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SHIELDING DESIGN FOR PWR IN FRANCE

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ABSTRACT

Shielding calculation scheme used in France for PWR is presented here for 900 MWe and 1300 MWe plants built by EDF the french utility giving electricity.

Neutron dose rate at areas accessible by personnel during the reactor operation is calculated and compared with the measurements which were carried out in 900 MWe units up to now. Measurements on the first french 1300 MWe reactor are foreseen at the end of 1983.

SHIELDING PROBLEMS IN PWR

During the reactor full power operation the checked accessibility to some areas of the reactor building is required for inspection and servicing,(see fig1) or for fast interventions :

- peripheral area (specially in front of primary coolant pump casemate and steam generator casemate)
- lower part of the reactor building behind the shielding door of access to the reactor cavity
- shielded areas on the operating floor, behind steam generator shielding for example.

Reactor shielding has to answer to conflicting criteria :

- safety requires a sufficient pressure release mechanism for a postulated loss-of-coolant accident, then large openings in the cavity shield wall. That provides an escaping path for radiations, then an increasing of biological dose.
- however the operating dose rate limit for short interventions is 200 mrem/h, and the additional shield provided for limiting the neutron dose rate during operation must not block access areas for maintenance operations when the reactor is shut down for refueling.

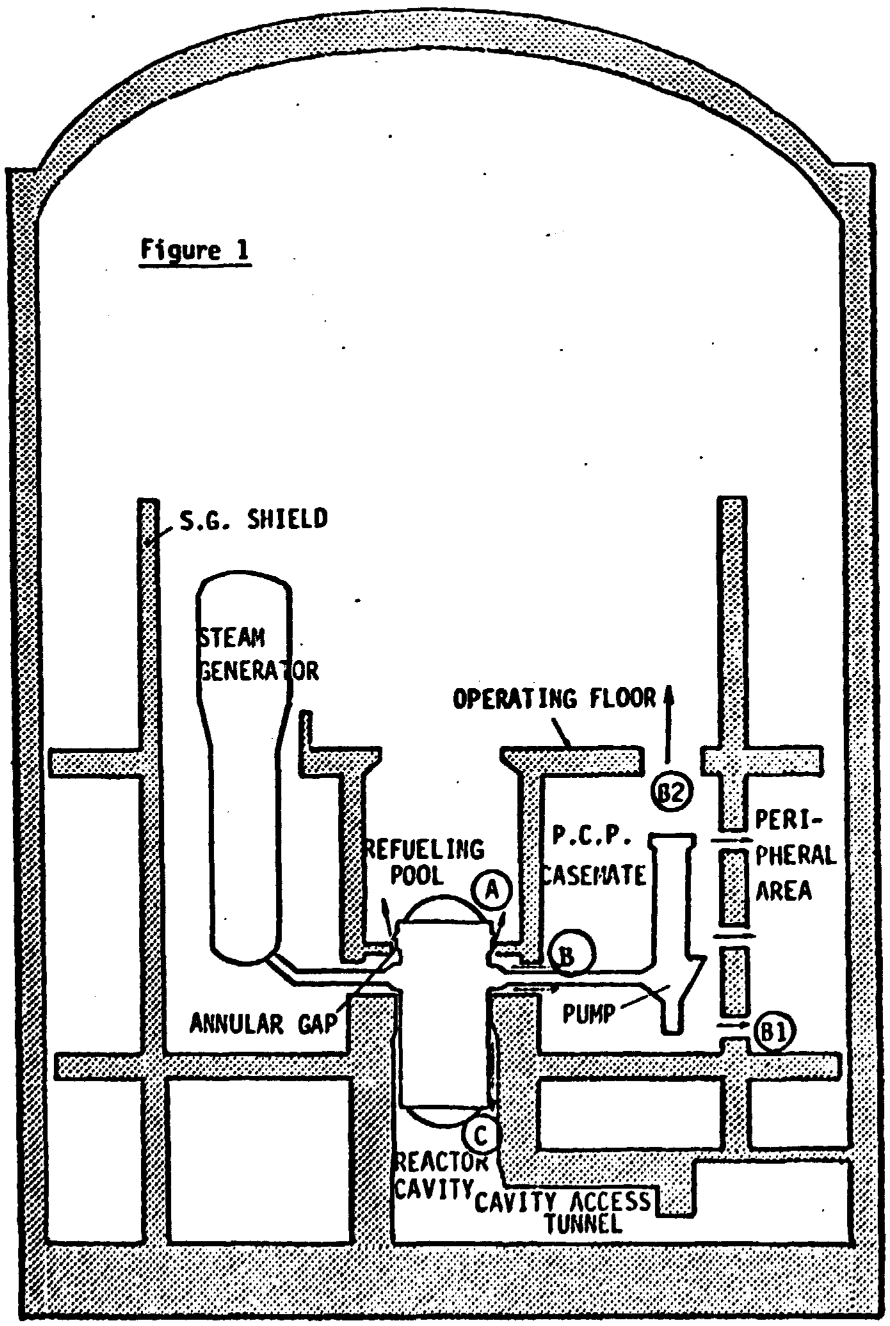
Neutron dose rate evaluation was the greatest part of our work because it is more important and more difficult to be calculated than the gamma dose rate. Neutrons leak essentially from the reactor cavity through three openings (see fig.1) :

- A the upper part of the reactor cavity : the annular air gap between the vessel and the biological shield wall provides paths for neutron streaming into the refueling cavity and hence to the operating floor
- B around coolant pipes : neutrons escape into primary coolant pump casemates, and then into the peripheral area (B_1) and containment building above the operating floor (B_2)
- C the lower part of the reactor cavity into cavity-access tunnel.

The principal objectives of these studies are to limit the neutron dose rate by additional shields (access tunnel at the bottom of the cavity), to choose the best place for the components which have to be accessible or to predefine the accessible places where the dose rate is less that 200 mrem/h (peripheral area).

An other part of our studies is to calculate the radiation embrittlement of the pressure vessel and of the surveillance specimens.

Figure 1



CALCULATION SCHEME AND USED CODES

Neutron studies

- ANISN[1] code for one dimension calculation in the core midplane with 100 energy groups, angular quadratures of order 8, cross-section Legendre expansions of order P_3 ; cross sections are given by the ENDFB library. Three types of results are calculated by this code :
  - 19 group colapsed cross sections for materials located in reactor vessel and for the reactor cavity concrete
  - line-of-sight attenuation cross sections with one energy group for several damage responses to be further used by the three-dimensional MERCURE-4[2] code which calculates damage rate of specimens and of vessel steel
  - 19 group angular fluxes exiting the pressure vessel near the core midplane.
- DOT[3] code with 19 energy groups, an S8 angular quadrature, P3-expanded cross-sections and in a RxZ geometry which describes a part of the core, inner structures, the pressure vessel, the reactor cavity and concrete walls. This code uses the collapsed cross sections obtained by ANISN and gives the angular and energy distribution of neutrons exiting the pressure vessel between the core midplane and the primary coolant pipe axis.
- TRIPOLI-2[4] is used to calculate neutron dose rate at places out of the vessel such as operating floor, primary coolant pump cavity and peripheral area in front of it and lower cavity access tunnel. This three dimensional Monte Carlo code has been used because of the complexity of the geometry, its great heterogeneity and of the importance of multiple neutron diffusions.

Monte Carlo calculations are done with as exact as possible geometries ; neutron paths in matter (concrete, steel, etc....) are studied without using an albedo. Sophisticated biasing techniques are used to reduce computing-time of neutron diffusion in concrete (exponential transformation[4]) and special techniques allow to treat small-angle streaming in voids (collision biasing[4]).

Sources for TRIPOLI calculations are given by angular fluxes exiting the reactor vessel calculated by DOT; automatical connection between DOT and TRIPOLI is carried out by the DOTTRI-code[5]. In a first step, the flux of the neutrons, escaping from the cavity at A and entering into the refueling pool, has been evaluated by the VIDDOT code which calculates the collimated flux from angular fluxes leaving the vessel. This step will be replaced by using DOTTRI and TRIPOLI in order to take into account real geometry with nozzles, pipes and their supporting blocks..

Gamma-ray studies

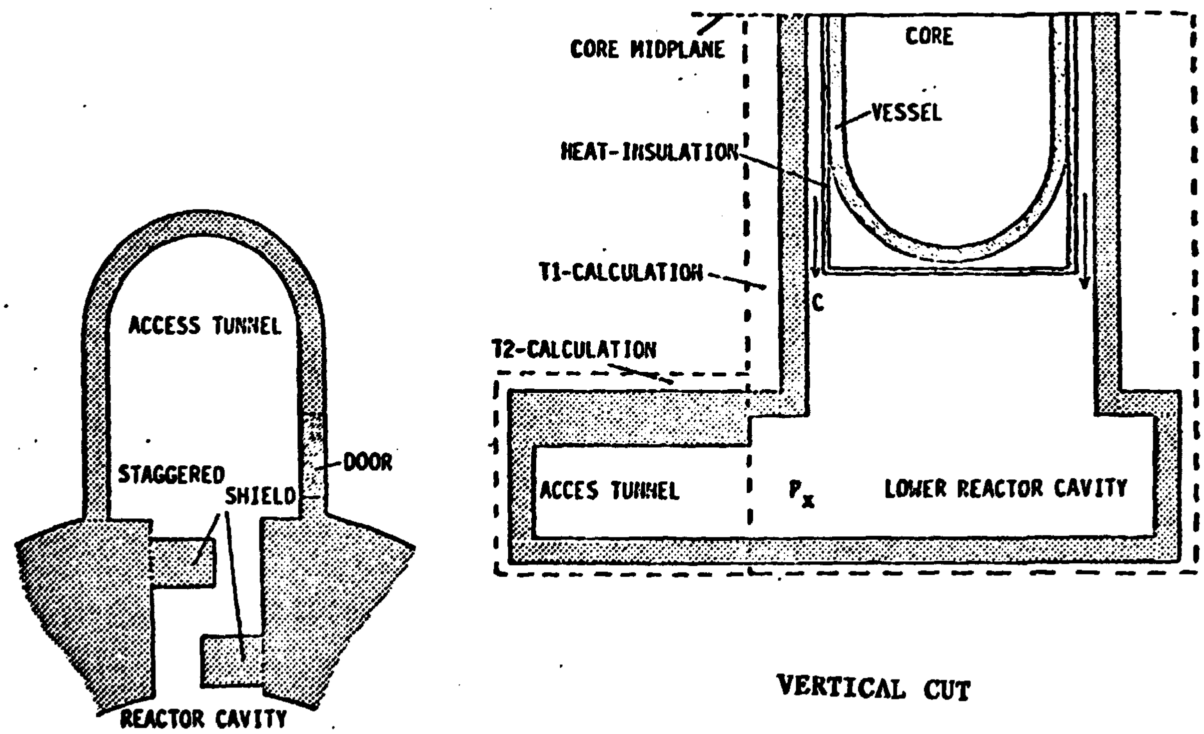
The dose rate due to gamma rays emitted by circuits during the reactor operation was calculated[6] by MERCURE-4. Contribution of structure activation (thermal insulation steel) to the gamma dose rate during shut down in the lower part of the reactor cavity was calculated[7] with TRINISHI[8] ; this code determines first collision sources then uses a line-of-sight point attenuation kernel method. Studies about corrosion products are communicated in an other paper[9]. Gamma problems are simpler than neutron ones and are not related in detail here.

MAIN RESOLVED PROBLEMS AND RESULTS

Reactor cavity access tunnel (lower part of the cavity) .

Neutron dose rate behind the tunnel door was calculated in two steps (see fig.2) :

- T1 TRIPOLI calculation of neutron streaming (C) through the gap between the vessel and the concrete wall ; DOTTRI is used to obtain TRIPOLI sources
- T2 TRIPOLI calculation of neutron propagation in the tunnel.



HORIZONTAL CUT

VERTICAL CUT

FIGURE 2

Calculated dose rate at P (for a 1300 MWe reactor) is compared with measurements in a 900 MWe reactor (comparison is possible because the flux in both cases are close to each other);agreement is good :

- measurements	1 219 mrem/h	1 299 mrem/h
- TRIPOLI calculation T1	1 205 mrem/h	

T2 calculation was done for a 900 MWe reactor.

Placing staggered shields in the tunnel lowers the neutron dose rate to 200 mrem/h and the gamma dose rate to less than 200 mrem/h behind the door. In 1300 MWe units the door has an additional shield (10 cm of Robatel N10-compound) to reduce neutron dose rate.

Primary coolant pump casemate

Interest of this calculation is not to study accessibility which is not allowed during the reactor operation, but to define :

- neutron streaming through the opening in the casemate roof emerging on the operating floor (B2)

- neutron streaming through the casemate-access doors (B1) which lead to the peripheral area which has to be accessible by personnel.

A TRIPOLI calculation (T3) with a realistic geometric model of the casemate (see fig.3) defines the dose rate at any point due to the neutrons streaming through the annular gap (B) around primary pipes. Principal results are :

- streaming through B2-opening
  dose rate 2087 mrem/h
  total flux $5.8\ 10^5\ n/cm^2.s$

- dose rate on the back wall, specially at the place of doors (B1) : from 5 000 to 22 000 mrem/h (see fig.4).

Another TRIPOLI calculation (T4) with a geometric model of the peripheral area on both sides of a door allowed to define the area (± 4 m from the door) where the dose rate is greater than the admissible value of 200 mrem/h and gave proof that a concrete wall in front of the door reduces the dose rate by a factor 2.

Containment building above the operating floor

A TRIPOLI calculation (T5) was run for a 900 MWe plant. The source is given by the neutrons streaming through the annular gap between the reactor vessel and the concrete into the refueling pool (see fig.5 : A). Angular and energy source distribution was calculated by VIDOT and ajusted at measurements with activation detectors in the upper part of the reactor cavity, VIDOT cannot effectively take into account complex geometry details in this part of the cavity.

TRIPOLI geometric model is very realistic : the containment building, steam generators, the pressurizer, the refueling pool, the seal ledge and the heat insulation are modelized (see fig.5,6,7). Fig.7 is a cross section above the operating floor and shows calculated and measured dose rate at any place in several units. Two spectra are presented in figures 8 and 9 : the current entering the refueling pool and the flux on the operating floor near refueling-pool edge.

A similar TRIPOLI calculation (T6) was run for a 1300 MWe unit which has four loops. Annular source (from the reactor cavity) was determined by VIDOT and the source of neutrons streaming from primary pump casemates was given by TRIPOLI (T3 calculation). This calculation will be run again with using the connecting DOTTRI code and TRIPOLI to calculate neutron streaming by the annular gap. Results will be compared with measurements on the 1300 MWe Paluel unit about the end of 1983.

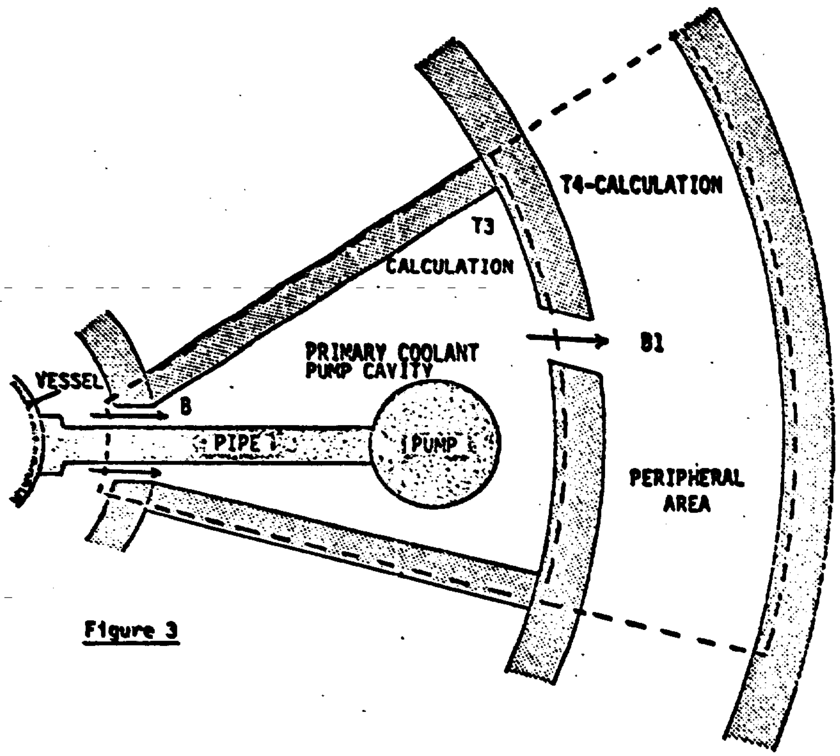


Figure 3

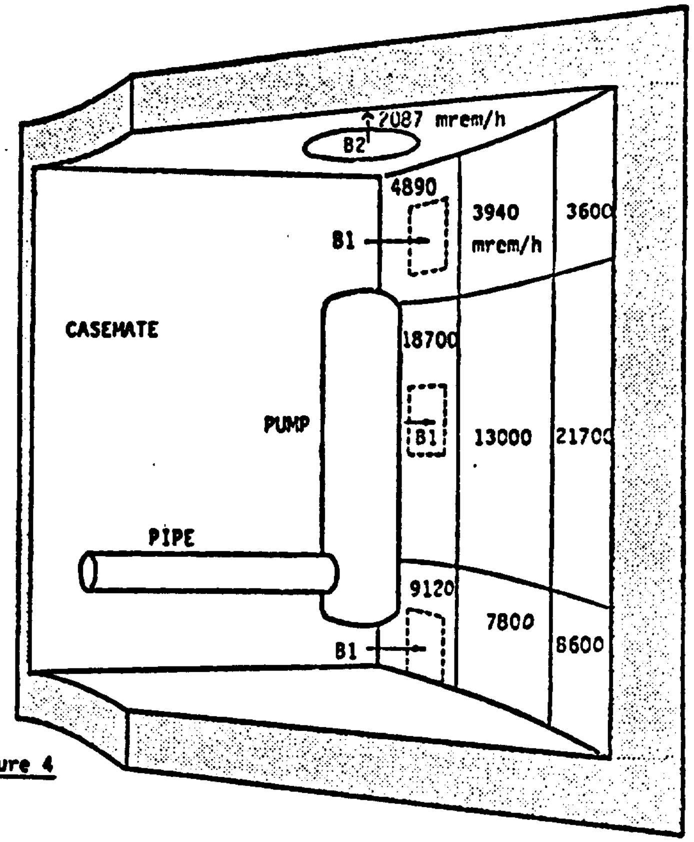


Figure 4

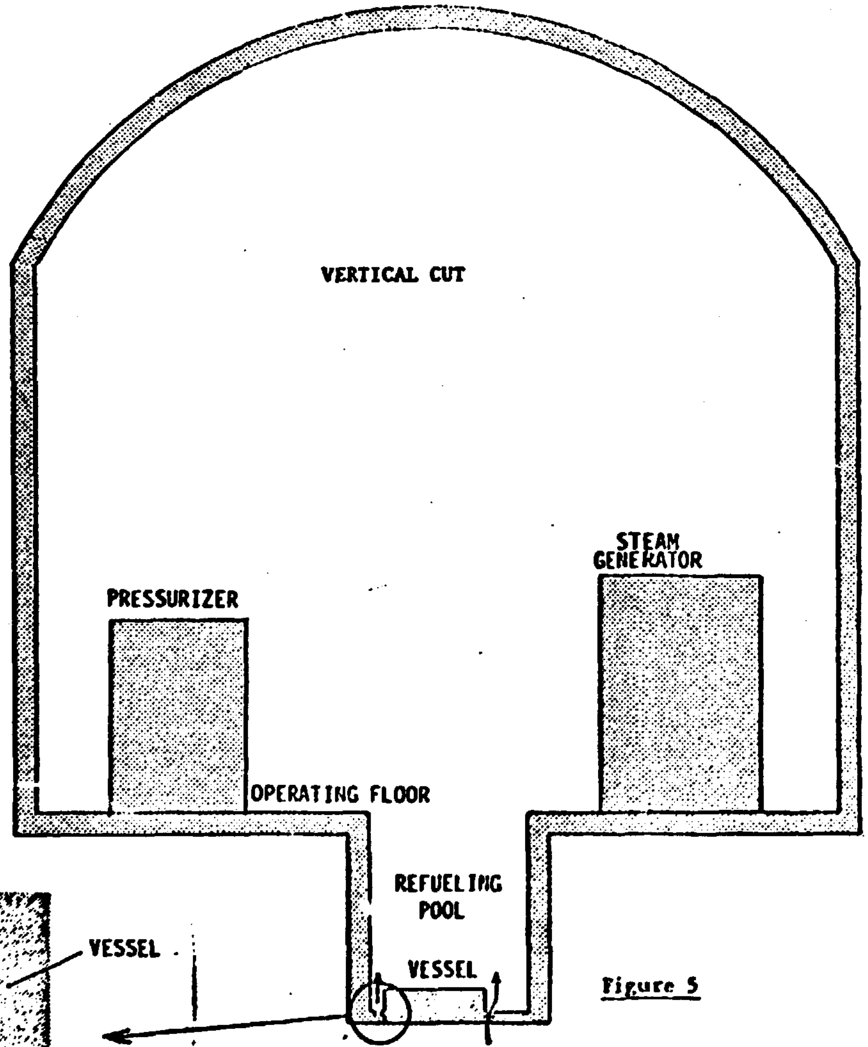


Figure 5

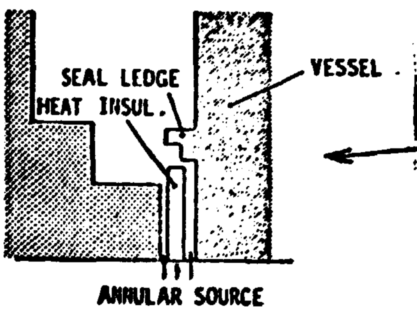


Figure 6

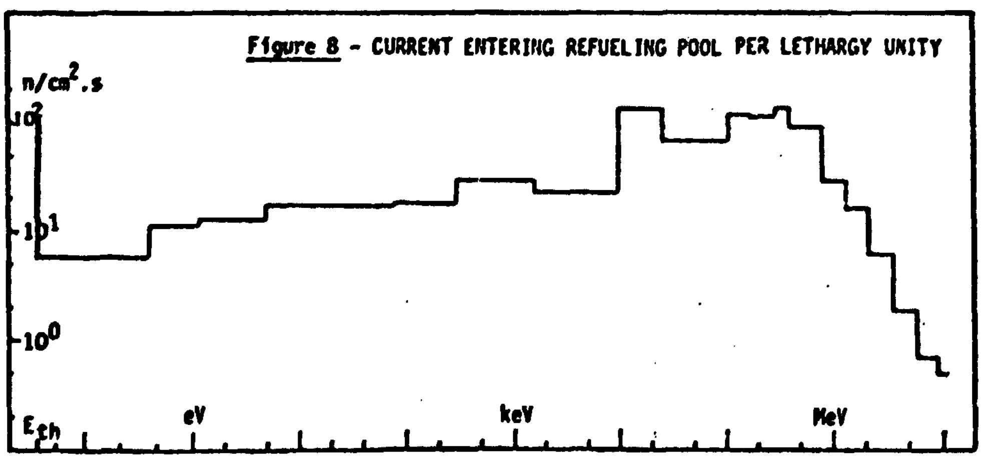


Figure 8 - CURRENT ENTERING REFUELING POOL PER LETHARGY UNITY

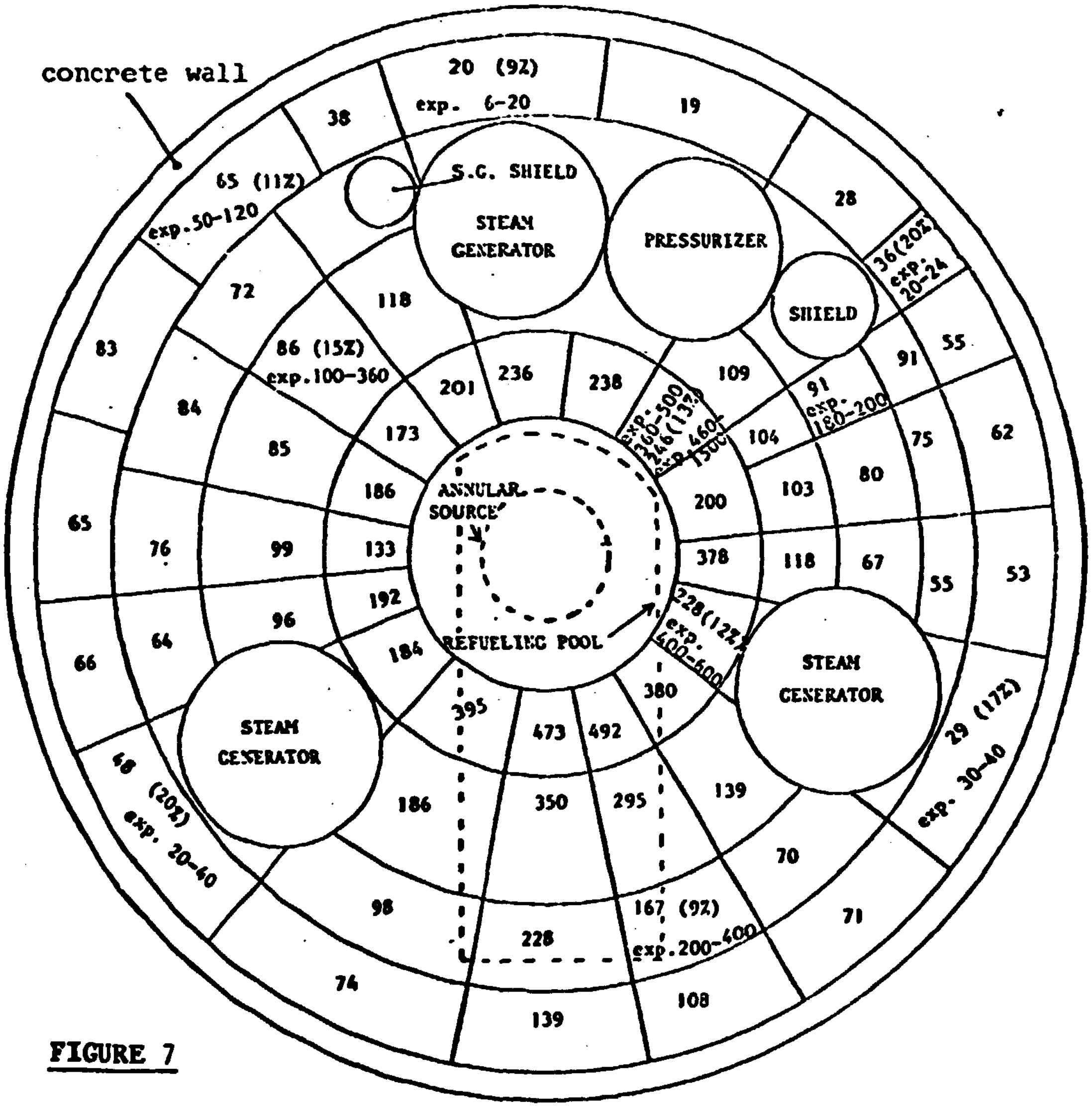


FIGURE 7

TS - TRIPOLI RESULTS COMPARED WITH MEASUREMENTS DOSE RATE ON THE OPERATING FLOOR

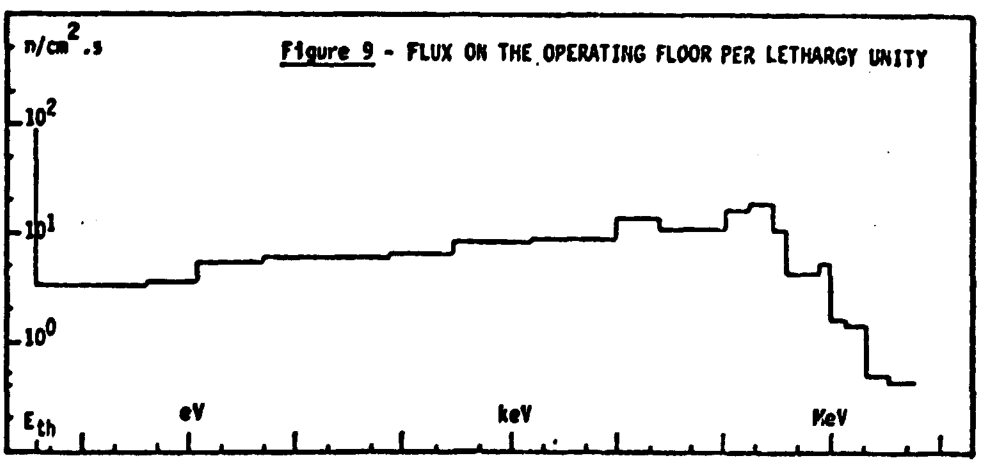


Figure 9 - FLUX ON THE OPERATING FLOOR PER LETHARGY UNITY

Damage rate in the reactor vessel and surveillance specimens

Several ANISN transport calculations were run with a cylindrical geometry along different directions corresponding to different water thicknesses (see fig.10). Then successions of MERCURE-4 calculations were run with the same geometry and the same cylindrised source density ; one succession of calculations was done for each response function : flux greater than 1 MeV, damage cross section. These calculations furnish line-of-sight attenuation coefficients for each zone and each material by adjusting with the reference ANISN calculations. Then a final three-dimensional MERCURE-4 calculation is run for specimens and for the vessel using three dimensional neutron sources.

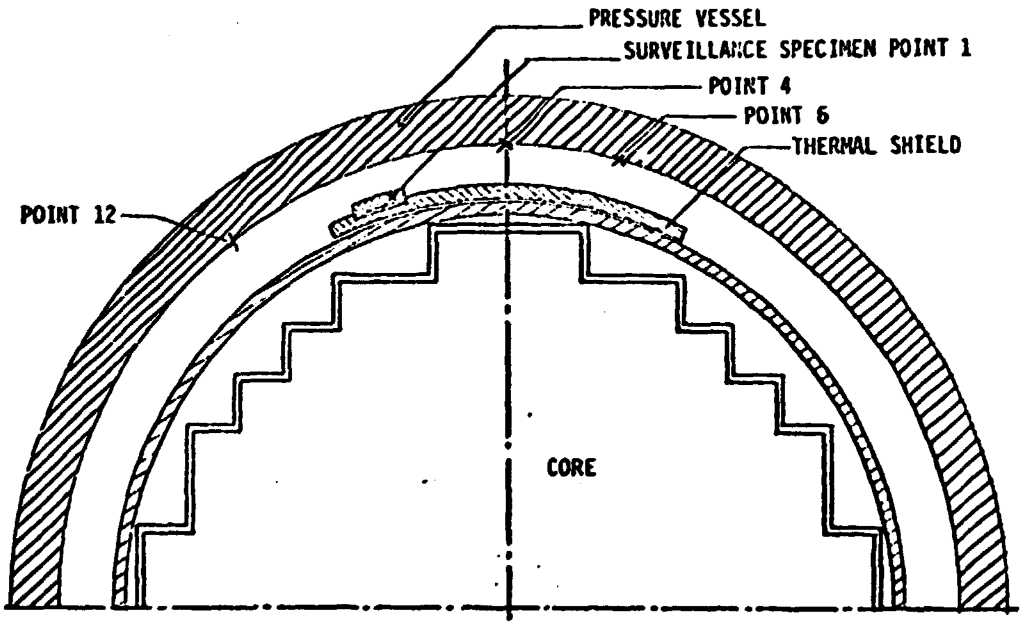


FIGURE 10

This calculation scheme has been tested on the C.A.P. demonstration reactor ; several comparisons[10] between calculations and measurements are shown in table 2.

Threshold detectors are placed in surveillance specimen boxes which are hung on the thermal shield.

Reaction rate $\sigma\Phi$	Cu	Nb	Np 237	U 238
Measurement	1.62 E-9	1.84 E-7	1.86 E-6	4.25 E-7
Calculated value	1.94 E-9	2.35 E-7	2.12 E-6	4.24 E-7
Reaction	$^{63}Cu(n,\alpha)^{60}Co$	$^{93}Nb(n,n')^{93m}Nb$	$^{237}Np(n,f)FP$	$^{238}U(n,f)$ FP
Cross section $\bar{\sigma}$	0.5 mb	153 mb	1312 mb	305 mb

CONCLUSION

We propose the following scheme for neutron shielding calculations in PWR : ANISN and DOT SN-codes for simple 1-D or 2-D configurations and the Monte Carlo TRIPOLI code which is a powerfull and not very expensive tool (about 30 mn of IBM 3033) for design problems in complex geometries. Connection between SN and Monte Carlo codes is done by DOTTRI.

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