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GAMMA RAY BENCHMARK ON THE SPENT FUEL SHIPPING CASK TN 12

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ABSTRACT

The purpose of this benchmark is to compare measurements and calculation of gamma-ray dose rates around a shipping cask loaded with 12 spent fuel elements of FESSENHEIM PWR type. The benchmark provides a means to verify gamma-ray sources and gamma-ray transport calculation methods in shipping cask configurations. The comparison between measurements and calculations shows a good agreement except near the fuel element top where the discrepancy reaches a factor 2.

INTRODUCTION AND EXPERIMENTAL CONDITIONS

The benchmark definition, data collection and measurements have been performed by TRANSNUCLEAIRE. The interpretation by calculation has been made by the COMMISSARIAT A L'ENERGIE ATOMIQUE. This work was sponsored by the COMMISSION OF THE EUROPEAN COMMUNITIES[1].

The measurements have been made on a TN 12 Transnucléaire shipping cask (see figure 1). This cask contained 12 PWR spent fuel elements with following burn-up : 17040 MWd/MTU for 5 of them and 25090 MWd/MTU for the remaining seven. The associated cooling times were respectively 815 days and 382 days.

The package was slowly moved continuously in front of the detector which allowed to record the gamma dose rate measurements. The experimental area was such that the nearest walls were always more than 7 meters away from cask external surfaces and the back ground noise was equal 0.3 10^{-5} gray per hour. This latter one has been substracted from the measurements. The gamma detector was an ionization chamber NARDEUX 31 A with a diameter of 78 mm and a length of 115 mm. This chamber was calibrated with a ^{60}Co source by the Metrology Laboratory of Saclay (LMRI). The calibration uncertainties were about 7,5 %. The devise allowed to simultaneously record the gamma dose rate and the related cask axial location with an accuracy of 12 mm.

The measurements were made :

- at several distances : 7 cm (contact), 1 m, 2 m away from the envelope external surface
- at different angles around the cask.

The comparison between measurements and calculations has concerned only the generatrix corresponding to the highest dose rate (153°).

SHIPPING CASK TN12

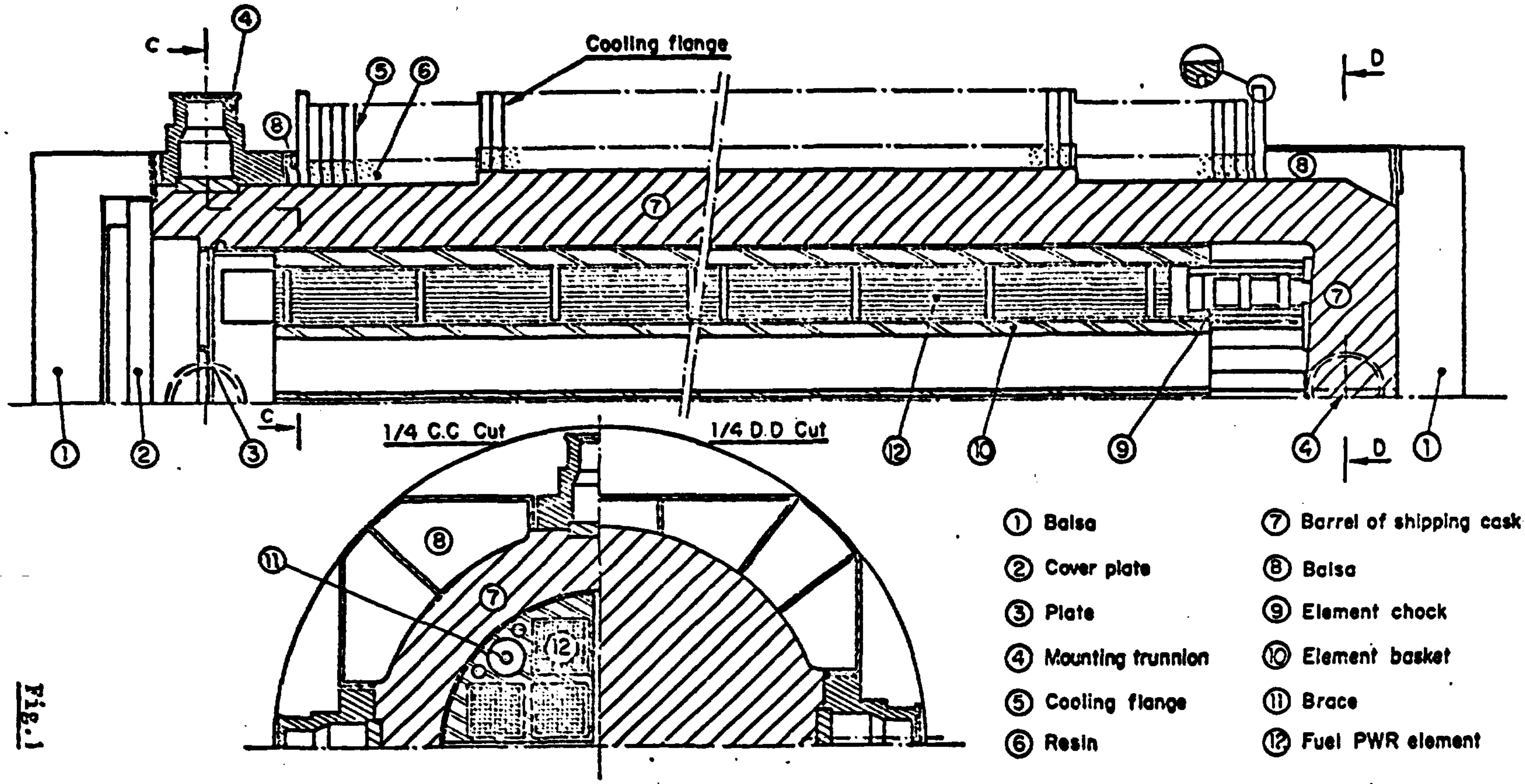


Fig.1

CALCULATION SCHEME

The calculation scheme contains the following codes :

- APOLLO[2] to compute the neutronic cell characteristics
- PEPIN[3] to compute the fission product gamma spectra
- ANISN[4] to determine the fast and thermal fluxes in the lower and upper fuel element parts
- MERCURE[5] to calculate the gamma dose rate when the gamma sources are known.

Cell calculations by APOLLO

This transport equation resolution in the fuel cell using the collision probabilities method gives versus burn up the following informations necessary to the exact evaluation of fission products :

- thermal and epithermal neutron fluxes in order to determine the capture rate on the fission products during operating duration
- fractional part of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu fast or/and thermal fissions.

Fission product inventory and corresponding gamma spectra

The fission product gamma rays are calculated by the PEPIN code. It solves the coupled differential equations describing the radioactive decay and capture chains for 635 nuclides. Two types of equation are solved :

- during operating duration :

$$\frac{dC_i(t)}{dt} = \sum_F \gamma_{iF}\, \tau_F + \sum_j C_j(t)\, \tau_j\, b_{j\rightarrow i} + \sum_k C_k(t)\, \lambda_k\, b_{k\rightarrow i} - \tau_i\, C_i(t) - \lambda_i\, C_i(t) \qquad (1)$$

$C_i(t)$ is the nuclide i concentration at time t, γ_{iF} is the independant yield for the fissile nuclide F, τ is the fission or capture reaction rate; $b_{k\rightarrow i}$ the branching ratio, λ_k the radioactive decay constant.

- during the cooling time

$$\frac{dC_i(t)}{dt} = \sum_k C_k(t)\, \lambda_k\, b_{k\rightarrow i} - \lambda_i\, C_i(t) \qquad (2)$$

PEPIN calculates the gamma spectra from activities $\lambda_i\ C_i\ (t_R)$ using the french CEA decay library[6].

The power diagram is described in steps where the neutronic characteristics are constant. We consider 2 steps and 4 steps which respectively correspond to both burn up : 17040 MWd/MTU and 25090 MWd/MTU. As each fuel element did not have exactly one of these considered burn up values, we performed the following corrections :

- axial interpolation to take into account the axial burn up distribution
- multiplication of the gamma spectra by ratio between the real burn up and the considered one.

Top and bottom structural material activations

The structural materials are composed of inconel and stainless steel. Therefore we consider the following activation reactions :

Co^{59} (n, γ) Co^{60} Ni^{58} (n, p) Co^{58} Fe^{54} (n, p) Mn^{54}

In fact only the first one is important. The activity A_i after the i^{th} step is given by the relation (3) :

$$A_i = A_{i-1}\, e^{-\lambda D_{i-1}} + \sigma \phi_{i-1}\, P_o \left[1 - e^{-\lambda D_{i-1}}\right] \qquad (3)$$

where λ is the radioactive decay constant, D_i the step duration, σ the microscopic cross section for fast or thermal neutrons, ϕ_i the corresponding flux, P_o the initial target nucleus concentration. The $\sigma\phi^i$ product is evaluated by the transport code ANISN in upper and lower part of the reactor versus the distance to the fissile part. Two imperfections exist in these calculations : the control rods and the soluble boron in water are missing.

Gamma dose rate calculation by MERCURE IV

MERCURE IV is a three dimensional multigroup, line of sight point attenuation kernel code developed at Saclay. This code calculates gamma and neutron dose rates and gamma heating. The dose rate $D(\vec{r})$ is evaluated by the relation (4) :

$$D(\vec{r}) = \iiint_{V_S} d\vec{ro} \sum_{g=1}^{N} S(\vec{r_o}, g)\ G(\vec{r}, \vec{r_o}, g) \qquad (4)$$

$S(\vec{r_o}, g)$ is the source density in the group g

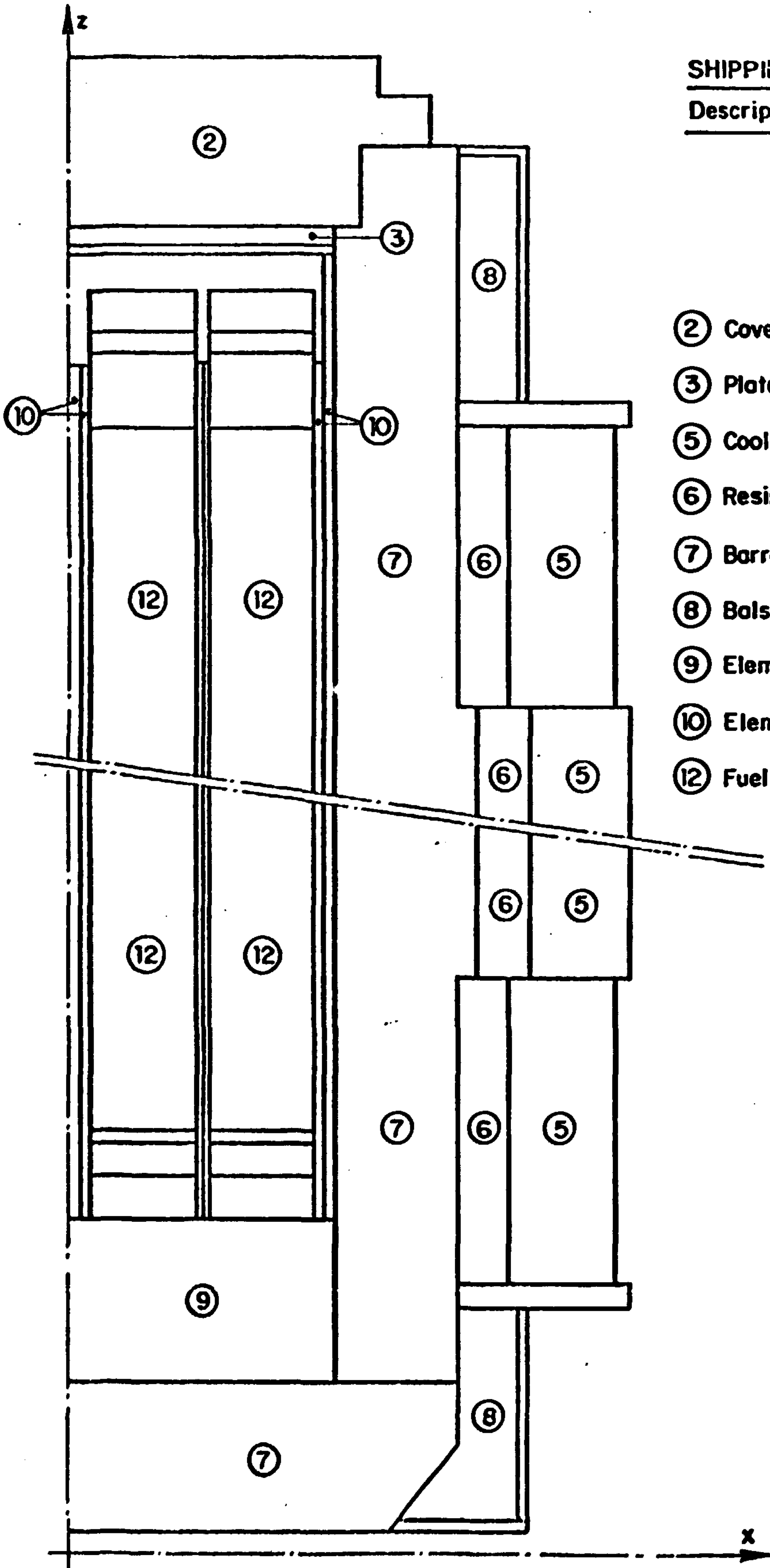
$G(\vec{r}, \vec{r_o}, g)$ is the attenuation kernel between $\vec{r_o}$ and $\vec{r}$

The integration over the source volume V_S and the summation of the N groups are performed by a Monte Carlo method using an optimized importance function.

In our benchmark a very sophisticated description of the geometry is used : 414 homogeneous volumes discribe the cask (see figure 2) ; figure 3 shows a cask section in its middle part.

SHIPPING CASK TN 12

Description of the geometry



(2) Cover plate (steel)

(3) Plate (steel)

(5) Cooling flange (copper)

(6) Resin

(7) Barrel of shipping cask

(8) Balsa

(9) Element chock

(10) Element basket

(12) Fuel PWR element

Fig.2

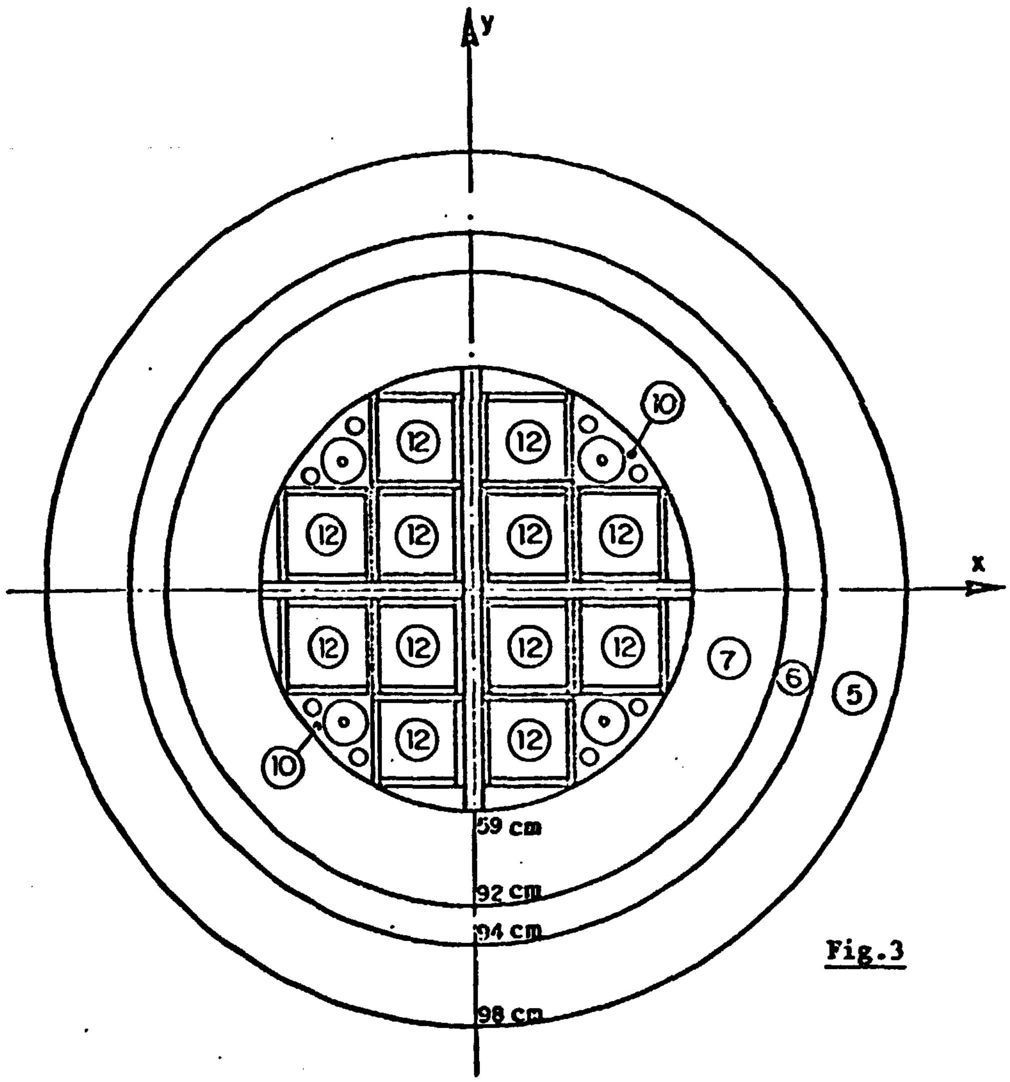


Fig.3

In MERCURE IV, we take into account diffusion in using a build up factor. This last one uses the Kitazume formula to treat the alternation of crossing materials. The statistical incertainty due to Monte Carlo use is about 4 % with 95 % of confidence.

MOST IMPORTANT RESULTS AND COMPARISON WITH EXPERIMENTAL VALUES

Figures 4 and 5 show the comparison between experimental and calculated dose rate values at 7 cm and at 1 m away from the cask external envoloppe surface.

Examination of the contact dose rate is specially instructive because it is very sensitive to a very detailed description of geometry and of source spatial distribution. The dose rate has two peaks which are due to a smaller thickness of the iron shield at the cask ends.

At the middle of the cask the calculated gamma dose rate values are very close to the experimental ones :

Fig. 4 _ Comparison between experimental and calculated dose rate values.at 7 cm.

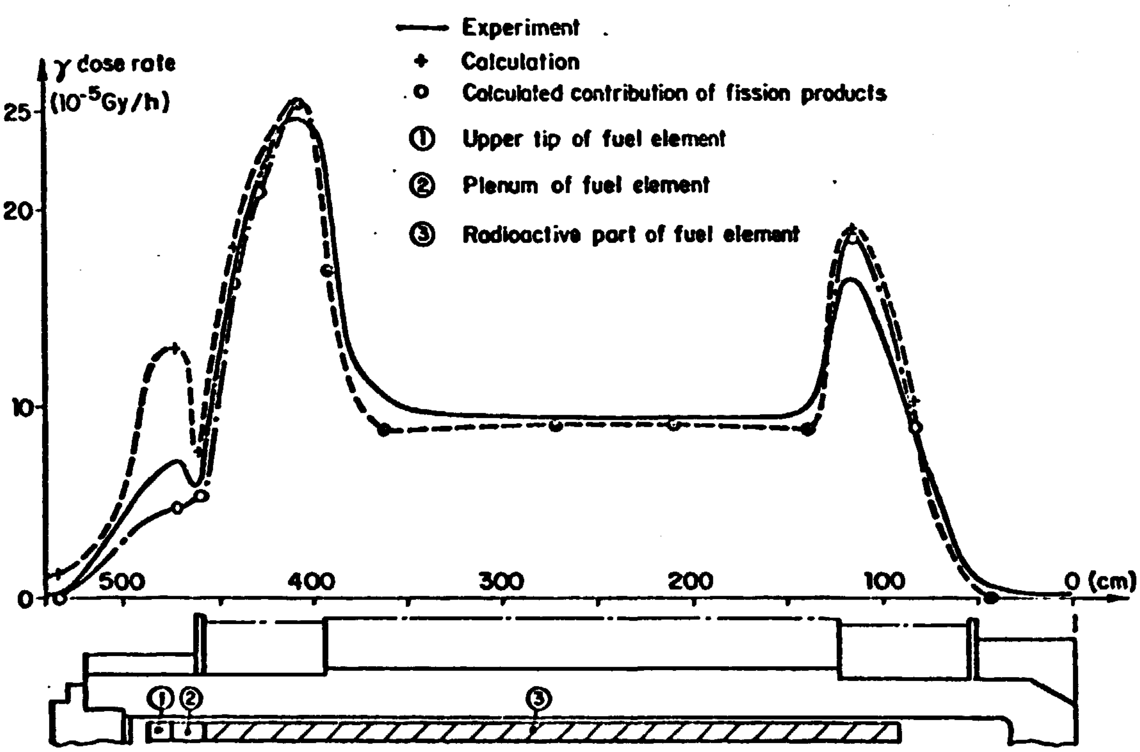
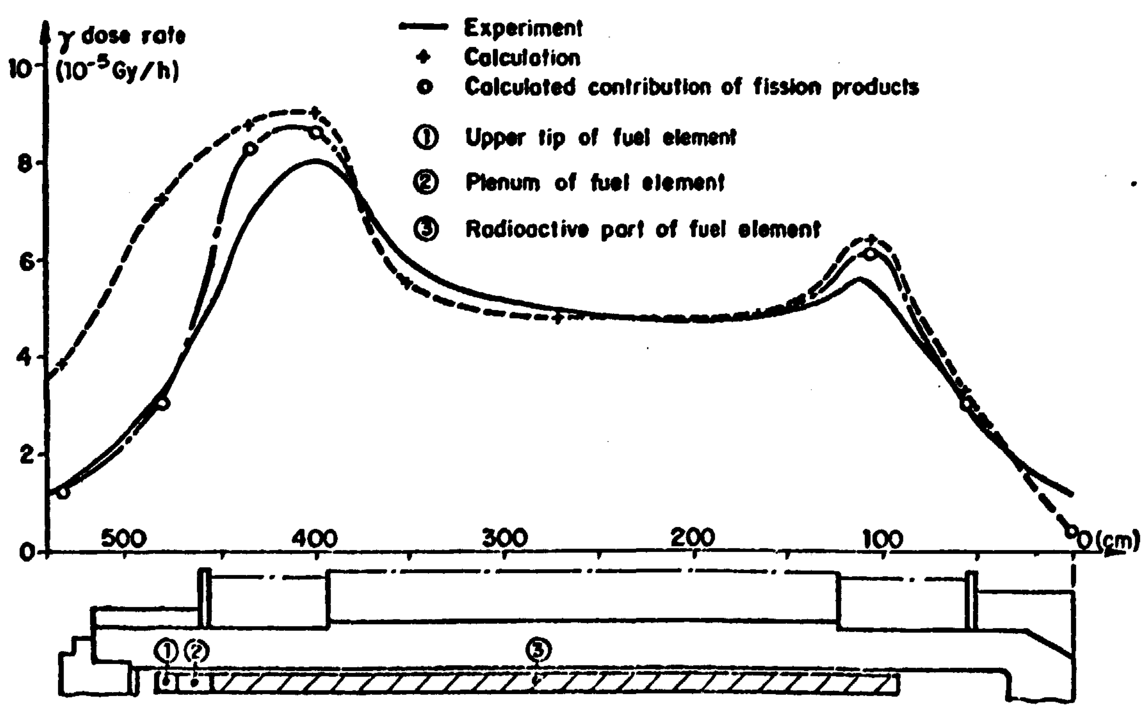


Fig. 5 _ Comparison between experimental and calculated dose rate values. at 1 m.



	Calculated values	Experimental values
at 7 cm	89 µSV/h	94 µSV/h
at 1 m	47,9 µSV/h	49,7 µSV/h

CONCLUSION

Three cases may be distinguished in comparison between measurements and calculations :

- at the bottom level of the fuel element : the agreement is good
- at the fissile part level : the agreement is excellent
- at the top level of the fuel element : the discripency reaches a factor 2. This phenomena is due in great part to the following facts :
  - the control rods are not taken into account for calculation of the thermal flux in core along the upper part
  - the boron capture in water is missing outside the fissile part.

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