Calculation Codes for Decommissioning

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Abstract

Decommissioning operations require repeated nuclear measurement to be performed, in order to get a precise knowledge of the radiological condition of the facilities : without reliable basic data, it is difficult to plan the operating conditions.

A first is the definition of the activation levels of neutron irradiated components, like the pressure vessel of a reactor core. This task requires different kinds of calculation codes, and is generally performed by highly specialised laboratories.

Secondly, calculation codes are commonly associated to nuclear measurements.During the interpretation phase, they allow the conversion of the measured physical quantity (generally photon fluence rates or dose rates) to activity values (Bq).

Endly, when manual operations are being prepared, calculation codes enable to predict the irradiation levels and help to define the possible radiological protection to be provided.

This paper summarizes different possible applications of calculation codes in the field of decommissioning. The way they can be coupled with nuclear measurement devices is explained as well. The most commonly operated codes are presented. Together with performances and limits, forthcoming trends are discussed.

I. Introduction

Calculation codes are effective and useful tools for decommissioning projects, especially for preparing and tracking operations.

One fundamental application is to define the initial radiological state of a facility, which is decisive for preparing the operational scenarios. It is difficult indeed to make decisions on a course of action without reliable basic data. Calculation codes are used to estimate used the component activation levels in reactors or accelerators subjected to a neutron flux (vessels, shielding, etc.).

Another increasingly widespread application involves coupling calculation codes with *in situ* measurement systems, generally to perform a “numeric calibration” for the purpose of converting the measured values (flux densities, dose rates) into activity levels.

Dosimetry predictions correspond to a third requirement in compliance with the ALARA principle. This is a routine maintenance practice that can be applied to decommissioning operations.

This article describes these three procedures, indicates the system performance and limits, and discusses some current trends.

II. Defining the Initial Radiological State

In a reactor the most important contribution to the radiological inventory is by far due to components activated by neutron radiation. However, determining the activity distribution in the core internals is a difficult step that can be long and expensive. The purpose is to identify the majority radioelements and the distribution of their activity levels within the main reactor core components.

This inventory can be established theoretically using calculation codes. However, this assumes the following basic data are known:

- fuel composition during reactor operation,
- geometric characteristics of the reactor,
- location and detailed composition of construction materials,
- irradiation history throughout the operating period.

Activity calculations are performed in two main steps:

- establish the space and energy distribution map of neutron fluxes in all the structures. This step uses particle transport codes in 3D geometry;
- estimate the activation levels of the structures after subdividing them into elementary volumes using source evolution calculation codes.

Several applications have already been carried out in

France to calculate core activation levels in heavy-water or gas-graphite reactors [1]. A CEA laboratory is specialized in this type of studies. One example of an application is the radiological inventory of the core internals of the Brennilis heavy water reactor.

The quantities generally estimated are the specific activities for each radionuclide as of the reactor shutdown date. The evolution of the activity levels over time is then inferred together with the dose rate distribution map in the reactor. The results are used to establish different decommissioning scenarios, to predict the quantity and activity of the waste produced, and to estimate the operational dosimetry.

These are complex calculations, and some basic data may be of doubtful accuracy, such as the composition of the component materials. It is therefore important to corroborate the theoretical results with experimental values: dose rate measurements are generally simple to perform, and can be used to check the orders of magnitude of the values obtained. Gamma spectrometry in the reactor block is more difficult, but more precise for estimating the activity of the principal γ-emitters. Both methods provide only relative data on the βγ-emitters, although this may be sufficient to correct the calculated values. More accurate validation can be obtained by sampling for destructive laboratory analysis. Separation and analysis techniques are currently capable of quantifying most of the radioelements present. This approach has some limitations:

- sampling may be costly, particularly in the case of highly active components such as the reactor vessel.
- transporting highly active samples is expensive.
- laboratory analysis is a lengthy and expensive process.

III. COUPLING MEASUREMENTS WITH CALCULATION CODES

Since the early 1990s, calculation codes have been used to interpret measurement results. Particle transport codes are employed to convert the physical quantity measured, generally the dose rate or flux density, to an activity value.

Irradiation and gamma spectrometry measurements are among the principal methods used for decontamination and decommissioning operations [2]. The first technique provides a dose rate; the second yields a spectrum from which the photopeak count rates are recorded. The count rates are then converted to flux density values after laboratory calibration.

The measurement configuration must then be modeled: source geometry, shielding (if any), detector position. By assigning an activity—generally a unit value (1 Bq·cm^{-2} or 1 Bq·cm^{-3})—to the source, the code calculates the dose rate or flux density at the measured data point. The actual activity of the measured component is then determined by simple proportionality. In other words, the measurement of a standard source is replaced by a calculation, hence the term “numeric calibration”.

This approach was first applied in the late 1980s, notably by the CEA [3]. It has since become widely used, and industrial systems based on similar methods are now commercially available.

Experience has shown, however, that this method has some limitations, the most important of which is the measurement uncertainty, which is a combination several uncertainties:

- the counting statistics, which generate only slight uncertainty if the measuring system is properly dimensioned and a suitable counting time is observed;
- the uncertainty arising from detector calibration, which can be minimized by routine rigorous calibration;
- the uncertainty on the calculation itself, which is dependent on each configuration, and is generally indicated;
- the uncertainty on the modeling hypotheses: the component geometry is generally simplified, particularly in the case of complex components, and the position, extent and even the nature of the contamination are not accurately known. Some hypotheses must therefore be postulated (surface source, volume source, contamination density or gradient). Experience has shown that this uncertainty largely predominates over the four preceding components: the validity of the final measurement result depends on the realism of the initial hypotheses. Imaging systems—notably gamma imaging [4]—can be used to localize the activity more precisely, and thus to reduce the measurement uncertainty. (See figure 1)

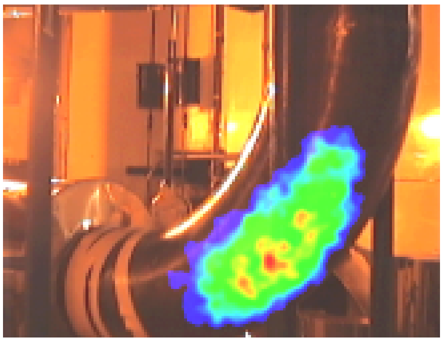

Figure 1 : Gamma imaging on a pipe. The exact position of the source is provided.

Another constraint is related to the use of calculation codes. The most powerful codes generally have rudimentary graphic interfaces, and modeling complex geometric configurations is a long process with a high risk of error. In recent years, calculation codes have been developed with more user-friendly (3D geometry) interfaces; this should facilitate modeling and limit geometric errors.

The CEA's development department for decontamination and dismantling has chosen to use the following codes:

- *MCNP* [5], a calculation code developed by LANL, used by the principal nuclear measurement laboratories to simulate electron, neutron or photon transport phenomena. This is probably one of the most versatile codes available. It is used in decommissioning to simulate flux densities or photon spectra, a particularly useful function for designing gamma spectrometry measurement systems. Figure 2 shows the result of a spectrum simulation for the design of an underwater gamma spectrometry system. The simulation provides valuable data: it is capable of predicting the measuring system response before the operation, and checking whether it matches the conditions encountered. This calculation code can also be used to interpret neutronic measurements, although they are applicable more to measurements of waste containing actinides than to *in situ* measurements, as *in situ* neutronic measurements are not routine operations. Reference [6] discusses an example using neutron interrogation to estimate the uranium retention in an enrichment plant. This code is reserved for experienced users. Moreover, the simulations may be very long, particularly in complex or diffusive media (e.g. underwater measurements).
- *Mercure* [7], a transport code capable of calculating the uncollided photon fluence rate or the dose rate at a specific point. It was developed by the CEA, and is routinely used to interpret gamma measurements (dose rates or spectrometry). Although it has fewer features than MCNP, it is considerably easier to use and the calculation times are generally acceptable.

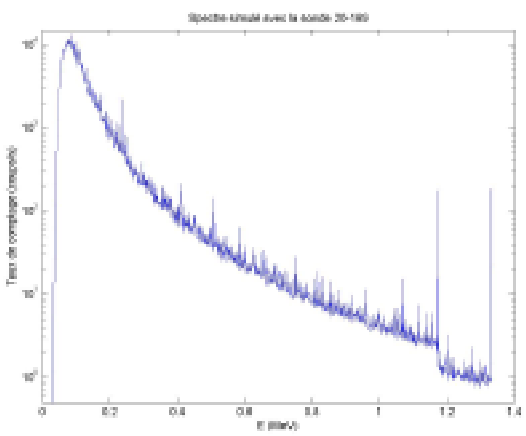

Figure 2. Example of a spectrum obtained by modeling with MCNP

- *Microshield*, which is intended more for radiological protection applications than for nuclear measurements. It is very easy to use, but is capable of modeling only very simple geometries (homogeneous waste drums, wall sections, etc.).

IV. RADIOLOGICAL PROTECTION

Controlling and optimizing occupational doses is essential during operations in irradiating environments.

Simulation software has recently been developed to estimate the dose received by an operator (or by a mechanical device) in a 3D environment. The input data are the following:

- a 3D description of the scene: geometries and materials; this phase is generally based on a combination of simple geometries (planes, spheres, cylinders, truncated cones, etc.);
- the spatial position, nature and intensity of the activity;
- the operator paths and the duration of the operation.

The objective is to define procedures that will reduce the doses received. The following developments are now in progress in this area [8]:

- Importing 3D models created in a standard format from drawings or 3D reconstruction techniques. Importing files avoids the need for the often lengthy and error-prone phase of describing the geometry in 3D. It is now possible to transfer complex 3D models without replacing each element by a standard geometric shape.
- Highly accurate estimates of personnel doses based on algorithms from calculation codes for static configurations, as in the Mercure code (see section III).

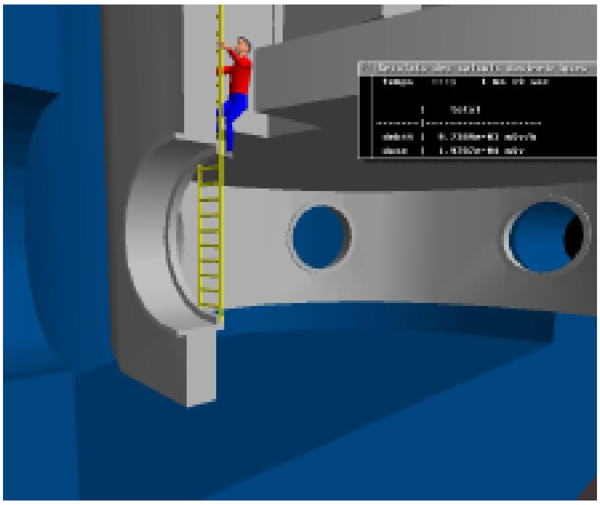

Figure 3. Simulation of a maintenance operation
(source: CEA/DRT–STMI)

- The shortest possible calculation time, through the use of optimized algorithms or powerful computers, with the ultimate objective of obtaining a realistic real-time dose estimate. Recent developments now make it possible, for example, to virtually move an operator in a scene comprising both irradiating elements and shielding, with a display of the dose rate and integrated dose.

Figure 3 shows a simulation performed by the CEA's technological research division for a French company working in ionizing environments. The purpose was to estimate the dose received by an operator during a steam generator maintenance operation. The operator's movements can be simulated, with the dose rate and the total integrated dose displayed at all times in a popup window. This project is now under development, and is currently functional only with point sources.

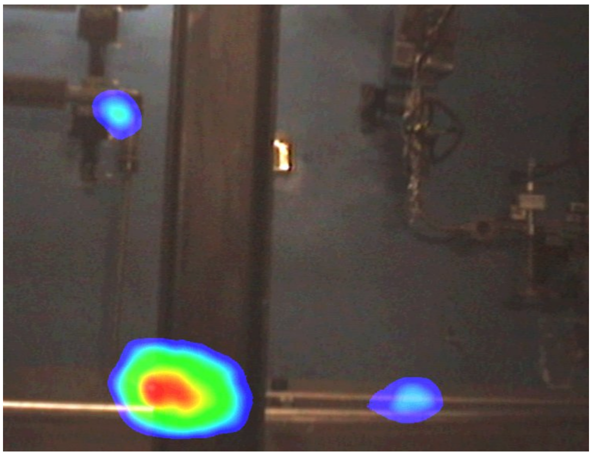

Figure 4 : hot spots localization before maintenance intervention in a nuclear power plant, operated with a CEA developed gamma imaging device

It should be noted that this approach requires adequate knowledge of the geometric and radiological conditions of the work zone, and thus assumes a prior reconnaissance phase, in order to localize the most irradiating sources and quantify their activity. Hot spots "hunting" either with gamma imaging (see figure 4)or more traditional means, activity quantification following the methods discussed in paragraph II.

V. Conclusion

Calculation codes are increasingly used before or during dismantling operations, thanks mainly to the rapid increase in computing power.

Determining the activation of materials with a calculation code is still a complex procedure, accessible only to specialized laboratories.

Calculation codes are used to design measurement systems and interpret the results obtained. Combining codes and measurements provides a means of estimating the activity of the components, an indispensable step during decontamination and decommissioning operations. Their performance is already satisfactory, but is continually improving, notably through the use of revised data bases. The human-machine interface is often lacking in sophistication, leading to errors and lost time. Work has recently been undertaken to make these codes more user-friendly, and significant progress can be expected in the near future.

Software is increasingly used to predict occupational doses in order to maintain them as low as reasonably achievable. Today they can be coupled with environments modeled in 3D, and in some cases by importing environmental geometries modeled using standard formats. More interactive and more precise systems will no doubt be developed by applying virtual reality techniques coupled with optimized numeric methods, but this will require extensive knowledge of the radiological conditions (source position and activity) to obtain reliable results.

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