IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Faire avancer la sûreté nucléaire

Neutron-gamma flux and dose calculations in a Pressurized Water Reactor (PWR)

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ANS SUMMER Meeting

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- Monte Carlo calculations in the PWR
  - ❖ Reactor model – Tihange-1
  - ❖ Fission distribution in the core
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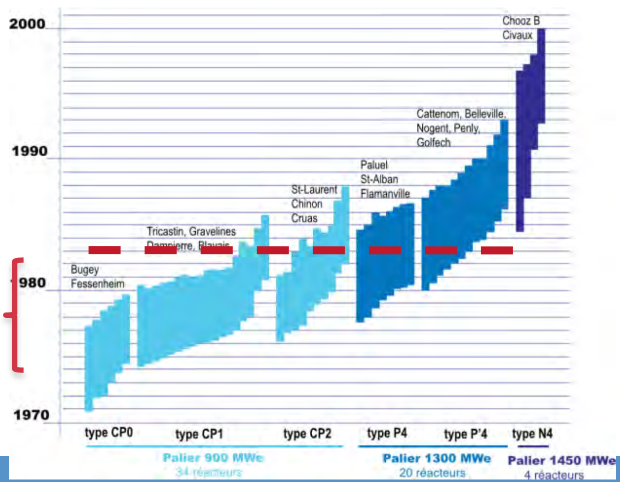
General context

- EDF NPPs initially <u>designed for 40 years</u>
- Feasibility study to extend the duration of NPPs operation
- IRSN: R&D as a support of expertise on safety
  - Mainly concerning non replaceable components: <u>reactor vessel and internals, containment, electrical cables</u>
  - Radiation environment calculation in the nuclear reactor

40 years old in 2020



R&D project

DISCOMS: DIstributed Sensing for COrium Monitoring and Safety

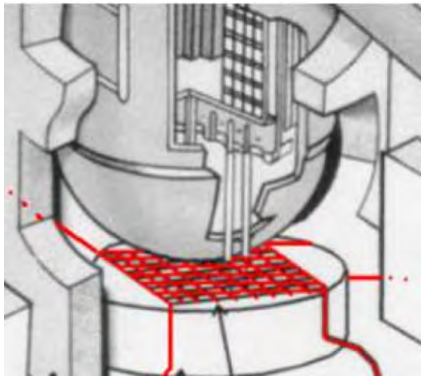
<u>Feasibility study of instrumentation implementation:</u>

- Optical Fiber Sensors (OFS)
- Self Powered Neutron Detectors (SPNDs)

<u>Monitor:</u>

- Status of the third barrier of confinement
- Pressure vessel breakthrough, concrete floor erosion and corium cooling

IRSN in charge of radiation environment calculations:
neutron and gamma dose, flux spectra



Monte Carlo simulation of the nuclear reactor

Monte Carlo codes:

Solve the n-γ transport equation using precise geometry & continuous energy cross sections

Core:

Dimensions ~3.6m x ~3m

Neutron mean free path ~1-2 cm

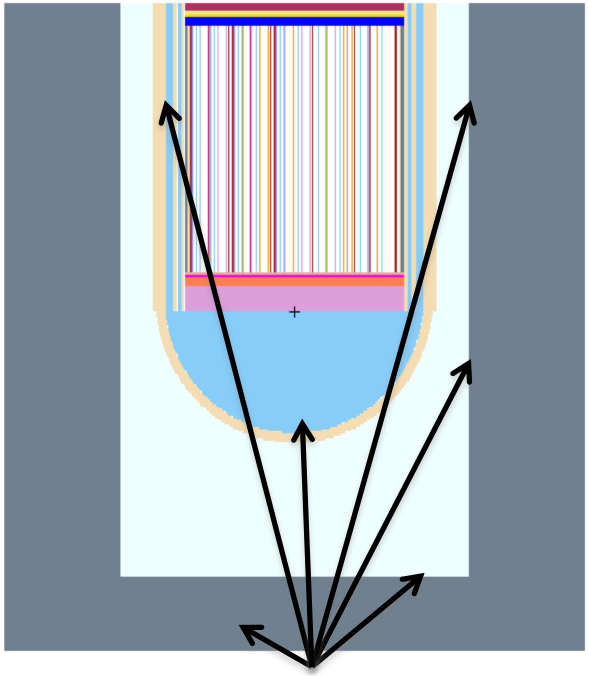
→ To cover the reactor core, **a lot of simulated particles are needed**

→ Outside the core the flux is strongly attenuated: **more simulated particles are needed increasing strongly the calculation time**

To rise this challenge

→ **Variance reduction techniques** are used to increase the statistics for the researched responses

↳ *Collaboration with ENEA, Bologna*



Calculation scheme:

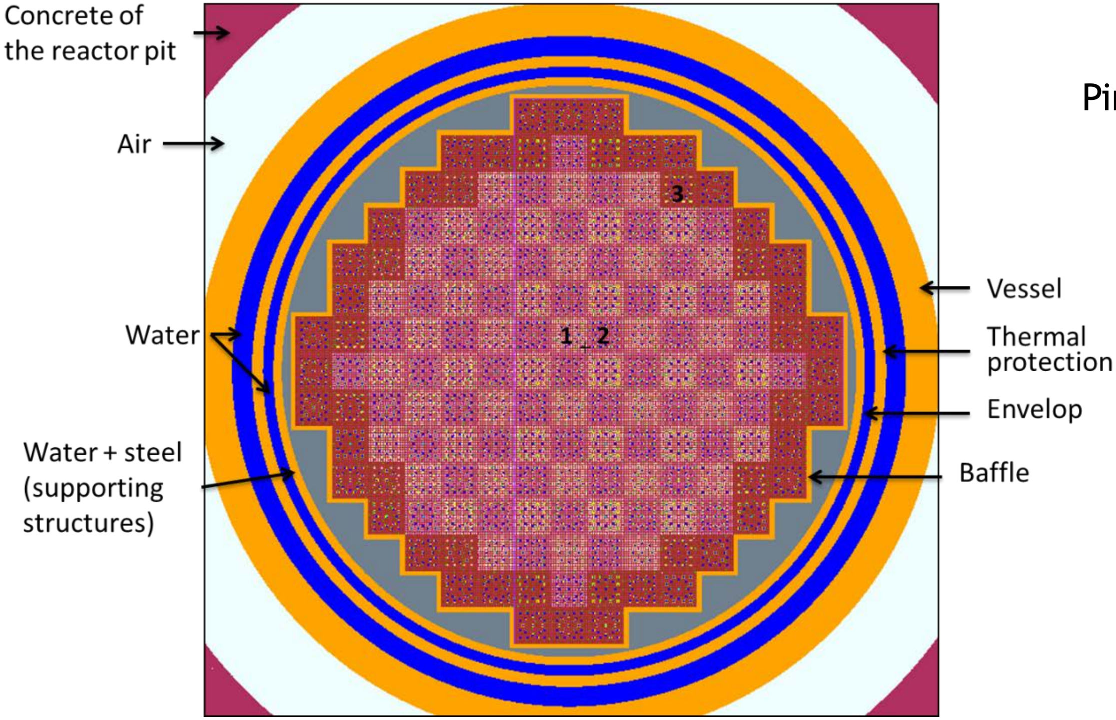
Fission distribution in the core ⇨ Variance reduction parameters optimisation ⇨ Monte Carlo particles transport towards detectors

PWR reactor modelling with MCNP

Core is based on the Tihange-1 reactor in Belgium (public data available, with associated flux in-core distribution measurements, similar to French PWRs)

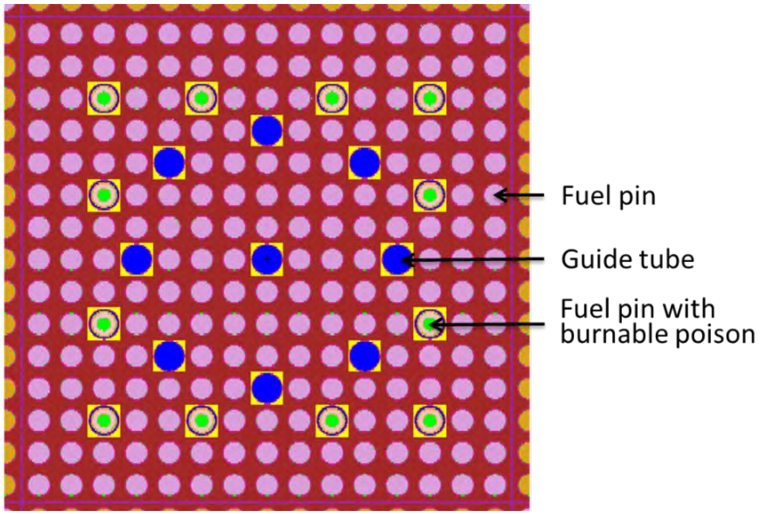
Hot zero power (no thermal gradient)

Fresh fuel (3 types of enrichment with the highest enrichment on the edge of the core)



3 types of fuel enrichment
(1: 2,55 %; 2: 1,95 % and 3: 3,10 %)

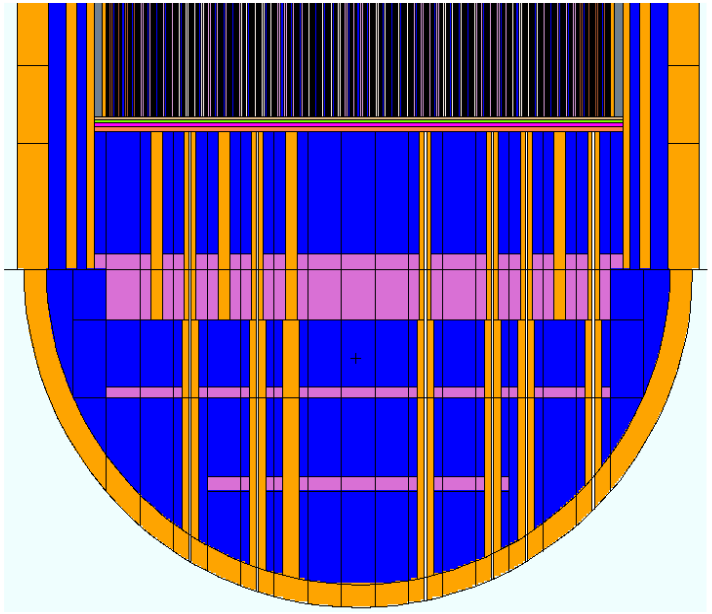
Pin by pin modelling in the MCNP model

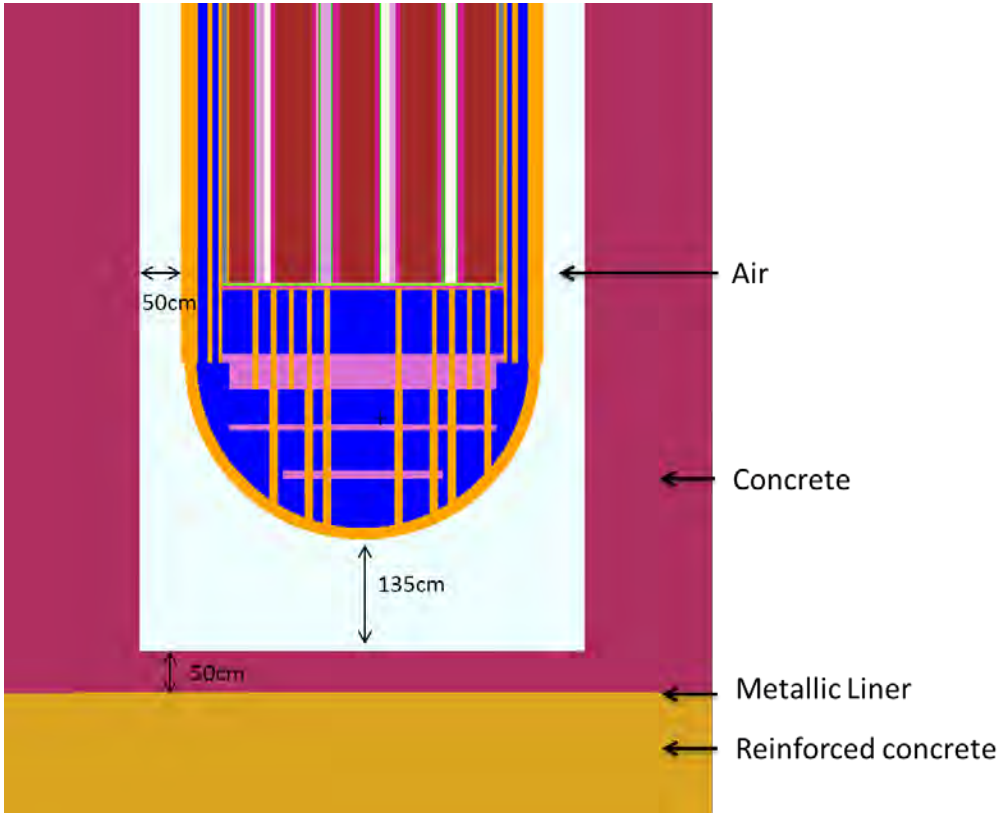


PWR reactor modelling with MCNP

Modelling including structures in the bottom of the vessel and the reactor pit

(supporting structures, piercing instrumentation tubes ...)



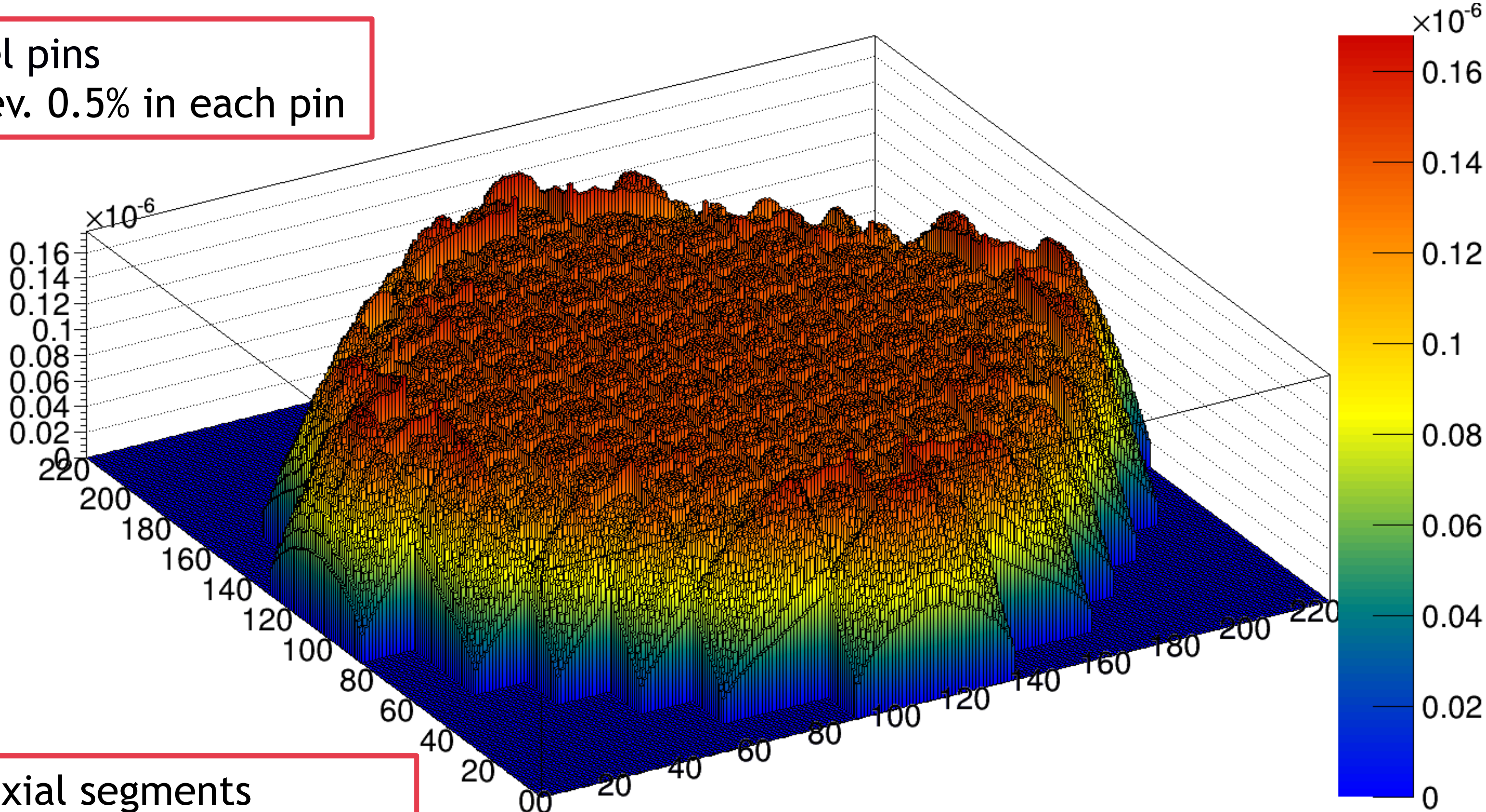


Calculated fission distribution in the core

First step:

- Converge the fission distribution in the core (criticality calculation ~10^{10} particles)
- Tally the fission reaction rate in each pin

≈35 000 fuel pins
max std. dev. 0.5% in each pin



200 axial segments
(unique distribution for the core)

Used nuclear data base ENDF/B-7.1

Comparison of calculated fission distribution to experimental data

Flux distribution measurements available with two separate measurement campaigns by FRAMATOME and by EDF

→ No measurement uncertainties available

Flux distribution in 1/8th of the core:

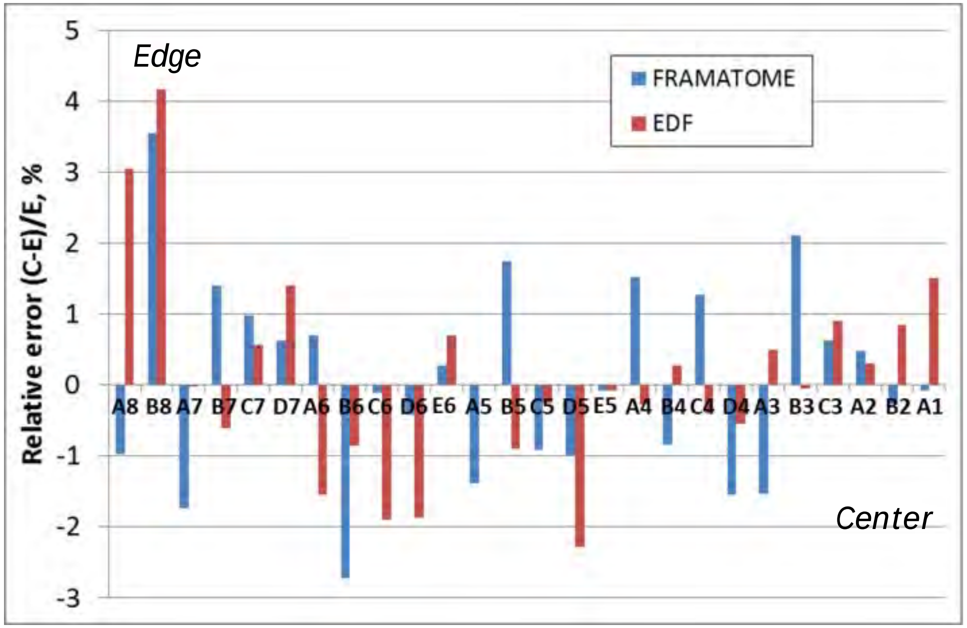
Center

	A	B	C	D	E
1	1.086 1.069 **1.085**				
2	1.087 1.089 **1.092**	1.081 1.069 **1.078**			
3	1.090 1.068 **1.073**	1.066 1.089 **1.088**	1.074 1.071 **1.081**		
4	1.066 1.085 **1.082**	1.078 1.066 **1.069**	1.076 1.094 **1.090**	1.099 1.088 **1.082**	
5	1.074 1.059 **1.059**	1.050 1.078 **1.068**	1.066 1.059 **1.056**	1.075 1.089 **1.064**	0.901 0.901 **0.900**
6	1.049 1.073 **1.056**	1.065 1.045 **1.036**	0.986 1.004 **0.985**	0.940 0.954 **0.936**	0.709 0.706 **0.711**
7	1.033 1.015 **1.015**	1.037 1.058 **1.052**	0.968 0.972 **0.977**	0.648 0.643 **0.652**	
8	0.921 0.885 **0.912**	0.675 0.671 **0.699**			

Edge

Framatome
EDF
MCNP

Comparison of the calculated values:



Highest error at the edge of the core (4%)

C-E : same order as discrepancies between both measurements

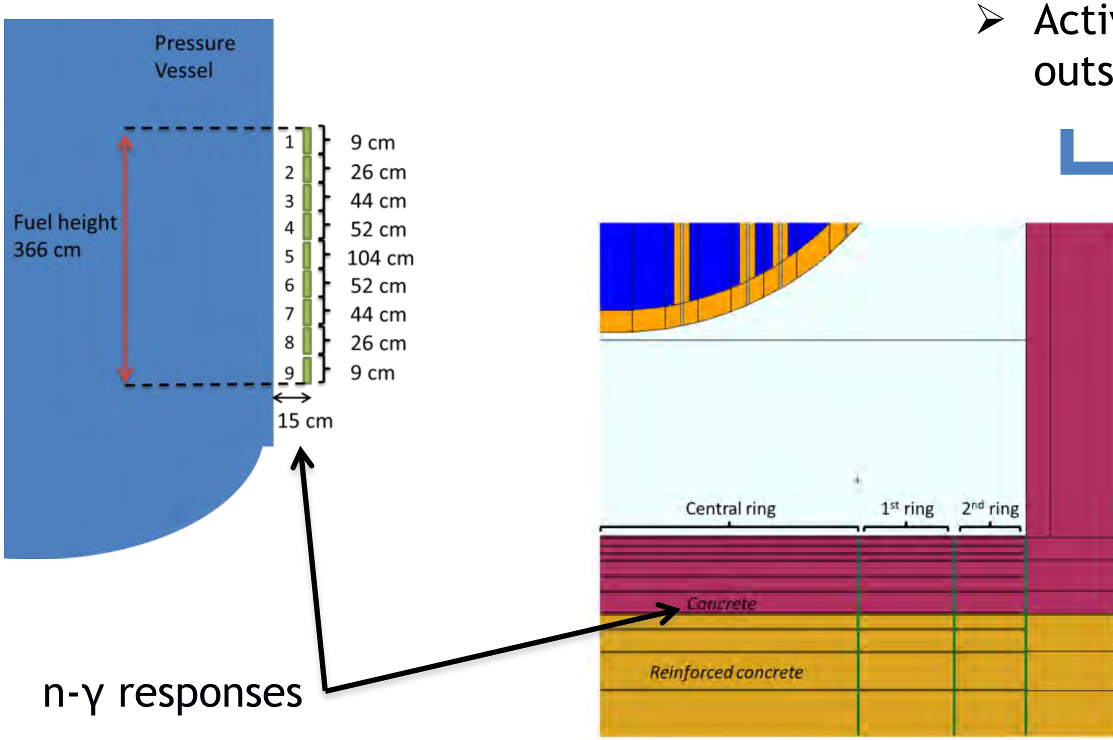
Used nuclear data base ENDF/B-7.1

Neutron and gamma responses

DISCOMS:

OFS
- fast neutron fluence and gamma dose
- possibility to embedded in the concrete

SPNDs
- neutron and gamma flux spectra



Ageing:

- Neutron fluence azimuthal pic on the inner side of the vessel & vessel bottom
- Neutron & gamma fluxes on the concrete
- Activation of internals + structures outside the vessel

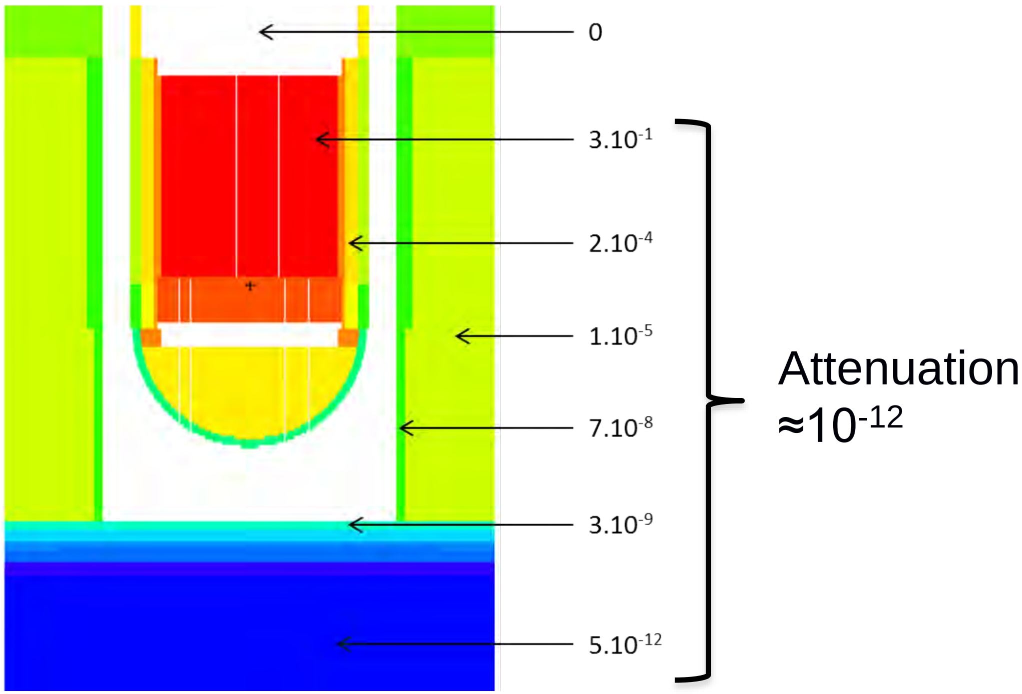
Work still ongoing

Variance reduction parameters

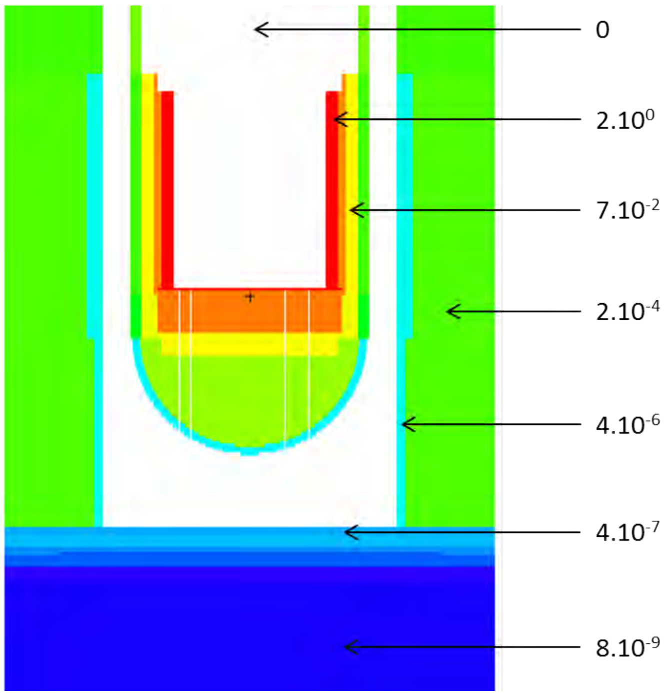
IRSN-ENEA collaboration:

- Kenneth W. Burn (ENEA) calculated the variance reduction parameters using the DSA (Direct Statistical Approach) method optimizing the parameters for the set of the responses of interest: inside and outside the vessel, in the concrete for n and γ.

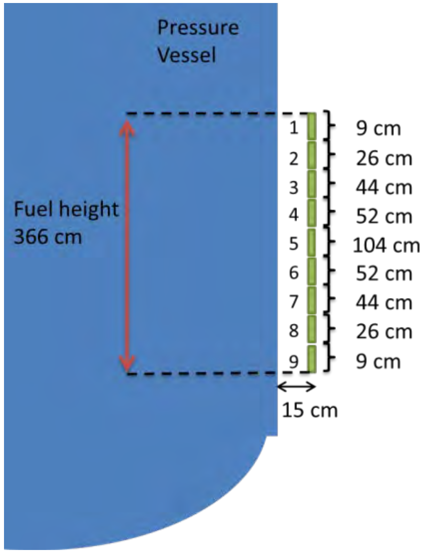
Weights for neutrons of 2-5 MeV (using Weight Windows)

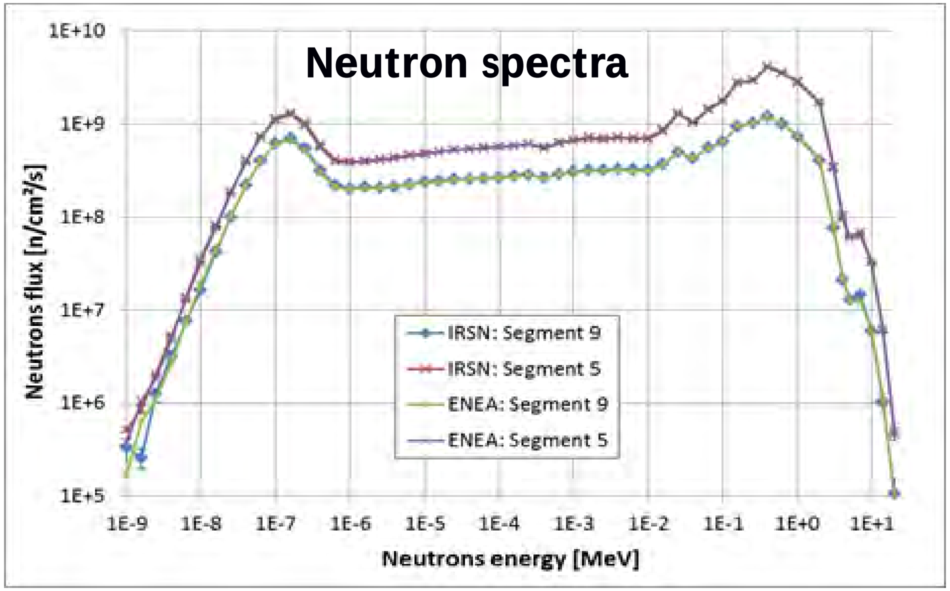


Weights for gamma rays of 0.7-1.5 MeV (using Weight Windows)



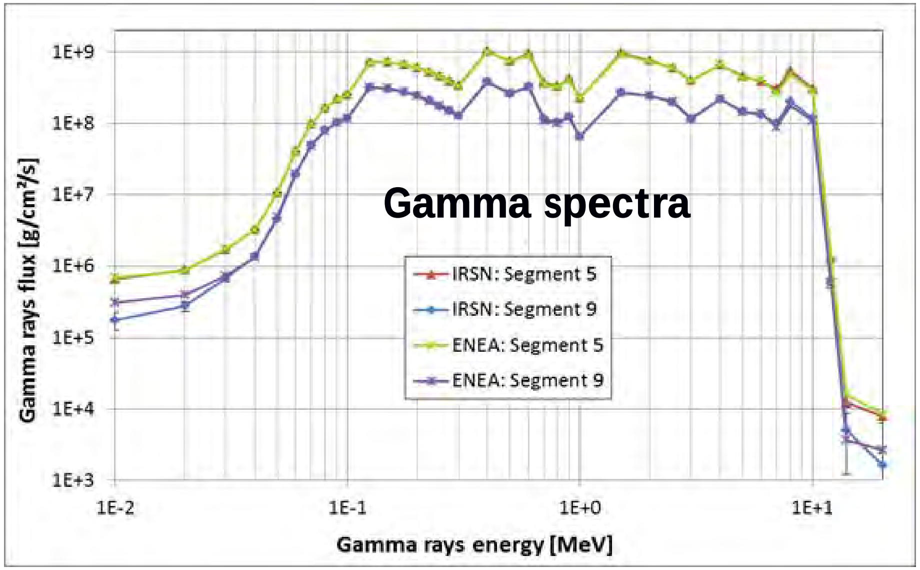
Flux on the side of the vessel





IRSN & ENEA calculations:

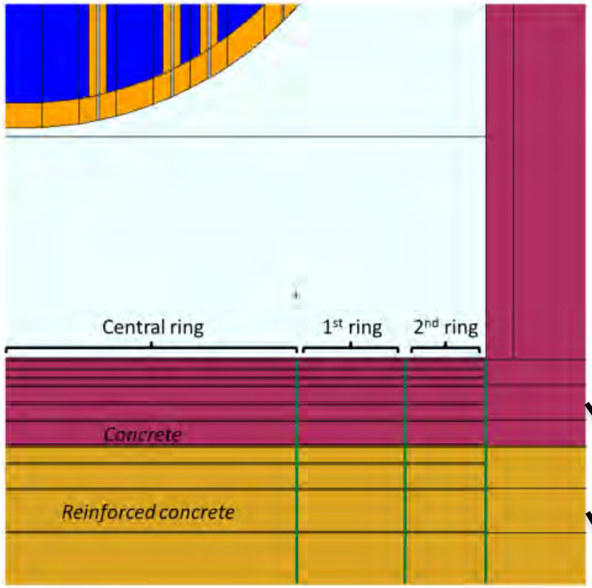
- Use of different source distributions
- Nuclear Data bases
  - IRSN: ENDF/B-VII.I
  - ENEA: JEFF-3.1



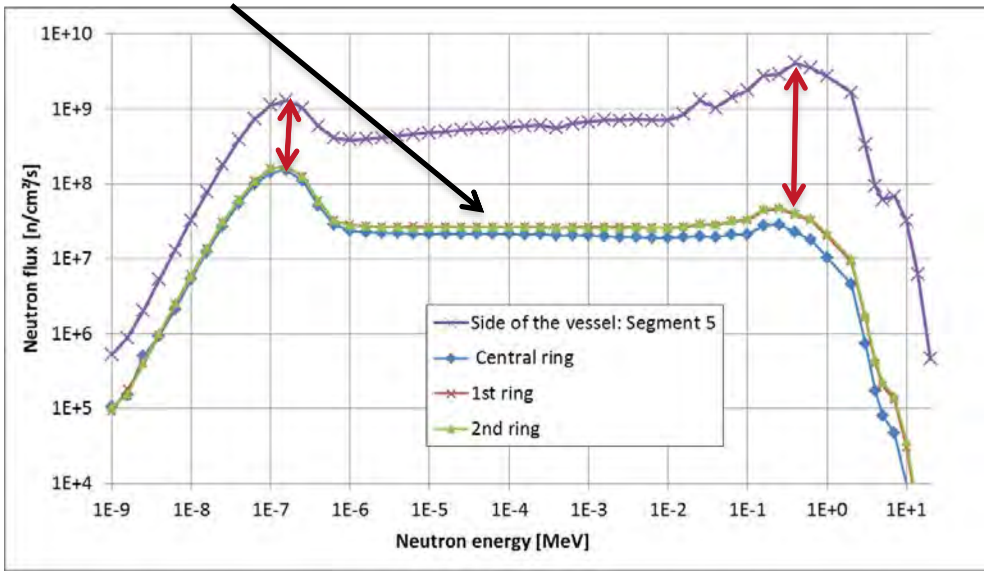
➡ **Good agreement IRSN/ENEA**

➡ **Axial position: no significant impact on n/γ spectra**

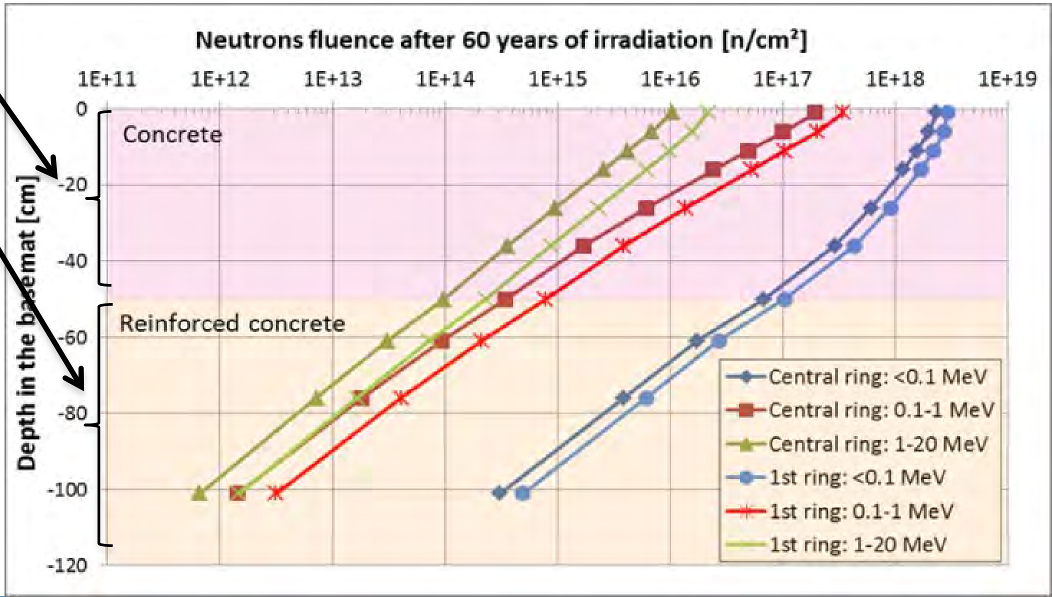
Neutron flux in the basemat



Neutron flux on the surface of the basemat



Neutron fluence attenuation in the concrete



IRSN calculations:

On the basemat surface:

Strong impact of the position

- **Fast neutrons $\times 10^{-2}$**
- **Thermal neutrons $\times 10^{-1}$**

Strong attenuation in the concrete

Sensitivity study

Example:

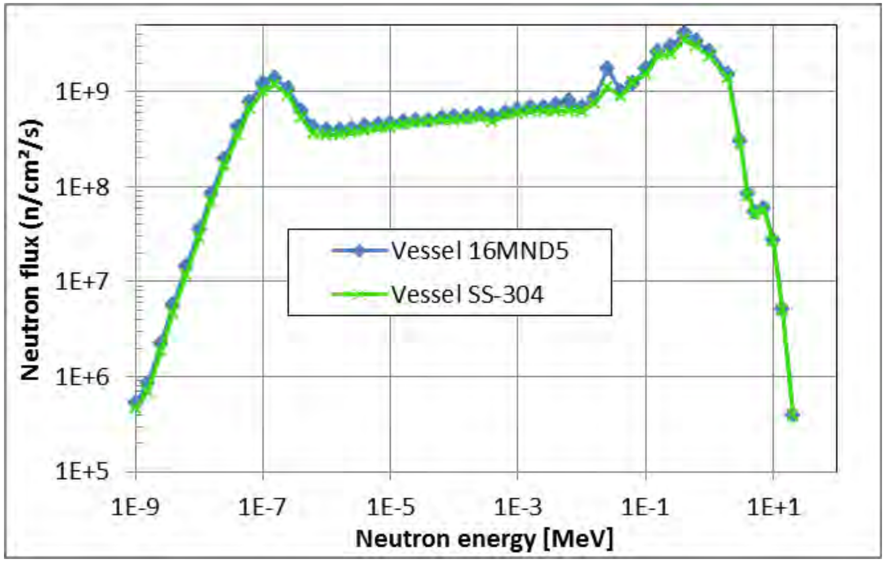
Pressure vessel material replacement

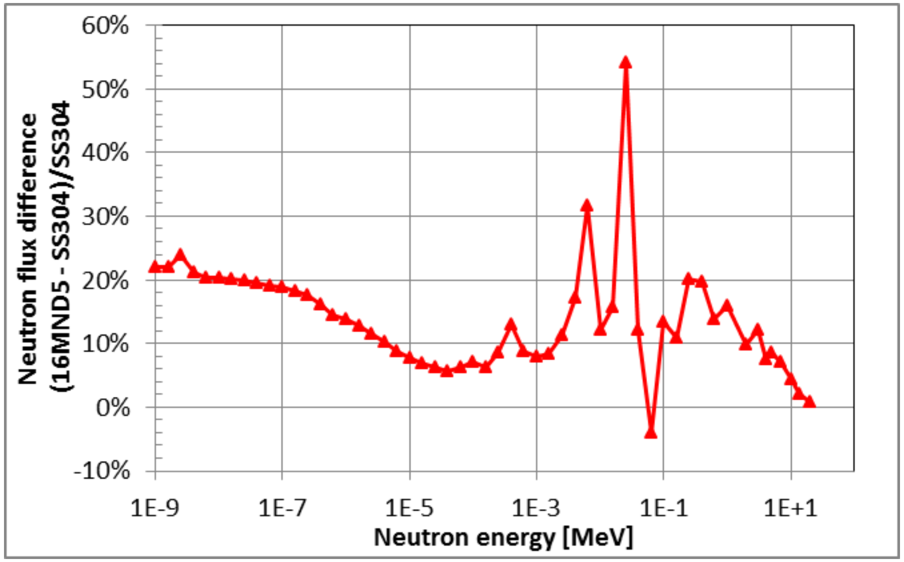
Stainless steel 304 → replaced by 16MND5

Goal: To see the impact of the vessel composition

Main elements	Fe	Cr	Ni	Mn
16MND5	96.8 %	0.3 %	0.7 %	1.4 %
SS-304	66.7 %	20.2 %	11.2 %	2.0 %

Neutron flux outside the pressure vessel





Flux outside the vessel is sensitive to steel composition

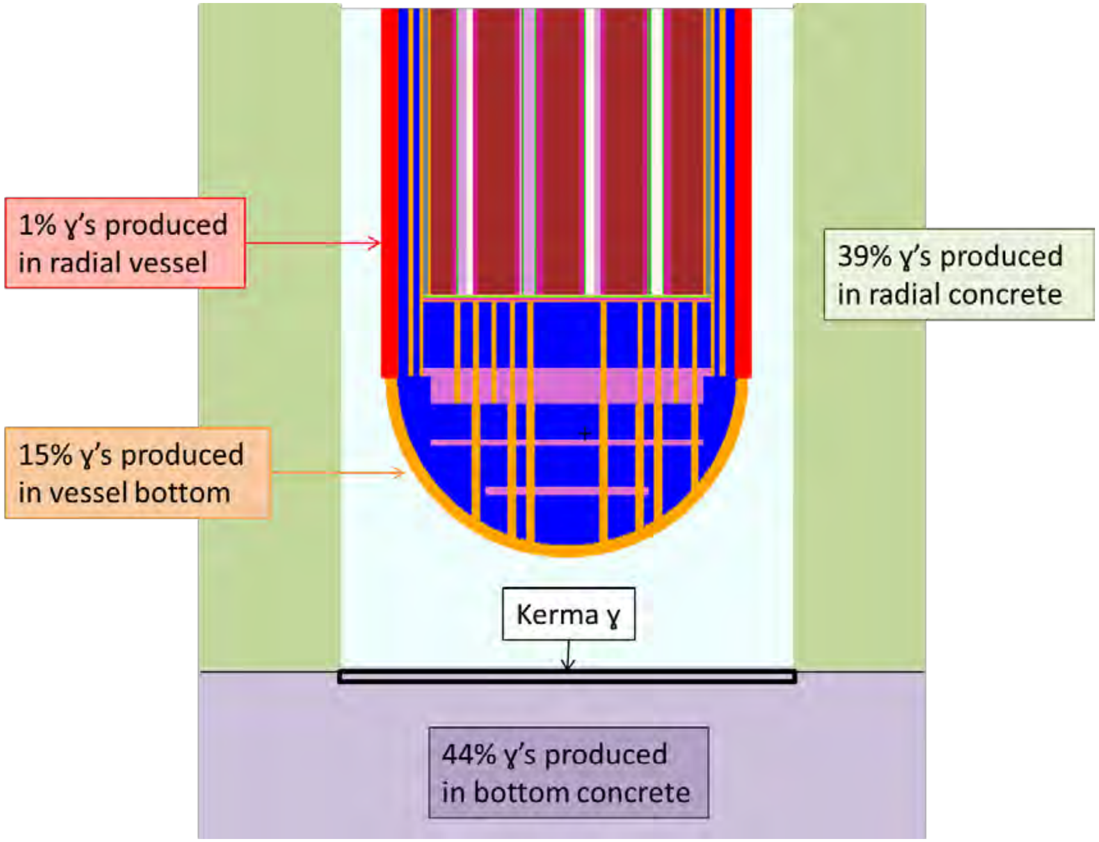
Gamma production analysis

Example:

Studied response: Kerma γ on the basemat

Flagging of cells where the gammas are produced

Goal: To know which materials are important (need to be known precisely)



Gammas are produced mainly in the concrete and only to 16% in the vessel

Radiation calculations for ageing issue & R&D project

- Radiation environment can be characterized with **"pure"** Monte Carlo simulations using variance reduction methods

  - **Time consuming** calculations, but **feasible** (burn-up, fission source distribution, variance reduction parameters and final transport calculations)

  - Needs **good knowledge** of the core and other structures in the reactor

    ⇨ Analysis/Sensitivity study to search for important parameters (compositions, …)

- Ageing issue: different physical quantities can be derived from the neutron/gamma flux calculations as Kerma/Dose, DPA, activation, He/H production, …

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