Nuclear Engineering and Technology 52 (2020) 2415–2421

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original Article

Evaluation of coolant density history effect in RBMK type fuel modelling

Aurimas Tonkunas*, Raimondas Pabarcius, Andrius Slavickas

Lithuanian Energy Institute, Lithuania

A R T I C L E I N F O

Article history:
Received 24 November 2019
Received in revised form
27 March 2020
Accepted 10 April 2020
Available online 17 April 2020

Keywords:
History effect
Water density
RBMK
WIMS

A B S T R A C T

The axial heterogeneous void distribution in a fuel channel is a relevant and important issue during nuclear reactor analysis for LWR, especially for boiling water channel-type reactors. Variation of the coolant density in fuel channel has an effect on the neutron spectrum that will in turn have an impact on the values of absolute reactivity, the void reactivity coefficient, and the fuel isotopic compositions during irradiation. This effect is referring to as the history effect in light water reactor calculations. As the void reactivity effect is positive in RBMK type reactors, the underestimation of water density heterogeneity in 3D reactor core numerical calculations could cause an uncertainty during assessment of safe operation of nuclear reactor. Thus, this issue is analysed with different cross-section libraries which were generated with WIMS8 code at different reference water densities. The libraries were applied in single fuel model of the nodal code of QUABOX-CUBBOX/HYCA. The thermohydraulic part of HYCA allowed to simulate axial water distribution along fuel assembly model and to estimate water density history effect for RBMK type fuel.

1. Introduction

Many nuclear reactors have a rather strong coupling between coolant density and neutronic behaviour. For example, in LWRs cores, in which the water coolant also serves as a moderator, a local decrease in water density will cause a decrease in moderation and hence a decrease in local power density. As a coolant passes up through the core, it absorbs the heat from the fuel elements and eventually will initiate subcooled/bulk boiling. Hence, there will be an increase in coolant void fraction. The formation of steam void in the core plays a significant role in BWRs [1]. A similar, but weaker and opposite playing, coupling between the coolant properties and core neutronic behaviour can arise in non-water reactors, for example, sodium fast reactors. Because liquid metal, which serves as coolant, causes significant additional moderation or softening of neutron energy spectrum in fast reactor core, a local reactivity is decreased, hence a coolant density decrease leads to an increase in local reactivity/power density. Such thermal-neutronic coupling that arises in LWRs and SFRs is much more weaker in the gas-cooled reactors because the coolant phase change does not occur, and also because the coolant does not provide significant moderation in the core due to its very low density. Thereby BWR core physics are more complex than physics of other reactors when the coolant enters the core in a single phase and different two-phase flow regimes with very strong axial dependence are developed rapidly. It is worth to mention that a reactor core physics becomes even more complicated in recent years because modern fuel assemblies have new sophisticated features (the inner by-pass regions, part-length rods), introducing larger uncertainties in relation to axial void distributions and their impact on integral reactor physics parameters [2]. The axial heterogeneous void distribution is very actual for channel-type reactors (CANDU, RBMK) cooled by light water. Such variation of coolant density in fuel channel may have an effect on the spectrum and, hence, the fuel depletion characteristics. This effect is referred to as a history effect in LWR calculations, of importance in prediction of reactor transient behaviour. Thus, the question could be raised– can it be estimated accurately by using an average coolant density of 0.5 g/cm^3 for all burnup calculations, as it is a standard approach for the RBMK-1500 reactor? Such discussion is the subject of this paper. An object of this investigation – the RMBK reactor – was selected because such channel-type reactor represents specific axial water density heterogeneity in the core, and authors have accumulated experience on the peculiarities of such reactor type.

* Corresponding author.
E-mail address: aurimas.tonkunas@lei.lt (A. Tonkunas).

https://doi.org/10.1016/j.net.2020.04.015

The RBMK is a boiling light water reactor [3] therefore the processes identified for BWR are similar. Nevertheless, the presence of channels and graphite as a moderator could influence other processes, not identified for vessel-type reactors. The purpose of this paper was to determine the importance of the water density history effect on the RBMK-type reactor and to quantify the impact of this on the values of absolute reactivity, the void reactivity effect, and the fuel isotopic compositions during irradiation. A separate calculation has been performed to generate individual cross-section data for each axial region, each with a different reference coolant density. For this purpose lattice code WIMS8 was employed. The evaluation is made on the Ignalina NPP RBMK-1500 reactor fuel assembly (FA) design, described in Section 2. Calculation results presented in this paper cover the investigations of two types fuel cells: loaded with 2.6% uranium-erbium poisoned fuel (U–Er) and fuel cell without burnable poison (2% enriched uranium dioxide fuel). 3D neutronic QUABOX/CUBBOX code was used and the brief description is placed in Section 3. Reference parameters selected for the preparation of cross-section libraries that were used during this investigation are discussed in Section 4. The results of analysis are given in Section 5. First, the variation of reactivity with fuel irradiation for each value of the water density has been investigated. Then, void reactivity effect has been determined for a change from three different states to void, and its variation with fuel irradiation for selected value of the water density was examined. In this context, the calculation results were compared to those obtained accounting thermal hydraulic feedbacks. Finally, an examination of the variation with irradiation of the fuel nuclide compositions for different values of the water density was performed.

2. RBMK fuel assembly design

The RBMK reactor is a channel-type graphite-moderated boiling water reactor [3]. It has a graphite block structure as the moderator that slows down the neutrons produced by fission. The reactor core has a 7.0 m fuel region and a 0.5 m reflector region above and below the fuel region. The feature of RBMK type reactor is that each FA (1661 vertical tubes overall in RBMK-1500) is positioned in its own vertical fuel channel, which is individually cooled by pressurized water. The water is supplied to fuel channels where it is heated to saturation and partially evaporates. The RBMK FA consists of two fuel bundles placed one above another. The fuel bundle contains a total of 18 fuel rods arranged in two rings, with six rods in the inner ring and twelve rods in the outer ring. The rods are arranged around a central carrier rod and held in place with stainless steel spacers separated by a distance of 36 cm. An active height of the fuel assemblies is 682 cm. The fuel elements are located within a graphite block in the form of rectangular parallelepipeds with a base of 25 × 25 cm.

The FA is located within a vertical pressure tube where the coolant density varies sharply (Fig. 1). Such water density profile in BWR design is smoother [4]. The water density axial profile has a breaking point at which water boiling commences. The position of the boiling point is different in each fuel channel as each fuel channel has a different power. The water density below the boiling point is relatively constant.

Nowadays, reactors of RBMK design are operated only in Russia, 10 units at Kursk, Leningrad and Smolensk NPPs [5]. In Lithuania, the last unit of the most powerful RBMK at Ignalina NPP rated at 4200MWth was shut down at the end of 2009, while the first unit in 2004. Both RBMK-1500 reactors installed at Ignalina NPP belong to the second-generation RBMKs. During the entire duration of RBMK-1500 units operation the reactor core safety was constantly upgraded, with the aim of improving the undesirable features of

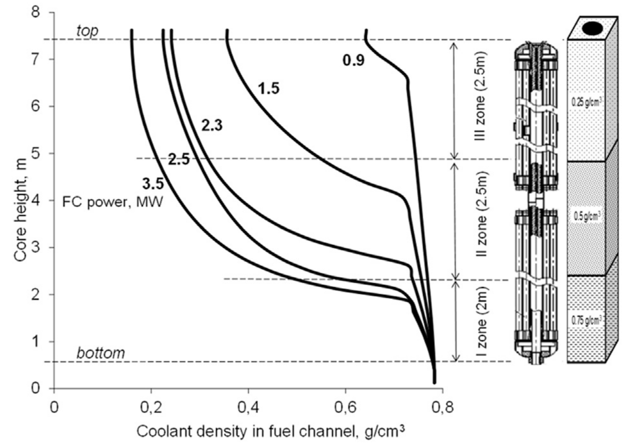

Fig. 1. Water density profiles within RBMK fuel channels operated at different power levels.

the Soviet-designed RBMK reactor cores; namely, to decrease the magnitude of the positive void reactivity coefficient and the positive reactivity insertion that occurred during reactivity initiated accidents [6]. This was achieved by implementing new types of fuel assemblies with a higher uranium enrichment and by inclusion of erbium poison, combined with the implementation of new control rod assemblies with an advanced design. The uranium fuel of 2% ^{235}U enrichment (U), and 2.4%, 2.6%, 2.8% ^{235}U enrichment with burnable erbium absorber were used during operation of RBMK-1500.

3. Applicable software for analysis

The QUABOX/CUBBOX-HYCA code [7] was employed in this research study to simulate the representative single fuel model of RBMK-1500 reactor core. The code was developed originally by GRS for core calculations of PWR and BWR type reactors. Since 1990 the code was adapted to the features of RBMK-1000 reactors and since 1995 additionally adapted to account for the specifics of RBMK-1500 reactors [8]. When Ignalina NPP was in operation the code was continuously used for various RBMK-1500 core calculations, as well as for the surveillance of the reactivity behaviour during the changing of the reactor core loading. QUABOX/CUBBOX-HYCA code was used for audit calculations during the review of the Ignalina NPP Safety Analysis Report (SAR) for Unit 1, for the preparation of SAR for Unit 2 and for the independent assessment of new installed Diverse Shutdown System at Unit 2.

The neutronic part of the code solves the two-energy group neutron diffusion equations in Cartesian geometry considering the reactivity feedback caused by changes of coolant flow conditions and by variations of fuel rod temperatures. The used flux expansion method based on the spatial neutron flux approximation by local polynomials of up to the 4th order results in a high accuracy of the spatial flux distribution even for a coarse mesh. The integration of the time dependent equations is done by a stable implicit matrix decomposition technique [9]. For the determination of thermal hydraulic feedback parameters the reactor core model includes the fuel rod model and the thermal-hydraulic module HYCA, which provide fuel and coolant temperatures and densities, as well as moderator (graphite) temperatures for accounting the feedback effects. The two-group library used with the QUABOX/CUBBOX-HYCA code, adapted to model the physical and nuclear processes in RBMK-1500 reactor core, was created using the WIMSD reactor physics cell code and its associated nuclear data library (prepared by the Russian Research Centre

"Kurchatov Institute" using the ENDF/B based library). The verifiable sensitivity and uncertainty analysis showed that measured void and power reactivity coefficients of Ignalina NPP are within tolerance limits of calculated results [10].

2 group cross section libraries preparations for this study have been performed with WIMS8 [11,12] using the JEF2.2 based library. The neutron flux was calculated for a few spatial regions in the full 172 energy group scheme. The energy group sequence in the energy decrease order was subdivided as follows: 92 epi-thermal energy groups, where 45 fast and 47 resonance energy groups are represented, and 80 thermal energy groups. The boundary energy value between the thermal and epi-thermal energy groups was taken to be 4 eV.

3.1. 2-Group library generation scheme

For analysis of history effect the two-group cross section libraries for fuel channels with two different fuel loading and assuming different reference conditions were prepared by employing WIMS8 code. The reference parameters for 2% U and 2.6% U–Er fuel are presented in Table 1.

The libraries were created for 11 burnup steps, starting from 0.1 MW days/kg and then follows 3, 6 and so on with the burnup step increase by 3 until 30 MW days/kg. Reference water and fuel temperatures were assumed 550/557/565 K and 870/1000/1150 K for water density values 0.75/0.5/0.25 g/cm^3 respectively, and Xe concentration was assumed at equilibrium. The temperature of graphite was assumed the same 750 K for all investigated cases because of inertia.

4. Calculation results

According to applicable practice for modelling the processes in RBMK-1500 reactor core it is used the 2 group cross section library prepared at reference water density of 0.5 g/cm^3 as initial for the whole fuel depletion range. Actually, the coolant density in axial direction varies from ~0.78 g/cm^3 in the bottom third of the fuel channel up to ~0.25 g/cm^3 in the top third (Fig. 1), and this would lead to different fuel isotopic composition during irradiation process due to different neutron spectrum along the FA. Consequently, it should have an effect on the modelling results as a core reactivity, various reactivity effects or coefficients. However, in practice a few-group constants frequently depend on relatively few parameters involving core temperatures and material densities in general. Hence, it is far more efficient to construct tables of the few-group constants for several values of these parameters, and then to use interpolation schemes to evaluate the group constants when necessary. Such approach was employed in this study to determine the importance of mentioned history effects for the RBMK reactor. Realizing that water density distribution curve has one extreme breaking point along axial direction (Fig. 1), there are two possibilities to represent the axial variation in water density:

Table 1
Reference parameters for preparation of the cross section libraries (WIMS8).

Parameter	2% (0.25) 2.6% (0.25)	2% (0.50) 2.6% (0.5)	2% (0.75) 2.6% (0.75)
Fuel burnup, MW d/kg	0 ÷ 30		
Water density[a], g/cm^3	0.001, 0.25	0.001, 0.50	0.001, 0.75
Water temperature, K	565	557	550
Fuel temperature, K	870	1000	1150
Graphite temperature, K	750		
Xe concentration	equilibrium		

[a] Only two water density points for each cross section preparation were assumed.

- water density in the lower axial region from the water inlet to the boiling point can be fixed, and then the remaining interval can be divided into *n* axial regions. This segmentation would be different for each fuel channel, depending on fuel channel power;
- the entire core height is divided into fixed *n* axial regions, i.e. the segmentation is the same for all fuel channels.

The first approach is more accurate as the real position of the boiling point can be determined for each individual fuel channel. However, according to the calculation route the flux within each QUABOX/CUBBOX core model node is determined applying a two-group cross-section library at first. It is difficult to describe the real position of the boiling point within each RBMK-1500 fuel channel. Therefore, it is only possible to divide the core height into a fixed number of axial regions, and such division is uniform for all fuel channels. This option has therefore been adopted during the investigation of water density history effects.

For estimation of the water density axial distribution in fuel channels a real database of Ignalina NPP Unit 2 operating at nominal power was used. At that time almost whole reactor core was loaded with uranium erbium fuel, i.e. 49/1356/200 fuel channels were loaded with mixed 2.4/2.6/2.8% U–Er type fuel, respectively. Other fuel channels contained 2% U FA. There were more than 80% fuel assemblies with 2.6% U–Er type fuel in the core. Taking into account that the fuel is placed in separate channels the water density distribution depends on power in each fuel channel. When analysing selected core state, it was determined that the magnitude of maximal/average/minimum powers of single fuel channel is 3.5/2.5/0.9 MW respectively. The number of fuel channels with a power greater than 2 MW was 1308 (~80%). The number of fuel channels with a power less than 1.5 MW was 100 (~6%), and these fuel channels were located at the periphery of the core near the radial reflector (Table 2).

As presented in Fig. 1, the position of boiling point is directly related to the fuel channel power. Water boiling begins in the upper part of the core if the fuel channel power is less than the core average value. 80% of the fuel channels in the core have a power that is greater than 2 MW and so the boiling point in these channels is located in the range from 1.4 to 2.8 m from the bottom of the core. The first axial region can therefore be defined to have a height of 2 m, starting from the bottom of the core. The remaining section of the channel has then been divided into two equal parts, each 2.5 m in length. The water density varies from 0.78 to 0.73 g/cm^3, from 0.73 to 0.3 g/cm^3, and from 0.3 to 0.2 g/cm^3 in each of the three selected axial regions, respectively. This provides for corresponding average values for the water density of 0.755, 0.515 and 0.25 g/cm^3, respectively. The following reference water density values of 0.75, 0.5 and 0.25 g/cm^3 were used in this study for libraries preparation.

For the explicit modelling of the axial variation in the coolant density a 3D QUABOX/CUBBOX single fuel cell model (Fig. 1) was established where FA with pressure tube is placed within a graphite block in the form of rectangular parallelepipeds with a base of 25 × 25 cm and height of 700 cm which represents the active fuel part. Single fuel cell model is divided into 28 axial nodes with height of 25 cm each. The model was divided into three axial regions which were assigned with corresponding 2 group cross section libraries, prepared at different refence coolant densities (i.e. 0.75, 0.5, 0.25 g/cm^3). Two types of fuel (2.0% U and 2.6% U–Er fuel) were analysed with the objective to determine the effect of the axial heterogeneity of water density in the fuel channel on reactivity parameters and fuel isotopic composition versus burnup. The average burnup of 2.0% U type FA equals to 1695 MWd/FA and for 2.6 U–Er FA type equal to 2638 MWd/FA respectively at Ignalina NPP.

Table 2
RBMK-1500 fuel channel power distribution at nominal power level.

Power, MW	≤0.5	0.51 ÷ 1.0	1.01 ÷ 1.5	1.51 ÷ 2.0	2.01 ÷ 2.5	2.51 ÷ 3.0	>3.0
Amount of fuel channels, %	0	0.5	5.6	15.0	29.9	35.8	13.3

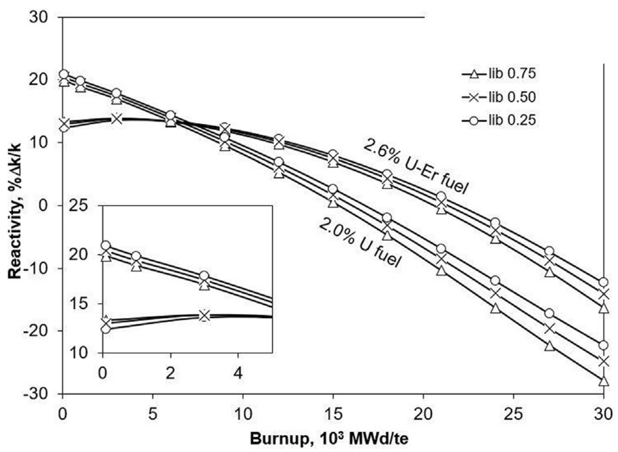

Fig. 2. Reactivity variation in fuel cell during irradiation (uniform water density profile).

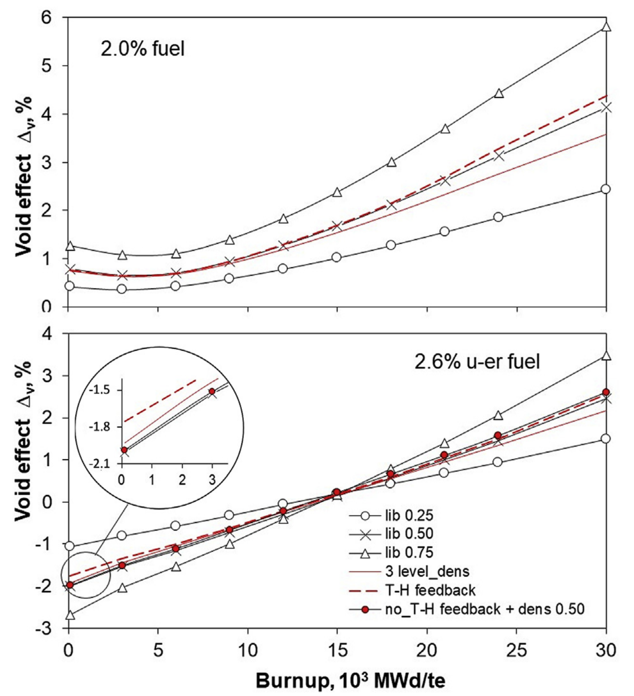

Fig. 3. Variation of void reactivity effect during fuel irradiation (fuel cell model).

4.1. Effect on system reactivity

The reactivity variation in the fuel cell during irradiation applying different uniform water density axial profiles for both analysed fuel types is shown in Fig. 2. The results indicate the magnitude of the impact on the depletion characteristics due to the coolant density change. Reactivity was defined as $\rho_{inf} = 1\text{-}1/K_{inf}$ in %, where K_{inf} is infinite multiplication factor. The trend of curves representing 2.6% U–Er fuel is different compared to 2% U case up to 12000 MWd/te because of burning process of erbium in the fuel. The differences between the predicted values of reactivity for each analysed water density value become more important with fuel irradiation increase. At beginning of fuel irradiation there is a small discrepancy between *lib0.25* and *lib0.75* in absolute value and does not exceed 1.0 $\%\Delta K_{inf}/K_{inf}$ (Fig. 2). During fuel irradiation the discrepancy increases for both fuel comparing results of the previous mentioned libraries and reach maximal discrepancy in absolute value up to 5.6 $\%\Delta K_{inf}/K_{inf}$.

4.2. Effect on coolant void reactivity effect

The coolant void reactivity effect Δ_v has been determined as the reactivity change from the state with fixed water density to void, and expressed as ratio $\Delta_v = 1 - K^{(2)}{}_{inf}/K^{(1)}{}_{inf}$. The state (1) represents fuel cell with water density 0.25, 0.50 or 0.75 g/cm^3, and state (2) when the water density is 0.001 g/cm^3 (void case). The calculation results for both analysed fuel types are presented in Fig. 3. Note, the uniform axial water density profile was selected for cases illustrated in the figure as the curves *lib0.75, lib0.50, lib0.25*. Again, the information presented in Fig. 3 indicates the magnitude of the impact on the depletion characteristics, and hence on the void reactivity effect due to variation in coolant density. Calculation results show, the observed differences are dependent on the fuel type. For the case of fuel cell with 2.0% U fuel Δ_v is directly proportional to the magnitude of the change in the water density and the discrepancies increase with fuel irradiation. For 2.6% U–Er type FA Δ_v is more positive at low fuel irradiation for smaller changes in the water density. This trend reverses at higher irradiation (>15 10^3 MWd/te), where Δ_v is now more positive for larger changes in the water density.

Coolant void reactivity effect Δ_v was also calculated for two additional cases when the fuel channel was divided into three axial regions with different water densities (Fig. 3). The curve named '*3level_dens*' corresponds to a QUABOX/CUBBOX calculations when the HYCA module is switched off (without thermal hydraulic calculations), and applied cross-sections correspond to those of the water density in each axial node (see Fig. 1). Another curve named '*T-H feedback*' corresponds to a QUABOX-CUBBOX calculation set when HYCA module is switched on, and a thermal hydraulic calculations are considered. The cross-sections used in calculations are then interpolated between two available values of the water density for each axial node according to the calculated in advance axial power profile and resulting profile for the coolant density. Analysing results in Fig. 3, as it is presented that the value of the void reactivity effect obtained assuming a uniform water density of 0.5 g/cm^3 (curve *lib0.50*), is in good agreement with '*T-H feedback*' case, where the axial variation in the water density is explicitly represented by interpolation according to the axial power profile. The credit of history effect influences the essential RBMK parameter, i.e. the positive void reactivity effect, which slightly increases especially at higher fuel burnup levels.

For 2% U fuel there is, as expected, a good agreement for fresh fuel between the void effect obtained with the thermal hydraulic calculation switched off (*3level_dens*), and that obtained when thermal hydraulic feedback is included (*T-H feedback*). For both cases the average water density in the fuel channel is ~0.5 g/cm^3 (see Fig. 3). For 2.6% U–Er fuel there is a discrepancy in results

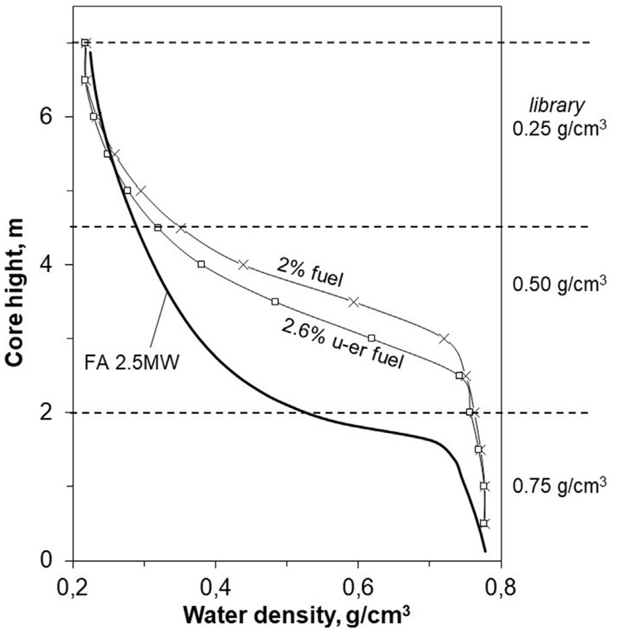

Fig. 4. Axial water density profile in fuel channel (at average burnup 12000 MWd/te).

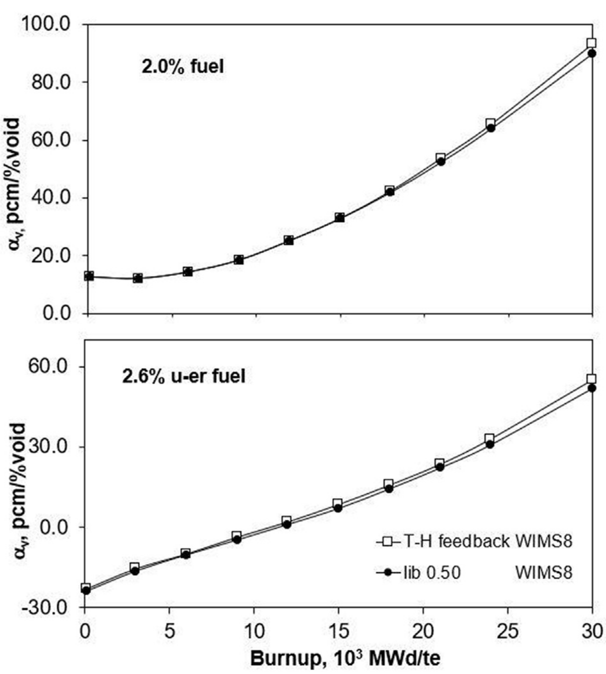

Fig. 5. Comparison of calculated void reactivity coefficient.

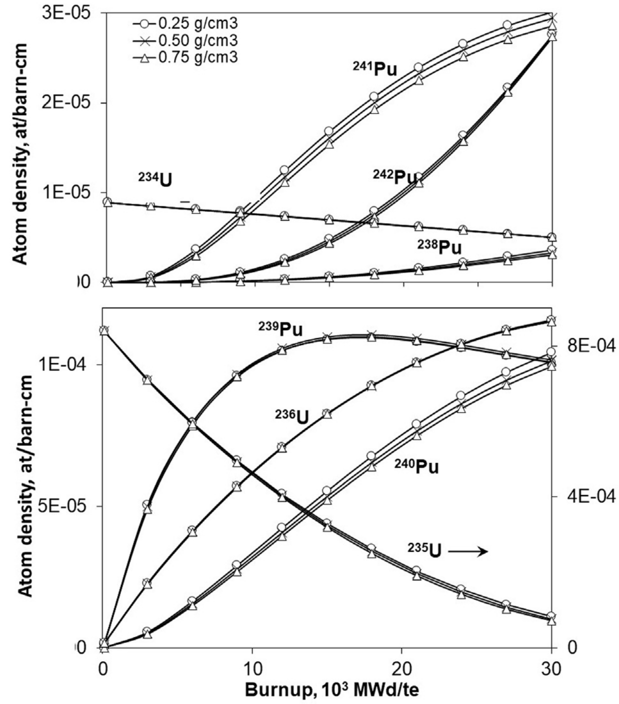

Fig. 6. Variation of nuclide concentration for 2.0% fuel.

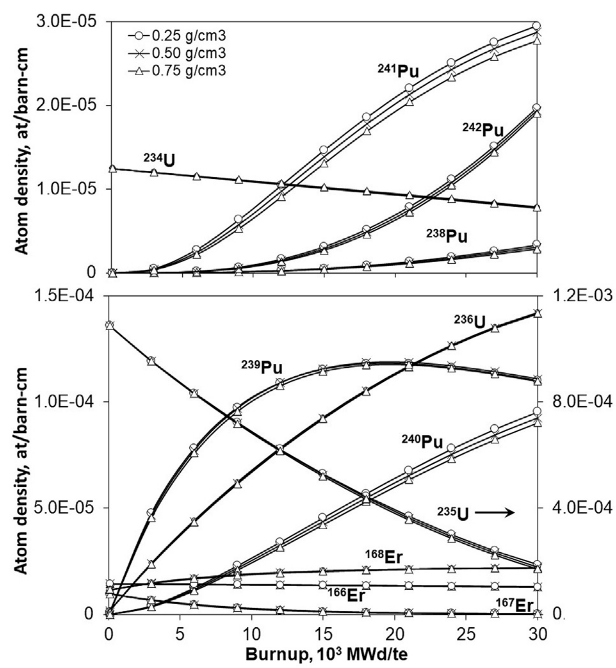

Fig. 7. Variation of nuclide concentration for 2.6% uranium-erbium fuel.

between '*3level_dens* and *T-H feedback*' cases for fresh fuel, because the average water density was assumed 0.45 g/cm^3 when the HYCA module was used in calculations. As presented on the expanded scale in Fig. 3, when thermal hydraulic calculations are performed and the average water density in the fuel channel is fixed to 0.5 g/cm^3, there is a good agreement between the cases '*no_T-H feedback + dens0.5*' and '*lib0.5*' for fresh fuel state.

The axial distribution of the water density obtained in the '*T-H feedback*" (Fig. 3) case when the HYCA module is used to perform a thermal hydraulic calculation is shown below in Fig. 4. For comparison purpose in Fig. 4 it is presented a curve of axial water density distribution for fuel channel operated at average power of 2.5 MW. The difference is caused due to the presence of graphite blocks at top and bottom of the fuel channel that have an impact on the axial power distribution and, consequently, on axial water density profile. The axial water density profile in the simulated single fuel cell loaded with 2.0% U or 2.6 U–Er FA was determined with no presence of graphite blocks at top and bottom parts of the model (Fig. 4). In this case, in each axial region, the cross-section

Table 3
Comparison of the average atom density of 2.6% U–Er fuel (divided into 3 coolant regions) with FA of 0.5 g/cm^3 coolant density.

	0.1 MWd/kgU			9 MWd/kgU			27 MWd/kgU		
	0.5 g/cm^3	0.5[a], g/cm^3	diff, %	0.5 g/cm^3	0.5[a], g/cm^3	diff, %	0.5 g/cm^3	0.5[a], g/cm^3	diff, %
^{238}Pu	1.243E-12	1.251E-12	0.61	1.203E-07	1.212E-07	0.76	2.365E-06	2.376E-06	0.46
^{239}Pu	1.078E-06	1.077E-06	−0.05	9.699E-05	9.663E-05	−0.37	1.142E-04	1.136E-04	−0.57
^{240}Pu	2.782E-09	2.806E-09	0.85	2.177E-05	2.183E-05	0.29	8.441E-05	8.467E-05	0.30
^{241}Pu	1.106E-11	1.133E-11	2.43	5.793E-06	5.815E-06	0.38	2.679E-05	2.672E-05	−0.24
^{242}Pu	8.972E-15	9.267E-15	3.18	5.987E-07	6.045E-07	0.95	1.474E-05	1.474E-05	0.01
^{234}U	1.247E-05	1.247E-05	0.00	1.108E-05	1.108E-05	−0.02	8.279E-06	8.276E-06	−0.04
^{235}U	1.087E-03	1.087E-03	0.00	7.207E-04	7.207E-04	0.00	2.320E-04	2.320E-04	−0.01
^{236}U	1.775E-06	1.776E-06	0.06	6.177E-05	6.181E-05	0.08	1.349E-04	1.350E-04	0.10
^{166}Er	1.157E-04	1.157E-04	0.00	1.127E-04	1.126E-04	−0.01	1.054E-04	1.054E-04	−0.04
^{167}Er	7.752E-05	7.751E-05	−0.01	2.444E-05	2.431E-05	−0.52	3.053E-06	3.031E-06	−0.71
^{168}Er	9.309E-05	9.309E-05	0.01	1.483E-04	1.484E-04	0.09	1.741E-04	1.741E-04	0.02

[a] Averaged values estimated from libraries prepared at water densities of 0.25, 0.5 and 0.75 g/cm.[3].

data were interpolated between two available water density values. For example, in the second axial region (2–4.5 m), where the water density varies from ~0.77 g/cm^3 to ~0.3 g/cm^3, the cross-sections for the 0.5 g/cm^3 library were obtained by interpolation using the void (0.001 g/cm^3) and 0.5 g/cm^3 water density values.

4.3. Comparison of calculated void reactivity

To verify obtained findings the calculation results were compared in another scaling, very similar to the one that was employed during a void reactivity analysis for BWR assembly [13]. For this case the void reactivity coefficient was defined as $\alpha_{vi} = \Delta\rho_{inf}(v_i)/\Delta v_i$, where $\Delta v_i = v_i\text{-}v_0$ is void fraction difference between non- and voided states. The difference $\Delta\rho_{inf}(v) = \rho_{inf}(v)\text{-}\rho_{inf}(v_0)$ is the reactivity variation with void normalized with respect to the reactivity of the voided system. Because during voiding the water density in RBMK fuel channel can vary from 0.78 to ~0 g/cm^3 (0.001 g/cm^3 selected in this study) consequently void fraction v_i ranking from pure liquid ($v = 100\%$) to pure steam ($v = 0\%$) was selected. This means that the void fraction 66% corresponds to water density 0.50 g/cm^3. Such normalization of results helps to visualize and qualitatively estimate variations of void reactivity coefficient in term of water history effect assessment.

WIMS8 calculation results presented in Fig. 5 confirm the above observed findings. The void reactivity coefficient obtained with WIMS8 assuming a uniform water density of 0.5 g/cm^3 (curve *lib0.50*) is in good agreement, and very slightly more underestimated at the higher irradiations, that the values obtained from the '*T-H feedback*' case where the axial variation in the water density was explicitly represented by interpolation according to the calculated axial power profile. The discrepancy is less than 4.6 pcm/%void in absolute value of α_v for both types of fuel in the whole irradiation range.

4.4. Effect on nuclide compositions

Further examination of the variation with fuel irradiation the nuclide compositions for different values of the water density was performed to interpret the basic trend observed. Three different libraries, i.e. *lib0.25, lib0.50* and *lib0.75*, prepared with WIMS8 (see Section 4) were employed for this analysis. Analysis of individual isotopic compositions shows the effect of the neutron spectrum due to water density on fuel depletion (Figs. 6, 7). It is presented that water density change is associated almost with the change of ^{235}U amount and consequently with plutonium isotopes, i.e. mostly ^{240}Pu and ^{241}Pu. The maximum discrepancy does not exceed 6% for all nuclide compositions through the whole investigated irradiation range, bigger difference can be observed for 2.6% U–Er type fuel. The differences between the predicted values of isotopic composition for each nuclide become more important as the fuel irradiation increases.

The comparison of averaged isotopes atom density values from three different libraries versus the reference case (evaluated at water density 0.5 g/cm^3) is showed in Table 3. The highest discrepancy is observed only at beginning of irradiation up to 3% for ^{241}Pu and ^{242}Pu isotopes while for the rest isotopes discrepancy does not exceed 1% percent.

5. Conclusions

The history effect due to different coolant density along the fuel channel on reactor physics parameters have been investigated for 2% U and 2.6% U–Er RBMK type FAs. Three irradiation dependent cross-section data using WIMS8 code were prepared for each of the three axial regions of the fuel channel. A comparison analysis was performed applying QUABOX/CUBBOX-HYCA single fuel model with different libraries. The single fuel model having the thermo-hydraulic part of HYCA allowed to estimate the axial water profile in fuel channel depending on power profile and evaluate the influence of history effect. Finally, it could be concluded that history effect has impact on the most essential RBMK parameter, i.e. void reactivity coefficient, causing in its slight increase. The maximal discrepancy does not exceed 4.6 pcm/%void in absolute value of α_v for both investigated fuel types in whole irradiation range. The different water density plays a big role to fuel isotopic composition. The biggest isotopic composition difference up to 6% is observed comparing 0.25 g/cm^3 and 0.75 g/cm^3 libraries at high fuel burnup depths.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgments

Authors of this paper are grateful to Serco Assurance staff who were involved in the past collaborative investigations, for their support at early stage of research on this topic.

Appendix A. Supplementary data

Supplementary data related to this article can be found at https://doi.org/10.1016/j.net.2020.04.015.

References

[1] Y. Maruyama, H. Asaka, A. Satou, H. Nakamura, Single rod experiments on transient void behavior during low-pressure reactivity-initiated accidents in light water reactors, Nucl. Eng. Des. 236 (2006) 1693–1700.
[2] Advanced Fuel Pellet Materials and Fuel Rod Design for Water Cooled Reactors vol. 1654, IAEA-TECDOC, 2010.
[3] K. Almenas, A. Kaliatka, E. Uspuras. Ignalina RBMK-1500. A Source Book. Extended and Updated Version, Lithuanian Energy Institute, Kaunas, Lithuania (1998).
[4] M. Fratoni, E. Greenspan, Neutronic design of hydride fueled BWRs, Nucl. Eng. Des. 239 (2009) 1531–1543.
[5] Nuclear Power Reactors in the world, 2018 edition), Reference Data Series No. 2, IAEA AEA-RDS-2/38.
[6] E. Uspuras, State of the art of the Ignalina RBMK-1500 safety, Sci. Technol. Nucl. Install. 2010 (2010) 11.
[7] S. Langenbuch, W. Maurer, W. Werner, Coarse mesh flux expansion method for the analysis of space-time effects in large LWR cores, Nucl. Sci. Eng. 63 (1977) 437–456.
[8] S. Langenbuch, A. Petri, J. Steinborn, I.A. Stenbok, A.I. Suslow, Adaptation of GRS Calculation Codes for Soviet Reactors, Web, 1994.
[9] S. Langenbuch, W. Maurer, W. Werner, High-Order schemes for neutron kinetics calculation based on a local polynomial approximation, Nucl. Sci. Eng. 64 (1977) 508–516.
[10] R. Pabarcius, A. Tonkunas, E. Bubelis, M. Clemente, Uncertainty and sensitivity analysis of void and power reactivity coefficients in an RBMK-1500 reactor core, Kerntechnik. ISSN 70 (2005) 114–119, 0932-3902.
[11] W.I.M.S. The Answers Software Package, A Modular Scheme for Neutronics Calculations, in: User Guide for Version 8, ANSWERS/WIMS, 1999.
[12] M.J. Halsall, WIMS8 – Speed with Accuracy, ANS Topical Meeting, Long Island, USA, 1998.
[13] F. Jatuff, G. Perret, M.F. Murphy, F. Giust, R. Chawla, Void reactivity coefficient benchmark results for a 10x10 BWR assembly in the full 0-100% void fraction range, Ann. Nucl. Energy 36 (2009) 853–858.