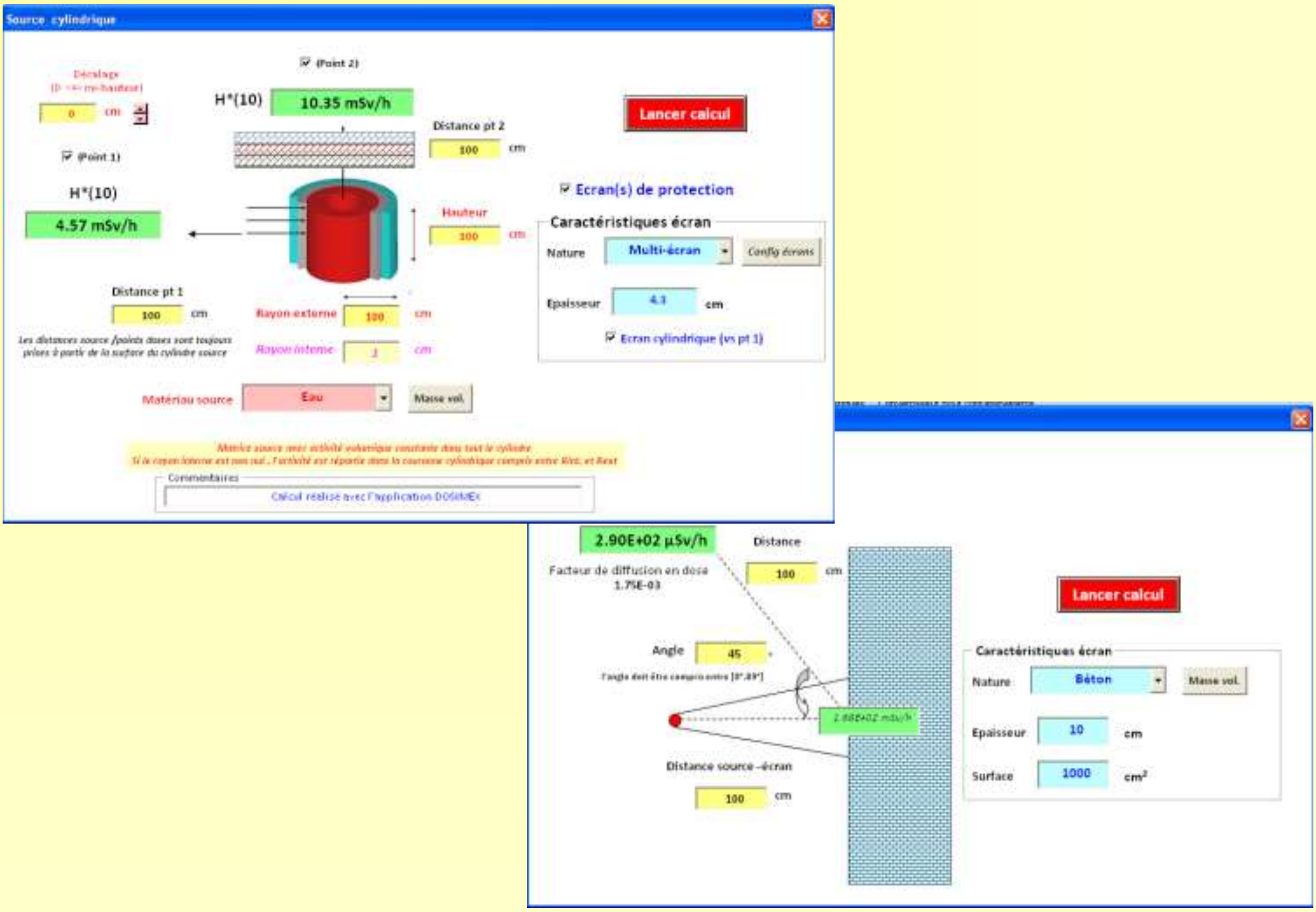
DOSIMEX-GX 3.0

GAMMA DOSE CALCULATION CODE

✓ *USER HAND-BOOK*



Alain VIVIER, Gérald LOPEZ

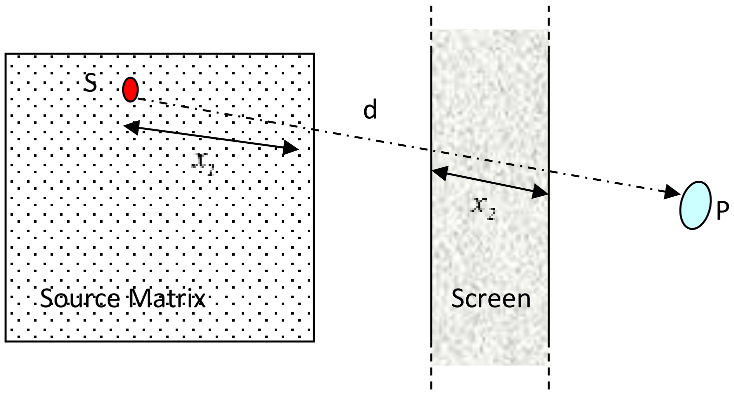
JUNE 2020

Partie I. PREAMBLE

I.1 PRINCIPLES OF OPERATION OF THE DOSIMEX-GX 2.2 CODE

The DOSIMEX-GX code is a code used to calculate the rates of gamma and X dose equivalents generated by emitters of ionizing radiation of the radionuclide type (gamma photons) or X generators. It is a deterministic type code implementing attenuation calculations in straight lines with build-up correction.

BASIC REMINDERS ON THE PRINCIPLE OF A DETERMINISTIC CODE



Schematic diagram of the calculation of attenuation in a straight line

For each thickness x of material crossed, we obtain a straight line attenuation factor equal to $e^{-\mu x}$, with μ the coefficient of linear attenuation of the material for the energy of the photons considered. It is always interesting to express these thicknesses in terms of the number of relaxation lengths $n = \mu x$. This dimensionless number is fundamental in the expression of the build-up

In the figure above, the fluence flow from the source point S is attenuated, before arriving at point P, by crossing the source matrix and the screen. The two terms of attenuations $e^{-\mu_1 x_1}$ and $e^{-\mu_2 x_2}$ are multiplicative, and the total attenuation in a straight line is equal to:

$$e^{-\mu_1 x_1} \times e^{-\mu_2 x_2} = e^{-\left[\mu_1 x_1 + \mu_2 x_2\right]}.$$

The fluence flow after attenuation in a straight line is therefore equal to:

$$\varphi_{att} = \mathcal{A}\ I_\gamma \frac{\Omega}{4\pi}\ e^{-\left[\mu_1 x_1 + \mu_2 x_2\right]}$$

With $\mathcal{A}$ source activity, I_γ emission intensity and $\frac{\Omega}{4\pi}$ the solid angle fraction, essentially related to distance.

Important note: some software uses the law in "1 / d2" to calculate the solid angle fraction. This model is not suitable for short distances (<1 cm) and can lead to totally irrelevant overestimates. On this subject, see chapter 6.1 of the book "dose calculation generated by ionizing radiation" and especially scenario 2 of the validation file)

Insofar as the calculation in attenuation in a straight line only takes into account the photons having retained their initial energy, the conversion coefficient allowing to pass from the flow of fluence attenuated to the flow of dose equivalent is unchanged. The dose rate after attenuation in a straight line is written:

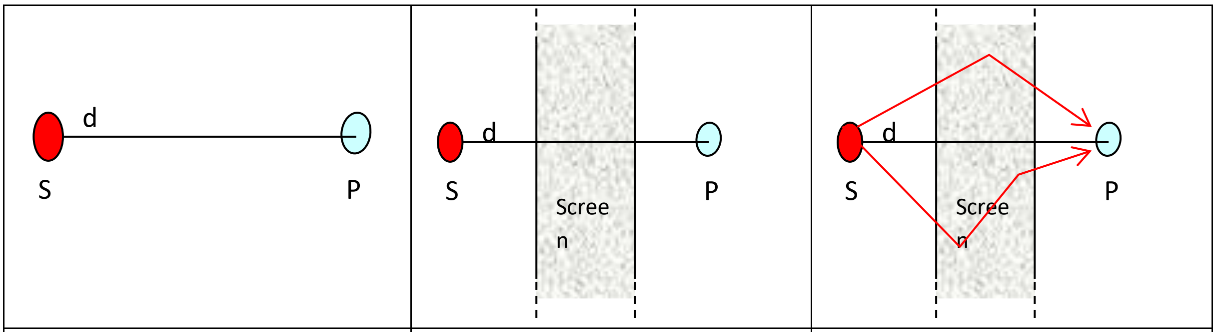
$$H^*(10)_{att.} = h^*(10,E_\gamma)\varphi_{att} = h^*(10,E_\gamma)\mathcal{A}\ I_\gamma \frac{\Omega}{4\pi} e^{-[\mu_1 x_1+\mu_2 x_2]}$$

In this sense, the presence of a screen reduces the simple diagram of a purely spherical radiative model (diagrams below)). In some cases, the presence of a screen can even increase the total value of the dose rate.

The value $\dot{H}_{att}$, result of the attenuation calculation in a straight line, must therefore be corrected to obtain an estimate of the total dose equivalent flow $\dot{H}_{tot}$.

The correction is formalized by a multiplicative factor B with

$$\dot{H}_{tot} = B \times \dot{H}_{att.}$$



A) Bare source calculation	B) Straight line attenuation	C) Build-up correction
$h^*(10,E_\gamma)\dfrac{\mathcal{A}\ I_\gamma}{4\pi d^2}$ *	$h^*(10,E_\gamma)\dfrac{\mathcal{A}\ I_\gamma}{4\pi d^2}e^{-\mu x}$	$B_\infty\ h^*(10,E_\gamma)\dfrac{\mathcal{A}\ I_\gamma}{4\pi d^2}e^{-\mu x}$

Principle of straight line attenuation and diffusion correction

** we have left here for educational purposes the law in "1 / d2"*

The term $\left[B_\infty \times e^{-\mu x}\right]$ represents the dose attenuation factor with diffusion correction.

The build-up resulting from the diffusion in two successive thicknesses of two materials of different natures can be estimated by the expression:

$$B_{m1+m2}(E_\gamma,\mu_1 x_1,\mu_2 x_2) \approx B_{m2}(E_\gamma,\mu_1 x_1+\mu_2 x_2)+\left[B_{m1}(E_\gamma,\mu_1 x_1)-B_{m2}(E_\gamma,\mu_2 x_2)\right]$$

the source element (voxel) at point S is written:

$$d\dot{D} = B_{total}(\mu_1 x_1, \mu_2 x_2)e^{-[\mu_1 x_1 + \mu_2 x_2]} d_\Phi I_\gamma \frac{\Omega}{4\pi} \mathcal{A}_{vol} \, dV$$

With $\mathcal{A}_{vol}$ the volume activity of the source (considering a uniform volume activity) and dV the elementary volume surrounding point S.

The total dose rate is obtained by integrating this expression on the total volume of the source:

$$\dot{D} = \int_{Source} B_{total}(\mu_1 x_1, \mu_2 x_2)e^{-[\mu_1 x_1 + \mu_2 x_2]} d_\Phi I_\gamma \frac{\Omega}{4\pi} \mathcal{A}_{vol} \, dV$$

In practice, this continuous integral is replaced by a discrete sum obtained by a discrete cutting (voxel) of the source.

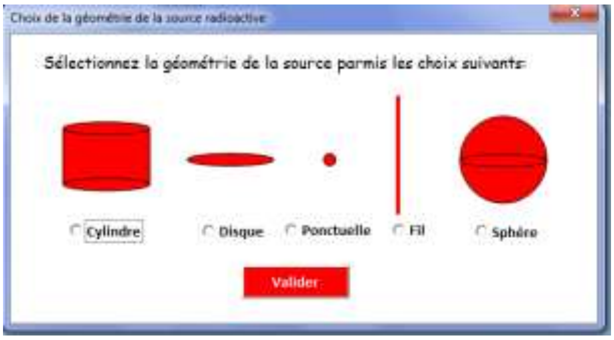
From this expression, we can also define and calculate the resulting build –up by the following weighted average:

$$\bar{B} = \frac{\displaystyle\int_{Source} B_{total}(\mu_1 x_1, \mu_2 x_2)e^{-[\mu_1 x_1 + \mu_2 x_2]} d_\Phi I_\gamma \frac{\Omega}{4\pi} \mathcal{A}_{vol} \, dV}{\displaystyle\int_{Source} e^{-[\mu_1 x_1 + \mu_2 x_2]} d_\Phi I_\gamma \frac{\Omega}{4\pi} \mathcal{A}_{vol} \, dV}$$

For these calculations, DOSIMEX-GX 2.1 uses the following databases

- For the attenuation coefficients μ : the XCOM / NIST database: http://physics.nist.gov/PhysRefData/Xcom/html/xcom1.html
- For fluence to dose equivalent conversion coefficients d_Φ : the values given in ICRU REPORT 57 (International Commission on Radiations Units and Measurements, 7910 Woodmont Avenue, Bethesda, maryland 20814, USA)
- For build –up coefficients: the values given in document ANSI / ANS-6.4.3-1991 (American Nuclear Society, Standarts Committee Working group ANS 6.4.3, 555 North Kensington Avenue La grange Park, Illinoid 60525 USA)
- *For gamma emission tables, the data available on the Laraweb site: http://laraweb.free.fr*

I.2 MAIN FIXES AND CHANGES TOTHE ORIGINAL VERSION (CD-ROM VERSION 1ST EDITION, JULY 2012)



DOSIMEX-G 1.0

VERSION 1.2

- Fixed bug for Bq.cm options^{-3} and Bq.cm-1 on cylinder, sphere and wire geometries.
- Correction of a self-absorption calculation defect in hollow spheres.
- Improvement of the composition law of build-ups, in particular between a thin screen and a thick screen, in very different materials, such configurations being liable to generate, in the initial version, aberrant build-ups.
- Extension of the X Com database (linear attenuation coefficients) from 3 MeV to 15 MeV.
- Transition from a uniform mesh to a power mesh, adapted to the energy of the photons and the nature of the materials, in order to be able to take into account very large sources ('see §II.6).
- Consideration of the "Air" material
- Creation of an option allowing to modify the densities of the source and screen materials.
- Addition of a monoenergetic pseudo-source (choice of “Mono E” element from 10 keV to 15 MeV) allowing parametric studies as a function of energy.

VERSION 1.3.1

- Addition of the rectangular source geometry.
- Possibility of two screens with point source
- Calculation of dose generator X medical or industrial and calculation according to standard NFC 15-160
- Modification of the build-up calculation. Abandonment of the Taylor model (see §II.2)

VERSION 1.3.2

- Modification of the NFC 15-160 calculation: creation of the abacus calculation method
- Possibility to create custom source spectrum
- Correction of a coding defect in the consideration of multiple radionuclides

VERSION 1.3.3

- Implementation of alternative methods of calculating the NFC 15-160 standard
- Multiple screen option for gamma sources
- Option for calculating the dose generated by braking and / or annihilation radiation (β^+) from a Beta source.

VERSION 1.4

- Calculation of dose rates within contaminated volumes on internal surfaces
- Inverse calculation of activity vs isotopic spectrum and measured dose rate
- New option to determine radiological zoning
- Choice of the nature of the anode with the X generator.

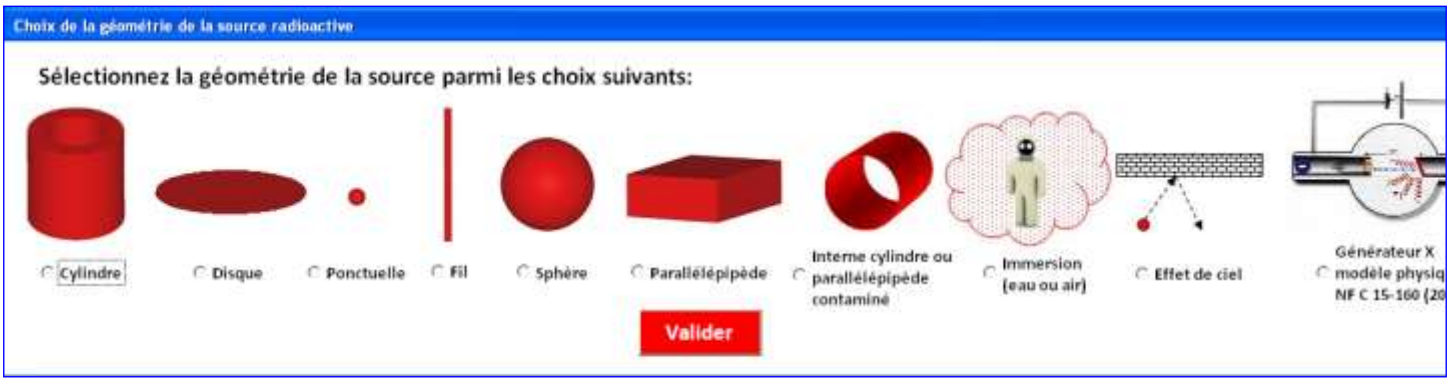
VERSION 1.4.1

- Correction of a bug on the calculation of the area of peaks X of fluorescence versus intensity (mA) in the calculation of dose rate generator X
- Diffuse dose rate and spectrum calculation in the X generator calculation application
- Calculation of the average build-up generated by the screen lit by the primary of generator X
- Calculation of the dose rate due to direct fluence in the “gamma skyshine” calculation
- Addition of a field in the dialog box allowing to add a comment in the summary sheet

VERSION 2.0 (SECOND EDITION, MARCH 2016)

Version 2.0 incorporates the previous changes +:

- Hp (3) calculation (crystalline dose)
- Installation of a screen on the broadcast path in the "X generator" option
- Possibility of multi-screen in X generator modeling
- Decay option and mass-activity relationship
- DED calculation for variable heights compared to the cylindrical and wire source
- Modification of gamma emission databases: taking into account low energy X emission (10 keV, 30 keV) for approximately 80 radionuclides (Am 241, I 125…). See scenario 1 validation file
- Display H * (10) directly on the dialog boxes in place of the kerma air
- Beta transmitter database for the braking radiation option
- Operational zoning in the zoning option
- Option to convert from mass to mass, isotope or activity fractions



VERSION 3.0 (2019)

RADIONUCLIDE PART

- Implementation of the hollow cylindrical source geometry with cylindrical protective screens
- Possibility to fill this hollow cylinder with water (pipe modeling)
- External database for storing new materials and new emission tables.
- Addition of source "immersion in a cloud" (common with Dosimex-I)
- Creation of a file to save the results
- Calculation of the effective anteroposterior dose (E (AP) in the summary sheet
- Uranium material in the braking radiation option
- Depleted gamma U source taking into account braking radiation
- Implementation of around forty radionuclides (rare earths) tested in nuclear medicine (Tm, Gd, Dy ..)
- Taking into account a technical calculation note carried out by AREVA NT carrying out MCNP calculations on the scenarios in the Dosimex-GX validation file
- Modification of build-up coefficient for light materials: water, concrete, Aluminum, calcium, air
- Calculation of screen attenuation factor in the point source model
- Contributions of each gamma line to the dose and contribution of each radionuclide if necessary in the point source model

GENERATOR PART X

- Improvement of the calculation time on the calculation of the diffuse with the generator model X
- Reinforced validation on X generator modeling vs CEA-R 6457 report
- Forward scattered calculation for generator X
- Writing of a more complete specific manual for the X generator including the validation file
- X-ray fluorescence implementation for diffusion calculation on a screen, used in the “skyshine” function in gamma and with the X generator.
- Calculation in H * (10) with generator X instead of kerma air
- Option to directly calculate the dose as a function of the load in mA.min
- Implementation of standard NF C 15-160 of 2018

- In general, significant strengthening of validation files

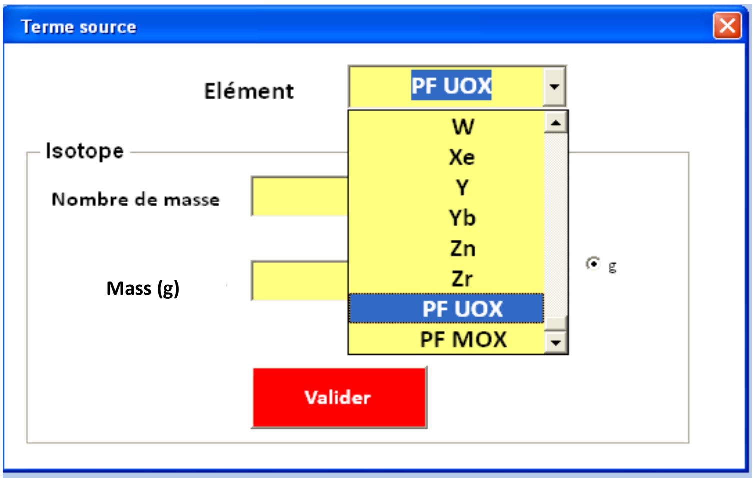
Partie II. DOSIMEX-GX USE / GAMMA SOURCES

II.1 RADIONUCLIDES DATABASE:

This database was established to limit computation times by grouping certain gamma lines of close energy. The application then uses an average gamma energy, weighted by the emission intensity and a total intensity equal to the sum of the intensities. Thus, for example, the spectrum of americium 241 which includes 191 gamma lines identified in the complete database, is limited to 6 gamma lines in the complete database with grouped gamma lines. During a dose rate calculation, the energies and intensities of the gamma lines used are displayed at the top right on the summary page.

In this database has been added:
The possibility of choosing a monoenergetic pseudo-transmitter**"Mono E"**for faster parametric calculations. The available energies are limited to precise values (10 keV, 20, keV, etc.) in a logarithmic progression up to 15 MeV. To choose any energy, see below "completing a database"

Typical emission tables for irradiated fuels of UOX or MOX type as a function of cooling times. These tables are given according to mass and not activity. See appendix at the end of the manual



❖ "MEDICAL" DATABASE & "ELECTRO-NUCLEAR" DATABASES

These two databases are reduced databases extracted from the complete database with grouped gamma rays, and limited to the radionuclides usually encountered in these two specific fields.

❖ COMPREHENSIVE DATABASE

This database does not use the principle of grouping of gamma lines and in principle gives more precise results. However, it will generally lead to longer calculation times. Indeed the computation time, for a given configuration, is proportional to the numbers of gamma lines taken into account

❖ COMPLETE A DATABASE

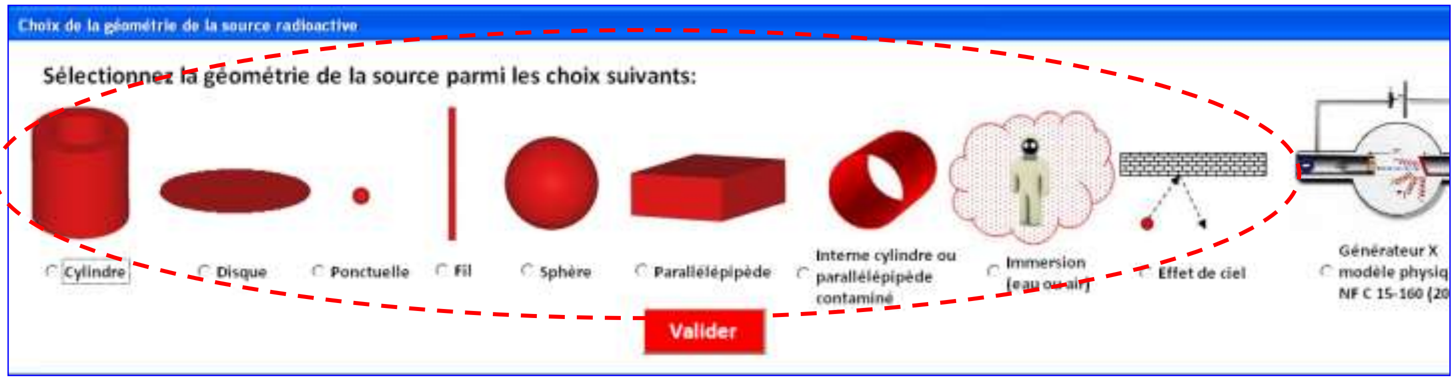
It is possible to add radionuclides to the database. To do this, you will need to go to the "**OPTION »**select "Manually define a gamma emission spectrum": see chapter on options §I.3.Once you have chosen your database, you are ready to use DOSIMEX-GX 3.0.

II.2 Gamma dose rate calculation.

II.2.1 Choice of geometry

Just click on the active button "Calculation of gamma and X dose" to bring up the dialog box offering the different possible sources, gamma or X:

A dialog box opens, offering the following configurations:



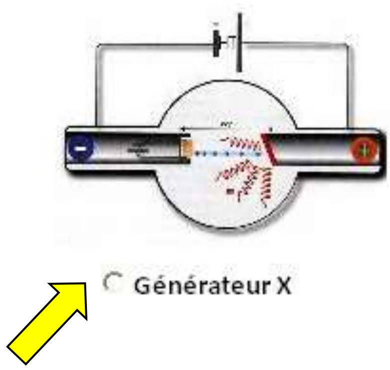
The first 9 configurations relate to the source geometries containing the gamma emitting radionuclide (s) chosen. The last configuration corresponds to the calculations for the X-ray generators. This specific application is presented in part III of this document.
For use in gamma dose calculation, the application allows calculation for cylindrical, disc, point, wire, spherical or parallelepiped sources.

For volume sources (resp. Surface, line), we consider that the volume activity (resp. Surface, line) is constant and homogeneous in the source.

The “contaminated cylinder or parallelepiped” geometry, added with version 1.4, makes it possible, for example, to determine the dose rate inside a pipe as a function of the surface contamination of the internal wall.

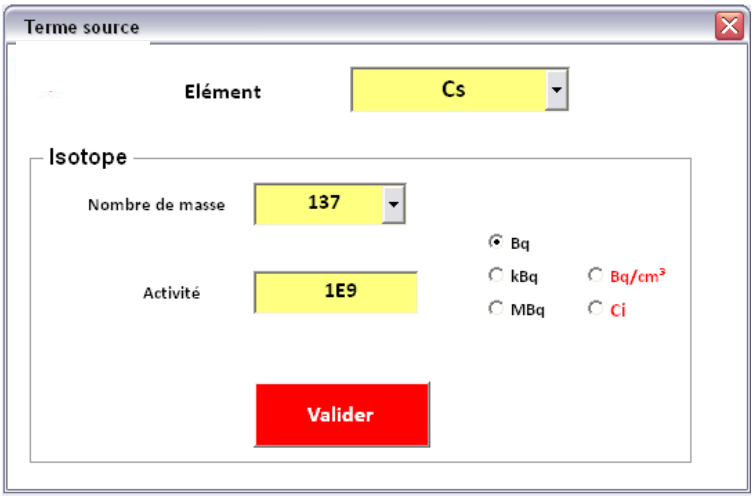
The “skyshine” option makes it possible to determine the dose equivalent flow rate generated by the diffusion of primary radiation on a plate of a determined mater

For the generator X option see specific manual



II.2.2 CHOICE OF SOURCE TERM

After choosing a source geometry, the “Source term” dialog box allowing you to choose the radionuclide (s) present, as well as their respective activities, opens:



You must first enter the chemical nature of the element (Cs or Co etc.); The dialog box then offers you to choose the desired isotope by specifying its mass number and its activity

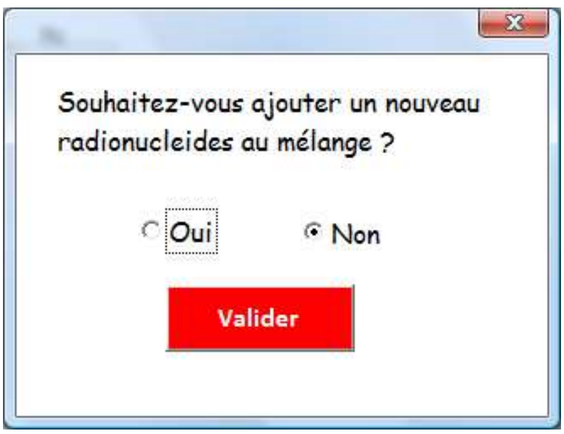
Once the information has been entered Click on "Validate" to save the entry.
Notes:

- *If you want to use a radionuclide that you added to the database using the manual option, it will be found at the end of the "Element" drop-down menu. In this case avoid the automatic entry of the symbol associated with your element because it is possible that an element similar to the one you entered is already saved in the database.*

- *Entering the activity accepts the scientific notations example for 1GBq you can enter 1E9 then validate the button "Bq".*

- *The specific activities evolve according to the type of source selected: Bq.cm-^{3} for volume sources, Bq.cm-2 for surface sources, Bq.cm-1 for linear sources.*

After validation, the application then offers you the possibility of entering another radionuclide:



If you select "yes" the application will open the "Source Term" window again.
You can use up to 15 different radionuclides in your calculation

If you select "no" the application will then ask you to choose the type of build-up calculation model that you want to use for your calculations. What can for certain geometries, in particular cylinder and parallelepiped, considerably increase the computation time.

II.2.3

II.2.4 CALCULATION RESULTS

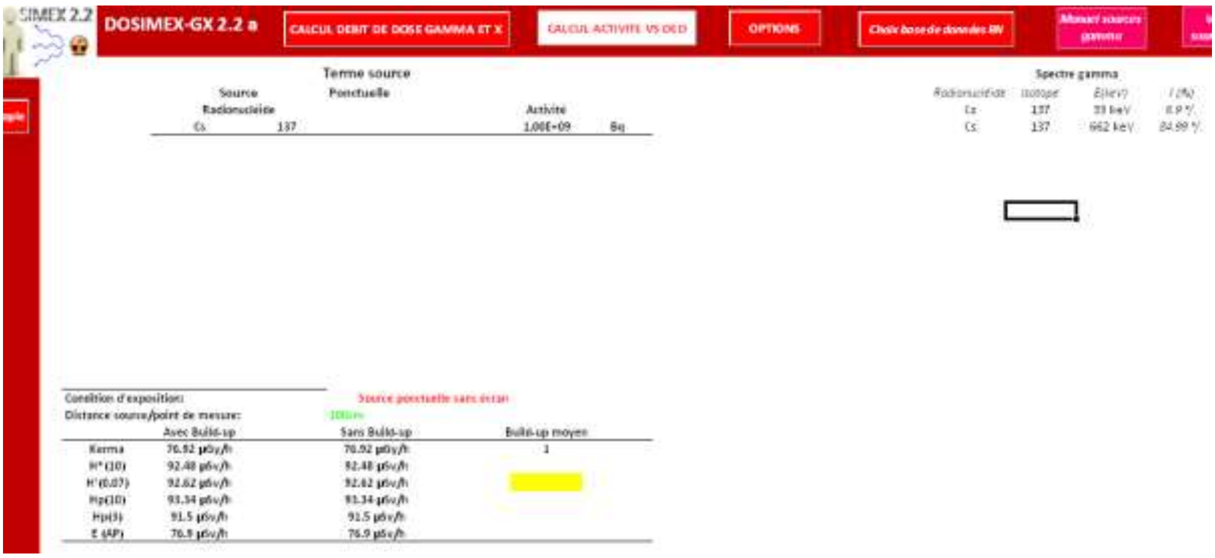
The value displayed in the dialog box corresponds to the ambient dose equivalent flow rate H * (10). All the essential regulatory radiometric quantities (CIPR 74) are indicated on the summary sheet:

- **Kerma flow in the air $\dot{K}_a$**
- **Ambient dose equivalent rate $\dot{H}*(10)$**
- **Individual dose equivalent rate $\dot{H}_p(10)$**
- **Crystal dose equivalent rate $\dot{H}_p(3)$ ((*see CEA report, see validation file*)**
- **Directional dose equivalent rate $\dot{H}'(0,07)$**
- **Effective antero-posterior dose rate E (AP) (version 2.2)**

The values are given both taking into account the build-up and without taking into account the build-up.

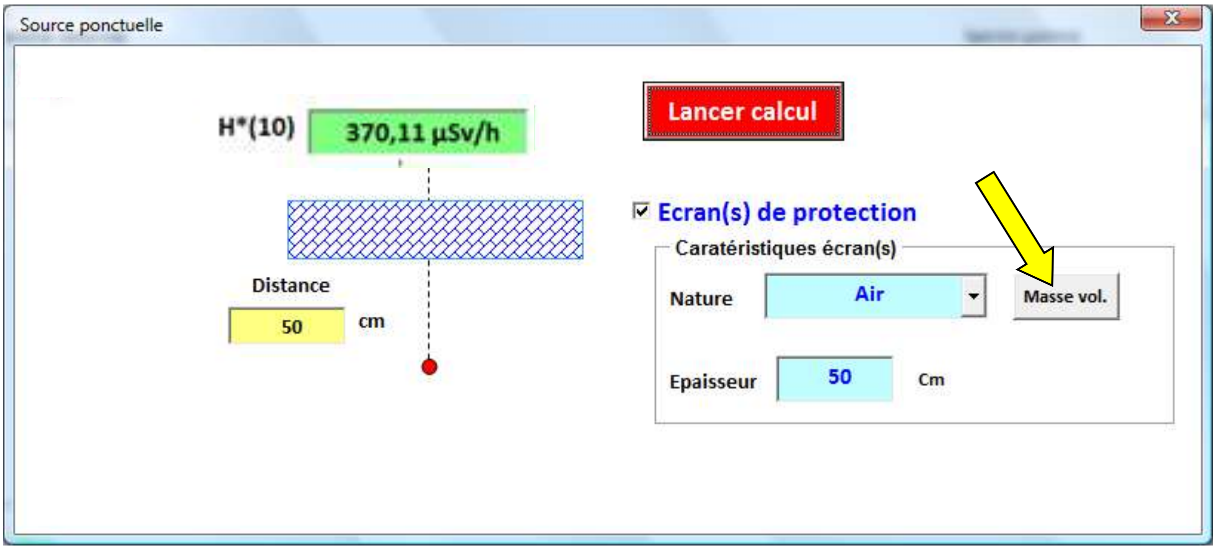
You will also find on this summary sheet:

- **The gamma spectrum used**
- **Listing of seized radionuclides**
- **Their respective activities**
- **The nature of the build-up used**
- **The source-detector distance**
- **The type of source matrix, geometry and material**
- **The thickness and nature of the screens**

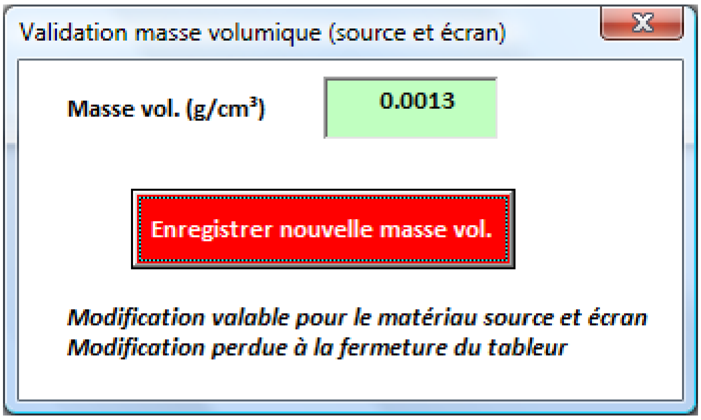


II.2.5 SCREEN CONFIGURATION

It is possible to modify the density of the screens used. To do this, select a screen in the dialog window associated with the calculation configuration, then click on the "Vol. "



A dialog box then tells you the density value currently entered for this material, you can then modify it and save this new value.

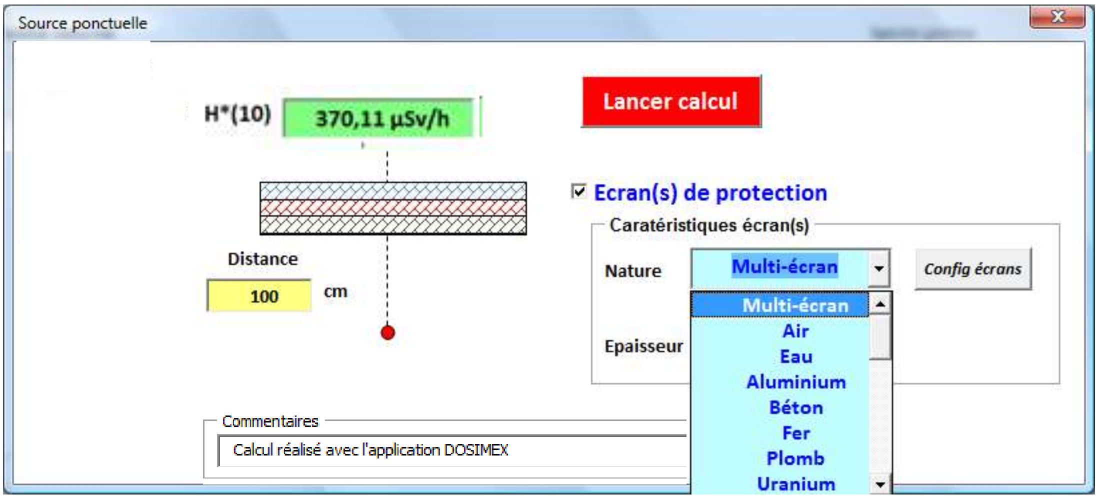


Notes:

- *As in the case of the modification of the database, your density modifications will not be saved when the application is closed.*
- *During the entire period of use of the application, the new density will be used for all the calculations.*

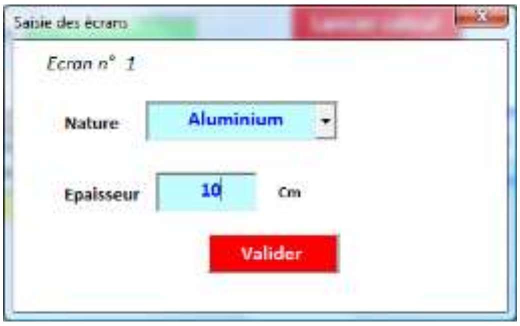
II.2.6 MULTI-LAYER SCREEN

It is also possible to create, after choosing a multilayer screen made up of a succession of various materials and various thicknesses. For this, you must select "Multi-screen" in the nature of the material:



By default the multi-screen consists of 100cm of air.

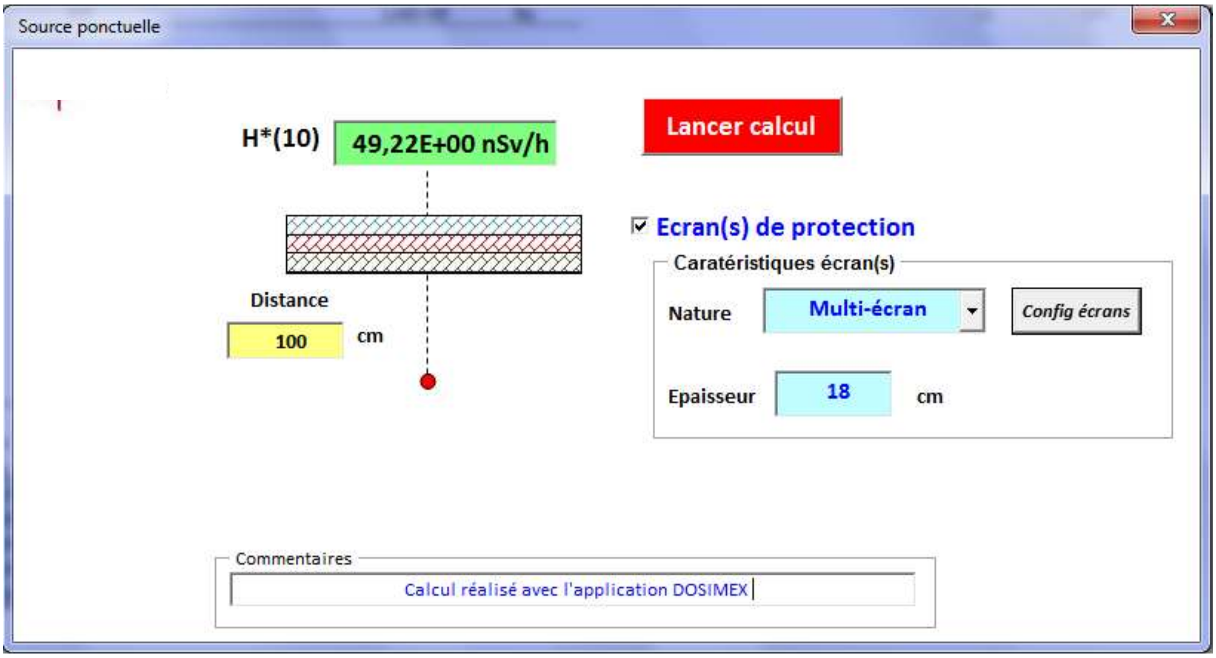
To configure it according to your needs, click on the “Config. Screens ”and enter one by one the nature and thickness of each of the screens which constitute it:



As you enter the screens, the number of the screen indicated at the top left of the "Enter screens" dialog box increments. With each new entry, the application reminds you of the constitution of the multilayer screen:

When you have entered all the screens making up your multilayer screen, close the "Screen capture" window.

In this calculation configuration, although the thickness window is visible, you will no longer be able to modify the value.



The summary sheet will then tell you that the calculation has been made in the case of a multi screen and will tell you at the bottom of the page the composition of this multi screen.

Type de Build-up Taylor

Condition d'exposition: Source ponctuelle avec écran de Multi-écran de 18cm

Distance source/point de mesure: 100cm

	Avec Build-up	Sans Build-up	Build-up moyen
Kerma	40,95E+00 nGy/h	16,12E+00 nGy/h	2,54
H*(10)	49,22E+00 nSv/h	19,38E+00 nSv/h	
H'(0,07)	49,26E+00 nSv/h	19,39E+00 nSv/h	
Hp(10)	49,69E+00 nSv/h	19,56E+00 nSv/h	
Hp(3)	48,64E+00 nSv/h	19,15E+00 nSv/h	

de: Aluminium 10 cm/ Fer 5 cm/ Plomb 3 cm/

Note :
Each time you click on the "screen config" button, the screen entered is destroyed and replaced by default with 100cm of air if you choose nothing.

II.2.7 CONTRIBUTION OF ENERGIES AND RN TO THE DOSE IN THE POINT MODEL

Option valid only in the point model:

In the case of a multi-radionuclide source, the contribution of each RN to the total DED is indicated in the summary sheet

In the example below with 1 MBq of Co 60 and 10 MBq of Cs 137. We then see the predominant weight expected from Cs 137 in the absence of a screen:

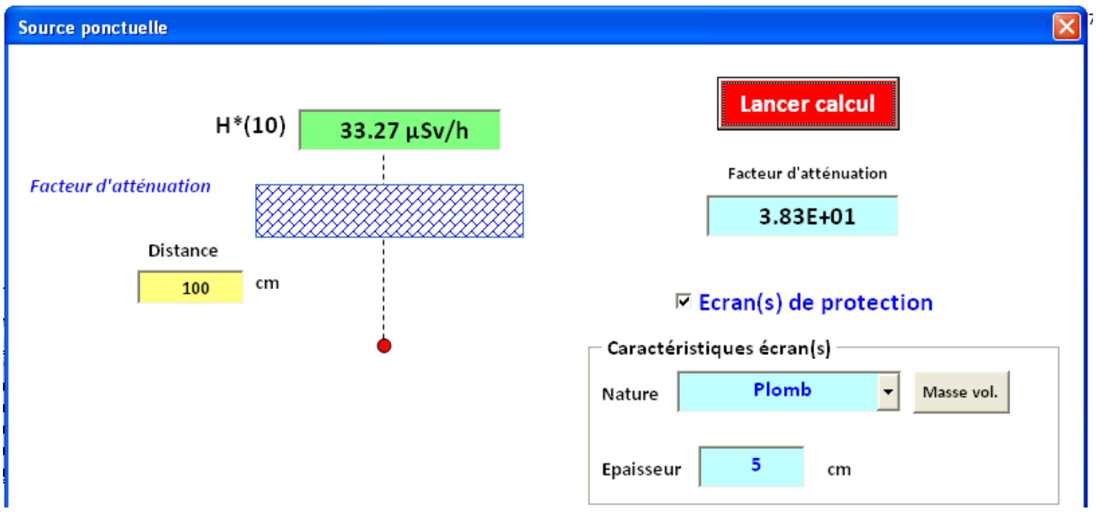
Terme source

Source		Ponctuelle			
Radionucléide			Activité		contribution DeD total
Co	60		1.00E+09	Bq	27.39 %
Cs	137		1.00E+10	Bq	72.61 %

We then see these weights changing in the presence of a screen, in the example below with 5 cm lead :

Source		Ponctuelle			
Radionucléide			Activité		contribution DeD total
Co	60		1.00E+09	Bq	85.46 %
Cs	137		1.00E+10	Bq	14.54 %

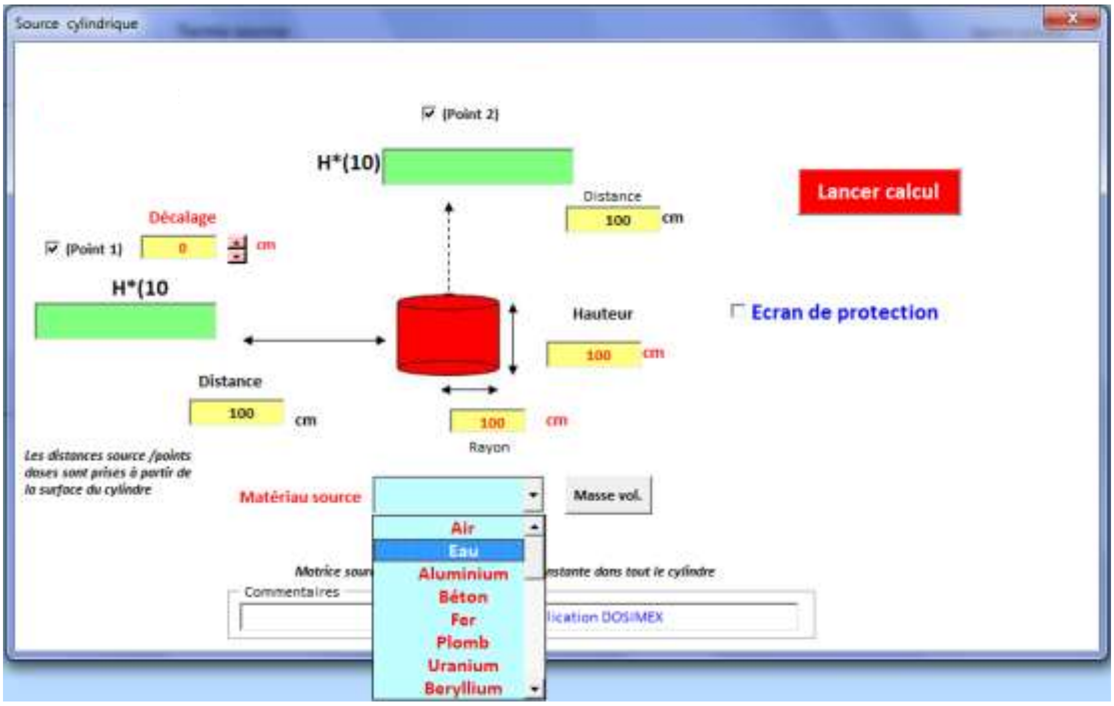
In addition, the dialog box displays the attenuation factor averaged over the entire spectrum:



II.3 Specificities on source geometries.

II.3.1 SPECIAL CASE OF VOLUME SOURCES

Volume sources are considered to be homogeneously contaminated in the matrix that constitutes them. You must therefore indicate the nature of this matrix. The constituent materials of the proposed source matrix are identical to those proposed for the screens, depending on the build-up model chosen.



For volume sources, we generally propose 2 points for calculating the dose rate. The summary table of results then specifies the values calculated for each of these points:

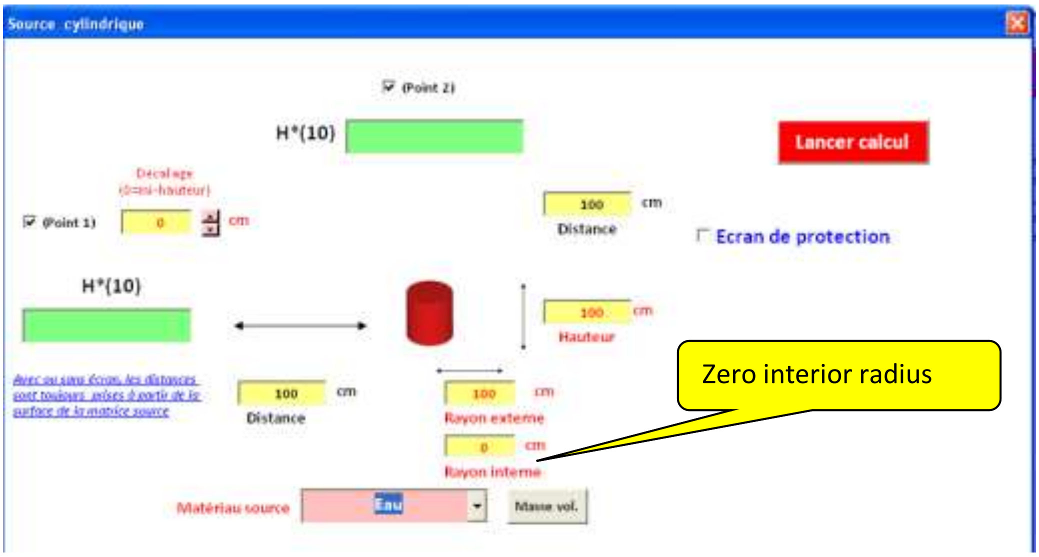
	Condition d'exposition		Cylindre R=100 cm / H=100 cm de Eau sans écran	
	Distance source/Pt1		100cm	Distance source/Pt2
		Avec Build-up	Sans Build-up	Build-up moyen
Pt1	Kerma	4,5 µGy/h	2,33 µGy/h	1,93
	H*(10)	5,4 µSv/h	2,8 µSv/h	
	H'(0,07)	5,41 µSv/h	2,8 µSv/h	
	Hp(10)	5,4 µSv/h	2,8 µSv/h	
	Hp(3)	5,34 µSv/h	2,77 µSv/h	
Pt2	Kerma	8,97 µGy/h	4,71 µGy/h	1,91
	H*(10)	10,79 µSv/h	5,65 µSv/h	
	H'(0,07)	10,79 µSv/h	5,65 µSv/h	
	Hp(10)	10,89 µSv/h	5,7 µSv/h	
	Hp(3)	16 µSv/h	8,38 µSv/h	

It is possible to choose only one point, which can be a time saver

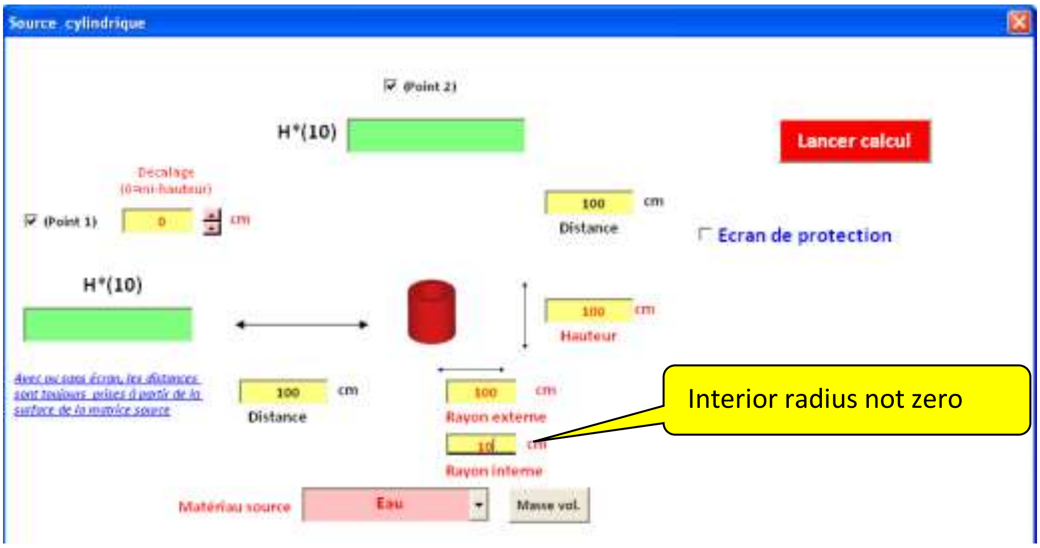
II.3.2 CYLINDRICAL SOURCE MATRIX FROM VERSION 2.1 (APRIL 2017) AND CYLINDRICAL SCREEN

From version 2.1 it is possible to use a hollow cylindrical matrix ("pipe" geometry).
A new parameter appears: the interior radius.

By default this internal radius is equal to 0, allowing to find the cylindrical source matrix

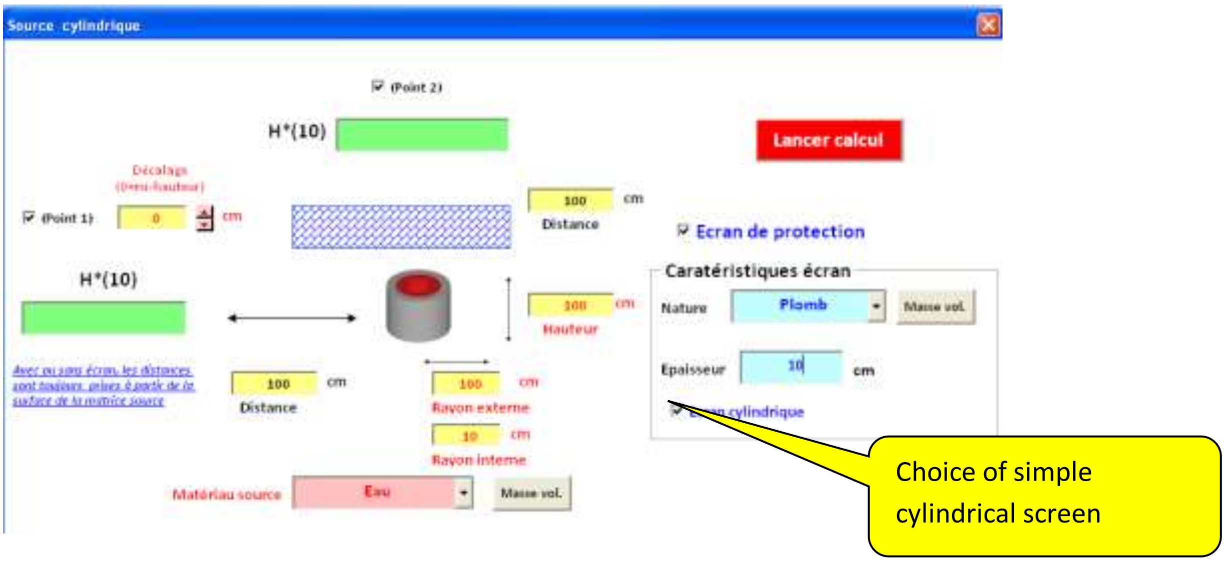


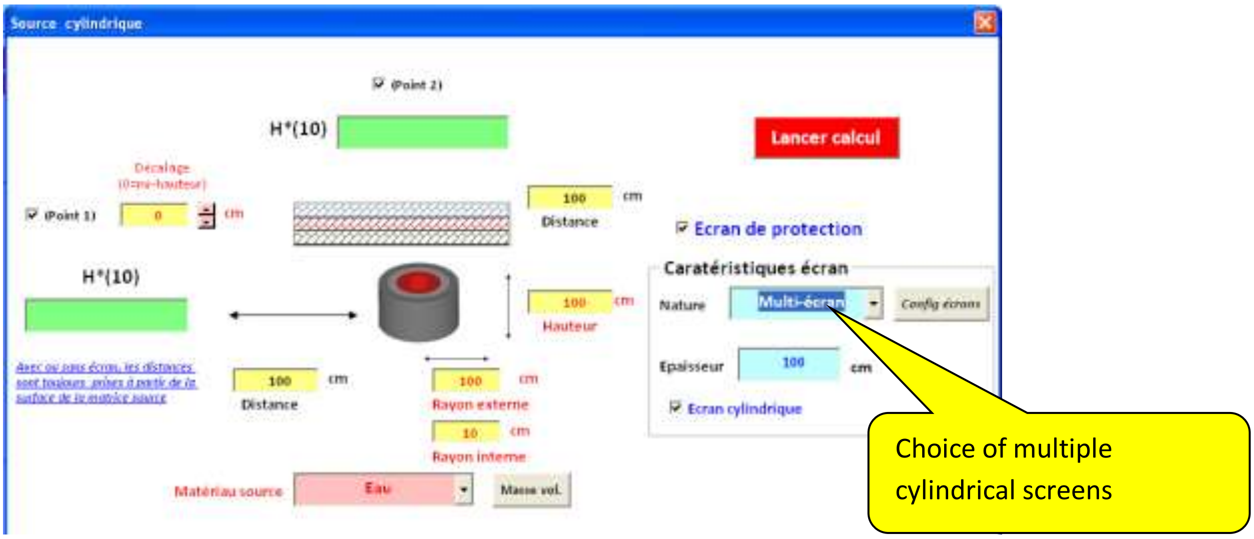
By taking a non-zero interior radius, we thus define a hollow cylindrical matrix:



The cylinder design then reproduces a hollow cylinder.

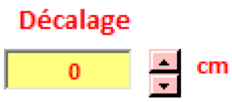
It is also possible to choose in the two previous cases one or more cylindrical screens





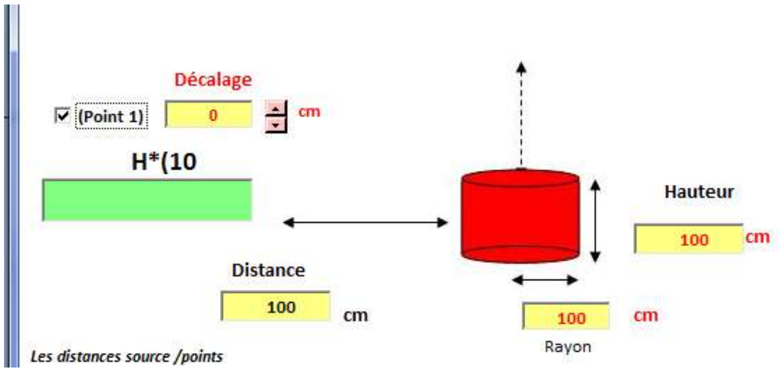
This new geometry makes it possible to better model geometries of the "Pipe" type containing a fluid (full cylinder) or for example a pipe contaminated on the internal surface by taking a thickness that is low in air.

II.3.3 OFFSET FUNCTION WITH CYLINDRICAL SOURCE AND WIRE

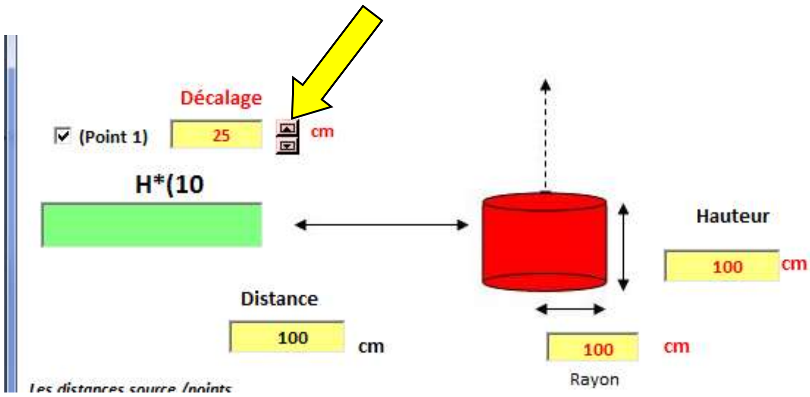


The “offset” function allows you to modify the position of the point located next to the cylinder (point 1).

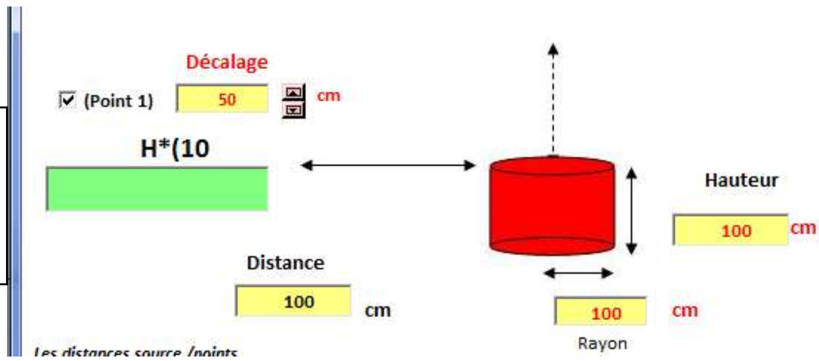
By default the initial position is halfway up the cylinder: offset 0



It is possible to modify this position ("altitude") by setting an offset from the middle position.
This modification can be done using the router or by manually entering the offset value

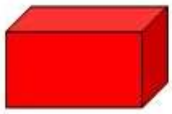


The maximum offset is equal to the mid-height of the cylinder (or half-height of the wire



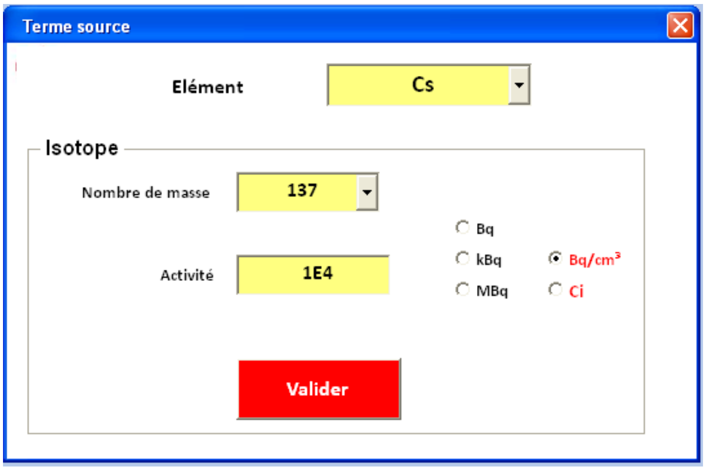
II.3.4 VOLUME ACTIVITY GRADIENT WITHSOURCE PARALLELEPIPED

The possibility of defining a variable volume activity has been integrated into the parallelepiped geometry, along the horizontal axis. This type of situation can be encountered for example with activities induced by a neutron fluence, or by slow processes of chemical migrations in the material (contamination)



Parallélépipède

This option can only be activated by choosing an activity in terms of volume activity. The value chosen is in all cases the value corresponding to the maximum volume activity $A_{vol.\,max.}$.



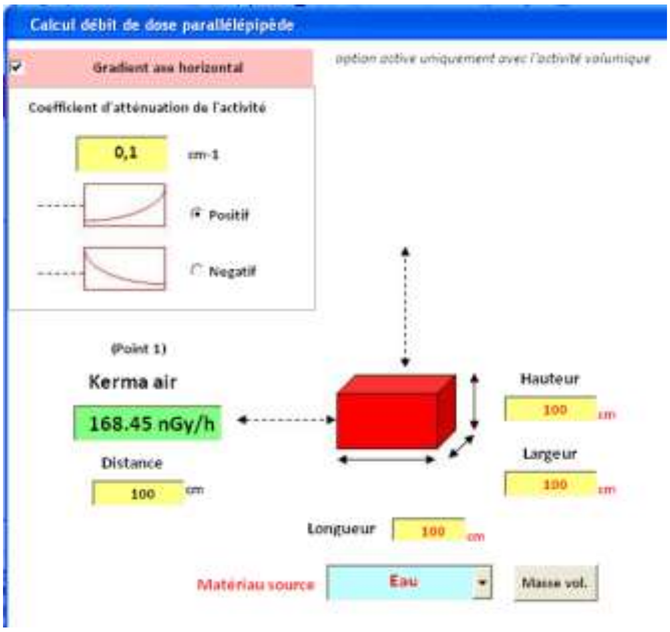
By choosing the "horizontal axis gradient" option on the parallelepiped dialog box, you can define an exponential gradient along the horizontal axis with two possibilities:

- ✓ **Positive gradient**: the face measured is then the one which is the least active. the gradient is then defined by: $A_{vol}(x) = A_{vol.min.} exp(\mu x)$.

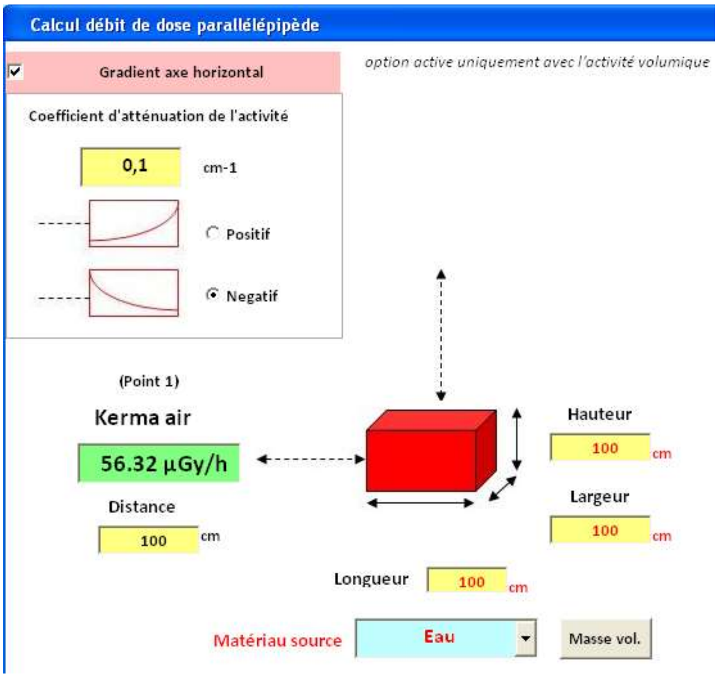
The minimum activity is determined from $A_{vol.\,max.}$ by : $A_{vol.\,min.} = A_{vol.\,max.} exp(-\mu L)$ with:

- μ : the attenuation (or relaxation) coefficient of the volume activity chosen by the user based on experimental data (core samples for example)
- L : the total length of the parallelepiped along the horizontal axis

The gradient is then expressed according to the relation $A_{vol}(x) = A_{vol.max.} exp\left[\mu\left(x - L\right)\right]$



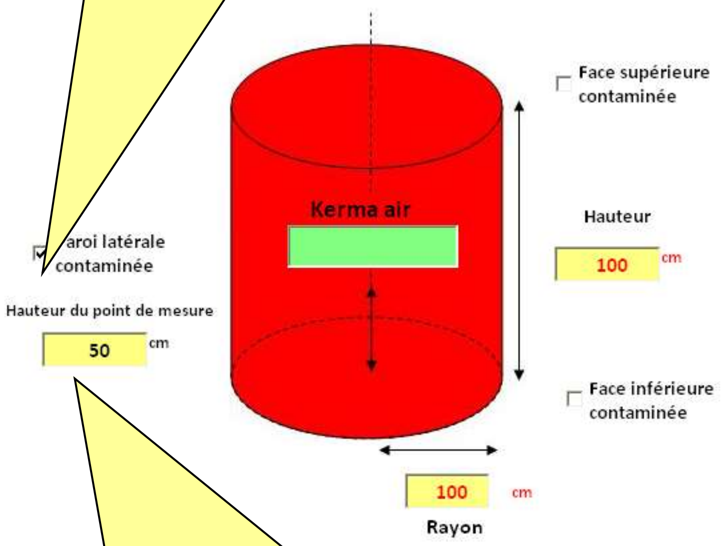
- ✓ **Negative gradient**: the face measured is then the one that is most active. the gradient is then defined by: $A_{vol}(x) = A_{vol.max.} exp(-\mu x)$.



II.3.5 SPECIAL CASES OF “PIPE” TYPE SOURCES

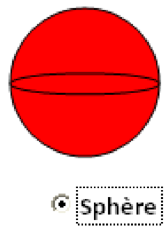
The sources "Contaminated cylinder or parallelepiped" allow the calculation of dose equivalent flow rates within these contaminated

You can choose contaminated surfaces, lateral or superior.
With this in mind, it is preferable to choose surface activities (Bq / cm2)

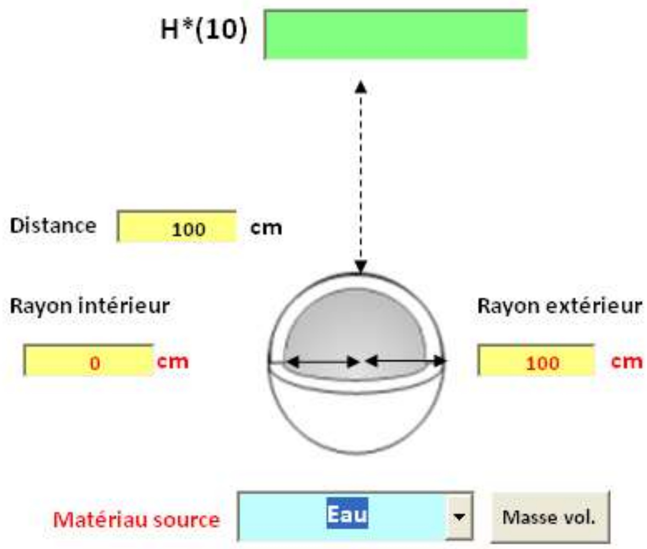


We can the height of the dose point on the internal central axis of the geometry

II.3.6 SPECIAL FEATURES OF THE SPHERICAL SOURCE

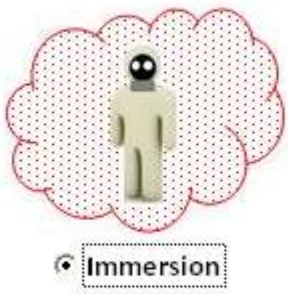


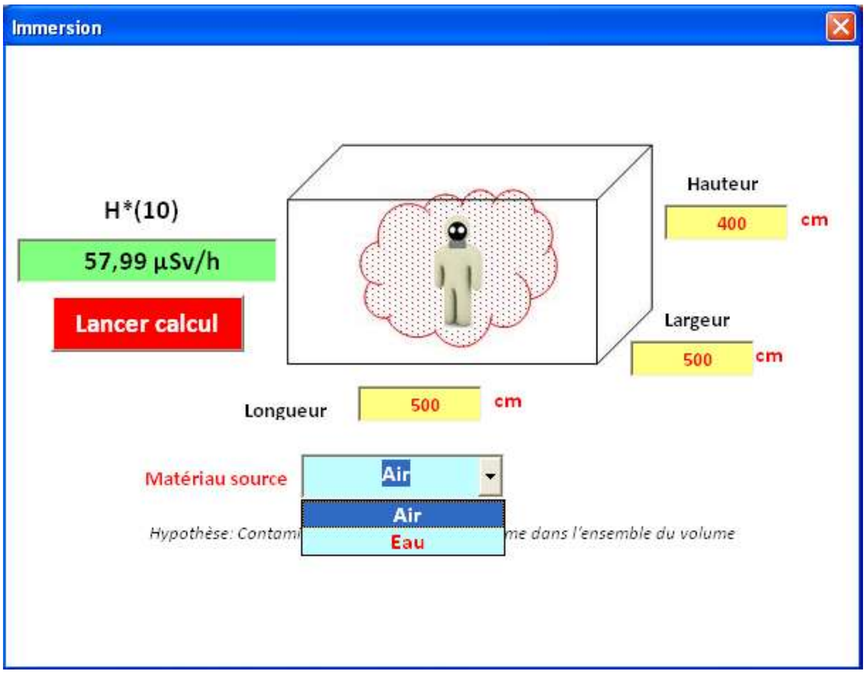
A particularity of the spherical source is to be able to define either a full spherical matrix or a contaminated crown with an empty central volume



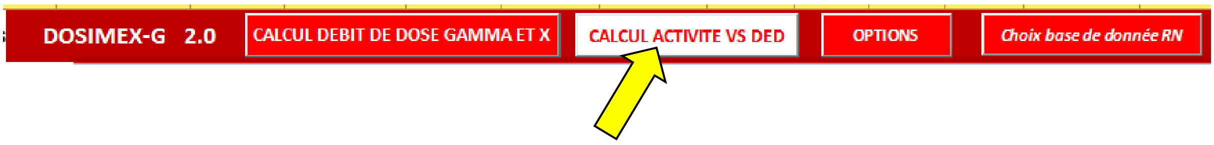
II.3.7 CLOUD IMMERSION

This option, common with Dosimex-I 2.1, makes it possible to calculate the radiological impact of external exposure for an individual immersed either in a volume of contaminated air with a uniform volume activity, or a volume of water (swimming pool). In this sense only the external gamma component is taken into account. (see Dosimex-I for the other components





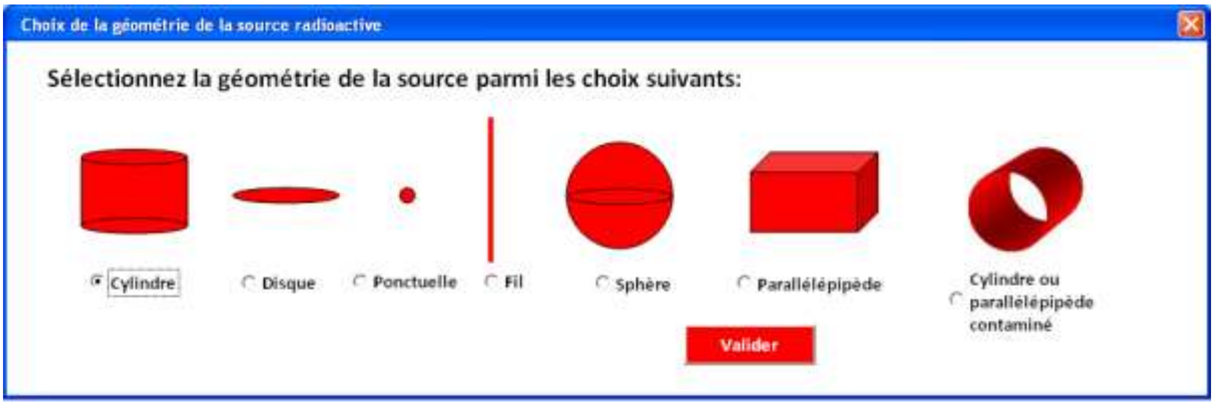
II.4 REVERSE ACTIVITY VS DOSE RATE CALCULATION



This option makes it possible to calculate the activity of a radioactive source as a function of the dose rate measured in H * (10) as a function:

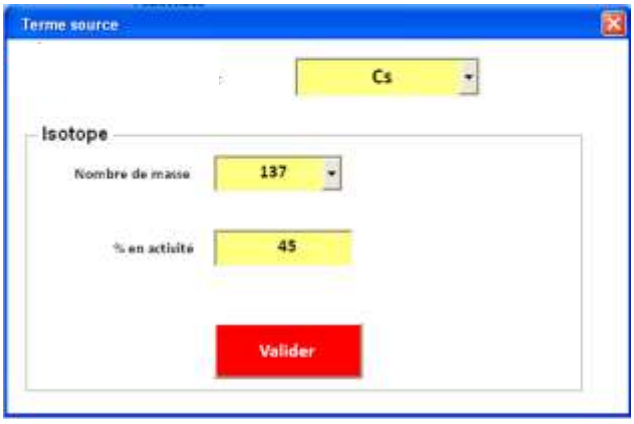
- Source-detector distance
- Source configuration (geometry, screen, etc.)
- Radionuclide spectrum in terms of proportions in activities

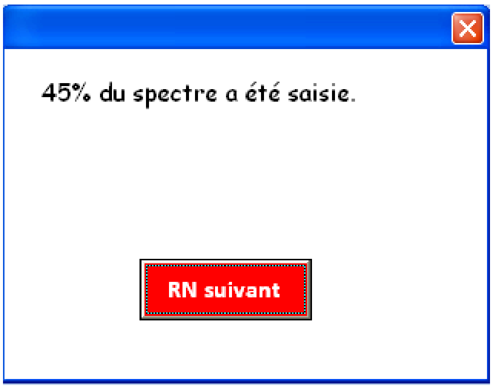
Choice of source configuration:

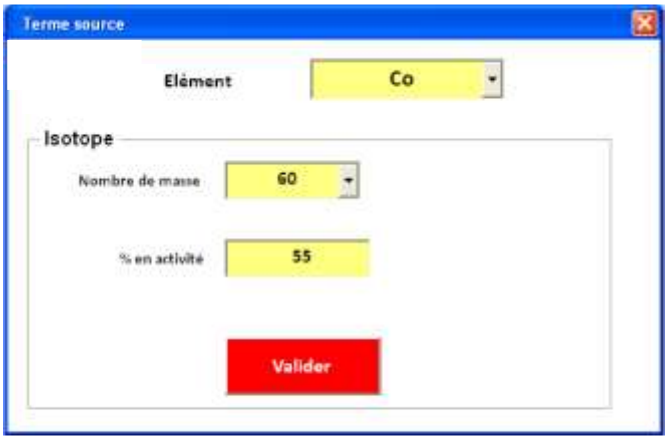


Determination of radionuclides and their proportion in activity in the spectrum:

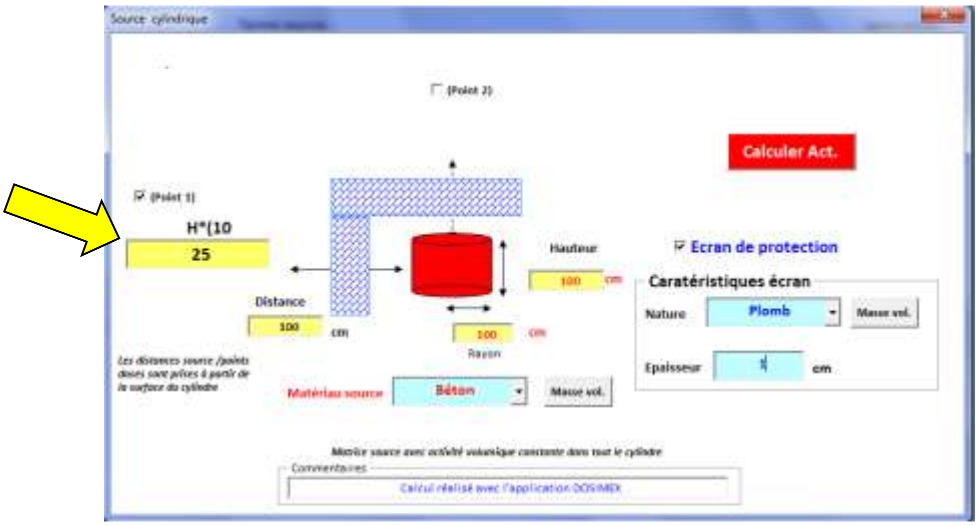
Example for 45% in DHW 137 activity and 55% in CO 60



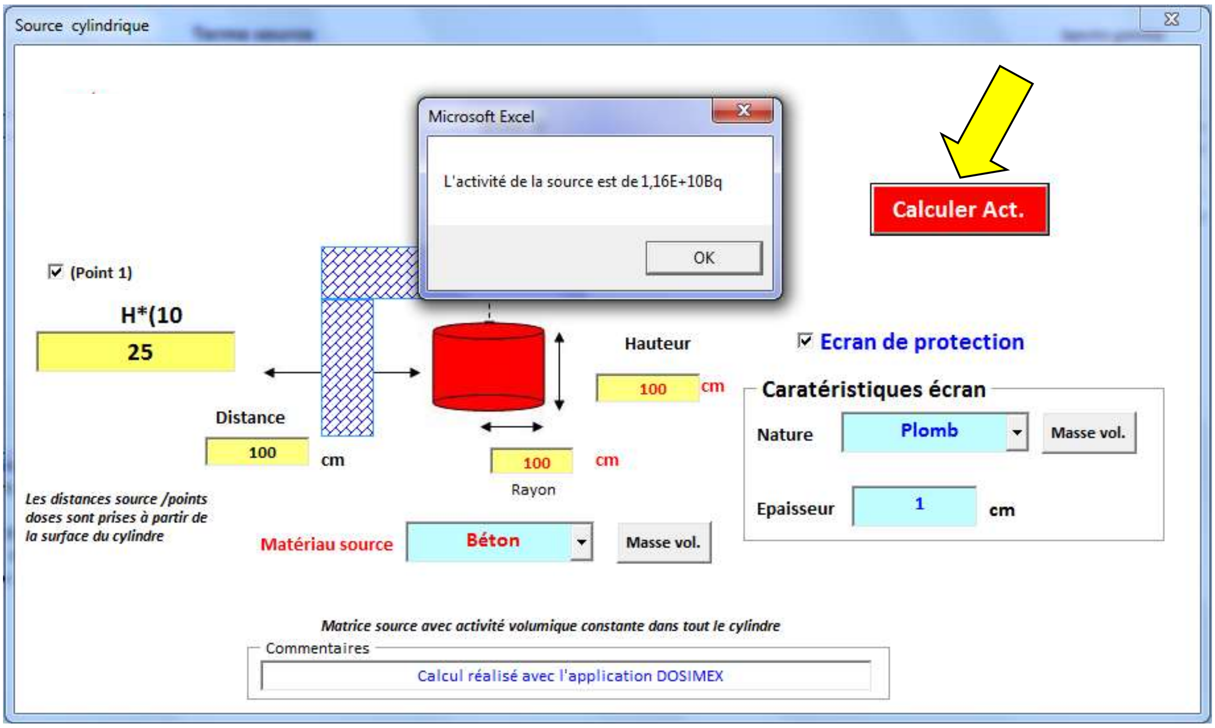




When 100% of the spectrum has been entered, the dialog box for the selected source appears and allows you to enter the measured dose rate:



It is then possible to launch the calculation to first obtain the total activity of the source:

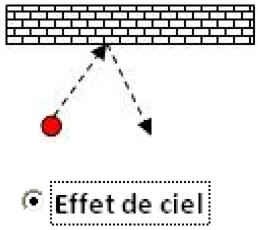


The summary sheet displays the activity by radionuclide:

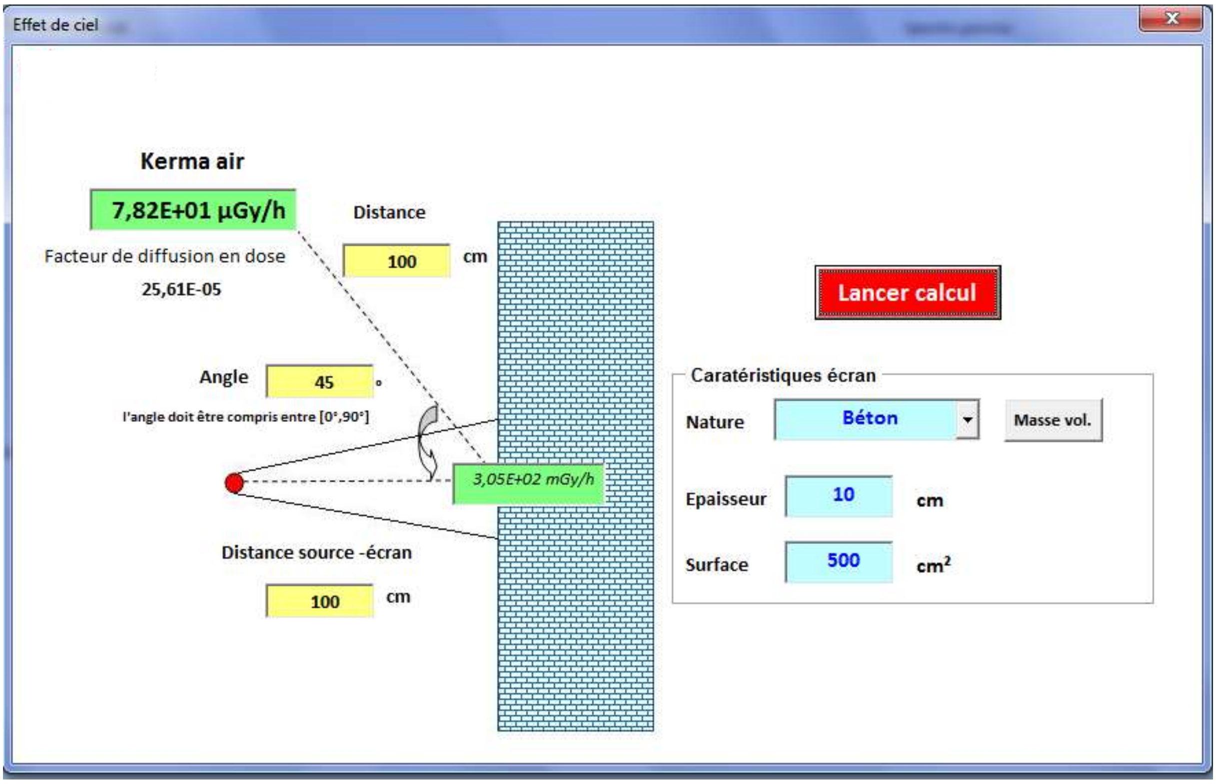
II.5 SKYSHINE CALCULATION

II.5.1 IMPLEMENTATION

The "skyshine" option makes it possible to determine the dose equivalent rate generated by the diffusion of primary radiation generated by a point source consisting of one or more radionuclides on a plate of a determined material.



After entering the various dialog boxes identical to the point source options (choice RN, activity), the skyshine dialog box appears.



Dialog box "skyshine", calculated here for 1TBq of Co 60

The parameters taken into account are:

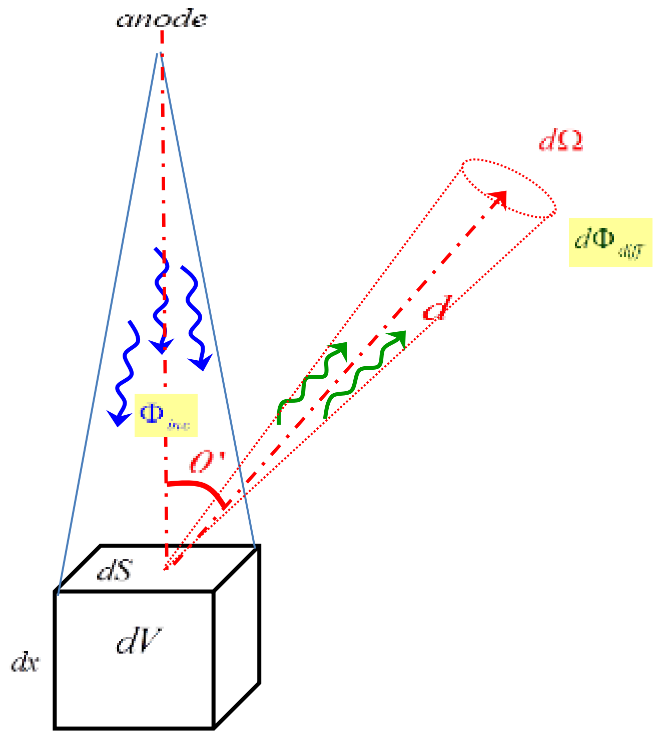
- The distance from the source to the right of the screen
- The angle between the normal angle of incidence of primary radiation and the position of the source. This angle is between 0 ° (backscatter towards the source) and 90 ° (direction parallel to the screen)
- The distance on the scattering axis at which one wishes to calculate the dose rate of the scattered radiation.
- The nature of the plaque
- Its thickness
- The irradiated surface, generally a function of a collimation angle (it is up to the user to determine this surface). This area is limited to 25.104 cm2 (25 m2)

- The calculation is based on the differential cross section of Klein and Nishina (see appendix H: MRI photon) and account on the one hand for the attenuation of the primary fluence in the irradiated plate as well as for the absorption of the scattered photons before emergence of the plaque

The code gives the value of the dose equivalent flow in the incident beam at the screen as well as the value of the flow generated by the component scattered at the desired point. This value does not take into account possible irradiation in a straight line at this point. If necessary, such a component can be calculated directly with the point source option

The diffusion factor is the ratio of the flow generated by the diffusion to the flow in the primary beam.

II.5.2 METHOD IMPLEMENTED BY DOSIMEX-GX 2.0 FOR THE CALCULATION OF THE SCATTERED RADIATION.



Schematic diagram: diffusion in a volume element.

The dimensions of the target being small compared to the average free path, we can consider that the fluence is homogeneous in the target volume of elementary volume $dV = dx\,dS$. Under these conditions, we can show that the number of photons scattered throughout the space is equal to:

$$dN_{diff.} = \mu_{Compt.}\Phi_{inc.}\,dV \quad (1)$$

With $\mu_{Compt.}$ the Compton linear attenuation coefficient of the material for the energy of incident photons still equal to $\mu_{Compt.} = n\,\sigma_{Compt.}$ with:

- n : the density of the target in number of nucleus
- $\sigma_{Compt.}$: the Compton cross section (ie all angles combined)

If the scattering was isotropic, the fluence scattered by this target element at a distance d would be equal to:

$$d\Phi_{diff} = \frac{dN_{diff.}}{4\pi d^2} = n\frac{\sigma_{Compt.}}{4\pi d^2}\Phi_{inc.}dV = n\frac{\sigma_{Compt.}}{4\pi}\Phi_{inc.}dVd\Omega \quad (2a)$$

Or again, taking into account that $\frac{1}{d^2}$ equals the solid corner element $d\Omega$:

$$d\Phi_{diff} = n\frac{\sigma_{Compt.}}{4\pi}\Phi_{inc.}dVd\Omega \quad (2b)$$

By taking into account the non-isotropic character of the Compton scattering through the differential cross section of Klein and Nishina, a function of the scattering angle and of the energy of the photons, we can write

$$\frac{d\sigma_{e-,Compt.}}{d\Omega}(\theta) = \frac{1}{2}r_0^2\frac{1+\cos^2\theta}{\left(1+\alpha(1-\cos\theta)\right)^2}\left[1+\frac{\alpha^2(1-\cos\theta)^2}{\left(1+\cos^2\theta\right)\left(1+\alpha(1-\cos\theta)\right)}\right] \text{ with } \alpha = \frac{E_\gamma}{m_e c^2}$$

The relation (2b) then becomes:

$$d\Phi_{diff}(\theta) = n\frac{d\sigma_{e-,Compt.}}{d\Omega}(\theta)\Phi_{inc.}d\Omega dV = n\frac{d\sigma_{e-,Compt.}}{d\Omega}(\theta)\Phi_{inc.}d\Omega\, dx\, dS \quad (3)$$

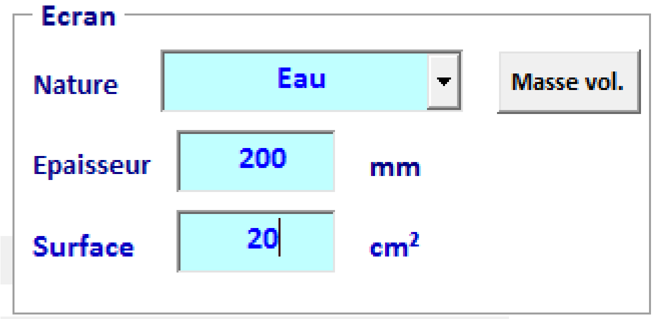
Note: compared to the block diagram above, the angle θ considered in the above formula is equal to $\theta = 180° - \theta'$

The complete calculation of the scattered radiation for a large target (several average free paths) is done by integrating this relationship on all the energies of the X spectrum but especially on the entire surface of the target and all the depth, taking into account the absorption of the incident and scattered beam.

<u>This relation shows, for fairly large distances in front of the dimensions of the target, that the incident fluence is, among other things, proportional to the irradiated surface S of the target.</u>

The DOSIMEX-GX 2.0 Code implements this model to calculate the incident fluence at a given angle and distance based on:

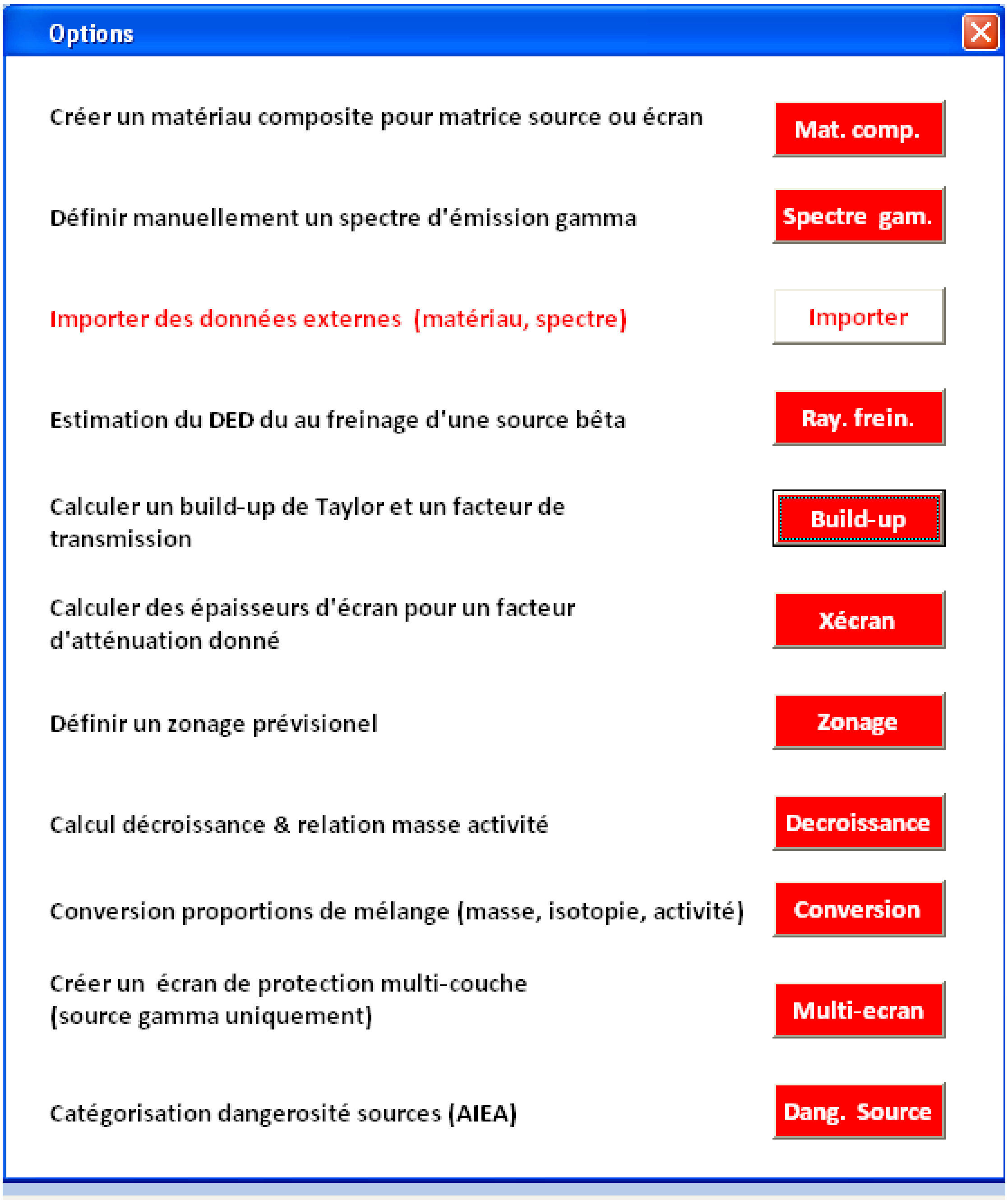
- The nature of the target
- Of its thickness
- From its surface



II.6 Options.

The DOSIMEX-G application also offers a panel of options accessible by clicking on the active "Options" button:

A dialog box offering 11 options opens:

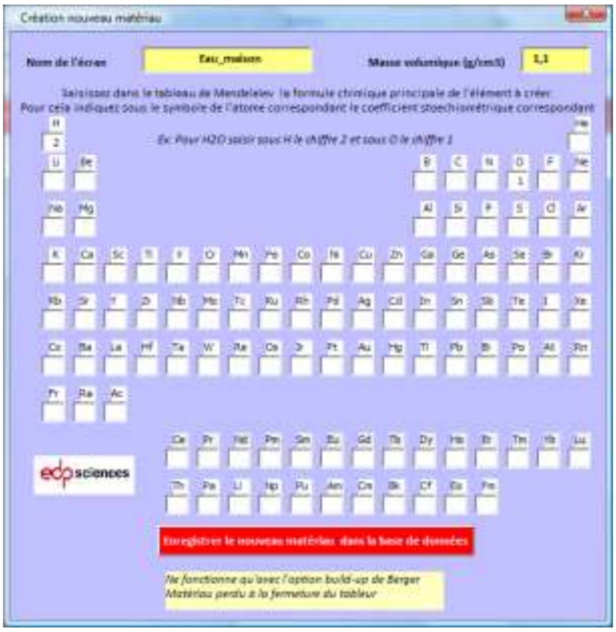


II.6.1 "COMPOSITE MATERIAL" OPTION

Mat. comp.

The choice of the option "Berger build-up model" allows the use of any materials, simple or composite, both for the source matrix and for the screens installed. The nature of the material must be defined beforehand with the "Mat.comp" option:

A dialog box type periodic table appears and offers you to define the chemical composition of your screen via atomic proportions (stoichiometric coefficients) as well as its total density.



When this dialog box is closed, your material will be accessible for all calculation configurations (excluding X generator) using the Berger build-up.

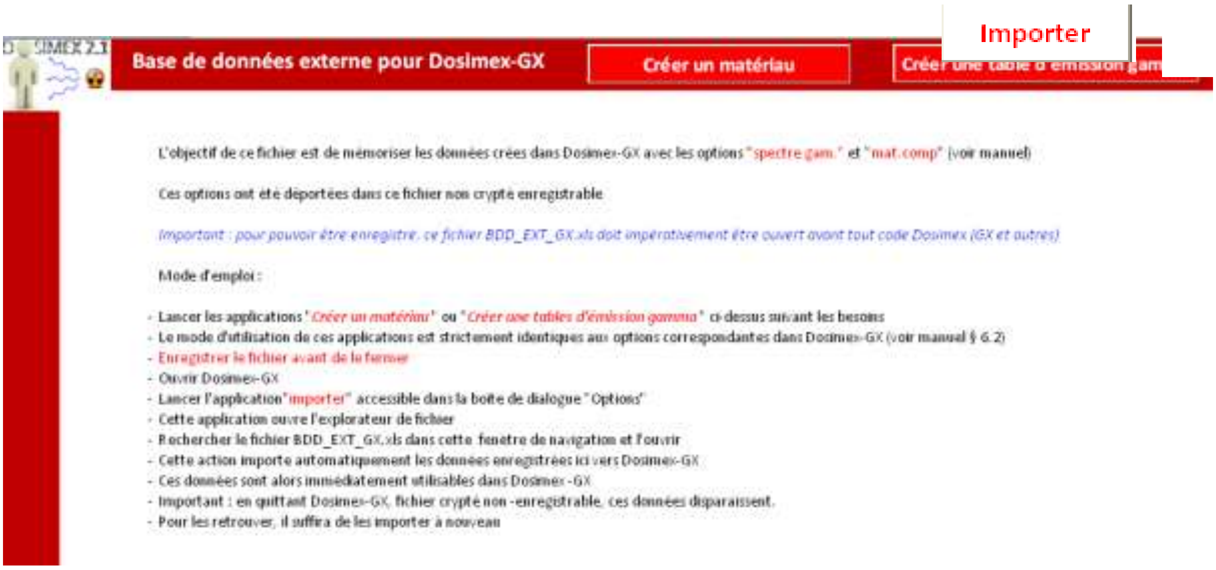
II.6.2 OPTION "MANUALLY DEFINE A GAMMA EMISSION SPECTRUM"

Spectre

Enter the radionuclide symbol (X), its mass number (A). Then define the emission spectrum of 1 to 5 possible lines by specifying energy (s) and intensity (s) of emission and validate by recording it. Several standard spectra can be entered successively. The radionuclides entered will be placed at the end of the radionuclide database in use.

If the radionuclide to be recorded has more than 5 gamma lines, record your first 5 data then repeat the operation by entering the same symbol and the same mass number. In order for your radionuclide to be correctly taken into account later, enter the information relating to a new radionuclide only after having fully completed the first one.

II.6.3 IMPORT OF MATERIAL DATA OR EMISSION TABLES FROM AN EXTERNAL FILE



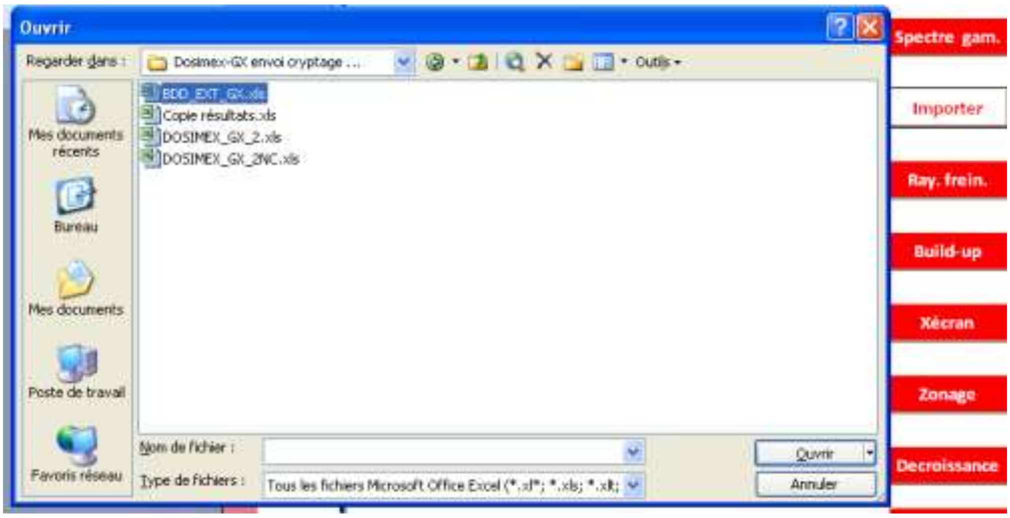
The purpose of this file **BDD_EXT_GX.xls** is to store the data created in Dosimex-GX with the options "gam spectrum. " and "mat.comp"previously defined.

These options have been deported to this recordable unencrypted file

Important: to be able to be saved, this BDD_EXT_GX.xls file must be opened before any Dosimex code (GX and others)

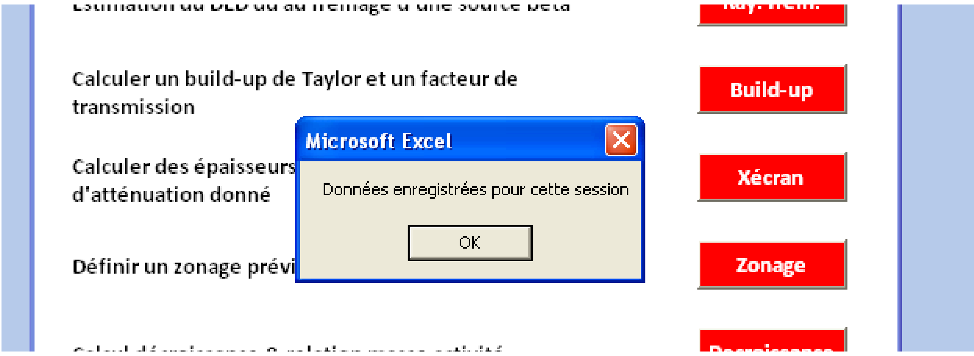
Manual :

- Launch applications "***Create a material***" or "***Create a gamma emission table***"which have been deported in this file (even if they still exist in Dosimex-GX 2.1)
- The mode of use of these applications is strictly identical to the corresponding options in Dosimex-GX
- **Save the file then close it**
- Open Dosimex-GX
- Launch the application"import"accessible in the" Options "dialog box
- This application opens the file explorer:



- Find file **BDD_EXT_GX.xls** in this navigation window (Dosimex-GX folder in the operational pack) and open it

- This action automatically imports the data saved here to Dosimex-GX



- This data can then be immediately used in Dosimex –GX

- Important: when leaving Dosimex-GX, an encrypted non-recordable file, this data disappears.
- To find them, simply import them again from**BDD_EXT_GX.xls**

Note: you can duplicate this file as much as you want by storing them in additional folders.

II.6.4 OPTION "CALCULATION OF THE DOSE RATE DUE TO THE BRAKING RADIATION OF A BETA SOURCE"

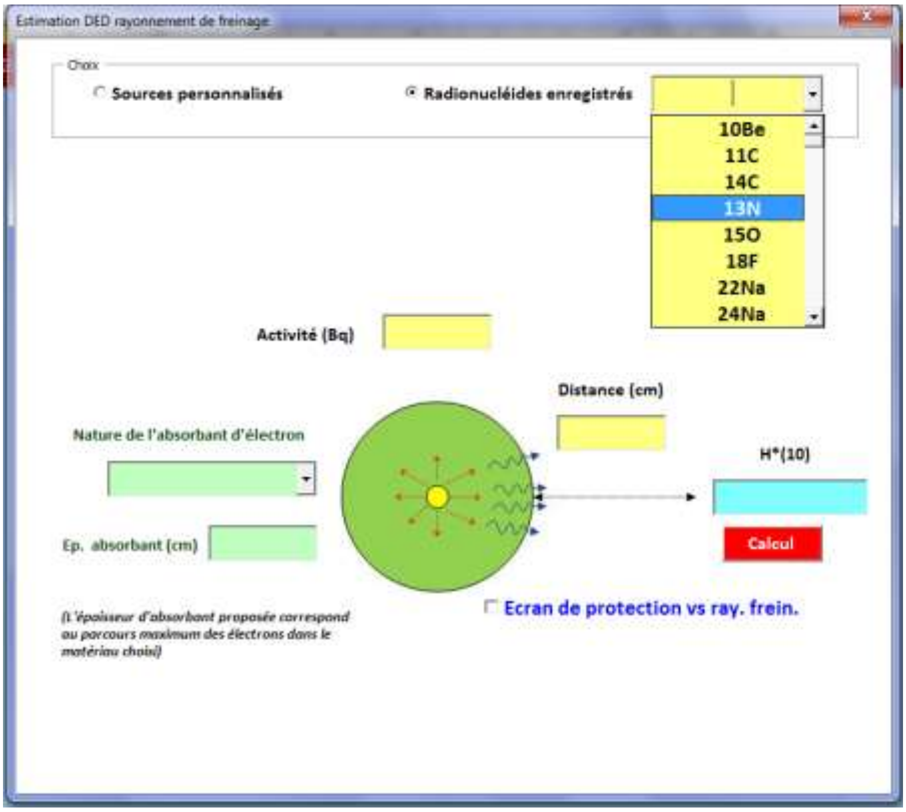
This option makes it possible to calculate the dose rate due to the braking radiation generated in a material acting as an electron absorber (in green) by the beta emission of a source. The thickness of the absorber is such that the electron / X conversion is total, which amounts to saying that no electron emerges from this absorber. The value of the dose obtained is thus only a dose "X". The dialog box does not take into account primary screen thicknesses less than the maximum range of electrons, calculated when the operator chooses the nature of the screen

In the opposite case, for an absorber with a thickness less than the maximum range of electrons, the DOSIMEX-B code must be used to calculate the “beta” dose.

The application determines the beta spectrum of the radiation source, then the braking X spectrum to finally determine the dose rate due to the braking radiation. It is possible to set up a secondary screen (in blue) to calculate the attenuation of the braking X-ray. In the case of a transmitter β^+ , this application takes into account the emission of 511 keV annihilation photons.

There are two possibilities to determine the beta transmitter:

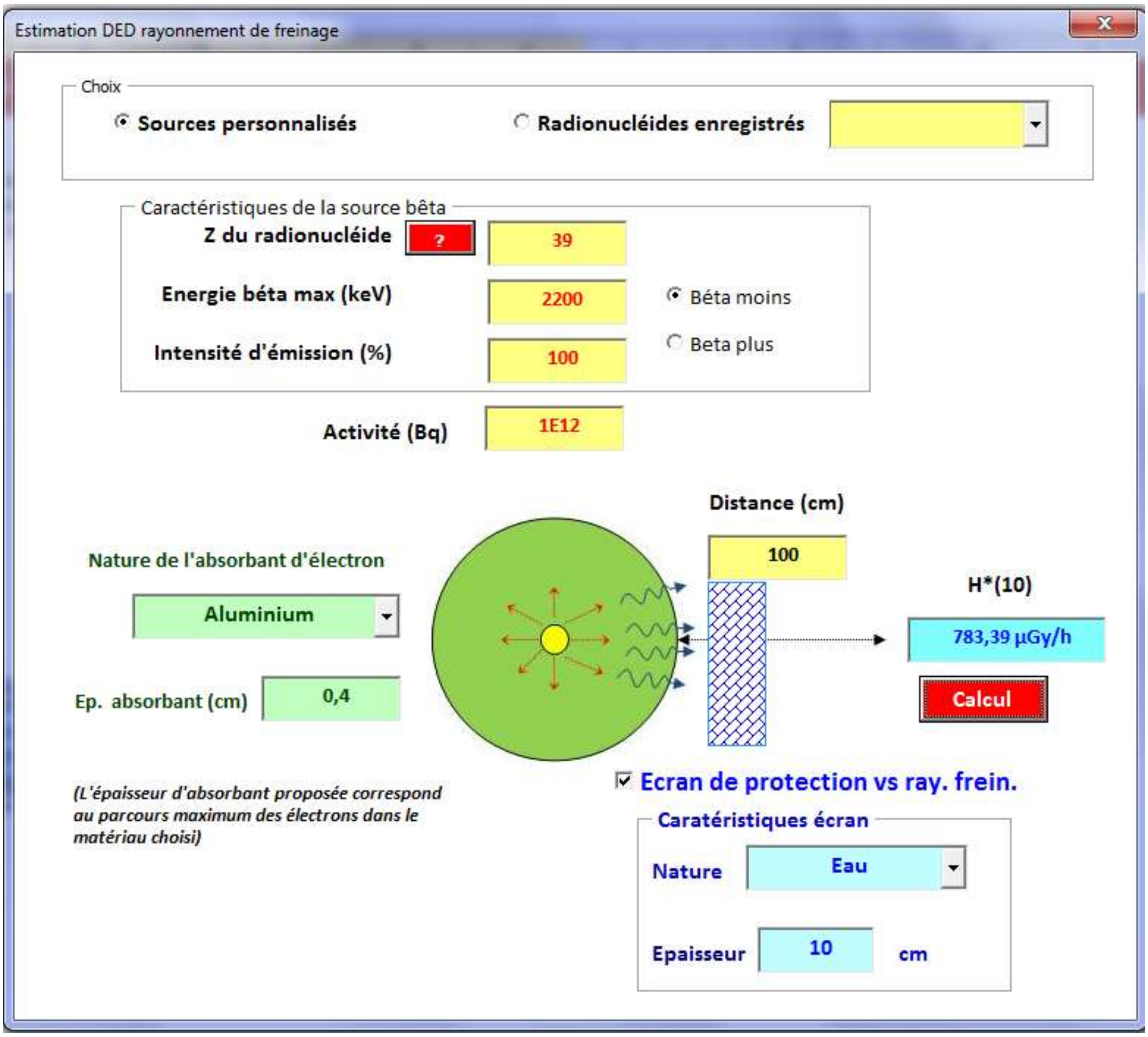
1) Or a drop-down list offering the essentials of the radionuclides taken into account in the Practical guide (Delacroix et al.)



Note: the code only calculating from a single component, the emission spectrum is approximated by an average Qmax.

2) Or by manually entering the characteristics of the beta component:

- Beta - or +
- Qmax
- Emission intensity



When using this option, a summary sheet is edited presenting the elements associated with the simulation:

DOSIMEX-G 2.0 | CALCUL DEBIT DE DOSE GAMMA ET X | CALCUL ACTIVITE VS DED

Terme source

Source

Source béta - de Z= 39 — A/I beta

Source gainée de 0,4cm de Aluminium — 1E12Bq / 100%

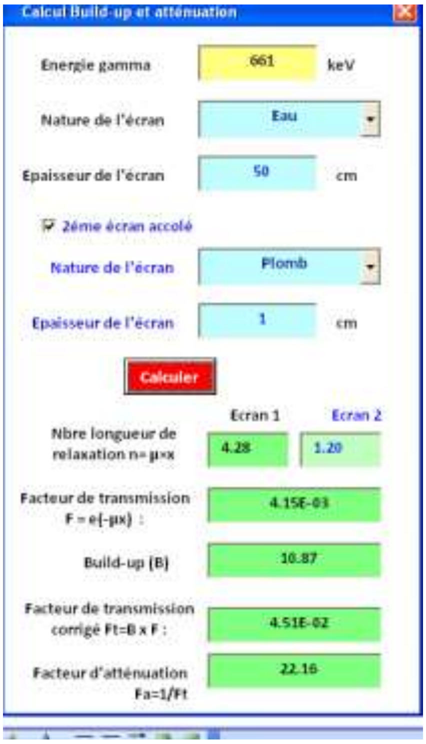
Type de build-up: Taylor — Configuration avec écran de Eau de 10cm

	Avec Build-up	Sans Build-up	Build-up moyen
Kerma	608,58 µGy/h	208,98 µGy/h	2,91
H*(10)	783,39 µSv/h	269,21 µSv/h	
H'(0,07)	772,19 µSv/h	265,36 µSv/h	
Hp(10)	806,64 µSv/h	277,2 µSv/h	
Hp(3)	768,19 µSv/h	263,98 µSv/h	

II.6.5 "BUILD-UP" OPTION

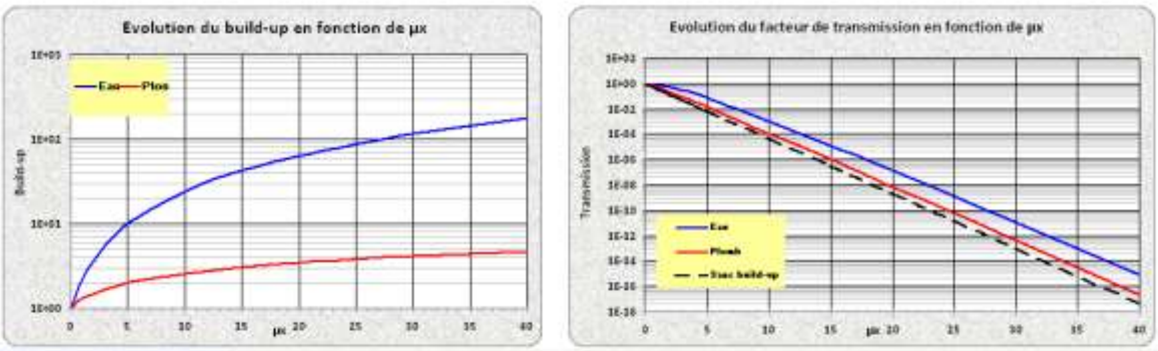
Build-up

This option allows a calculation of build-up (BU), of the transmission factor (F), and the product of the 2, for simple configurations, equivalent to a monoenergetic point source (limited from 10 keV to 15 MeV) for one or two adjoining screens.



The summary sheet then indicates the curves and tables of evolution of the Taylor build-up of the transmission factor.

661 keV		Eau			Plomb		
μX	Transmission (F)	X (cm)	Build-up (B)	F*B	X (cm)	Build-up (B)	F*B
0	1.00E+00	0.00E+00	1.00E+00	1.00E+00	0.00E+00	1.00E+00	1.00E+00
1	3.68E-01	1.17E+01	2.14E+00	7.88E-01	8.33E-01	1.30E+00	4.78E-01
2	1.35E-01	2.33E+01	3.70E+00	5.00E-01	1.67E+00	1.50E+00	2.03E-01
3	4.98E-02	3.50E+01	5.57E+00	2.77E-01	2.50E+00	1.69E+00	8.42E-02
4	1.83E-02	4.67E+01	7.69E+00	1.41E-01	3.34E+00	1.84E+00	3.37E-02
5	6.74E-03	5.83E+01	9.97E+00	6.72E-02	4.17E+00	1.98E+00	1.33E-02
10	4.54E-05	1.17E+02	2.41E+01	1.09E-03	8.34E+00	2.59E+00	1.18E-04
15	3.06E-07	1.75E+02	4.24E+01	1.30E-05	1.25E+01	3.08E+00	9.43E-07
20	2.06E-09	2.33E+02	6.39E+01	1.32E-07	1.67E+01	3.48E+00	7.17E-09
30	9.36E-14	3.50E+02	1.15E+02	1.08E-11	2.50E+01	4.11E+00	3.85E-13
40	4.25E-18	4.67E+02	1.76E+02	7.48E-16	3.34E+01	4.63E+00	1.97E-17



II.6.6 "CALCULATE SCREEN THICKNESSES FOR 

UATION FACTOR" OPTION

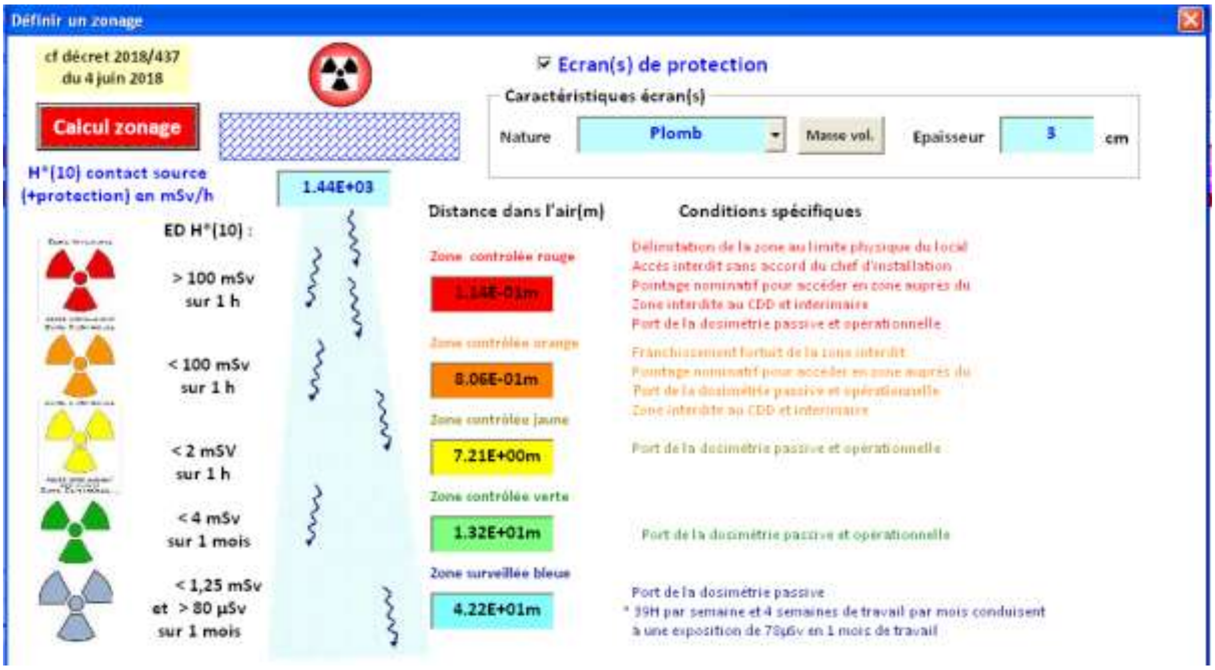
This option makes it possible to define, for a given energy and a given material, the screen thickness necessary to obtain any attenuation factor chosen by the operator (1/2, 1/10… 1 / n etc ..).
Two screen thicknesses are calculated, the first, resulting from the classical calculation of exponential attenuation, not taking the build-up into account, and the second, invariably greater than the previous one, taking into account the build-up. It is possible to use the two build-up models, Taylor and Berger. The interest, with the build-up of Berger, being to be able to use this option with a composite screen defined in the option "Composite material"

Example of tenth screen calculation for 1332 keV in lead with the Taylor model:

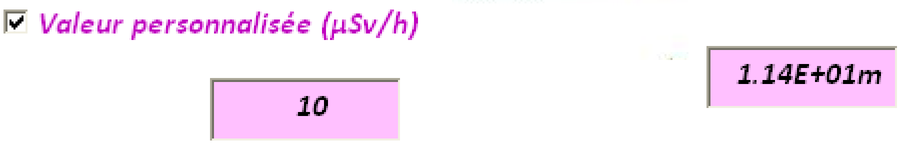
II.6.7 "DEFINE A PROVISIONAL ZONING" OPTION

This option allows you to define the distances corresponding to the different radiological zones for a point source (1 or more possible radionuclides) with or without a protective screen. The calculations (solver) take into account attenuation and build-up in the air.

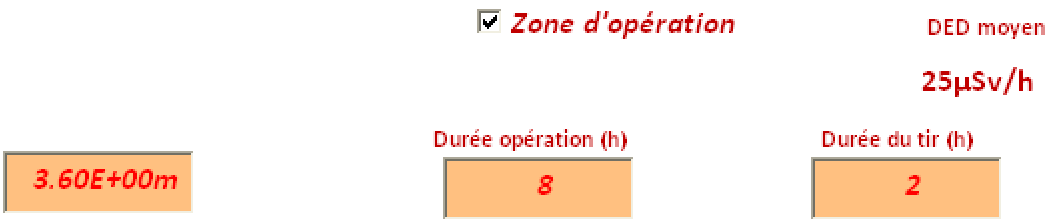
The zoning takes into account the latest modifications made by decree 2018/437 of June 7, 2018. Example for 40 Ci of iridium 192 with a 3 cm lead screen.



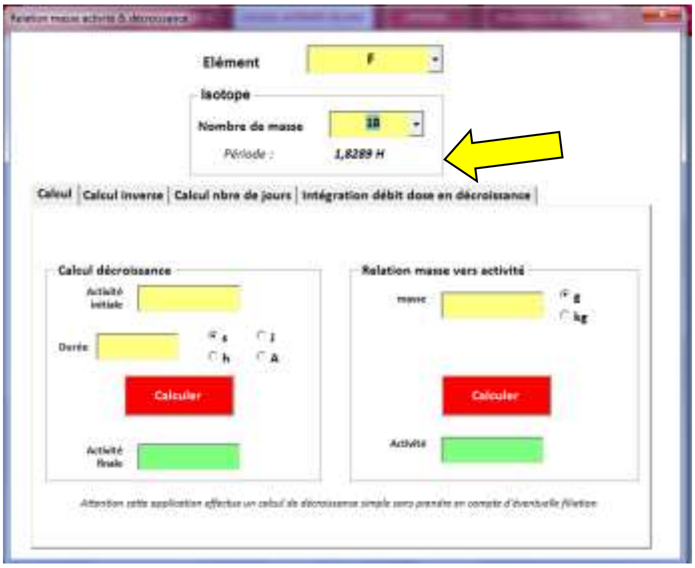
A first additional option makes it possible to know the distance from an area for a chosen value of the dose rate:



A second option makes it possible to define an area of operation in proportion to time:

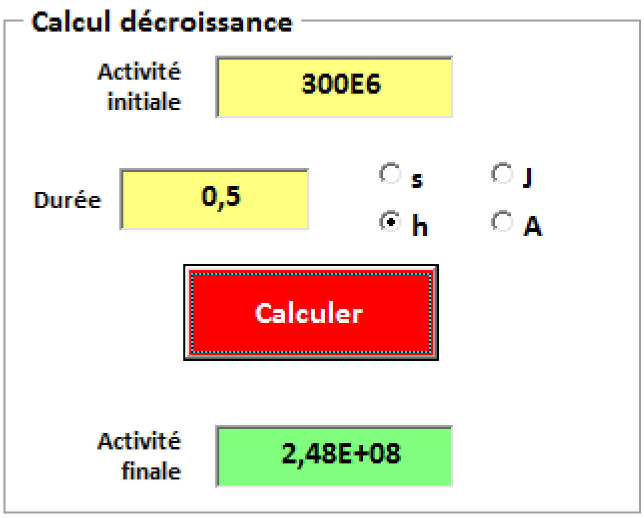


II.6.8 "DECAY" OPTION

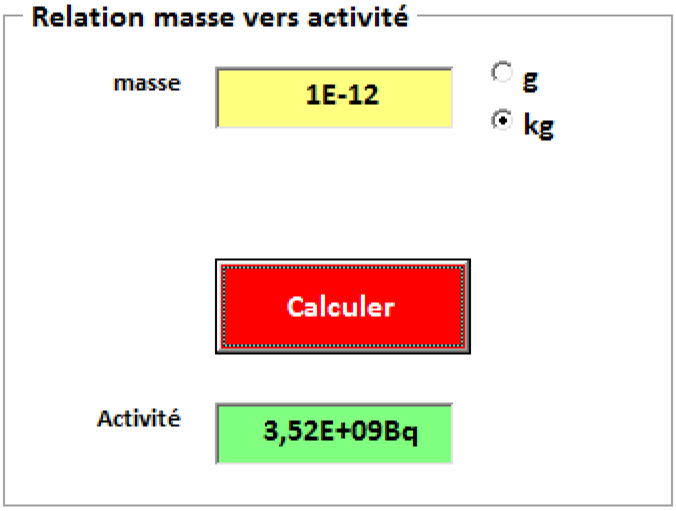


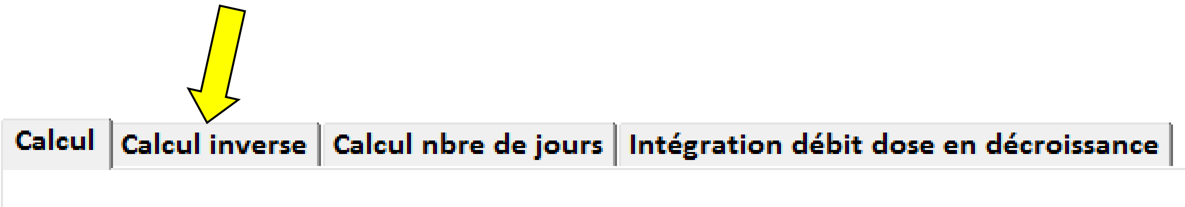
After choosing a radionuclide, we obtain the first information: its period in seconds (s), hour (h), days (d) or year (A) as the case may be
(example here with Fluor 18)

The first tab "Calculation" allows you to calculate a decay correction
By returning to the initial activity and the duration of decline

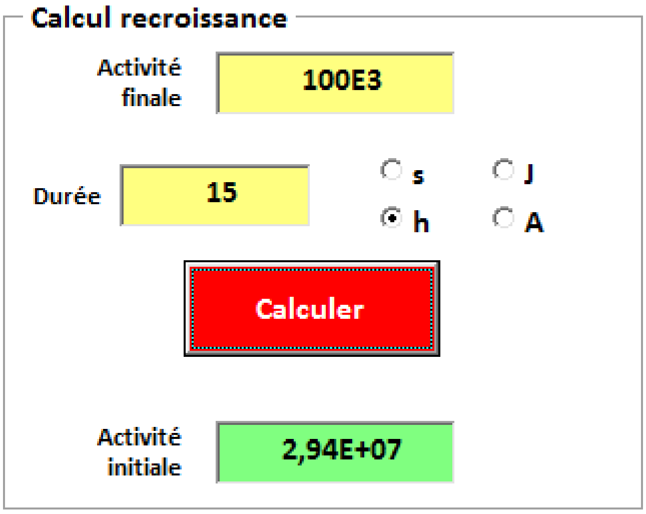


On the same tab it is possible to convert a mass (g or kg) into activity (Bq)

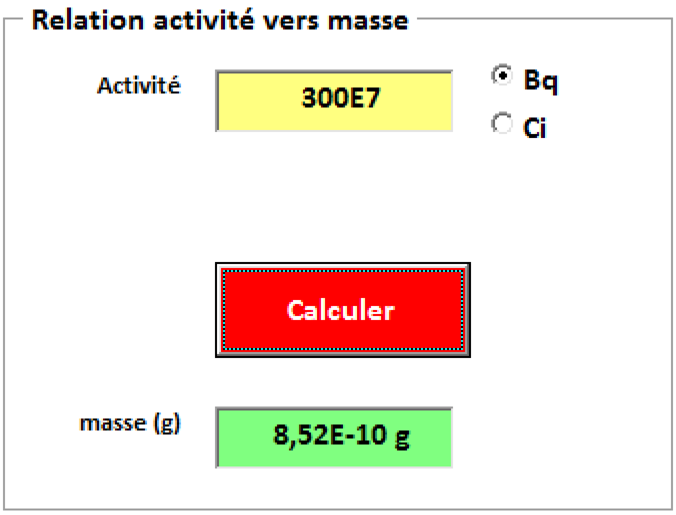




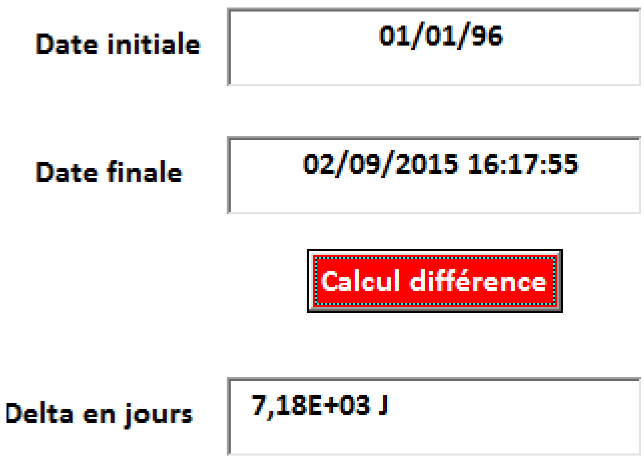
The second tab "Reverse calculation" allows you to calculate a decrease correction "in reverse", that is to say to go back to the initial activity of a source having decreased



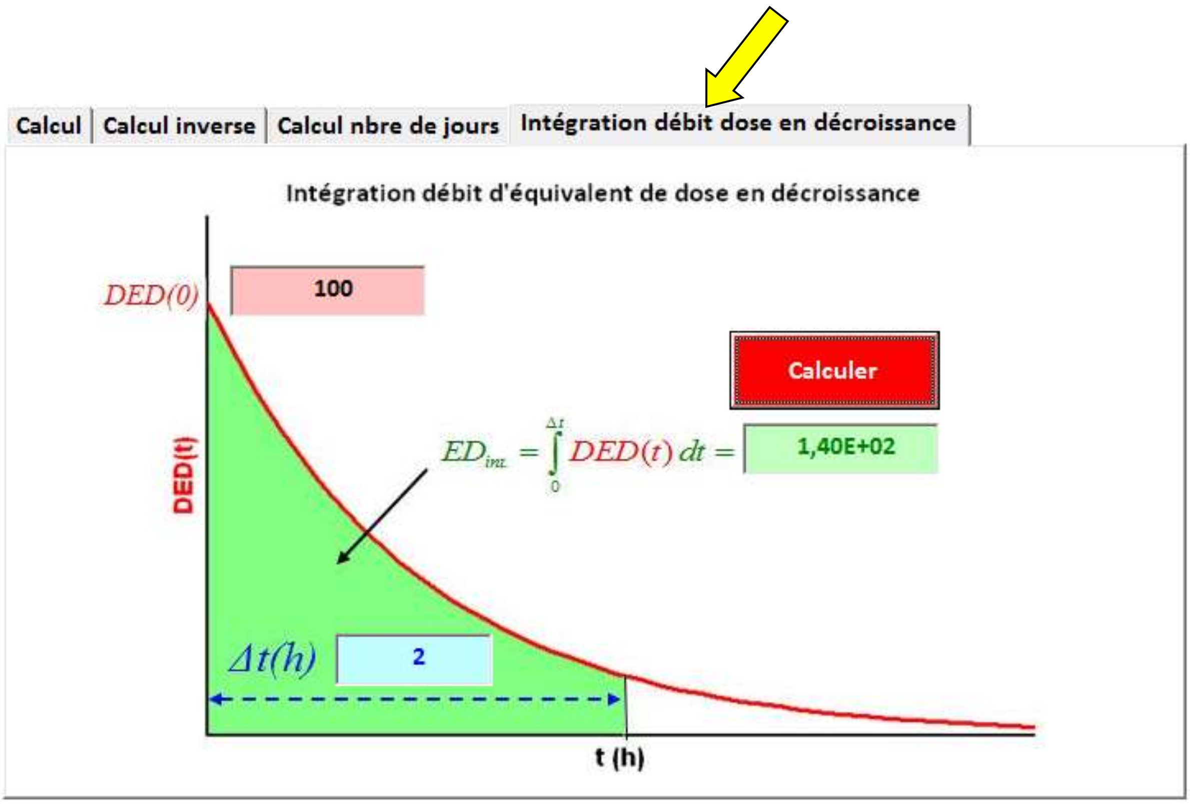
On the same tab it is possible to convert an activity (Bq) into mass (g or kg)



The "Calculation of days" tab allows you to convert a date difference into days, which can be used with the previous tabs
Initially the fields contain the current date ("today" function of Excel)



The last tab lets you know the equivalent dose integrated over a period of time, taking into account the decrease in the source. The duration considered is expressed in decimal hours, so that If the dose rate is considered in µSv / h, then the integrated dose will be expressed in µSv.



II.6.9 OPTION "CONVERSION OF PROPORTIONS OF MIXTURES"

Conversion

A. PRINCIPLES OF CALCULATIONS MIXTURE OF K ELEMENTS

Rating:

- ω_i : proportion by mass $\left(\sum_i \omega_i = 100\,\%\right)$
- I_i : isotopic proportion $\left(\sum_i I_i = 100\,\%\right)$
- Act_i : proportion by mass $\left(\sum_i Act_i = 100\,\%\right)$
- ✓ T_i : period in s and $\lambda_i = \ln(2)/T_i$ in s-1
- ✓ A_i : mass number

$$\text{Case 1: } I_i \text{ known} \Rightarrow \begin{cases} \omega_i = \dfrac{I_i\, A_i}{\sum_k I_k\, A_k} \\[2ex] Act_i = \dfrac{\lambda_i I_i}{\sum_k \lambda_k I_k} \end{cases}$$

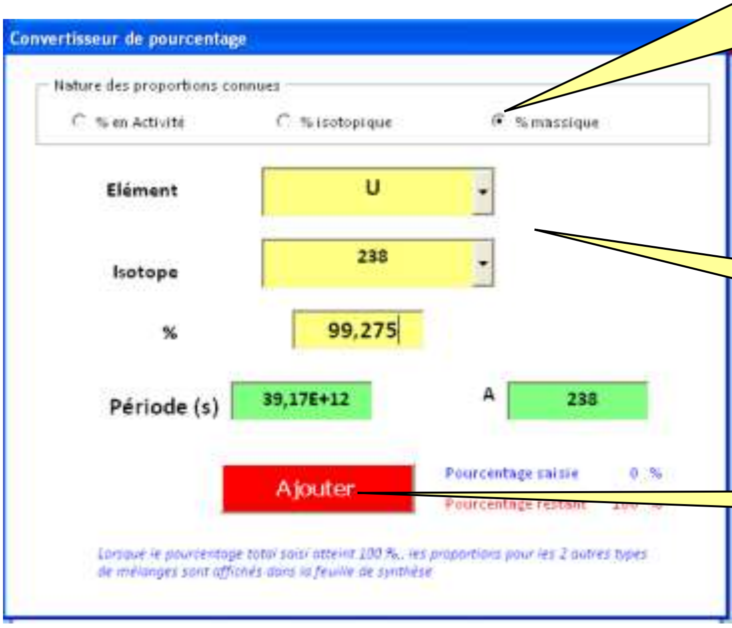
$$\text{Case 2: } \omega_i \text{ known} \Rightarrow \begin{cases} calcul\,1: I_i = \dfrac{\omega_i / A_i}{\sum_k \omega_k / A_k} \\[2ex] Puis\,calcul\,2: Act_i = \dfrac{\lambda_i I_i}{\sum_k \lambda_k I_k} \end{cases}$$

$$\text{Case 3: } Act_i \text{ known} \Rightarrow \begin{cases} calcul\ 1: \omega_i = \dfrac{Act_i\ A_i / \lambda_i}{\sum_k Act_k\ A_k / \lambda_k} \\[2ex] Puis\,calcul\ 2: I_i = \dfrac{\omega_i / A_i}{\sum_k \omega_k / A_k} \end{cases}$$

B. IMPLEMENTATION

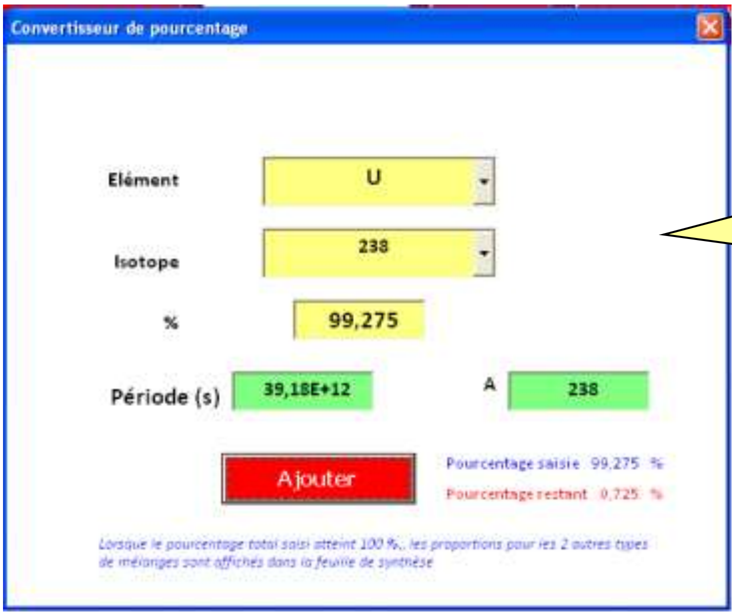
Example with the mass mixture of natural uranium

First choose the nature of the proportions of mixtures: Isotopy, mass or activity



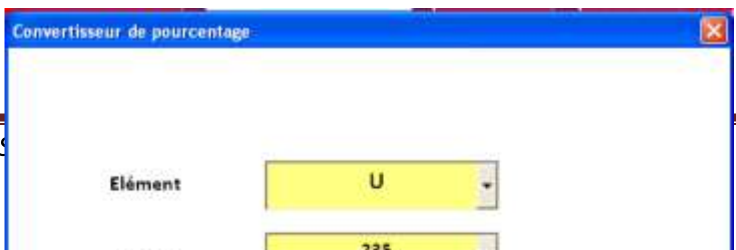
Then choose the first radionuclide and its mixing rate

Validate this choice



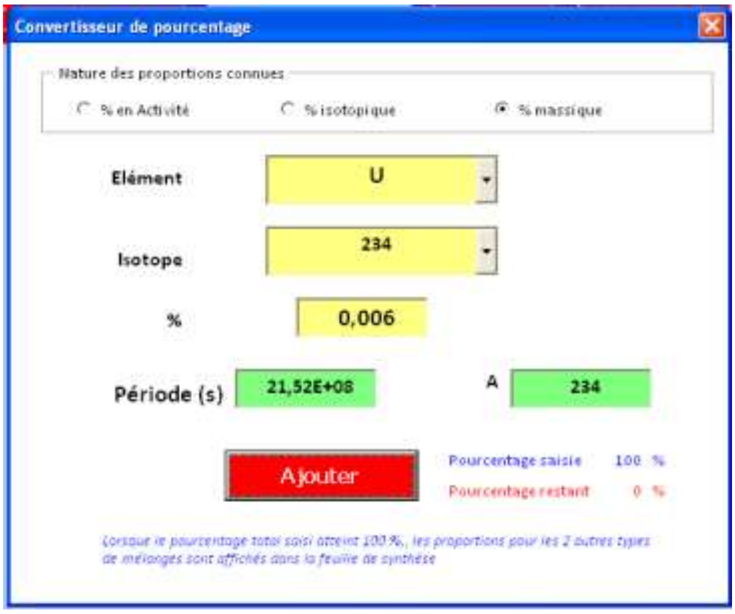
After validation the choice on the nature of the mixture disappears, to avoid any handling error

The proportion entered and the proportion remaining to be entered are displayed to facilitate the entire operation



Enter the second radionuclide, its proportion then validate

Enter the last radionuclide.



The entry stops when the sum of the proportions reaches 100%.
If the 100% is exceeded, an alert message appears and automatically adjusts

The results of the conversions for the other types of mixture are then displayed on the summary sheet:

RN	Lambda	A	% en Act	% en isot.	% en masse
U238	49,14E-19	2,38E+02	46,17E+00	99,27E+00	99,28E+00
U235	31,20E-18	2,35E+02	21,50E-01	72,81E-02	71,90E-02
U234	89,47E-15	2,34E+02	51,68E+00	61,02E-04	60,00E-04

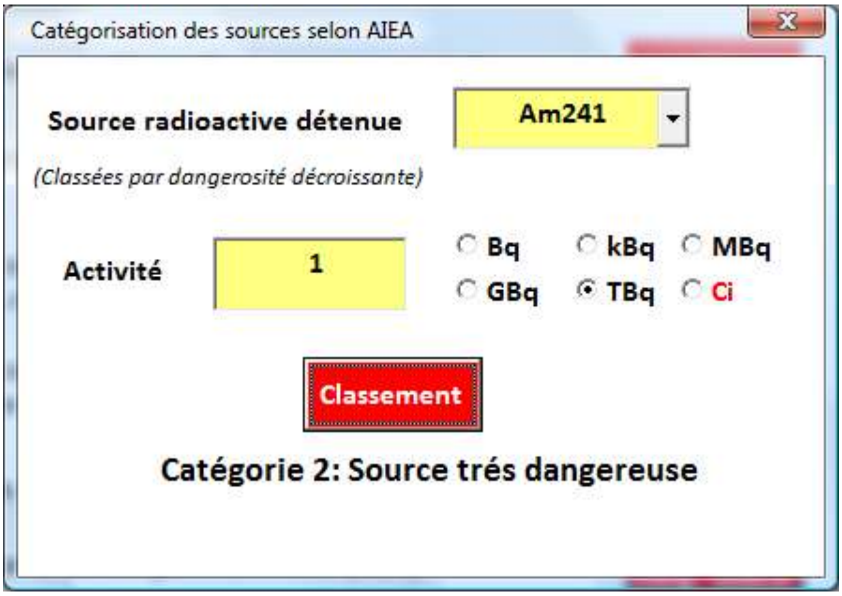
II.6.10 MULTI-SCREEN OPTION

Multi-ecran

This option allows you to define a multi-screen prior to a gamma dose calculation. Its principle of use is identical to that described above (§ I.2.E)

II.6.11 "HAZARDOUS SOURCES CATEGORIZATION" OPTION

This option defines the danger class of a source according to the IAEA criteria (IAEA / Safety guide / RS-C-1.9*http://www-pub.iaea.org/MTCD/publications/PDF/Pub1227_web.pdf*)

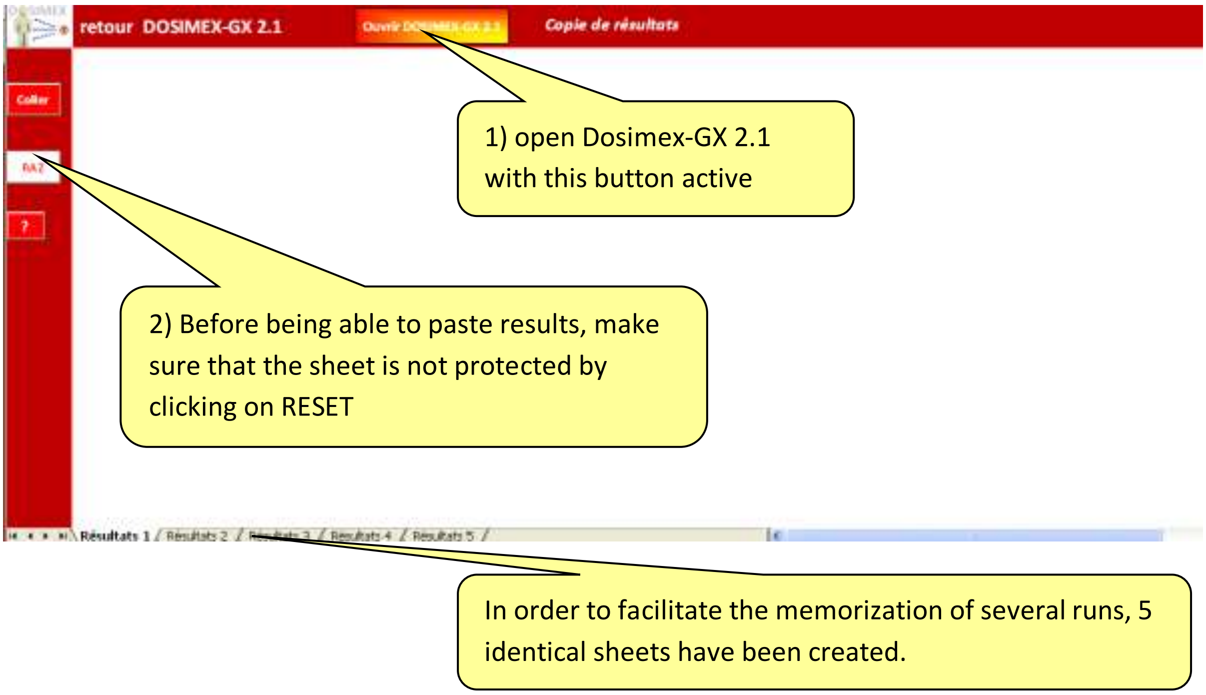


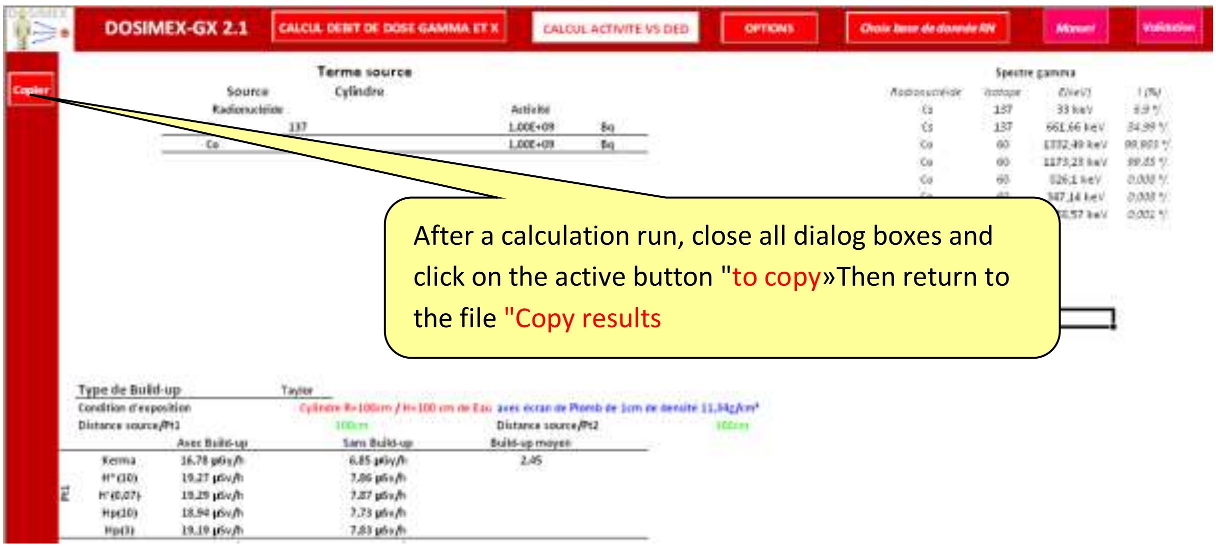
II.7 Copy of results to a dedicated file

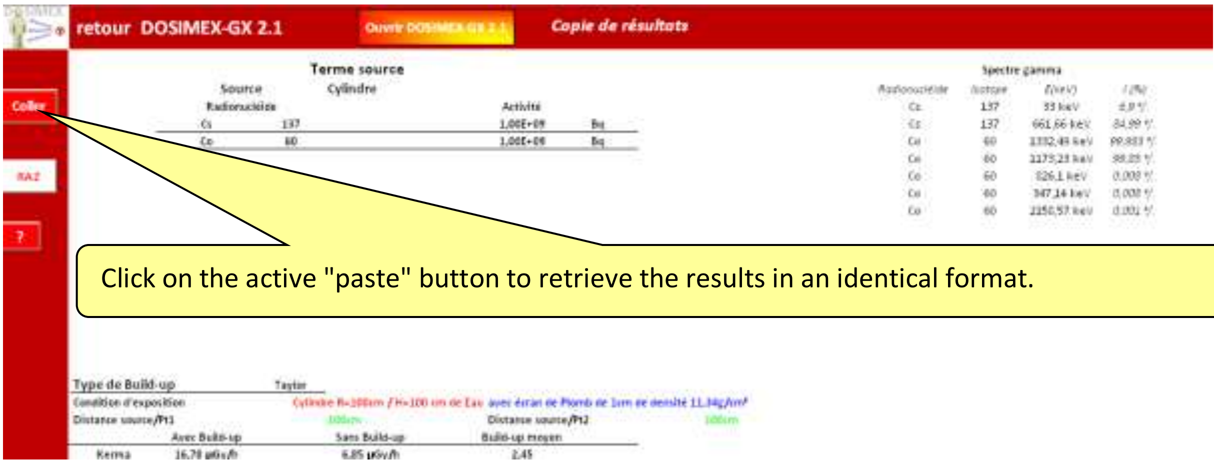
In order to facilitate the recording of results, a specific file has been created: "Copy results.xls"
This unencrypted file must imperatively be opened before any encrypted file of the Dosimex pack to be able to be saved. Otherwise, it will be opened in “read only” mode, which cannot be saved.

A specific link then opens Dosimex-GX 2.1 directly. For this the fileCopy results.xls original must remain in the “Dosimex-GX” folder, but it can be saved under another name in the directory of your choice after acquisition of results.

II.7.1 CASE OF RESULTS WITHOUT GRAPH

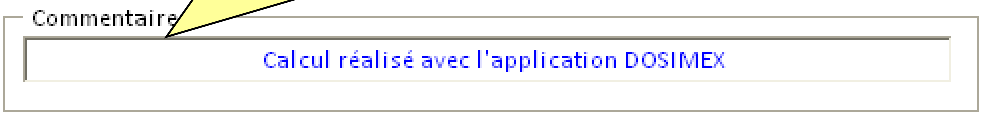






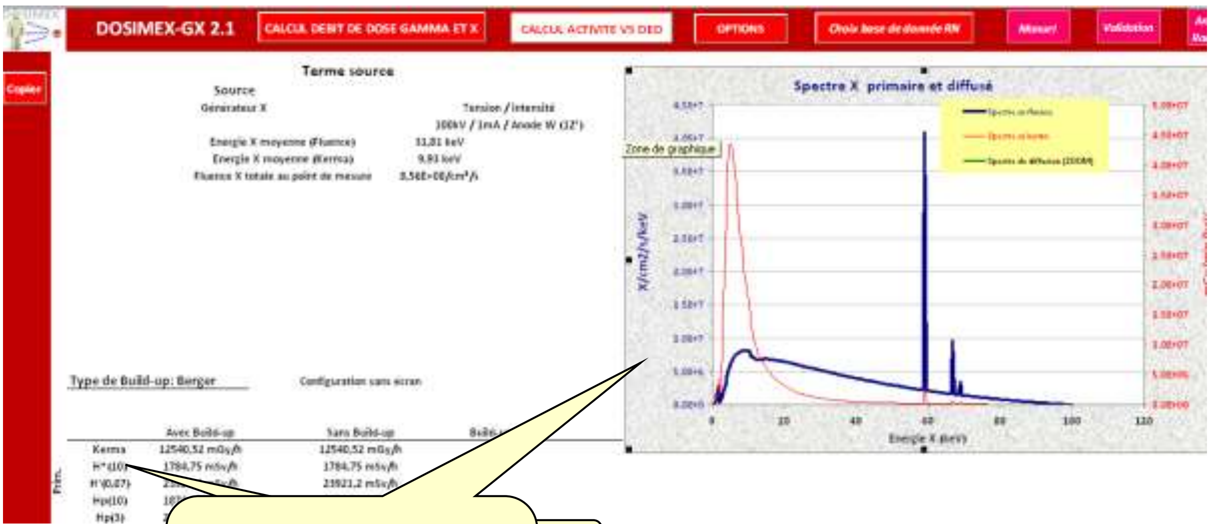
Click on the active "paste" button to retrieve the results in an identical format.

The sheet is then write-protected. Remember to introduce comments if necessary in the Dosimex – GX dialog before launching the calculation



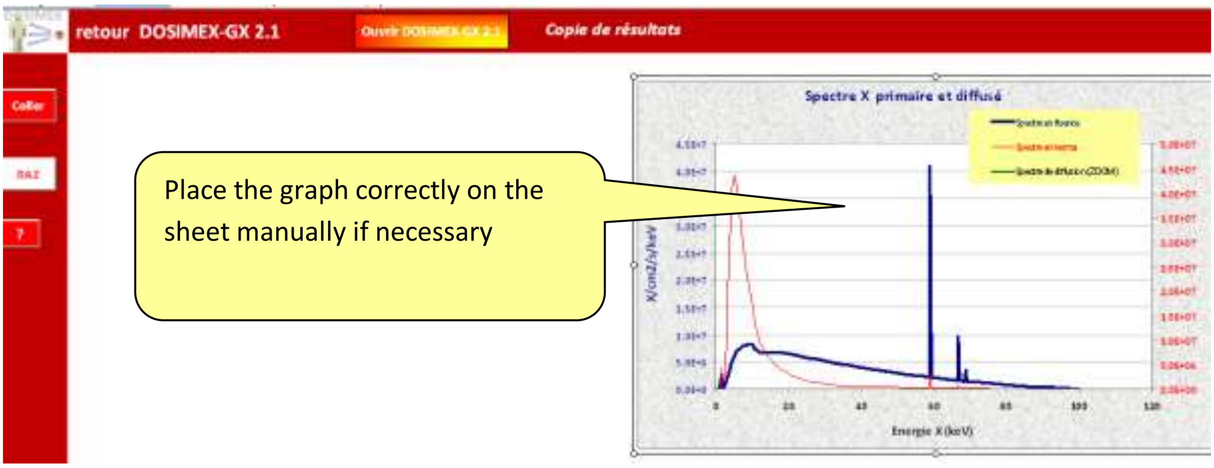
II.7.2 CASE OF RESULTS WITH GRAPH

The “copy” function cannot directly transfer a graph from an encrypted file to an unencrypted file. To save a graph (Build-up options and X generator modeling) it is necessary to manually transfer them according to the following procedure:



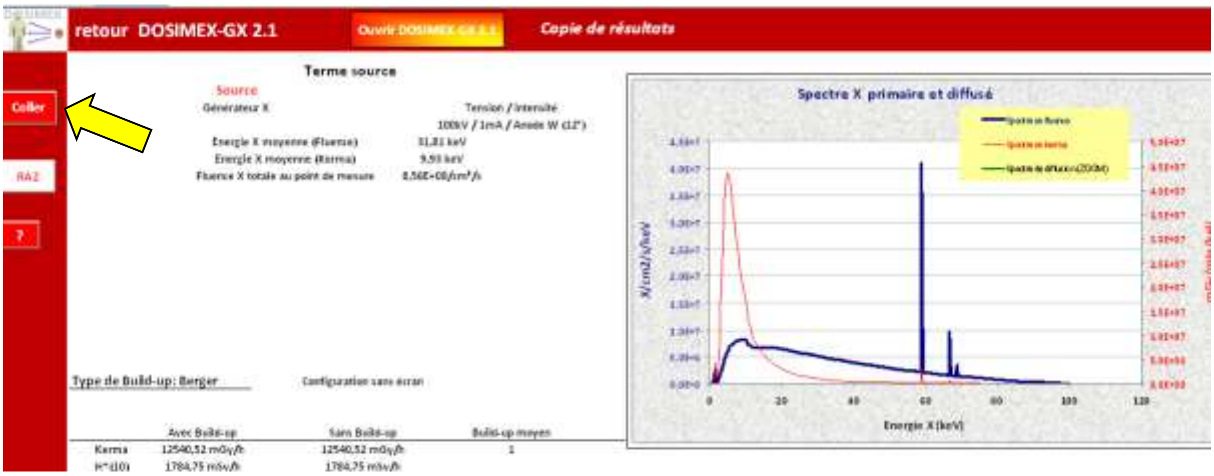
2) manually copy the graph (Ctrl + C)

Go back to the "Copy results" file then copy the graph manually: Ctrl + V on an empty sheet (RESET function to perform if necessary before the previous manipulation)



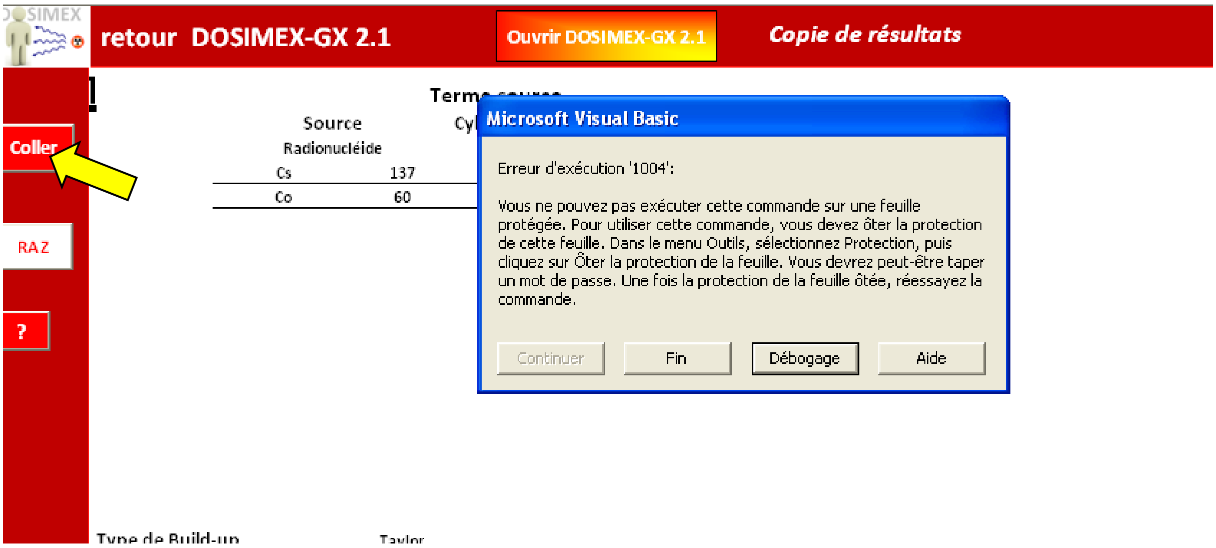
Attention: the copy of the graph in this configuration is longer than in normal use between unencrypted files

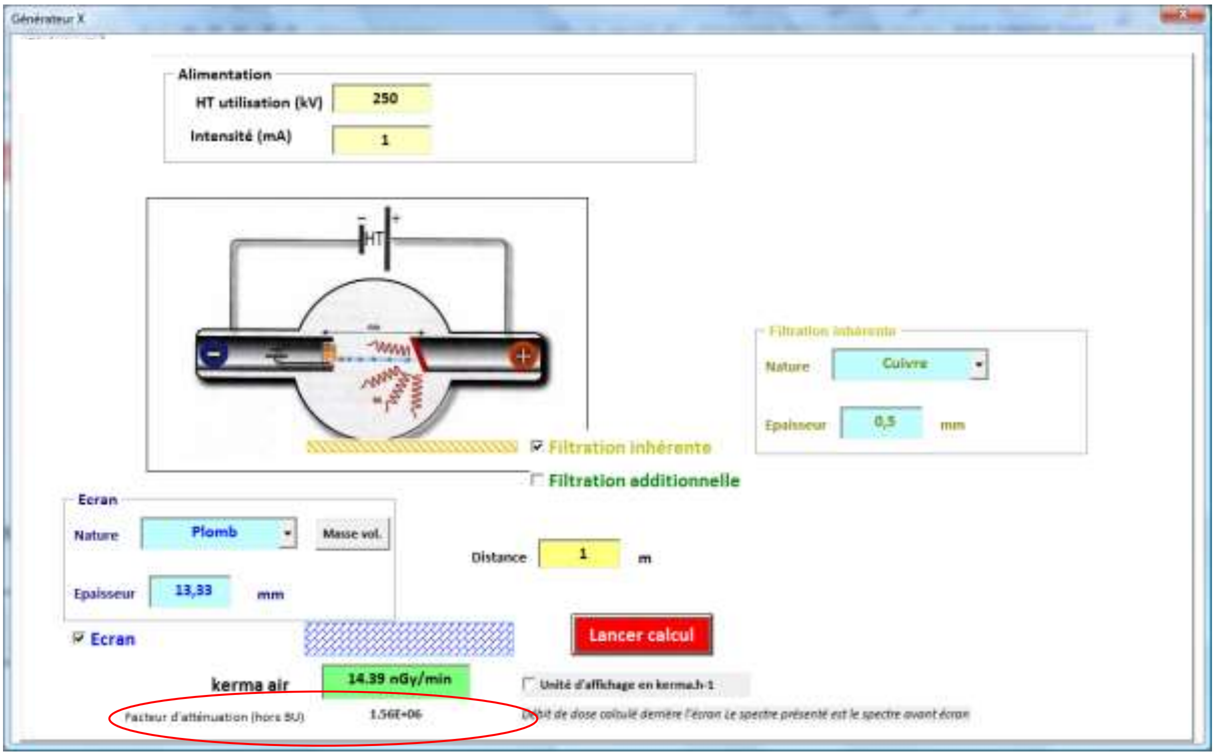
Then go back to Dosimex-GX (link available on the file) then make a copy-paste as in the previous case (copy without graph). The text is then superimposed on the graph already present.



Tip: keep the original version of the “Copy results” file and save the results under another name and possibly in another folder. This will allow you to keep blank sheets allowing you to immediately paste results

If you try to copy results to an already filled sheet, the Excel file will be put in error:




Générateur X
Alimentation
HT utilisation (kV)
250
Intensité (mA)
1
HT
Filtration inhérente
Nature
Cuivre
Epaisseur
0,5
mm
Filtration inhérente
Filtration additionnelle
Ecran
Nature
Plomb
Masse vol.
Epaisseur
13,33
mm
Distance
1
m
Ecran
Lancer calcul
kerma air
14.39 nGy/min
Unité d'affichage en kerma.h-1
Facteur d'atténuation (hors BU)
1.56E+06
Débit de dose calculé derrière l'écran Le spectre présenté est le spectre avant écran

Partie III. SPECIFIC GAMMA SOURCES

III.1 UOX AND MOX SOURCES

III.1.1 SOURCE DATA

These data were provided by Gilles BAROUCH, of the CEA / Cadarache Radiological Protection Service

GOALS : Evaluate the dose equivalent flow rates generated by UOX and MOX type fuels after irradiation and cooling.

INITIAL DATA: Dthey types of REP fuels have been studied, a UOX and a MOX, for which the following initial isotopic spectra are considered:

Typical UOX mass ratios		
U234%	U 235%	U 238%
0.05%	4.95%	95.00%

MOX type mass proportions					
Pu238	Pu239	Pu240	Pu241	Pu242	Am241
1.62%	58.16%	22.43%	11.10%	5.59%	1.10%

III.1.2 CALCULATION METHOD

✓ **Phase 1** : calculation of neutron irradiation to 99 groups performed with the code APOLLO1

In both cases, irradiation campaigns of 10 GWd / t followed by a 90-day shutdown were simulated until:

- 120 GWd / ton for UOX
- 90 GWd / ton for MOX

✓ **Phase 2** : cooling calculation at 6 months, 1 year, 3 years then finally 5 years using the CESAR-4 p evolution code

III.1.3 UOX FUEL RESULTS

UOX - Total activity of fission products (TBq / tonne) (1/2)				
ISOTOPE	**6 month refr**	**1 year refr**	**3 year refr**	**5 years refr**
ZN 72	2.95E-29	0.00E + 00	0.00E + 00	0.00E + 00
GA 72	4.24E-29	0.00E + 00	0.00E + 00	0.00E + 00
AS 77	6.41E-33	0.00E + 00	0.00E + 00	0.00E + 00
SE 79	2.22E-03	2.22E-03	2.22E-03	2.22E-03
BR 82	1.77E-35	0.00E + 00	0.00E + 00	0.00E + 00
KR 85	7.29E + 02	7.06E + 02	6.20E + 02	5.45 + 02
RB 86	2.02E-01	2.34E-04	3.77E-16	6.31E-28
RB 87	2.15E-06	2.15E-06	2.15E-06	2.15E-06
SR 89	1.52E + 03	1.25E + 02	5.50E-03	2.455E-07
SR 90	6.48E + 03	6.41E + 03	6.11E + 03	5.82E + 03
Y 90	6.48E + 03	6.41E + 03	6.11E + 03	5.83E + 03
Y 91	3.11E + 03	3.61E + 02	6.25E-02	1.10E-05
ZR 93	1.95-01	1.95-01	1.95-01	1.95-01
ZR 95	6.84E + 03	9.52E + 02	3.46E-01	1.27E-04
NB 93M	4.74E-02	5.05E-02	6.22E-02	7.29E-02
NB 95	1.35E + 04	2.07E + 03	7.69E-01	2.83E-04
MO 99	6.53E-16	7.91E-36	0.00E + 00	0.00E + 00
TC 99	1.45E + 00	1.45E + 00	1.45E + 00	1.45E + 00
TC 99M	6.25E-16	7.58E-36	0.00E + 00	0.00E + 00
RU103	3.08E + 03	1.25E + 02	3.19E-04	8.31E-10
RU106	2.74E + 04	1.95 + 04	4.98E + 03	1.28E + 03
RH103M	3.07E + 03	1.25E + 02	3.19E-04	8.30E-10
RH105	2.50E-33	0.00E + 00	0.00E + 00	0.00E + 00
RH106	2.74E + 04	1.95 + 04	4.98E + 03	1.28E + 03
PD107	2.15E-02	2.15E-02	2.15E-02	2.15E-02
AG108M	1.21E-06	1.21E-06	1.20E-06	1.20E-06
AG110M	6.52E + 03	3.94E + 03	5.18E + 02	6.84E + 01
AG111	3.59E-04	1.59E-11	0.00E + 00	0.00E + 00
CD113	1.67E-15	1.67E-15	1.67E-15	1.67E-15
CD113M	7.51E + 00	7.33E + 00	6.62E + 00	5.97E + 00
CD115	7.55E-23	0.00E + 00	0.00E + 00	0.00E + 00
CD115M	6.08E + 00	3.59E-01	4.18E-06	4.95E-11
IN115	5.31E-13	5.32E-13	5.32E-13	5.32E-13
SN121	1.01E-01	1.01E-01	9.81E-02	9.57E-02
SN121M	1.31E-01	1.30E-01	1.26E-01	1.23E-01
SN123	2.15E + 01	8.09E + 00	1.60E-01	3.19E-03
SN125	7.01E-04	1.45E-09	2.17E-32	0.00E + 00
SN126	9.96E-02	9.96E-02	9.96E-02	9.96E-02
SB122	6.44E-19	0.00E + 00	0.00E + 00	0.00E + 00
SB124	1.65E + 01	2.03E + 00	4.50E-04	1.01E-07
SB125	5.64E + 02	4.98E + 02	3.02E + 02	1.83E + 02
SB126	1.01E-01	9.96E-02	9.96E-02	9.96E-02

UOX - Total activity of fission products (TBq / tonne) (1/2)				
ISOTOPE	6 month refr	1 year refr	3 year refr	5 years refr
SB127	2.20E-11	1.29E-25	0.00E + 00	0.00E + 00
TE123	5.48E-13	6.35E-13	6.81E-13	6.82E-13
TE123M	4.08E + 00	1.42E + 00	2.06E-02	3.01E-04
TE125M	1.29E + 02	1.15E + 02	6.98E + 01	4.23E + 01
TE127	1.77E + 02	5.57E + 01	5.33E-01	5.14E-03
TE127M	1.81E + 02	5.68E + 01	5.44E-01	5.24E-03
TE129M	5.18E + 01	1.21E + 00	3.42E-07	9.87E-14
TE132	6.78E-13	1.04E-29	0.00E + 00	0.00E + 00
I 129	3,95E-03	3,95E-03	3,95E-03	3,95E-03
I 131	5.73E-03	8.78E-10	3,75E-37	0.00E + 00
XE131M	3.34E-02	9.04E-07	4.01E-25	0.00E + 00
XE133	2.92E-06	1.05E-16	0.00E + 00	0.00E + 00
XE133M	2.55E-22	0.00E + 00	0.00E + 00	0.00E + 00
CS134	2.51E + 04	2.12E + 04	1.08E + 04	5.52E + 03
CS135	5.98E-02	5.98E-02	5.98E-02	5.98E-02
CS136	5.51E-01	3.78E-05	7.19E-22	1.44E-38
CS137	1.27E + 04	1.25E + 04	1.20E + 04	1.14E + 04
BA137M	1.20E + 04	1.19E + 04	1.13E + 04	1.08E + 04
BA140	2.88E + 00	1.45E-04	8.01E-22	0.00E + 00
LA140	3.31E + 00	1.67E-04	9.22E-22	0.00E + 00
CE141	1.14E + 03	2.35 + 01	3.99E-06	6.90E-13
CE142	7.02E-12	7.02E-12	7.02E-12	7.02E-12
CE143	4.31E-36	0.00E + 00	0.00E + 00	0.00E + 00
CE144	2.30E + 04	1.48E + 04	2.50E + 03	4.23E + 02
PR143	4.81E + 00	4.44E-04	2.78E-20	1.83E-36
PR144	2.30E + 04	1.48E + 04	2.50E + 03	4.23E + 02
ND144	2.38E-10	2.41E-10	2.46E-10	2.47E-10
ND147	2.34E-01	2.42E-06	2.30E-26	0.00E + 00
PM147	5.31E + 03	4.66E + 03	2.75 + 03	1.62E + 03
PM148	3.85E + 00	1.82E-01	8.53E-07	4.08E-12
PM148M	7.29E + 01	3.43E + 00	1.61E-05	7.71E-11
PM149	3.16E-21	0.00E + 00	0.00E + 00	0.00E + 00
SM147	1.93E-07	2.09E-07	2.56E-07	2.84E-07
SM148	8.80E-12	8.80E-12	8.80E-12	8.80E-12
SM149	9.06E-13	9.06E-13	9.06E-13	9.06E-13
SM151	2.53E + 01	2.52E + 01	2.48E + 01	2.44E + 01
SM153	1.95-24	0.00E + 00	0.00E + 00	0.00E + 00
EU152	1.07E + 00	1.05E + 00	9.42E-01	8.49E-01
EU154	1.10E + 03	1.06E + 03	9.02E + 02	7.70E + 02
EU155	5.32E + 02	4.96E + 02	3.75 + 02	2.84E + 02
EU156	8.61E + 00	2.13E-03	6.95E-18	2.37E-32
TB160	2.47E + 01	4.32E + 00	3.91E-03	3.57E-06
TB161	9.82E-07	1.16E-14	0.00E + 00	0.00E + 00
HO166M	4.62E-03	4.62E-03	4.62E-03	4.61E-03
Total	2.11E + 05	1.42E + 05	6.69E + 04	4.64E + 04

UOX - Gamma source of fission products (gamma / sec / ton)				
GROUP (keV)	**Cooled 6 months**	**Cooled 1 year**	**Cooled 3 years**	**Cooled 5 years**
10	4.78E + 14	3.06E + 14	5.80E + 13	1.32E + 13
18	2.74E + 11	2.56E + 11	1.94E + 11	1.46E + 11
25	3.39E + 14	4.94E + 13	6.65 + 12	2.26E + 12
38	2.65 + 15	1.87E + 15	6.22E + 14	3.23E + 14
53	4.48E + 13	2.94E + 13	1.25E + 13	8.08E + 12
68	6.89E + 10	4.73E + 06	8.99E-11	1.80E-27
88	4.92E + 14	3.62E + 14	1.51E + 14	9.40E + 13
108	1.10E + 14	1.03E + 14	7.75 + 13	5.86E + 13
135	3.56E + 15	2.08E + 15	6.43E + 14	3.59E + 14
175	3.80E + 13	3.34E + 13	2.02E + 13	1.23E + 13
250	9.39E + 13	7.58E + 13	6.18E + 13	5.20E + 13
350	7.15E + 12	4.76E + 12	2.12E + 12	1.09E + 12
425	4.97E + 13	2.21E + 13	5.23E + 12	1.40E + 12
480	3.36E + 15	6.28E + 14	2.82E + 14	1.56E + 14
555	1.17E + 16	9.04E + 15	3.60E + 15	1.58E + 15
650	3.88E + 16	3.38E + 16	2.14E + 16	1.53E + 16
750	4.20E + 16	2.14E + 16	9.43E + 15	4.88E + 15
900	2.77E + 15	2.35 + 15	1.32E + 15	7.92E + 14
1165	1.66E + 15	1.35E + 15	7.20E + 14	4.53E + 14
1415	8.33E + 14	6.89E + 14	3.36E + 14	1.69E + 14
1580	4.50E + 13	2.98E + 13	7.61E + 12	1.95 + 12
1830	3.99E + 13	2.37E + 13	5.80E + 12	1.48E + 12
2250	1.90E + 14	1.23E + 14	2.25E + 13	4.25E + 12
2750	4.13E + 12	2.86E + 12	7.26E + 11	1.85E + 11
3250	4.25E + 11	3.02E + 11	7.73E + 10	1.98E + 10
TOTAL	1.09E + 17	7.44E + 16	3.88E + 16	2.43E + 16

III.1.4 MOX FUEL RESULTS

MOX - Total activity of fission products (TBq / tonne) (1/2)				
ISOTOPES	**Cooled 6 months**	**Cooled 1 year**	**Cooled 3 years**	**Cooled 5 years**
ZN 72	2.81E-29	0.00E + 00	0.00E + 00	0.00E + 00
GA 72	4.03E-29	0.00E + 00	0.00E + 00	0.00E + 00
AS 77	6.00E-33	0.00E + 00	0.00E + 00	0.00E + 00
SE 79	2.77E-02	2.77E-02	2.77E-02	2.77E-02
BR 82	8.90E-36	0.00E + 00	0.00E + 00	0.00E + 00
KR 85	4.25E + 02	4.11E + 02	3.61E + 02	3.18E + 02
RB 86	6.83E-02	7.91E-05	1.28E-16	2.13E-28
RB 87	1.03E-06	1.03E-06	1.03E-06	1.03E-06
SR 89	1.44E + 03	1.19E + 02	5.21E-03	2.32E-07
SR 90	3.17E + 03	3.14E + 03	2.99E + 03	2.85 + 03
Y 90	3.17E + 03	3.14E + 03	2.99E + 03	2.85 + 03
Y 91	2.97E + 03	3.44E + 02	5.97E-02	1.05E-05
ZR 93	1.18E-01	1.18E-01	1.18E-01	1.18E-01
ZR 95	6.70E + 03	9.33E + 02	3.39E-01	1.25E-04
NB 95	1.32E + 04	2.03E + 03	7.53E-01	2.77E-04
MO 99	6.44E-16	7.80E-36	0.00E + 00	0.00E + 00
TC 99	1.28E + 00	1.28E + 00	1.28E + 00	1.28E + 00
TC 99M	6.16E-16	7.47E-36	0.00E + 00	0.00E + 00
RU103	3.02E + 03	1.22E + 02	3.13E-04	8.15E-10
RU106	2.94E + 04	2.09E + 04	5.35E + 03	1.37E + 03
RH103M	3.02E + 03	1.22E + 02	3.13E-04	8.14E-10
RH105	2.72E-33	0.00E + 00	0.00E + 00	0.00E + 00
RH106	2.94E + 04	2.09E + 04	5.35E + 03	1.37E + 03
PD107	2.57E-02	2.57E-02	2.57E-02	2.57E-02
AG110M	7.14E + 03	4.31E + 03	5.67E + 02	7.49E + 01
AG111	2,85E-04	1.26E-11	0.00E + 00	0.00E + 00
CD113	1.33E-14	1.33E-14	1.33E-14	1.33E-14
CD113M	6.58E + 00	6.41E + 00	5.79E + 00	5.23E + 00
CD115	7.51E-23	0.00E + 00	0.00E + 00	0.00E + 00
CD115M	5.54E + 00	3.27E-01	3.81E-06	4.51E-11
IN115	1.10E-12	1.10E-12	1.10E-12	1.10E-12
SN121	2.09E + 00	2.08E + 00	2.03E + 00	1.98E + 00
SN121M	2.70E + 00	2.68E + 00	2.61E + 00	2.55E + 00
SN123	2.06E + 01	7.75 + 00	1.54E-01	3.06E-03
SN125	6.76E-04	1.40E-09	2.09E-32	0.00E + 00
SN126	9.56E-02	9.56E-02	9.56E-02	9.56E-02
SB122	3.73E-19	0.00E + 00	0.00E + 00	0.00E + 00
SB124	1.03E + 01	1.27E + 00	2.80E-04	6.27E-08
SB125	5.71E + 02	5.04E + 02	3.05E + 02	1.85E + 02
SB126	9.66E-02	9.56E-02	9.56E-02	9.56E-02
SB127	2.13E-11	1.26E-25	0.00E + 00	0.00E + 00
TE123	2.59E-13	2.88E-13	3.03E-13	3.03E-13

MOX - Total activity of fission products (TBq / tonne) (2/2)				
ISOTOPES	Cooled 6 months	Cooled 1 year	Cooled 3 years	Cooled 5 years
TE123M	1.35E + 00	4.72E-01	6.84E-03	9.99E-05
TE125M	1.31E + 02	1.16E + 02	7.06E + 01	4.28E + 01
TE127	1.74E + 02	5.48E + 01	5.24E-01	5.05E-03
TE127M	1.78E + 02	5.59E + 01	5.35E-01	5.16E-03
TE129M	5.05E + 01	1.18E + 00	3.34E-07	9.63E-14
TE132	6.73E-13	1.03E-29	0.00E + 00	0.00E + 00
I 129	3.64E-03	3.64E-03	3.64E-03	3.64E-03
I 131	5.66E-03	8.67E-10	3.71E-37	0.00E + 00
XE131M	3.33E-02	9.00E-07	3.99E-25	0.00E + 00
XE133	2.90E-06	1.04E-16	0.00E + 00	0.00E + 00
XE133M	2.54E-22	0.00E + 00	0.00E + 00	0.00E + 00
CS134	1.39E + 04	1.17E + 04	5.99E + 03	3.06E + 03
CS135	1.12E-01	1.12E-01	1.12E-01	1.12E-01
CS136	6.17E-01	4.24E-05	8.06E-22	1.61E-38
CS137	9.87E + 03	9.76E + 03	9.32E + 03	8.90E + 03
BA137M	9.35 + 03	9.24E + 03	8.83E + 03	8.43E + 03
BA140	2.89E + 00	1.46E-04	8.03E-22	0.00E + 00
LA140	3.33E + 00	1.68E-04	9.25E-22	0.00E + 00
CE141	1.13E + 03	2.33E + 01	3.955E-06	6.85E-13
CE142	4.87E-12	4.87E-12	4.87E-12	4.87E-12
CE143	4.26E-36	0.00E + 00	0.00E + 00	0.00E + 00
CE144	2.29E + 04	1.47E + 04	2.48E + 03	4.20E + 02
PR143	4.79E + 00	4.43E-04	2.77E-20	1.82E-36
PR144	2.29E + 04	1.47E + 04	2.48E + 03	4.20E + 02
ND144	9.81E-11	1.01E-10	1.06E-10	1.06E-10
ND147	2.29E-01	2.37E-06	2.25E-26	0.00E + 00
PM147	7.33E + 03	6.43E + 03	3.79E + 03	2.23E + 03
PM148	7.53E + 00	3.55E-01	1.67E-06	7.97E-12
PM148M	1.42E + 02	6.72E + 00	3.16E-05	1.51E-10
PM149	2.47E-21	0.00E + 00	0.00E + 00	0.00E + 00
SM147	2,65E-07	2.87E-07	3.52E-07	3.91E-07
SM148	5.81E-12	5.81E-12	5.81E-12	5.81E-12
SM149	3.40E-12	3.40E-12	3.40E-12	3.40E-12
SM151	1.06E + 02	1.06E + 02	1.04E + 02	1.03E + 02
SM153	1.24E-24	0.00E + 00	0.00E + 00	0.00E + 00
EU152	9.15E + 00	8.92E + 00	8.03E + 00	7.24E + 00
EU154	1.56E + 03	1.50E + 03	1.28E + 03	1.09E + 03
EU155	1.77E + 03	1.65E + 03	1.25E + 03	9.46E + 02
EU156	1.94E + 00	4.79E-04	1.56E-18	5.33E-33
TB160	2.41E + 01	4.20E + 00	3.80E-03	3.47E-06
TB161	7.83E-07	9.22E-15	0.00E + 00	0.00E + 00
TOTAL	**1.95 + 05**	**1.27E + 05**	**5.35E + 04**	**3.47E + 04**

MOX - Gamma source of fission products (gamma / sec / ton)				
GROUP (keV)	Cooled 6 months	Cooled 1 year	Cooled 3 years	Cooled 5 years
10	4,558E + 11	3.246E + 11	8.302E + 10	2.127E + 10
18	4.420E + 12	3.061E + 12	7.790E + 11	1,989E + 11
25	1.904E + 14	1,240E + 14	2,276E + 13	4.320E + 12
38	3.921E + 13	2,499E + 13	6.225E + 12	1.593E + 12
53	4.805E + 13	3.196E + 13	8.171E + 12	2,093E + 12
68	5.257E + 14	4.288E + 14	2.034E + 14	1.017E + 14
88	1.665E + 15	1.368E + 15	7.980E + 14	5.574E + 14
108	2.163E + 15	1.858E + 15	1.154E + 15	7.814E + 14
135	3,293E + 16	1,400E + 16	5.751E + 15	3.035E + 15
175	2.665E + 16	2,315E + 16	1.489E + 16	1.099E + 16
250	9.701E + 15	7.276E + 15	2,618E + 15	1.059E + 15
350	3.164E + 15	4.984E + 14	2.174E + 14	1.227E + 14
425	6.868E + 13	2,435E + 13	5.602E + 12	1.494E + 12
480	7.922E + 12	5.631E + 12	3.311E + 12	2,392E + 12
555	1,440E + 14	1.164E + 14	9.679E + 13	8.223E + 13
650	3.841E + 13	3.377E + 13	2,047E + 13	1.241E + 13
750	3.803E + 15	2,331E + 15	8.637E + 14	5.492E + 14
900	2,064E + 14	1,925E + 14	1.454E + 14	1,100E + 14
1165	6.343E + 14	4.958E + 14	2,536E + 14	1.714E + 14
1415	7.725E + 10	5.306E + 06	1.008E-10	2,020E-27
1580	6.149E + 13	4,536E + 13	2,543E + 13	1.856E + 13
1830	2,684E + 15	1,919E + 15	6.969E + 14	3.998E + 14
2250	3.367E + 14	5.187E + 13	8.085E + 12	3,160E + 12
2750	5.158E + 11	4.811E + 11	3,637E + 11	2,751E + 11
3250	4.652E + 14	2.967E + 14	5.375E + 13	1.119E + 13
TOTAL	**8.553E + 16**	**5.428E + 16**	**2,784E + 16**	**1.801E + 16**

III.2 SPECTRUM TYPE FISSION PRODUCTS S060

Isotopic spectrum: (ref APM DDCO NTDMED 03058)

RN	AT	% activity
Am	241	8,65E-03
This	144	1.43E-118
Pr	144	1.43E-118
Co	60	2.33E-20
Cs	134	2,85E-47
Cs	137	1.36E-04
Had	154	8.80E-14
I	129	5.18E-07
Mn	54	3.96E-110
Mo	93	5.17E-07
Nb	94	9.57E-10
Could	238	2.89E-03
Could	239	3.76E-02
Could	240	2.41E-02
Could	241	2.81E-05
Could	242	1.46E-05
Ru	106	6.44E-01
Sb	125	1.62E-02
Sn	121-m	3.52E-05
Tc	99	1.81E-05
U	234	7.09E-05
U	235	3.37E-06
U	238	6,75E-05
Y	90	7.31E-02
Zr	93	2.78E-06

Full emission table:

RN	AT	E	Intrinsic intensity	% activity	intensity mix
Am	241	60 keV	3.60E + 01	1.40E-02	5,043E-01
Am	241	14 keV	3.31E + 01	1.40E-02	4.630E-01
Am	241	27 keV	3.29E + 00	1.40E-02	4,599E-02
Am	241	104 keV	5.41E-02	1.40E-02	7.575E-04
Am	241	299 keV	2.33E-03	1.40E-02	3,264E-05
Am	241	683 keV	7.61E-04	1.40E-02	1.066E-05
This	144	90 keV	2.12E + 01	8.64E-03	1,830E-01
This	144	696 keV	1.34E + 00	8.64E-03	1,161E-02
Pr	144	696 keV	1.34E + 00	8.64E-03	1,161E-02
Pr	144	836 keV	5.90E-03	8.64E-03	5,098E-05
Cs	134	596 keV	1.23E + 02	1.21E-03	1.487E-01
Cs	134	796 keV	9.42E + 01	1.21E-03	1,140E-01
Cs	134	33 keV	8.30E-01	1.21E-03	1.004E-03
Cs	134	276 keV	3.50E-02	1.21E-03	4.235E-05
Cs	137	662 keV	8.50E + 01	1.38E-01	1.173E + 01
Had	154	93 keV	6.61E + 01	2.87E-03	1.897E-01
Had	154	731 keV	2.54E + 01	2.87E-03	7,285E-02
Had	154	614 keV	8.78E + 00	2.87E-03	2,521E-02
Had	154	250 keV	7.12E + 00	2.87E-03	2,044E-02
Had	154	452 keV	1.26E + 00	2.87E-03	3,613E-03
I	129	36 keV	2.06E + 01	5.18E-07	1.069E-05
Mn	54	835 keV	1.00E + 02	6.98E-04	6.978E-02
Mo	93	17 keV	6.33E + 01	5.49E-07	3.475E-05
Mo	93	31 keV	5.00E-04	5.49E-07	2,745E-10
Nb	94	703 keV	9.79E + 01	9.67E-10	9.467E-08
Could	238	16 keV	1.06E + 01	3.15E-02	3.348E-01
Could	238	44 keV	3.97E-02	3.15E-02	1,251E-03
Could	238	100 keV	7.45E-03	3.15E-02	2,347E-04
Could	238	153 keV	9.30E-04	3.15E-02	2.930E-05
Could	238	111 keV	6.17E-05	3.15E-02	1,944E-06
Could	238	200 keV	3.92E-05	3.15E-02	1,235E-06
Could	238	742 keV	5.10E-06	3.15E-02	1.607E-07

Could	239	12 keV	4.70E + 00	3.79E-02	1.781E-01
Could	239	52 keV	2.81E-02	3.79E-02	1.064E-03
Could	239	109 keV	2.22E-02	3.79E-02	8.395E-04
Could	239	39 keV	1.00E-02	3.79E-02	3,790E-04
Could	239	394 keV	3.00E-03	3.79E-02	1,139E-04
Could	240	45 keV	4.47E-02	2.49E-02	1,113E-03
Could	240	104 keV	7.23E-03	2.49E-02	1,800E-04
Could	241	107 keV	1.27E-03	2.81E-05	3.565E-08
Could	242	17 keV	8.70E + 00	1.46E-05	1,270E-04
Could	242	45 keV	3,75E-02	1.46E-05	5.475E-07
Could	242	104 keV	2.40E-03	1.46E-05	3.504E-08
Ru	106	511 keV	2.06E + 01	6.44E-01	1.328E + 01
Ru	106	622 keV	1.07E + 01	6.44E-01	6.905E + 00
U	235	13	2.32E + 01	3.37E-06	7.803E-05
U	234	15.75 keV	1.04E + 01	7.09E-05	7.338E-04
U	238	15.75 keV	7.50E + 00	6,75E-05	5,063E-04
Zr	93	16.936 keV	1.29E + 00	2.78E-06	3,582E-06
Sn	121-m	28 keV	1.72E + 01	3.52E-05	6.068E-04
Zr	93	3.08E + 01	5.57E-04	2.78E-06	1,548E-09
Sb	125	34 keV	1.96E + 01	1.62E-02	3,179E-01
Sn	121-m	36 keV	2.34E + 00	3.52E-05	8,237E-05
U	235	38.488991	1.06E-01	3.37E-06	3,566E-07
U	238	49.55 keV	6.20E-02	6,75E-05	4,185E-06
U	234	53.2 keV	1.23E-01	7.09E-05	8.721E-06
U	235	67.98993	2.44E-01	3.37E-06	8,236E-07
U	234	92.064 keV	6.68E-03	7.09E-05	4.736E-07
U	235	96.30316	1.33E + 01	3.37E-06	4.483E-05
U	238	110.2 keV	1.25E-02	6,75E-05	8,410E-07
U	234	120.9 keV	3.42E-02	7.09E-05	2,425E-06
U	235	149.85356	1.63E + 01	3.37E-06	5,498E-05
Sb	125	179 keV	7.81E + 00	1.62E-02	1,266E-01
U	235	185.8021	5.82E + 01	3.37E-06	1,960E-04
U	235	205.2988	6.28E + 00	3.37E-06	2.116E-05
U	235	251.63438	1.80E-01	3.37E-06	6.061E-07
U	235	290.58938	6.40E-02	3.37E-06	2.155E-07
U	235	367.38661	9.42E-02	3.37E-06	3.1755E-07

Sb	125	434 keV	4.22E + 01	1.62E-02	6.837E-01
Y	90	511	6.38E-03	7.31E-02	4.664E-04
Sb	125	616 keV	3.59E + 01	1.62E-02	5.817E-01
Ru	106	751 keV	1.00E-03	6.44E-01	6.440E-04
Nb	94	871 keV	9.99E + 01	9.67E-10	9.660E-08
Ru	106	874 keV	4.31E-01	6.44E-01	2,773E-01
Had	154	875 keV	1.46E + 01	2.87E-03	4,195E-02
Had	154	1002 keV	2.88E + 01	2.87E-03	8,278E-02
Cs	134	1039 keV	9.91E-01	1.21E-03	1.199E-03
Ru	106	1067 keV	1.90E + 00	6.44E-01	1.226E + 00
Cs	134	1168 keV	1.79E + 00	1.21E-03	2.168E-03
Co	60	1173 keV	9.99E + 01	3.19E-03	3.1855E-01
Ru	106	1194 keV	7.02E-02	6.44E-01	4,521E-02
Had	154	1271 keV	3.68E + 01	2.87E-03	1.055E-01
Co	60	1332 keV	1.00E + 02	3.19E-03	3,189E-01
Ru	106	1349 keV	1.19E-02	6.44E-01	7.670E-03
Cs	134	1365 keV	3.02E + 00	1.21E-03	3,648E-03
Pr	144	1387 keV	6.79E-03	8.64E-03	5.867E-05
This	144	1388 keV	6.40E-03	8.64E-03	5,530E-05
This	144	1489 keV	2.79E-01	8.64E-03	2,411E-03
Pr	144	1489 keV	2.79E-01	8.64E-03	2,412E-03
Had	154	1491 keV	8.32E-01	2.87E-03	2,387E-03
Ru	106	1553 keV	1.78E-01	6.44E-01	1,149E-01
Had	154	1597 keV	1.82E + 00	2.87E-03	5.233E-03
Ru	106	1777 keV	5.86E-02	6.44E-01	3,771E-02
Ru	106	1982 keV	2.79E-02	6.44E-01	1,800E-02
Pr	144	2006 keV	1.44E-03	8.64E-03	1,244E-05
Ru	106	2122 keV	4.00E-02	6.44E-01	2,574E-02
This	144	2186 keV	7.00E-01	8.64E-03	6.048E-03
Pr	144	2186 keV	7.00E-01	8.64E-03	6.048E-03
Y	90	2,186,254	1.40E-06	7.31E-02	1.023E-07
Pr	144	2368 keV	4.20E-05	8.64E-03	3,629E-07
Ru	106	2369 keV	5.96E-02	6.44E-01	3,840E-02
Ru	106	2542 keV	5.24E-03	6.44E-01	3.375E-03
Pr	144	2654 keV	2.00E-04	8.64E-03	1,728E-06
Ru	106	2705 keV	7.23E-03	6.44E-01	4.656E-03

Pr	144	2843 keV	1.10E-04	8.64E-03	9.504E-07
Ru	106	2852 keV	2.80E-03	6.44E-01	1.803E-03
Ru	106	3041 keV	1.38E-03	6.44E-01	8.887E-04
Tc	99	90 keV	0.00058	1.81E-05	1,050E-08

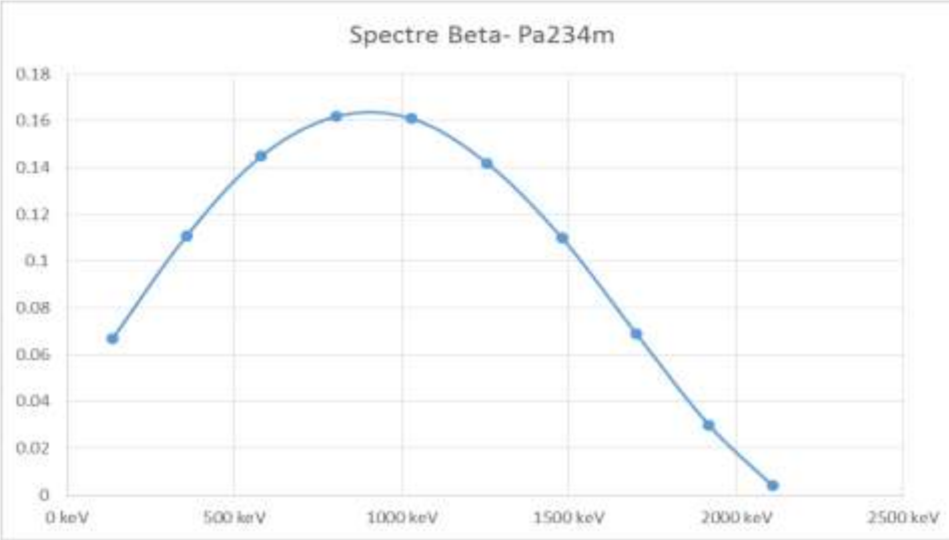
We mainly find Cs 137 (13.8%) and Ru-Rh 106 (64.4%). . In order not to neglect anything the gamma emissions of all the radionuclides of the spectrum So60 a been taken into account in an energy condensed spectrum on 12 groups:

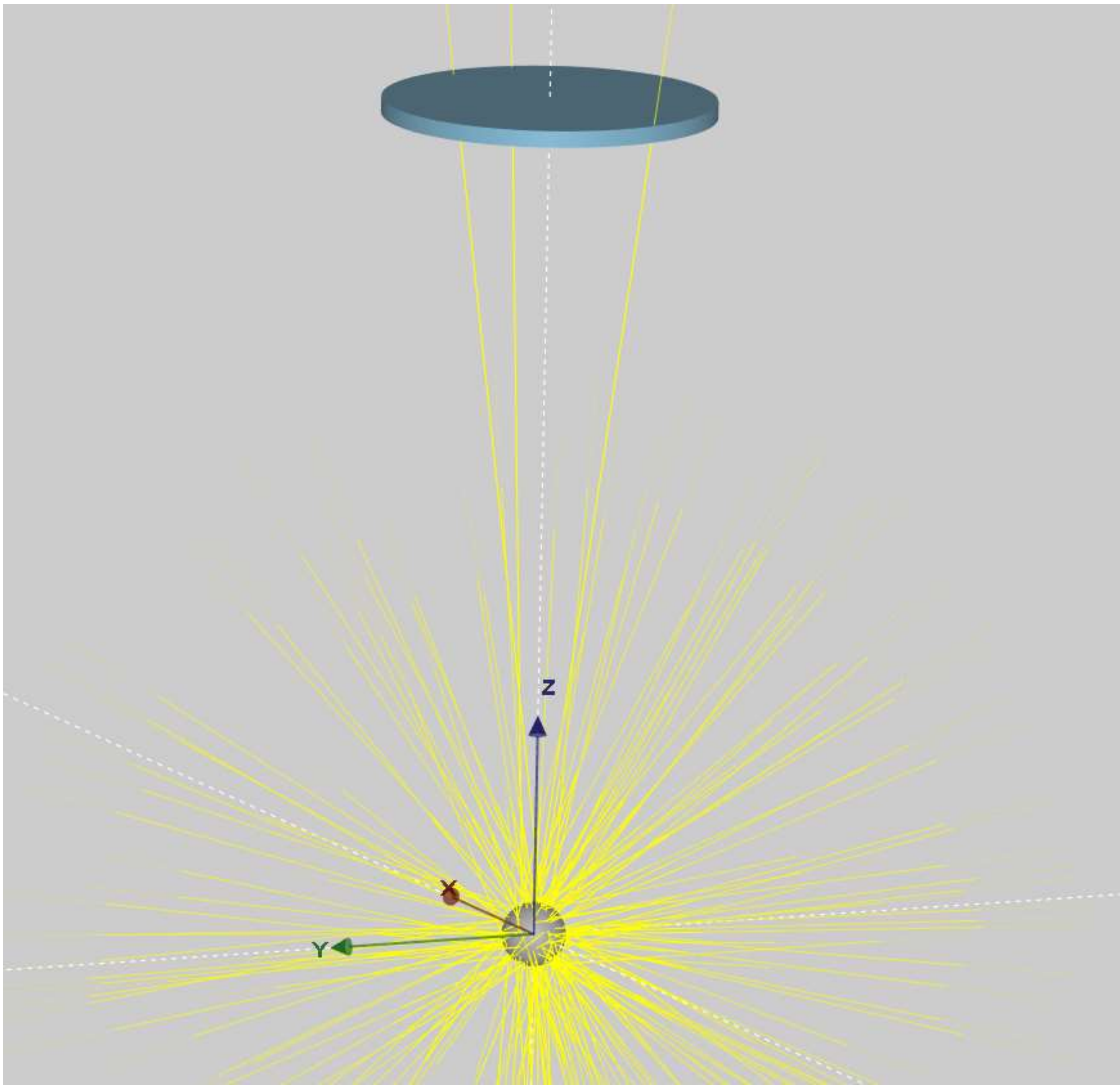
Condensed gamma spectrum recorded with Dosimex-GX 2.0 (see end of element list:Pr. fission spectrum –type # SO 60)

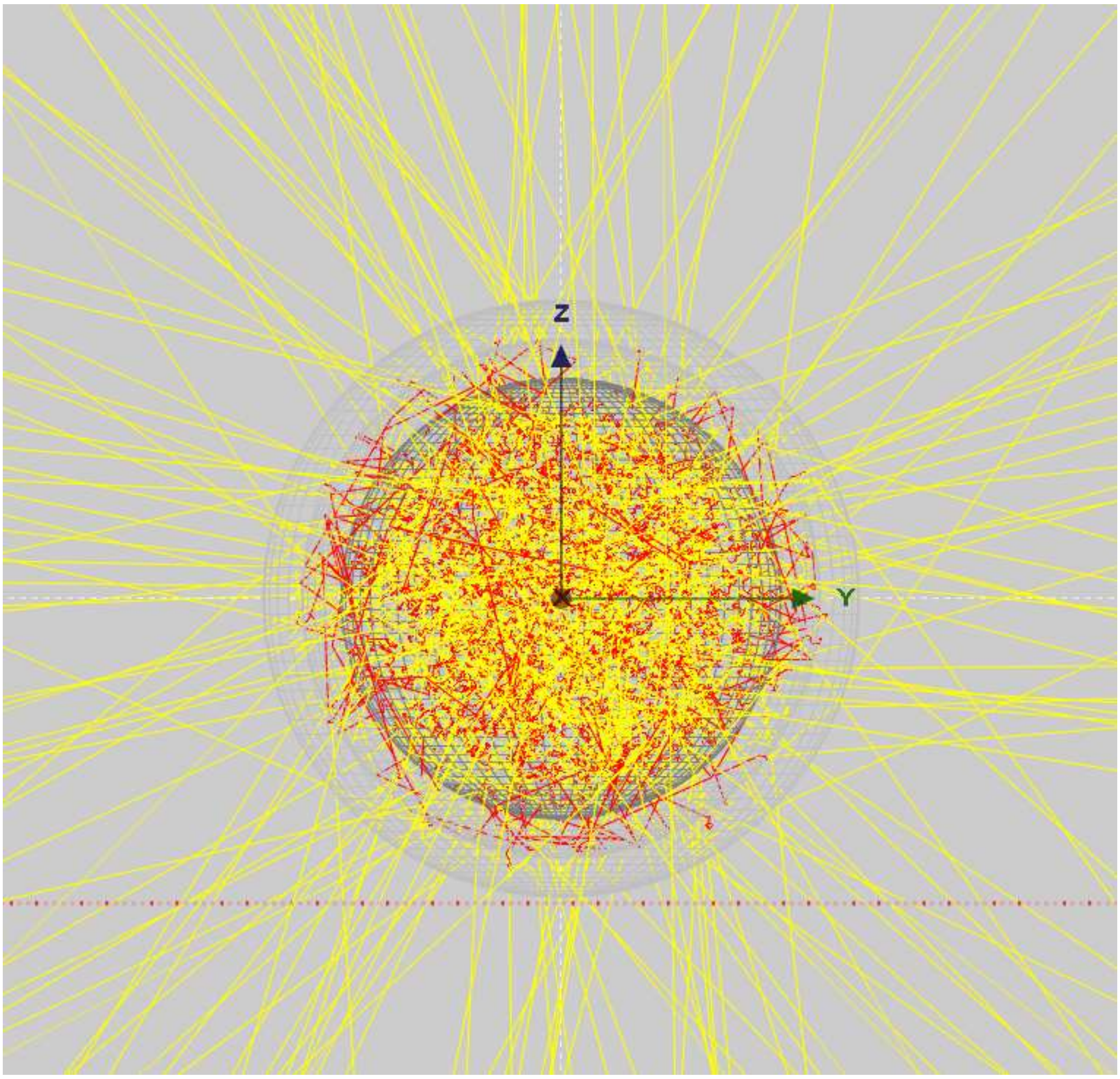
Energy	Intensity (%)
20 keV	1.35E + 00
73 keV	8.79E-01
189 keV	1.47E-01
509 keV	1.41E + 01
646 keV	1.93E + 01
788 keV	2.57E-01
874 keV	3.19E-01
1134 keV	2.10E + 00
1535 keV	1.39E-01
1964 keV	9.36E-02
2383 keV	4.18E-02
2781 keV	7.35E-03

III.3 Depleted uranium + braking radiation

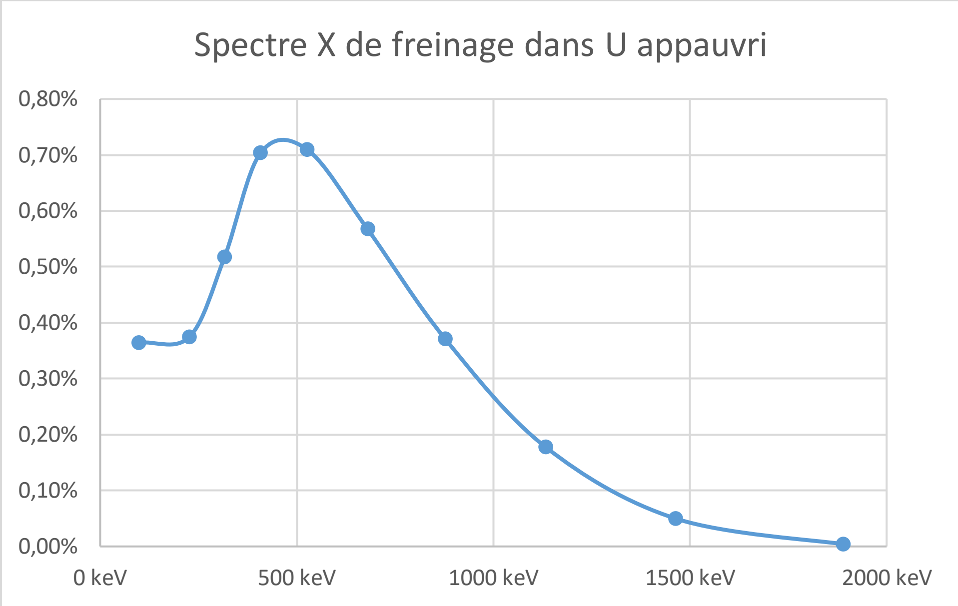
E (keV)	Intensity
137 keV	0.067
358 keV	0.111
581 keV	0.145
806 keV	0.162
1031 keV	0.161
1255 keV	0.142
1479 keV	0.11
1701 keV	0.069
1918 keV	0.03
2108 keV	0.004




Z
X
Y



Emission table X depleted U braking	
E	Intensity
98 keV	0.36%
227 keV	0.37%
316 keV	0.52%
408 keV	0.70%
527 keV	0.71%
680 keV	0.57%
878 keV	0.37%
1133 keV	0.18%
1463 keV	0.05%
1889 keV	0.00%


Spectre X de freinage dans U appauvri
0,80%
0,70%
0,60%
0,50%
0,40%
0,30%
0,20%
0,10%
0,00%
0 keV
500 keV
1000 keV
1500 keV
2000 keV