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Analyze the Effect of Void Fraction on the Main Operating Parameters of the VVER-1200

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The VVER-1200/AES-2006 is one of the most promising nuclear reactors for power generation. This work provides an in-depth analysis of the void fraction effect on the operation of VVER-1200 using MCNPX code. The void fraction in the VVER-1200 may be a result from the loss of coolant accident (LOCA) or boiling the moderator/coolant materials. The fission multiplicities (υ), the estimated of the recoverable energy per fission (Q) and the effective delayed neutrons fraction (β_{eff}) have been investigated at different fuel enrichment and a different void fraction. It is beneficial to calculate the β_{eff} at the different void fraction and different enrichment, due to its impact on the reactor power change rate. The excess reactivity in the reaction can be controlled using the gadolinium dioxide or boron carbide. Increasing the void fraction decrease the burnup rate of gadolinium dioxide and boron carbide in the VVER-1200 assembly. The Doppler coefficient has a large effect on the reactivity of the reaction. Therefore, it is important to calculate the Doppler coefficient at different fuel temperature. The existence of the void fraction increases the concentration of the most hazardous radioactive nuclear waste products.

Keywords: *VVER-1200- Void fraction- LOCA- Delayed neutrons- Doppler effect- Spent fuel.*

Introduction

Light water reactors (LWRs) are the main suppliers of nuclear electricity. The VVER reactors belong to the family of the pressurized water reactors [1]. The VVER-1200/AES-2006 is the evolution of the VVER-1000 by improving plant performance and increasing plant safety. The VVER-1200 is one of the most promising Generation +III reactors. The main goal in the evolutionary development process of VVER-1000 to VVER-1200 was to reduce costs without substantially changing the basic configuration of the nuclear steam supply system, while at the same time increasing safety. The thermal power was increased to 3200 MWt and additional passive safety systems introduced for the management of beyond design basis accidents. The VVER 1200 will produce 1200 MWe of power.

The VVER-1200 design is based on the principle of safety assurance for the personnel, population and environment. The principle meets the requirements for the standards of radioactive substance release into the environment and their content at normal operation, at anticipated operational occurrences including the design basis conditions. Safety features include a containment building and missile shield. It has full emergency systems that include an emergency core cooling system, emergency backup diesel power supply, advanced refueling machine, computerized reactor control systems and backup feed water supply. The nuclear reactor and associated systems are in a single building with another building for the turbo-generators. The main building comprises the reactor, refueling machine and diesel backup power supply, steam generators and reactor control

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systems. There are two families of VVER-1200/AES-2006 plant designs; the first is the V-392M version that was developed by Moscow Atomenergoproekt on the basis of the AES-92 design. The other family of VVER-1200/AES-2006 designs, the V-491 version, was developed by St Petersburg Atomenergoproekt on the basis of the AES-91 design developed for China, as successfully deployed at Tianwan [2].

The LOCA is postulated to occur as a break between the reactor vessel and the main circulation pump. The break results in the continuous, uncontrolled loss of reactor coolant to the containment or the auxiliary/turbine building. The LOCA is considered one of the main important design basis accidents for the LWRs. To mitigate the consequences of LOCAs from the pipe break, the emergency core cooling system is usually activated in LWRs to maintain the core inventory and remove the decay heat during all phases of the accident. Without an adequate source of emergency cooling water, the subsequent fuel heat up could result in fuel damage and the release of fission products from the fuel [3, 4]. Safe operation of nuclear power plants is assured by maintaining basic safety functions such as reactivity control, fuel cooling, and confinement. Among these functions, fuel cooling is the chief function that can maintain the thermal integrity of fuel channels in a pressurized heavy water reactor during event progression of a large break loss-of-coolant accident [5].

In VVER-1200 using boron carbide (B_4C) / gadolinium dioxide (Gd_2O_3) and varying its concentration with time serves to compensate for excess reactivity. Burnable poisons (BP) embedded in the fuel or other core constituents offer an additional means for limiting excess reactivity as well as mitigating localized power peaking. Burnable absorber (BA) is one of the most important control elements in with respect to the nuclear reactor safety. Boron carbide and boron steel are currently used in control rods of Russian power water reactors. These absorber materials accumulate large radiation-induced damages caused by (n,α) reactions on ^{10}B isotopes, helium formation and swelling. The first failure of the rod cluster control assembly of VVER-1000 caused by B_4C swelling [6]. The performance of B_4C and Gd_2O_3 was also analyzed in this study.

Large quantities of nuclear waste plutonium and long-lived radioactive actinides have been accumulated in the LWRs using conventional fuel. Transmuted uranium (TRU) elements represent the most hazardous radioactive nuclear waste products because of their long-term, high-level radioactivity. Plutonium constitutes the majority and most dangerous component of nuclear waste. Plutonium component has serious hazard potential because of the high radio-toxicity and high radioactivity [7]. There is a great efforts have been made to reduce the inventory of long-lived radioactive actinides resulting from the fuel burning process in LWRs. Most of these TRU elements are fissionable and thus can be considered as fissile fuel materials in the form of mixed fuel with thorium or natural uranium in the latter. Therefore, it is important to study the stockpiled of the TRU elements resulting from the burning process of conventional fuel (UO_2) under different condition.

Description of the reactor

The VVER-1200/AES-2006 reactor is a pressurized water moderated reactor that consists of 163 fuel assemblies. Each fuel assembly includes 311 fuel rods, 18 guide channels and one tube for instrumentation. The control and the burnable absorber rods have been placed in the guiding channels. The model of the fuel assembly is shown in Fig. 1. The main reactor operating conditions and design parameters are presented in Table 1 [1]. The fuel assemblies of VVER-1000 and VVER-1200 are actually identical in design. Both fuel assemblies comprise a top nozzle, bottom nozzle, and the bundle of fuel elements in a rigid welded framework. They only differ in the fuel column height 353 cm for VVER-1000 and 375 cm for VVER-1200. All calculations were performed with a three-dimensional transport and depletion computer code MCNPX 2.7 code. The Monte Carlo technique can be beneficial. Since the Monte Carlo method simulates individual particle tracks through a given system, it can provide a very accurate probabilistic transport solution.

Results and Discussion

The fuel elements in the VVER-1200 core are modeled exactly, including all gaps and zircaloy cladding. The VVER-1200 assemblies are modeled at the beginning of life (BOL). The output results of the MCNPX model was calculated with standard deviation ±0.00021.

Table (1): Characteristics and dimensions of VVER-1200 core [1, 8].

Parameter	Value
Core	
Reactor thermal power (MWt)	3200
Reactor electrical power (MWt)	1200
Number of fuel assembly in the core	163
Coolant temperature at the reactor inlet (˚C)	298 ± 2.5
Average coolant temperature at the reactor outlet (˚C)	329 ± 5.0
Active core diameter (cm)	316
Active core height (cm)	375
Refueling frequency (months	12 (18)
Primary coolant temp. at core inlet (˚C)	298.2
Primary coolant temp. at core outlet (˚C)	328.9
Primary coolant pressure at reactor vessel outlet (MPa)	16.2
Feed water temperature at SG inlet (˚C)	225
Assembly	
Fuel assembly form	Hexagonal
Pitch between the assemblies	23.6
Number of fuel rod in the fuel assembly	311
Number of guide tubes	18
Fresh fuel assembly enrichment	1.6%, 2.4%, 3.6%
Fuel rod	
Hole diameter in the fuel pellet (mm)	1.5
Fuel pellet outside diameter (mm)	7.57
Cladding inside diameter (mm)	7.73
Cladding outside diameter (mm)	9.1
Fuel rod pitch (mm)	12.75
Fuel pellet material	UO^2
Cladding material	Alloy Zr + 1% Nb
Fuel pellet density (g/cm^3)	10.4–10.7

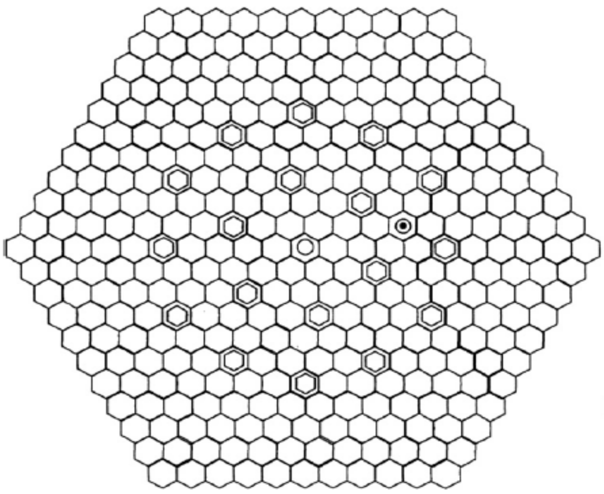

Fig. (1): VVER-1200 assembly configuration

The fuel is ceramic uranium dioxide pellets (UO_2 with a melting point of 2800 °C). Four types of enriched fuels (1.6%, 2.4%, 3.6% and 4.95%) are used in designing the VVER-1200 core. The cylindrical pellets are then put into tubes of a corrosion-resistant zirconium metal alloy plus 1% Nb which are back filled with helium to aid heat conduction. Sintered UO_2 pellets with different enrichment are stacked inside the cladding. Fig.2 illustrates a horizontal cross-section of the VVER-1200 core modeling. Table 2 describes the ten different types of fuel assemblies. To control and maintain the safety of the reactor, groups of control rod are employed. The locations of the control rod groups in the reactor assembly are shown in Fig. 3, where the control rods are inserted in the guide tube. The multiplication factor K_{inf} values for the different fuel enrichments suggested have been investigated as illustrated in Fig. 4 it is observed that the K_{inf} values increase when the fuel enrichment increases. The K_{inf} values for different fuel enrichments decreases with burnup until reaching steady state at which the rate of ^{235}U burnt equal to the rate of ^{239}Pu gained. In the VVER-1200 assembly, boron carbide is used in the guide tube to manage the excess reactivity. Fig.5 illustrates the variation of the K_{inf} values for different fuel enrichment at different B_4C concentration. The ability of B_4C to absorb the thermal neutrons prevents more fission reaction occurs. Therefore, the presence of the B_4C in the guide tube decreases the K_{inf} values of the assembly. The insertion of the B_4C rod with 20% decreases the K_{inf} values by 390 pcm, while in the case of 36% decreases by 928 pcm.

The variation of the K_{inf} values of the core with burnup is illustrated in Fig.6. The core performance is estimated from the assembly burnup analysis assuming a 10-batch fuel-management. The average K_{inf} of the core can be calculated from equation 1.

$$K_{inf} = 1/(f_1/k_{inf,1} + f_2/k_{inf,2} + f_3/k_{inf,3} + \ldots + f_i/k_{inf,n}) \ldots \quad (1)$$

Where n is the number of batches, $k_{inf,i}$ is the multiplication value calculated from the unit cell for the batch i and f_i is the fraction of the total core power generated by batch i. The value of f_i can be determined from a 3-D analysis of a representative core. The attainable discharge burnup is the burnup of the fuel at the EOC from the n^{th} batch.

The thermal neutron flux is the main responsible for the fission chain reaction in light water reactor. Therefore, it is important to investigate the thermal neutron flux distribution through the fuel assembly. The radial thermal neutron flux (E<0.625ev) visualization for VVER-1200 assembly fueled with UO_2 with enrichment 4.95% is presented in Fig. 7. The Local thermal flux peaks inside the control rod tubes that are filled with water when they are pulled out. This is because of the ability of the hydrogen atoms to reduce the energy of the fast neutrons that produce from the fission reaction. The radial total power distribution is presented in Fig. 8 it can be seen that there is no power in the position of the control rod that is filled with water. The power values of the fuel rods near the water rods are larger than other fuel rods due to their subjection of a large number of thermal neutrons

MCNPX code has been used to calculate the υ and Q value in the VVER-1200 assemblies at different conditions. MCNPX takes the prompt recoverable energy per fission (Q-prompt) and multiplies that value by the constant normalization factor that was calculated by dividing 200 MeV by the prompt fission energy release of ^{235}U. The β_{eff} is determined in order to quantify the contribution of delayed neutrons to the fission as safety and control measurement. The presence of delayed neutrons plays a significant role in reactor control due to its impact on reactor power change rates. The effective delayed neutron fraction (β_{eff}) can be calculated using equation 2.

$$\beta_{\mathrm{eff}} = 1 - \left(\frac{K_P}{K}\right) \quad (2)$$

Where k is the total effective eigenvalue for both prompt and delayed neutrons and k_p is the effective prompt eigenvalue. Both eigenvalues can be obtained from the straight calculation mode of MCNPX [9]. Table 3 illustrates the main neutronic parameters of the VVER-1200 assembly at different fuel enrichment. By analyzing these results, the values of υ, Q-value and the β_{eff} were found to decrease as the fuel enrichment percentage increased. This is because, the concentration of the produced Pu isotopes at low fuel enrichment is higher than that produced at high fuel enrichment as illustrated in Fig.9. ^{238}U is the main responsible for the formation of Pu and its isotopes. At low fuel enrichment, a large quantity of ^{238}U is more subject to thermal neutrons than at higher fuel enrichment so the concentration of the ^{238}U consumed per initial heavy material (IHM) in the case of FA1 is larger than that consumed in the case of FA2 as illustrated in Fig.10. With the onset of nuclear fission, the concentration of ^{235}U per IHM decreased with fuel burnup as illustrated in Fig.11 and the Pu isotopes are produced. In the case of ^{235}U and ^{238}U, the total neutron yield per fission is 2.4355 ± 0.0023 and 2.819 ± 0.020 respectively, while in the case of ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu is 3.00 ± 0.14, 2.8836 ± 0.0047, 3.086 ± 0.025, 2.9479 ± 0.0055 and 3.189 ± 0.035 respectively. Therefore, the υ values increase as the fuel enrichment decreases. Also, the β_{eff} values decrease with increasing the fuel enrichment. This is due to the formation of Pu isotopes and the deformation of ^{235}U. The delayed neutron yield in case of ^{235}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu is 0.0162 ± 0.0005, 0.0047 ± 0.0005, 0.0065 ± 0.0003, 0.0090 ± 0.0004, 0.0160 ± 0.0008 and 0.0183 ± 0.0010.

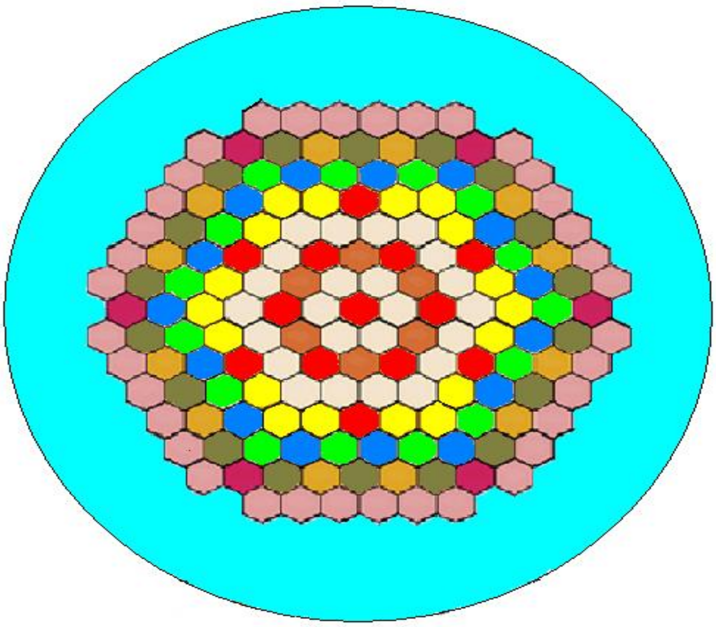

Fig. (2): Horizontal cross-section of the VVER-1200 core modeling

Table (2): Description of the ten different types of the fuel assemblies

Color	Fuel type	Fuel enrichment (%)	No. burnable poison (boron content, g/cm^3)	Symbol	No. assembly
	16	1.6	------	FA1	24
	24	2.4	------	FA2	13
	36	3.6	------	FA3	18
	4.95	4.95	------	FA4	36
	24B20	2.4	18 (0.020)	FA5	6
	24B36	2.4	18 (0.036)	FA6	18
	36B36	3.6	18 (0.036)	FA7	15
	36B20	3.6	18 (0.020)	FA8	12
	49B20	4.95	18 (0.020)	FA9	6
	49B36	4.95	18 (0.036)	FA10	15

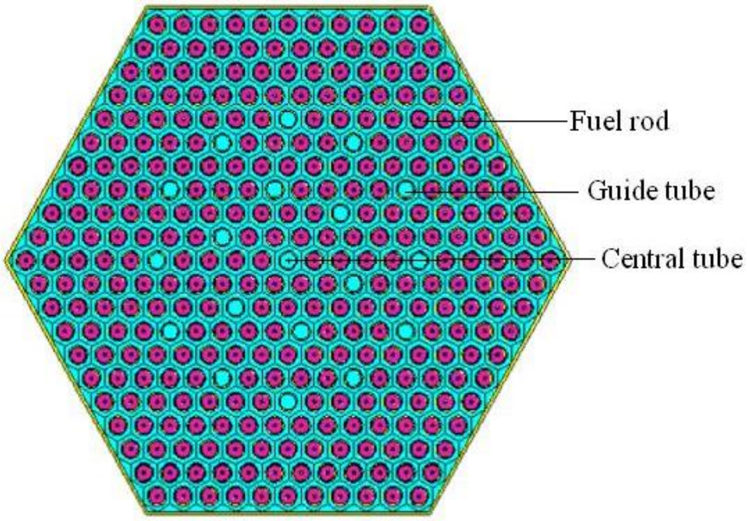

Fig. (3): Horizontal cross section of MCNPX model of VVER-1200 assembly

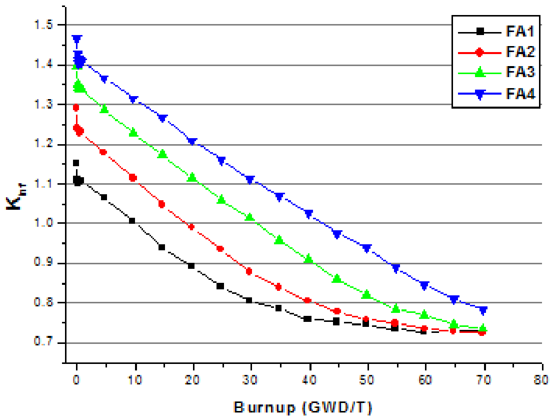

Fig. (4): Variation of the K_{inf} values with burnup at different fuel enrichment

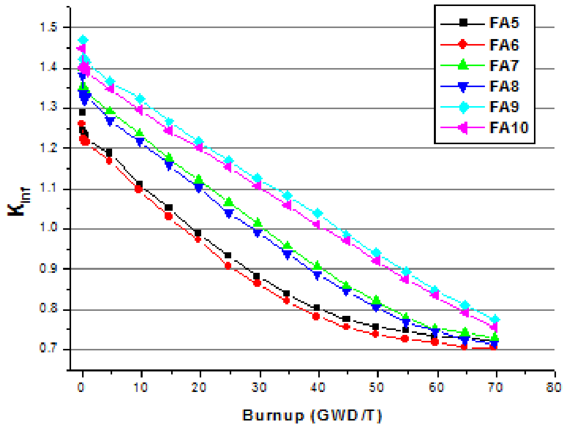

Fig. (5): Variation of the K_{inf} values with burnup at different concentration of burnable poison

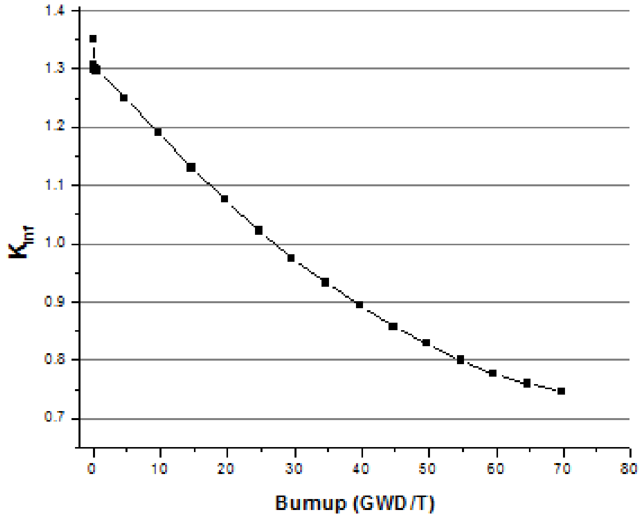

Fig. (6): Variation of the K_{inf} values of the core with burnup

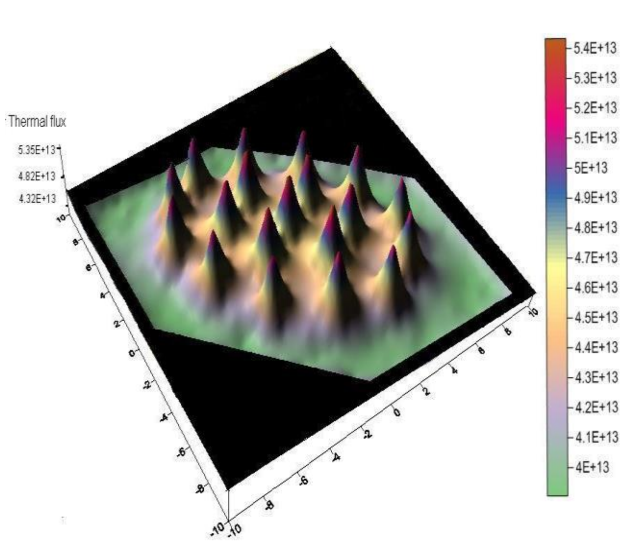

Fig. (7): Radial thermal neutron flux distribution through VVER-1200 assembly

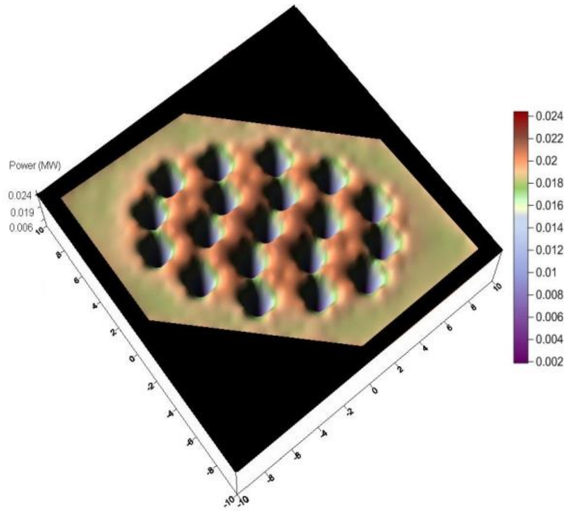

Fig. (8): Radial power distribution through VVER-1200 assembly

Table 3 Neutronics parameters of the VVER-1200 assembly at different fuel enrichment.

Parameters	Fuel enrichment			
	1.6%	2.4%	3.6%	4.95%
υ	2.68	2.65	2.61	2.58
Q-values (MeV)	205.65	205.10	204.34	203.59
β_{eff} (pcm)	811	699	637	595

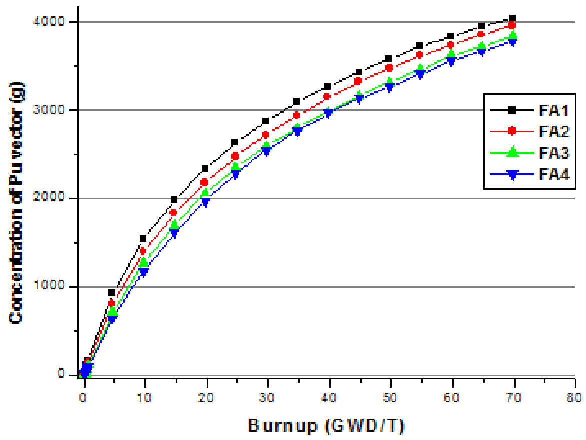

Fig. (10): Variation of the U-238/IHM (wt%) with burnup for different fuel assembly

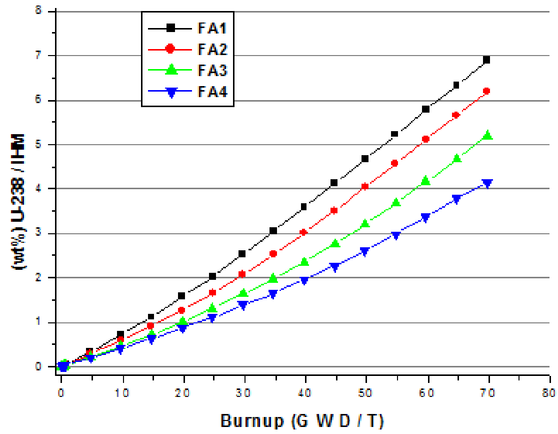

Fig. (11): Variation of the U-235/IHM (wt%) with burnup for different fuel assembly

Void fraction

In the VVER-1200, the void fraction may be formed as a result of two reasons; firstly, if the coolant/ moderator (water) is lost from the reactor in some types of accident (loss of coolant accident). Secondly, if the temperature inside the reactor rises, the moderator/coolant boils. This boiling leads to voids inside the reactor (steam bubbles). The author

suggested theoretically that the fuel was burnt in the existence of voids to analyze what will be happening. Fig. 12 illustrates the K_{inf} of the VVER-1200 assembly fueled with UO_2 with enrichment 4.95% at the different void fraction. At the begin of the cycle (BOC), the K_{inf} values at 20% voids are smaller than that at 0% voids while at the end of the cycle (EOC), the K_{inf} values at 20% voids is larger than that at 0% voids. Fig. 13 illustrates the variation of ^{235}U/IHM (wt %) concentration with burnup at the different void fraction. As the water become steam, its density decreases and its ability to slow down the fast neutrons decreases. The fast neutrons are hardly absorbed by fissile nuclei than the slow neutrons. Therefore, less number of nuclei is undergoing fission and, consequently, the K_{inf} values decrease at the BOC and increases at the EOC as the void fraction increases. The neutronic parameters of the VVER-1200 assembly fueled with UO_2 with enrichment 4.95% at different voids are listed in Table 4. It is observed that the investigated parameter values increase when the void fraction increases. This is because the concentration of the Pu vector concentration increases when the void fraction increases as illustrated in Fig. 14. The void fraction inside the reactor has a significant effect on the reactivity of the reactor. Therefore, it is important to calculate the void coefficient of the reactivity. It is observed that the negative of the reactivity increase as the void fraction increases. A void coefficient of reactivity (α_v) is the change in reactivity per change in the void fraction. It can be calculated using equation 3.

$$\alpha_v = \frac{d\rho}{dx} \qquad (3)$$

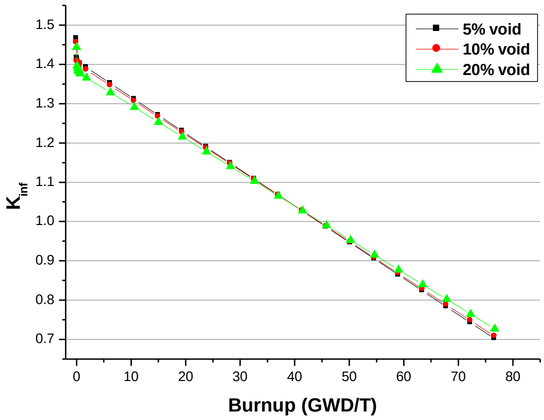

Fig. (12): Variation of the K_{inf} of the VVER-1200 assembly fueled with UO_2 with enrichment 4.95% at the different void fraction

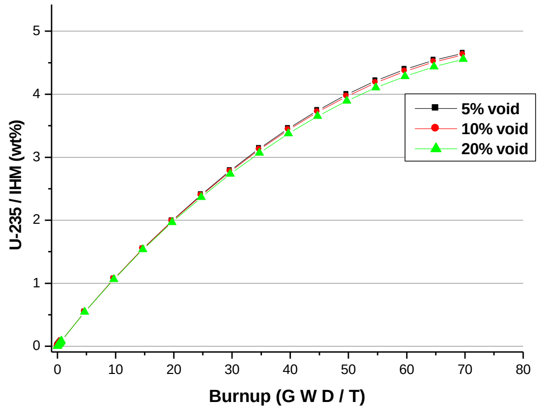

Fig. (13): Variation of the ^{235}U/IHM concentration of the VVER-1200 assembly fueled with $UO_2$with enrichment 4.95% at the different void fraction

Table (4): Neutronics parameters of the VVER-1200 assembly at the different void percentage

Parameters	Void fraction		
	5 %	10 %	20%
υ	2.582	2.583	2.584
Q-values (MeV)	203.6	203.62	203.65
β_{eff} (pcm)	605	628	757
α_v (pcm/%)	-15	-55	-82

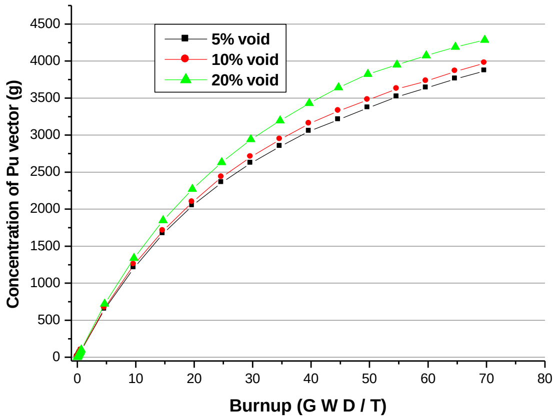

Fig. (14): Variation of the Pu vector of the VVER-1200 assembly fueled with UO_2 with enrichment 4.95% at the different void fraction

Burnable absorber

The BA is a material that is used to control the nuclear fission reaction in the reactor. The BA is a strong neutron absorber that is converted into a less neutron absorbent nuclide after capturing a neutron. Two general types of BAs can be used with VVER-1200 fuels: Integral burnable absorbers (IBAs) and Burnable poison rods (BPRs). IBAs are non-removable, neutron-absorbing materials used as fixed components of a fuel assembly [10]. BPRs are rods containing neutron absorbing materials that can be inserted into the guide tubes of a VVER-1200 assembly during operation and are commonly used for reactivity control and enhancing fuel utilization. To compensate the excess reactivity present after reloading UO_2 with enrichment 4.95%, sufficient quantities of gadolinium were added to the fuel in the assembly or boron carbide rods were inserted into the fuel assembly. Table 5 illustrates the effect of the BAs at the different void fraction on the reactivity of the VVER-1200 assembly. The existences of gadolinium or boron carbide at BOC suppress the reactivity (i.e less quantity of ^{235}U was burnt). This negative feedback effect is one of the safety features of VVER-1200. At the EOC, the reactivity values became positive due to two reasons; firstly, the degradation of the BAs materials. Secondly, the concentration of the remaining ^{235}U with BAs was larger than that without BAs. The existence of the void fraction affects the reactivity of the reactor, as the thermal neutron flux decrease when the void fraction increases. At the BOC, The reactivity at 20% voids is more negative than at 0% voids while at the EOC, the reactivity at 20% voids is more positive than at 0% voids. This is due to the small quantity of ^{235}U that burnt in the case of 20% at the BOC.

The natural abundance of the ^{155}Gd equals 15%, whilst the natural abundance of ^{157}Gd is 16 %. The neutron capture on ^{155}Gd and ^{157}Gd follows the formation of ^{156}Gd and ^{158}Gd with much lower absorption cross-sections as illustrated in Fig. 15. This process is described as gadolinium burnout. The thermal absorption cross-section of natural gadolinium exceeds fuel absorption cross-section, thus it burns faster in neutron field. The absorption cross section of ^{157}Gd for thermal neutrons is larger than fast neutron so the conversion of ^{157}Gd to ^{158}Gd at 0% voids is larger than at 20% voids.

The ability of boron carbide to absorb neutrons without forming long-lived radioactive nuclides makes it attractive as an absorbent for neutron radiation arising in nuclear power plants [11]. Natural boron contains two isotopes, namely ^{10}B and ^{11}B. Absorption of neutron mainly relies on ^{10}B whose thermal neutron section can be up to 3800 barns. Abundance of ^{10}B in boron carbide for reactor should be larger than 19% and the reaction equation with neutron is

$$^{10}B + 1n \rightarrow {}^{7}Li + 4He.$$

A disadvantage of boron is that it captures neutrons by an (n,α) reaction so that while it is in an operating reactor helium atom accumulate within the crystals of the boron carbide. These tend to form little bubbles of gas that disrupt the structure of the crystals and damage the material (swelling). This, together with the loss of the ^{10}B, limits the life of a rod used for controlling the reactor while it is operating. The absorption cross section of ^{10}B for thermal neutrons is larger than fast neutron so the conversion of ^{10}B to ^{3}Li at 0% voids is larger than at 50% voids as illustrated in Fig.16.

Fissile inventory ratio (FIR)

Fissile inventory ratio (FIR) is defined as the core fissile inventory at a specific time divided by the initial core fissile inventory. The FIR can be calculated by equation 4.

$$\mathrm{FIR}(t) = \frac{N_{U235}(t) + N_{Pu239}(t) + N_{Pu241}(t) + N_{Np239}(t)}{N_{U235}(0) + N_{Pu239}(0) + N_{Pu241}(0) + N_{Np239}(0)} \quad \text{……. (4)}$$

$N_{fissile}(t)$ denotes the atom density of the fissile material at a given time, and $N_{fissile}(o)$ is the initial atom density of the fissile material. FIR (t) can be used to evaluate the breeding ability more intuitively. For uranium dioxide fuel, ^{239}Np atoms are counted as a fissile nuclide, as in the case of core shutdown, all ^{239}Np relatively quickly decay into fissile ^{239}Pu [12]. The FIR values are calculated for UO_2 with enrichment 4.95% at the different void fraction. The FIR in the case of 0% and 20% void decrease with burnup due to the fission of the ^{235}U as illustrated in Fig.17. The variation of the coolant/moderator density affects the fission reaction rate as the coolant/moderator is the main responsible for slowing down the fast neutron until they reach the thermal energy. The fast neutron absorption cross-section of ^{235}U is very small, so less ^{235}U is burned at 20% voids than at 0% voids. The absorption cross section of ^{238}U at fast neutrons is larger than that at thermal neutrons. Therefore, more ^{238}U atoms absorb neutrons and convert to ^{239}Np and ^{239}Pu at 20% voids than at 0% voids. Consequently, The FIR in the case of 0% voids is smaller than that in the case of 20% voids

Table (5): Effect of BAs at different void fraction on the reactivity of the VVER-1200 assembly

Fuel cycle	Reactivity (pcm)			
	IBAs (3% Gd_2O_3)		BPRs (B_4C)	
	0% Void	20% Void	0% Void	20% Void
BOC	-10952	-18271	-20669	-38025
MOC	-324	-8015	-14552	-25235
EOC	1161	6334	6923	8370

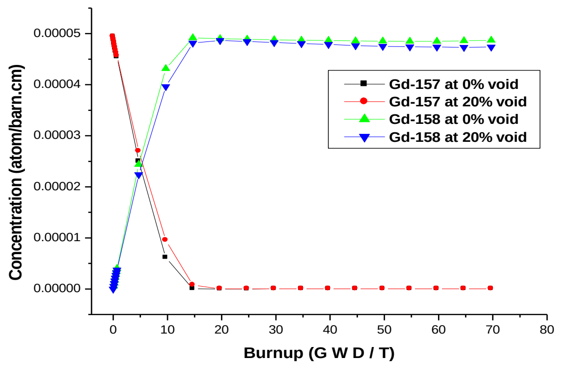

Fig. (15): Variation of the gadolinium concentration with burnup

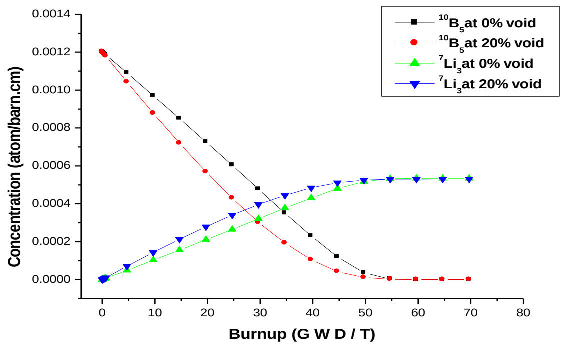

Fig. (16): Variation of the gadolinium concentration with burnup

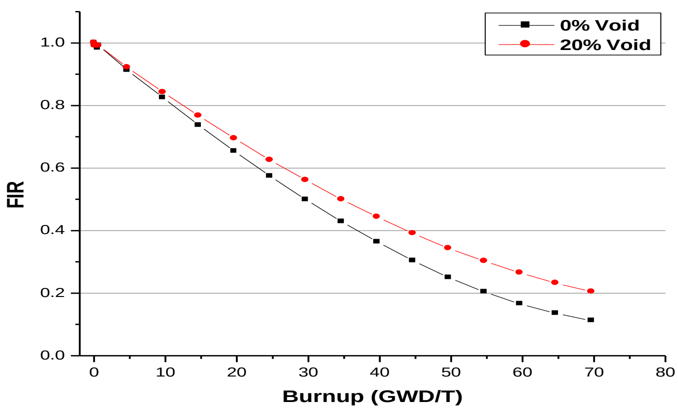

Fig. (17): Variation of the FIR with burnup at different voids

Spent fuel
The toxicity of plutonium is significant, and a long cooling time of approximately 100,000 years is necessary for it to decay to a level below that of natural uranium. The toxicities of plutonium isotopes are generated from ^{238}U. Furthermore, ^{235}U generates neptunium. However, its toxicity is negligible compared with that of natural uranium. If ^{238}U is removed from the fuel, the spent fuel has low toxicity and the cooling time to decay the toxicity to a level below that of natural uranium level might be drastically reduced to approximately 300 years [13].
Table 6 illustrates the effect of the fuel enrichment and void fraction on the TRU in the spent fuel of a VVER-1200 assembly. These elements represent the most hazardous radioactive nuclear waste products because of their long-term, high-level radioactivity. For UO_2 fuel with enrichment 1.66%, nearly 80% of spent fuel consists of plutonium isotopes and 20% of other TRU elements, while for UO_2 fuel with enrichment 4.95%, only 44.5 % of spent fuel consist of plutonium isotopes. The plutonium concentration in the spent fuel depends on (a) the ^{238}U concentration (b) The coolant or moderator density. The TRU elements concentrations were investigated for UO_2 with enrichment 4.95% at different void concentration. It is observed that most of the TRU element concentration increases when the void fraction increases.^{236}U is generally considered a nuisance and long-lived radioactive waste. It is found in spent nuclear fuel and in the reprocessed uranium made from spent nuclear fuel. ^{239}U is usually produced by exposing ^{238}U to neutron radiation in a nuclear reactor. ^{239}U decays into ^{239}Np through beta decay. ^{239}Np decays in a second important step that ultimately produces fissile ^{239}Pu. Plutonium constitutes the majority and most dangerous component of nuclear waste.

Doppler Effect
Increasing the fuel temperature is considered one of the main problems that result from the LOCA. The fuel temperature plays an effective role in the reactivity of the reaction due to the Doppler broadening of the resonance absorption in the UO_2 fuel. The Doppler reactivity coefficient is the change in the multiplication factor K_{inf} of the system caused by variations in the fuel cross-sections from temperature changes. The Doppler coefficient must be negative to be considered safe which means that the k_{inf} decreases as the fuel temperature increases as illustrated in Table 6. This negativity is useful in an accident scenario in which the core overheats and the fuel temperature rises. The higher temperature causes the reactor to fission less and decrease in power.

Table (6): The composition of TRU in the spent fuel of a VVER-1200 assembly at different condition

Isotopes	Atom concentration (g)			
	Fuel enrichment		Void	
	1.66%	4.95%	Void 5%	Void 20%
^{236}U	8.421E+02	3.296E+03	3.292E+03	3.295E+03
^{239}U	4.734E-01	3.780E-01	3.775E-01	3.659E-01
^{237}Np	1.205E+02	2.916E+02	3.079E+02	3.457E+02
^{238}Pu	6.810E+01	2.257E+02	2.346E+02	2.655E+02
^{239}Pu	8.551E+02	9.728E+02	1.032E+03	1.260E+03
^{240}Pu	1.201E+03	1.247E+03	1.273E+03	1.321E+03
^{241}Pu	4.463E+02	4.999E+02	5.338E+02	6.428E+02
^{242}Pu	1.414E+03	8.343E+02	8.317E+02	7.956E+02

Table (6): Doppler reactivity coefficient at different temperature

Temperature (k)	Doppler reactivity coefficient (pcm/k)
400	-331.48
600	-482.903
900	-513.557
1200	-784.153

Conclusion

A model of MCNPX for VVER-1200 has been designed and applied for void fraction analysis. This model has been used to investigate the variation of the K_{inf} at different fuel assembly enrichment and different void fraction. It is observed that the variation of the υ, Q and β_{eff} values depends on the Pu isotopes concentrations that are breeding with burnup. The presence of voids in the reactor increases the production of the Pu isotopes. It is found that the FIR values increases when the void fraction increases. This is because two reasons; the first, a small quantity of ^{235}U undergoes fission as the void fraction increases. The second, the presences of void fractions increase the breeding of Pu isotopes. The presence of void fraction with either IBAs or BPRs increases the negativity of the reaction. The effect of the void fraction on the B_4C is larger than Gd_2O_3. The occurrence of the void fraction leads to increase the fuel temperature, consequently, the negativity of the Doppler coefficient of VVER-1200 increases. The concentration of the most hazard radioactive nuclear waste increases when the void fraction increases,

References

1. Obaidurrahman, K., Doshi, J.B. (2011) Spatial instability analysis in pressurized water reactors. Annals of Nuclear Energy. 38, 286-294.
2. Status report 108 - VVER-1200 (V-491) (VVER-1200 (V-491)).
3. Nikitin, K., Mueller, P., Martin, J., Doesburg, W.V., Hiltbrand, D. (2016) BWR loss of coolant accident simulation by means of RELAP5. Nuclear Engineering and Design. 309, 113–121.
4. Rassame, S., Hibiki, T., Ishii, M. (2017) ESBWR passive safety system performance under loss of coolant accidents. Progress in Nuclear Energy. 96, 1-17.
5. Yu, S.O., Cho, Y.J., Kim, S.J. (2017) Effect of emergency core cooling system flow reduction on channel temperature during recirculation phase of large break loss-of-coolant accident at wolsong unit 1. Nuclear Engineering and Technology. 49, 5, 979-988.
6. Risovany,V.D., Varlashova,E.E., Suslov, D.N. (2000) Dysprosium titanate as an absorber material for control rods. Journal of Nuclear Materials. 281, 84-89.
7. Sahin, S., Ahmed, R., Khan, M.J. (2012) Assessment of criticality and burn up behavior of candu reactors with nuclear waste trans uranium fuel. Progress in Nuclear Energy. 60, 19-26.
8. Mozafari, M.A., Faghihi, F. (2013) Design of annular fuels for a typical VVER-1000 core: Neutronic investigation, pitch optimization and MDNBR calculation. Annals of Nuclear Energy. 60, 226–234.
9. Bretscher, M.M. (1997) Evaluation of reactor kinetic parameters without the need for perturbation codes. International Meeting on Reduced Enrichment for Research and Test Reactors. Argonne National Laboratory, Illinois 60439-4841. Jackson Hole, Wyoming, USA.
10. Galahom, A.A. (2017) Study of the possibility of using Europium and Pyrex alloy as burnable absorber in PWR. Annals of Nuclear Energy. 110, 1127-1133.
11. Sokhansanj, A., Hadian, A.M. (2012) Purification of Attrition Milled Nano-size Boron Carbide Powder. 2nd International Conference on Ultrafine Grained & Nanostructured Materials. International Journal of Modern Physics: Conference Series. 5, 94-101.
12. Liu, S., Cai, J. (2013) Convergence analysis of neutronic/ thermohydraulic coupling behavior of SCWR. Nuclear Engineering and Design. 265, 53- 62.
13. Fukaya, Y., Goto, M., Nishihara, T. (2015) Study on erbium loading method to improve reactivity coefficients for low radiotoxic spent fuel HTGR. Nuclear Engineering and Design. 293, 30-37.