

Impact of Fission Spectrum Modeling on Aging Estimations of Pressurized Water Reactor Vessels

02/12/2021

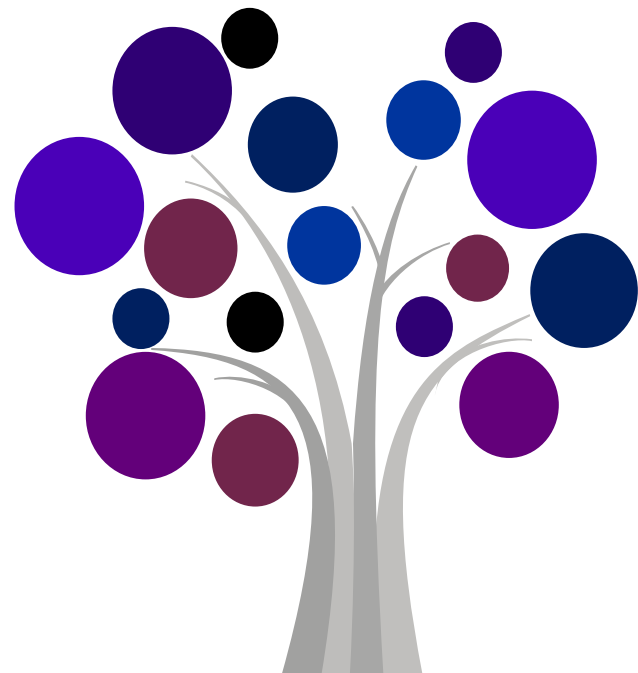
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 **ANS** Winter Meeting & Expo

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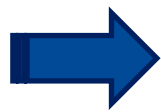


Introduction

Study context

- Extension of PWRs operation duration
 - Vessel = 2nd confinement barrier → essential to accurately characterize its structural integrity
 - Neutron irradiation → embrittlement of the vessel material
- Monitoring of embrittlement : Surveillance Programs
 - Accelerated irradiation of vessel material samples → mechanical strength tests

➤ *Embrittlement = f(vessel composition, $\varphi_{>1\text{ MeV}}$)* *fast neutron fluence*



Limit not to exceed

$$\varphi_{>1\text{ MeV}} = \int_0^T \int_{E>1\text{ MeV}} \phi(t, E) \cdot dE \cdot dt$$

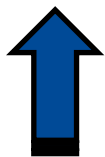
- $\sigma_{\varphi_{>1\text{ MeV}}} \sim 10\% - 20\%$

Reactors are getting closer to the limit → need to better analyse the global bias and uncertainty on $\varphi_{>1\text{ MeV}}$ assessments

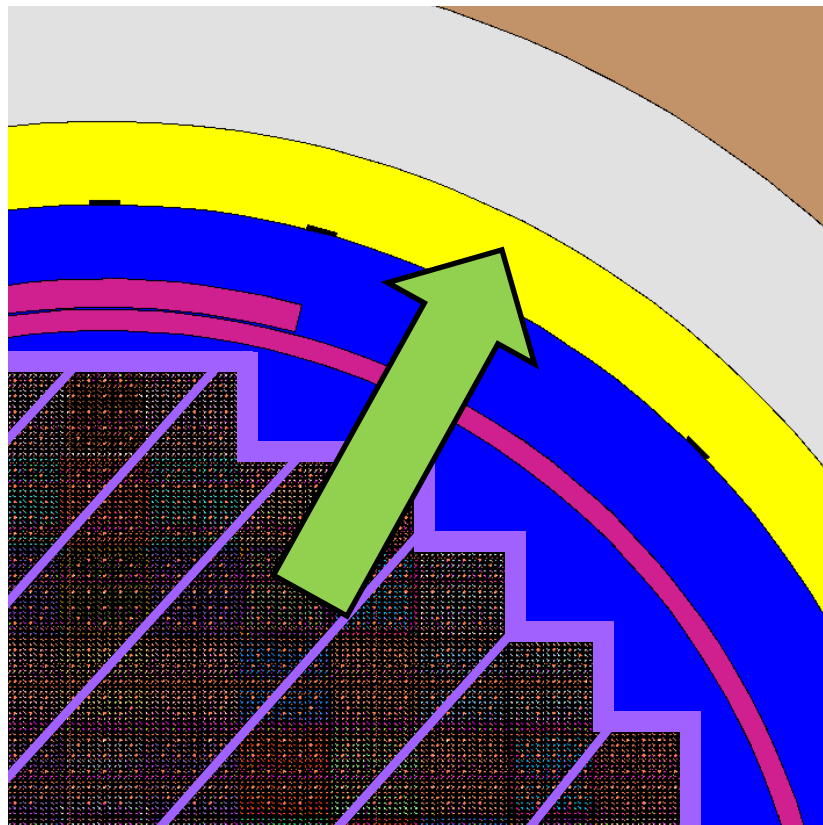
Usual calculation scheme for vessel fluence assessments

First step:
fission neutron
source term
evaluation

Where are the
fission neutrons
emitted and what
is their energy ?



Approximations on
the fission spectrum
modeling



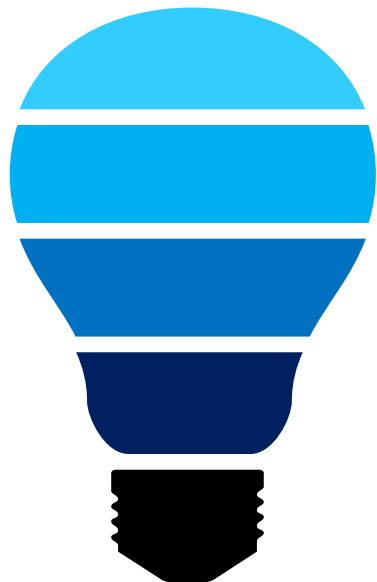
Second step:
modeling of neutron
attenuation from the core
to the sites of interest
(vessel, capsules)

For a quantity of
emitted neutrons, how
many reach the vessel
and what is their
energy ?

Questioning and objectives



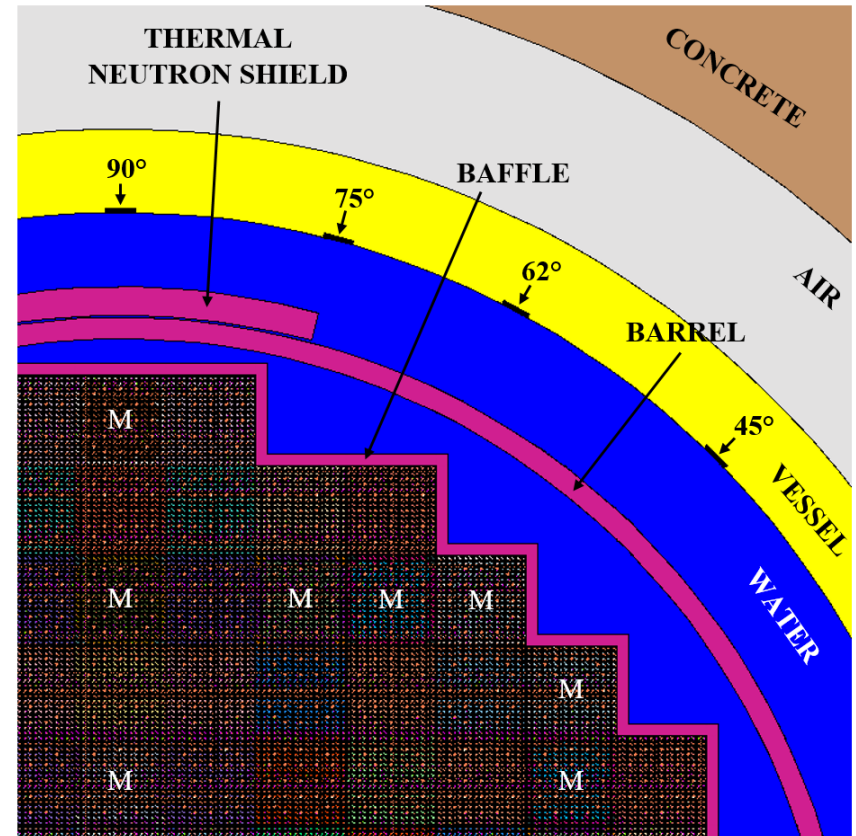
- What is the impact of the modeling approximations made on the fission spectrum in two-step schemes on the accuracy of aging assessments?
- What modeling should be used to limit bias on aging assessments?



Comparison between a one-step reference case and a decoupled approach with several fission spectrum modeling

Case of study

- Representative of a French 900 MW_e PWR
- Fuel composition → end of a UOX/MOX equilibrium cycle
- Several in-vessel detectors*
 - Neutron flux > 1 MeV
 - Neutron flux > 100 keV



* : axially located at the middle of the active height of the core



Modeling and Methods

One-step reference calculation

MCNP6 eigenvalue calculation : **no spatial or energy averaging**

E = energy of incident neutron
 E' = energy of emitted neutron
 i = isotope

$$S(\vec{r}, E') = \sum_i \int_E R_{f,i}(\vec{r}, E) \cdot \nu_i \cdot \chi_i(E, E') \cdot dE$$

Fission rate (points to $R_{f,i}$)
Number of neutrons emitted per fission (points to ν_i)
Fission spectrum (points to χ_i)

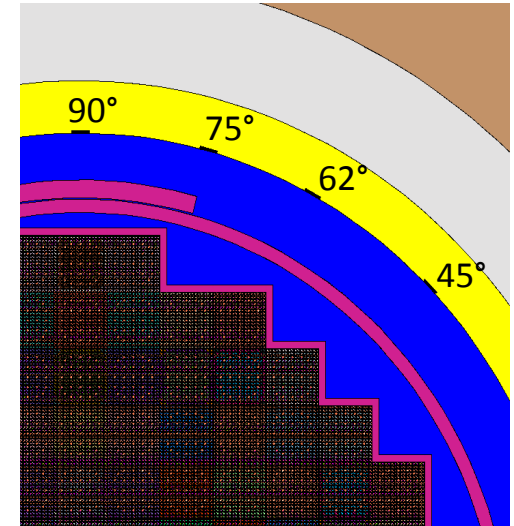
Variance reduction method: Direct Statistical Approach (DSA)

- Based on splitting and Russian roulette
- Optimization of local detectors while maintaining the fundamental mode
- Usage of neutron **superhistories**

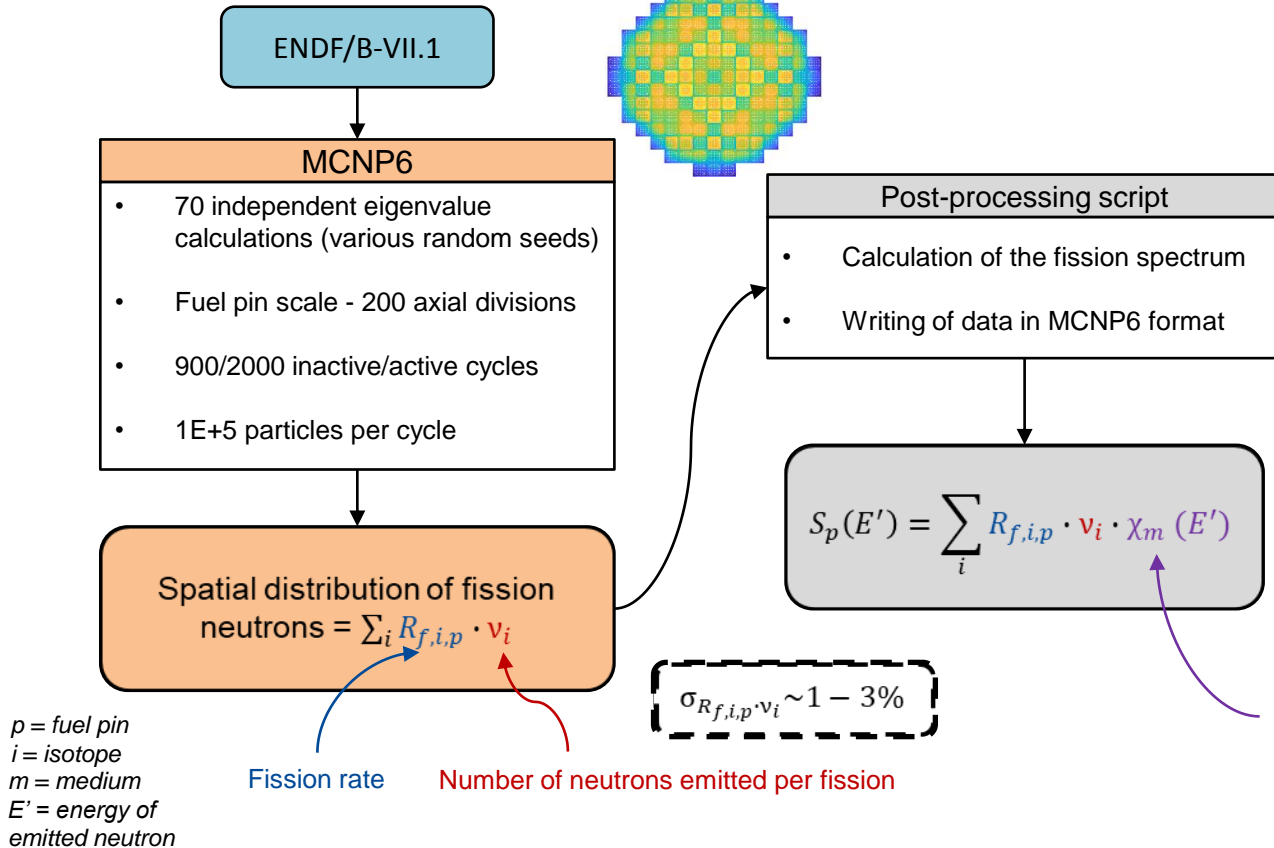
Simulation parameters

- 150 active cycles
- 2E+6 neutrons histories per cycle
- 10 fission generation per superhistory

$$\sigma_{\Phi} < 0,4\%$$



Decoupled approach – first step



Tested scales for averaging

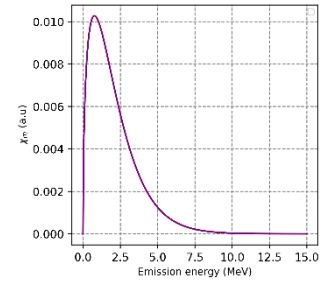
- Reactor
- Fuel assembly

χ_{235U} and χ_{239Pu}
as points of comparison
(reactor scale)

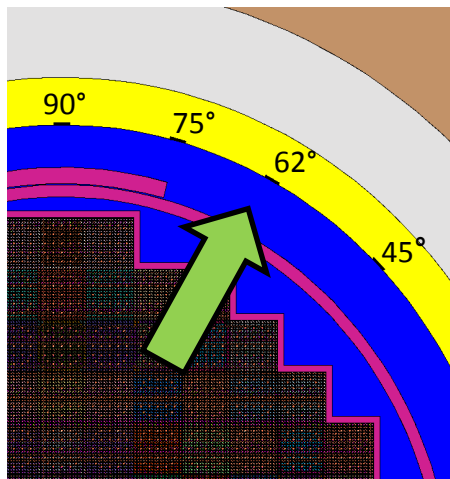
Probability that the neutron emission comes from isotope *i*

$$\chi_m(E') = \frac{\sum_i R_{f,i,m} \cdot \nu_i \cdot \chi_i(E')}{\sum_i R_{f,i,m} \cdot \nu_i}$$

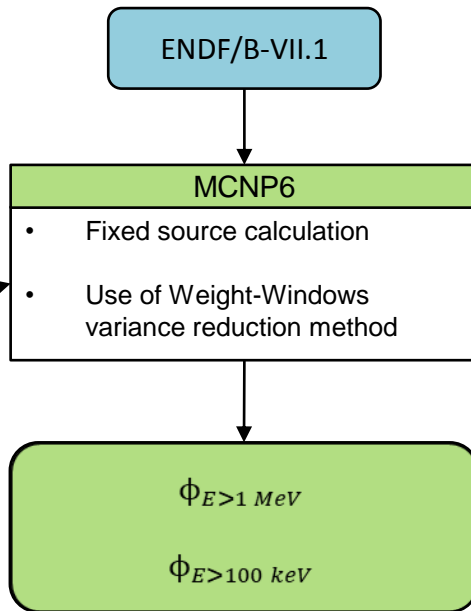
Average fission spectrum of a considered medium



Decoupled approach – second step

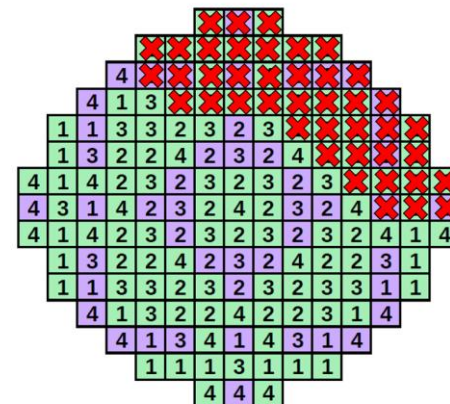


$$S_p(E') = \sum_i R_{f,i,p} \cdot v_i \cdot \chi_m(E')$$



$$\sigma_{\Phi} < 0,5\%$$

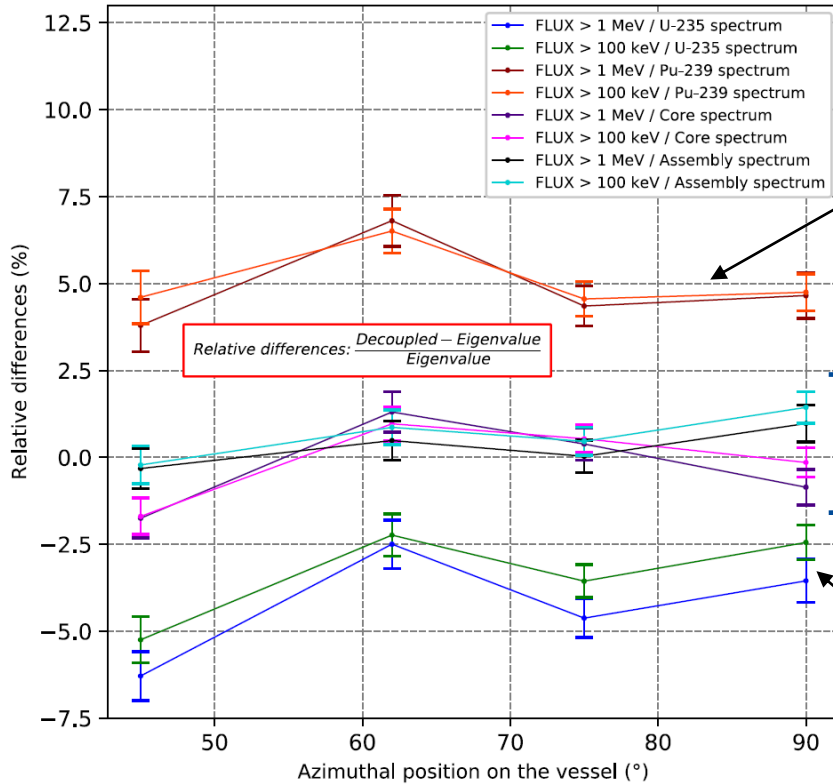
- Reactor scale and isotopic fission spectrum → a single fixed source calculation
- Assembly scale fission spectrum → one fixed source calculation per assembly (for a total of 42)





Results

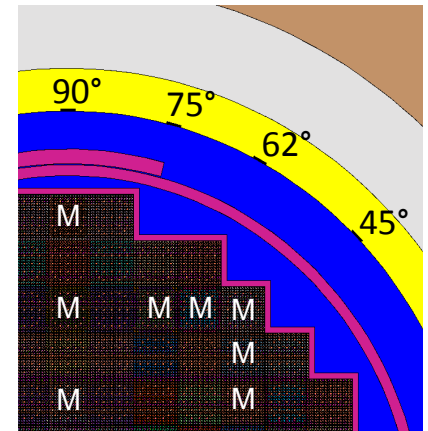
Obtained results



$\chi_{239Pu} \rightarrow$ Overestimation
 $\in [+3,8\% ; +6,8\%]$

Core spectrum $\in [-1,7\% ; +1,3\%]$
 Assembly spectrum $\in [-0,2\% ; +1,4\%]$

$\chi_{235U} \rightarrow$ Underestimation
 $\in [-6,3\% ; -2,2\%]$





Conclusion

Conclusion

- What is the impact of the modeling approximations made on the fission spectrum in two-step schemes on the accuracy of aging assessments?
- What modeling should be used to limit the bias on aging assessments?

Use of isotopic fission spectrum significantly biases the neutron flux estimates

χ_{239Pu} → Overestimation
∈ [+3,8% ; +6,8%]

χ_{235U} → Underestimation
∈ [-6,3% ; -2,2%]

Core scale fission spectrum → differences ≤ 2%

Assembly scale fission spectrum → differences ≤ ±3σ

Recommended for decoupled schemes

Methodology developed at IRSN for reactor vessel aging analysis

R. VUIART et al. (2021) A Versatile Methodology for Reactor Pressure Vessel Aging Assessments, Nuclear Science and Engineering, DOI: [10.1080/00295639.2021.1991761](https://doi.org/10.1080/00295639.2021.1991761)

Capabilities

- neutron flux + displacements per atom (dpa)
- wide axial and azimuthal range in the vessel and capsules of the French surveillance program
- consideration of the history of the reactor



Validity of the fast neutron fluence with respect to dpa



Thank you for your attention

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IRSN

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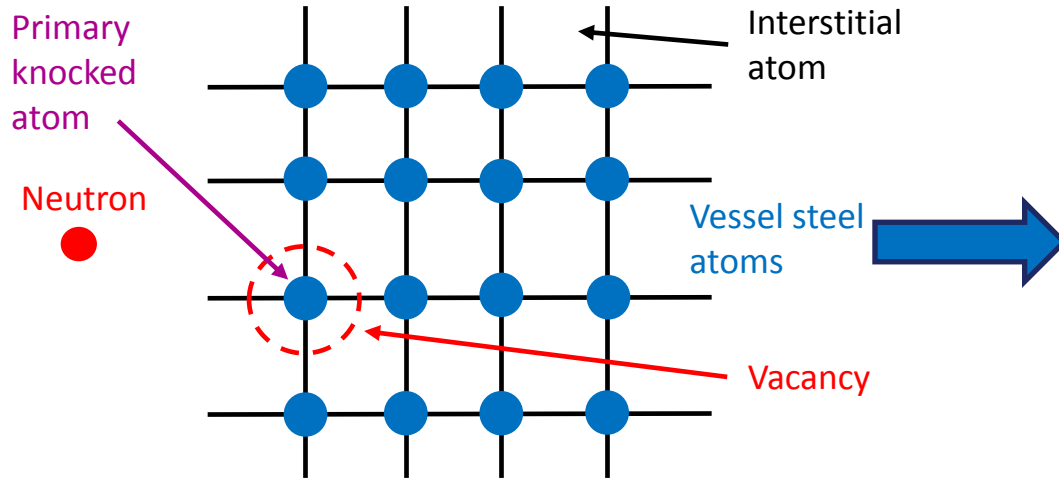
The image features a solid blue background. In the top-left corner, there are two white circles of different sizes. A white line forms a rectangular frame that starts from the top-left corner and extends across the top and down the right side. The text 'BACK-UP' is centered within this frame. In the bottom-right corner, there are two overlapping semi-transparent blue circles of different sizes. The text 'Romain VUIART' is located in the bottom-right corner of the image.

BACK-UP

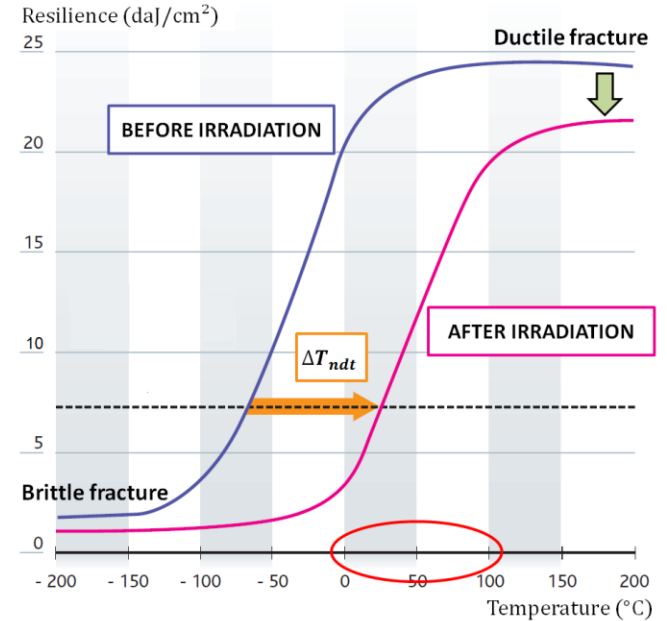
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Effect of neutron irradiation of the RPV material

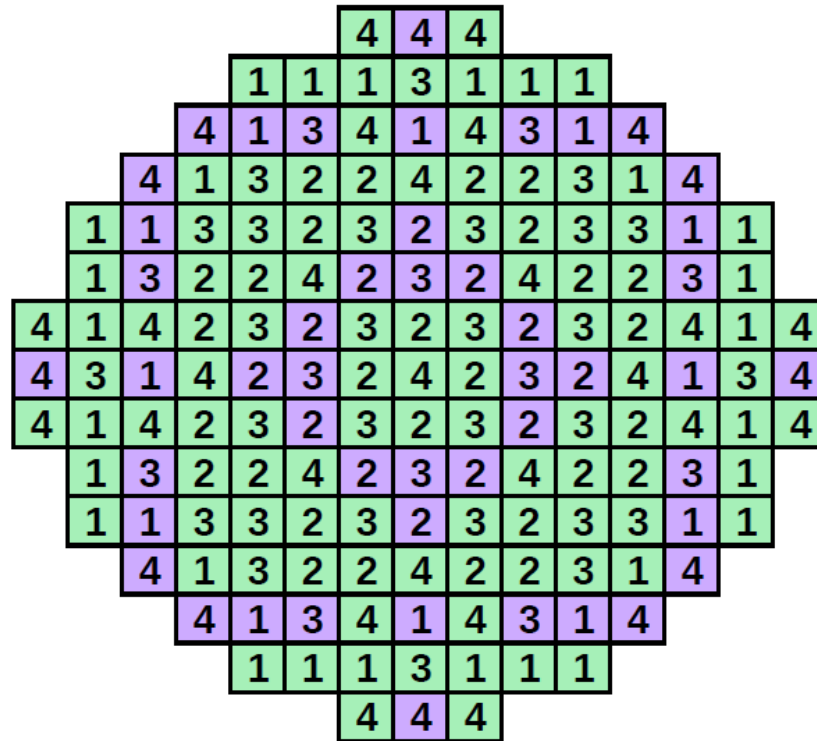
Neutron irradiation → material embrittlement



Recombination time
($t_c < t_f$ few ps)



Fuel loading



 UOX

 MOX

number = irradiation cycle

DSA – Direct Statistical Approach

Basic Idea (Dubi '81):

The DSA theory is based on the possibility to obtain exact explicit expressions for the dependence of the second moment and calculation time on the splitting parameters.

➤ Implemented in MCNP6.1

➤ Automatic optimization of the splitting parameters to maximize the FOM

$$\text{FOM} = 1/(\text{Variance} * \text{Time})$$

➤ Application of DSA method for multi-detectors problem:

▪ Fixed source problem

- K. W. Burn, "Optimizing Monte Carlo to Multiple Responses: the Direct Statistical Approach, 10 Years On," *Nucl. Technol.* **175**, pp.138–145 (2011)
- K. W. Burn, "Extending the Direct Statistical Approach to Include Particle Bifurcation between the Splitting Surfaces," *Nucl. Sci. Eng.*, **119**, 44 (1995)
- K.W. Burn, "Complete Optimization of Space/Energy Cell Importances with the DSA Cell Importance Model," *Ann. Nucl. Energy*, **19–2**, 65 (1992)

▪ Eigenvalue mode

○ Usage of superhistories

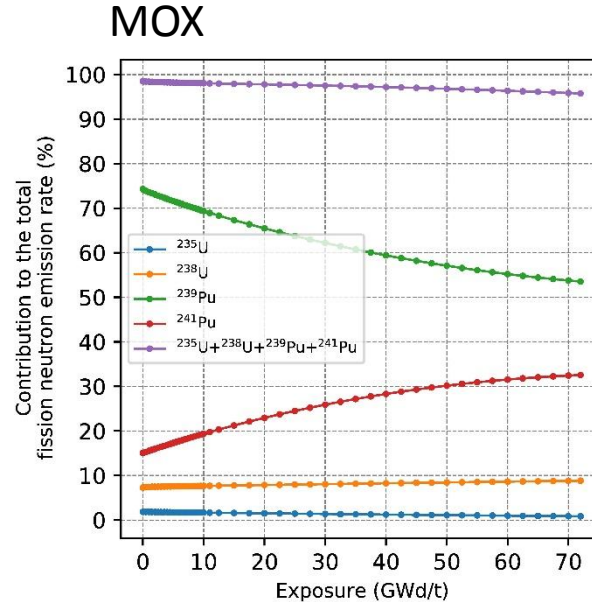
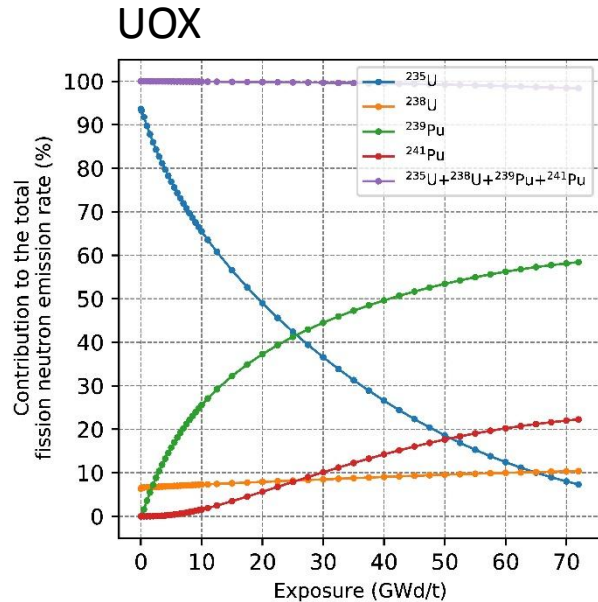
○ Simultaneous optimization of the local detectors and maintaining the fundamental mode

- K.W. Burn, "Optimizing variance reduction in Monte Carlo eigenvalue calculations that employ the source iteration approach," *Ann. Nucl. Energy*, **73**, pp. 218–240 (2014)
- K.W. Burn, "A correction and a clarification to 'Optimizing variance reduction in Monte Carlo eigenvalue calculations that employ the source iteration approach'," *Ann. Nucl. Energy*, **85**, pp. 776–777 (2015)
- K.W. Burn and P. Console Camprini, "Radiation transport out from the reactor core: to decouple or not to decouple," *Proceedings of the ICRS-13*, Paris, Oct. 3–6, 2016, EPJ Web of Conferences **153**, 05007 (2017)
- K.W. Burn and P. Console Camprini, "A Consistent Monte Carlo Treatment of Radiation Responses in and around Critical Configurations," *Proceedings of PHYSOR-2018*, Cancun, April 22 – 26, pp. 576–586 (2018)

Isotopic fission spectrum

TABLE I. Data used for fission spectrum modeling

Nuclide	Incident neutron energy	Fission Spectrum type	a (MeV)	b (MeV ⁻¹)
²³⁵ U	Thermal	Watt	0.988	2.249
²³⁸ U	2.6 MeV	Watt	0.920	3.121
²³⁹ Pu	Thermal	Watt	0.966	2.842
²⁴¹ Pu	Thermal	Maxwell	1.3597	-



[Vuiart et al., NSE, 2021]

Assembly case: Response reconstruction

$$D_{tot} = \sum_a \left(\sum_i R_{f,i,a} \cdot \nu_i \right) / \left(\sum_a \sum_i R_{f,i,a} \cdot \nu_i \right) \cdot D_a \quad (4)$$

$$\leftrightarrow D_{tot} = \sum_a W_a \cdot D_a \quad (5)$$

where, for an isotope i and an assembly a , $R_{f,i,a}$ is the fission rate (s^{-1}), ν_i is the number of neutrons emitted per fission, W_a is the weight of the assembly a and D_a is the detector response for the fixed source calculation of the assembly a .

Then, the uncertainty of this total detector response is evaluated according to Eq. (6). All detector responses calculated obtained with the decoupled scheme have a relative standard deviation inferior to 0.5%.

$$\sigma(D_{tot}) = \sqrt{\sum_a (W_a \cdot \sigma(D_a))^2} \quad (6)$$

where W_a is the weight of the assembly a (defined in Eq. (5)) and $\sigma(D_a)$ is the uncertainty of the detector response for the fixed source calculation of the assembly a .