Measuring, Cutting and Sorting Irradiated Control Rods to Limit the Volume of Intermediate Level Long-lived Waste - 17001

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ABSTRACT

During operation, the PWR control rods are either completely extracted (scram or safety rods) or partially inserted in the operating zone of the core (fine control or operating rods). At the nominal rating conditions or close to them, the operating zone is limited to some steps of insertion in the order of a few tens of cm. In fact, only the lower part of the control rods are subjected to a significant neutron flux with an impact on the neutron activation and so on the waste classification (Low Level Short-lived Waste versus Intermediate Level Long-lived Waste).

Our calculative as well as experimental experience confirms the significant presence of radionuclides in the control rods, but only in the lower parts. Measuring, cutting and finally sorting control rod scraps can reduce the volume of Intermediate Level Long-lived Waste planned for our storage deposit facility.

This paper presents the EDF/DP2D methodologies followed to determine the sort criteria between the Low Level Short-lived Waste versus the Intermediate Level Long-lived Waste zones for the Silver-Indium-Cadmium control rods.

The irradiated control rods are stored in cases without their head. Each case can contain several control rods, and the cases are stored under water in the fuel assembly pool.

To calculate the radioactive inventories, we have developed a calculation scheme which is composed of a 3-dimensional mapping of the neutron flux, a calculation of the evolution of the isotopic compositions under neutron flux and finally a comparison between calculated and measured values. The main hypotheses of the control rods calculation take into account :

- 3-dimensional geometry and 3-dimensional neutron sources distribution,
- the isotopic compositions including impurities,
- the history of irradiation resulting mainly from the daily power production, the positions of the control rods in the loading patterns and the positions in the axial operating zone.

In order to validate the calculations, measurements were made 20 meters under water in the fuel assembly pool. The experimental results show the presence of many gamma emitter radionuclides. Only Ag-108m was retained for validating the calculations. This is due to the fact that silver is considered as the "standard"

chemical element for our control rods and silver produces Ag-108m which is a long-lived gamma emitter radionuclide.

The analysis of the calculated results and the measured values allow us to build a simplified physics model for the simulation of the control rods activation.

Finally three sort criteria can be proposed :

- Without measurement devices, the sorting is based on an empirical sort criterion. Cuttings must then be made on the 2 back-ends of the control rod pins. This option is due to the difficulty in knowing the orientation of the pins in the storage cases. The cutting lines are placed at approximately 1 m from the 2 back-ends of the pins. This approach does not allow a great optimization of the sorting, less than 50% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.
- With a simple dose rate measurement device, the knowledge of the orientation of the pins in the storage cases allows us to limit cuttings to the lower part of the control rods. Depending on the "top to tail" cases, between approximately 50% and 75% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.
- With an accurate measurement device, the sorting can be based on a specific activity of Ag-108m. Over 4400 Bq/g of Ag-108m (corresponding to the Intermediate Level Long-lived Waste limit), the control rod pieces can be considered as a long-lived waste. The measurements must be made over the 2 back-ends of control rod pins to examine the cases of "top to tail" storage. This approach allows a great optimization of the sorting. Depending on the history of irradiation, more than 75% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.

INTRODUCTION

The knowledge of the radionuclide content of radioactive waste is of utmost importance for safety and waste management reasons.

State of EDF nuclear reactor fleet

The EDF nuclear reactor fleet consists of fifty eight pressurized water reactors (PWRs) spread across nineteen sites. Currently, three types of reactor can be noted, thirty four "900 MWe", twenty "1300 MWe" and four "1500 MWe".

As EDF is the only civil nuclear operator in France, according to French law, EDF is responsible for dismantling their power plants. EDF is also responsible for managing the operating waste like the control rods. Numerical simulations are used by DP2D (i.e. decommissioning branch of EDF) to anticipate dismantling and radioactive waste management. For this, DP2D developed a calculation scheme including the mapping of the 3 dimensional neutron flux, and radioactive inventory calculations (Ref. [1]).

Radioactive waste management in France

A waste classification can be set according to the radioactive inventory of each component or subcomponent and the waste classification criteria.

In France, five classes of nuclear waste are defined. Each refers to a particular level of specific activity and radio-toxicity. The different classes from the least to the most penalizing are : “Very Low Level”, “Low Level Short-Lived”, “Low Level Long-Lived”, “Intermediate Level Long-lived”, and “High Level Long-Lived”.

Typically it is necessary to know the radioactive inventory in terms of specific activity (Bq/g) based on a list of 143 radionuclides.

Due to the types of particles emitted, most of these radionuclides cannot be measured, this is why numerical simulations must be made. Then during the dismantling work, gamma measurements are made on the wastes. Depending on the measured values, the results of the calculated radioactive inventory can be corrected.

Issues

Table I presents the quantity of control rods estimated for the French NPP fleet based on average hypothesis of forty years operating for each reactor and ten years under flux for the controls rods.

Table I
Estimation of the French NPP control rods quantity (number of pins & weight)

	900 MWe 34 plants	**1300 MWe** 20 plants	**1500 MWe** 4 plants	**sum** 58 plants
stainless steel [weight]	22 000	15 000	1 000	38 000 [100 t]
Ag-In-Cd [weight]	157 000	8 000	1 000	166 000 [450 t]
Ag-In-Cd/B_4C [weight]	0	102 000	25 000	127 000 [450 t]
			global (weight)	331 000 [1000 t]

With the storage capacity of 2 metric tons per cask (that means approximately 500 “Intermediate Level Long-lived” casks) it’s easy to understand the interest of optimizing and reducing the quantity of “Intermediate Level Long-lived” waste to the absolute minimum.

Classical calculations of the radioactive inventory

The main numerical activation schemes comprise a calculation of the 3 dimensional neutron flux map obtained on the basis of neutron propagation calculations. The codes used solve the transport equation called the Boltzmann equation without simplification. The Monte Carlo method is retained to solve the Boltzmann equation. Random series of numbers are used to simulate the lives of millions of neutrons. The codes follow each neutron individually, from birth to disappearance by leakage, absorption, or fission. Then the neutron flux map is calculated at the nominal power rating conditions.

The main drawback to the Monte Carlo codes is the large computing time needed to converge within acceptable statistical criteria. This computing time is incompatible with the multiple fuel managements used by EDF to operate its NPPs because a Monte Carlo calculation is necessary for each fuel loading pattern.

The solution developed by EDF-DP2D is to use the importance factors of Green's Functions to simplify the computing of the 3-dimensional neutron flux maps.

Numerical activation scheme using Green's Functions

Figure 1 presents the numerical simulation scheme based on Green's Functions approach used by EDF-DP2D to calculate the activation by neutron flux (Ref. [2]).

This scheme comprises five steps :

- Step 1 : Computing of 3-dimensional response matrices to unitary sources. These matrices (i.e. normalized source contributions or importance factors of Green's Functions) are built for each tally of interest. The code used is MCNP (Ref. [2]). It was developed by the American Los Alamos National Laboratory to solve the Boltzmann equation. The input data covers the microscopic cross-sections, the 3-dimensional geometry, the chemical compositions with no impurities and the neutron unitary sources emitted by the fuel assemblies.
- Step 2 : Computing of a 3-dimensional multigroup neutron flux map. The neutron flux mapping is btained on the basis of the convolution of the response matrices to unitary sources (step 1) with the neutron sources emitted by the fuel assemblies. The neutrons emitted by the fuel assemblies are computed with the core code COCCINELLE (deterministic code developed by EDF). The matrix convolution is made by using the Python language. The neutron flux maps are calculated for the nominal power rating conditions, and each flux is homogenized into a limited number of energy groups.
- Step 3 : Calculation of the activities. The activities are calculated for each component or sub-component of interest. The code used is DARWIN-PEPIN (Ref. [3]). It was developed by the French laboratory Commissariat à l'Energie Atomique (CEA) to solve a system of Bateman equations. The input data covers the 3-dimensional neutron flux maps calculated in step 2, the microscopic cross sections, the radioactive decay series associated with the radioactive half lives, the chemical compositions including impurities and the

irradiation history resulting from the daily power production. The output data is the radioactive inventory of each component or sub-component of interest limited to a list of 143 radionuclides.

- Step 4 : Comparisons between calculated results and measured results. For this, we need results of measurements and results of calculations linked with the measurements. The comparisons are based on calculation measurement ratios (i.e. “C/M”). A value greater than 1 corresponds to an overestimated calculation, and a value less than 1 corresponds to an underestimated calculation. Depending on the results, the input data may be redefined to make a new simulation.
- Step 5 : Waste classification. By using the radioactive inventory of each component or sub-component, and the waste classification criteria, a waste classification can be made. The criteria are based on specific activity and radiotoxicity levels of 143 radionuclides. The distinction between the “Long-lived” and the “Short-Lived” waste is based on a list of specific activity limits for 40 radionuclides. If none of these limits are crossed, a weighted specific activity level is used to separate “Very Low Level” and “Low Level Short Lived” waste. The weighted specific activity value is obtained by taking into account the specific activity levels balanced with the levels of radio-toxicity of 143 radionuclides.

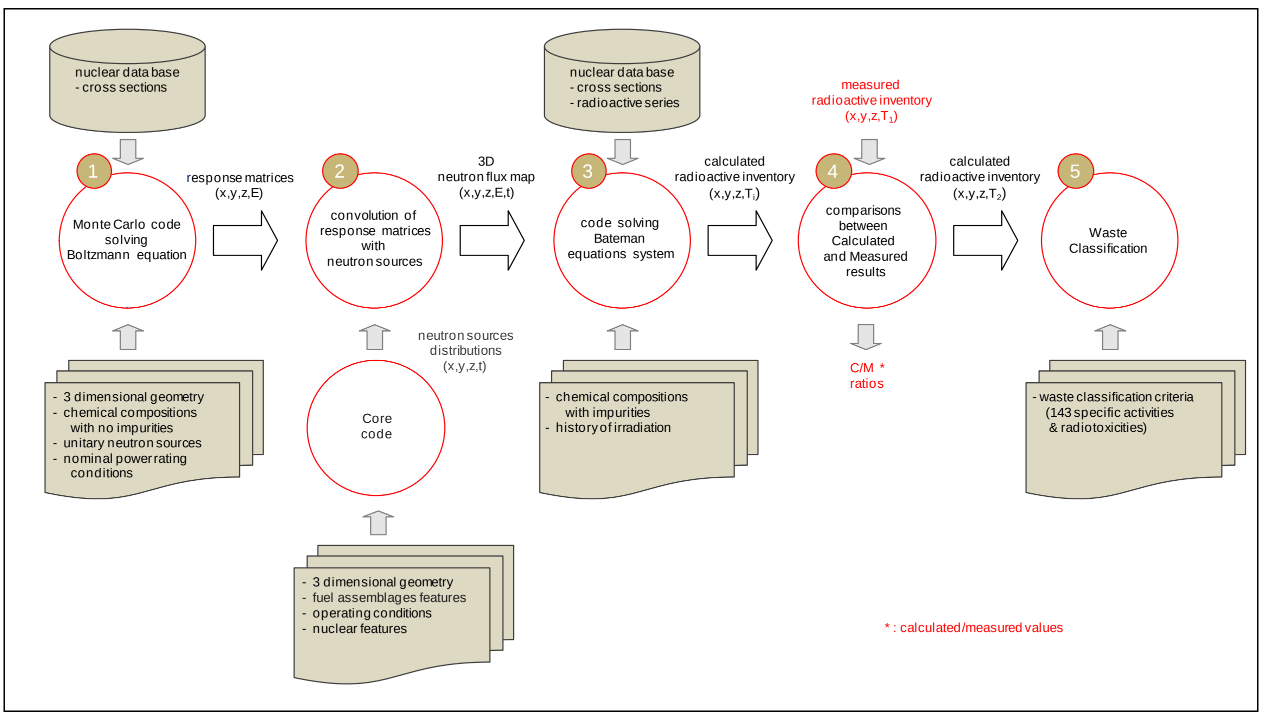

FIGURE 1

Numerical activation scheme based on Green’s Functions

RESULTS

This paper refers to "Bugey 2" a "900 MW" PWR plant. The results concern Silver Indium Cadmium control rods.

Measured values

As examples, figure 2 shows measured values. The measurements were made 20 meters under water in the fuel assembly pool. The experimental results provide the presence of many gamma emitter radionuclides (mainly Co-60, Ag-108m and Ag-110m).

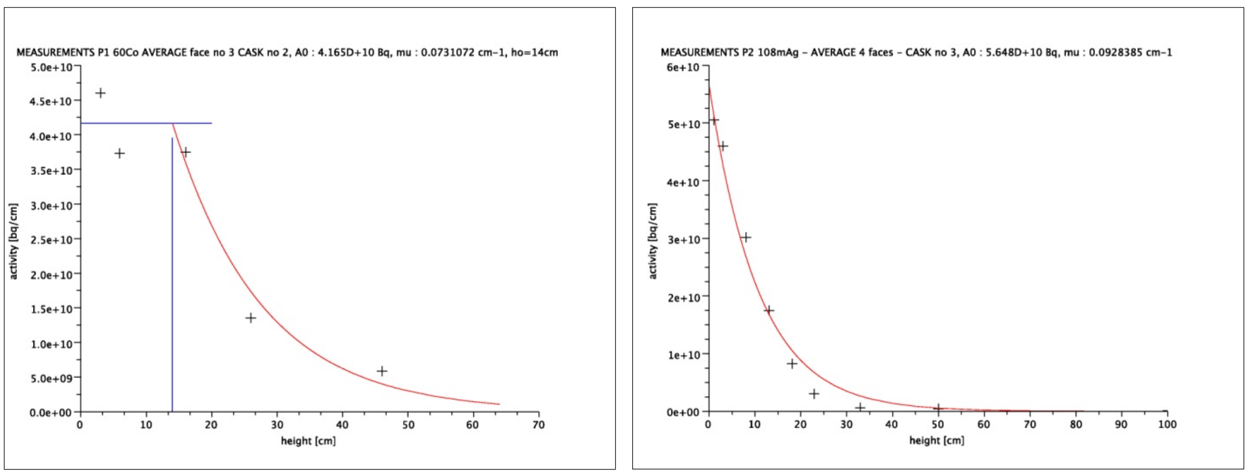

FIGURE 2
Examples of measured values on a storage cask

Principal hypothesis of the simulations

A 3-dimensional geometry was taken into account in the Monte Carlo code to compute the mapping of the neutron source contributions (see Figure 1, step 1). Depending on the tally localization, the fuel assemblies are described pin by pin or homogenized. If the propagation distance is high ($\gg$1 meter) the fuel must be homogenized to reduce the simulation time and the output data size. Therefore, the set of non-external fuel assemblies is homogenized for calculations concerning the vessel. In the same way regarding the control rods, the bottoms of the fuel assemblies are homogenized. All the internals are fully described 3-dimensionaly, including the different plates, the control rods and the guide tubes. The security absorbers are fully extracted from the core, whereas the control rods are partly extracted from the core (i.e. lower boundary operating zone). Three temperatures are defined for the primary water (core inlet, fuel assembly zone, core outlet).

The neutron sources calculated with a core code (see Figure 1, step 2) are given pin by pin according to axial distributions.

The isotopic compositions are reduced to the major chemical elements to calculate the neutron flux because impurities or traces do not affect the neutron propagation code (see Figure 1, step 1). However, to calculate the activation (see Figure 1, step 3), it is necessary to use the complete chemical compositions including impurities and traces as these minor impurities can directly impact the waste classification.

The flux mapping is calculated at the nominal power rating conditions (see Figure 1, step 2) while the isotopic evolution uses the irradiation history (see Figure 1, step 3). However this history was simplified to a limited number of steps, corresponding to the fuel campaigns.

Validation of the use of Green's Functions approach

An overall validation of the numerical simulation scheme can be made by using the activation of "standard" chemical elements (Ref. [1]). Certain criteria must be satisfied in order to qualify a chemical element as "standard" : high concentration, activation into large cross section range, low concentration uncertainties, production of radionuclide having significant half life (greater than several months) and no difficulties measuring the radionuclide produced.

In order to optimize the use of the fuel assemblies and to limit the vessel neutron fluence (i.e. neutron flux integrated in a period of time), different fuel loading patterns were defined for each reactor. Each fuel loading pattern is linked with a particular 3 dimensional power distribution. The radioactive inventories of the vessel and the internals due to neutron flux activation are a result of the history of the different power distributions. The main point in using the importance factors of Green's Functions is that the normalized source contributions matrices can be used for all the power distributions and thus reducing the Monte Carlo computing time.

The validation of the use of Green's Functions approach is based on the activation of "Bugey 2" control rods (Ref [2]). The control rods are composed of a mixture of "Silver", "Indium" and "Cadmium". "Silver" was retained as the "standard" chemical element, and the γ emitter measured by γ spectrometry is the Ag-108m ($T_{1/2}$ = 418 years).

Seven control rod storage cases were measured. Each case contained between 5 and 11 control rods. Before storage, the control rods were irradiated during 3 to 10 fuel cycles, in different core positions (i.e. radial positions), at different depths in the core (i.e. axial positions) and in different periods of time. All of these parameters were used in our simulations.

Table II provides the Calculated/Measured values (C/M) relative to the 7 measured cases. The key parameters also are presented in this table. It can be noted that the numerical simulation produces a slight overestimation of the Ag-108m inventory (+30%).

Table II
Calculation/Measurement ratios

case n°	number of rods	average irradiation time [years]	average cooling time [years]	Ag-108m C/M
1	11	11	23	1.0
2	8	10	8	1.0
3	5	4	18	1.5
4	11	11	25	1.7
5	10	9	14	0.9
6	11	11	22	1.3
7	11	8	17	1.2
			average value	1.3

The reasons for these overestimations are directly linked to the measurement uncertainties, the nuclear data uncertainties, the Monte Carlo statistical uncertainties and the hypotheses retained for the irradiation conditions history (like the position of the control rods, the temperature of the primary water, the average monthly power level, etc). However, it is difficult to give a weighting to each of these parameters.

DISCUSSION

Physical model of the radiological activity axial distribution

The analysis of the measured values (see Figure 2) allows us to propose a simplified physical model of the control rod activity axial distribution.

Close to the back-end of the control rod (meaning the part inserted in the core for operating), the activity $A(\square\square x\square\square)$ can be considered as constant :

$$A(\square\square x\square\square) = A0$$
(Eq. 1)

where :

x = the distance from the bottom of the control rod

$A0$ = the activity due to the insertion of the rod in the core for a long period of time

Above the constant part (meaning the part extracted from the fuel assembly zone), the activity $A(\square\square x'\square\square)$ can be considered as decreasing exponentially :

$$A(\square x'\square\square) \quad = A0 \,.\, \exp(-\frac{\ln(2)}{l_{1/2}} x')$$

(Eq. 2)

where :

x' = the distance out of the fuel assembly zone

$A0$ = the constant activity due the insertion of the rod in the core

$l_{1/2}$ =the half distance (i.e. distance that halves the activity)

This simplified model can be used to define the empirical sort criterion.

Based on the penalizing hypothesis of control rods permanently inserted in the fuel assemblies operating zone (i.e. 24 cm corresponding to the middle of the control rod operating zone), a half distance of 6.3 cm and the higher values measured on the seven Bugey casks, the empirical sort criterion can be placed at 102 cm from the bottom of the control rods.

Figure 3 illustrates the basis of our simplified physical model of control rods activation.

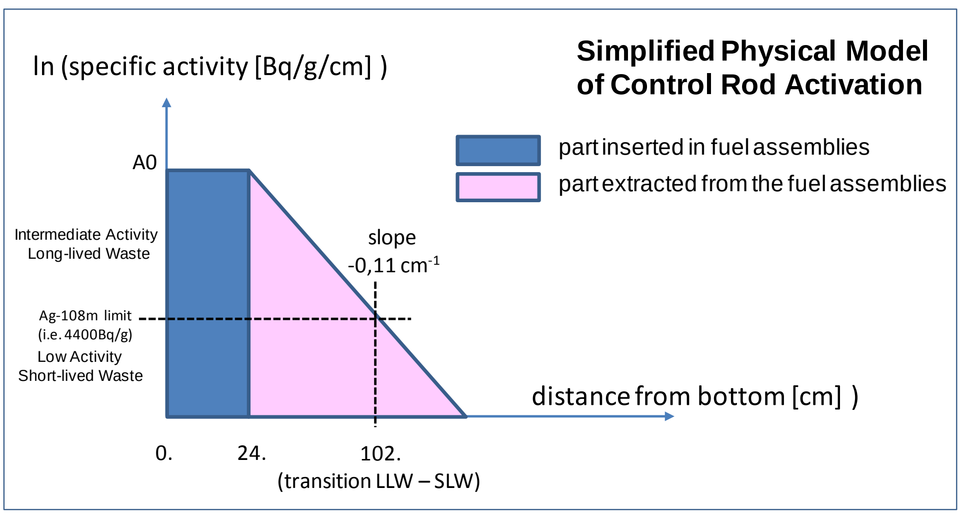

FIGURE 3
Simplified Physical Model

Sorting criteria

Three sort criteria can be proposed :

- Without measurement devices, the sorting is based on an empirical sort criterion. Cuttings must then be made on the 2 back-ends of the control rod pins. This option is due to the difficulty in knowing the orientation of the pins in the storage cases. The cutting lines are placed at approximately 1 m from the 2 back-ends of the pins. This approach does not allow a great optimization of the sorting, less than 50% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.
- With a simple dose rate measurement device, the knowledge of the orientation of the pins in the storage cases allows us to limit cuttings to the lower part of the control rods. Depending on the “top to tail” cases, between approximately 50% and 75% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.
- With an accurate measurement device, the sorting can be based on a specific activity of Ag-108m. Over 4400 Bq/g of Ag-108m (corresponding to the Intermediate Level Long-lived Waste limit), the control rod pieces can be considered as a long-lived waste. The measurements must be made over the 2 back-ends of control rod pins to examine the cases of “top to tail” storage. This approach allows a great optimization of the sorting. Depending on the history of irradiation, more than 75% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.

CONCLUSION

Knowing the radionuclide content of radioactive waste is of utmost importance for safety and waste management reasons. Numerical simulations are used by EDF-DP2D to anticipate dismantling and radioactive waste management.

Our calculative as well as experimental experience confirms the significant presence of radionuclides in the control rods, but only in the lower parts. Measuring, cutting and finally sorting control rod pieces can reduce the volume of Intermediate Level Long-lived Waste.

Three sort criteria can be proposed :

- Without measurement devices, based on an empirical sort criterion, less than 50% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.
- With a simple dose rate measurement device, depending on the “top to tail” cases, between 50% and 75% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.
- With an accurate measurement device, based on Ag-108m, more than 75% of the length of the control rods can be saved from Intermediate Level Long-lived Waste storage.

The next steps will be :

- to finalize the study of optimizing a packaging cell ;
- to validate the procedure of measuring, cutting and sorting with irradiated control rods ;
- to extend the procedure to other type of control rods (stainless steel, AIC+B_4C) ;
- to extend the procedure to other type reactors (1300 Mwe and 1500 Mwe).

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