



HAL
open science

Impact of core power variations on the fast neutron flux incident on pressurized water reactor vessels

Romain Vuiart, Mariya Brovchenko, Julien Taforeau

► To cite this version:

Romain Vuiart, Mariya Brovchenko, Julien Taforeau. Impact of core power variations on the fast neutron flux incident on pressurized water reactor vessels. International Congress on Advances in Nuclear Power Plants (ICAPP 2019), May 2019, JUAN LES PINS, France. hal-02463775

HAL Id: hal-02463775

<https://hal.science/hal-02463775v1>

Submitted on 12 Jun 2020

HAL is a multi-disciplinary open access archive for the deposit and dissemination of scientific research documents, whether they are published or not. The documents may come from teaching and research institutions in France or abroad, or from public or private research centers.

L'archive ouverte pluridisciplinaire **HAL**, est destinée au dépôt et à la diffusion de documents scientifiques de niveau recherche, publiés ou non, émanant des établissements d'enseignement et de recherche français ou étrangers, des laboratoires publics ou privés.

Copyright

Impact of core power variations on the fast neutron flux incident on pressurized water reactor vessels

Romain Vuiart^{1,*}, Mariya Brovchenko¹, Julien Taforeau¹.

¹Institut de Radioprotection et de Sûreté Nucléaire (IRSN), 31 avenue de la Division Leclerc, 92260, Fontenay-aux-Roses, France.

*Corresponding Author, E-mail: romain.vuiart@irsn.fr

Abstract

As many French nuclear 900 MW_e Pressurized Water Reactors (PWRs) approach the end of their design lifetime (40 years), it is of great importance, in the perspective of a possible extension of operation, to be able to accurately characterize the structural integrity of the reactor pressure vessel for safety reasons. Therefore, the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) consider the possibility to develop a best-estimate calculation scheme for the fast neutron fluence (neutrons with energies greater than 1 MeV) at the vessel of PWRs, which is used to predict the material embrittlement. However, before building this type of model, one prior key point is to determine which parameters need to be considered. The aim of this paper is to assess the need to take into account the power history of the reactor core and the induced variation of its operational parameters (core power, boron concentration and control rods insertion) in order to perform fast neutron fluence calculations. The analysis has been performed using a calculation scheme coupling a deterministic approach (CASMO5 and PARCS codes) to evaluate the core fission distribution and a Monte-Carlo modeling (MCNP6 code) to estimate the attenuation of the neutrons source from the core to the vessel.

Keywords: *neutronics, fast neutron fluence, vessel embrittlement, core power variation.*

1. Introduction

The lifetime of a nuclear reactor and its possible extension are strongly related to the aging of the reactor pressure vessel under neutron irradiation. Indeed, the embrittlement of the vessel material is primarily induced by the bombardment of fast neutrons (with energies greater than 1 MeV) [1].

The fast neutron fluence (φ^T) at the reactor vessel accumulated over a period T can be defined as:

$$\varphi^T = \int_0^T \Phi_{fast}(t) \cdot dt = \int_0^T \int_{E>1\text{ MeV}} \Phi(t, E) \cdot dE \cdot dt \quad (1)$$

where t is the time (s), E is the neutron energy (MeV), $\Phi_{fast}(t)$ is the time dependent, energy integrated, flux of neutrons with energies above 1 MeV at the vessel ($n.cm^{-2}.s^{-1}$).

This quantity can be derived from experimental activity measurements of dosimeters inside the reactor vessel or directly evaluated by numerical simulations. Many approaches have been developed worldwide to perform vessel fluence calculations ([2], [3], [4] and [5]). However, in most of these studies, there is no precision on how the power history of the reactor is taken into account and, to our knowledge, there is no public study testing this hypothesis.

A conventional way to compute the fast neutron fluence is first to evaluate the core fission distribution, then to model the neutron transport from the core to the pressure vessel. In this first step a classical approach is to use a fission distribution averaged over a fuel cycle (often based on operational data of the reactor).

In order to evaluate how the vessel flux may be influenced by the power history of the reactor core, the authors performed a sensitivity analysis using a deterministic approach to evaluate instantaneous fission distributions and a Monte-Carlo code to model the neutron attenuation from the core to the vessel.

This choice is motivated by the lower computational time of this methodology compared to a full Monte-Carlo simulation. Despite the approximations due to deterministic methods, the precision of this model is in line with our needs since the aim of this study is to evaluate the order of magnitude of the impact of core power variations on the vessel flux.

After a brief presentation of the considered reactor core, the following section introduces the modeling used to perform the analysis. Afterward, section 3 details the methodology used to evaluate the impact of core power history on the fast neutron flux at the vessel. Results are presented in section 4 and after a discussion in section 5, conclusions are drawn in section 6.

2. Modeling

Fission distributions are evaluated using a deterministic approach coupling the CASMO5 [6] and PARCS [7] codes and the vessel fast neutron flux $\Phi_{fast}(t)$ is then calculated using the Monte-Carlo code MCNP6 [8]. The second calculation is a fixed source problem using neutron source description derived from the fission distribution evaluated in the first step by the deterministic approach. However, before going into the details of the used calculation scheme, the considered reactor core is briefly presented.

2.1) Reactor core description

The reactor core is representative of a French 900 MW_e PWR core with a hybrid UOX/MOX fuel management. The reactor core layout is shown in **Figure 1** and each fuel assembly is referenced in the classical naval coordinates system (ex: J01 corresponds to the assembly located in the J column, first line). In this figure, colors are just there for the sake of display.

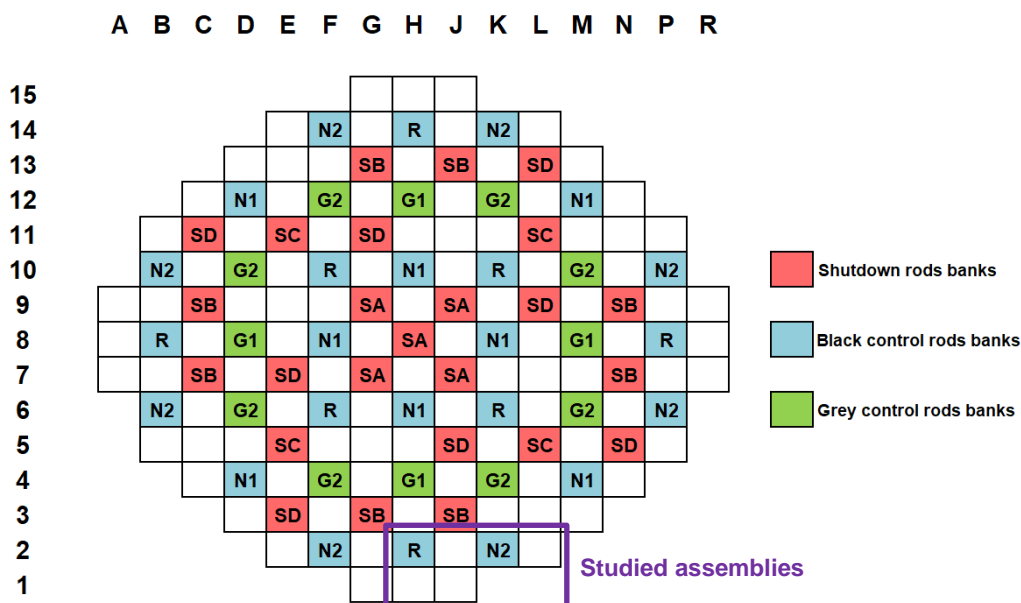


Figure 1: Control rods banks and shutdown rods banks positions [9]

For this reactor type, the control rods are composed of a mixture of silver, indium and cadmium. These are separated in 2 grey control rod banks (G1 and G2), which have a low impact on reactivity and 3 black control rod banks (R, N1 and N2) which are stronger neutron absorbers. Furthermore, the considered control rods management is the French “G” mode [9] that is widely used in the French nuclear fleet. For this management the R control rod bank has continuously, during the cycle, few steps inserted to regulate the coolant temperature while the G1, G2, N1 and N2 banks are progressively inserted (in that order) as the core power decreases. There are also 4 shutdown rod banks (SA, SB, SC and SD) but these are permanently withdrawn when the reactor is operating.

2.2) Neutron source description

The spatially homogenized neutron source within a fuel assembly m can be defined as:

$$S_m(E) = R_{f,m} \cdot \nu_m \cdot \chi_m(E) = \frac{P_m}{E_{Rm}} \cdot \nu_m \cdot \chi_m(E) \quad (2)$$

where $R_{f,m}$ is the fission reaction rate (s^{-1}), P_m is the radial thermal power (W), ν_m is the average number of neutron emitted by a fission, E_{Rm} is the mean recoverable energy per fission (J), E is the energy of the emitted neutron (eV) and $\chi_m(E)$ is the average fission spectrum.

However, the pin neutron source used in the MCNP6 modeling is deduced from the assembly power distribution according to the following equation:

$$S_p(E) = R_{f,p} \cdot \chi(E) = \frac{P_m}{N_{pin}} \cdot \chi(E) \quad (3)$$

where $S_p(E)$ is the homogeneous neutron source in pin p , $R_{f,p}$ is the fission reaction rate for the pin p , N_{pin} is the number of fuel pins and $\chi(E)$ is the fission spectrum for the whole reactor core. In this study, a Watt fission spectrum has been used with values of 0.965 MeV for the a coefficient and 2.29 MeV⁻¹ for the b coefficient [10].

In equation (3) the fission reaction rate distribution ($R_{f,p}$) is approximated by the core power distribution calculated by the CASMO5-PARCS scheme (P_m/N_{pin}). Furthermore, the authors assumed that the average number of neutrons emitted by a fission (ν_m) and the recoverable energy per fission (E_{Rm}) are not varying from one assembly to another by setting ν_m/E_{Rm} equal to 1. This approximation should not have an important impact on the estimated relative differences since only cores with the same average burnup are compared in this study. Moreover, in the used modeling, the source distribution has no pin dependency but takes into account an axial profile for each assembly (generated by the PARCS code).

2.3) Evaluation of the core power distribution

In order to estimate the core power distribution (i.e each individual P_m) at different burnup steps covering the whole cycle duration, a deterministic approach based on CASMO5 and PARCS has been used.

First of all, the CASMO5 code is used to generate fuel assembly homogenized two-group macroscopic cross-section data that the PARCS code will use as a multi-parametric database (since these cross sections depend on energy group, moderator density...) to compute 3D core power distributions at multiple steps of the cycle duration (10.43 GWd/t).

On one hand, CASMO5 solves the 2D stationary transport equation (the axial dimension being considered as infinite) at an assembly level using a 586-group data library based on the ENDF/B-VII.1 nuclear data. The CASMO5 calculation scheme is a two-level one using the so called P_{ij} method [11] with 586 energy groups and the MOC method [11] with 26 energy groups. On the other hand, PARCS solves the 3D diffusion equation with 2 energy groups for the whole core and for different burnup steps to evaluate several quantities (radial and axial power, burnup, xenon and samarium concentration...). Regarding the spatial discretization, each assembly is divided into 4 radial meshes and 14 axial meshes along its height (including reflectors). An advantage of using PARCS regarding Monte-Carlo methods is that it includes a module that takes into account simplified thermohydraulic feedbacks. Xenon-135 and Samarium-149 concentrations are calculated to be at equilibrium, meaning that Xenon-135 concentration is constant between two burnup steps while the Samarium-149 one is fixed for the whole cycle. This

choice is motivated to prevent any Xenon instability that may appear after a strong power variation. Furthermore, an equilibrium cycle has been considered and the cycle duration has been divided in 150 burnup steps so the authors assumed that this approximation is not impacting significantly the obtained results. Finally, boron concentration is adjusted by PARCS to have critical core.

2.4) Modeling of the neutron attenuation

The geometry used in the MCNP6 model (**Figure 2**) is representative of a French 900MW_e reactor and a previous version has been detailed in Reference [12]. However, some modifications have been made on this model, including a modification of the thermal shield from a symmetric to an asymmetric configuration and the addition of a simplified modeling of the capsules used in the French surveillance program [13] containing dosimeters and vessel material samples. Fuel compositions are representative of the beginning of cycle for the studied fuel management (described in section 3.1). Furthermore, the moderator temperature has been set equal to 304.6°C, meaning that moderator density is uniform over the whole reactor core. According to Reference [2], this approximation may have a non-negligible (around 3.5%) impact on Φ_{fast} evaluations but should not impact too much relative differences between two calculations. However, in reality, the axial temperature gradient varies according to the core power and using the same planar-averaged temperature profile for two simulations at different core power may have some impact on the evaluated relative differences. This matter is discussed in section 6. Moreover, the boron concentration has been fixed to 10 ppm since preliminary studies [12] demonstrated that its impact is negligible for a Φ_{fast} estimation because boron-10 is a thermal absorber and has almost no effect on fast neutrons.

Fast neutron flux detectors (1 MeV < E < 20 MeV) have been modeled at different azimuthal positions on the vessel, axially located at the center of the active part, since this axial position corresponds to the maximum flux at the beginning of the cycle. The detectors are indicated by black sections in **Figure 2**.

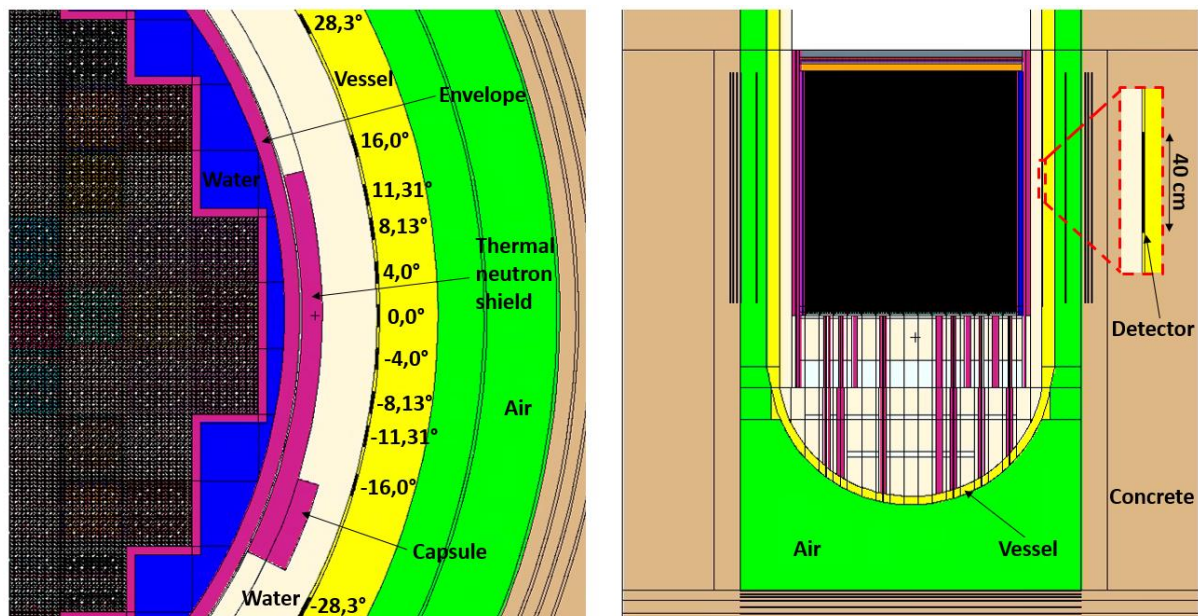


Figure 2 : Horizontal cross section of the MCNP6 reactor modeling with the considered detectors (left side) and vertical cut of the modeling (right side)

To reach sufficient statistical uncertainties with a reasonable computational cost, the weight-window variance reduction method has been used. The weight-window parameters were calculated for 19 neutron energy groups (with energy bounds up to 17.3 MeV) by the mean of the ADVANTG 3.0 [14] tool, based on the so called Forward Weighted Cadis method [15], using the deterministic code Denovo [16]. An example of such weight windows is presented for the second energy group (with 1.00 MeV < E < 1.35 MeV) in **Figure 3**. The high importance regions shown in blue close to the detectors (indicated with red dots) have lower weight-window parameters ensuring enhanced sampling.

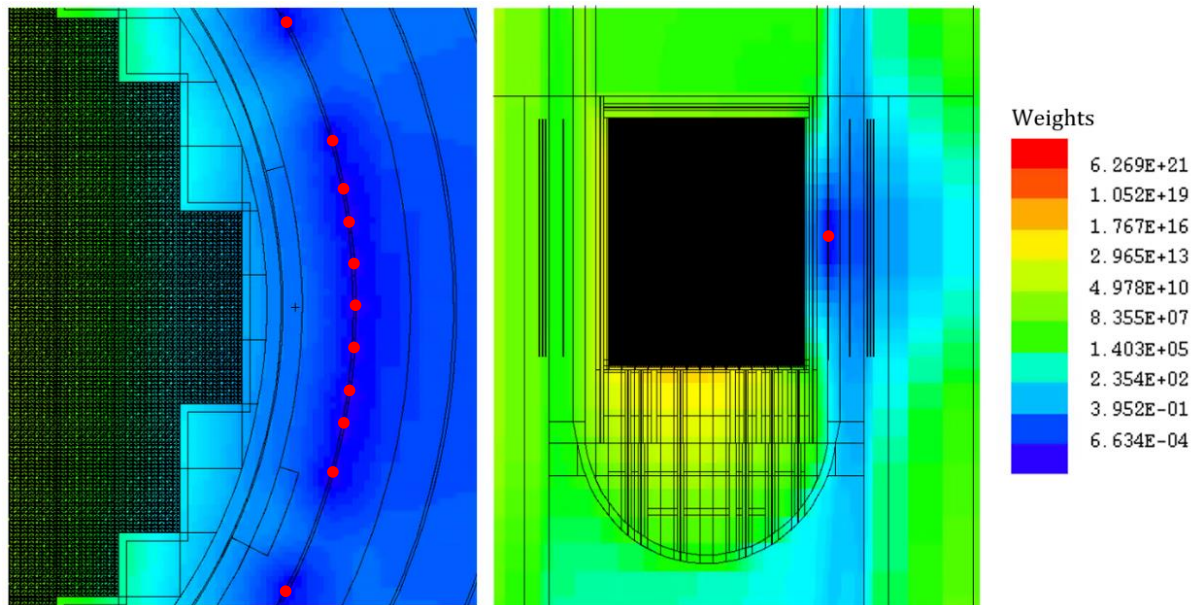


Figure 3 : Weight-window map for the neutrons between 1.00 and 1.35 MeV and red dots indicating detectors location (horizontal (left side) and vertical (right side) cross section)

Finally, Φ_{fast} calculations have been performed using data library based on the ENDF-B/VII.1 nuclear data and $4 \cdot 10^8$ source neutrons have been simulated, leading to statistical uncertainties about 1-2%.

3. Methodology to evaluate the impact of core power history

As mentioned previously, the goal of this work is to quantify the impact of core power variations on the fast neutron flux reaching the vessel. Since the second step of vessel flux evaluation can be time consuming, a more extensive sensitivity study was performed focusing on the energy released by the assemblies on the border of the core during the whole cycle. Indeed, the assemblies in the first two rows (**Figure 1**) are the main contributors to the flux at the 0° detector (**Figure 2**). After analysis of the effect of core power variations on the assemblies of interest, some selected power distribution maps were used for fast neutron flux calculations at the vessel to quantify the impact of core power history.

3.1) Energy released by the border assemblies

As described before (section 2.3), the cycle length has been discretized in N_{step} burnup steps. The energy released per units of heavy metal mass E_m (which is equivalent to a burnup exposure) during a complete cycle by a fuel assembly m has been computed using the following formula:

$$E_m = \sum_{i=1}^{N_{step}} P_{m,i} \cdot \Delta t_i \quad (4)$$

where $P_{m,i}$ is the normalized radial power of the assembly m at the burnup step i (which corresponds to the total power over the assembly) and Δt_i is the length of the burnup step i (GWd/t). In the PARCS code, $P_{m,i}$ is normalized to the average fuel assembly power.

In order to quantify the impact of core power variations on the energy released by the considered fuel assemblies, three different cases with the same final core burnup have been compared (**Figure 4**):

- case “variable power”: a set of simulations where the reactor core experiences a significant number of power drops corresponding thus to different cycle mean power values (for each cycle mean power, several simulations with different number and depths of power drops have been considered, within an uncertainty of 1% on the cycle mean power),
- case “constant power”: a set of simulations where the reactor core power is constant during the whole cycle length (for each cycle mean power corresponds only one simulation),
- case “nominal power”: a simulation where the reactor core is at nominal power with all the control rods withdrawn.

Notes: in cases “variable power” and “constant power”, reactor core operational parameters (control rods) vary according to the core power as described in section 3.1. The cycle mean powers for the “variable power” simulations vary between approximately 30% and 90% of the core nominal power.

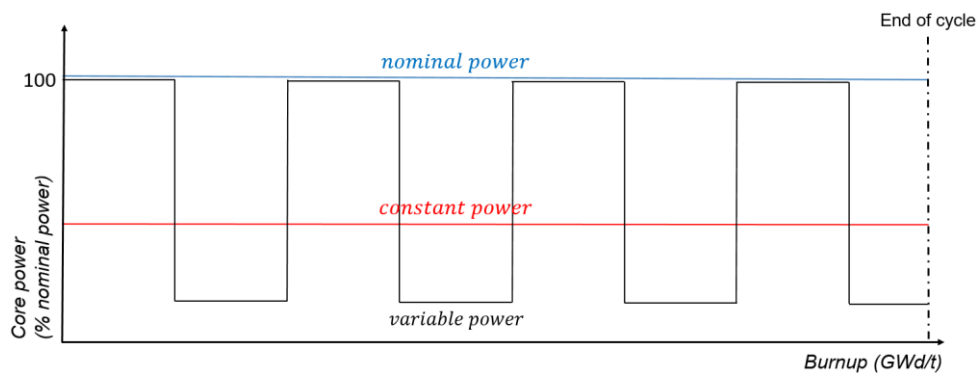


Figure 4 : Scheme of the various tested cases of reactor power variation

3.2) Fast neutron flux at the vessel

In order to quantify the impact of core power variations on the fast neutron flux at the vessel, a study has been performed using the MCNP6 model described in section 2.4. Power distributions were extracted from the “constant power” case described in section 3.1 and converted into a pin neutron source for the MCNP6 modeling using the methodology described in section 2.2. This methodology implies that every pin of an assembly has the same radial and axial power distribution. Two core power maps have been compared, both at a core burnup equal to 2.03 GWd/t. This burnup step was chosen to be close to the cycle beginning while avoiding any bias that may be introduced on the core power distribution by Xenon-135 transient. The first power map (referred to as “32%”) corresponds to a reactor core operating at 32% of its nominal power and the other one (referred to as “80%”) to a reactor core operating at 80% of its nominal power. The relative assembly radial power differences ΔP_{32-80} (cf. equation (5)) between these two maps are presented in **Figure 5**.

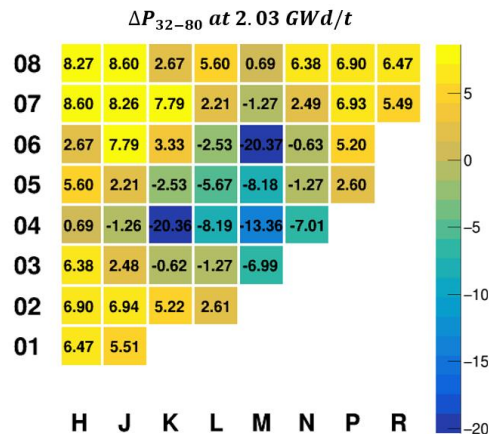


Figure 5: Relative core power distribution differences (%) between case 32% and 80% at 2.03 GWd/t

4. Results

4.1) Impact on the energy released by the border assemblies

The **Figure 6** presents the obtained results for the J01 assembly. This figure highlights the fact that, for the most realistic case in regards to a reactor core operating in load following mode (“variable power” represented by black dots), the energy released increases linearly as the core cycle mean power decreases. Furthermore, dispersion appears around this linear behavior induced by the variation of the number and depths of power drops during the cycle. However, since the maximal dispersion for a given value of the cycle mean power is about 1%, a linear regression constitutes a good model to reproduce the behavior of the “variable power” case. Nevertheless, the generation of such a model would be rather time-consuming and will have to be reproduced for each core management.

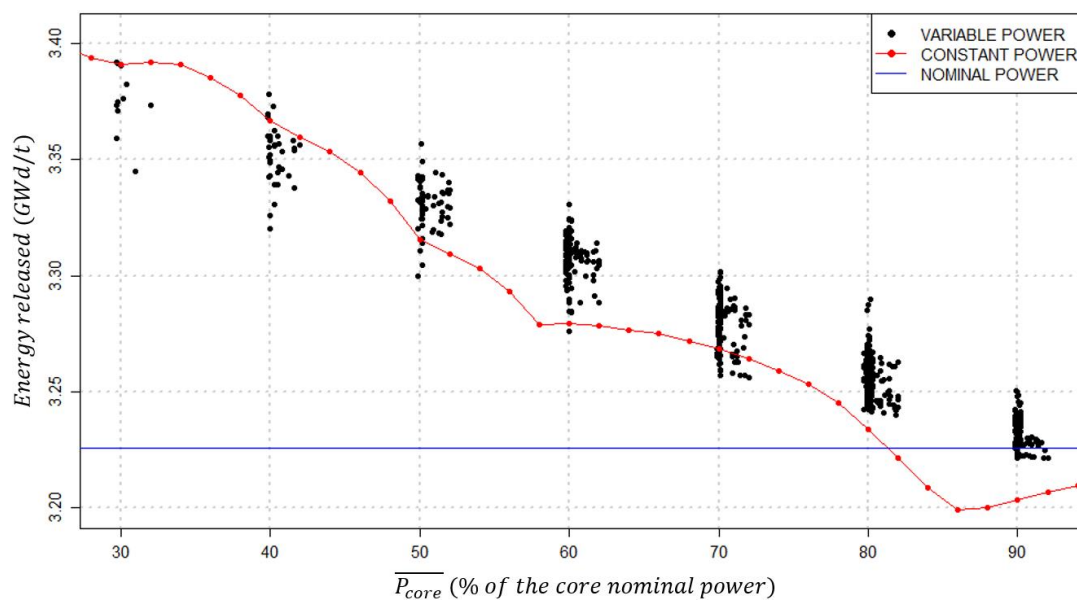


Figure 6: Energy released by the J01 assembly as a function of cycle mean power¹

Results are summarized in **Table 1**. In this table, all calculations were performed using the following relative difference formula:

$$\Delta X_{ref-test} = \frac{X_{ref} - X_{test}}{X_{test}} \quad (5)$$

where X_{test} is the evaluated quantity for the test case and X_{ref} is the evaluated quantity for the reference case.

In this table, the averaged absolute bias to the “variable power” case $|\overline{\Delta E_{VAR-model}}|$ is an indicator of the model quality since a low value of this parameter induces that the considered model follows properly the most realistic case. Furthermore, the extremum of the bias to the reference case $\Delta E_{VAR-model\ min-max}$ measures the extent to which the considered model can deviates from the most realistic case. A positive value indicates an underestimation of the assembly energy while a negative one indicates an overestimation.

¹ The “nominal power” case, corresponding to 100% of the nominal power, has only one value of the released energy by each assembly and no dependence on the core mean power. In order to visualize the bias with other cases, it was chosen to be represented as a line.

Quantity / Assembly	J01	H01	L02	K02	J02	H02
E_{NOM} (GWd/t)	3.22	4.38	7.40	10.24	11.10	8.17
$ \Delta E_{VAR-NOM} $ (%)	2.48	3.53	3.50	4.41	4.18	2.46
$\Delta E_{VAR-NOM_{min-max}}$ (%)	5.15	6.77	5.96	8.30	8.47	5.88
$ \Delta E_{VAR-CST} $ (%)	0.61	0.73	0.52	0.36	0.88	1.10
$\Delta E_{VAR-CST_{min-max}}$ (%)	1.73	2.07	-1.18	1.19	2.11	2.53

Table 1: Summary of the results for different assemblies where “VAR”, “NOM” and “CST” refers to the “variable power”, “nominal power” and “constant power” cases, respectively.

Before analyzing the differences between these simulations, it is important to recall that all simulations were performed with a final core burnup equal to 10.43 GWd/t, meaning that comparisons are not biased by any cycle length difference. Regarding the “nominal power” case (blue line in **Figure 6**), the bias to the “variable power” case increases as core cycle mean power decreases, leading to a non-negligible underestimation (up to 8.47%) of the energy released by the considered assemblies (**Table 1**). However, even if the “constant power” case (red line in **Figure 6**) remains a simple model, it is largely less biased than the “nominal power” case because the impact of the control rods insertion and the impact of the core power drop-off are taken into account. On the contrary, the impact of their variations during a cycle is not modeled in the “constant power” case, which explains the differences with the “variable power” case. By comparison, the bias between these two cases is always between -1.18% and 2.53%. Finally, even if the “constant power” case constitutes a good compromise between model simplicity and precision (since $|\Delta E_{VAR-CST}|$ is always inferior to 1.10%), results are slightly underestimated for the J01, H01, J02 and H02 assemblies (**Figure 1**) when the cycle mean power is close to 85% (**Figure 6**). As consequence, this case could be adjusted at high cycle mean powers to improve its predictions.

4.2) Impact on the fast neutron flux at the vessel

The fast neutron flux in each detector described in section 2.4 and the differences $\Delta\Phi_{32-80}$ resulting from the power maps of cores operating at 32% and 80% of their nominal power (section 3.2) are presented in **Figure 7**. These results show that core power variations have a significant (mostly superior to the 3 sigma statistical uncertainty) impact on the fast flux. Furthermore, comparing **Figure 5** and **Figure 7**, it can be deduced that an increase of 1% on the radial power of the studied assemblies results almost in an increase of 1% on Φ_{fast} , meaning that the effect on the attenuation is of the same order.

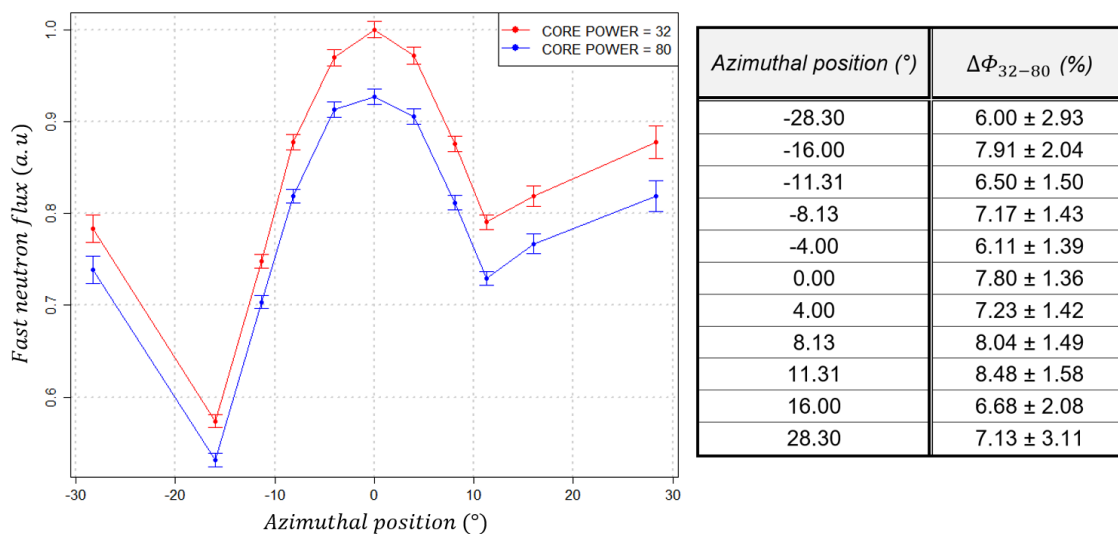


Figure 7 : Fast neutron flux azimuthal distribution at the core mid plane at 2.03 GWd/t (left side) and fast flux differences between cases 32% and 80% (right side).

5. Discussion

The results of this study show that core power variations have a significant impact (up to 8.5%) on the fast neutron flux at the vessel. However, some of the approximations used could impact the result.

Firstly, the neutron sources have been determined by a deterministic code solving the two-group diffusion equation. Even if the number of energy groups employed in the calculations is expected to have low impact on the evaluated core power distributions (inferior to 1% according to Reference [17]), the diffusion approximation implies potentially large uncertainties at the core periphery boundary. Thus, an improvement to this methodology could be the use of 3D Monte-Carlo simulations to determine the neutron source (inducing a more time consuming calculation scheme).

Secondly, a preliminary study has shown that the fuel pins that are the closest to the vessel contributes up to two order of magnitude more than the ones at the opposite side of the assembly. However, the model used in this paper considered power distributions on assembly level and this approximation has a non-negligible impact on the calculated fast neutron flux, as already seen in Reference [18]. Therefore, a good improvement to this modeling will be to take into account a pin power distribution instead of an assembly one. Additionally, the pin power distribution is also impacted by the control rod insertion that may influence the fast neutron fluence.

Thirdly, in the MCNP6 modeling, the moderator temperature is uniform over the whole reactor core. In reality, the moderator temperature at the axial center of the vessel decreases of approximately 5 degrees when the core power drops from 80% to 30% of the nominal power. Based on the sensitivity study performed in Reference [12], the authors deduced that the used modeling could lead to an overestimation of $\Delta\Phi_{32-80}$ about 2%. Therefore, operating in average at 32% instead of 80% of the core nominal power can lead to an increase of the fast neutron flux at the vessel up to 6.5% (rather than 8.5%).

Finally, several hypotheses were made on the neutron source definition at the interface between the two steps of the calculation scheme. In this study, the number of neutrons emitted and the recoverable energy per fission have been neglected (cf. equation (3)) and this approximation may have an impact on the estimated fast flux differences. Therefore, defining these parameters for each assembly could constitute an improvement. Moreover, the fission spectrum used in the MCNP6 simulation has been defined for the entire reactor core despite the fact that fuel composition varies from one assembly to another, thus, defining a fission spectrum for each assembly could also constitute an improvement.

6. Conclusions

One of the most important issue regarding possible lifetime extensions of nuclear PWRs is the embrittlement of the reactor pressure vessel material mainly induced by fast neutron (with energies > 1 MeV) irradiation. Therefore, models would need to be developed to quantify with precision and accuracy the fast neutron fluence accumulated at the reactor pressure vessel during its operating time. However, before building such a model, one prior key point is to determine which parameters need to be considered in the simulation.

The study presented in this paper assessed the need to take into account the power history of the reactor core and the induced variation of its operational parameters. For this purpose, the impact of core power variations on the neutron fast flux at the reactor pressure vessel has been studied. This analysis has been performed using a calculation scheme coupling a deterministic approach (CASMO5 and PARCS codes) to evaluate the fission distribution inside the reactor core and a Monte-Carlo modeling (MCNP6 code) to estimate the attenuation of the neutrons source from the core to the vessel.

The accomplished work has shown that reactor core power variations have a significant impact on the fast neutron flux at the reactor pressure vessel. Indeed, the vessel of a reactor operating in average at 32% of the nominal power will be exposed to 6.5% - 8.5% more neutron flux than the vessel of a reactor operating in average at 80% of the nominal power. However, a reactor that would operate in average at 32% of the nominal power over the cycle may be very rare. As a result, evaluating the vessel fluence using power distributions from a reactor core operating at nominal power could lead to an underestimation of the vessel fluence. Consequently, to accurately evaluate the vessel fluence for reactors operating

at a power significantly inferior to their nominal one, it may be necessary to take into account their power history. To this end, realistic power histories can be approximated by a simulation where the reactor core power is constant during the whole cycle length (at a value equal to the core cycle mean power). Indeed, the study presented in this paper shows that this model follows in an acceptable way the behavior of a realistic one but could be improved by some corrections for high cycle mean power. In the end, this model constitutes a good compromise between simplicity and precision.

References

- [1] Pichon C., Brillaud, C., Deydier, D., Alberman, A., and Soulat, P., "Neutron Spectrum Effect and Damage Analysis on Pressure Vessel Steel Irradiation Behaviour", Effects of radiation on Materials: 19th International Symposium, ASTM STP 1366, M. L. Hamilton, A. S. Kumar, S. T. Rosinski, and M. L. Grossbeck, Eds., American Society for Testing and Materials West Conshohocken, PA, 2000. DOI : 10.1520/STP12384S
- [2] A. Dupré et al., "Towards modeling and validation enhancements of the PSI MCNPX fast neutron fluence computational scheme based on recent PWR experimental data". Elsevier Ltd, Annals of Nuclear Energy, Volume 85, Pages 820-829, November 2015. DOI : 10.1016/j.anucene.2015.06.040
- [3] K. Matsushita & M. Kurosawa. "Neutron fluence analyses around the reactor pressure vessel of BWR using MCNP with a heterogeneous and homogeneous mixed core model". Progress in Nuclear Science and Technology, Volume 4, pp. 463-466 (2014). DOI: 10.15669/pnst.4.463
- [4] John C. Wagner, Alireza Haghghat & Bojan G. Petrovic (1996) Monte Carlo Transport Calculations and Analysis for Reactor Pressure Vessel Neutron Fluence, Nuclear Technology, 114:3, 373-398. DOI: 10.13182/NT96-A35241
- [5] S. Bourganel, O. Petit, C. M. Diop, "Three-Dimensional Particle Transport Using Green's Functions in TRIP-OLI-4 Monte Carlo Code: Application to PWR Neutron Fluence and Ex-Core Response Studies", Nuclear Technology, 184:1, 29-41 (2013). DOI: 10.13182/NT13-A19866
- [6] J. Rhodes et al., "CASMO5 A Fuel Assembly Burnup Program User's Manual". Rev. 8, Studsvik Scandpower Inc (2014).
- [7] T. Downar et al., "PARCS, U.S. NRC Core Neutronics Simulator – USER MANUAL". University of Michigan (2009).
- [8] T. Goorley et al., "Initial MCNP6 Release Overview –MCNP6 version 1.0", LA-UR-13-22934, (2013).
- [9] Book: H. Gard, « Physique, fonctionnement et sûreté des REP – le réacteur en production ». EDP-SCIENCES, Collection Génie Atomique (2013).
- [10] MCNP6 Users Manual - Code Version 6.1, LA-CP-13-00634 (May 2013).
- [11] Book: A. Hébert, "Applied Reactor Physics". Presses internationales Polytechnique (2009).
- [12] M. Brovchenko et al., "Neutron-gamma flux and dose calculations in a Pressurized Water Reactor (PWR)". EPJ Web Conf. Volume 153, 05008, ICRS-13 & RPSD-2016 (2017). DOI: <https://doi.org/10.1051/epjconf/201715305008>
- [13] Book: "Irradiation Embrittlement of Reactor Pressure Vessels (RPVs) in Nuclear Power Plants". 1st edition, Naoki Soneda, Woodhead Publishing (2014).
- [14] Mosher S. W. et al. "ADVANTG - An Automated Variance Reduction Parameter Generator". ORNL/TM-2013/416 Rev. 1, Oak Ridge National Laboratory (2015).
- [15] John C. Wagner, Alireza Haghghat, "Automated Variance Reduction of Monte Carlo Shielding Calculations Using the Discrete Ordinates Adjoint Function". Nuclear Science and Engineering, 128:2, 186-208 (1998). DOI: <https://doi.org/10.13182/NSE98-2>
- [16] Evans T. M. et al., "Denovo: A New Three-Dimensional Parallel Discrete Ordinates Code in SCALE". Nucl. Technol., 171, 171 (2010). DOI : <http://dx.doi.org/10.13182/NT10-5>
- [17] T. Bahadir et S-Ö Lindahl, "SIMULATE-4 Pin power Calculations", ANS Topical Meeting in Mathematics on Reactor Physics, Organized and hosted by the Canadian Nuclear Society, Vancouver, BC, Canada (2006).
- [18] M. Brovchenko et al., "Neutron-gamma flux and dose calculations for feasibility study of DISCOMS instrumentation in case of severe accident in a GEN 3 reactor". EPJ Web Conf. Volume 153, 07030, ICRS-13 & RPSD-2016 (2017). DOI: <https://doi.org/10.1051/epjconf/201715307030>