Topical meeting on reactor physics and shielding
Chicago, IL (USA) 17 - 19 sep 1984
CEA-CONF--7466

NEUTRON TRANSPORT BY TRIPOLI-2 CODE IN THE LOWER PART OF A PWR PIT AND IN THE PIT-ACCESS CELL

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SUMMARY

The neutrons, which exit the reactor vessel, leak between the reactor vessel and the primary concrete shield and provide high dose rates in the lower part of the reactor pit. In this part of the reactor the biological shield is reduced at the level of the pit-access cell door and of the ventilation duct. Consequently there is a pin-point increase of dose rate at reactor place where access must be free during power operation. Then it is necessary to estimate with a good precision the consequences of this reduction to decide actions which have to be undertaken.

Studies concern 900 MWe and 1300 MWe plants ; calculation results are compared with dose rate measurements which have been done at several interesting points of reactor. Neutron transport is studied in two steps :

- first Monte Carlo calculation carried out by the TRIPOLI-2 system gives the current of neutrons entering into the pit-access cell
- a second calculation is about neutron transport in this cell ; it is also performed with the TRIPOLI-2 code which uses a very realistic model for cell geometry. Then the door efficiency is evaluated by a SN one dimension code.

Neutron transport is well studied with a Monte Carlo code which allows to treat correctly complex and real three dimensional geometries and to take into account neutron multiple diffusions.

The neutrons, which exit the reactor vessel, leak between the reactor vessel and the primary concrete shield and provide high dose rates in the lower part of the reactor pit. In this part of the reactor the biological shield is reduced at the level of the pit-access cell door and of the ventilation duct. Therefore there is a pinpoint increase of dose rate at reactor places where access must be free during power operation. It is not possible to obturate the cell because it is used to inject cooling air into the pit and to avoid over pressure in case of a postulated loss-of-coolant accident. Then it is necessary to estimate the dose rate with the actual configuration to decide actions which have to be undertaken.

Studies concern 1300 MWe plants and calculation results will be compared with measurements done at the french PALUEL plant when these one will be available.

The calculation scheme used for these studies can be cut into four steps :

- calculation of the flux of neutrons exiting the reactor vessel by ANISN (1) and DOT-3.5 (2) codes
- neutron transport in the pit by the TRIPOLI-2 (3) system and determination of neutron current entering the pit-access cell
- neutron transport in the cell by TRIPOLI-2
- evaluation of cell-shield efficiency by ANISN code

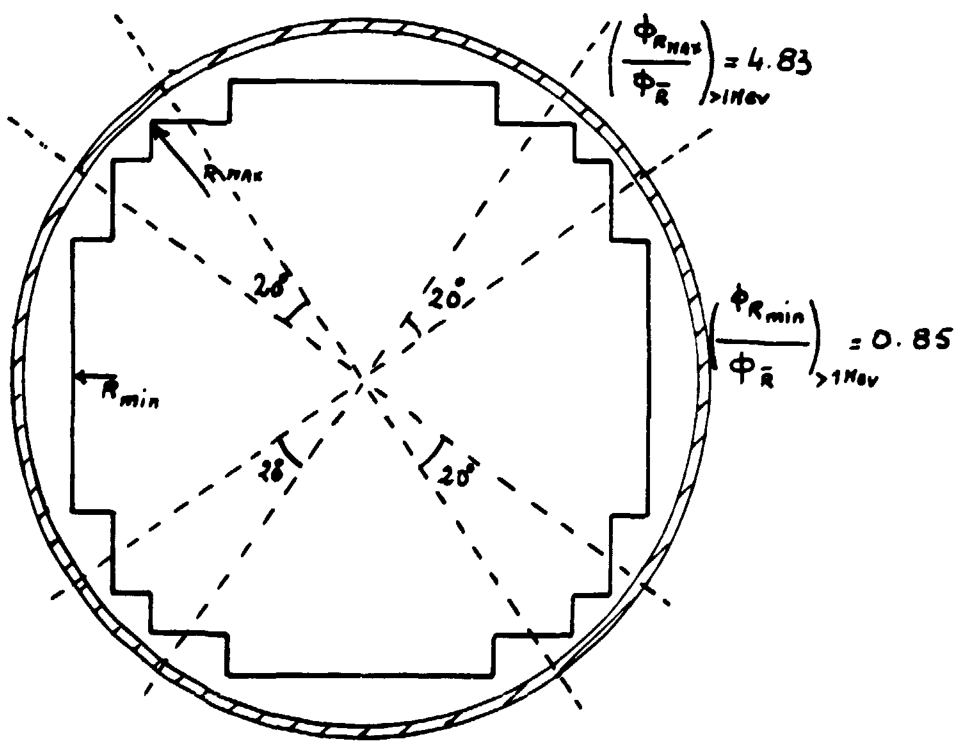
NEUTRONS EXITING THE REACTOR VESSEL

Calculation in the core midplane is performed with ANISN code, in one-dimension cylindrical geometry using 100 energy groups, 16 angular directions and cross-section Legendre expansions of order P3. Fluxes are supposed to be independant of the z-axis between 0 and 90 cm from the core midplane.

DOT-3.5 code with 19 energy groups, 16 angular directions, P3-expanded cross sections uses a R x Z geometry to describe a part of the core, inner structures, the pressure vessel, the reactor cavity and concrete walls. This code uses the collapsed cross sections by zone obtained by ANISN and gives the angular and energy distribution of neutrons exiting the vessel between 90 cm and the vessel bend.

The ANISN calculation used a core cylindrical modelisation with an averaged value $\overline{R}$ ($\overline{R}$ = 168 cm) of the core radius, which allowed to respect the core surface. The true form of the core is shown on the figure 1 and it can be considered that the radius varies between a minimum value R_{min} = 161 cm and a maximum value R_{max} = 185 cm. 19 groups ANISN calculations were performed with these two radius and scalar fluxes greater than 1 MeV were compared to each other.

Figure 1 - Azimuthal variation of flux exiting the reactor vessel



$$C_{min} = \Phi\ (E > 1\ \text{MeV}, R_{min})/\Phi\ (E > 1\ \text{MeV}, \bar{R}) = 0.86$$

$$C_{max} = \Phi\ (E > 1\ \text{MeV}, R_{max})/\Phi\ (E > 1\ \text{MeV}, \bar{R}) = 4.83$$

Two hypothesis are done: the variation of angular flux versus core radius is the same as the variation of scalar flux greater than 1 MeV ; this variation is conserved all along the vessel.

The azimuthal variation of angular fluxes is taken into account by multiplying $\bar{R}$ values by a C-factor ($C = C_{min}$ or $C = C_{max}$) constant by azimuthal regions of 20° width for $C = C_{max}$ and 70° width for $C = C_{min}$. Mean value of this factor on 2π is equal to 1.75.

This process is rough and probably overestimates fluxes.

NEUTRON TRANSPORT IN THE REACTOR PIT

It is performed with the three dimensional Monte Carlo code TRIPOLI-2 because of the geometry complexity and of the importance of multiple neutron diffusions.

Reactor geometry is described between core midplane and the pit bottom in a realistic way : vessel, heat insulation steel and pit. A recess represents the entrance of the pit-access cell. (figure 2).

Cross sections are finely described in a 270 step mode.

Biasing technics of TRIPOLI-2 are used for reducing computing time (an importance is attributed at each point and the simulation is processed according to this importance). For this typical problem of diffusion in void the general idea is : any point at the void boundary is equi-important and the importance decreases in the matter versus the distance to void boundary. A special biasing technic allows to better study neutrons which stream through the annular void between the vessel and the concrete.

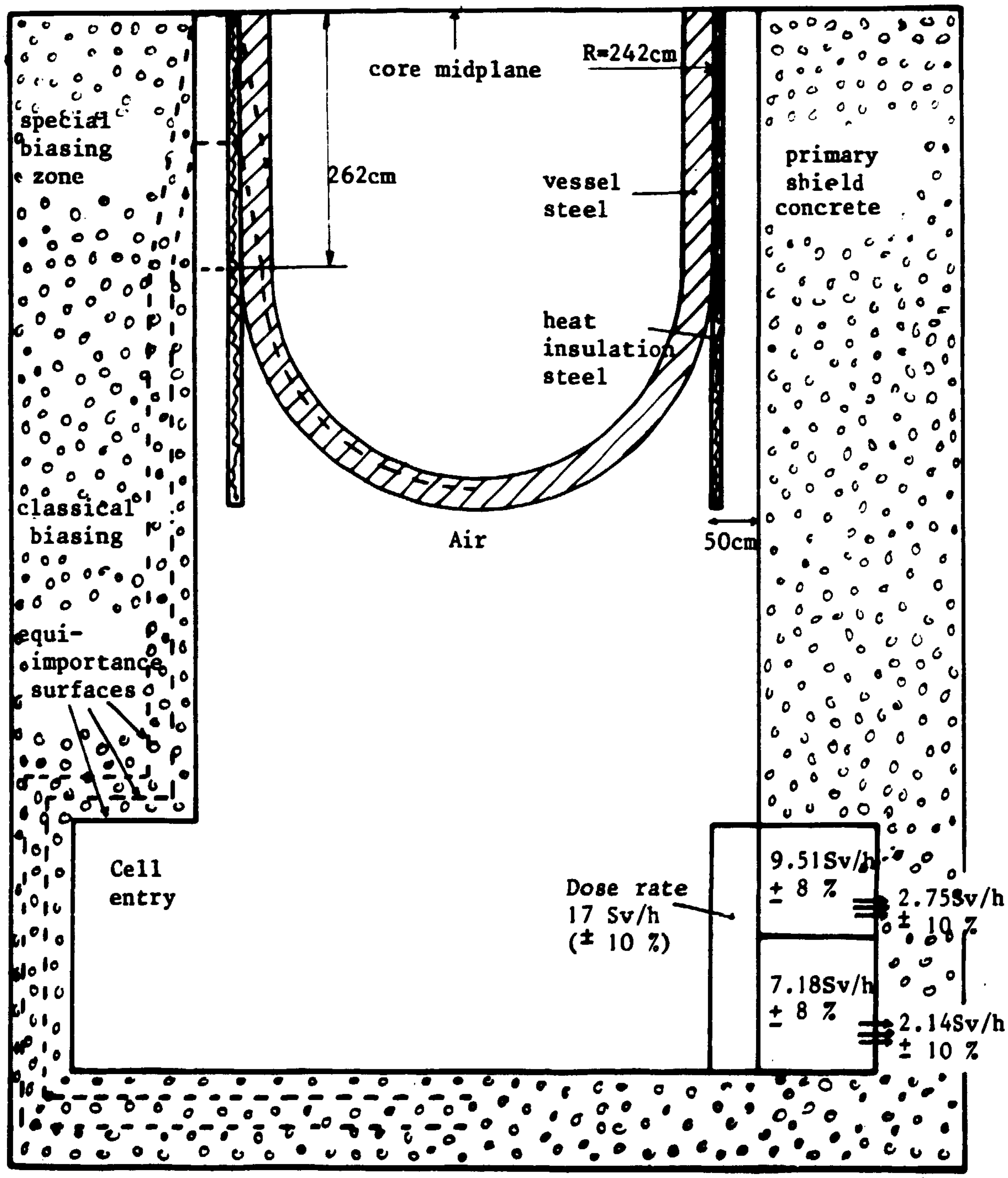
Sources for TRIPOLI are given by the angular fluxes exiting the vessel calculated by DOT. Automatic connection between DOT and TRIPOLI is carried out by the DOTTRI module (4).

This calculation furnishes the dose rate and flux spectrum verticaly under the annular void. Values can be compared with measurements in a 900 MWe plant (5).

		900 MWe plant measurements	1300 MWe plan calculation
	Dose rate	from 12 Sv/h to 13 Sv/h	17 Sv/h ($\pm$ 8 %)
	> 1 MeV	from 2.02 10^4 to 2.3 10^4	from 0.77 10^6 to 1.9 10^6
Flux	thermal	8.5 10^7	4.9 10^7
$n.cm^{-2}.s^{-1}$	TOTAL	2.36 10^8	1.1 10^8 ($\pm$ 5 %)

Figure 2 - Neutron transport in the reactor pit

Geometry modelized for TRIPOLI - vertical cut -



Thermal flux, then total flux is smaller in 1300 MWe because the thickness of core water slab and of vessel is more important. But the fast neutron flux is higher because the vessel-pit interval is 50 cm wide in 1300 MWe and only 40 cm wide in 900 MWe, then fast neutrons can stream more easily.

This calculation furnishes also the energy and angular distribution of the current of neutrons entering the cell. It must be noted that the current spectrum is downgraded: more than 50 % of neutrons are thermal. The angular distribution is very sharped toward the access cell (90 % of neutrons are at less than 30° from the cell axis).

Flux and dose rate at the cell entry

Upper part	Dose rate	9.51 Sv/h ($\pm$ 8 %)
	Flux > 1 MeV	5.78 10^5 n/cm^2.s
	Thermal flux	4.65 10^7 n/cm^2.s
	Total flux	8.8 10^7 n/cm^2.s ($\pm$ 6 %)
	Current entering the cell	2.7 Sv/h ($\pm$ 10 %)
Lower part	Dose rate	7.18 Sv/h ($\pm$ 8 %)
	Current entering the cell	2.14 Sv/h ($\pm$ 10 %)

NEUTRON TRANSPORT IN THE PIT-ACCESS CELL

It is also performed with the TRIPOLI-2 system. The geometry of the cell is given by figure 3 : it can be decomposed into two parts : a narrow, 5 m long tunnel and a room with 5 m^2 section and 7 m high where are placed an access-door and a ventilation duct in the upper part. Geometry is modelized by a parallelepiped and by a cylinder (fig. 4,5) and takes into account the ventilation duct. The door is considered as in concrete.

Sources for this calculation are given by the energy and angular distribution of the current entering the cell, which has been calculated in the precedent step by TRIPOLI.

Dose rate is calculated at several places : (see fig. 5)

- in front of the access door 0,17 Sv/h ($\pm$ 3 %)
- in front of the ventilation duct 1,3 10^{-2} Sv/h ($\pm$ 4 %)
- current exciting the cell by the ventilation duct 4,3 10^{-3} Sv/h ($\pm$ 3 %)

High values of dose rate are found. Measurements done in 900 MWe plants indicated a dose rate equal to 0,40 Sv/h in front of the door. Studies have been performed with TRIPOLI-2 to calculate efficiency of concrete mazes set in the cell :these ones reduce dose rate to 6.10^{-2} Sv/h.

Figure 3 - Cell geometry horizontal cut -

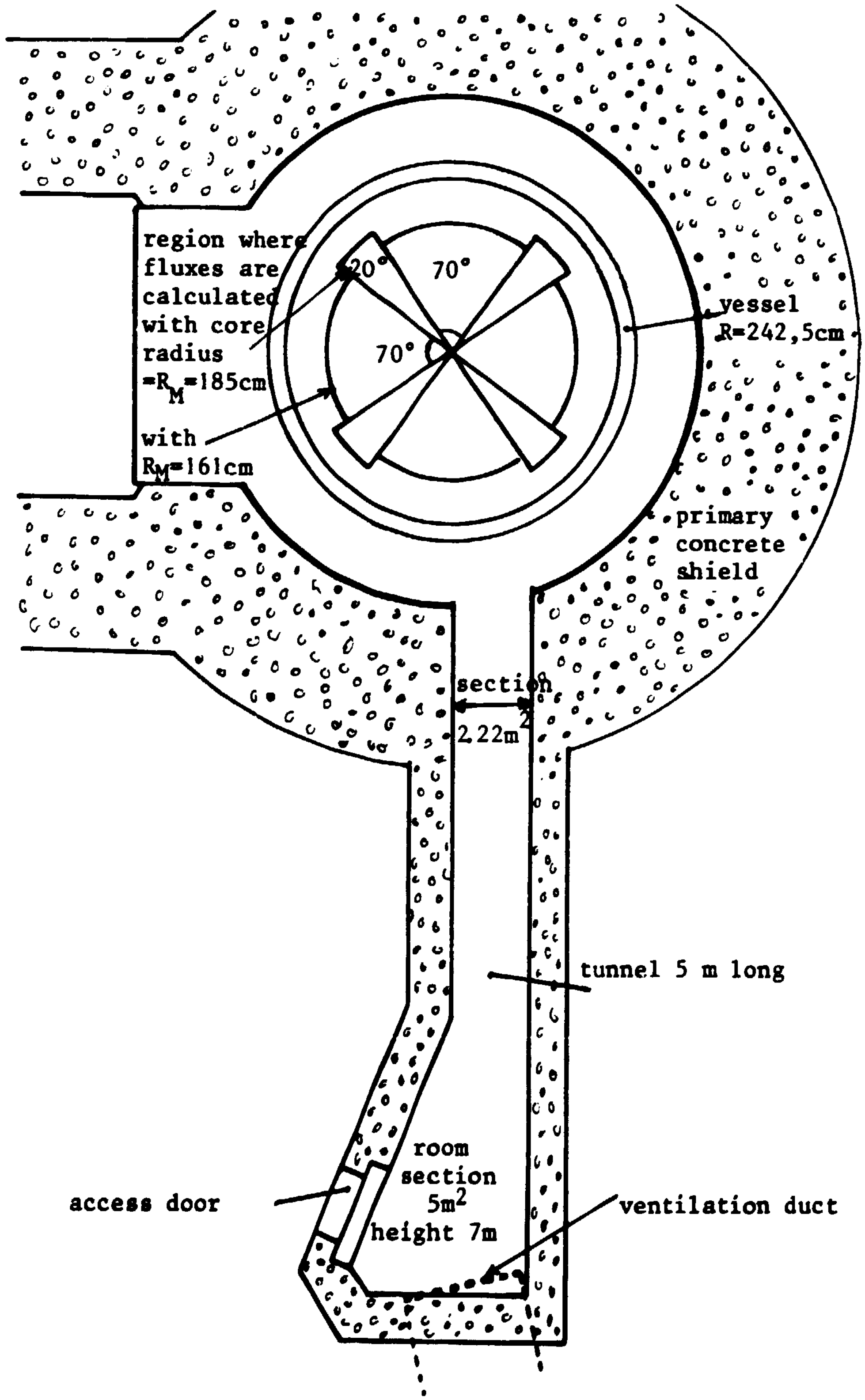


Figure 4 - Cell-geometry (modelization for TRIPOLI) - Horizontal cut -

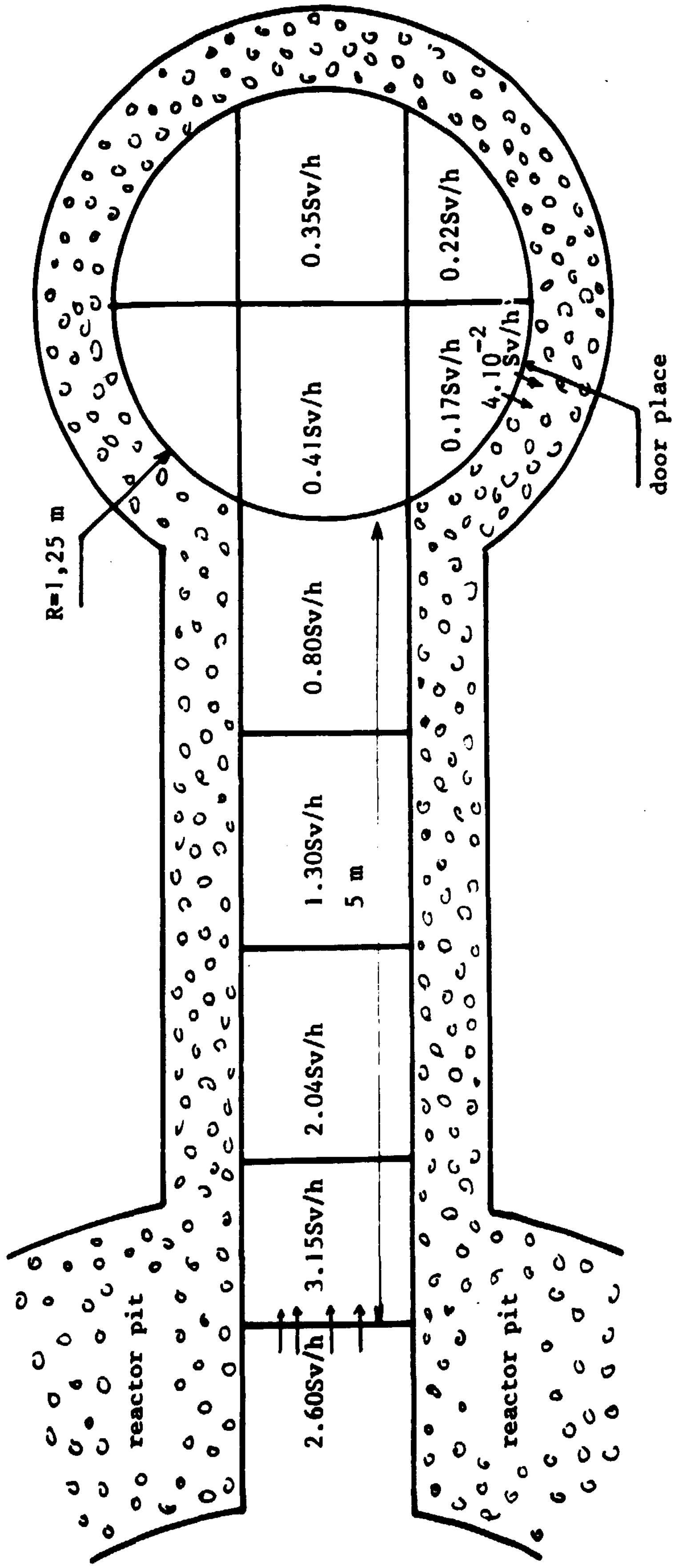
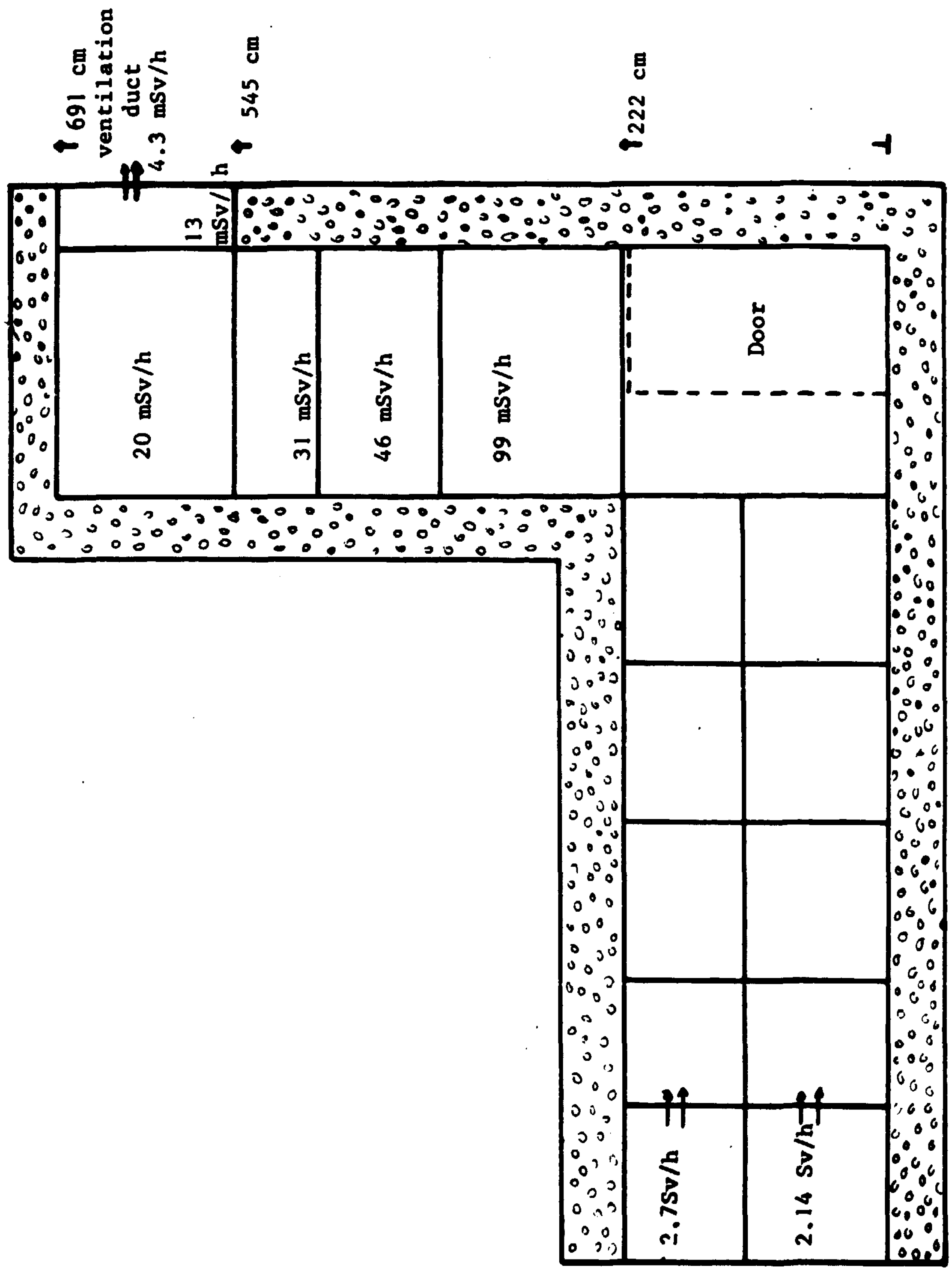


Figure 5 - Cell geometry (modelization for TRIPOLI) - vertical cut -



EVALUATION OF CELL-SHIELD EFFICIENCY

ANISN code is used to calculate the door efficiency. The door is constitued by :

10 cm of Robatel-10 compound

7 cm of lead

3.2 cm of steel

Neutron flux entering the door	$2.8\ 10^{6}\ n/cm^{2}.s$
Associated dose rate	0.13 Sv/h
Dose rate behind the door	140 µSv/h

CONCLUSION

Calculated dose rate values are high, particularly near the ventilation duct, but not very different from the admissible limit (2 m Sv/h) ; then no additionnal shield is provided now.

Comparison between calculation and measurements will be done at each step of the calculation scheme. In France the first 1300 MWe plant is just operating and measurements are no complete enough and too recent to be used in comparisons. However calculation scheme seems correct.The TRIPOLI-2 system is a powerfull and easy to use tool for such studies with complex geometries and multiple neutron diffusion problems.

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