Sensitivity of the neutron multiplication factor to gadolinium isotopes' nuclear data for light water reactor fuel assemblies in the peak reactivity burnup range

Go Chiba*,[a]

[a]*Hokkaido University, Kita-ku, Sapporo, Hokkaido 060-8628, Japan*

Abstract

The reactivity of a fuel assembly including burnable absorbers can become the largest at low fuel burnup, so the accuracy of reactivity calculations for such systems is important. To investigate this issue, the impact of gadolinium isotopes' nuclear data on neutron multiplication factor k is quantified. Sensitivities of k to nuclear data are calculated from 0 to 20 GWD/t for a BWR 3×3 multicell model. Sensitivity to gadolinium-157 (n,γ) cross section becomes the largest at the zero burnup. Sensitivity to gadolinium-155 (n,γ) cross sections takes the two largest values and the second one is observed around fuel burnup where the reactivity reaches its peak. Sensitivities are also calculated for BWR and PWR assemblies, and similar trends are observed. Finally, nuclear data-induced uncertainties of k are quantified. Gadolinium-157 contribution is the largest at zero burnup, and gadolinium-155 contribution is relatively important around fuel burnup corresponding to the reactivity peak.

Key words: Burnable absorber, gadolinium, depletion perturbation theory, uncertainty quantification

1. Introduction

The use of burnable absorbers in nuclear fuels to suppress the excess reactivity at the beginning of nuclear reactor operations has been quite a familiar concept in thermal reactors, and generally gadolinium-oxide (or gadolinia) is used by mixing it with uranium-dioxide. There are six stable gadolinium isotopes, and it is well known that gadolinium-155 and -157 have significantly large (n,γ) cross sections, especially for thermal neutrons. By adopting these burnable absorbers, the neutron multiplication factor at the beginning of reactor operation is dramatically reduced, so the initial excess reactivity can be suppressed and the reactivity management becomes relatively easy. With reactor operation, these burnable absorbers are depleted, and other gadolinium isotopes with low (n,γ) cross sections are generated. Thus the neutron multiplication factor increases until, after a certain burnup period, most of nuclides initially present in the

*Corresponding author, Tel: +81-10-706-6683, Fax: +81-10-706-6683
Email address: go_chiba@eng.hokudai.ac.jp (Go Chiba)

burnable absorbers have been depleted and converted to other nuclides, so the multiplication factor begins to decrease with burnup as in nuclear fuels without burnable absorbers. This means that there should be a peak of reactivity during fuel burnup in nuclear fuels with burnable absorbers. Generally, nuclear fuels are burned in a nuclear reactor core until they reach on average around 35 GWD/t burnup at discharge, but due to various reasons, there is a possibility that irradiated nuclear fuels with low fuel burnup are also treated outside of a reactor core, such as in a water pool. For a water pool containing irradiated nuclear fuels, proper criticality management should be conducted; reactivity of a system should be accurately estimated and sub-criticality of this system should be assured. Reactivity prediction generally relies on numerical simulation, and the fact that the largest reactivity can be found not in fresh fuel but in low-burnup fuel if nuclear fuel in this system includes burnable absorbers should be carefully considered. This suggests that numerical prediction of the neutron multiplication factor around fuel burnup corresponding to full depletion of burnable absorbers is important.

The numerical prediction of the evolution of the neutron multiplication factor of a system along burnup should include various types of uncertainties, one of them being the uncertainty induced by nuclear data which are used in nuclear fuel burnup calculations and reactivity calculations of neutron multiplying systems. It is needless to say that nuclear data of major actinides such as uranium and plutonium are important, but nuclear data of other nuclides including burnable absorbers (i.e., gadolinium isotopes) should also be considered for the above-mentioned reasons.

To quantify nuclear data-induced uncertainty of neutron multiplication factor during nuclear fuel burnup, the Monte Carlo sampling procedure based on a number of calculation cases with randomly-sampled nuclear data has been widely used in the field of reactor physics, and a lot of works and results have been accumulated(Rochman, 2012, 2014; Yamamoto, 2015; Leray, 2017). However, as far as the authors know, no such studies focusing on the neutron multiplication factor around full depletion of burnable absorbers have been published so far.

The present work addresses nuclear data-induced uncertainties of the neutron multiplication factor around full depletion of burnable absorbers in light water reactor (LWR) fuel assemblies. Uncertainty quantification will be carried out not with the Monte Carlo sampling procedure but with a sensitivity-based approach; sensitivities of the neutron multiplication factor with respect to nuclear data are initially calculated, and then nuclear data-induced uncertainties are quantified with these sensitivities and covariance data of nuclear data. The capability to calculate sensitivities of reactor physics parameters to nuclear data during nuclear fuel burnup in LWR fuel assemblies including burnable absorbers has been recently developed(Chiba, 2016a, 2018a), based on the generalized perturbation theory for spatially-dependent nuclear fuel depletion problems, the *depletion perturbation theory* (DPT), established by Gandini(Gandini, 1969) and Williams(Williams, 1979). With this new capability, sensitivities can be easily calculated. Another merit of this approach is that it can help understand how nuclear data affects calculation results, i.e., neutron

multiplication factor change during fuel burnup.

When conducting nuclear data-induced uncertainty quantification, reliable covariance data are mandatory. Among the widely-used evaluated nuclear data libraries, ENDF/B, JEFF and JENDL, the ENDF/B and JEFF libraries provide covariance data for gadolinium isotopes. However, the latest version of these libraries, ENDF/B-VIII.0 and JEFF-3.3, adopt the identical data obtained in 2009 through the international collaboration in the framework of the Working Party of International Evaluation Co-operation of the Nuclear Energy Agency within the Organisation for Economic Co-operation and Development(NEA, 2009). In the present work, these covariance data will be used, however it is important to underline that uncertainty quantification analyses are highly dependent on the covariance data used. For this reason, the analyses will be focused on the sensitivities.

The present paper is organized as follows. Section 2 describes basic data, theory and procedure about sensitivity analyses and uncertainty quantification for the neutron multiplication factor during nuclear fuel burnup. Section 3 presents problem specifications for nuclear fuel burnup and all the numerical results, and the conclusion is given in Section 4.

2. Data, theory and procedure of sensitivity analyses and uncertainty quantification

2.1. Nuclear data and their covariance data

In the present work, sensitivities are calculated with the JENDL library: JENDL-4.0(Shibata, 2011) for neutron-induced reaction cross sections, JENDL/FPY-2011 and JENDL/FPD-2011(Katakura, 2011) for fission yield data and decay data of fission products (FP). Calculated sensitivities are used to quantify nuclear data-induced uncertainties of the neutron multiplication factor. The present work mainly focuses on gadolinium isotopes' nuclear data, but to assess the impact of gadolinium nuclear data uncertainty in the total uncertainty, uncertainties of reaction cross sections of major actinides, fission yields, decay constants and decay branching ratios are also taken into account. Generally, uncertainty information on these nuclear data are provided as covariance data in evaluated nuclear data libraries. In JENDL-4.0, covariance data are evaluated for almost all the important actinides, and those are taken into account in the present study. However, as mentioned in the introduction, JENDL-4.0 provides no covariance data for reaction cross sections of gadolinium isotopes, so we will use covariance data for these nuclear data taken from ENDF/B-VIII.0(Brown, 2018). In JENDL/FPY-2011, fission yield data are given to 1,299 FP. Only variance data are provided to fission yield data in JENDL/FPY-2011 due to the format limitation, but it has been pointed out by many authors that correlations in fission yield data among different nuclides should be taken into account to properly conduct uncertainty propagation calculations(Katakura, 2011; Fiorito, 2016). While there have been several methods to consider correlations, we will adopt the generalized least-square (GLS) updating procedure which introduces information about chain yields and some physical

constraints about fission reactions(Fiorito, 2016). On the decay branching ratio, correlations due to the physical constraints (the normalization condition) are taken into account.

2.2. Sensitivity calculations with the depletion perturbation theory

For a certain nuclide, the relative sensitivity of neutron multiplication factor k to nuclear data σ_i is defined as

$$s_i = \frac{dk}{d\sigma_i} \cdot \frac{\sigma_i}{k}. \tag{1}$$

Since quite a large number of nuclear data are used in nuclear fuel burnup calculations, numerical differentiation to obtain s_i is unrealistic. Therefore, DPT will be used to calculate sensitivities of k during nuclear fuel burnup with respect to nuclear data such as neutron-induced reaction cross sections, fission yields, decay constants and decay branching ratios. Details about DPT can be found in references(Gandini, 1969; Williams, 1979).

Numerical calculations of sensitivities with DPT are generally based on numerical methods for nuclear fuel burnup calculations, which consist of neutron flux distribution calculations by solving the neutron transport equation and nuclides transmutation calculations. In the present study, all these calculations are carried out with a reactor physics code system CBZ(Chiba, 2019) which has been developed at Hokkaido University.

In nuclear fuel burnup calculations, generation and transmutation of FP should be numerically simulated. Since there are a large number of FP, over 1,000, it is computationally expensive to treat all these FP explicitly in nuclide transmutation calculations, especially in a complicated system like LWR fuel assemblies. Furthermore, the computational burden increases even more when the DPT calculations are carried out since a huge amount of data during nuclear fuel burnup should be stored in computer memory. In order to solve these problems, a simplified nuclide chain model is used consisting of 138 FP(Chiba, 2015) with which reactivity during fuel burnup can be accurately calculated.

For resonance calculations, the advanced Bondarenko method(Chiba, 2016b) is adopted, and medium-wise 107-group cross section data are generated. Energy structure of the 107-group is that of the SRAC code developed for thermal reactor analyses(Okumura, 2006). Background cross sections required in the advanced Bondarenko method are calculated from the Dancoff factors evaluated with a whole assembly model(Sugimura, 2006). Neutron transport calculations with the 107-group cross sections including generalized adjoint neutron flux calculations for DPT are performed with a neutron transport calculation module MEC(van Rooijen, 2011) based on the method of characteristics, and reaction rates are calculated for all the media containing nuclear fuels. Note that angular integration of bilinear functions of neutron flux and generalized adjoint neutron flux in DPT should be carefully carried out in a system including strong neutron absorbers(Chiba, 2018a). Scattering anisotropy is taken into account by the P0 transport approximation.

Nuclides transmutation calculations are carried out with the simplified nuclide chain model as mentioned above. Nuclides transmutation equations including adjoint problems in DPT are solved by the matrix exponential method with the mini-max polynomial approximation method(Kawamoto, 2015). To reduce the time-discretization error, the predictor-corrector method is employed.

The relative sensitivity of k at fuel burnup t to σ can be calculated as follows:

$$\frac{dk}{d\sigma}\cdot\frac{\sigma}{k}=\left\{\frac{\partial k}{\partial\sigma}+\frac{\partial k}{\partial\mathbf{\Phi}(t)}\cdot\frac{d\mathbf{\Phi}(t)}{d\sigma}+\frac{\partial k}{\partial\mathbf{N}(t)}\cdot\frac{d\mathbf{N}(t)}{d\sigma}\right\}\cdot\frac{\sigma}{k}, \tag{2}$$

where $\mathbf{\Phi}(t)$ and $\mathbf{N}(t)$ are vectors representing neutron flux and nuclides number density at t. The first and second terms in the parenthesis of right-hand-side of this equation can be easily calculated with the classical perturbation theory, and are referred to as *static component*. On the other hand, the third term can be calculated with the classical perturbation theory and DPT, and is referred to as *burnup component*(Chiba, 2016a).

2.3. Uncertainty quantification with sensitivities and covariance data

Nuclear data-induced uncertainties of k can be easily quantified if sensitivities of k with respect to nuclear data are available. The relative uncertainty in k due to nuclear data can be calculated with the following equation:

$$\Delta k/k=\sqrt{\sum_i\sum_{i'}s_i s_{i'}\mathrm{cov}(\sigma_i,\sigma_{i'})}, \tag{3}$$

where $\mathrm{cov}(\sigma_i,\sigma_{i'})$ is a relative covariance between σ_i and $\sigma_{i'}$, and a covariance matrix covering all the nuclear data considered is defined as $\mathbf{V}_\sigma$ here. When we define a sensitivity vector $\mathbf{s}$ as $\mathbf{s}=(s_1\ \ s_2\ \ \cdots\ \ s_I)^T$, where T is for vector transposition, $\Delta k/k$ can be rewritten using $\mathbf{V}_\sigma$ and $\mathbf{s}$ as

$$\Delta k/k=\sqrt{\mathbf{s}^T\mathbf{V}_\sigma\mathbf{s}}. \tag{4}$$

Since a simplified nuclide chain model is used, sensitivities of k to nuclear data are calculated within the framework of this simplified model, so in the uncertainty quantification calculations, covariance data consistent with the simplified chain model should be used. Discussion about this issue, as well as the numerical method and tool developed, can be found in reference(Chiba, 2018b) and will be used in the following.

3. Numerical tests and results

3.1. Problem specifications

Sensitivities of k with respect to nuclear data including (n,γ) cross sections of gadolinium isotopes during nuclear fuel burnup are calculated with high resolution (per 0.1 GWD/t) for a 3×3 multicell model. In this model, a gadolinium-bearing fuel pin is loaded at the center position and it is surrounded by uranium fuel

pins whose uranium-235 enrichment is 3.9 wt%. Geometrical and material specifications are taken from a BWR fuel assembly model developed through the OECD/NEA burnup credit benchmark phase-IIIC(NEA, 2015). Reflective boundary conditions are adopted. The geometrical specification of the 3×3 multicell model and the original BWR assembly model is shown in **Fig. 1**. As for coolant condition, two different void fractions of 0% and 40% are considered. The infinite neutron multiplication factor during fuel burnup of this 3×3 multicell model is shown in **Fig. 2**. The reactivity peaks are observed at around 13 GWD/t. In the original BWR fuel assembly model, a square-shaped large water channel is located inside the assembly, so neutron moderation by water in this 3×3 model is less enhanced than the original assembly model. With the calculated sensitivities and nuclear data covariance matrix, nuclear data-induced uncertainties of k can be also quantified with high resolution for fuel burnup.

In addition to the 3×3 multicell model, sensitivities of k with respect to (n,γ) cross sections of gadolinium isotopes are calculated for the original BWR assembly with three different void conditions (0%, 40% and 70%) and two PWR assemblies prepared in the VERA depletion benchmark suite(Kim, 2015) with low resolution (per 1.0 GWD/t). The geometrical specification of these PWR assembly models, 2O and 2P, is shown in **Fig. 3**. Weight percent of Gd_2O_3 in gadolinium-bearing fuel rods is 5.0% in both the BWR and PWR assemblies. The infinite neutron multiplication factor of these BWR and PWR assemblies with fuel burnup is shown in **Figs. 4** and **5**, respectively. As shown in these figures, fuel burnup where full depletion of gadolinium isotopes is attained is different from each other: the gadolinium isotopes burn out earlier in the PWR assemblies (around 8 GWD/t) than in the BWR assembly (around 12 GWD/t) because of a difference in fuel pin diameter.

3.2. Detailed sensitivity profiles of k to (n,γ) cross sections of gadolinium isotopes in the 3×3 multicell model

In this section, sensitivities of k to (n,γ) cross section of gadolinium isotopes in the 3×3 multicell model from 0 to 20 GWD/t are presented.

The k sensitivities can be calculated to every group cross section, and sensitivities to one-group (energy-averaged) cross section can be easily calculated by summing up all these sensitivities. One-group sensitivity is a useful quantity since it can be used to estimate the impact on k of average change in the cross section. Sensitivities of k to one-group (n,γ) cross section of gadolinium isotopes are shown in **Figs. 6** to **9**. In addition to total sensitivities, static and burnup components are also presented.

Among all the isotopes and during the whole fuel burnup period, maximum sensitivity is observed in gadolinium-157 cross section at zero fuel burnup.

Regarding gadolinium-155 and -157, which are strong neutron absorbing isotopes, large negative static components are observed at the beginning of fuel burnup. The gadolinium-157 static component becomes small with fuel burnup because of their depletion. The gadolinium-155 static component increases with

fuel burnup at the beginning, and begins to decrease around 4 GWD/t. The static components of energy-dependent sensitivity to gadolinium-155 (n,γ) cross section at the 0% void condition are shown in **Fig. 10**. Since (n,γ) cross section of gadolinium-157 is larger than that of gadolinium-155, gadolinium-157 is depleted quickly at the beginning of fuel burnup, and shifting of the neutron flux or neutron spectrum to lower energies is expected. The (n,γ) cross section of gadolinium-155 is large in low energy range, so this neutron flux energy spectra shifting enhances neutron capture of gadolinium-155 with fuel burnup. Burnup components of sensitivities to gadolinium-155 and -157 increase from zero with fuel burnup in the low burnup range. Let us consider k at certain fuel burnup. If (n,γ) cross section of these isotopes is increased, neutron absorption rate at this fuel burnup should be increased, but the depletion of these isotopes becomes fast and number densities of these isotopes at this fuel burnup should be small. The former and latter correspond to the static and burnup components, respectively, and the cancellation between these two components can be expected. As a result of this cancellation, there exists fuel burnup where the total sensitivity becomes zero. After full depletion of gadolinium-155 and -157, the total sensitivities of these isotopes become close to zero. On gadolinium-157, total sensitivity takes its maximum at zero fuel burnup because of large contribution of the static component. On the other hand, on gadolinium-155, two peaks can be found around fuel burnup of 2 GWD/t and at 11 GWD/t in the 0% void fraction case and at 14 GWD/t in the 40% void fraction case. This is because (n,γ) cross section of gadolinium-155 is smaller than that of gadolinium-157 and the gadolinium-155 depletion should follow the depletion of gadolinium-157.

Regarding gadolinium-154 and -156, which can be converted to gadolinium-155 and -157 via (n,γ) reaction, both the components are negative and the burnup component becomes large with fuel burnup by the burnup of 10 GWD/t. Thus the total sensitivities also become large with fuel burnup in this burnup range. The reason why the burnup component is negative is that (n,γ) reaction of these isotopes generate more neutron absorbing isotopes: gadolinium-155 and -157. The total sensitivity of gadolinium-156 is larger than that of gadolinium-154. After fuel burnup corresponding to full depletion of gadolinium-155 and -157, the burnup components tend to decrease with fuel burnup, and this decrease in gadolinium-154 is more significant than in gadolinium-156 since gadolinium-154 has no parent nuclide in a nuclide chain.

Energy-dependent sensitivities of k at 0.1 GWD/t and 12 GWD/t (or 10 GWD/t) in void fraction of 0% are shown in **Fig. 11**. Sensitive energy ranges of each isotope can be specified from these figures. Contributions of sensitivities in thermal neutron energy range are dominant in gadolinium-155 and -157. In gadolinium-154, contributions of epi-thermal neutron energy range are comparable with contributions of thermal energy range, and in gadolinium-156 contribution of epi-thermal neutron energy range is dominant. Energy-dependent sensitivities in void fraction of 40% are also shown in **Fig. 12**.

3.3. Sensitivity profiles of k to (n,γ) cross sections of gadolinium isotopes in the BWR and PWR assemblies

In the preceding section, component-wise sensitivites of k to (n,γ) cross sections of gadolinium isotopes have been carefully examined with high resolution for fuel burnup, and energy-dependent sensitivities have been also presented in the 3×3 multicell model. In order to obtain more general conclusions, sensitivities for the BWR and PWR assemblies with lower resolution for fuel burnup are calculated

Total sensitivities of the BWR assembly with different void conditions are shown in **Fig. 13**. Since the BWR assembly includes the central water channel and the water gap channel outside the channel box, neutron moderation is enhanced in comparison with the 3×3 multicell model. As the result, absolute values of sensivities to (n,γ) cross sections of gadolinium-154, -155 and -157, in which contribution of thermal neutron energy range is dominant as shown in Fig. 11, becomes larger than those of the 3×3 multicell model by less than a factor of 2. Dependence on void fractions and on fuel burnup is quite similar with the 3×3 multicell model. Sensitivities to gadolinium-156 in the BWR assembly is generally smaller than those in the 3×3 multicell model, but the difference is not significant.

Total sensitivities of the PWR assemblies are shown in **Fig. 14**. Since gadolinium isotopes in these PWR assemblies burn out earlier than in the BWR 3×3 multicell model, dependence of sensitivities on fuel burnup is slightly different from that in the 3×3 model, but the trend is similar. Absolute values of sensitivities of gadolinium-155 and -157 in 2P are comparable with those in the BWR 3×3 multicell model and the BWR assembly, and absolute values of sensitivities of the other gadolinium isotopes are generally smaller than those in the BWR problems.

From these results, we have confirmed that trends in sensitivities of k to (n,γ) cross sections of gadolinium isotopes are very similar among the BWR 3×3 multicell model, the BWR assembly and the PWR assemblies.

3.4. Uncertainty quantification in k due to neutron-induced nuclear data for the 3×3 multicell model

To assess the impact of the (n,γ) cross section uncertainties of gadolinium isotopes in k, uncertainty propagation calculations are performed for the BWR 3×3 multicell model with sensitivities calculated in the preceding section.

As mentioned above, ENDF/B-VIII.0 is the reference for the uncertainty of (n,γ) cross sections of gadolinium isotopes. Relative standard deviations of 107-group (n,γ) cross sections of gadolinum isotopes, obtained from the evaluated data in ENDF/B-VIII.0, are shown in **Fig. 15**, and correlation matrices among different energy groups for each isotope are shown in **Fig. 16**.

With covariance data of nuclear data including (n,γ) cross sections of gadolinium isotopes, uncertainty of k is quantified as shown in **Fig. 17**. Uncertainty is the largest at zero fuel burnup, and becomes almost constant after around 5 GWD/t. In the result of the void condition of 40%, a small local peak is observed at around 12 GWD/t. Component-wise uncertainty of k is shown in **Fig. 18**, and it can be found that neutron-induced reaction cross section uncertainties are dominant in the k uncertainties. This result is

consistent with the previous works if fission yield covariance data are generated with the GLS (or GLS-like) method(Leray, 2017; Chiba, 2018b). **Figure 19** shows reaction cross section-induced uncertainties of k decomposed into heavy metal (HM) components and gadolinium isotopes (Gd) components. Based on covariance data of heavy metal nuclides from JENDL-4.0 and those of gadolinium isotopes from ENDF/B-VIII.0, uncertainty induced by nuclear data of heavy metal nuclides is dominant in the k uncertainties. In the gadolinium nuclear data-induced uncertainties, the largest value is observed at zero fuel burnup and small local peaks can be observed at around 12 GWD/t. Gadolinium isotope-wise k uncertainties induced by (n,γ) cross sections are shown in **Fig. 20**. Contribution of gadolinium-157 is dominant when the fuel is fresh, and it decreases with fuel burnup by 6 GWD/t. Around 10 GWD/t, there is a peak in the gadolinium-157 contribution, but the height of this peak is much lower than that at zero fuel burnup. This can be understood by seeing sensitivities shown in Fig. 9. In gadolinium-155-induced uncertainties, two peaks can be observed around 2 GWD/t and 12 GWD/t in the 0% void fraction case and 0 GWD/t and 14 GWD/t in the 40% void fraction case, and heights of these two peaks are comparabale with each other as expected from sensitivities shown in Fig. 7. This second peak contributes to a small local peak observed in the k uncertainties shown in Fig. 17. Fuel burnup where this second peak is observed corresponds to that of full depletion of gadolinium isotopes and that where k takes the largest value during fuel burnup. This result suggests that gadolinium-155 nuclear data is more important than gadolinium-157 nuclear data from a view point of reactivity management of irradiated nuclear fuel with low fuel burnup. It is also interesting to point out that gadolinium-156 contribution is not negligible because of ralatively large standard deviation of around 15% and large positive correlation in sensitive energy ranges from 1 eV to 1 keV evaluated in ENDF/B-VIII.0.

4. Conclusion

The reactivity (or neutron multiplication factor) of a fuel assembly including burnable absorbers can become the largest not at zero fuel burnup but at low fuel burnup, so the accuracy of reactivity calculations for such systems at low fuel burnup is important. In order to investigate this issue, the impact of gadolinium isotopes' nuclear data of on k has been quantified.

Sensitivities of k with respect to various nuclear data including neutron-induced reaction cross sections of gadolinium isotopes have been calculated with the CBZ code system, in which a capability based on the depletion perturbation theory has been implemented. Total and component-wise sensitivities of k to (n,γ) cross sections of gadolinium isotopes have been calculated with high resolution for fuel burnup for the BWR 3×3 multicell model, and physical interpretations have been provided. From the observations of the sensitivities, the following have been obtained: (1) sensitivity to gadolinium-157 cross section becomes the largest at zero burnup, and (2) sensitivity to gadolinium-155 cross sections takes the two largest values

during fuel burnup and the second one is observed at fuel burnup where reactivity peak is observed. The burnup where the largest sensitivity values are observed are dependent on the void fraction; 2 GWD/t and 11 GWD/t in the 0% void fraction case and 2 GWD/t and 14 GWD/t in the 40% void fraction case. Energy-dependent sensitivities have been also provided.

In order to obtain more general conclusions, sensitivities of k to (n,γ) cross sections of gadolinium isotopes have been calculated for the BWR and PWR assemblies, and it has been confirmed that the trend observed in the BWR 3×3 multicell model also appears in these assemblies.

Finally, with the sensitivities calculated with the depletion perturbation theory, nuclear data-induced uncertainties of k have been quantified. In this calculation, covariance data given in JENDL-4.0 and ENDF/B-VIII.0 have been used for cross sections, and those given in JENDL/FPY-2011 and JENDL/FPD-2011 have been used for fission yield and decay-relevant data. Among these nuclear data, cross sections of heavy metal nuclides are dominant contributors to total k uncertainties. On k uncertainties induced by gadolinium isotopes' nuclear data, gadolinium-157 contribution is the largest at zero fuel burnup, but gadolinium-155 contribution is relatively important around fuel burnup of 13 GWD/t where k takes the largest value during fuel burnup.

Based on the above results, it can be concluded that uncertainties of k induced by gadolinium isotopes nuclear data are not significant during nuclear fuel burnup. This conclusion deserves a careful consideration, since uncertainty quantification analyses are strongly dependent on the input covariance data and evaluators and new state-of-the-art covariance matrices are released, it is strongly recommended that the impact of gadolinium nuclear data uncertainties on k during burnup is once again examined.

Acknowledgements

This work is motivated by a suggestion from Dr. Toshihisa Yamamoto and Dr. Tatsuya Fujita of The Secretariat of the Nuclear Regulation Authority (S/NRA) in Japan.

References

Brown, D.A., et al., 2018. ENDF/B-VIII.0: The 8th major release of the nuclear reaction data library with CIELO-project cross sections, new standards and thermal scattering data, Nucl. Data Sheets, 148, 1-142.

Chiba, G., et al., 2015. Important fission product nuclides identification method for simplified burnup chain construction, J. Nucl. Sci. Technol., 52, 953-960.

Chiba, G., et al., 2016. Development of a fuel depletion sensitvity calculation module for multicell models in a deterministic reactor physics code CBZ. Ann. Nucl. Energy, 96, 313-323.

Chiba, G., et al., 2016. Advanced Bondarenko method for resonance self-shielding calculations in deterministic reactor physics code system CBZ, Ann. Nucl. Energy, 96, 277-286.

Chiba, G., 2018a. Perturbation theory for nuclear fuel depletion calculations with predictor-corrector method. J. Nucl. Sci. Technol., 55. 290-300.

Chiba, G., Okumura, S., 2018b. Uncertainty quantification of neutron multiplication factors of light water reactor fuels during depletion. J. Nucl. Sci. Technol., 55, 1043-1053.

Chiba, G., Endo, T., 2019. Numerical benchmark problem of solid-moderated enriched-uranium-loaded core at Kyoto university critical assembly, J. Nucl. Sci. Technol., 57, 187-195.

Fiorito, L., et al., 2016. Generation of fission yield covariances to correct discrepancies in the nuclear data libraries, Ann. Nucl. Energy, 88, 12-23.

Gandini, A., 1969. A method of correlation of burnup measurements for physics prediction of fast power-reactor life. Nucl. Sci. Eng., 38, 1-7.

Katakura, J., 2011. JENDL FP decay data file 2011 and fission yields data file 2011, JAEA-Data/Code 2011-025, Japan Atomic Energy Agency.

Kawamoto, Y., Chiba, G., Tsuji, M., et al. 2015. Numerical solution of matrix exponential in burn-up equation using mini-max polynomial approximation, J. Nucl. Sci. Technol., 80, 219-224.

Kim, K.S., 2015. Specification for the VERA depletion benchmark suite, CASL-U-2015-PHI.CVS.P11.02-010, Oak Ridge National Laboratory.

Leray, O., et al., 2017. Uncertainty propagation of fission product yields to nuclide composition and decay heat for a PWR UO2 fuel assembly. Prog. Nucl. Energy, 101, 486-495.

Nuclear Energy Agency (NEA), 2009. Evaluated data library for the bulk of fission products, Vol. 23, International Evaluation Co-operation, A report by the Working Party on International Evaluation Co-operation of the NEA Nuclear Science Committee, NEA/WPEC-23, OECD/NEA, Paris.

Nuclear Energy Agency (NEA), 2015. Burn-up credit criticality safety benchmark phase III-C. NEA/NSC/R/(2015)6, OECD/NEA, Paris.

Okumura. K., et al., 2006. SRAC2006: a comprehensive neutronics calculation code system, JAEA-Data/Code 2007-004, Japan Atomic Energy Agency.

Rochman, D., et al., 2012. Propagation of ^{235,236,238}U and ^{239}Pu nuclear data uncertainties for typical PWR fuel element. Nucl. Technol., 179, 323-338.

Rochman, D., et al., 2014. Efficient use of Monte Carlo: uncertainty propagation. Nucl. Sci. Eng., 177, 337-349.

van Rooijen, W.F.G., Chiba, G., 2011. Diffusion coefficient for LMFBR cells calculated with MOC and Monte Carlo methods. Ann. Nucl. Energy, 38, 133-144.

Shibata, K., et al., 2011. JENDL-4.0: a new library for nuclear science and engineering, J. Nucl. Sci. Technol., 48, 1-30.

Sugimura, N., Yamamoto, A., 2006. Evaluation of Dancoff factors in complicated geometry using the method of characteristics, J. Nucl. Sci. Technol., 43, 1182-1187.

Williams, M.L., 1979. Development of depletion perturbation theory for coupled neutron/nuclide fields. Nucl. Sci. Eng., 70, 20-36.

Yamamoto, A., et al., 2015. Uncertainty quantification of LWR core characteristics using random sampling method. Nucl. Sci. Eng., 181, 160-174.

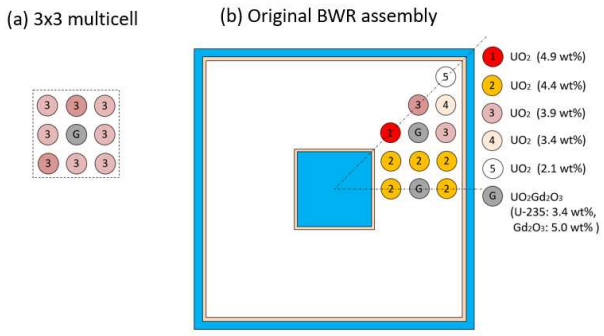

Figure 1: Geometrical specification of the 3×3 multicell model and the BWR assembly model

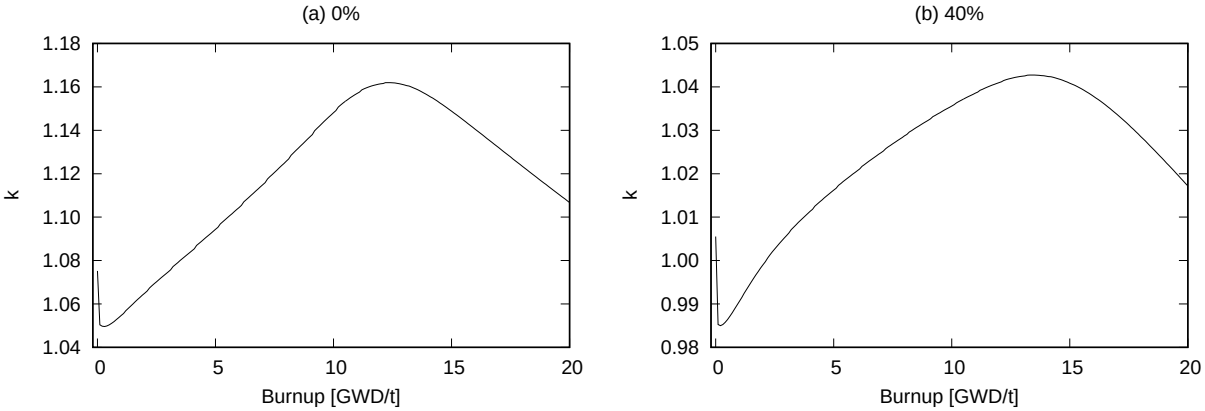

Figure 2: Infinite neutron multiplication factor of the 3×3 multicell model during nuclear fuel burnup

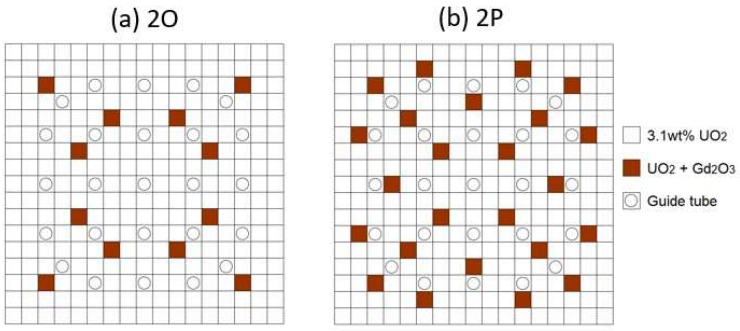

Figure 3: Geometrical specification of the PWR assembly models in the VERA depletion benchmark suite

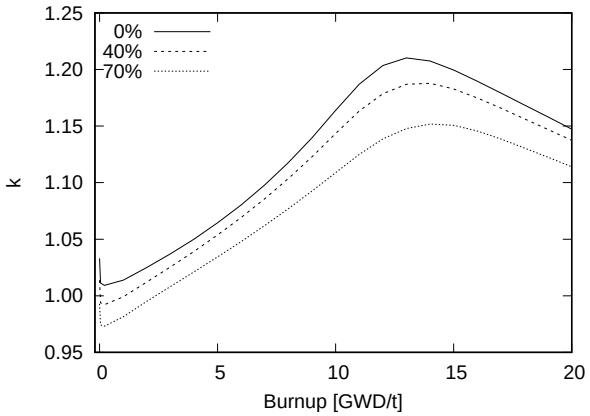

Figure 4: Infinite neutron multiplication factor of the BWR assembly with different void fraction

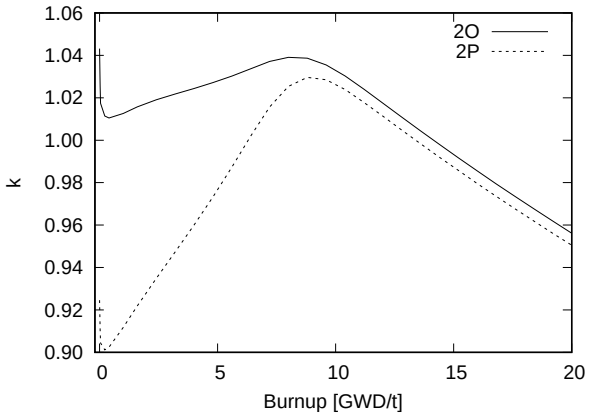

Figure 5: Infinite neutron multiplication factor of the PWR assemblies

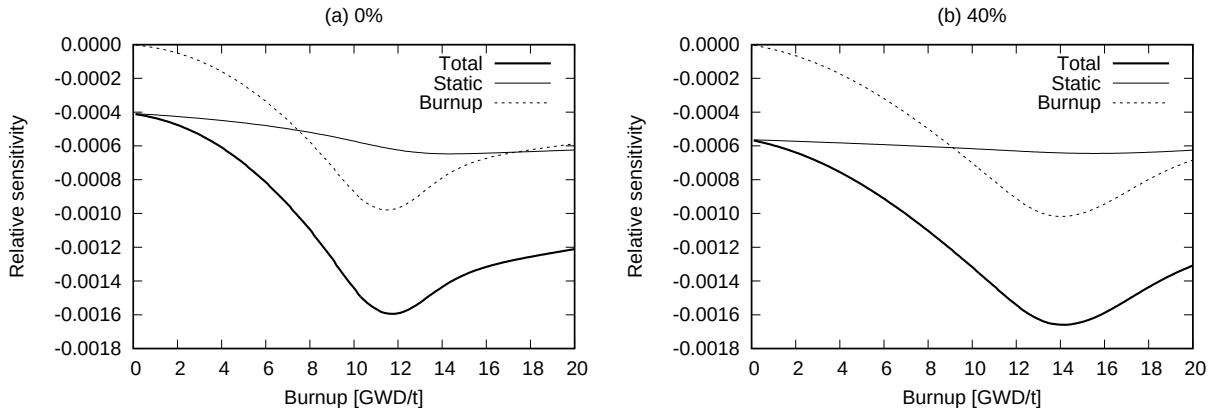

Figure 6: Sensitivity of k to one-group (n,γ) cross sections of gadolinium-154 during nuclear fuel burnup in the 3×3 multicell model

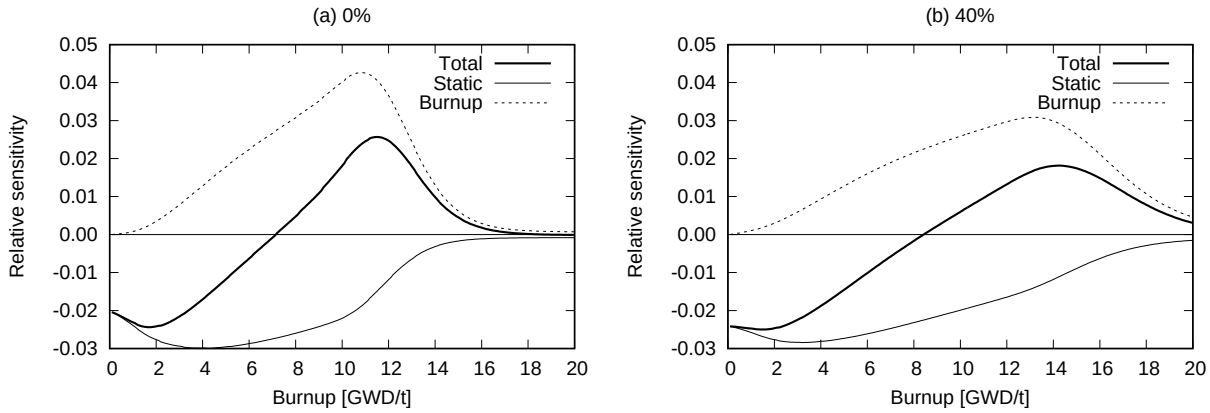

Figure 7: Sensitivity of k to one-group (n,γ) cross sections of gadolinium-155 during nuclear fuel burnup in the 3×3 multicell model

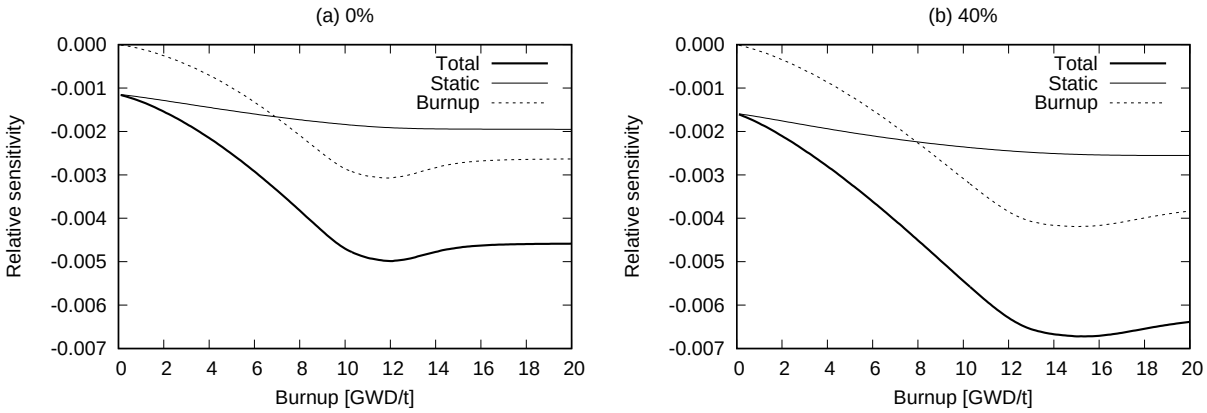

Figure 8: Sensitivity of k to one-group (n,γ) cross sections of gadolinium-156 during nuclear fuel burnup in the 3×3 multicell model

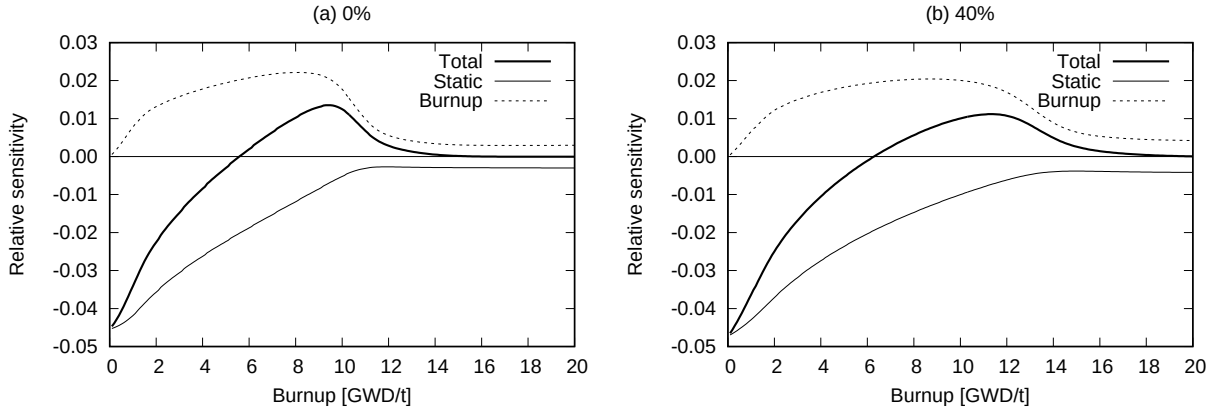

Figure 9: Sensitivity of k to one-group (n,γ) cross sections of gadolinium-157 during nuclear fuel burnup in the 3×3 multicell model

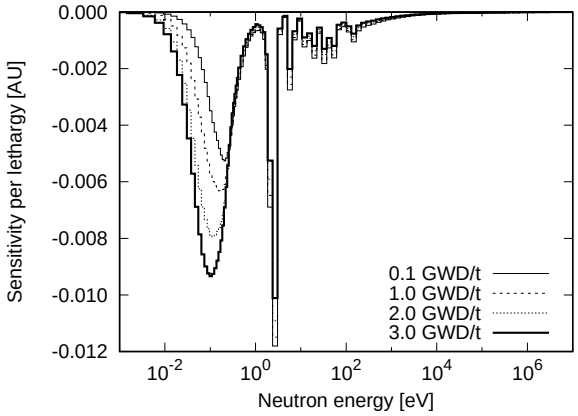

Figure 10: Static component of energy-dependent sensitivity of k to (n,γ) cross sections of gadolinium-155 in the 3×3 multicell model

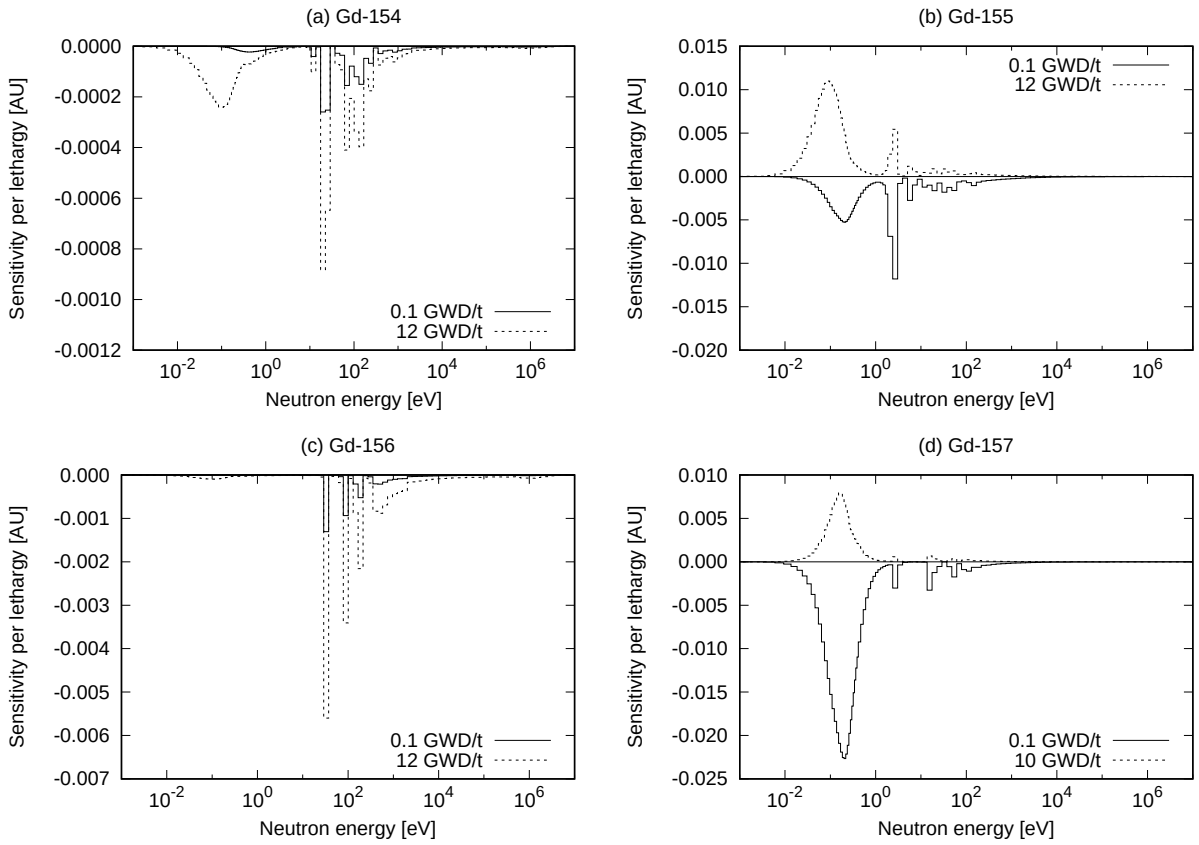

Figure 11: Energy-dependent sensitivity of k to (n,γ) cross sections of gadolinium isotopes in the 3×3 multicell model with 0% void fraction

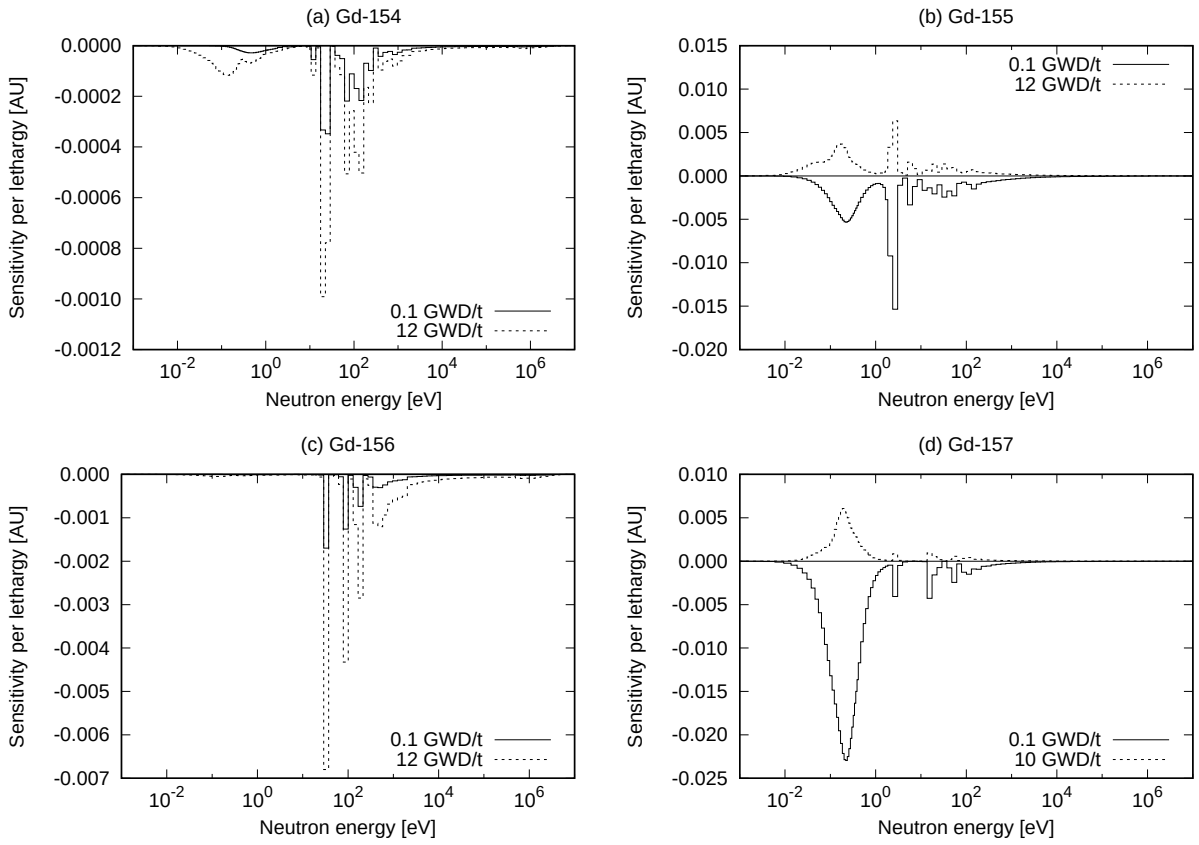

Figure 12: Energy-dependent sensitivity of k to (n,γ) cross sections of gadolinium isotopes in the 3×3 multicell model with 40% void fraction

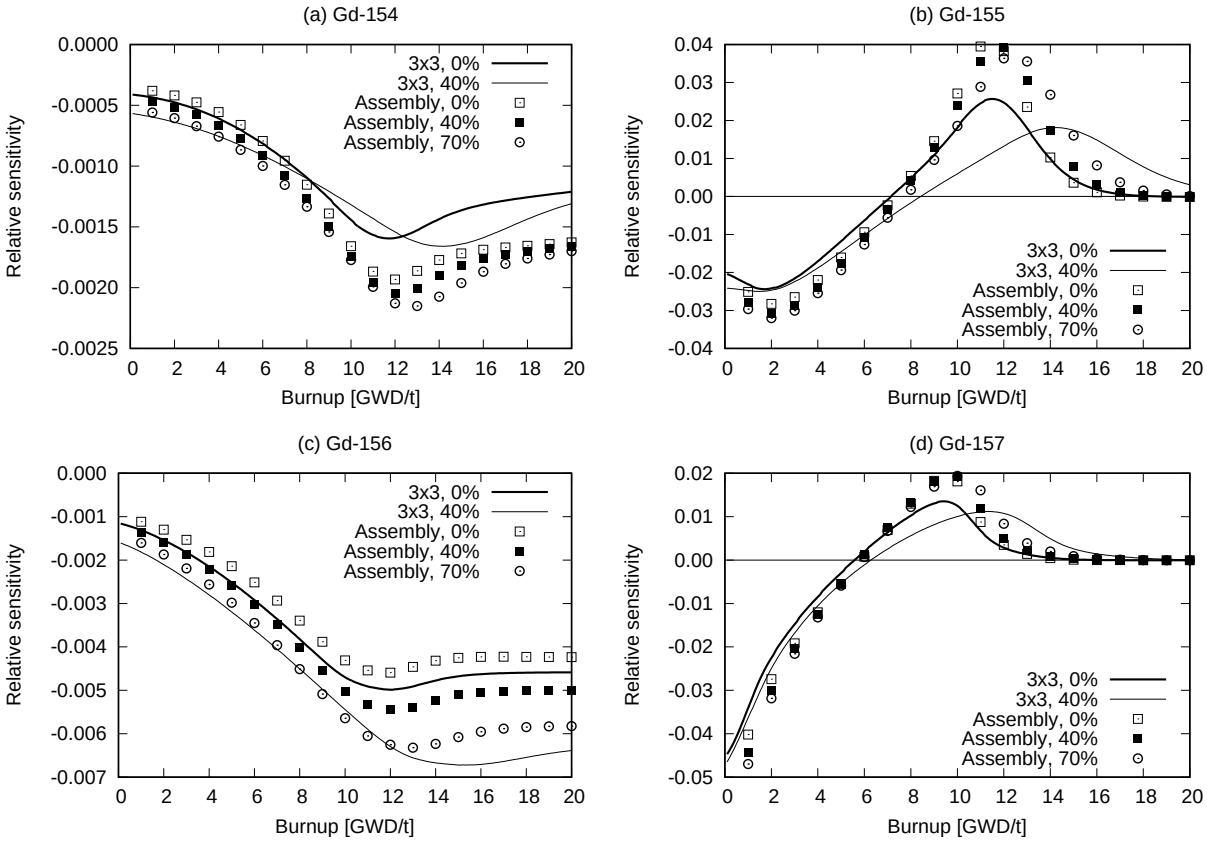

Figure 13: Total sensitivity of k to one-group (n,γ) cross sections of gadolinium isotopes in the BWR assembly

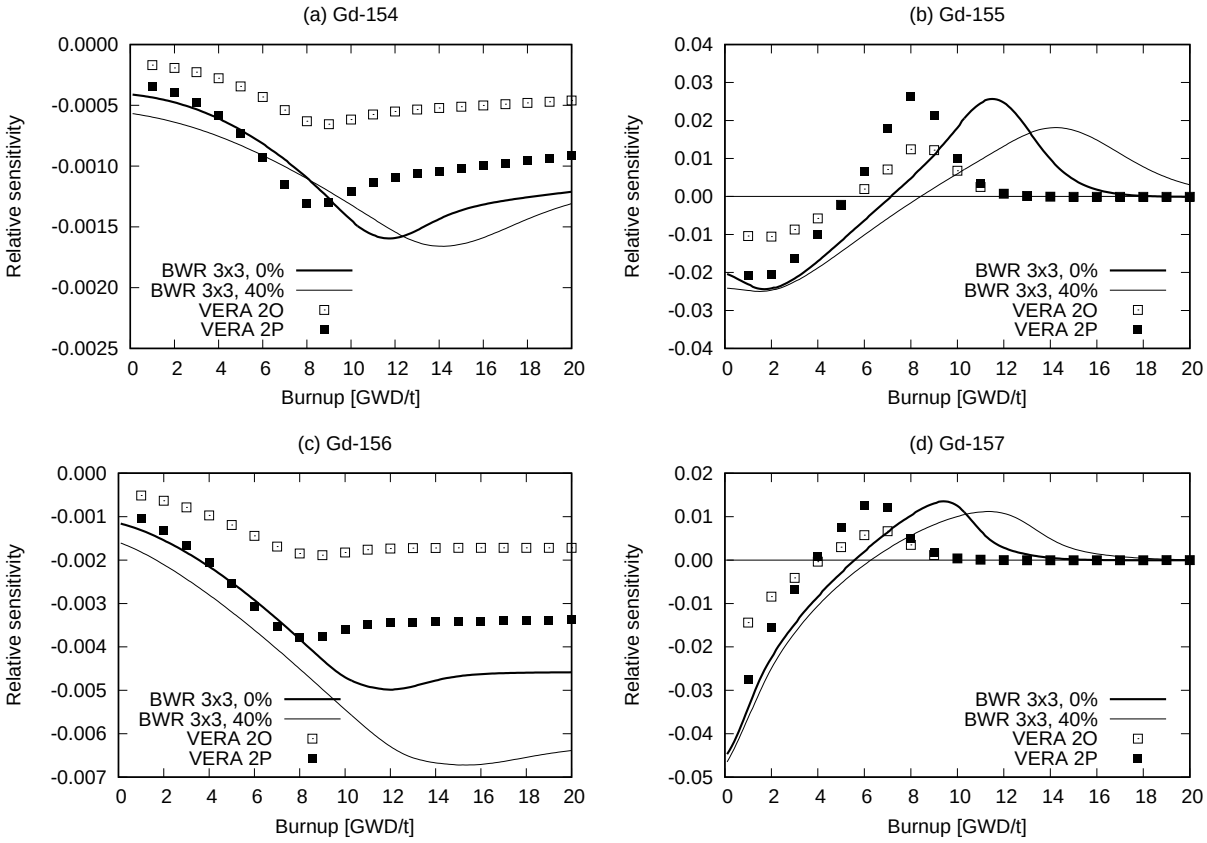

Figure 14: Total sensitivity of k to one-group (n,γ) cross sections of gadolinium isotopes in the PWR assemblies

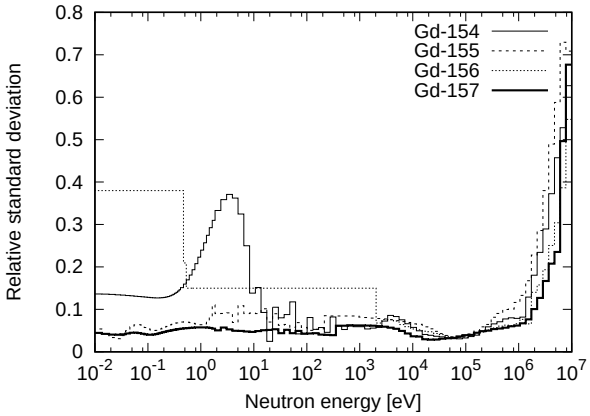

Figure 15: Relative standard deviations of energy group-wise (n,γ) cross sections of gadolinium isotopes

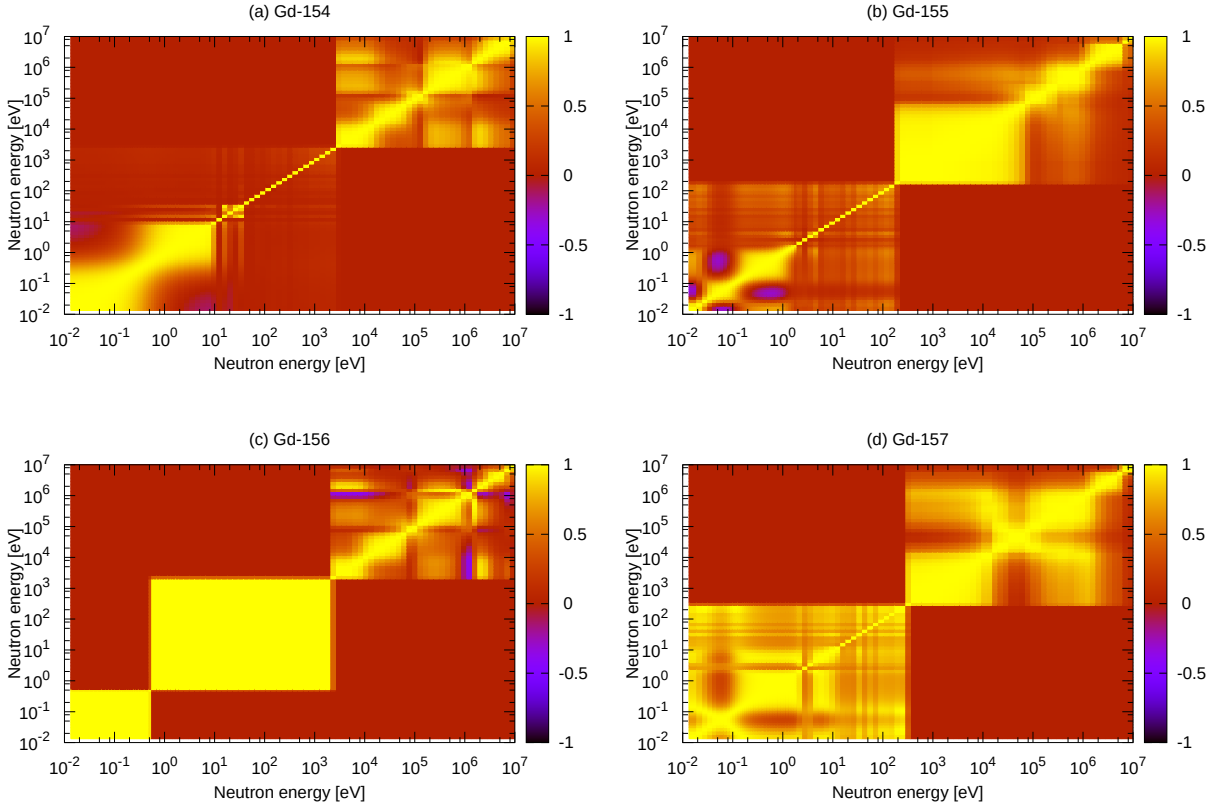

Figure 16: Correlation matrices of energy group-wise (n,γ) cross sections of gadolinium isotopes

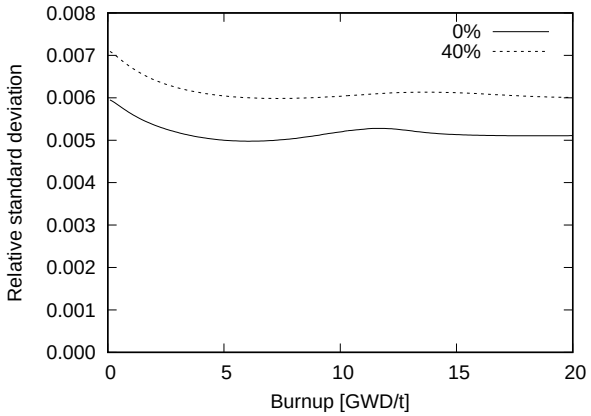

Figure 17: Nuclear data-induced uncertainty of k during nuclear fuel burnup in the 3×3 multicell model

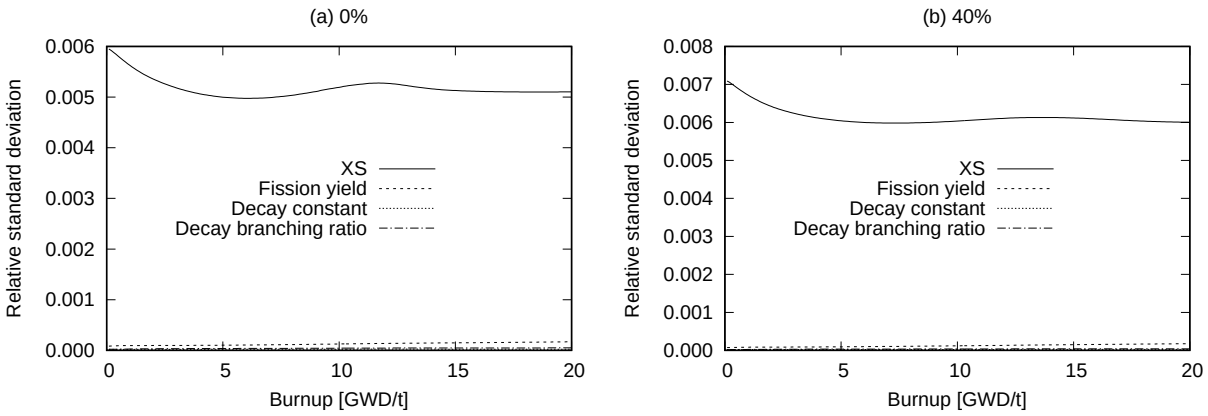

Figure 18: Component-wise nuclear data-induced uncertainty of k during nuclear fuel burnup in the 3×3 multicell model

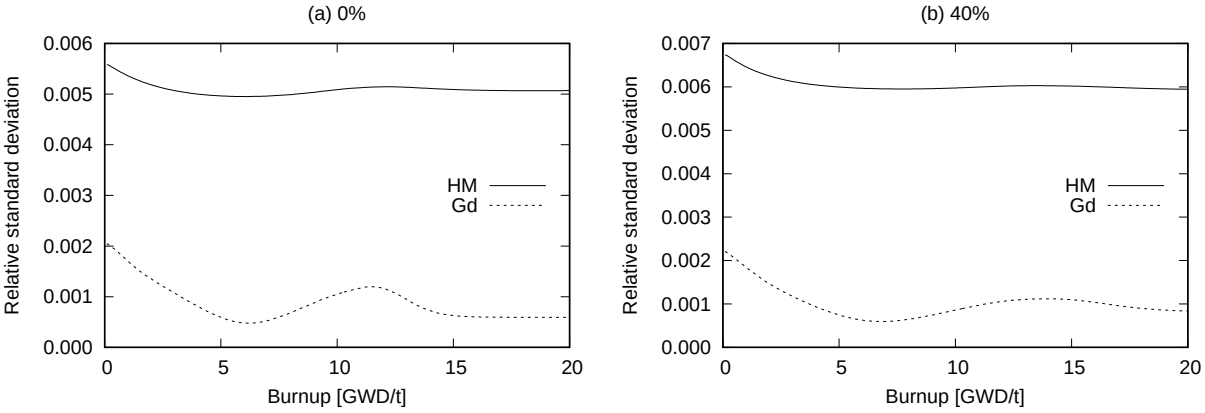

Figure 19: Reaction cross sections-induced uncertainty of k during nuclear fuel burnup in the 3×3 multicell model

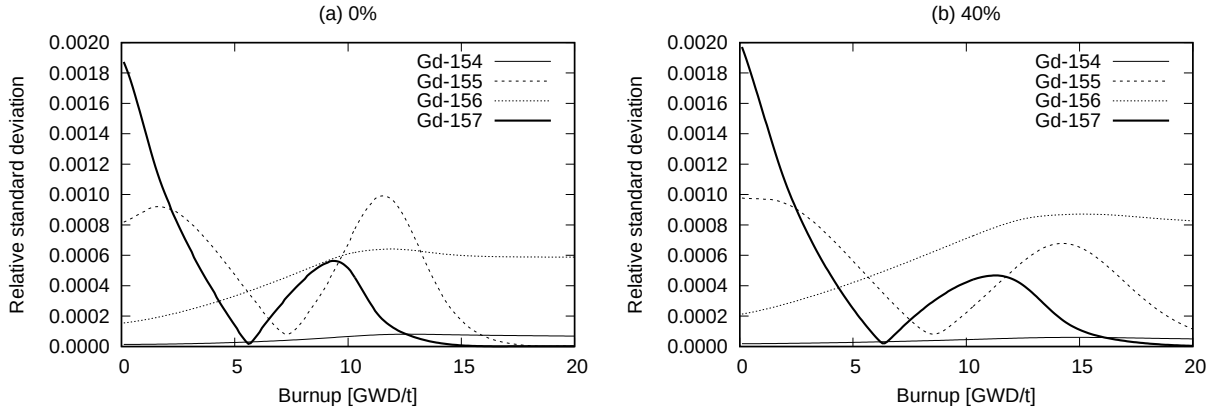

Figure 20: Gadolinium isotope-wise reaction cross sections-induced uncertainty of k during nuclear fuel burnup in the 3×3 multicell model

List of Figure Captions