239}Pu and ^{241}Pu isotope ;

- MU5, MU8, MPu9 et MPu1 are the masses in grams, respectively, of ^{235}U , ^{238}U , ^{239}Pu and ^{241}Pu.

Two additional equations are used to solve the system.

The MU5/MU8 ratio ranges between 0.5% and 1.5% for the fuel assemblies received at the La Hague site. In addition, the MPu1/MPu9 ratio affords a linear correlation with the ratio of the ^{154}Eu and ^{137}Cs activities throughout the range of UOX fuel assemblies from light water reactors. Like the others, this correlation was determined using the CESAR code.

5.3   COMBINED NEUTRON/GAMMA MEASUREMENTS INTERPRETATION

Combined interpretation of neutron and gamma measurements is used to determine CSD-C characteristics at the compacting facility's outgoing measurement station. The measurements are interpreted in three blocks:

1. interpretation of gamma measurements ;

2. interpretation of neutron measurements ;

3. combined neutron/gamma measurements interpretation to determine CSD-C activities in terms of masses and decay heat.
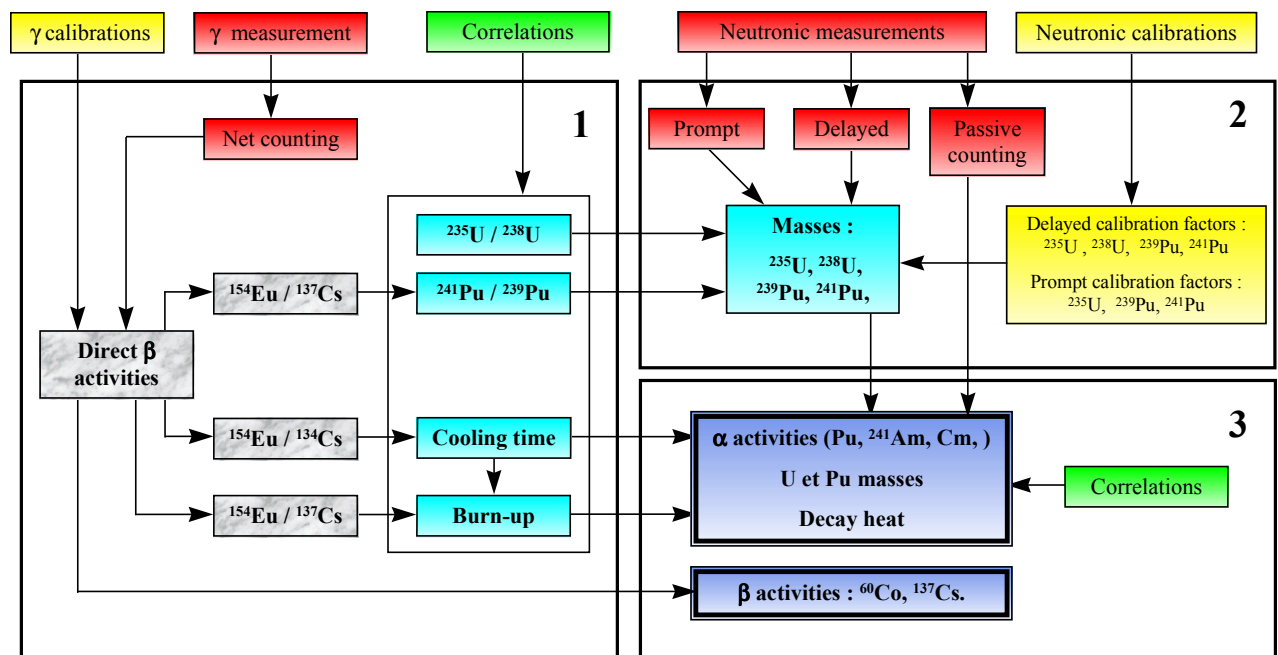

Figure 9

The diagram in Figure 9 summarizes the three measurement interpretation phases. Below is an example of how activity is determined.

Using the CESAR code, a correlation was established between the plutonium alpha activity ratio (AαPu) and burn-up. This correlation is of the form :

$$\frac{A\alpha Pu}{MPu9} = c_5 . \exp\left(c_6 . BU\right)$$

As for all the others, this correlation was determined for the fuel range covered, which enabled defining a minimum and maximum for each c_4 and c_5 value.

Burn-up is determined from interpretation block 1 (as described in 5.1.3.2.); ^{239}Pu mass is determined from block 2 interpretation (as described in 5.2.3.).

The following variables are subjected to a similar characterization :

- Pu alpha activity ;

- ^{241}Pu beta activity ;

- Alpha activity of the emitters with a half-life of more than 50 years ;

- Curium alpha activity ;

- ^{137}Cs beta activity ;

- ^{90}Sr-^{90}Y beta activity ;

- Thermal power.

A mean value and a related uncertainty are determined for each variable. An innovative uncertainty propagation technique was developed, as detailed in 6.3.

6. MAIN INNOVATIVE FEATURES

6.1 OPTIMIZED NEUTRON MEASUREMENT CELL DESCRIPTION

As described in 5.2.2.2, the measurement cell was optimized particularly to separate the U and Pu contributions not only using a single measurement cell but also in the same operation. Unlike a conventional active neutron measurement cell, the optimized cell was dethermalized to detect the ^{238}U contribution to the delayed signal. Accordingly, as shown in Figure 6, no thermalizing material was placed behind the two neutron generators. This cell design enables eliminating the ^{235}U proportion in the prompt signal, which helps to improve the plutonium 0mass assessment and determine the waste package's activity alpha.

However, this increase of the ^{238}U signal proportion in the delayed signal substantially reduced the fission rate in the cell. A new type of generator (GENIE 36) therefore had to be developed with 20 times higher neutron emission than previous generators (GENIE 26).

6.2 NEW NEUTRON GENERATOR DEVELOPMENT

A new neutron generator, the GENIE 36, has been designed by SODERN on the requirements issued by SGN for this specific application. The development has been done after the experience obtained by COGEMA and SODERN with the previous GENIE 26.

The main characteristics are the following ones :

- average neutron output : 2×10^9 n/s (14 MeV neutrons),

- life-span for the tube : 1500 hours of running,

- dark emission between pulses : less than 50 n/s,

- high level of reliability and no need of routine conditioning.

A new neutron tube has been designed to meet the challenge. It is equipped with a coaxial ion source which delivers a spread beam of deuterons onto the target, in order to reduce its erosion. The walls are made of alumina for a better electrical insulation. It works under an accelerating voltage of 140 kV, delivers neutron pulses with a duration in the 100 μs range, and is cooled with circulating oil.

The tube with its electrical and mechanical interfaces is connected to the ancillaries through 15-meter cables and pipes, and the equipment is fully controlled with an external computer using the specific COGEMA protocol. The control unit of the GENIE 36 can control one or two equipment, which then deliver their neutron pulses synchronously.

The first tests performed by CEA have validated the foreseen characteristics.

6.3 INNOVATIVE UNCERTAINTY PROPAGATION TECHNIQUE SELECTION

As described in section 5, the entire measurement interpretation algorithm (Figure 9) is based on a combined gamma spectrometry and neutron measurement interpretation. The algorithm uses numerous measurements and correlations. Uncertainty calculations for each variable, and the propagation of this uncertainty throughout the algorithm with the standard variance propagation method, proved to be one of the difficulties of the project. A novel method was therefore applied to overcome this difficulty.

6.3.1 Principle of the Method

Instead of assigning a mean value and a standard deviation to each variable, each variable is considered as a distribution in the mathematical sense of the term (a function in discrete elements in n intervals with a probability associated with each interval).

Accordingly, all the measurements are expressed in discrete elements in the form of a Gaussian function and the number of intervals n was optimized at 21. The algorithm was divided up into subsystems and a rigorous, systematic propagation of the uncertainties was implemented for each subsystem. Accordingly, a simple multiplication of two variables a and b entails a multiplication of two distribution laws in the algorithm or a total of $n2=441$ operations. When the subsystem includes p variables, the total number of operations is therefore np.

The general algorithm was divided up into subsystems with less than five variables, except for block 2 shown in Figure 9. This block, which cannot be split up, contains 11 variables (two measurements, two correlations, and seven calibration coefficients). Since it was not feasible to achieve all possible combinations for block 2, processing using the Monte-Carlo method was implemented for this block. An extremely fast convergence of the algorithm was observed after less than 50,000 runs. Details of the algorithm are presented in Reference 2. An example of its implementation is, however, provided below.

6.3.2 Uncertainty Propagation Example

The uncertainty propagation technique employed is explained with the example of plutonium alpha activity evaluation for a block 3 subsystem (Figure 9).

We first consider the results obtained from block 2, which indicate the general outline of the ^{239}Pu mass distribution law (Figure 10). It should be noted that, for this figure which was determined using the preliminary version of the software that SGN developed to optimize the variables, n is greater than 21.
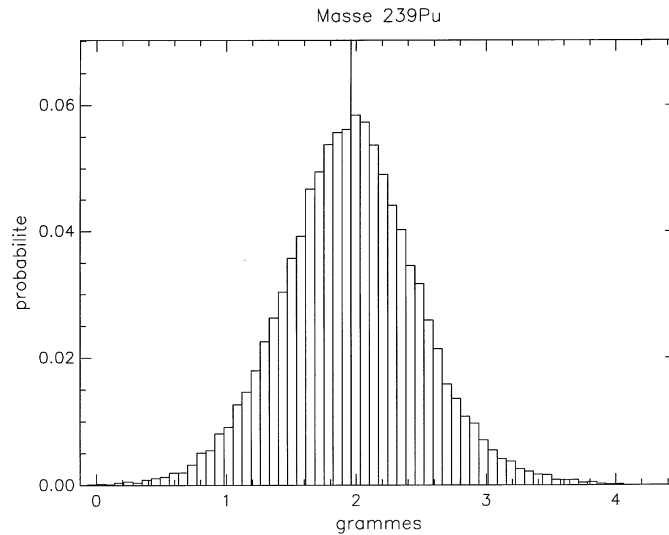

Figure 10

Then we consider the burn-up distribution law, which is obtained from block 1 results (Figure 11).
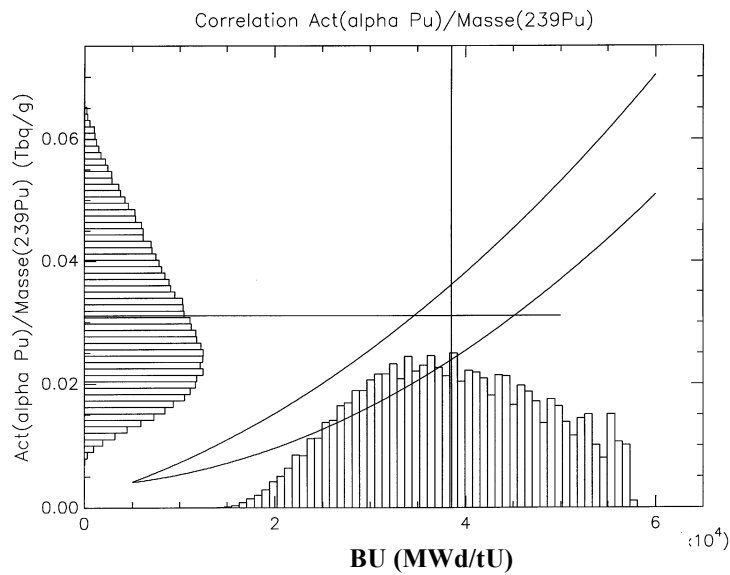

Figure 11

We see that the burn-up distribution deviates somewhat from a Gaussien function, which indicates that the method of systematic uncertainty propagation using distribution laws is far more rigorous than the variance propagation method. Due to the unknown covariances terms, an uncertainty propagation method using distribution laws has been selected.

Between the two exponential curves designated f_{min} and f_{max}, the graph in Figure 11 also shows the validity range of the correlation which links the R = (Pu alpha activity)/(^{239}Pu

mass) ratio and burn-up. The values of R may range between $R_{min} = f_{min} (BU_{min})$ and $R_{max} = f_{max} (BU_{max})$. BU_{min} is the mean burn-up on the first of n intervals and BU_{max} is the mean burn-up on the last of n intervals.

The range of R between R_{min} and R_{max} is split up into n registers called R_i.

For each abscissa, which corresponds to an interval xi of the burn-up distribution law, the y validity range of the correlation is discretized into n isoprobable intervals, each of which is designated y_i.

Let $P(x_i)$ be the normalized probability that corresponds to the interval xi burn-up. For each y_i, interval the mean corresponding value of R falls within one of the R_j registers. The value (1/n) x $P(x_i)$ is then added in register Rj. This operation is performed for all n y_i intervals of each of the n x_i intervals, which enables construction of the histogram in ordinates as shown in Figure 11.

The histogram R = (Pu alpha activity)/(^{239}Pu mass) is then cross-correlated with the ^{239}Pu mass histogram to determine the Pu alpha activity distribution law.

7. EXPECTED PERFORMANCES

7.1 DETECTION LIMITS

Depending on the passive neutron emission the detection limits are :

- 100 mg of ^{235}U + ^{239}Pu + ^{241}Pu for the prompt neutrons

- 100 g of ^{235}U + ^{238}U + ^{239}Pu + ^{241}Pu for the delayed neutron ($\Leftrightarrow \approx$1g of ^{235}U + ^{239}Pu + ^{241}Pu)

7.2 UNCERTAINTIES

7.2.1 Measurement and calibration factors

The statistical uncertainties coming from the measurement will be :
    $\approx$ 5% for the prompt neutron measurement
    $\approx$ 10% for the delayed neutron measurement

For the calibration factors the uncertainties will be between 15% and 30%.

7.2.2 Masses and activities

Concerning the 4 masses calculated in the block two of interpretation (see figure 9 in chapter 5.3.), the uncertainties will be for ^{238}U and ^{239}Pu $\approx$50%, and for ^{235}U and ^{241}Pu $\approx$100% because of the uncertainties coming from the correlation in block 1.

Depending on the initial caracteristics of the hulls, the other activities will be determined with the following uncertainties

Activity	Uncertainty
Alpha Cm	≈20%
Beta ^{241}Pu	≈100%
Alpha Pu	≈100%
> 50 years alpha emitters	≈100%
Beta ^{137}Cs	≈30%
Beta ^{90}Sr-^{90}Y	≈50%

The Cm is directly detected by the passive neutron emission. As a result the uncertainty will be the uncertainty of the measurement which is about 20%.
The beta activity of ^{241}Pu will have the uncertainty of the ^{241}Pu mass evaluation (≈100%).
The alpha activity of Pu and of alpha emitters with a > 50 years half time will be determined with an uncertainty of ≈100% which comes from the uncertainty of the ^{239}Pu mass evaluation (≈50%) and from the correlation.
The beta activity of ^{137}Cs is directly measured, the uncertainty will be around 30%. The Beta activity of ^{90}Sr-^{90}Y will be determined with a correlation from ^{137}Cs. The uncertainty will be around 50%.

8. CONCLUSION

Based on joint developments by COGEMA and CEA, the neutron and gamma measurement implemented in the hulls and end pieces facility are very innovating.
The innovations concern :

- the optimization of the neutron measurement cell to separate U and Pu signal contribution ;

- the development of a new neutron generator with an emission of 2.10^9 n/s ;

- the development of a combined gamma spectrometry and neutron measurement interpretation ;

- the development of an innovative uncertainty propagation technique.

The expected performances, will be confirmed by the measurement of the first drum in this facility before the middle of the year 2001.

9. REFERENCES

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