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ORIGINAL PAPER

Study of boron dilution phenomenon in the core and fuel assemblies of Bushehr VVER-1000 reactor in normal operating conditions

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Abstract. The spatial temperature distributions in fuel and coolant, results in appearing local changes in those elements densities in the reactor core, and also due to the complete solubility of boric acid in the coolant, there will be a direct correlation between the changes in the boron concentration and the coolant density. Because of the gradual reduction of boron concentration, first a local positive reactivity will be inserted into the core which will cause slight thermo-neutronic fluctuations in the reactor core. Of course, the trend of this process in the case of excessive reduction of the density of the coolant and evaporation of water (accident scenarios) will be reversed and subsequently the negative reactivity will be given to the system. With regard to the importance of this phenomenon, the spatial changes of boron concentration in the core and fuel assemblies of Bushehr VVER-1000 reactor have been examined. In line with this, by designing a complete thermo-neutronic cycle and by using CITATION, WIMS D-5 and COBRAN-EN codes, coolant temperature distribution and boron concentration will be calculated through this procedure, which first by using the output results of WIMS and CITATION codes, the thermal power of each fuel assembly will be calculated and finally, by linking these data to COBRA-EN code and using core and sub-channel analysis methods, the three-dimensional (3D) calculations of boron dilution will be obtained in the core as well as the fuel assemblies of the reactor.

Key words: VVER-1000 • boron dilution • WIMS D-5 • CITATION • COBRA-EN • core and sub-channel analysis

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Introduction

The study of boron dilution is one of the important subjects in the analysis of nuclear reactor behavior in the steady state and transients. By solving neutron transport equation in the defined geometry, the WIMS D-5 [6] code calculates neutronic group constants and in the continuation, the CITATION [3] code by solving diffusion equations, calculates neutron flux and thermal power for the entire defined areas. By linking the thermal power distribution data (CITATION output) to COBRA-EN [1] code and then mesh specifying and solving mass, energy and momentum balance equations in each of these meshes, and using the finite difference method (semi-explicit mode), the temperature distribution as well as boric acid concentration can be calculated. It is worth mentioning that due to the prevalence of calculation error in using the symmetrical geometry and modeling only the 1/6th of the core, we decided to conduct the modeling in the complete real geometry for both core and fuel assemblies.

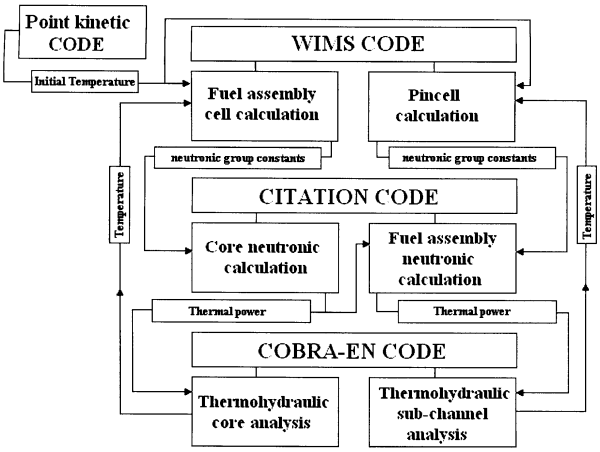

Fig. 1. Applied calculation flow chart.

In this work, WIMS D-5 and CITATION codes were used in the neutronic calculations while COBRA--EN code was used in the thermohydraulic modeling. Figure 1 shows the calculation flow chart that we have used.

A procedure of neutronic calculations in the full core modeling

In the first stage, using WIMS D-5 code, the neutronic group constants which are needed in the CITATION code were calculated by the following method. With regards to the core structure and arrangement of the fuel assemblies at the beginning of life (BOL) mode, we modeled the fuel assemblies with 1.6, 2.4 and 3.62% enrichments and a reflector. In order to calculate the neutron flux and power density using CITATION code, we coupled the codes and linked the output results of WIMS D-5 to CITATION input file. The cyclic process continued throughout the whole calculation until the criterion was satisfied.

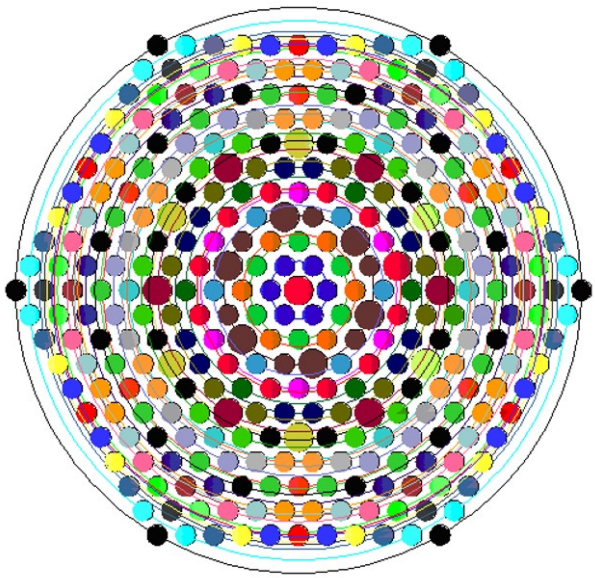

Fig. 2. The manner of regulating arrays used in modeling fuel assembly by WIMS D-5 code.

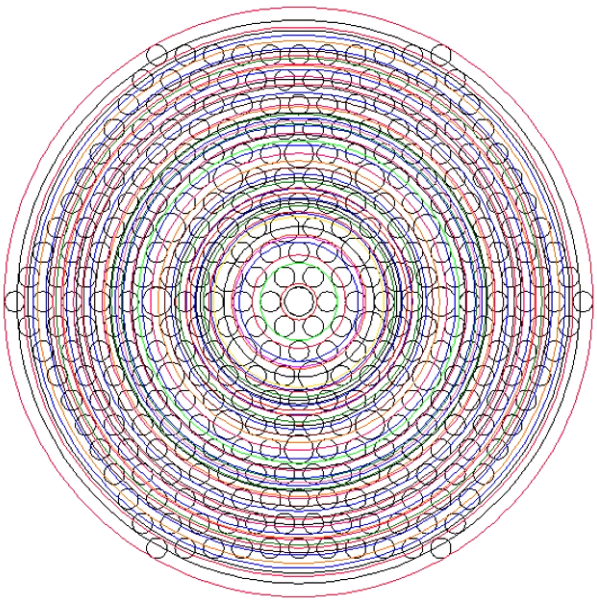

Fig. 3. Regulating annuluses used in modeling fuel assembly by WIMS D-5 code.

Regulating arrays of fuel assemblies in WIMS D-5 code

The position of fuel rods in each fuel assembly will be defined in the 36 fold arrays which have been shown in Fig. 2.

Also, in Fig. 3, the defining of complemented space between rods depicted as 54 fold annulus in each fuel assembly by WIMS D-5 code.

It should be noted that the initial temperatures of the elements used in WIMS D-5 input file, are first obtained on the basis of Point Kinetic model [4]. After completion of the calculation cycle and attainment of the correct temperature of the elements (by using the link of COBRA-EN, WIMS D-5 and CITATION codes), these element temperatures will be corrected and the process will be repeated for the improvement of the computational precision.

After this stage and attainment of each fuel assembly group constants, by linking these data to the CITATION input file (card 008-Macroscopic cross section), the thermal power distribution will be calculated in each fuel assembly, whose mesh specifying process in the reactor core through CITATION code has been shown in Fig. 4.

In the next stage, after calculation of the thermal power distribution in the reactor core, through linking these data to the COBRA-EN code, the rate of temperature distribution in fuel and coolant will be gained as well as the changes in the concentration of boric acid dissolved in the coolant.

In Fig. 5, the meshes specifying process and arrangement of channels in COBRA-EN code (by using core analysis method) have been shown.

Now, by using the results gained through COBRA--EN code, the spatial changes in boron concentration will be calculated in the core, which have been shown in Figs. 6, 7 and 8.

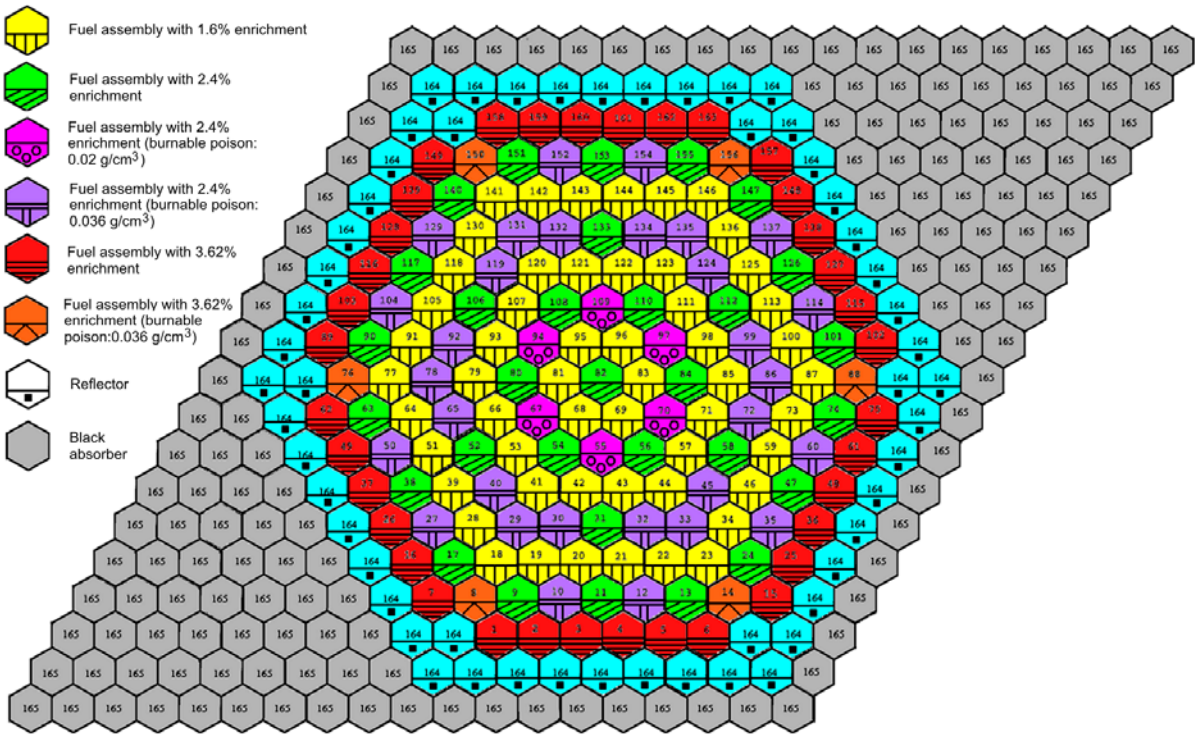

Fig. 4. Mesh specifying process in the reactor core by using CITATION code.

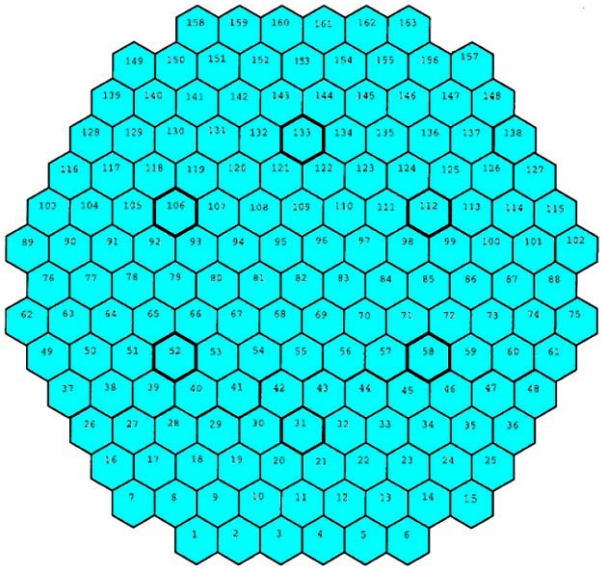

Fig. 5. Mesh specifying process and arrangement of channels in COBRA-EN code on the basis of core analysis method.

Calculations of boron concentration changes in fuel assemblies of the VVER-1000 reactor

At this stage, by using the results of CITATION code (full core mode) in accordance with the conducted mesh, the thermal power of each fuel assembly will be calculated, and subsequently, in line with attaining the thermal power distribution within each fuel assembly, the input files of CITATION for each assembly will be regulated.

The modeling description and cell processing of fuel assemblies, has been shown in Figs. 9–13.

It should be noted that with regard to the varieties of fuel assemblies and their comprising elements

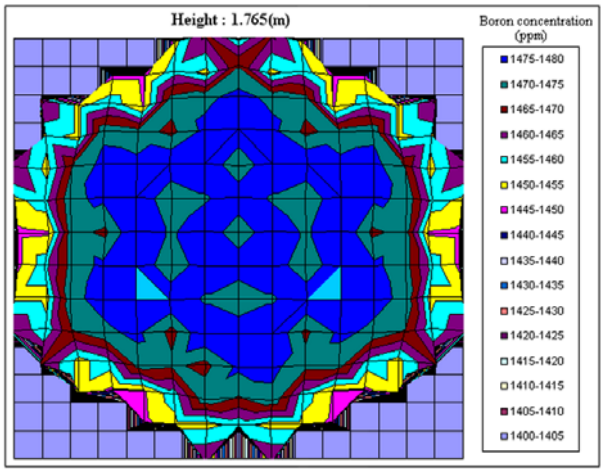

Fig. 6. Rate of changes in boron concentration in radial direction in the middle of the core.

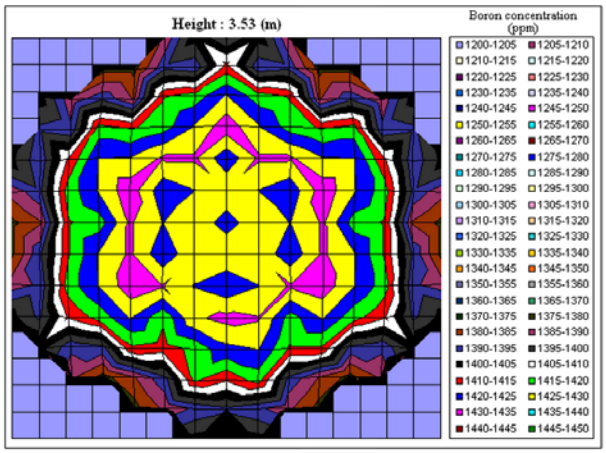

Fig. 7. Rate of changes in boron concentration in radial direction in the end of the core.

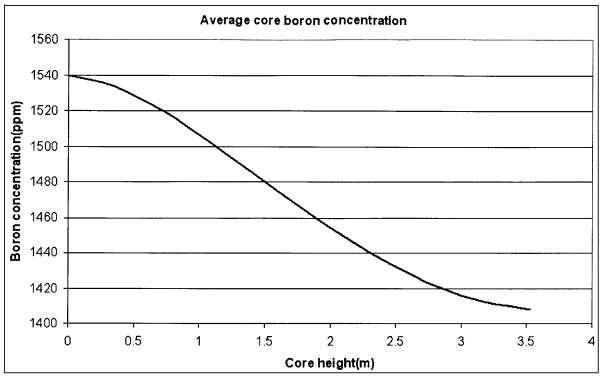

Fig. 8. Rate of changes in boron concentration in the core in axial direction of the core.

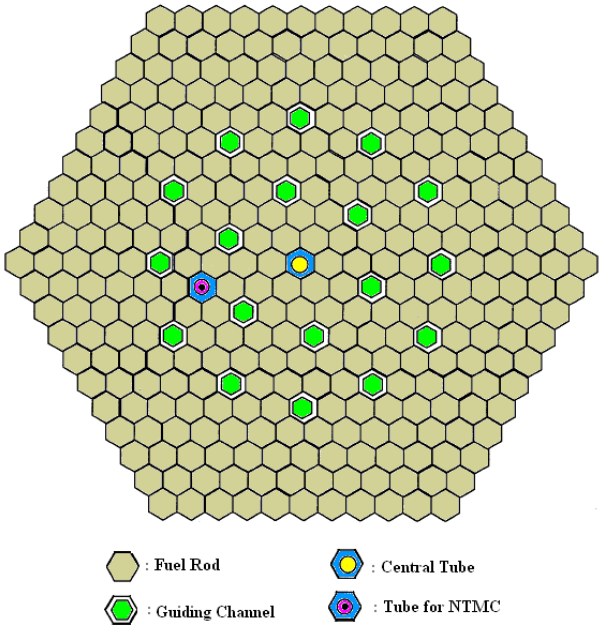

Fig. 9. The arrangement of fuel and control rods in 1.6 and 2.4% assemblies.

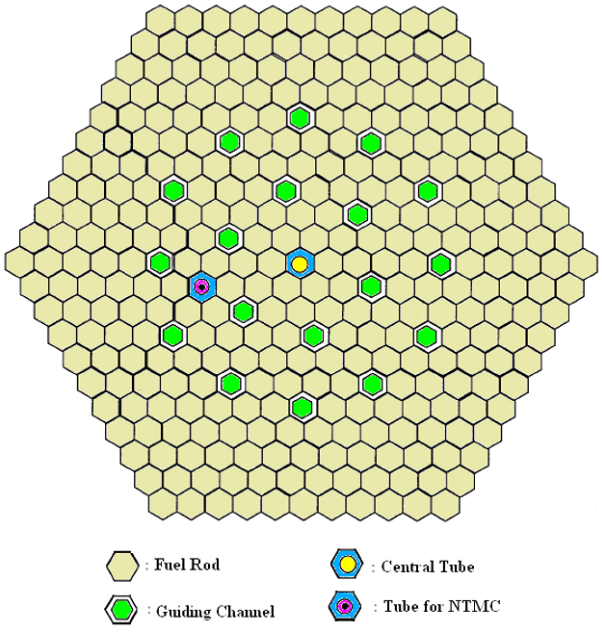

Fig. 10. Cell processing of 1.6 and 2.4% assemblies.

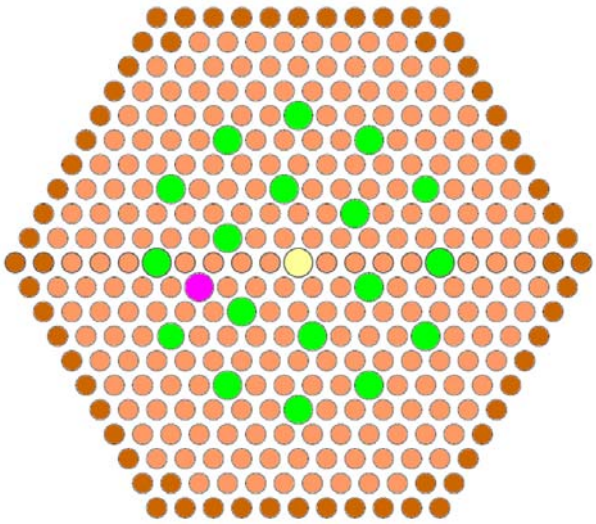

Fig. 11. The arrangement of fuel rods of type 1 and 2 and control rods in 3.62% fuel assembly.

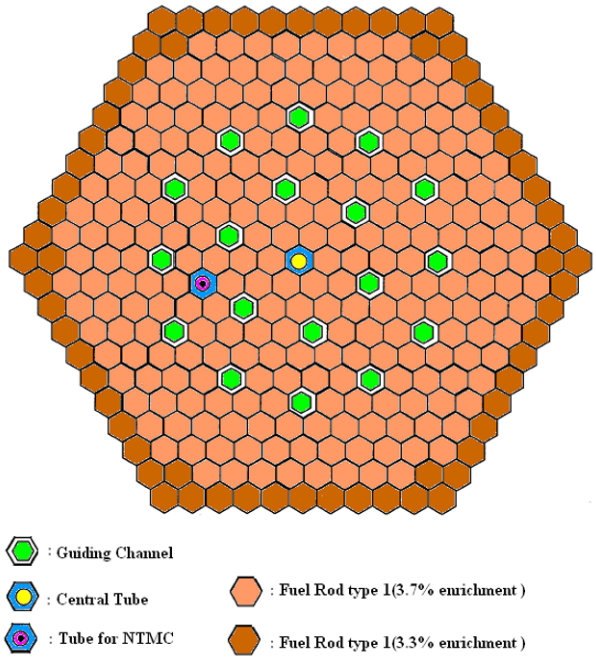

Fig. 12. Cell processing of 3.62% fuel assembly.

[2], cell calculations have been conducted by using WIMS code for cells with the enrichment 1.6, 2.4, 3.3, 3.7%, control rod, reflector and also burnable poisons. Thereafter, the distribution of neutron flux and thermal power will be calculated by CITATION code in radial and axial directions (for each fuel assembly) which in

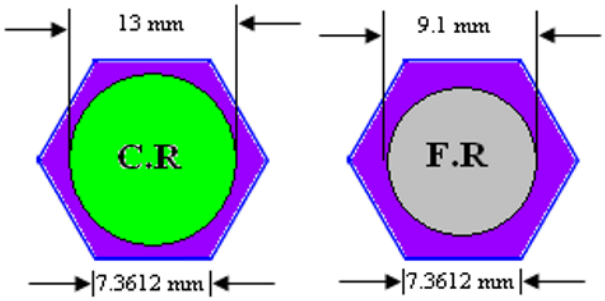

Fig. 13. The definition of cells in WIMS D-5 code.

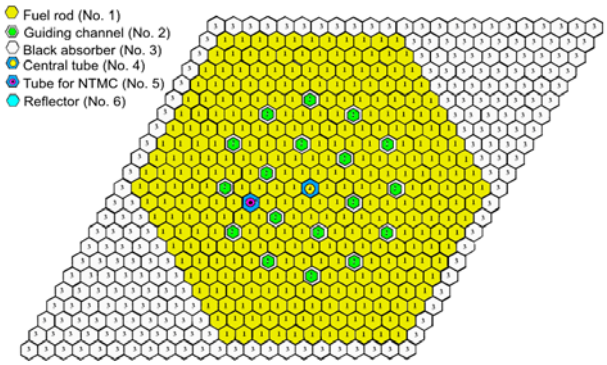

Fig. 14. Mesh specifying for fuel assemblies with enrichments 1.6 and 2.4% in CITATION code.

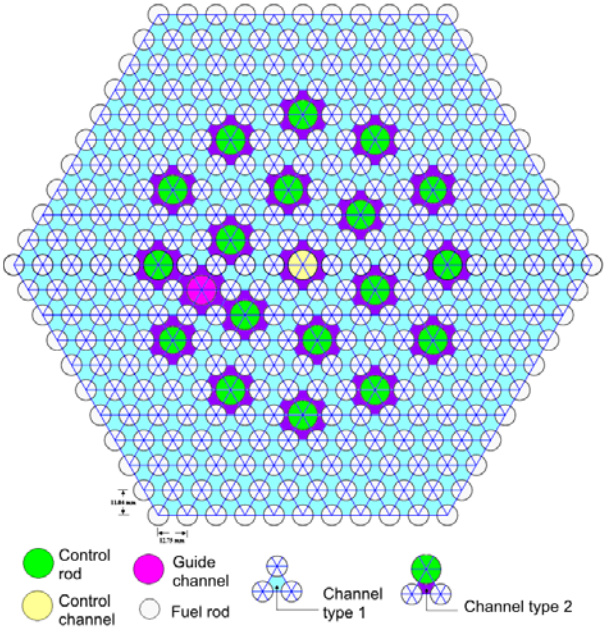

Fig. 15. The arrangement of fuel rods and coolant channel in fuel assembly.

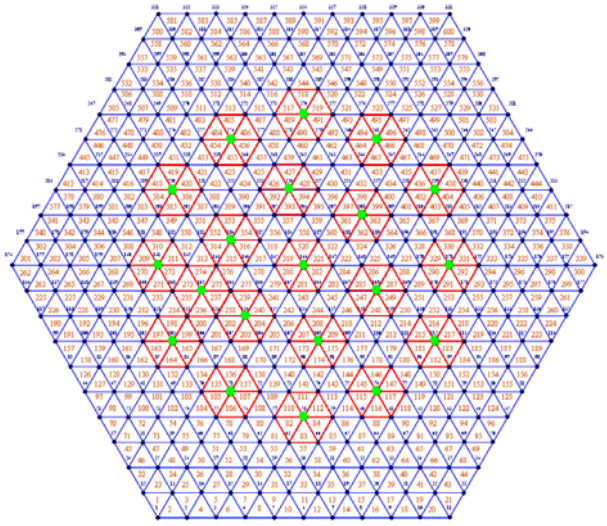

Fig. 16. Process of channeling fuel assembly according to triangular channels (COBRA-EN code).

Fig. 14 the mesh specifying process has been shown for an exemplary fuel assembly.

In the final stage of this study, by using COBRA-EN code, the temperature distribution in fuel and the coolant as well as changes in the concentration of soluble boric acid will be calculated.

It should be noted that "sub-channel analysis" model has been used in the thermohydraulic calculations of each fuel assembly.

In Figs. 15 and 16, the arrangements of channels and the exerted meshes have been shown, respectively.

As it has been shown in Figs. 15 and 16, in order to increase the accuracy of calculations, the total area of a fuel assembly has been divided into 600 triangular channels.

The important point, which was found out in this study is that, contrary to the general belief, the symmetrical susceptibility in the reactor core will not be considered as a logical reason for the definition of modeling geometry only for a symmetrical sector. This circumstance will reduce the accuracy of calculations and will cause occurrence of a false equal distribution in the radial direction of the reactor core.

The rate of radial and axial changes of boron dilution for the hot fuel assembly of VVER-1000 reactor has been shown in Figs. 17–19. In Fig. 20, the rate of axial changes of boron dilution for different fuel assemblies has been shown, which, of course, due to the high number of output diagrams, reference has only been made to the results in fuel assemblies that are considerably different from each other.

Method of calculating positive reactivity insertion caused by the boron dilution phenomenon in the reactor core

With regard to the relation of boron concentration changes to the temperature distribution in the coolant, the need is being felt for calculations in the field of reactivity insertion caused by this phenomenon.

In this study, by using the following correlation, the positive reactivity caused by the boron dilution will be calculated [5]:

$$\rho_B = 1.92 \times 10^{-3} C_B(1 - f_0) \quad (1)$$

where: ρ_B – reactivity resulted by boric acid concentration changes; C_B – difference in boric acid concentration in each region compared to the original concentration; f_0 – the thermal utilization factor in the absence of boron.

Where the f_0 parameter is given by:

$$f_0 = \frac{(2\sigma_a^O + e\sigma_a^{u^{235}} g_a^{u^{235}}(T_{\text{fuel}}) + (1-e)\sigma_a^{u^{238}} g_a^{u^{238}}(T_{\text{fuel}}))\frac{\sqrt{T_{\text{fuel}}}}{T_{\text{fuel}}}}{(2\sigma_a^O + e\sigma_a^{u^{235}} g_a^{u^{235}}(T_{\text{fuel}}) + (1-e)\sigma_a^{u^{238}} g_a^{u^{238}}(T_{\text{fuel}}))\frac{\sqrt{T_{\text{fuel}}}}{T_{\text{fuel}}} + } + (28.95\frac{\rho_{\text{mod}}}{\rho_{\text{fuel}}})\sigma_a^{\text{mod}}\frac{\sqrt{T_{\text{mod}}}}{T_{\text{mod}}}(\frac{\Phi_{\text{mod}}}{\Phi_{\text{fuel}}})(\frac{V_{\text{mod}}}{V_{\text{fuel}}}) \quad (2)$$

where: ρ_{fuel} and ρ_{mod} – fuel and moderator density (Kg/m^3); e – uranium enrichment ratio; T_{fuel} and T_{mod} – fuel and moderator temperatures (k); Φ_{fuel} and Φ_{mod} – thermal

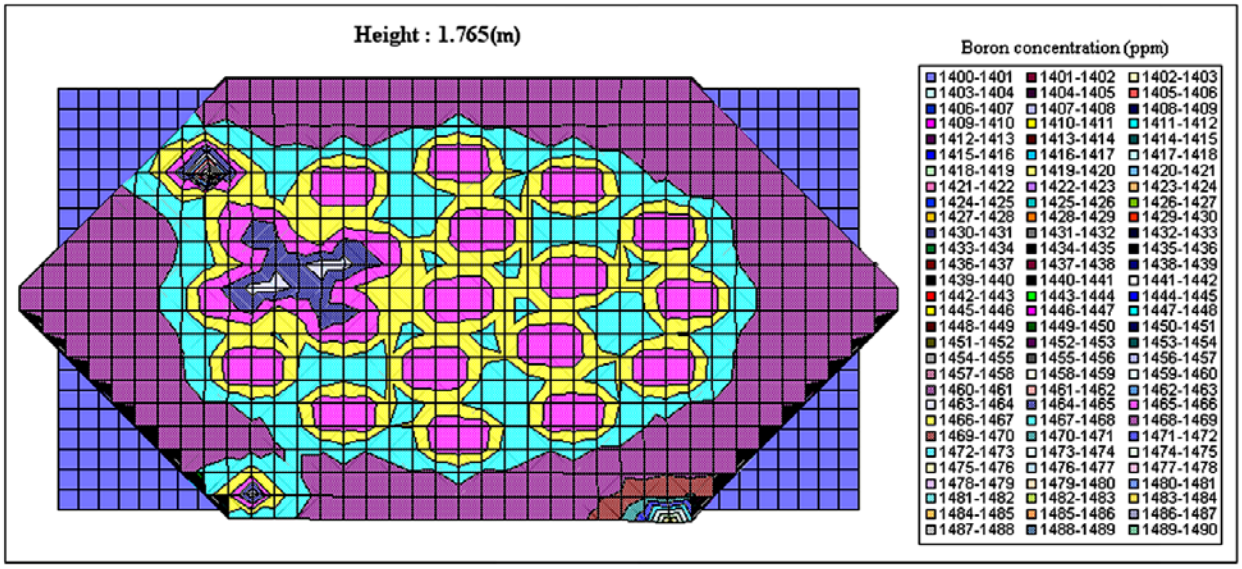

Fig. 17. Boron concentration changes in radial direction at the middle of the hot fuel assembly.

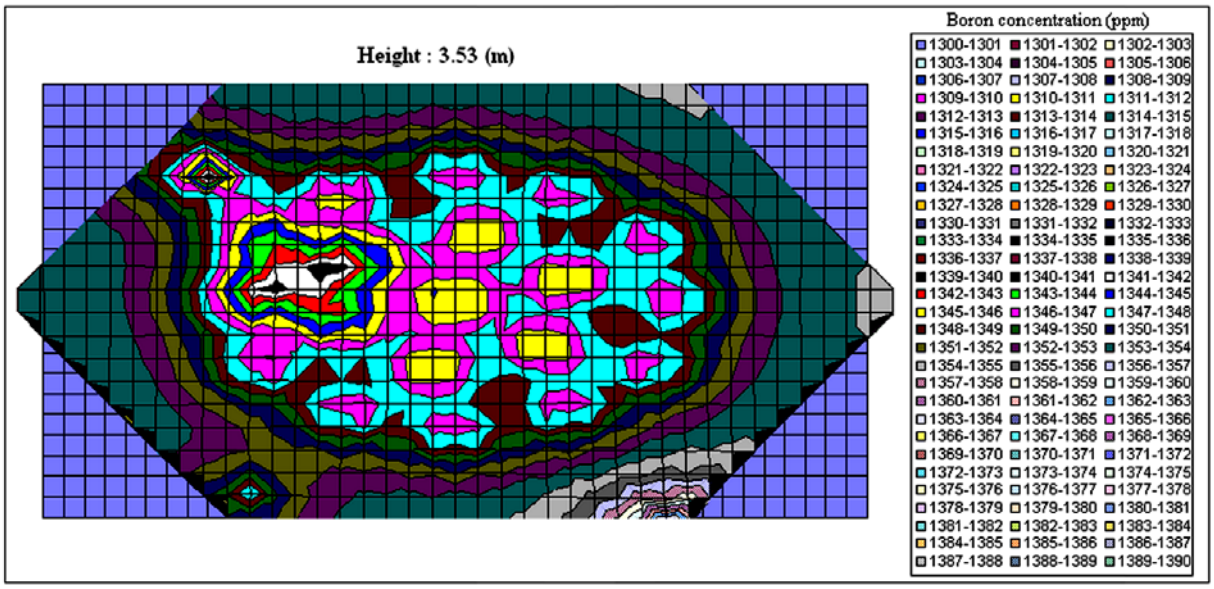

Fig. 18. Boron concentration changes in radial direction at the end of the hot fuel assembly.

neutron flux in fuel and moderator (#/m^3·s); V_{fuel} and V_{mod} – fuel and moderator volumes (m^3).

It should be noted that in the said correlation the parameters of absorption cross sections, g_a (non–1/v factor) and density, are calculated in each zone. In the next step, the neutron flux in each fuel assembly will be derived by using CITATION code.

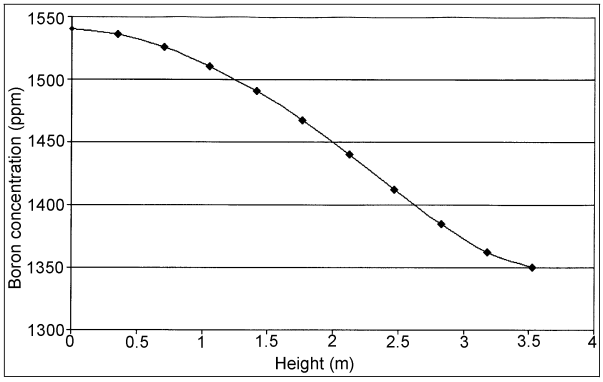

Fig. 19. Boron concentration changes at axial direction of the core in hot fuel assembly.

In continuation, the temperature relation of each of the said parameters will be calculated in each zone in smart mode through programming. Furthermore, with regard to the prevalence of fuels with different enrichments in the reactor core (1.6, 2.4, and 3.62%) we substitute the "e"parameter in accordance with the existing fuel enrichment in each zone in Eq. (2).

For the calculation of the absorption cross sections, according to temperature changes in the fuel and coolant, the following correlation is used [5].

$$X = X_0 + \alpha(T_{fuel} - T_{fuel_0}) + \beta(T_{coolant} - T_{coolant_0}) + \gamma(T_{fuel} - T_{fuel_0})^2 + \delta(T_{coolant} - T_{coolant_0})^2 \quad (3)$$

where: α, β and γ – temperature coefficients; X, X_0 – the temperature dependent and initial values of physical constant; T_{fuel}, $T_{coolant}$ – new values of fuel and coolant

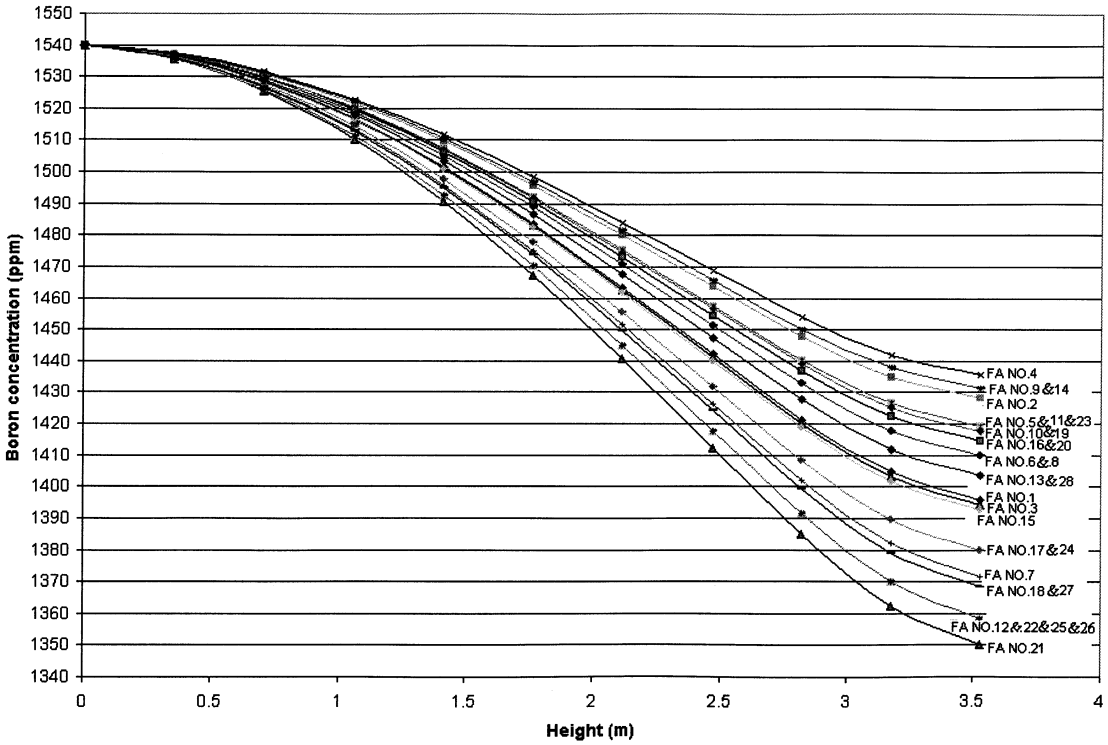

Fig. 20. Boron concentration changes in axial direction of the reactor core in different fuel assemblies.

temperatures; $T_{\text{fuel}0}$, $T_{\text{coolant}0}$ – fuel and coolant reference temperatures.

Now, with respect to the fact that in the above equation for the calculation of neutronic cross sections, there is a need to know the quadruple temperature coefficients, therefore in order to solve this problem, by using the WIMS D-5 code for each fuel assembly in the four different temperatures, we calculate the absorption cross sections for different elements, then by putting them in the below equations and subsequently simultaneous solving of these four fold equations, the temperature coefficients will be calculated [4].

$$
\begin{aligned}
\sigma_{a_1} &= \sigma_{a_0} + \alpha(T_{\text{fuel}_1} - T_{\text{fuel}_0}) + \beta(T_{\text{coolant}_1} - T_{\text{coolant}_0}) \\
&\quad + \gamma(T_{\text{fuel}_1} - T_{\text{fuel}_0})^2 + \delta(T_{\text{coolant}_1} - T_{\text{coolant}_0})^2 \\
\sigma_{a_2} &= \sigma_{a_0} + \alpha(T_{\text{fuel}_2} - T_{\text{f}_{\text{fuel}_0}}) + \beta(T_{\text{coolant}_2} - T_{\text{coolant}_0}) \\
&\quad + \gamma(T_{\text{fuel}_2} - T_{\text{fuel}_0})^2 + \delta(T_{\text{coolant}_2} - T_{\text{coolant}_0})^2 \\
\sigma_{a_3} &= \sigma_{a_0} + \alpha(T_{\text{fuel}_3} - T_{\text{fuel}_0}) + \beta(T_{\text{coolant}_3} - T_{\text{coolant}_0}) \\
&\quad + \gamma(T_{\text{fuel}_3} - T_{\text{fuel}_0})^2 + \delta(T_{\text{coolant}_3} - T_{\text{coolant}_0})^2 \\
\sigma_{a_4} &= \sigma_{a_0} + \alpha(T_{\text{fuel}_4} - T_{\text{fuel}_0}) + \beta(T_{\text{coolant}_4} - T_{\text{coolant}_0}) \\
&\quad + \gamma(T_{\text{fuel}_4} - T_{\text{fuel}_0})^2 + \delta(T_{\text{coolant}_4} - T_{\text{coolant}_0})^2
\end{aligned}
\tag{4}
$$

At the end, by using these temperature coefficients, we could easily calculate the absorption cross sections according to temperature changes in the fuel and moderator and then substitute them in Eq. (2).

The rate of radial and axial changes of positive reactivity insertion for VVER-1000 reactor core has been shown in Figs. 21–23.

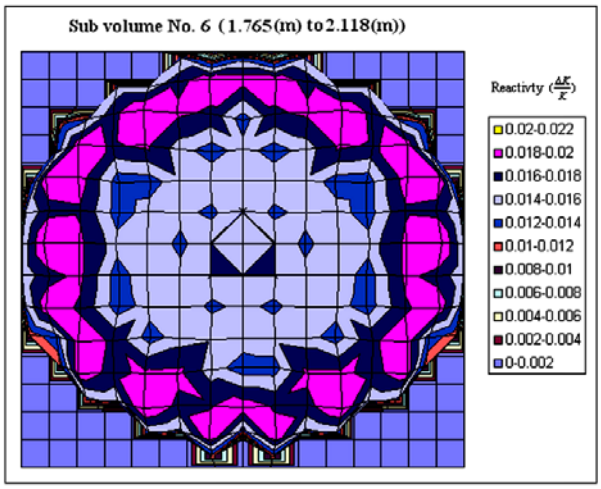

Fig. 21. The rate of positive reactivity insertion (due to boron dilution phenomenon) in radial direction at the middle of the core.

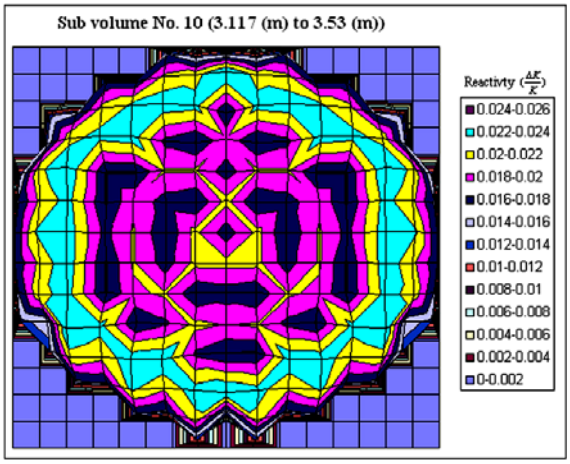

Fig. 22. The rate of positive reactivity insertion in radial direction at the end of the core.

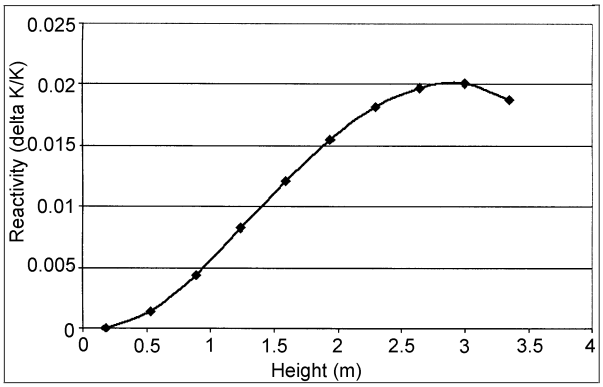

Fig. 23. Core average reactivity in each sub-volume (in axial direction).

Conclusion

By observing charts 6 and 7, it will be noticed that, due to the monotonous distribution of power in the radial direction of reactor core, the boron concentration changes will result in minor in the middle and end of the axial sub-volumes.

In Fig. 8, which shows the axial changes in boron concentration in the reactor core, it is noticed that contrary to the results of previous diagrams, due to the existence of cosine distribution of thermal power, the remarkable changes are made in the boron concentration. This phenomenon at the time of the occurrence of events could cause considerable and destructive fluctuations in the reactor core, which is proposed that the boric acid injection be applied locally in these mentioned sub-volumes.

Also, it is noticed in Figs. 17 and 18, the radial distribution of boron acid in the hot fuel assembly has a lower concentration in control rod channels, which is due to non-arrival of control rods, and then, consequently, these channels are filled with the coolant.

With regard to Figs. 19 and 20 it is noticed that, due to the cosine distribution of thermal power in the axial direction of fuel assemblies, boron concentration changes in this direction of the core will start reduction of density in the coolant.

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