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RAPID model to predict radial burnup distribution in LWR UO_2 fuel

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Abstract

The RAPID (RAdial power and burnup Prediction by following fissile Isotope Distribution in the pellet) model was developed to predict the radial distribution of power, burnup and fissionable nuclide densities in the LWR UO_2 fuel pellets to be used in the fuel performance analysis code. It considers the specific radial variations of the neutron reactions of all the fissionable nuclides as functions of burnup and ^{235}U enrichment in the pellet. Comparison of the RAPID prediction with the measured data of the irradiated fuels and the predicted results by other codes showed good agreement, and therefore, the RAPID model can be used for UO_2 fuel of up to 10 w/o ^{235}U enrichment and 150 MWD/(kg U) pellet average burnup under the LWR environment. © 2000 Elsevier Science B.V. All rights reserved.

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1. Introduction

Due to the radial variation of the neutron flux and energy spectrum inside UO_2 fuel, the fission density and the fissile isotope production rates vary radially in the pellet. During irradiation, fissile elements are generated by the neutron absorption of fertile elements while being destroyed by fission. Detailed prediction of those reactions has usually been performed by reactor physics codes such as HELIOS [1], WIMS-E [2] and APOLLO-2 [3], which take into account the neutron flux and neutron energy spectrum and their spatial variation. Since a detailed reactor physics code cannot be directly used in the fuel rod performance analysis code, a simple empirical model has been used instead to predict the radial power distribution. However, it has limitations in the prediction of local variations of power and burnup inside the pellet.

By the resonance neutron absorption of ^{238}U to become ^{239}Pu of the fissile nuclide near the edge of the fuel pellet, the local power and burnup at the pellet edge increase by a factor of 2–3 as the burnup increases [4,5]. This is called the 'neutronic rim effect'. For high burnup fuel, the variation of the local burnup becomes significant, and a high burnup structure (HBS) or rim structure may develop near the pellet edge when the local burnup is higher than the threshold burnup of around 60–80 MWD/(kg U). In the region where HBS develops, grains are finely divided into very small grains with a size of 0.1–0.5 μm, and fission gas bubbles with an average diameter of 1–2 μm are distributed uniformly between the subgrains [6]. Therefore, it is essential to accurately predict the local burnup in the pellet to analyze the neutronic rim effect in the fuel performance code; so the RAPID model, which can calculate the radial variations of local power and burnup by following the changes of the fissionable nuclides with the burnup and radial position, was developed [7].

2. Models of the RAPID program

The first developed program to model the buildup of ^{239}Pu near the pellet edge for the analysis of fuel performance is RADAR [8], which was validated by

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comparison with the general physics code, WIMS-E [2]. RADAR considers only ^{239}Pu, and other plutonium isotopes, such as ^{240}Pu, ^{241}Pu and ^{242}Pu, are not considered. However, the TUBRNP model [9] considers such plutonium isotopes as ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. To model the resonance neutron capture of ^{238}U near the pellet edge, a special radial function with an exponential term is employed for the absorption cross-section of ^{238}U, which is similar to the RADAR model. To model the radial variation of the neutron reactions of the nuclides, the neutron flux was derived based upon the one-group diffusion theory, while the neutron absorption and fission cross-sections of all the nuclides were assumed to be uniform radially. Therefore, except for the neutron absorption reaction of ^{238}U, all the other neutron reaction rates of all the nuclides were assumed to have the same radial variation in the pellet, regardless of nuclide, burnup and ^{235}U enrichment.

In the RAPID model, on the other hand, radial variation of the neutron reactions with burnup and ^{235}U enrichment are considered for all the fissionable nuclides, since there exist specific radial variations in the neutron reaction rates of uranium and plutonium isotopes such as ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. The HELIOS code [1] was used as the reference reactor physics code. The HELIOS code is based on a two-dimensional transport theory to calculate fuel burnup and gamma dose in the fuel assembly, and can calculate the radial variations of fissionable nuclides inside the pellet with more advanced features. HELIOS calculation was performed with a 35-energy library for the PWR case, and the total number of nodes inside the pellet was 10, with finer nodalization near the pellet edge. In the RAPID model, one-group neutron flux and neutron reaction cross-sections of the elements were used, and their values were derived from HELIOS prediction results. Dependence of the neutron reaction cross-sections and neutron flux upon the radial position, burnup and ^{235}U enrichment were considered for all the fissionable nuclides.

Fig. 1 shows the radial distributions of fissile nuclides such as ^{235}U, ^{239}Pu and ^{241}Pu with the burnup for 4 w/o ^{235}U fuel calculated by the HELIOS code. As the burnup increases, ^{239}Pu and ^{241}Pu build up, and at a higher pellet average burnup like ~60 MWD/(kg U), the ^{239}Pu nuclide concentration becomes higher than that of ^{235}U. At a pellet average burnup of ~90 MWD/(kg U), ^{241}Pu can contribute a certain fraction to the total fissile nuclides. Therefore, in the RAPID model, the specific models for ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu as well as ^{235}U and ^{238}U were used for more accurate prediction for the very high burnup fuel.

Based on the one-group and one-dimensional diffusion theory, the balance equations of the fissionable nuclides such as ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu can be set as follows:

$$\frac{\partial}{\partial t}N_{235}(t,r) = -N_{235}(t,r)\sigma_{\mathrm{a}}^{235}(t,r)\phi(t,r), \tag{1}$$

$$\frac{\partial}{\partial t}N_{238}(t,r) = -N_{238}(t,r)\sigma_{\mathrm{a}}^{238}(t,r)\phi(t,r), \tag{2}$$

$$\begin{aligned}\frac{\partial}{\partial t}N_{239}(t,r) = {} & -N_{239}(t,r)(\sigma_{\mathrm{a}}^{239}(t,r)\phi(t,r)+\lambda_{239}) \\ & + N_{238}(t,r)\sigma_{\mathrm{c}}^{238}(t,r)\phi(t,r),\end{aligned} \tag{3}$$

$$\begin{aligned}\frac{\partial}{\partial t}N_{240}(t,r) = {} & -N_{240}(t,r)(\sigma_{\mathrm{a}}^{240}(t,r)\phi(t,r)+\lambda_{240}) \\ & + N_{239}(t,r)\sigma_{\mathrm{c}}^{239}(t,r)\phi(t,r),\end{aligned} \tag{4}$$

$$\begin{aligned}\frac{\partial}{\partial t}N_{241}(t,r) = {} & -N_{241}(t,r)(\sigma_{\mathrm{a}}^{241}(t,r)\phi(t,r)+\lambda_{241}) \\ & + N_{240}(t,r)\sigma_{\mathrm{c}}^{240}(t,r)\phi(t,r),\end{aligned} \tag{5}$$

$$\begin{aligned}\frac{\partial}{\partial t}N_{242}(t,r) = {} & -N_{242}(t,r)(\sigma_{\mathrm{a}}^{242}(t,r)\phi(t,r)+\lambda_{242}) \\ & + N_{241}(t,r)\sigma_{\mathrm{c}}^{241}(t,r)\phi(t,r),\end{aligned} \tag{6}$$

where $N_i(t,r)$ is the atomic density of nuclide i (atoms/m^3), σ_{a}^{i} the effective neutron absorption cross-section of nuclide i (m^2), σ_{c}^{i} the effective neutron capture cross-section of nuclide i (m^2), λ_i the decay constant of nuclide i (s^{-1}) and $\phi(t,r)$ is the one-group neutron flux (n/cm^2 s).

The neutron reaction cross-section in the one-group approach represents the effective neutron reaction cross-section, which takes into account both the neutron energy spectrum and the energy-dependent neutron reaction cross-section. The neutron absorption cross-section is a sum of the neutron capture and fission cross-sections. In fact, there exist radial variations in both the neutron reaction cross-sections of the nuclides and the neutron fluxes with different energies. However, in the RAPID model, radial variation of the neutron reaction is modeled by assuming radial variation of the cross-section of each nuclide while letting the neutron flux of one energy group remain constant in the radial direction. On the other hand, the TUBRNP model assumes fixed radial distribution of the neutron flux regardless of burnup and ^{235}U enrichment while assuming the cross-sections of the nuclides, except the absorption cross-section of ^{238}U, to be constant independent of burnup and ^{235}U enrichment. Therefore, specific radial variations of neutron reactions of the different nuclides with respect to burnup and ^{235}U enrichment can be taken into account only by the RAPID model. Variation of reaction cross-sections with burnup results from the hardening of the neutron energy spectrum. It is caused by the absorption of thermal neutrons by the accumulated fission products with burnup.

Fig. 2 shows the radial distribution of one-group neutron flux with burnup, which is quite constant

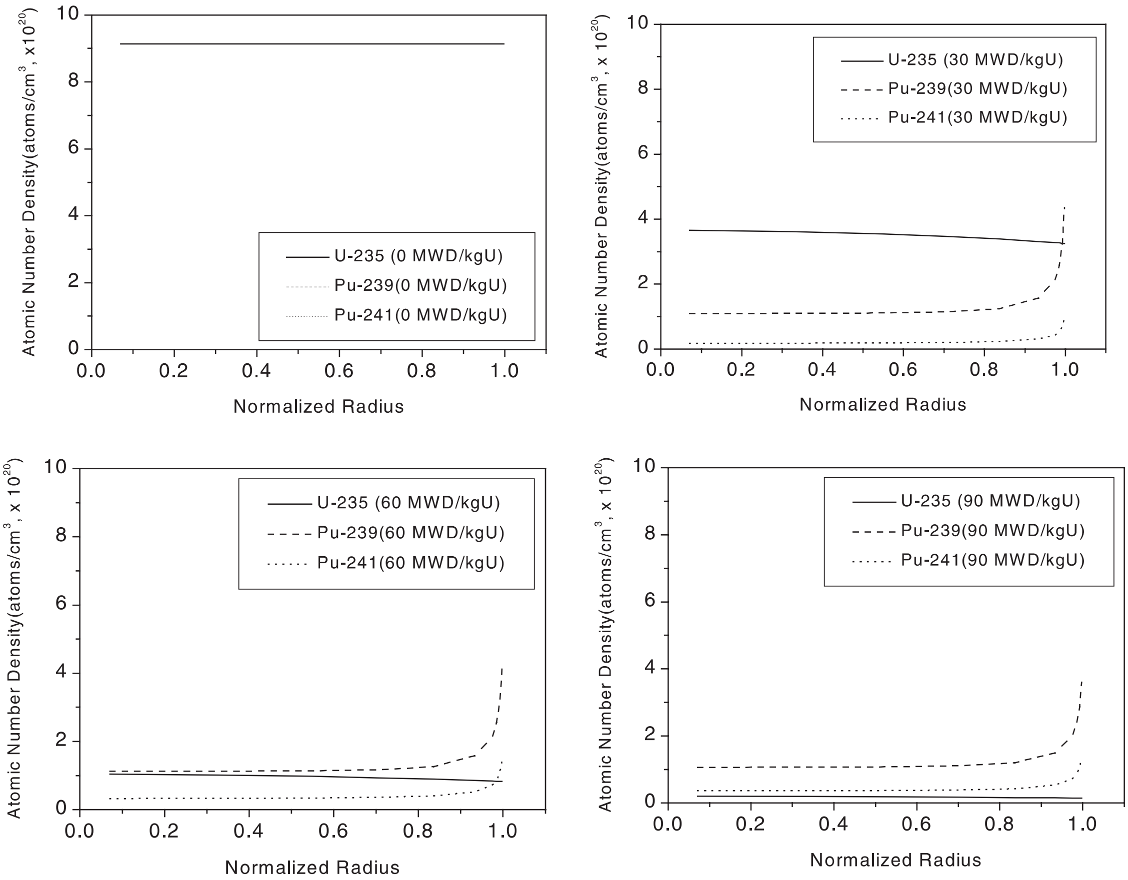

Fig. 1. Variation of radial fissile atomic density distribution with the burnup obtained by HELIOS calculation (4 w/o ^{235}U).

regardless of the radial position and burnup obtained by HELIOS calculation. Slight deviations in both central and peripheral regions which may result from the self-shielding effect of the thermal neutrons can be neglected. Therefore, one-group neutron flux, $\phi(t,r)$, can be assumed constant in the radial direction and be expressed as a function of the power level (POWDEN, w/cm^3), burnup (BU, MWD/(kg U)) and ^{235}U enrichment (EN, w/o) as follows:

$$\phi(t,r) = ((C_1^n + C_2^n\mathrm{EN} + C_3^n\mathrm{EN}^2) + C_4^n\mathrm{BU} + C_5^n\mathrm{BU}^2)\mathrm{POWDEN}, \tag{7}$$

where C_i^ns are constants.

The absorption cross-sections of ^{235}U, ^{239}Pu and ^{241}Pu and the fission cross-sections of ^{235}U, ^{239}Pu and ^{241}Pu, of which neutron reactions occur mainly by reaction with thermal neutrons, can be expressed by similar functions of radial position, burnup and ^{235}U enrichment. The jth neutron reaction cross-section of the ith nuclide, $\sigma_j^i(t,r)$, depends on burnup, ^{235}U enrichment and radial position, so that it can be separated by

$$\sigma_j^i(t,r) = f_j^i(r)\sigma_j^i(t), \tag{8}$$

where $f_j^i(r)$ and $\sigma_j^i(t)$ are the radial and time-dependent functions, respectively.

Since the radial function, $f_j^i(r)$ depends on ^{235}U enrichment, it is assumed to be a third-order polynomial of the radial variable and a first-order polynomial of ^{235}U enrichment as follows:

$$f_j^i(r) = C_1^a + C_2^a\mathrm{EN} + (C_3^a + C_4^a\mathrm{EN})r + (C_5^a + C_6^a\mathrm{EN})r^2 + (C_7^a + C_8^a\mathrm{EN})r^3. \tag{9}$$

The time-dependent function, $\sigma_j^i(t)$, can be expressed as a function of burnup and ^{235}U enrichment as follows:

$$\sigma_j^i(r) = (C_1^b + C_2^b\mathrm{EN} + C_3^b\mathrm{EN}^2)(C_4^b + (C_5^b + C_6^b\mathrm{EN})\mathrm{BU} + (C_7^b + C_8^b\mathrm{EN})\mathrm{BU}^2). \tag{10}$$

For the neutron absorption reaction of ^{238}U, the resonance capture of neutrons with energy in the epithermal region occurs near the pellet edge, as shown in Fig. 3. It decreases slightly with ^{235}U enrichment and increases slightly with burnup. Therefore, special treat-

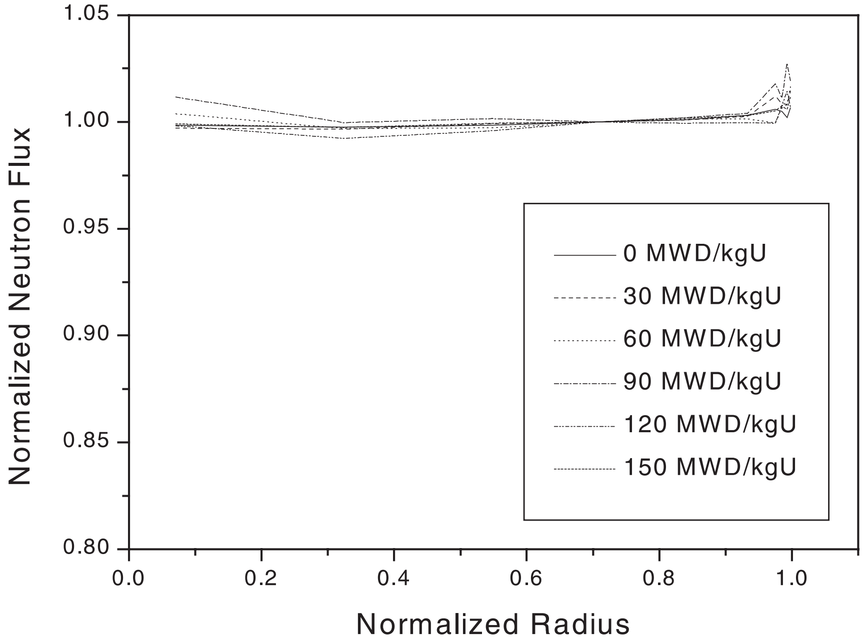

Fig. 2. Radial variation of one group neutron flux obtained by HELIOS calculation (4 w/o ^{235}U).

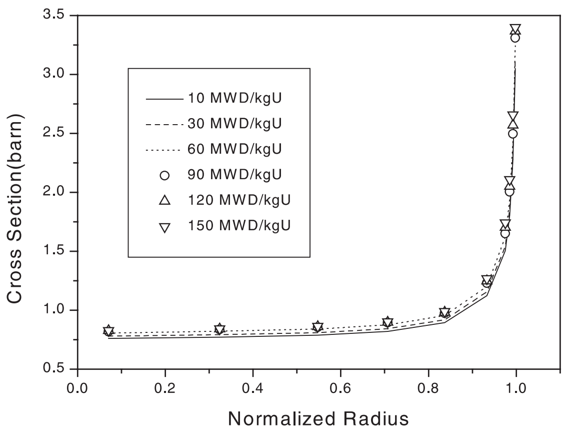

Fig. 3. Radial variation of ^{238}U absorption cross-section with the burnup obtained by HELIOS calculation (4 w/o ^{235}U).

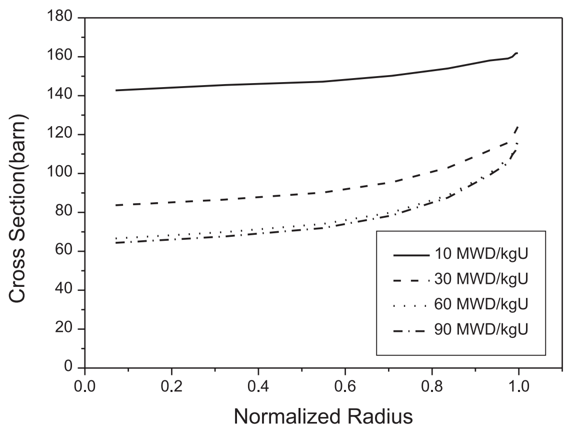

Fig. 4. Radial variation of ^{240}Pu absorption cross-section with the burnup obtained by HELIOS calculation (4 w/o ^{235}U).

ment is given to the radial dependence of the ^{238}U absorption cross-section as follows:

$$\sigma_a^{238}(t,r) = f_a^{238}(r)\sigma_a^{238}(t). \quad (11)$$

The radial function, $f_a^{238}(r)$ is set as an exponential function to simulate the decrease of neutron resonance absorption reaction from the outer region of the pellet by the self-shielding effect, which is similar to those used in the RADAR [8] and TUBRNP [9] models as follows:

$$f_a^{238}(r) = C_1^d + (C_2^d + C_3^d \mathrm{EN}) \exp\left(C_4^d (1-r)^{C_5^d}\right). \quad (12)$$

The time-dependent function, $\sigma_a^{238}(t)$, is expressed as a third-order polynomial of burnup and the first order of ^{235}U enrichment as follows:

$$\sigma_a^{238}(t) = (C_1^e + C_2^e \mathrm{EN})(C_3^e + C_4^e \mathrm{BU} + C_5^e \mathrm{BU}^2 + C_6^e \mathrm{BU}^3). \quad (13)$$

The absorption cross-section of ^{240}Pu changes considerably with burnup, as shown in Fig. 4. It results from the hardening of the neutron energy spectrum due to the buildup of fission products with burnup as explained previously. The neutron absorption cross-section of the

^{240}Pu nuclide strongly depends upon the energy level of the neutron. Therefore, it is expanded by the higher order polynomials for the variables of burnup, ^{235}U enrichment and radial position as follows:

$$\sigma_{\mathrm{a}}^{240}(t,r) = f_{\mathrm{a}}^{240}(\mathrm{BU},\mathrm{EN})(C_1^{\mathrm{BU}} + C_2^{\mathrm{BU}} r + C_3^{\mathrm{BU}} r^2 + C_4^{\mathrm{BU}} r^3), \tag{14}$$

where

$$\begin{aligned} f_{\mathrm{a}}^{240}(\mathrm{BU},\mathrm{EN}) = {} & (C_1^{\mathrm{f}} + C_2^{\mathrm{f}}\mathrm{BU} + C_3^{\mathrm{f}}\mathrm{BU}^2 + C_4^{\mathrm{f}}\mathrm{BU}^3) \\ & \times (C_5^{\mathrm{f}} + C_6^{\mathrm{f}}\mathrm{BU} + C_7^{\mathrm{f}}\mathrm{BU}^2 + C_8^{\mathrm{f}}\mathrm{BU}^3 \\ & + (C_9^{\mathrm{f}} + C_{10}^{\mathrm{f}}\mathrm{BU} + C_{11}^{\mathrm{f}}\mathrm{BU}^2 \\ & + C_{12}^{\mathrm{f}}\mathrm{BU}^3)\mathrm{EN}), \end{aligned}$$

$$\begin{aligned} C_1^{\mathrm{BU}} &= C_1^{\mathrm{g}} + C_2^{\mathrm{g}}\mathrm{BU} + C_3^{\mathrm{g}}\mathrm{BU}^2 + C_4^{\mathrm{g}}\mathrm{BU}^3, \\ C_2^{\mathrm{BU}} &= C_1^{\mathrm{h}} + C_2^{\mathrm{h}}\mathrm{BU} + C_3^{\mathrm{h}}\mathrm{BU}^2 + C_4^{\mathrm{h}}\mathrm{BU}^3, \\ C_3^{\mathrm{BU}} &= C_1^{\mathrm{l}} + C_2^{\mathrm{l}}\mathrm{BU} + C_3^{\mathrm{l}}\mathrm{BU}^2 + C_4^{\mathrm{l}}\mathrm{BU}^3, \\ C_4^{\mathrm{BU}} &= C_1^{\mathrm{m}} + C_2^{\mathrm{m}}\mathrm{BU} + C_3^{\mathrm{m}}\mathrm{BU}^2 + C_4^{\mathrm{m}}\mathrm{BU}^3, \\ C_i^j &= \text{constant}. \end{aligned}$$

Radial variation of the ^{242}Pu absorption cross-section also showed significant radial variation with burnup so that it was also correlated with a high-order function of radial position, burnup and ^{235}U enrichment.

For the fission cross-sections of the fertile nuclides such as ^{238}U, ^{240}Pu and ^{242}Pu, the fission reaction occurs by reaction with the high-energy fast neutron, which has a quite uniform radial distribution inside the pellet compared with the thermal neutron, and therefore, their fission cross-sections are quite constant in the radial direction and do not vary much with burnup either. Therefore, the fission cross-sections of ^{238}U, ^{240}Pu and ^{242}Pu are assumed to be constant with regard to radial position and burnup.

The calculation scheme of the RAPID model is shown in Fig. 5. Initially, time- or burnup-independent data such as pellet dimension, density and ^{235}U enrichment are calculated and radial nodalization is performed. Then, from the power history of the fuel, the pellet average burnup is calculated for each time step. The neutron flux and radial distributions of neutron reaction cross-sections of the fissionable nuclides are estimated from burnup, radial position and ^{235}U enrichment. Selected fitting constants for neutron flux and neutron reaction cross-sections of the nuclides under PWR condition are given as an example in Table 1. Then, radial distributions of the atomic densities of the nuclides are calculated. Finally, radial power distribution is estimated from local atomic densities and fission cross-sections, and the local burnup is obtained by integration of the power density.

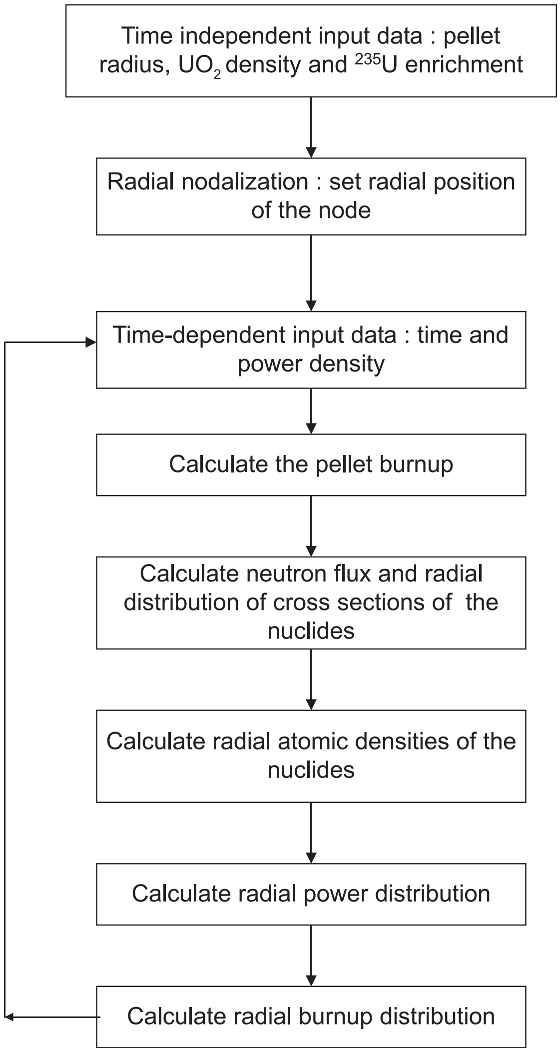

Fig. 5. RAPID calculation flow.

3. Evaluation and verification

By using the pellet data and the fuel irradiation history, RAPID calculates the radial distribution of power, burnup and atomic densities of fissionable nuclides such as ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. Fig. 6 compares variations of the radial burnup distribution with burnup for 4 w/o ^{235}U fuel predicted by HELIOS and RAPID, and shows their excellent agreement. Local peaking near the pellet edge is visible throughout the whole burnup range. Fig. 7 shows the dependence of radial burnup distribution upon ^{235}U enrichment of 3–10 w/o. Local power peaking near the pellet edge decreases with ^{235}U enrichment. There is good agreement between RAPID and HELIOS. Fig. 8 compares the plutonium isotope buildup at the mid-radius of the pellet as a

Table 1
Selected fitting constants for neutron flux and neutron reaction cross-sections under PWR condition

Neutron flux (n/cm^2 s)	C_i^n, $i = 1,5$	9.96E12, −5.59E11, 7.0E10, 6.73E9, −1.57E8
^{235}U fission cross-section (barn or 10^{-24} cm^2)	C_i^a, $i = 1,8$	0.987, −8.62E−3, −2.55E−4, 7.46E−3, 0.0132, 3.27E−3, 0.0226, 4.77E−3
	C_i^b, $i = 1,8$	8.18, −0.97, 0.0358, 6.72, 0.0231, 2.38E−3, −2.84E−4, 3.91E−5
^{238}U absorption cross-section (barn)	C_i^d, $i = 1,5$	0.92, 7.1, 0.069, −6.88, 0.329
	C_i^e, $i = 1,6$	0.94, −0.011, 0.9, 1.21E−3, −8.03E−7, −1.38E−8
^{240}Pu absorption cross-section (barn)	C_i^f, $i = 1,12$	183.2, −3.84, 4.18E−2, −1.42E−4, 1.14, −5.75E−3, 1.12E−4, −5.14E−7, −2.9E−2, 1.26E−3, −2.41E−5, 1.09E−7
	C_i^g, $i = 1,4$	0.992, −6.53E−3, 5.9E−5, −1.7E−7
	C_i^h, $i = 1,4$	−4.11E−2, 2.59E−2, −2.14E−4, 5.45E−7
	C_i^l, $i = 1,4$	0.195, −6.79E−2, 5.35E−4, −1.3E−6
	C_i^m, $i = 1,4$	−0.162, 0.061, −4.82E−4, 1.19E−6

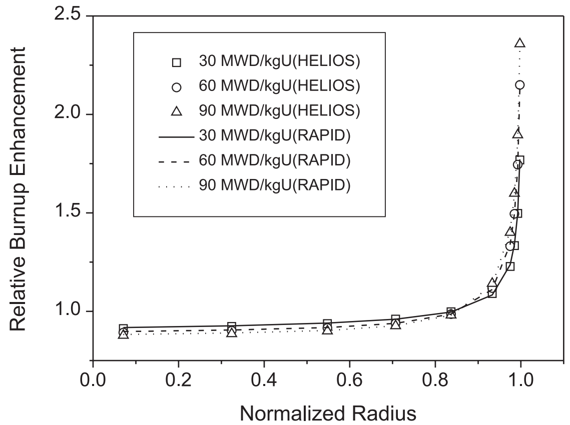

Fig. 6. Variation of radial burnup distribution with burnup (4 w/o ^{235}U).

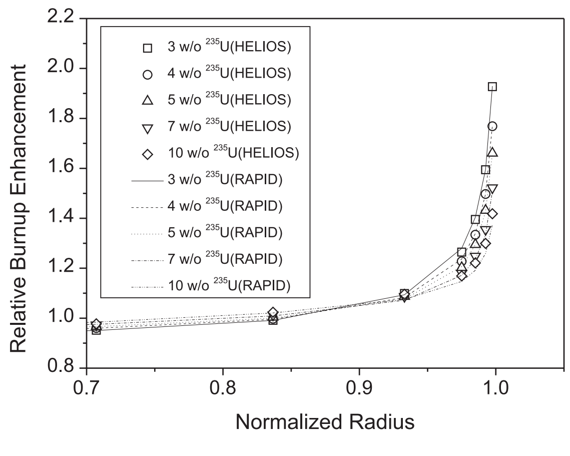

Fig. 7. Variation of radial burnup distribution with ^{235}U enrichment at the burnup of 30 MWD/(kg U).

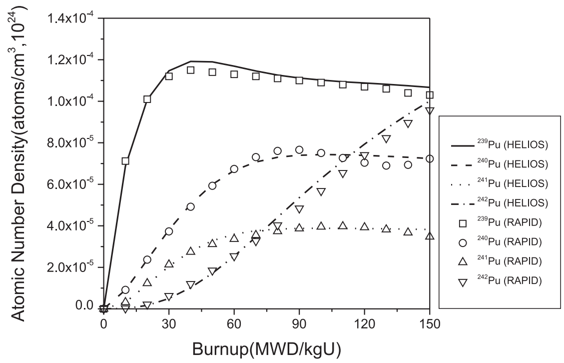

Fig. 8. Variation of atomic number density with the burnup at mid-radius (4 w/o ^{235}U).

function of the burnup predicted by the HELIOS and RAPID codes. There is a good agreement between them, which shows that variations of nuclide densities with burnup are well modeled in RAPID. Depending upon ^{235}U enrichment, the concentration of ^{239}Pu seems to saturate near a burnup of 25–40 MWD/(kg U), and those of ^{240}Pu and ^{241}Pu are saturated at 60–90 MWD/(kg U).

There exist a few measured data available for the radial distributions of plutonium isotopes and burnup of irradiated fuels. Measured data of GE and EPRI fuels were published by Lassmann et al. [9] in the validation of the TUBRNP model [9], so that they were used to evaluate the RAPID model along with the TUBRNP prediction results. The GE fuel was of BWR design and irradiated in a Millstone-1 power plant up to burnups of 23 and 39 MWD/(kg U). The EPRI fuels were of PWR design and irradiated in the BR3 reactor up to a burnups of 39.4 and 64 MWD/(kg U). Figs. 9 and 10 compare the predicted results of the radial distributions of total

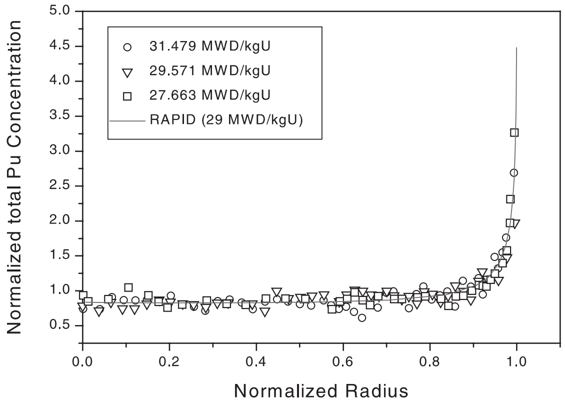

Fig. 9. Comparison of radial distribution of total Pu concentration with the measured GE fuel data at the burnup of 29 MWD/(kg U) [9].

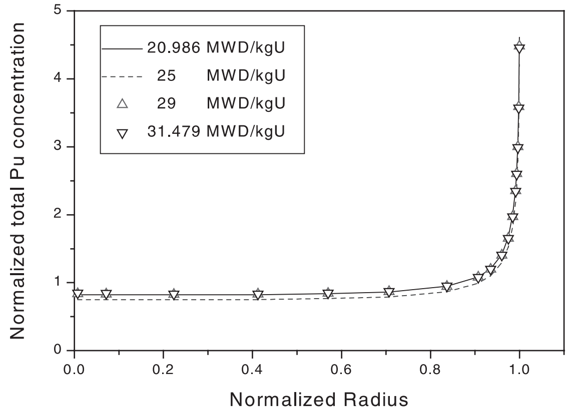

Fig. 11. Burnup dependence of radial distribution of total Pu concentration for GE fuel.

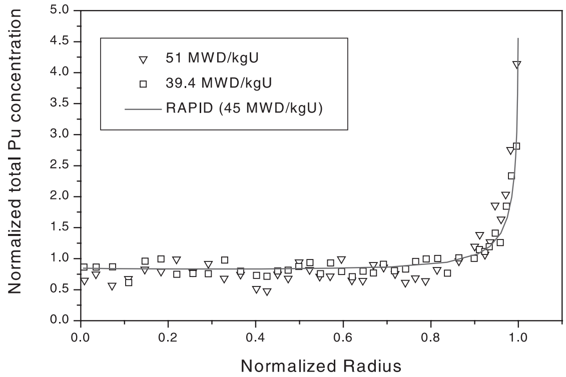

Fig. 10. Comparison of radial distribution of total Pu concentration with the measured EPRI data of 8.25 w/o ^{235}U at the burnup of 45 MWD/(kg U) [9].

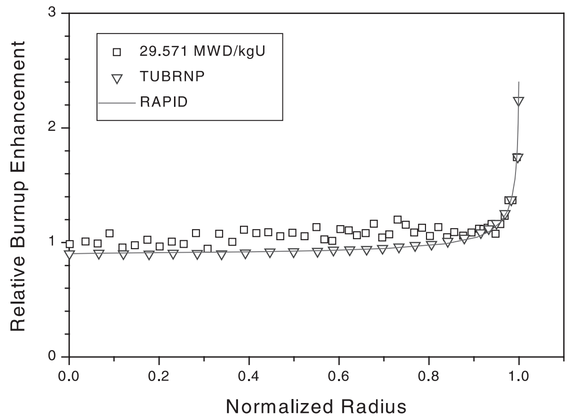

Fig. 12. Comparison of radial burnup distribution with the measured GE (2.9 w/o ^{235}U) data and predictions by RAPID and TUBRNP at the burnup of 29.571 MWD/(kg U) [9].

plutonium concentration by RAPID with the measured data of GE and EPRI fuels. There is a rather good agreement within the scatter band of the measured data. As can be seen from Fig. 11, the radial distribution of total plutonium buildup is quite invariant to burnup, and its minor variations are rather random. Therefore, the radial distribution of total plutonium concentration seems not to be a good indicator for the burnup dependence of the models. Fig. 12 compares the radial burnup distribution for the GE fuel (2.9 w/o ^{235}U) at a burnup of 29.571 MWD/(kg U). There is quite good agreement among RAPID, TUBRNP and the measured data within the scatter band of the measured data. Fig. 13 compares the radial burnup distribution of RISO (2.95 w/o ^{235}U) at a burnup of 44.105 MWD/(kg U) [9], which shows also a good agreement among them.

Fig. 14 shows the effect of ^{235}U enrichment upon radial burnup distribution, especially near the pellet edge. As shown in Fig. 7, depending on the ^{235}U

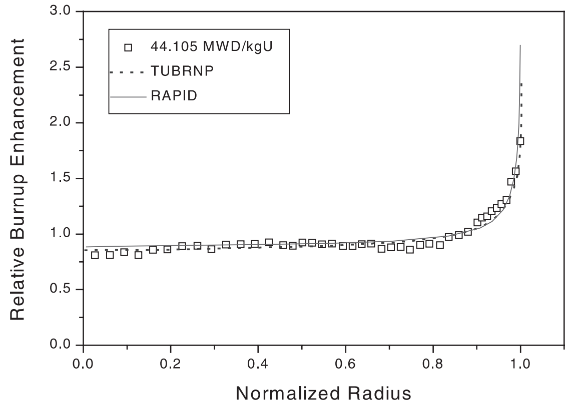

Fig. 13. Comparison of radial burnup distribution with the measured RISO (2.95 w/o ^{235}U) data and predictions by RAPID and TUBRNP at the burnup of 44.105 MWD/(kg U) [9].

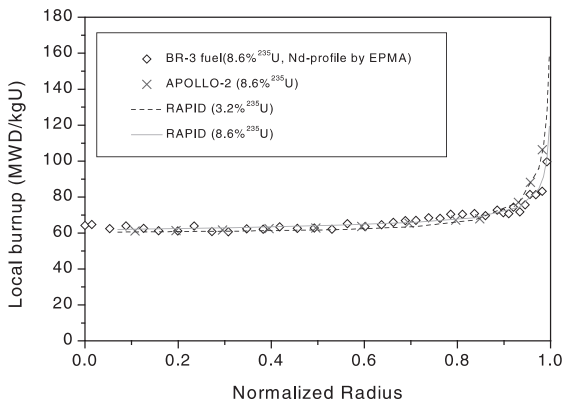

Fig. 14. Effect of ^{235}U enrichment upon radial burnup distribution in comparison with the measured BR-3 fuel (8.6 w/o ^{235}U) data [10].

enrichment, there are large variations near the pellet edge, where the high burnup structure first develops. Prediction by the RAPID model shows good agreement with the measured data and even better than that by the reactor physics code, APPOLLO-2 [10].

Fig. 15 compares measurements and predictions calculated by RAPID and TUBRNP for burnup dependence of pellet-averaged total plutonium of the GE fuel. It can be seen that up to a burnup of ~40 MWD/(kg U), the predictions of RAPID and TUBRNP are within the scatter band of the measured data. However, as the burnup increases, the value predicted by RAPID becomes higher than that by TUBRNP. Even though there are no data available for the high burnup fuel, it is expected that, since RAPID has more detailed models for ^{240}Pu, ^{241}Pu and ^{242}Pu nuclides than TUBRNP, such as the burnup dependence of the neutron reaction cross-sections of those nuclides, and the relative fractions of those nuclides among total fissile inventories increase

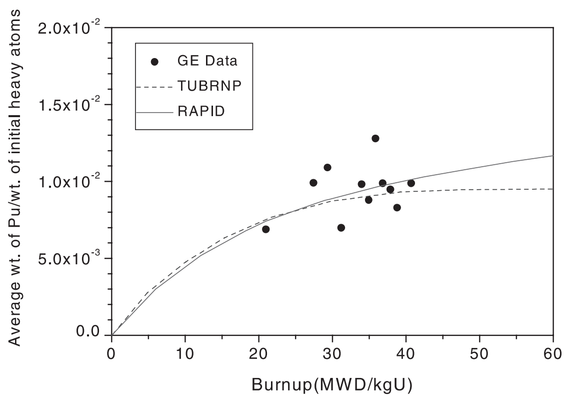

Fig. 15. Comparison of average Pu concentration with the measured GE fuel (2.9 w/o ^{235}U), RAPID and TUBRNP [9].

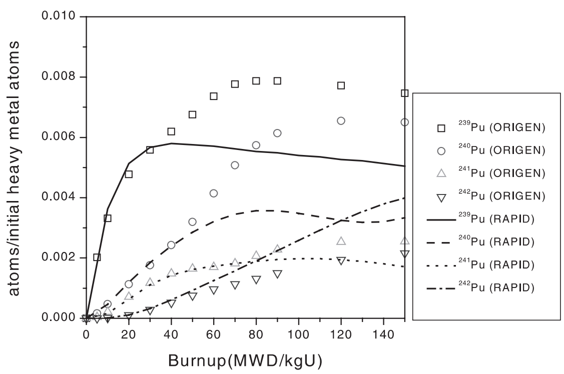

Fig. 16. Comparison of RAPID and ORIGEN calculations for buildup of pellet-averaged Pu during irradiation (4 w/o ^{235}U).

with the burnup, the predicted result of RAPID may be more accurate than that of TUBRNP, which has a simplified model for plutonium isotopes.

Fig. 16 compares total plutonium concentrations in the pellet predicted by RAPID and ORIGEN [11] codes for PWR fuel. There is a reasonable agreement up to a burnup of 40–60 MWD/(kg U). However, at burnup higher than that, the difference between them gets bigger with burnup. In the ORIGEN code, generation and fission of the plutonium nuclides are treated by one neutron energy group, and their dependence upon burnup and ^{235}U enrichment as well as radial position are not considered. Therefore, predictions of the RAPID model, which takes into account the dependence of each fissionable nuclide upon radial position, burnup and ^{235}U enrichment, and is validated by comparison with the detailed neutron reactor physics code, HELIOS, should be more reliable and applicable than those of the ORIGEN code.

4. Conclusions

The RAPID model to predict the radial distributions of power, burnup and fissionable nuclide densities in UO_2 pellets with burnup and ^{235}U enrichment was developed. It is based on and validated by the reactor physics code, HELIOS. The RAPID model considers specific radial variations of neutron reactions of nuclides with burnup, so that it may be more accurate than other programs with simplified models, specially in the high burnup region. The comparison of RAPID predictions with measured data of irradiated fuels showed good agreement. Therefore, RAPID can be used to calculate local variations of fissionable nuclide concentrations as well as local power and burnup inside the pellets as a function of burnup up to 10 w/o ^{235}U enrichment and 150 MWD/(kg U) pellet average burnup under the LWR environment.

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