



MARMARA UNIVERSITY
INSTITUTE FOR GRADUATE STUDIES
IN PURE AND APPLIED SCIENCES



ANALYSIS OF NEUTRONIC PARAMETERS OF THE VVER-1200, PWR AND TRIGA MARK II REACTOR

Doğukan AÇIKGÖZ

MASTER THESIS

Department of Physics

Thesis Supervisor

Prof. Dr. Kadir ESMER

Thesis co-Supervisor

Assoc. Prof. Dr. Tayfun AKYÜREK

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ABSTRACT

ANALYSIS OF NEUTRONIC PARAMETERS OF THE VVER-1200, PWR AND TRIGA MARK II REACTOR

In the first part of this study, we modeled PWR and VVER-1200 fuel assemblies using the MCNP 6.1 code and compared the neutronic parameters we obtained by simulating them for different scenarios and fuel assembly temperatures under the same conditions in the absence of control rods. In the second part of this study, we modeled the fuel element of the TRIGA Mark II research reactor at Istanbul Technical University using the MCNP 6.1 code and simulated it according to the information we obtained from the reactor logbook, and determined the average burnup value of the critical mass in the fuel element.

ÖZET

VVER-1200, PWR VE TRIGA MARK II REAKTÖRÜNÜN NÖTRONİK PARAMETRELERİNİN ANALİZİ

Çalışmanın ilk kısmında PWR ve VVER-1200 yakıt demetlerini MCNP 6.1 kodunu kullanarak modelleyip kontrol çubuklarının yokluğunda aynı şartlar altında, farklı senaryolar ve yakıt demeti sıcaklıkları için simüle ederek elde ettiğimiz nötronik parametreleri karşılaştırdık. Bu çalışmayı oluşturan ikinci kısımda ise İstanbul Teknik Üniversitesi'nde bulunan TRIGA Mark II araştırma reaktörünün yakıt elemanını MCNP 6.1 kodunu kullanarak modelleyip reaktör seyir defterinden elde ettiğimiz bilgilere göre simüle ederek yakıt elemanındaki kritik kütlenin ortalama yanma değerini elde ettik.

SYMBOLS

UO_2	: Uranium dioxide
${}_0^1n$	: Neutron
${}_{92}^{235}U$	: Uranium-235
${}_{92}^{236}U$	: Uranium-236
${}_{38}^{90}Sr$	: Strontium-90
${}_{54}^{144}Xe$	: Xenon-144
${}_{35}^{87}Br$	: Bromine-87
${}_{57}^{146}Ba$	: Barium-146
${}_{37}^{96}Rb$	: Rubidium-96
${}_{55}^{137}Cs$	: Cesium-137
${}_{40}^{97}Zr$	: Zirconium-90
${}_{52}^{137}Te$	: Tellurium-137
${}_{36}^{92}Kr$	: Krypton-92
${}_{56}^{141}Ba$	: Barium-141
${}_{90}^{232}Th$	: Thorium-232
${}_{90}^{233}Th$	: Thorium-233
β^-	: Beta decay
${}_{91}^{233}Pa$	: Protactinium-233
${}_{92}^{233}U$	: Uranium-233
${}_{92}^{232}U$	: Uranium-232
${}_{91}^{232}Pa$	: Protactinium-232
${}_{91}^{231}Pa$	: Protactinium-231
U_3O_8	: Triuranium octoxide
k	: Neutron multiplication factor
k_∞	: Infinite neutron multiplication factor
k_{eff}	: Effective neutron multiplication factor
P	: Non-leakage probability

P_T	: Non-leakage probability of thermal neutrons
P_F	: Non-leakage probability of fast neutrons
η	: Reproduction factor
ϵ	: Fast fission factor
f	: Thermal utilization factor
p	: Resonance escape probability
Gd_2O_3	: Gadolinium oxide
ZrB_2	: Zirconium diboride
${}^{239}_{93}Np$	: Neptunium-239
${}^{239}_{94}Pu$	: Plutonium-239
${}^{135}_{53}I$	: Iodine-135
${}^{135}_{54}Xe$	: Xenon-135
${}^{135}_{55}Cs$	: Cesium-135
${}^{135}_{56}Ba$	: Barium-135
${}^{149}_{60}Nd$	: Neodymium-149
${}^{149}_{61}Pm$	: Promethium-149
${}^{149}_{62}Sm$	: Samarium-149
${}^{150}_{62}Sm$	: Samarium-150
$U - ZrH_{1.6}$	: Uranium zirconium hydride

ABBREVIATIONS

CANDU	: Canada Deuterium Uranium
MCNP	: Monte Carlo N-Particle Transport Code
PWR	: Pressurized Water Reactor
VVER	: Voda-Vodyanoi Energetichesky Reaktor
IBA	: Integral Burnable Absorber
IFBA	: Integral Fuel Burnable Absorber



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1. INTRODUCTION

The discoveries and developments that took place at the end of the 19th and beginning of the 20th century in the fields of which we call as nuclear physics and particle physics today, led humanity to consider and build a new technology for production of energy. Although most of the discoveries that the theory of nuclear technology based on revealed themselves at those times, one of the most crucial and important one took place in the year of 1789, the discovery of the uranium, which was discovered by the German chemist Martin Klaproth [1]. More than a hundred years later from this discovery, in 1896, Henri Becquerel discovered that the uranium nucleus spontaneously splits and emits electromagnetic radiation [2]. Building on Becquerel's work, Marie Curie and Pierre Curie put their efforts and made different experiments, then as a result, they discovered 2 different nuclei which can spontaneously split and emit radiation (radium and polonium), and called the phenomenon of nucleus emitting radiation by splitting as radioactivity [3,4,5]. Becquerel and Curies shared the 1903 Nobel Prize as a result of their studies and discoveries on radioactivity. Discoveries of the relation between energy and mass of a body by Einstein in 1905 [6] and almost all of the mass of the atom is concentrated in the atom's nucleus by Rutherford in 1909, gave the first signals of the energy that can be obtained from the atom's nucleus. Theories and studies carried out to better understand the nucleus of the atom led to the discovery of the neutron, the particle that plays a critical role in the reactions that enable us to obtain energy from reactors. The possibility of existence of a neutron first was mentioned by Rutherford in 1920, and the existence of a neutron was discovered by Rutherford's student, James Chadwick in 1932 [7]. Chadwick won the Nobel Prize in 1935 after this great discovery. As a result of a better understanding of the structure of the atomic nucleus with the discovery of the neutron, scientists have carried out new studies by going beyond the discoveries made before. One of the most important study was made by German scientists, Fritz Strassman and Otto Hahn, in the year of 1938 [8]. They bombarded the uranium nucleus with neutrons and discovered that the interaction

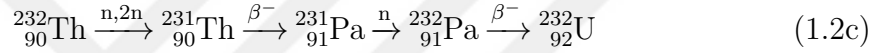
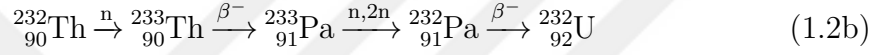
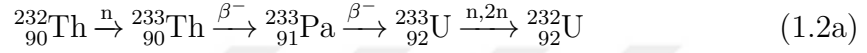
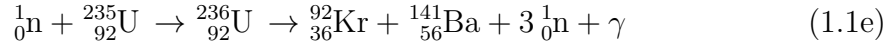
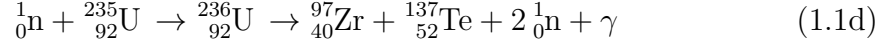
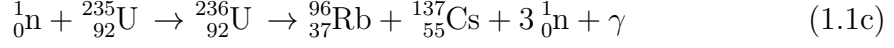
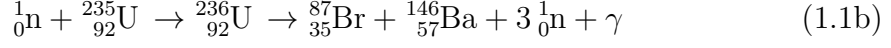
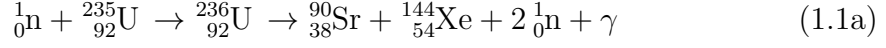
between neutron and uranium nucleus causes the barium to occur after the bombarding process. Lisa Meitner and Otto Frisch explained the beyond of the Hahn and Strassmann's discovery and named this event as the "fission" [9]. This "event" called fission is the reaction that makes it possible us to obtain energy from nuclear reactors. It is important to have a fission reaction that is sustainable and can feed itself to produce long term and efficient energy inside the reactors. Enrico Fermi obtained the first fission chain reaction while he was working on Manhattan Project in 1942. One of the first decision to use the nuclear energy for the benefit of the lives made by USA in 1946, by founding the Atomic Energy Commission and giving them the mission to build and produce energy for beneficial reasons [10]. As a result of this given mission to the commission, Experimental Breeder Reactor I was built in the year of 1951 and the first electricity generation from a reactor in the world was obtained. The first non-military commercial reactor in the world called "Obninsk Nuclear Plant" was built by the USSR in 1954. After a short time from this development, USA built its first commercial reactor called Shippingport Reactor in 1957. As a result of all these steps and developments in nuclear physics, reactors operating with different fuels and working principles have been developed, and thus nuclear energy has become one of the most important opportunities for humanity to produce energy from. Nuclear energy supply approximately 10% of humanity's electricity needs and since its use reduce the carbon emissions, it maintains its place as a prominent energy type today [11].

Nuclear reactors are the structures that produce energy as a result of controlling the fission reaction. Having a controllable fission reaction is important not only for to produce energy but also to obtain sustainable reaction. Therefore, a self sustaining series of fission reactions should be obtained in the reactor core. These kind of reactions are called fission chain reactions. The fission chain reaction is based on the fact that the neutron released from a fission reaction interacts with at least one of the nuclei around it to form a new fission reaction, and the neutrons released as a result of this fission continue the fission reaction in the same way.

Figure 1.1 Fission Chain Reaction Scheme

onable material and fuel assemblies that formed by these fuel rods. consist of elements that can be burned by the fission reaction. Properly, the height, geometry of the core and fuel assemblies inside the core of reactor to reactor. The number of fuel rods inside the assembly and the assembly inside the core are also vary for different kind of reactors. Core assembly and the core can be hexagonal, square or ring shaped [1]

neutron and fissile ^{235}U occurs. Fission reactions of ^{235}U and neutron absorption of the ^{232}Th can be seen in the following equations,



Reactor types that this study covers have enriched uranium based fuels. They are enriched in terms of ^{235}U , which is called fissile element for the reason of it can undergo fission reaction with neutrons with low energy, even almost 0 kinetic energy. However, the natural uranium extracted from the earth contains only 0.7% of the isotope uranium-235 (^{235}U), so the concentration of ^{235}U needs to be increased in fuels used for reactors. This process is known as enrichment. The amount of enriched fuel varies depending on the type of reactor. Generally, low enriched fuels with a maximum of 5% ^{235}U are used. However, there are also reactors such as CANDU that utilize unenriched, natural uranium as fuel [13]. Over time, the uranium element in the earth undergoes oxidation due to oxygen, resulting in its natural occurrence as triuranium octoxide (U_3O_8) in nature. Before this compound is used as a fuel, it goes through several operations such as purification and as a result of these processes the uranium dioxide (UO_2) compound is obtained. The obtained UO_2 compound has low thermal conductivity and high melting point. Low thermal conductivity prevents the fuel from being affected by the increased environment temperature with the energy released as a result of fission, and high boiling point ensures that the fuel does not change its phase in high temperatures. In order for neutrons to interact with ^{235}U and for the chain reaction to take place, the neutrons that will

interact with the uranium nucleus must be at the thermal energy, 0.025 eV [14]. The water inside the core acts as both a coolant and moderator. Neutrons formed as a result of the fission reaction have high energy and they lose their energy by interacting with the water in which the fuel rods are immersed. As a result of this interaction, they reach the thermal energy level and can interact with ^{235}U . In order to obtain a sustainable energy from the reactor, the number of neutrons that can react with nucleus is as important as the energies of the neutrons. The ratio of the number of neutrons released as a result of fission and creates new fission reaction in the core to the number of neutrons that initiate the reaction that causes the release of those neutrons is defined as the neutron multiplication factor and is denoted by k [15]. Neutron multiplication factor can be formulized as the following,

$$k = \frac{\text{number of fissions in one generation}}{\text{number of fissions in preceding generation}} \quad (1.3)$$

In order to keep the fission occuring in the reactor under control and the energy obtained to be sustainable and safe, the ratio of the number of neutrons released as a result of the reaction which will create a new fission to the number of neutrons that initiate the reaction is expected to be 1. The reactor is defined as supercritical if the ratio is greater than 1, critical if it equals to 1, and subcritical if it is less than 1. If the ratio is greater than 1, it means that the number of fission is increasing in each step, and if this cannot be controlled, the total energy released from reactions will increase gradually, which may prevent the controllability of the reactions and create a major safety problem. If it is less than 1, it means that the number of fission is decreasing in each step, which prevents a sustainable reaction from occuring and the energy obtained from the reaction gradually decreases. When the ratio is 1, equal number of fission reactions take place in each step and this provides both a continuous and safe energy [15]. The criticality level of the reactor can be increased or decreased by the people operating the reactor according to the power value desired to be obtained from the reactor. The average number of neutrons released from the fission is 2.43. If the neutron multiplication factor is interpreted over this average value, having a factor equals to 1 means that one of the 2.43 neutrons released interacts with other fissionable nuclei and undergoes a fission reaction, and the remaining 1.43 neutrons

are lost via leakage or neutron capture. Some of the neutrons released from fission leak out of the reactor and these neutrons cannot create any reaction in the core. This is called leakage. Some of the neutrons interact with nuclei with high neutron absorption cross section that is placed in assemblies in order to keep the reaction under control and are absorbed by those nuclei. Neutron multiplication factor can also be defined by considering these loss events and are defined as k_{eff} and k_{∞} [16]. The parameter called k_{∞} is defined for an infinite sized reactor and can be formulated as the following,

$$k_{\infty} = \frac{\text{number of neutrons generated from fission}}{\text{number of lost neutrons in preceding generation via absorption}} \quad (1.4)$$

k_{∞} is defined as the ratio of the number of neutrons produced by fission in a generation to the number of neutrons lost by being absorbed by other nuclei in the preceding generation. In the case of a reactor with infinite size, the leakage factor is not taken into account since it gives you a reactor that no neutron can escape to the outside of it, that is, there can be no leakage. k_{eff} , on the other hand, is by definition the ratio of number of neutron produced by fission in one generation to the number of neutrons lost by absorption and leakage in the preceding generation and is formulated as follows [16],

$$k_{\text{eff}} = \frac{\text{number of neutrons generated from fission}}{\text{number of lost neutrons in preceding generation via absorption and leakage}} \quad (1.5)$$

The ratio of the k_{eff} and k_{∞} gives the non-leakage probability. If the ratio of these two parameters is taken then one can get,

$$\frac{k_{\text{eff}}}{k_{\infty}} = \frac{\text{number of lost neutron in preceding generation via absorption}}{\text{number of lost neutron in preceding generation via absorption and leakage}} \quad (1.6)$$

the right side of the equality is called the non-leakage probability. It can also be shown as,

$$\frac{k_{\text{eff}}}{k_{\infty}} = P \quad (1.7)$$

the relationship between k_{eff} and k_{∞} can also be found by using equation 1.5,

$$k_{\text{eff}} = P k_{\infty} \quad (1.8)$$

The k_{eff} value is equal to the k_{∞} multiplied by the non-leakage probability value. If the value of non-leakage probability is 1, that is, if there is no loss in the number of neutrons due to leakage in the system, the value of k_{eff} will be equal to the value of k_{∞} . However, if there is a loss due to leakage in the system, the non-leakage probability value gradually decrease from 1, so the k_{eff} value will take a lower value than the k_{∞} value. Neutrons released as a result of fission are fast and have thermal energy by interacting with water. The neutrons that may leak from the reactor can be both fast neutrons released as a result of fission and thermal energetic neutrons. Therefore, while considering the non-leakage probability in the equation above, the leakage probabilities of neutrons with both high and thermal energies should be considered. Thus, the equation can be written as,

$$k_{\text{eff}} = k_{\infty} P_T P_F \quad (1.9)$$

Where P_T represents the non-leakage probability of the neutrons with thermal energy and P_F represents the non-leakage probability of the neutrons with high energy. The non-leakage probability of thermal energetic neutrons can be defined as the ratio of the number of neutrons with thermal energy that do not leak from the reactor to the total number of neutrons with thermal energy, and the non-leakage probability of fast neutrons can be defined as the ratio of the number of fast neutrons that do not leak from the reactor to the number of fast neutrons. The value of P_T for a thermal reactor is 0.99 and P_F is 0.97 [17]. They can be formulated as follows respectively,

$$P_T = \frac{\text{number of thermal energetic neutrons do not leak}}{\text{total number of thermal energetic neutrons}} \quad (1.10)$$

$$P_F = \frac{\text{number of fast neutrons do not leak}}{\text{total number of fast neutrons}} \quad (1.11)$$

Understanding the parameters such as the energy of the generated neutrons and the number of neutrons with these energies, how much of them are absorbed by the fuel

or by the absorbers in the core, how much of its energy can fall into thermal level and start a reaction, and how much of it may leak from the reactor is important in order to achieve a desired and safe reaction in the reactor. Therefore, there are a few more parameters that exist for a better understanding of the neutron life cycle. The first of these parameters is the reproduction factor, which gives how many fast neutrons are released as a result of the fission formed by thermal neutrons absorbed by the fuel. The reproduction factor is equal to the ratio of the fast neutrons released after the reaction and the number of thermal neutrons that initiate that thermal fission and are absorbed by the fuel. The value of this parameter for a thermal reactor is 1.65 [17]. It can be formulated as follows,

$$\eta = \frac{\text{number of fast neutrons released from thermal fission}}{\text{number of thermal neutrons absorbed by the fuel}} \quad (1.12)$$

^{235}U nuclei interact with thermal neutrons and release fast neutrons. However, fission can occur not only with thermal neutrons, but also with fast neutrons. ^{235}U nuclei do not interact with fast neutrons, but ^{238}U nuclei can interact and cause fission to occur. For this reason, the fast neutrons released can be not only from the fission caused by thermal neutrons, but also from the fission reaction caused by fast neutrons. Another related parameter is the fast fission factor, which gives the ratio of fast neutrons released from all fission reactions in the core and fast neutrons released only from fission caused by thermal neutrons. The value of this parameter for a thermal reactor is 1.02 [17]. It can be formulated as follows,

$$\epsilon = \frac{\text{number of fast neutrons released from all fissions}}{\text{number of fast neutrons released from thermal fissions}} \quad (1.13)$$

After the fast neutrons released get thermal energy by interacting with water, the thermal neutrons formed can be absorbed both by the fuel and the absorber in the core. The measure of how much neutrons are absorbed by which material is determined by the parameter defined as thermal utilization factor. This parameter is equal to the ratio of the number of thermal neutrons absorbed by the fuel to the number of neutrons absorbed by all absorber materials in the core. Other materials that can absorb the neutron apart from fuel can be water, which is called moderator, and fission products with high neutron absorption cross section, which are released

as a result of fission and called poison. The value of this parameter for a thermal reactor is 0.71 [17]. The thermal utilization factor is formulated as follows,

$$f = \frac{\text{number of thermal neutrons absorbed by the fuel}}{\text{number of thermal neutrons absorbed by all reactor materials}} \quad (1.14)$$

Not all neutrons, whose begin to obtain the thermal energy as a result of interaction with the moderator, can reach thermal energy. Neutrons that start to slow down can be absorbed by the absorber materials in the environment and can escape from the reactor core as a result of leakage event. The ratio of the number of fast neutrons that can get thermal energy to the neutrons that start to slow down after interacting with the moderator in the system is called the resonance escape probability. The value of this parameter for a thermal reactor is 0.87 [17]. It is formulated as follows,

$$p = \frac{\text{number of neutrons reach thermal energy}}{\text{number of fast neutrons starts to slow down}} \quad (1.15)$$

Equation 1.8, effective neutron multiplication factor, can be formulized by using the parameters mentioned above. Which can be seen in the following expression,

$$k_{eff} = \eta f p \epsilon P_T P_F \quad (1.16)$$

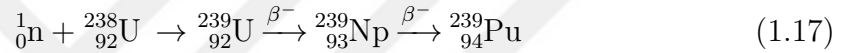
This equation is called six-factor formula. The six parameters mentioned above and which are the constituents of the six-factor formula are of great importance in order to understand the neutron life cycle in the reactor.

The losses due to absorption in the k_{eff} and k_{∞} parameters are due to the neutron absorber elements in the core. Neutron absorber elements are placed in fuel assemblies to control the reaction inside the reactor. These elements can be coated on the fuel rods or mixed with the UO_2 fuel and placed in the fuel rods. The rods in which the absorbers are coated on the fuel rods are called Integral Fuel Burnable Absorber (IFBA) rods, and the rods in which the absorbers are mixed with the fuel are called Integral Burnable Absorber (IBA) rods. In this study, IBA and IFBA rods were used in the fuel assembly. The amount of coating and the amount of absorber to be mixed with the fuel may vary according to the level of criticality desired to be achieved. As another option, these absorber elements can be placed

in structures called control rods, located at the top of the reactor core, which can be lowered into the reactor core at any time. As soon as the criticality level of the reactor increases, the control rods are lowered to the reactor core with the help of a mechanism and as a result of the absorbers inside these rods absorbing the fission neutrons, the number of neutrons that can undergo fission in the system is reduced, the number of reactions is reduced, thus reducing the criticality of the reactor. In this way, the power obtained from the reactor and the energy that will be released as a result of the reaction can be controlled. For this reason, neutron absorbers have an important role in the reactor in terms of both safety and reactor control. In this study, gadolinium oxide (Gd_2O_3) was used as IBA absorber and zirconium diboride (ZrB_2) was used as IFBA absorber. There are other elements used as IBA rods in the literature [18]. IBA and IFBA rods can be used simultaneously in reactor assemblies to achieve desired criticality levels. The cases where IBA and IFBA rods are used together are called hybrid rods and in this case, reactor criticality is expected to be lower than when IBA and IFBA rods are used separately, as there will be more neutron absorber material in the reactor assembly.

The total energy released from the fuel in the reactor core is called burnup which can be expressed in megawatt-day (MWd) [19]. This energy can be found by multiplying the power obtained from the reactor with the operation time of the reactor. The energy released from the unit mass of the fuel is called specific burnup and is expressed in megawatt day per metric tonnes (MWd/t) or megawatt day per kilogram (MWd/kg) [16]. Burnup varies depending on the temperature of the system, the amount of fuel used, the geometry of the reactor assembly, the reactor operation power and the number of fission reactions taking place in the system. As the temperature of the system increases, the energy of the neutrons formed as a result of fission will also increase, so it will be difficult for them to get the thermal energy. Therefore, as the system's temperature increases, the number of neutrons that can obtain thermal energy will also decrease, which will appear as a factor that reduces the number of interaction between thermal neutrons and ^{235}U nucleus. The reduction of neutron interaction with ^{235}U will also reduce the number of fission that

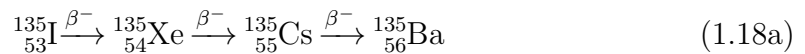
^{235}U nuclei can make in the system, thus preventing the decrease in the amount of ^{235}U in the system. Consequently, with increasing temperature, the reduction in the burnup value of the system becomes less pronounced. This is due to the decrease in the likelihood of interaction between ^{235}U and neutrons, while the probability of interaction between ^{238}U and neutrons increases. ^{238}U tends to interact with highly energetic fast neutrons compared to ^{235}U . The elevated temperatures in the system result in neutrons attaining even higher energies, increasing the likelihood of their interaction with ^{238}U . Consequently, ^{238}U absorbs these high-energy neutrons, leading to the formation of ^{239}U . ^{239}U then undergoes beta decay to form ^{239}Np , which subsequently undergoes further beta decay to produce ^{239}Pu . The decay scheme can be represented as follows,



Based on the aforementioned factors, in a system experiencing temperature increase, it is anticipated that the reduction in the concentration of ^{235}U will be less pronounced, while the decrease in the concentration of ^{238}U will become more significant. Furthermore, an elevated temperature is expected to lead to an increased concentration of ^{239}Pu resulting from the neutron absorption by ^{238}U , as compared to lower temperature conditions.

In the event of an excessive number of fission reactions occurring within the system, the fuel will undergo a higher degree of fission, resulting in increased burnup. Consequently, the ^{235}U nuclei, which serve as the target nuclei, will undergo additional splits, leading to the formation of different nuclei. Consequently, the overall number of fissionable nuclei present within the system gradually diminishes. This reduction in the number of fissionable nuclei leads to a decrease in the quantity of neutrons released as a result of fission during each subsequent generation. Consequently, the probability of fission events occurring within the system in each generation decreases. As a consequence, it is expected that the neutron multiplication factor will decrease with increasing burnup, thereby reducing the criticality of the reactor. The temperature of the system is also one of the important factors

effecting reactor criticality. As the temperature of the system increases, it will be more difficult for neutrons to reach thermal energy, so it will be more difficult to obtain a fission reaction at each step, and therefore the reactor criticality will be lower compared to lower temperatures. Another parameter that effects the criticality of the reactor is the fission products formed as a result of fission reaction. Some of these fission products absorb neutrons in the system due to their high neutron absorption cross sections and reduce the reactivity in the system. These products are called as poisons. The most important of these poisons are ^{149}Sm and ^{135}Xe . ^{135}Xe has almost 5000 times more neutron absorption capacity than ^{235}U , neutron absorption cross sections of ^{135}Xe and ^{235}U are 2.6 million barns and 507 barns respectively [20]. The formation of this nucleus as a fission product in the reactor will make it difficult to achieve a sustainable fission reaction in the reactor. Although ^{135}Xe is highly unlikely to form as the fission product of ^{235}U , approximately 0.5 % [21], this nucleus may also be released as a result of the decay of another fission product, ^{135}I . This nucleus is more likely to appear as a fission product than ^{135}Xe , approximately 6.3 %. For this reason, formation of ^{135}Xe in the reactor is usually due to the decay of ^{135}I . ^{135}I is a radioactive nucleus and undergoes beta decay, resulting in the formation ^{135}Xe . The concentration of ^{135}Xe in the reactor must be kept under control in order to create a sustainable fission reaction. ^{135}Xe nuclei formed as a result of ^{135}I decay are also radioactive and form ^{135}Cs nuclei by beta decay. ^{135}Cs nuclei decays beta and forms stable ^{135}Ba nuclei. However, this is not enough to eliminate the effect of ^{135}Xe on the reactor. This is because ^{135}I has a half life of 6.58 hours [22] and ^{135}Xe has a half life of about 9.13 hours [23]. This means that the ^{135}Xe formed remains in the reactor longer. The event that keeps the concentration of ^{135}Xe at the desired level is the interaction of ^{135}Xe with the neutrons released due to fission, and as a result, the stable ^{136}Xe is formed. The aforementioned formation and reactions of xenon can be seen in the equations below,



^{136}Xe nucleus is not a neutron absorber. Therefore, as long as this reaction between

^{135}Xe and neutrons continues to occur, the concentration of ^{135}Xe will remain at a certain point and will not effect the activity in the reactor. Although no further fission will take place in the environment when the reactor is shut down, fission products will be present in the reactor core. ^{135}Xe nucleus, which is released due to the beta decay of ^{135}I that is the fission product of ^{235}U , will not be able to interact with fission neutrons and form ^{136}Xe nuclei since the fission reactions stop in the reactor core. For this reason, the concentration of ^{135}Xe increases when the reactor is shut down. It takes approximately 60 hours to return to the level of concentration before it was shut down. Therefore, the ^{135}Xe concentration must return to the same level as before the reactor was shut down in order for the reactor to be operated again. In other words, reactor should not be operated for about 3 days. The figure 1.2 shows how the concentration of ^{135}Xe effects the reactivity after reactor shut down.

Figure 1.2 shows how the increase in concentration of ^{135}Xe after reactor shut down effects the reactivity according to different neutron flux values. Figure 1.2 indicates that when the neutron flux increases, the negative effect of ^{135}Xe on reactivity increases. As the neutron flux increases, the number of neutrons per unit surface per second in the reactor will be higher, and as a result, neutron interactions with ^{235}U will be higher. Thus, more iodine 135 nuclei will be formed in the system compared to low neutron fluxes and will cause more ^{135}Xe nuclei to form by beta decay. Therefore, as the flux increases, ^{135}Xe will decrease the reactivity more.

^{149}Sm nuclei are released from decay of another fission product, ^{149}Nd . ^{149}Nd is one of the fission products of ^{235}U , and the rate of release of this nucleus as a product is 1.4 %. ^{149}Nd nucleus formed from fission firstly undergoes beta decay to form the ^{149}Pm nucleus, and the ^{149}Pm nucleus beta decay to form the ^{149}Sm nucleus. Unlike the ^{135}Xe , the ^{149}Sm is not radioactive. Consequently, it does not undergo spontaneous decay, which can potentially result in more significant complications in the reactor. Keeping the ^{149}Sm concentration in equilibrium in the reactor is can be achievable only by the interaction of ^{149}Sm with the neutrons. ^{149}Sm nuclei interacting with neutrons absorb neutrons and form ^{150}Sm nuclei. The

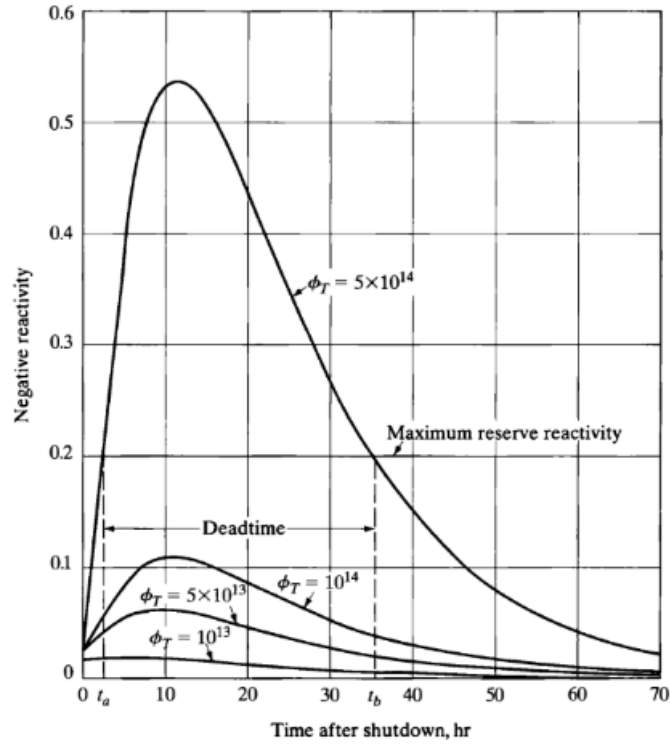


Figure 1.2 Negative Reactivity-Time After Shutdown graph for ^{135}Xe [24]

aforementioned formation and reactions of ^{149}Sm can be seen in the equations below,



The nucleus of ^{150}Sm exhibits a significantly low neutron absorption cross section. Therefore, this nucleus cannot play as much a role as ^{149}Sm in effecting the reactivity inside. The effect of ^{149}Sm concentration on reactivity with reactor shot down is shown in figure 1.3 for different neutron flux values.

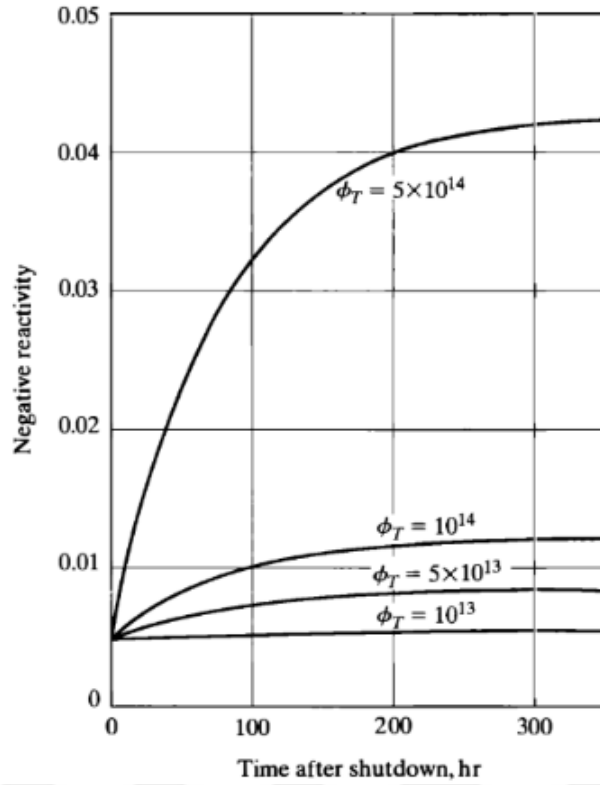


Figure 1.3 Negative Reactivity-Time After Shutdown graph for ^{149}Sm [25]

All these reactor parameters mentioned above can be obtained experimentally or it is possible to predict how neutronic parameters change during the operation period of a reactor by modelling it through a simulation program. Thanks to the simulation, the reactor parameters can also be obtained without the reactor being physically operated, so it may be possible to know how the reactor will behave during its operation, and to test how the reactor will be effected by changing some parameters in the core or fuel assembly. In contrast to the experimental approach, the simulation method, particularly in the field of nuclear studies, offers both economic and significantly reduced time requirements for obtaining results. The reactor that has the only operating license in Turkey and is operated for research is the ITU Triga Mark II reactor. There are programs such as ORIGEN, MCNP where nuclear reactors can be modeled and simulated. In this study, MCNP 6.1 program was used for modeling and simulation process. Monte Carlo N-Particle Code (MCNP), is a program in which all systems containing particles and where particle interactions take place can be modeled and simulated. This program can be used to model and

simulate nuclear reactors and particle accelerators, as well as in nuclear medicine. All neutronic parameters of the reactor, fuel burnup rate, isotope concentrations that change with burnup and reactor operation time can be obtained by using MCNP. In this study, the neutronic parameters of PWR and VVER-1200 assemblies for different scenarios and assembly temperatures were obtained and the results were compared. On the other hand, the burnup value of the fuel element in the core of the ITU TRIGA Mark II reactor was found by using MCNP 6.1 code.

To calculate the concentration of isotopes whose concentration changes with reactor operation time in a reactor core or assembly, MCNP solves set of differential equations, called Bateman Equations, which are used to find the isotopic depletion of any material in a system where the concentration of the isotope changing with time, as can be seen in the following,

$$\frac{dN_m(\vec{r}, t)}{dt} = -N_m(\vec{r}, t)\beta_m + \sum_{k \neq m} N_k(\vec{r}, t)\gamma_{k \rightarrow m}, \quad (1.20)$$

where β_m and $\gamma_{k \rightarrow m}$ are,

$$\beta_m = \lambda_m + \sum_r \int \sigma_{m,r}(E)\Phi(E, t)dE, \quad (1.21a)$$

$$\gamma_{k \rightarrow m} = \sum_{k \neq m} N_k L_{km} \lambda_k + \sum_{k \neq m} \sum_r \int Y_{km,r}(E) \sigma_{k,r}(E) \Phi(E, t) dE. \quad (1.21b)$$

In equation 1.15, $N_m(\vec{r}, t)$ and β_m terms are the atom density and total transmutation probability of isotope m respectively. $\gamma_{k \rightarrow m}$ is the transmutation probability of isotope k to isotope i by decay or absorption. In equation 1.16a, λ_m is the decay constant of isotope m and $\sigma_{m,r}(E)$ is the microscopic cross section of nuclear reaction r that generates isotope i. In equation 1.16b, L_{km} term is the transmutation probability of isotope k to m by decay and $Y_{km,r}$ represents the yield of fission product m from fissionable isotope k, and $\Phi(E, t)$ term is the neutron flux [26].

To calculate the burnup value of an individual material, MCNP calculates the

following equation,

$$Burnup = Burnup_{previous\ step} + \frac{Power\ Fraction \times Power\ Level \times PFRAC \times Time}{MTU} \quad (1.22)$$

Power fraction is fission power in a particular material divided by the fission power in the sum of all burned materials. PFRAC represents the power fraction applied to the reactor assembly at the specified time interval and its value can be changed with using PFRAC command in the input file according to the system to be simulated. Time is the reactor operation time interval in which burnup takes place. Time value also can be set in input file for the desired simulation process. MTU is the metric tonne unit. In this formula, MCNP takes the previous step term as 0 at the time of $t=0$, that is, at the beginning of the reactor operation, and calculates the burnup value in each step and adds it to it in the next step, and finds the burnup value for the specified time interval.

1.1. Pressurized Water Reactor

Pressurized Water Reactor (PWR) is a Western made pressurized and pool type reactor with thermal power of 1000 MW. Core and the assemblies are shaped as square and the core of the PWR can have total of 14×14 , 15×15 , 16×16 , 17×17 shaped assemblies which differs from reactor to reactor. Assembly of the reactor can be seen in section 2.2 figure 2.6. The chosen assembly shape for this study is 17×17 square and each assembly contains 264 fuel rods and control rods between 16-20. PWR uses UO_2 mixture as fuel element and the enrichment of ^{235}U can be up to 5 wt%. Each fuel element has cylindrical geometry and some of the fuel rods may be doped with Gd_2O_3 as burnable absorber to control the criticality inside the assembly. As the burnable absorber can be doped with the uranium fuel, it can also be used as coating material for the fuel rods. Some of the fuel rods have boron doped fuel rod coating which is also used to decrease the number of fission reactions inside the assembly to control the reactivity. The reactor core, where the fuels are located and the reactions take place, is located inside the structure called

18 guide channels and 1 instrumental channel. Fuel rods inside the assemblies can have different enrichment of Gd_2O_3 absorber which varies between 5 wt% and 9 wt%. Burnable absorber Gd_2O_3 is mixed homogeneously with UO_2 fuel and can be placed into selected positions inside the assemblies. In the VVER-1200 reactor, just as in the PWR, fuels and heated water are contained in the reactor pressure vessel. High pressure keeps the water in liquid form even when the core's temperature reaches roughly 600 K. The primary circuit, which is also seen in the picture below, is where the water is located. The water is heated by the energy generated as a result of fission. The tubes inside the steam generators receive heated water that has been transported from the primary circuit. But only about half of the water in the primary circuit is under pressure inside the structure known as the secondary unit. The steam generator also receives this water. Since the water in the secondary unit is less pressurized than the water in the primary unit, it absorbs the heat generated by the water coming from the primary unit inside the steam generator and evaporates. The resulting steam is transferred to the turbines and electricity is obtained. Diagram of the VVER-1200 can be seen in the figure 1.5.

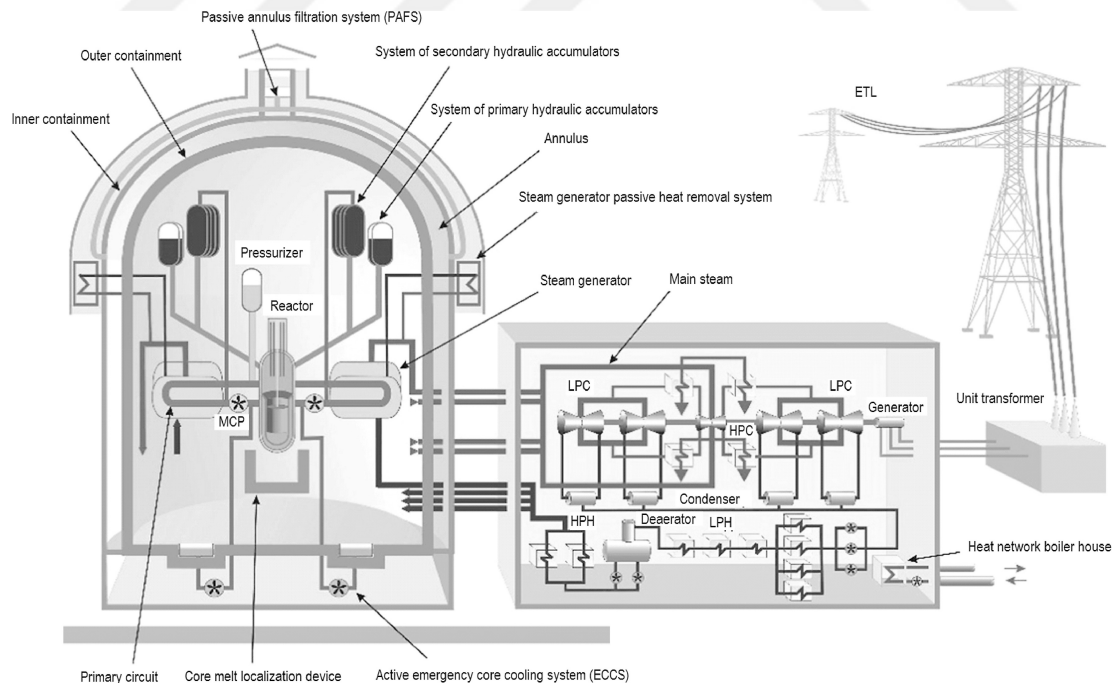


Figure 1.5 VVER-1200 Scheme [28]

1.3. TRIGA Mark II Research Reactor

TRIGA (Training, Research, Isotopes, General Atomics) reactors are one of the safest reactors in the world, built for training and experiments. No electricity is obtained from these reactors. The TRIGA Mark II reactor at Istanbul Technical University was built and started to operate in 1979 [29]. Its operating power is 250 kW. The reactor core is ring shaped and contains 69 cylindrical fuel elements. Fuel elements contain uranium zirconium hydride ($\text{U-ZrH}_{1.6}$) fuel enriched with 19.8 wt% in terms of ^{235}U . $\text{ZrH}_{1.6}$ is used as a moderator to slow the neutrons down in the system. Uranium concentration in the compound is 8.5 wt%. Fuel elements are cylindrical shaped and stainless steel is used as coating. In addition to the fuel elements in the reactor core, there are a total of 16 graphite rods in the outermost part of the core and these play the role of neutron reflector. The neutron hitting the graphite is reflected back to the reactor core and thus leakage is partially prevented. Since the fuel elements close to the graphite rods can undergo fission with the neutrons that collide with the graphite and reflected back to the core, it is expected that those fuel elements will burn more. The green areas in the core diagram represent the control rods. The control rods above the core are lowered to the green area as soon as the criticality of the reactor is wanted to be controlled. In the grey area, there is a neutron source that will initiate the fission chain reaction in the core. The rabbit system, shown in red, is the area that allows materials to be placed in the reactor core for various experiments, such as observing the effects of irradiation on the material. TRIGA Mark II diagram and fuel element are shown in the figure 1.6 and figure 1.7 respectively.

This study consists of two parts. The aim of the first part is to obtain the neutronic parameters of PWR and VVER-1200 assemblies and investigate the effect of standard, IBA, IFBA and hybrid rods to these neutronic parameters in the absence of control rods inside the fuel assembly. The assemblies of the PWR and VVER-1200 were modeled and simulated by using MCNP 6.1 code, operated under the same conditions and scenarios without control rods, and the neutronic

parameters of the reactors were obtained and compared. Uranium enrichment, IFBA rod coating thickness and number of IFBA rods, Gd_2O_3 enrichment of IBA rods and number of IBA rods and the operation time are same for both reactors. In the second part, the fuel element of the TRIGA Mark II reactor in Istanbul Technical University was modeled and simulated with the MCNP 6.1 program and the average burnup value for the fuel element of the TRIGA Mark II reactor was found by using the operation information obtained from the log book of the reactor.

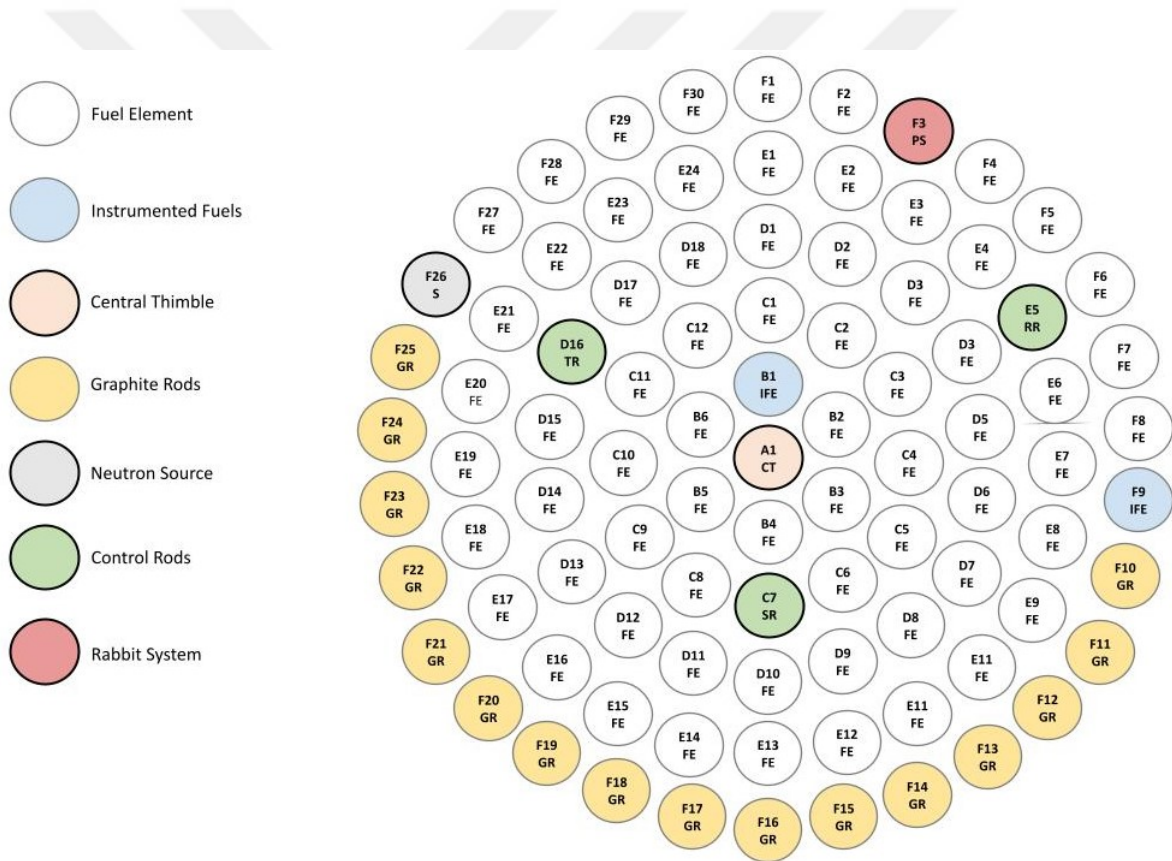


Figure 1.6 TRIGA Mark II Reactor Core Diagram

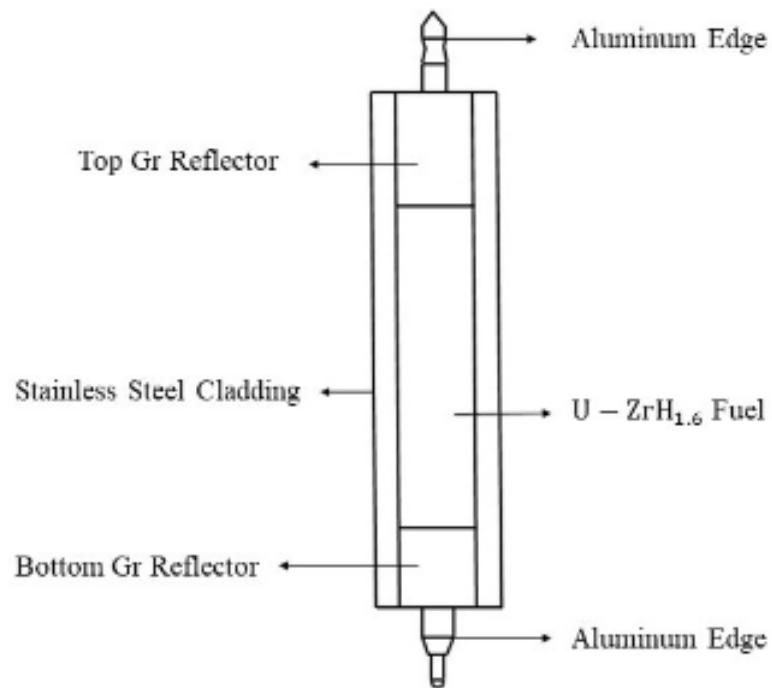


Figure 1.7 TRIGA Mark II Reactor Fuel Element

2. MATERIAL AND METHOD

2.1. Modeling the Pressurized Water Reactor Fuel Assembly

The pressurized water reactor has square shaped assembly. Square shaped assembly contains fuel rods embedded in water. The PWR assembly that studied in this work is 17×17 square shaped and it has 264 fuel rods and 25 empty rods filled with water. Since we want to obtain the neutronic parameters in the absence of control rods, the positions of the control rods in the assembly are filled with water. In this work, fuel material UO_2 has enrichment of 4.35 wt% ^{235}U . In scenarios where IBA rod is added to the fuel assembly, the Gd_2O_3 concentration in the IBA rods is chosen as 5 wt% and 9 wt%. The reason for choosing two different concentrations is to observe how increasing Gd_2O_3 enrichment effects reactor criticality. In scenarios where the assembly contains IFBA rods, the number of IFBA rods inside the assembly is 80. Since the aim of the work is to investigate and compare VVER-1200 and PWR neutronic parameters under same conditions, the dimensions of the PWR taken into account to calculate the ZrB_2 coating thickness. The calculated value of the ZrB_2 thickness was used for both reactors. The design parameters of the studied PWR fuel assembly can be seen in the table 2.1. Based on the parameters shown in Table 2.2, PWR fuel assembly was modeled and simulated for 4 different scenarios.

Scenario 1: The "Standard Fuel" scenario was implemented for PWR fuel assembly as it was for VVER-1200. In this scenario, reactor assembly has only fuel rods. No IBA or IFBA rods are used as neutron absorbers. The aim of this scenario is to investigate the evolution of the effective neutron multiplication factor with increasing temperature, burn time and burnup as well as the change in isotope concentrations which are important to understand the reactions taking place inside the assemblies. Investigations were made when there is no neutron absorber inside the assembly except the fuel material. Modeled assembly can be seen in figure 2.1.

Parameter	Value
Diameter of fuel pellet	0.819 cm
Inner diameter of fuel rod cladding	0.836 cm
Outer diameter of fuel rod cladding	0.95 cm
Fuel rod pitch	1.26 cm
Type of fuel	UO ₂ and UO ₂ -Gd ₂ O ₃
Gd ₂ O ₃ enrichment	5 wt% and 9 wt%
Number of IBA rods	8
Assembly Height	388.1 cm
Assembly Shape	17 × 17 Square
Number of fuel rods in assembly	264
Number of empty rods in assembly	25
Enrichment of fuel	4.35 wt % ²³⁵ U
Number of IFBA rods	80
IFBA coating thickness	0.0009410214 cm

Table 2.1 Design Parameters of PWR Fuel Assembly [30]

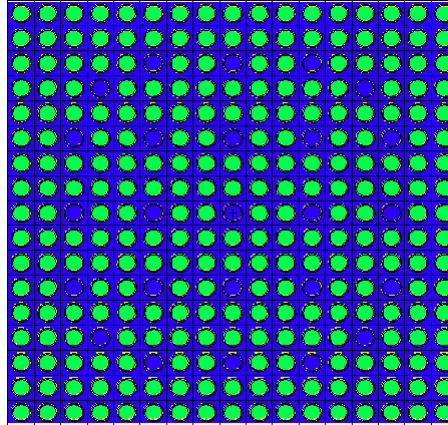


Figure 2.1 MCNP Representation of PWR Standard Fuel Assembly

Figure 2.1 represents the fuel assembly of PWR for standard fuel scenario. Green rods in the assembly are the fuel rods with the enrichment of 4.35 wt%. Blue colored areas and rods represents moderator material water. The number of fuel rods is 264 and water filled rods is 25.

Scenario 2: The "IFBA Fuel" scenario. In this scenario, 80 fuel rods inside the assembly have ZrB_2 coating. Coating thickness of the fuel rods is the same with VVER-1200. Modeled assembly can be seen in figure 2.2.

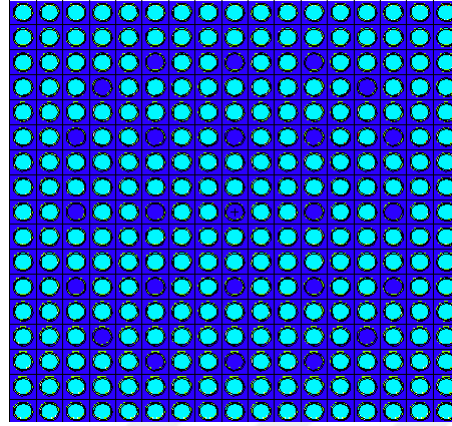


Figure 2.2 MCNP Representation of PWR IFBA Fuel Assembly

Figure 2.2 represents the PWR assembly for IFBA Fuel scenario. Aqua colored rods are the UO_2 fuel rods with the enrichment of 4.35 wt%, dark blue colored areas and rods are the moderator, water. In this model assembly has 264 fuel rods, 80 of which are coated with ZrB_2 and 25 water filled rods. Coating thickness of IFBA rods is 0.0009410214 cm. Cross section of coated fuel rods can be seen in figure 2.3.

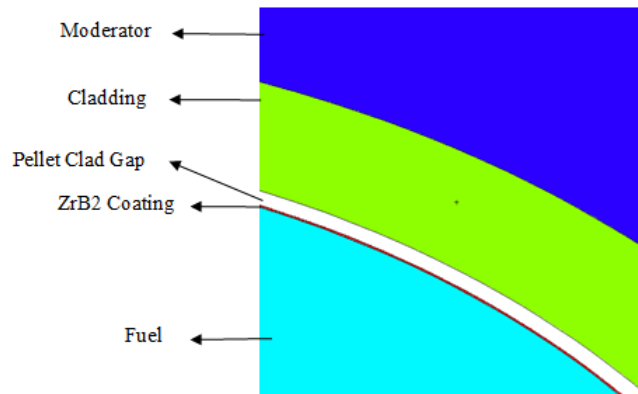


Figure 2.3 Cross Section of ZrB_2 Coated PWR Fuel Rod

It can be seen from figure 2.3 that the aqua colored area is the fuel material UO_2 , little area with red color is ZrB_2 coating of the fuel rod, white area is the gap between two cladding, green area is the zirconium cladding of the fuel rod, and the dark blue area is the moderator material water. Cladding calculations is shown in section 2.3

Scenario 3: The "IBA Fuel" scenario. In this scenario PWR assembly contains 8 IBA rods in addition to the fuel rods inside the assembly. Number of IBA rods and the concentration of Gd_2O_3 in $UO_2-Gd_2O_3$ mixture is the same as VVER-1200 model, 5 wt% and 9 wt%. Modelling and simulation process took place for these 2 different Gd_2O_3 concentrations, respectively. There is no IBA rods in this scenario. Modeled assembly can be seen in figure 2.4.

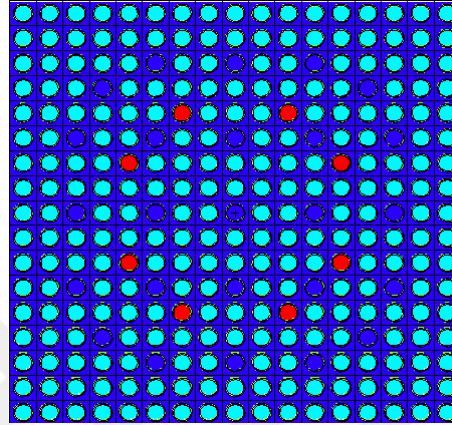


Figure 2.4 MCNP Representation of PWR IBA Fuel Assembly

In figure 2.4, red colored rods are IBA rods that is $UO_2-Gd_2O_3$ mixture, aqua colored rods are UO_2 fuel rods and dark blue colored areas are moderator, water. The positions of IBA rods inside the assembly can be seen in figure. Assembly has 8 IBA rods, $UO_2-Gd_2O_3$ mixture, and 258 UO_2 fuel rods. Assembly also has 25 water filled rods.

Scenario 4: The "Hybrid Fuel" scenario. In this scenario, scenario 2 and scenario 3 are combined. Therefore, the assembly of the PWR contains 8 IBA rods, 264 fuel rods 80 of which are coated with ZrB_2 material and 25 water filled rods. This scenario was also modeled and simulated for two different Gd_2O_3 concentration in IBA rods, 5 wt% and 9 wt% respectively. The modeled PWR assembly for this scenario can be seen in figure 2.5.

Aqua colored rods are the fuel rods with the enrichment of 4.35 wt%, dark blue colored areas and rods are the moderator material water, red colored rods are the $UO_2-Gd_2O_3$ material which are IBA rods. Although the assembly has 80 rods with ZrB_2 coating, it cannot be seen in the figure as another color because the coating

thickness of the rods is too small compared to the assembly size. Assembly has 8 IBA, 264 fuel rods 80 of which are coated with ZrB_2 material and 25 water filled rods.

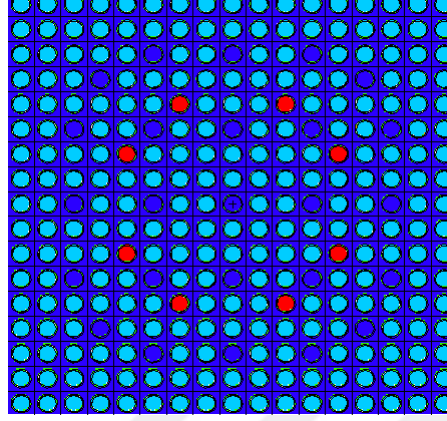


Figure 2.5 MCNP Representation of PWR Hybrid Fuel Assembly

All these scenarios for PWR were simulated for five different temperatures. The aim of simulating for different temperatures is to investigate the effect of increasing temperature on reactor criticality, fuel burnup, and material concentrations inside the PWR assembly. Selected temperatures are 300 K, 600 K, 900 K, 1200 K and 2500 K. ENDF/B-VII.1 library was used in MCNP input file to set the neutron cross sections of the used isotopes for different temperatures. KCODE were used to obtain the criticality and burnup calculations. 5000 neutrons, 300 active and 85 inactive cycles were used while simulating the assemblies under different temperatures as it was used for VVER-1200 simulation process. Assembly was operated for 1400 days and operation time divided to 21 different time steps with different time intervals. Total power applied to the assembly is different in each time interval. The power input of the assembly was chosen as 17.1 MW which is the average power per assembly for PWR [31]. Since the reactor contains more than 1 assembly and the power per assembly can differ in the reactor, we decided to chose the average power per assembly also for PWR because only 1 assembly was simulated for every scenario in this work. Operation time for both PWR and VVER-1200 assemblies and the applied power fractions can be seen in table 2.2. Although the given power fractions are same for both VVER-1200 and PWR in their input file, it should be remembered

that the given power value for their assemblies are different. Therefore, the operation power values for reactors will be different in each time step. As an example, at the time steps where the given power fraction is 1.0, VVER-1200 assembly operates with 19.415 MW and PWR assembly operates with 17.1 MW power. Reactor shut down period is also considered to see its effect on reactor criticality, fuel burnup and material concentrations.

Operation Time (Day)	Power Fraction
26.12855	1.0
26.12855	1.0
52.2571	1.0
104.5142	1.0
104.5142	1.0
104.5142	1.0
104.5142	1.0
30.0	0.0 (Down time)
25.23883	0.826
25.23883	0.826
50.47766	0.826
100.95532	0.826
100.95532	0.826
100.95532	0.826
100.95532	0.826
30.0	0.0 (Down Time)
26.0886	0.2814
26.0886	0.2814
52.1772	0.2814
104.3544	0.2814
104.3544	0.2814

Table 2.2 Given Power Fractions at Each Time Step for PWR and VVER-1200 Assemblies

2.2. Modeling the VVER-1200 Fuel Assembly

The VVER-1200 assembly has a hexagonal shape and contains fuel rods embedded in water. A typical VVER 1200 reactor has 312 fuel rods, 18 guide channels and 1 instrumented channel in the fuel assembly. In this study, there are 306 fuel rods in the VVER-1200 fuel assembly. The reason for this is that there are 25 control rods in the PWR and since we want to simulate the VVER-1200 and PWR assemblies under the same conditions, the number of control rods is chosen the same as the PWR for the VVER-1200. Since we want to obtain the neutronic parameters in the absence of the control rods, the positions of the control rods in the assembly are filled with water. There is no control rod in the assembly in any scenario. Also no guide channel or instrumental channel is considered. ^{235}U enrichment of a typical VVER-1200 can be up to 5 wt%. In this work inside the fuel rods there are UO_2 fuels with the enrichment of 4.35 wt% . In scenarios where IBA rod is added to the fuel assembly, the Gd_2O_3 concentration in the $\text{UO}_2\text{-Gd}_2\text{O}_3$ mixture is chosen as 5 wt% and 9 wt%. The reason for choosing two different concentrations is to observe how increasing Gd_2O_3 enrichment effects reactor criticality. In scenarios where the assembly contains IFBA rods, the number of IFBA rods inside the assembly is 80. Since the aim of the work is to investigate and compare the neutronic parameters of PWR and VVER-1200 assemblies under same conditions, the dimensions of the PWR taken into account to calculate the ZrB_2 coating thickness. The calculated value of the ZrB_2 thickness was used for both reactors. Design parameters of the typical VVER-1200 that are the height of the assembly, diameter of fuel pellet, inner and outer fuel rod cladding, fuel rod pitch were taken as reference while modeling the fuel assembly. The design parameters of the studied VVER-1200 fuel assembly can be seen in the table 2.3.

Parameter	Value
Diameter of fuel pellet	0.76 cm
Inner diameter of fuel rod cladding	0.773 cm
Outer diameter of fuel rod cladding	0.91 cm
Fuel rod pitch	1.275 cm
Type of fuel	UO ₂ and UO ₂ -Gd ₂ O ₃
Gd ₂ O ₃ enrichment	5 wt% and 9 wt%
Number of IBA rods	8
Assembly Height	373 cm
Assembly Shape	Hexagonal
Number of fuel rods in assembly	306
Number of empty rods in assembly	25
Enrichment of fuel	4.35 wt % ²³⁵ U
Number of IFBA rods	80
IFBA coating thickness	0.0009410214 cm

Table 2.3 Design Parameters of VVER-1200 Fuel Assembly [32]

Based on the parameters shown in Table 2.1, VVER-1200 fuel assembly was modeled and simulated for 4 different scenarios as follows.

Scenario 1: This scenario is called the "Standard Fuel" scenario. In this scenario, there is no IBA and IFBA rods inside the assembly. Assembly of the reactor contains 306 UO₂ fuel rods with 4.35 wt% enrichment and 25 empty rods filled with water. The aim of this scenario is to investigate the change in effective neutron multiplication factor with increasing temperature, burn time and burnup when there is no absorber material inside the reactor assemblies. Additionally, to find the burnup values of the PWR and VVER-1200 assemblies and the change in the concentrations of some specific isotopes which are important for understanding the reactions taking place inside both PWR and VVER-1200 assemblies. Modeled assembly by using MCNP 6.1 program for standard fuel scenario can be seen in the figure 2.6.

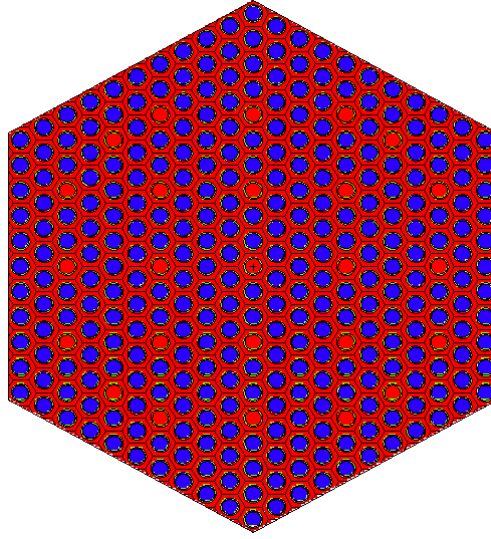


Figure 2.6 MCNP Representation of VVER-1200 Standard Fuel Assembly

Figure 2.6 represents the fuel assembly of VVER-1200 for the standard fuel scenario. Blue rods in the assembly are the fuel rods with enrichment of 4.35 wt%. Every blue color on the figure represents the fuel material UO_2 . There 306 fuel rods inside the assembly. Red color represents the moderator material, water. It can be seen that the fuel rods are immersed in water. Unlike the blue rods, red rods represent the water filled control rod positions. As it was assumed that there is no control rods in the assembly, instead it was filled with water.

Scenario 2: This scenario is called the "IFBA Fuel" scenario. In this scenario, there are fuel rods coated with ZrB_2 . The number of coated rods inside the assembly is 80. Number of fuel rods and the enrichment of the fuels are same with the standard fuel. The only difference between these 2 scenarios is in this scenario fuel rods in the assembly has 80 ZrB_2 coated fuel rods. The aim of this scenario is to investigate the change in effective neutron multiplication factor values with increasing assembly temperature, burn time and burnup in the presence of a neutron absorber material, ZrB_2 . Another aim is to find the burnup values of the PWR and VVER-1200 assemblies and the change in the concentrations of some specific isotopes which are important for understanding the reactions taking place inside both PWR and VVER-1200 assemblies. The modeled VVER-1200 assembly by using MCNP 6.1 program for the IFBA fuel scenario can be seen in the figure 2.7.

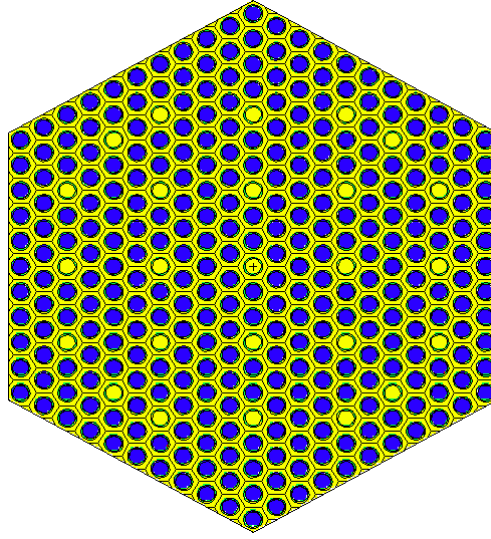


Figure 2.7 MCNP Representation of VVER-1200 IFBA Fuel Assembly

Figure 2.7 above represents the fuel assembly of VVER-1200 for IFBA fuel scenario. Blue rods in the assembly are the fuel rods with the enrichment of 4.35 wt%. Blue color on the figure represents the fuel material UO_2 as it is in standard fuel scenario. Yellow color represents the moderator material, water. The reason it is colored different than the standard fuel scenario is, in this scenario system has another material, ZrB_2 . MCNP visualizes every material with different color, so if there is a new material in the system material colors can change. Assembly has 306 fuel rods colored with blue, 25 water filled rods and 80 IFBA rods in this scenario. The coating thickness of the IFBA rods effects the value of the total mass of ^{10}B in the assembly, and this is a factor that directly effects the criticality conditions of the reactor assemblies. ZrB_2 coating thickness of 80 IFBA rods in the assembly was calculated according to the selected parameters. Calculation of IFBA rod ZrB_2 coating thickness can be seen in section 2.3 of the material method section. In order to simulate both reactor assemblies under the same conditions, the ZrB_2 coating thickness calculated according to the selected parameters has the same value in the IFBA rods in the reactor assemblies. The ZrB_2 coating thickness used in IFBA rods of the reactor assemblies was calculated as 0.000941021 cm. Cross section of coated fuel rod can be seen in figure 2.8.

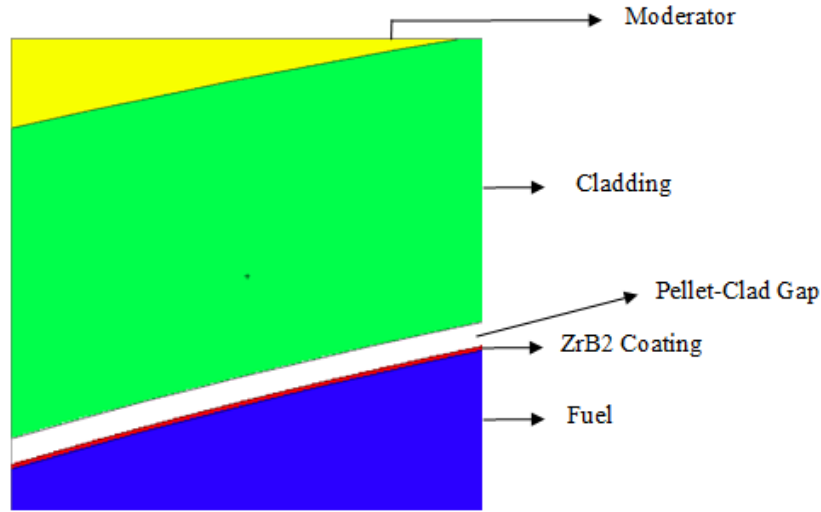


Figure 2.8 Cross Section of ZrB₂ Coated VVER-1200 Fuel Rod

Every material in the figure is colored different by MCNP 6.1 visualization tool. It can be seen that the blue colored area is the fuel material UO₂, and the area with red color is the ZrB₂ coating of the fuel rod. The reason why the red area is very small is because the coating thickness of the rod is too low. White area is the gap between two cladding and the green area is the zirconium cladding of the fuel rod as it is mentioned in introduction section. The yellow color is the moderator material water in the assembly.

Scenario 3: This scenario is called the "IBA Fuel" scenario. In this scenario, assembly has fuel UO₂ mixed with Gd₂O₃ material. The number of UO₂-Gd₂O₃ fuels inside the assembly is 8 and the concentrations of Gd₂O₃ inside the UO₂-Gd₂O₃ mixture are 5 wt% and 9 wt% respectively. Modelling and simulation process took place for two different concentrations. The enrichment of the fuel rods is the same with the first two scenarios. The aim of this scenario is to investigate the change in effective neutron multiplication factor with the increasing temperature, burn time and burnup with the presence and effect of IBA rods. Additionally, we want to find the burnup values of the PWR and VVER-1200 assemblies and the change in the concentrations of some specific isotopes which are important for understanding the reactions taking place inside both PWR and VVER-1200 assemblies. Modeled assembly for IBA fuel scenario can be seen in figure 2.9.

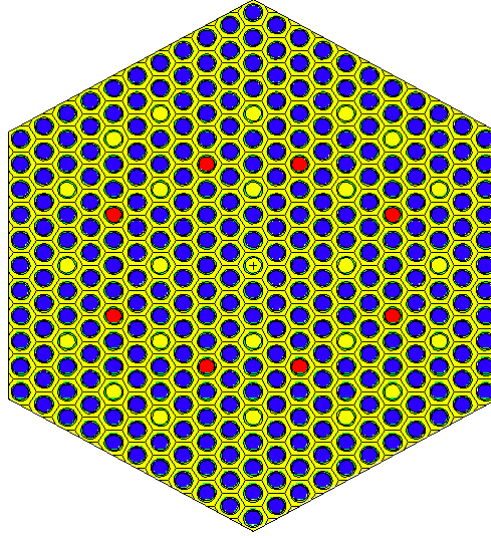


Figure 2.9 MCNP Representation of VVER-1200 IBA Fuel Assembly

In Figure 2.9, the blue-colored rods represent UO_2 fuel, the yellow-colored areas represent moderator water, and the red-colored rods represent IBA rods, which consist of a mixture of UO_2 and Gd_2O_3 . The positions of the IBA rods within the fuel assembly are illustrated in the figure. This particular assembly was subjected to modeling and simulation, considering two different concentrations of Gd_2O_3 . The assembly has 8 IBA rods, 25 water-filled rods, and 298 fuel rods.

Scenario 4: This scenario is called "Hybrid Fuel". In this scenario, assembly has both IBA and IFBA rods as neutron absorber materials. This scenario is the combination of scenario 2 and scenario 3. The aim of this scenario is to investigate evolution of the effective neutron multiplication factor with the increasing temperature, burn time and burnup with the presence of both IBA and IFBA rods inside the assembly. Additionally, to find the burnup values of the PWR and VVER-1200 assemblies and the change in the concentrations of some specific isotopes which are important for understanding the reactions taking place inside both PWR and VVER-1200 assemblies. Fuel enrichment value was chosen as the same with first 3 scenarios, 4.35 wt%. Concentration of Gd_2O_3 inside the IBA rods are the same with scenario 3, 5 wt% and 9 wt% respectively. This scenario was also modeled and simulated for these two different Gd_2O_3 concentrations. Modeled assembly for hybrid fuel scenario of the VVER-1200 can be seen in figure 2.10.

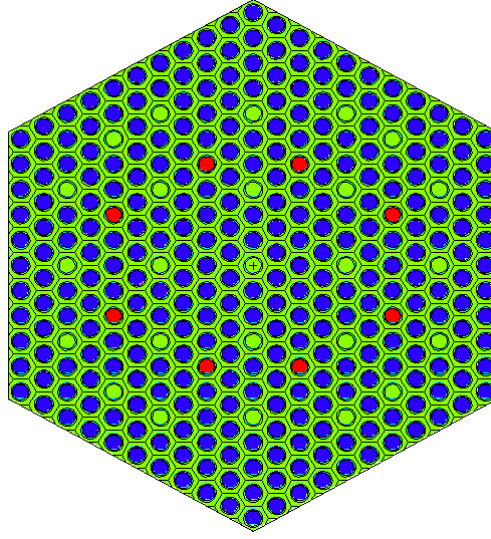


Figure 2.10 MCNP Representation of VVER-1200 Hybrid Fuel Assembly

In figure 2.10 blue colored rods are UO_2 fuel, green colored areas and rods are moderator water, red rods are IBA rods and there are 80 IFBA rods inside the assembly. Position of IFBA rods cannot be seen because the thickness of the ZrB_2 coating is too small compared to assembly size. Assembly in this scenario has 8 IBA rods, 25 water filled rods and 298 fuel rods, 80 of which are coated with ZrB_2 (IBA rods).

All these scenarios for VVER-1200 assembly were simulated for 5 different temperatures. The aim of simulating for different temperatures is to investigate the effect of increasing temperature on reactor criticality, fuel burnup, and material concentrations inside the assembly. Selected temperatures are 300 K, 600 K, 900 K, 1200 K and 2500 K. ENDF/B-VII.1 library was used in MCNP input file to set the neutron cross sections of the used isotopes for the different temperatures. KCODE were used to obtain the criticality and burnup calculations. 5000 neutrons, 300 active and 85 inactive cycles were used while simulating the assemblies under different temperatures. Assembly was operated for 1400 days and operation time divided to 21 different time steps with different time intervals. These time intervals also contain the reactor shut down period to see the effect of shut down time on the reactor criticality and burnup. Total power applied to the assembly is different in each time interval. The power input of the assembly was chosen as 19.415 MW which is the

average power per assembly for VVER-1200. Since the reactor contains more than 1 assembly and the power per assembly can differ in the reactor, we decided to choose the average power per assembly because only 1 assembly was simulated for every scenario in this work.

2.3. Calculation of IFBA Rod ZrB₂ Coating Thickness

Coating thickness of IFBA rod depends on the enrichment of boron isotopes inside the mixture, ¹⁰B and ¹¹B. Since the aim in this study is to analyze the PWR and VVER-1200 assemblies under same conditions, the dimensions of only one reactor were considered while calculating the coating thickness and the obtained thickness value was used for both reactors. Dimensions of the PWR assembly were used as a reference in the coating calculation and the result obtained was also used in the VVER-1200 fuel rod coating. All the isotopes of zirconium and boron were taken into account in order to calculate the coating thickness. In this study, ¹⁰B enrichment was chosen as 48 wt% and ¹¹B as 52 wt%. In order to find the coating thickness, the first thing to do is to find the amount of the isotopes of boron and zirconium in the ZrB₂ compound. The first thing to do is to find the molar masses of zirconium and diboride (B₂) then to find the molar mass of ZrB₂ mixture by making following calculations. Molar mass of boron is given by:

$$M_B = (M_{10B} \times 0.48 + M_{11B} \times 0.52) \quad (2.1)$$

Molar masses of ¹⁰B and ¹¹B are 10 g/mol and 11 g/mol respectively. Inserting them into equation 2.1 we find the molar mass of Boron as:

$$M_B = 10.52 \text{ g/mol} \quad (2.2)$$

Molar mass of Zr can be found as following:

$$M_{Zr} = (M_{90Zr} \times 0.5145 + M_{91Zr} \times 0.1122 + M_{92Zr} \times 0.1715 + M_{94Zr} \times 0.1738 + M_{96Zr} \times 0.028) \quad (2.3)$$

Molar masses of ^{90}Zr , ^{91}Zr , ^{92}Zr , ^{94}Zr and ^{96}Zr are 90, 91, 92, 94 and 96 respectively. The multiplicative factors (0.5145, 0.122, 0.1715, 0.1738 and 0.028 values) in the equation are natural abundance of zirconium isotopes in nature in terms of mole fraction. Inserting molar masses into equation the result becomes:

$$M_{\text{Zr}} = 91.3184 \text{ g/mol.} \quad (2.4)$$

Finding the molar masses of boron and zirconium led us to find the molar mass of ZrB_2 compound, which can be calculated as:

$$M_{\text{ZrB}_2} = M_{\text{Zr}} + 2M_{\text{B}}. \quad (2.5)$$

Inserting the molar mass values of zirconium and boron into the equation then the result is:

$$M_{\text{ZrB}_2} = 112.3564 \text{ g/mol.} \quad (2.6)$$

Now one can find the percentages of boron and zirconium isotopes in the ZrB_2 compound. Percentages of ^{10}B and ^{11}B in ZrB_2 can be calculated from following equation:

$$\%^{10}\text{B} = 0.48 \times \frac{2 \times 10}{112.3564} = 0.08544087491 \quad (2.7)$$

0.48, 10 and 112.3564 are mole fraction of isotope, molar mass of isotope and molar mass of ZrB_2 compound respectively. Since there are 2 boron atoms in ZrB_2 compound, it is multiplied by 2. Percentage of ^{11}B and zirconium isotopes in ZrB_2 can also be calculated with the same relation 2.7 by inserting the mole fraction and molar mass of ^{11}B and zirconium isotopes, then the results are:

$$\begin{aligned} \%^{11}\text{B} &= 0.1018170426, \\ \%^{90}\text{Zr} &= 0.4121187201, \\ \%^{91}\text{Zr} &= 0.09087171053, \\ \%^{92}\text{Zr} &= 0.140425638, \\ \%^{94}\text{Zr} &= 0.1454025689, \\ \%^{96}\text{Zr} &= 0.02392344498. \end{aligned} \quad (2.8)$$

In this study, it was considered that there are 80 IFBA rods in the reactor assembly. The height of the rods in the assembly is 388.1 cm (152.7952754 inches) and the diameter of a fuel pallet is 0.819 cm. Based on this information, it can be said that the surface area of a rod in the assembly is 998.5674772 cm². If it is desired to find the coating thickness using the isotope percentages and reactor parameters above, the mass of ZrB₂ per IFBA rod must first be found. To find this, first the ¹⁰B mass per IFBA rod will be found, and using this information, ZrB₂ per ifba rod will be found. Based on our decision, we determined that the mass of ¹⁰B per inch on an individual IFBA rod should be 0.0032 grams, and we performed our calculations accordingly. Based on the calculations conducted, ¹⁰B/inch/IFBA rod for VVER-1200 was determined as 0.0029 grams. ¹⁰B/inch/IFBA rod is 0.0032 g, multiplying this value with the rod height gives the ¹⁰B mass per IFBA rod, which can be calculated as:

$$\frac{M_{^{10}B}}{IFBA\ Rod} = 0.0032 \times 152.7952754 = 0.4889448814\ g \quad (2.9)$$

Dividing the result with ¹⁰B mass/ZrB₂ mass will give the ZrB₂ mass per IFBA rod. Result of the ¹⁰B mass/ZrB₂ mass have been found as 0.08544087491, dividing result with this value gives:

$$\frac{M_{ZrB_2}}{IFBA\ Rod} = \frac{0.4889448814}{0.08544087491} = 5.722610892\ g \quad (2.10)$$

Density of ZrB₂ compound is 6.09 g/cc. Coating thickness can be found as following equation:

$$Coating\ Thickness = \frac{\frac{Mass\ ZrB_2\ per\ IFBA}{Density\ of\ ZrB_2}}{Surface\ Area\ Per\ Rod} \quad (2.11)$$

Inserting the values inside the equation gives:

$$Coating\ Thickness = \frac{\frac{5.722610892}{6.09}}{998.5674772} = 0.0009410214\ cm \quad (2.12)$$

Coating thickness of IFBA rod is 0.0009410214 cm. As can be seen, the coating thickness is directly dependent on the enrichment value of ¹⁰B that we choose. The result found was used as the coating thickness of IFBA rods in both reactors.

2.4. Modeling the TRIGA Mark II Fuel Assembly

TRIGA Mark II reactor has a ring shaped core, with 69 fuel elements inside of it. Each fuel element has cylindrical shape, and consist fuel material uranium zirconium hydride ($\text{U-ZrH}_{1.6}$) with the enrichment of 19.9 wt% in terms of ^{235}U . Uranium concentration in ($\text{U-ZrH}_{1.6}$) is 8.5 wt%, concentration of zirconium is 89.9 wt% and the concentration of hydrogen in fuel material is 1.6 wt%. This study focuses on modeling a single fuel element within the reactor core, rather than the entire core itself. The aim of the study is to find the average burnup value for fuel element of the reactor by using the operation information in reactor log-book. Fuel elements of the reactor has stainless steel as cladding material. The dimensions and the representations of the modeled reactor fuel element can be seen in table 2.4, figure 2.11 and figure 2.12, respectively.

Fuel Element Parameter	Value
Length	38.10 cm
Cladding thickness	0.0508 cm
Radius	1.8161 cm
Radius of zirconium rod	0.3175 cm
Fuel enrichment	19.8 wt%
Axial bottom reflector length	8.8138 cm
Axial top reflector length	8.7376 cm
Total height	73.152 cm

Table 2.4 TRIGA Mark II Reactor Fuel Element Dimensions [29]

In figure 2.11, red areas at the top and bottom of the fuel element represent the graphite element used as reflector. Yellow area surrounding the fuel element is the stainless steel cladding. Green area inside the element is the fuel material ($\text{U-ZrH}_{1.6}$) and the blue area is the zirconium rod. A three-dimensional representation of the fuel element is shown in Figure 2.12. In this model, although the materials are colored differently from 2D, the blue and platinum colors represent the germanium

reflector, the purple area represents the fuel material, and the green area inside the element represents the zirconium rod.

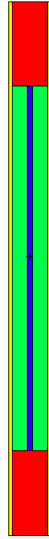


Figure 2.11 Fuel Element 2D MCNP Representation

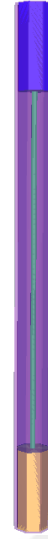


Figure 2.12 Fuel Element 3D MCNP Representation

The modeling and simulation process was carried out on the scenario of missing information in the reactor log book. It has been considered that we have information about how much energy was obtained in total from 1979 when the reactor started to operate until November 2017 when it was completely shut down, information such as how long the reactor was operated and how much power was applied to the reactor during these time intervals, the time interval when it was shut down is considered as not available. According to the information we have, the total energy obtained from the reactor from 1979 to November 2017 is 301522.934 kWh. Since we do not have the mentioned information and we cannot use that information in the input file while the modeling and simulation process is taking place, the following approach has been made on the energy obtained. It has been assumed that the reactor works without any shutdown, the total power applied to the reactor does not change during the operation period and it is homogeneously distributed on the fuel elements. Based on the available data, it was calculated how long the fuel element burned inside the reactor without ever shutting the reactor down. It was thought that the reactor operated at full power 250 kW without shut down during its operation time, then dividing the total amount of energy obtained by the maximum

operation power value gives how many hours the reactor needs to operate in order to obtain this energy. By dividing the parameters one can find,

$$t = \frac{301522.934 \text{ kWh}}{250 \text{ kW}} \approx 1206 \text{ hours} = 50.25 \text{ days}. \quad (2.13)$$

This value was used as reactor operation time in the MCNP input file. Time value should be converted from hour unit to days unit for the input file. After having the operation time, then the reactor operation power value should be set to input file. Since the aim is to find the average burnup for a fuel element of the reactor, the average power per 1 fuel element was used as power value for simulation process. For a TRIGA Mark II reactor operated under full power (250 kW), as it has been considered in this thesis, the maximum power value expected for one fuel element is 6.5 kW [33]. By using the reactor operation time value and the maximum power value for one fuel element, reactor fuel element was burned at 6.5 kW power for 50 days, and as a result the burnup value of the fuel was obtained.

In order to find the burnup value of the fuel element, the modeled environment was set up exactly the same with the experimental environment. The lead cask used in the experimental methods was modeled exactly same with MCNP and the fuel element was placed in the lead cask and the burnup value was determined. The experimental method takes place as follows. First, the burned fuel is removed from the reactor core. Then the burned fuel is placed in the cylinder cavity inside the lead cask. The fuel element placed inside the lead cask is moved up and down with the help of a mechanism to find out how much certain points of the fuel element have burned. Burnup determination is carried out with a detector placed in front of the lead cask. With the help of the detector placed in front of one of the holes placed at an angle of 120 degrees inside the lead cask, the isotopes released as a result of fission are tracked and how much the fuel element has burned through the spectrum is obtained. Since the reactor core was not modeled in this study, the fuel element was not burned inside the reactor, but was burned outside the reactor in accordance with the parameters inside the reactor. Then, the burning rate of the fuel element was found by placing it in lead cask. Modeled lead cask and fuel element by using

MCNP6.1 can be seen in the figure 2.13 and 2.14.

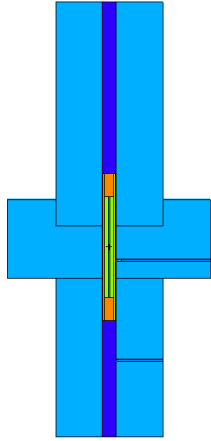


Figure 2.13 Fuel Element in Lead Cask
2D Representation

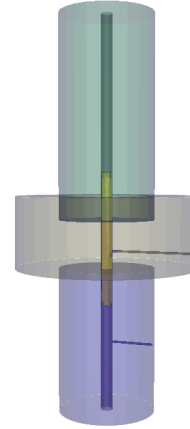


Figure 2.14 Fuel Element in Lead Cask
3D Representation

The aqua colored structure seen in Figure 2.13 is the lead cask. The modeled fuel element shown in Figure 2.11 can be seen inside of the lead cask. The area represented by dark blue between the lead cask and the fuel element is filled with air. The dark blue hole that extends from the fuel element to the edge of the lead cask is the opening mentioned in the paragraph above, which allows measurement by placing a detector in front of it.

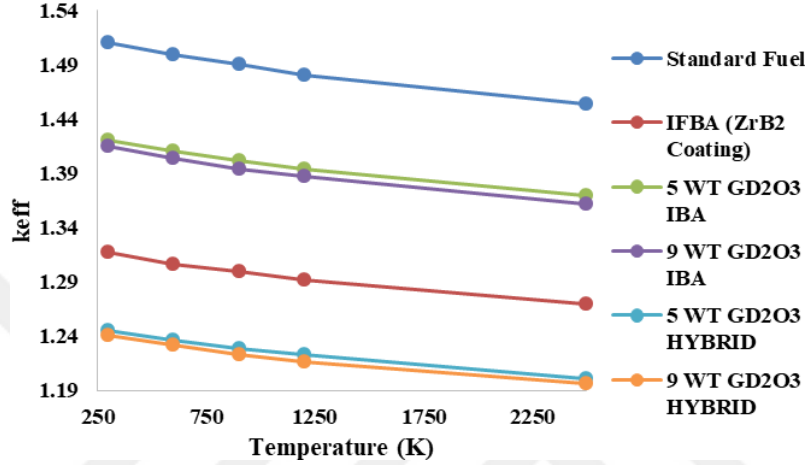
3. RESULTS

3.1. Effective Neutron Multiplication Factor and System Temperature

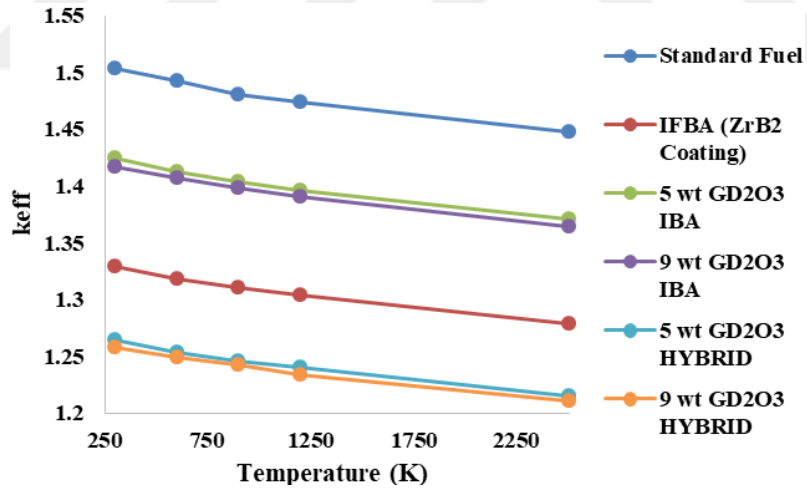
In this study, the effect of increasing temperature in the fuel assembly on the effective neutron multiplication factor, which is one of the most important reactor neutronic parameters, was observed. The energy released as a result of the fission reactions taking place in the fuel assembly can increase the system temperature. For this reason, we decided to examine how reactor criticality changes with increasing temperature.

Effect of increasing temperature on the reactor criticality for both PWR and VVER-1200 can be seen in figure 3.1. Each scenario is simulated for 300 K, 600 K, 900 K, 1200 K and 2500 K assembly temperature. As can be seen from the graphs 3.1(a) and 3.1(b), it is observed that the reactor criticality decreases as the system temperature increases in both PWR and VVER-1200 reactors. Increasing the temperature in the fuel assembly causes the high-energy neutrons released as a result of fission to gain even more kinetic energy. The higher the average kinetic energy of neutrons, the harder it is to reach thermal energy after interacting with water. Therefore, the thermal neutron population in the fuel assembly decreases as the temperature increases, resulting in lower probability of fission in the fuel assembly and reactor criticality starting to decrease as the temperature increases. Another reason for the decrease in reactor criticality with an increase in temperature can be explained as follows. The moderator water, which is used to slow down the high-energy neutrons released as a result of fission, is in liquid form in the assembly. As the temperature of the system increases, the liquid water becomes less dense. For this reason, the distance between the H_2O molecules in the water material increases, and this increase reduces the possibility of interaction between the neutrons released as a result of fission and the water molecules. As the probability of interaction decreases, the

high-energetic neutrons released are less likely to reach the energy level required to interact with ^{235}U , namely thermal energy. Temperature causes a decrease in moderator efficiency, causing a negative change in criticality. This effect of temperature on the moderator can also be observed on the fuel material. When the temperature in the fuel assembly starts to increase to very high values, it may reduce the density of the fuel material. This decrease in density also causes a decrease in reactivity.



(a)



(b)

Figure 3.1 (a) Effective multiplication factor-Temperature graph of every scenario for PWR (b) Effective multiplication factor-Temperature graph of every scenario for VVER-1200

Another result seen in the graph 3.1 is that the reactor criticality changes in each scenario. For both reactor types, the scenario with the highest criticality is the standard fuel scenario, followed by the other scenarios of 5 wt% IBA, 9 wt% IBA,

IFBA, 5 wt% hybrid and 9 wt% hybrid, respectively. The reason why the standard fuel scenario is the scenario with the highest criticality is that there is no neutron absorber in the fuel assembly compared to other scenarios. As there is no factor in the ensemble to reduce the thermal neutron population by neutron absorption, the highest criticality level was reached in standard fuel scenario in every temperature.

Neutron-absorbing materials Gd_2O_3 and ZrB_2 used in other scenarios absorb thermal neutrons in the assembly, causing the thermal neutron population to decrease, and consequently reducing the criticality level of the reactor. ^{157}Gd and ^{10}B isotopes in these materials are isotopes that act as neutron absorbers. Thermal neutron absorption cross sections of ^{157}Gd and ^{10}B isotopes are 252912 barns [34] and 3844 barns [34] respectively. Among the other Gd isotopes, except ^{157}Gd , the isotope with the highest thermal neutron cross section is ^{155}Gd and its cross section value is 60737 barns [31]. The other isotope of boron, ^{11}B , has a cross section of 0.005 barns [31] and its effect can be neglected because it has very low thermal neutron absorption cross section. Although the thermal absorption cross section of ^{157}Gd is much higher than that of ^{10}B , the criticality level in the IFBA scenario is lower than the criticality level in the IBA scenarios, as can be seen from the graphs. This is because the Gd enrichment ratio in the IBA rods is 5 wt% and 9 wt%, while the ^{10}B enrichment is 48 wt%. In addition, while the number of IBA rods in the assembly is 8, the number of IFBA rods is 80. Therefore, the neutron absorbing material in the IFBA rod scenario negatively affects the reactivity by absorbing more thermal neutrons than the neutron absorbing material in the IBA rod scenario. ^{157}Gd should not have high enrichment ratio in the assembly due to its high thermal neutron absorption cross section. Considering that gadolinium will have the same enrichment ratio and rod number as boron, its interaction with thermal neutrons greatly reduces the neutron population and brings the criticality level in the system below 1, that is, to the subcritical state. In such a case, it is not possible to obtain a self-sustaining and continuous fission reaction in the reactor. For this reason, if ^{157}Gd is to be used as a neutron absorber in the reactor, it should be used at low enrichment rates and the number of rods in the assembly should be kept as low as possible. In this study,

instead of using the gadolinium enrichment ratio from 1 wt% to 9 wt%, the scenarios were modeled by choosing the minimum and maximum values in the literature and the results were obtained accordingly.

In the hybrid scenarios, the effective multiplication factor has the lowest values because both IBA and IFBA rods are included in the assembly. The simultaneous use of two different neutron absorbers in the assembly and the increase in the total absorber concentration cause the thermal neutron population to decrease further compared to other scenarios. For this reason, criticality levels have the lowest values. Hybrid scenarios can be compared within themselves as follows. The only difference between the two hybrid scenarios is the Gd_2O_3 concentration in the IBA rods. The number of IBA and IFBA rods, ^{10}B enrichment of the IFBA rod is the same in both scenarios. Since Gd_2O_3 absorber material and UO_2 fuels material is found in the rod in mixture form, an increase in the Gd_2O_3 concentration means an increase in the amount of neutron absorber and a decrease in the UO_2 concentration in the system, that is, a decrease in fissionable material inside the assembly. For this reason, with the increasing Gd_2O_3 concentration, the probability of absorption of thermal neutrons increases, and the probability of fission in the system decreases as the number of fissionable nuclei will decrease. As a result, the 9 wt% hybrid scenario, which is the scenario with the highest Gd_2O_3 concentration, appears as the scenario with the lowest criticality compared to all other scenarios.

In this study, in addition to analyzing the variation of the criticality values of the reactors with increasing temperature in different scenarios for both reactors separately, the criticality values of the two reactors in the same scenario were also compared. Table 3.1, 3.2, 3.3, 3.4, 3.5 and 3.6 shows the deviation of effective neutron multiplication factor values for both PWR and VVER-1200 assemblies with simulated scenarios and temperatures. The graphs in Figure 3.2 show the comparison of criticality values of PWR and VVER-1200 reactors with increasing temperature in the same scenarios.

	PWR	VVER-1200
Temperature	k_{eff}	k_{eff}
300 K	1.50965 ± 0.00056	1.50337 ± 0.00055
600 K	1.49862 ± 0.00052	1.49234 ± 0.00053
900 K	1.48966 ± 0.00051	1.48103 ± 0.00052
1200 K	1.47987 ± 0.00054	1.47377 ± 0.00058
2500 K	1.45381 ± 0.00057	1.44736 ± 0.00052

Table 3.1 Deviation in Effective Neutron Multiplication Factor Values of PWR and VVER-1200 Assemblies for Standart Fuel Scenario with Simulated Temperatures

	PWR	VVER-1200
Temperature	k_{eff}	k_{eff}
300 K	1.31671 ± 0.00062	1.32940 ± 0.00064
600 K	1.30629 ± 0.00068	1.31811 ± 0.00069
900 K	1.29926 ± 0.00070	1.31027 ± 0.00060
1200 K	1.29183 ± 0.00063	1.30354 ± 0.00066
2500 K	1.26984 ± 0.00063	1.27908 ± 0.00063

Table 3.2 Deviation in Effective Neutron Multiplication Factor Values of PWR and VVER-1200 Assemblies for IFBA Fuel Scenario with Simulated Temperatures

	PWR	VVER-1200
Temperature	k_{eff}	k_{eff}
300 K	1.42061 ± 0.00062	1.42460 ± 0.00061
600 K	1.41013 ± 0.00058	1.41290 ± 0.00059
900 K	1.40146 ± 0.00059	1.40349 ± 0.00059
1200 K	1.39305 ± 0.00064	1.39600 ± 0.00063
2500 K	1.36876 ± 0.00063	1.37050 ± 0.00061

Table 3.3 Deviation in Effective Neutron Multiplication Factor Values of PWR and VVER-1200 Assemblies for 5 wt% Gd₂O₃ IBA Fuel Scenario with Simulated Temperatures

	PWR	VVER-1200
Temperature	k_{eff}	k_{eff}
300 K	1.41465 ± 0.00057	1.41751 ± 0.00054
600 K	1.40396 ± 0.00059	1.40686 ± 0.00061
900 K	1.39395 ± 0.00063	1.39837 ± 0.00064
1200 K	1.38701 ± 0.00061	1.39070 ± 0.00060
2500 K	1.36191 ± 0.00063	1.36446 ± 0.00061

Table 3.4 Deviation in Effective Neutron Multiplication Factor Values of PWR and VVER-1200 Assemblies for 9 wt% Gd_2O_3 IBA Fuel Scenario with Simulated Temperatures

	PWR	VVER-1200
Temperature	k_{eff}	k_{eff}
300 K	1.24538 ± 0.00064	1.26493 ± 0.00064
600 K	1.23633 ± 0.00067	1.25388 ± 0.00070
900 K	1.22779 ± 0.00068	1.24589 ± 0.00065
1200 K	1.22272 ± 0.00064	1.24014 ± 0.00068
2500 K	1.20075 ± 0.00067	1.21534 ± 0.00071

Table 3.5 Deviation in Effective Neutron Multiplication Factor Values of PWR and VVER-1200 Assemblies for 5 wt% Gd_2O_3 Hybrid Fuel Scenario with Simulated Temperatures

	PWR	VVER-1200
Temperature	k_{eff}	k_{eff}
300 K	1.24089 ± 0.00064	1.25793 ± 0.00064
600 K	1.23172 ± 0.00069	1.24889 ± 0.00066
900 K	1.22281 ± 0.00064	1.24203 ± 0.00068
1200 K	1.21582 ± 0.00061	1.23407 ± 0.00066
2500 K	1.19574 ± 0.00061	1.21057 ± 0.00066

Table 3.6 Deviation in Effective Neutron Multiplication Factor Values of PWR and VVER-1200 Assemblies for 9 wt% Gd_2O_3 Hybrid Fuel Scenario with Simulated Temperatures

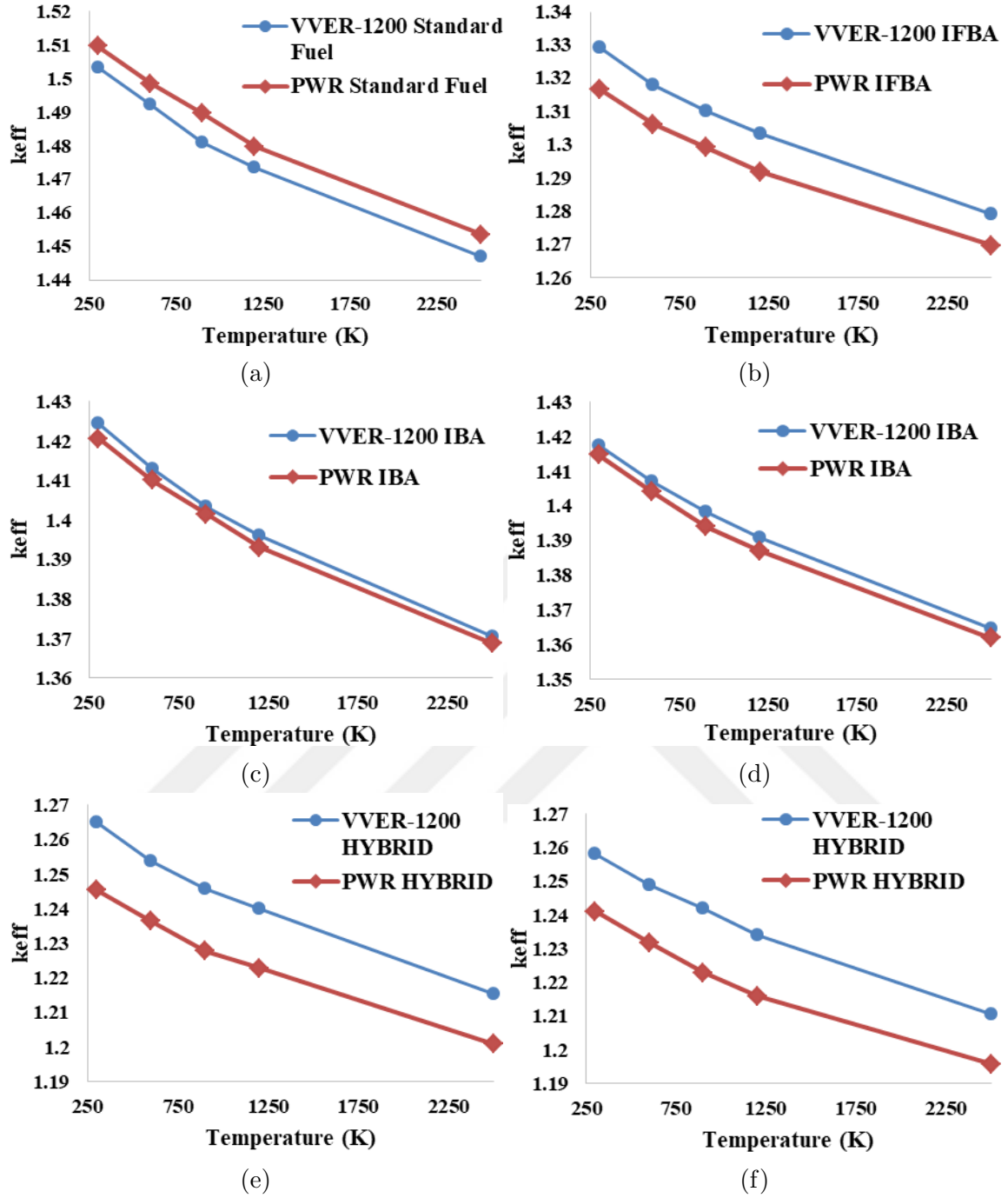


Figure 3.2 (a) Comparison of effective neutron multiplication factor with increasing temperature for PWR and VVER-1200 standard fuel scenarios (b) Comparison of effective neutron multiplication factor with increasing temperature for PWR and VVER-1200 IFBA fuel scenarios (c) Comparison of effective neutron multiplication factor with increasing temperature for PWR and VVER-1200 IBA fuel scenarios with 5 wt% Gd_2O_3 (d) Comparison of effective neutron multiplication factor with increasing temperature for PWR and VVER-1200 IBA fuel scenarios with 9 wt% Gd_2O_3 (e) Comparison of effective neutron multiplication factor with increasing temperature for PWR and VVER-1200 hybrid fuel scenarios with 5 wt% Gd_2O_3 (f) Comparison of effective neutron multiplication factor with increasing temperature for PWR and VVER-1200 hybrid fuel scenario with 9 wt% Gd_2O_3

Figure 3.2 (a) illustrates the comparison of effective neutron multiplication factors for different assembly temperature under standard fuel scenario. As can be seen in figure, the criticality value of PWR in the standard fuel scenario is higher than VVER-1200 at all temperatures. When the assembly temperature is 300 K, the criticality value of PWR at the start of operation is about 1.51 while VVER-1200 criticality value is around 1.50. The reason for such a situation to occur in the standard fuel scenario is that the critical mass in the PWR fuel assembly, namely the total concentration of ^{235}U , is higher than the VVER-1200. To advance the interpretation, the change of ^{235}U concentration with burn time at 300 K temperature for both reactor assemblies is shown in the figure. As can be seen from the figure 3.3, the total concentration of ^{235}U in the PWR assembly is greater than VVER-1200 assembly. The other notable thing is that the concentration of ^{235}U decreases as the burn time increases in both assemblies. The reduction of the concentration of ^{235}U with burn time will be carried out in the next sections for different scenarios and temperature. According to the modeled assemblies, total initial concentration of ^{235}U in the PWR assembly is 25690 g, while the initial amount of ^{235}U in the VVER-1200 assembly is 24720 g. Although the number of fuel rods in the VVER-1200 assembly is greater than that of the PWR, the diameter and assembly height of the fuel rods in the PWR assembly are greater than in the VVER-1200. This shows that the total fuel volume in the PWR assembly is greater than in the VVER-1200 assembly. The fact that the total volume of the fuel rods in which the fuels with the same enrichment and fuel composition are placed is greater in PWR indicates that it has a higher critical mass than VVER-1200. Higher critical mass also causes the PWR to have more fissionable material at the beginning of the operation, which leads to more fission reactions within the assembly. As it will be remembered from the equation 1.3, more fission reaction in each generation means more criticality value, since the effective neutron multiplication factor, that is, the reactor criticality, is proportional to the fission reactions that take place in each step.

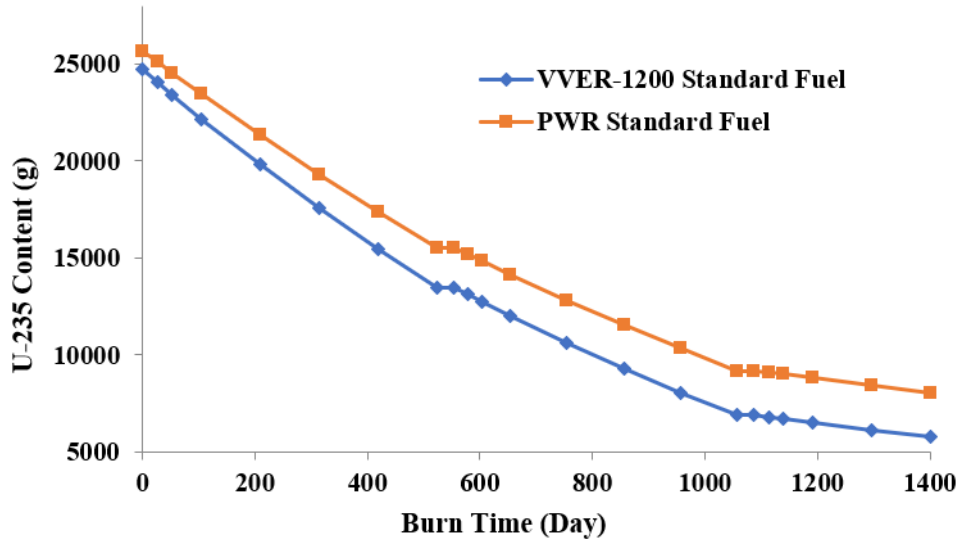


Figure 3.3 Change in ^{235}U concentration with increasing burn time for PWR and VVER-1200 assemblies standard fuel scenario at 300 K temperature

Effective multiplication factors of PWR and VVER-1200 with increasing temperature under IFBA fuel scenario can be seen in figure 3.2 (b). In this scenario, the criticality values of both reactors decrease due to the neutron absorber material ZrB_2 used in the assembly. If the criticality values at 300 K temperature are compared with the standard fuel scenario, the criticality value of PWR is decreased from standard fuel scenario value 1.51 to approximately 1.32, and VVER-1200 criticality value from 1.5 to approximately 1.33. It is an expected result that the criticality values will decrease due to the neutron absorber material in the assembly. However, as can be seen from the figure, PWR criticality is lower than VVER-1200 in this scenario. If we look at the ZrB_2 coatings of 80 IFBA rods used in the assemblies in this scenario by its volume, it is seen that the volume of the ZrB_2 coating in the PWR is greater because the fuel rod diameter and assembly height of the PWR are larger than the VVER-1200. For this reason, concentration of ^{10}B isotope in the coating material is higher in the PWR assembly than in VVER. In order to take the interpretation further, the change in ^{10}B concentrations with respect to burn time or assembly operation time at 300 K temperature for PWR and VVER-1200 assemblies in IFBA fuel scenario is shown in the figure 3.4. Looking at the $t=0$ beginning of the operation, it is seen that the concentration of ^{10}B in the PWR assembly is about 35 grams, and the concentration of ^{10}B in the VVER-1200 is about 31 grams. In

addition, it can be seen that the ^{10}B concentration approaches to 0 as the burn time increases. The reason for this is that the ^{10}B isotope, which absorbs the thermal neutron, forms ^{11}B , and it turns into two different nuclei of ^4_2He and ^7_3Li by the decay of ^{11}B . As the burn time increases, ^{10}B nuclei interact with more thermal neutrons to form ^{11}B and therefore the ^{10}B concentration in the assembly decreases. The greater the neutron absorber material concentration in the PWR assembly, the greater the interaction of the thermal neutrons with the absorber material, resulting in a lower criticality value at the beginning of the reactor operation.

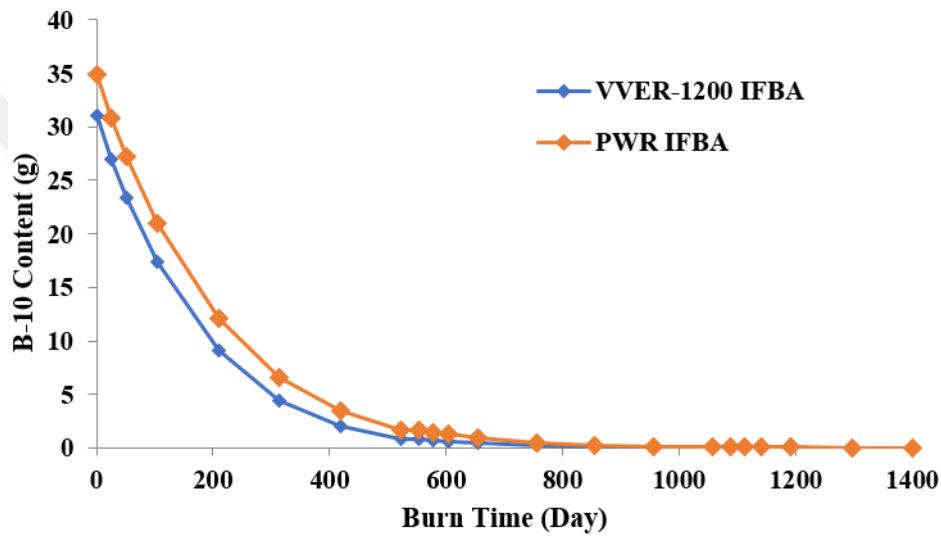


Figure 3.4 Change in ^{10}B concentration with increasing burn time for PWR and VVER-1200 assemblies IFBA fuel scenario at 300 K temperature

Figure 3.2 (c) and 3.2 (d) show a comparison of the criticality levels of reactor assemblies at the start of the reactor operation time in IBA scenarios with 5 wt% Gd_2O_3 and 9 wt% Gd_2O_3 concentrations. In both scenarios, unlike the standard fuel scenario such as the IFBA fuel scenario, the VVER-1200 assembly criticality level is higher than the PWR at the same system temperature values. In the IBA scenario, both assemblies have 8 IBA rods with a mixture of $\text{UO}_2\text{-Gd}_2\text{O}_3$. The total concentration of gadolinium isotopes, especially ^{157}Gd , in these rods is important in terms of how much the thermal neutron population in the system will decrease. Even though both reactor assemblies have the same number of IBA rods with the same Gd_2O_3 enrichment, the overall volume of the rods with the Gd_2O_3 neutron

absorber is also greater in the PWR because the fuel rods of the PWR has a larger diameter and height than the VVER-1200. Considering that the rods are completely filled with this material, it turns out that the amount of ^{157}Gd in the PWR assembly is greater than in the VVER-1200. As an example and to further the interpretation, Figure 3.5 shows the variation of ^{157}Gd concentration in assemblies with burn time for both reactors for the IBA fuel scenario with 5 wt% Gd_2O_3 concentration at 300 K temperature.

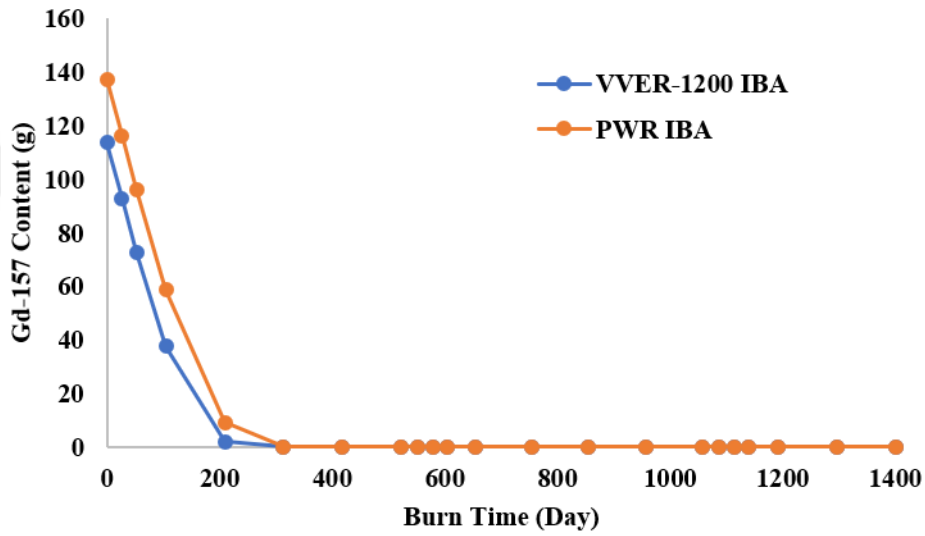
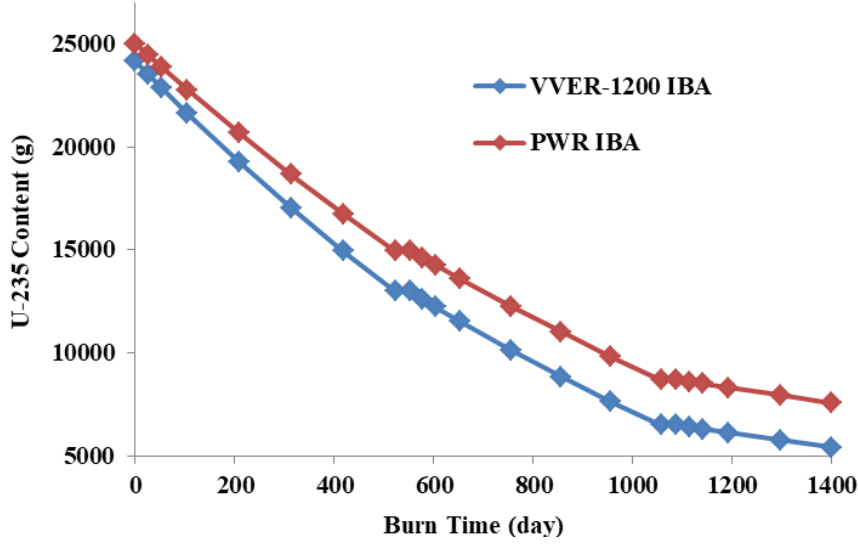
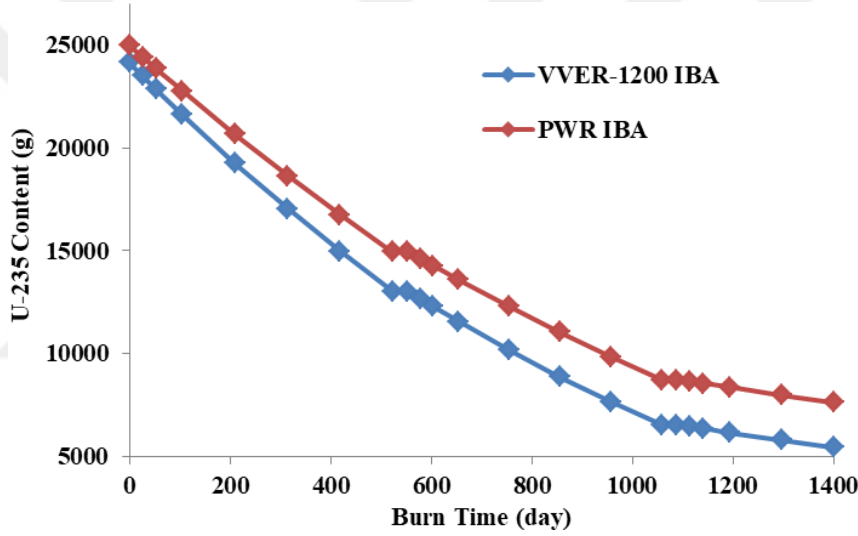


Figure 3.5 Change in ^{157}Gd concentration with increasing burn time for PWR and VVER-1200 assemblies 5 wt% Gd_2O_3 IBA fuel scenario at 300 K temperature

As can be seen from the figure 3.5, initially ^{157}Gd amount in the PWR assembly is larger. The amount of ^{157}Gd isotope in the PWR assembly is 137.4 grams compared to 113.8 grams in the VVER-1200 assembly. Contrary to the standard fuel and IFBA fuel scenarios, since Gd_2O_3 is mixed into the fuels containing UO_2 in this scenario, the UO_2 concentration in the system decreases with the increasing concentration of Gd_2O_3 mixed. This shows that the critical masses in the IBA rod scenarios are less than the first 2 scenarios mentioned. If the critical masses of the 5 wt% and 9 wt% scenarios are compared, the scenario with 9 wt% concentrations of $\text{UO}_2\text{-Gd}_2\text{O}_3$ fuels is the scenario with the least critical mass among all scenarios. In the 5 wt% scenario, the total mass of ^{235}U in the PWR assembly is 25010 grams, while in the VVER-1200 assembly it is 24160 grams.



(a)



(b)

Figure 3.6 (a) Change in ^{235}U concentration with increasing burn time for PWR and VVER-1200 assemblies under 5 wt% Gd_2O_3 IBA fuel scenario at 300 K temperature (b) Change in ^{235}U concentration with increasing burn time for PWR and VVER-1200 assemblies under 9 wt% Gd_2O_3 IBA fuel scenario at 300 K temperature

As an example, the figure 3.6 shows how ^{235}U concentrations change with burn time for the 5 wt% and 9 wt% IBA scenarios at 300 K assembly temperature. As can be seen from the figure 3.6, the PWR assembly has more critical mass than the VVER-1200 assembly in both scenarios. However, there is a little difference between the critical masses, less than 1000 grams, which explains why the PWR assembly with larger amount of ^{157}Gd is less critical. Due to this small difference between the critical masses, we see that there is a greater decrease in the thermal

neutron population in the PWR assembly where the ^{157}Gd concentration is higher, and therefore the effective neutron multiplication factor is less at the reactor start-up time at all temperatures. The criticality levels of the reactor assemblies may show different trends when the neutron-absorbing material approaches 0 and is completely burned inside the assembly.

Figure 3.2 (e) and 3.2 (f) shows the comparison of the effective neutron multiplication factor values for different temperatures in the hybrid fuel scenario of reactor assemblies. Since the hybrid fuel scenario is a combination of IBA and IFBA fuel scenarios, reactor criticality values are expected to change as in those two scenarios. In the hybrid fuel scenario, there are both ^{10}B and ^{157}Gd neutron absorbers in the assembly. In this scenario with 80 IFBA and 8 IBA rods, both the decrease in the total critical mass in the assemblies due to the mixture of $\text{UO}_2\text{-Gd}_2\text{O}_3$ and the number of thermal neutrons to be formed is higher than the standard fuel and IFBA rod scenarios, and the thermal neutrons formed in the system as a result of fission are higher than in other scenarios. The criticality values were expected to be lower because of the further reduction of the thermal neutron population due to exposure to the neutron absorber. As can be seen from the figures 3.2 (e) and 3.2 (f), in both scenarios with 5 wt% and 9 wt% Gd_2O_3 enrichment ratios, the criticality values at the start of assembly operation are lower than in other scenarios. It has been seen and discussed in the IFBA and IBA fuel scenarios that the criticality level of the PWR assembly is lower than that of the VVER-1200 assembly. In this scenario, the criticality level of VVER-1200 is higher than PWR at all temperatures, as it is a combination of the two scenarios mentioned. As the total masses of ^{10}B and ^{157}Gd in PWR is higher than that of VVER-1200 and the critical mass difference between the two reactor assemblies is less than 1 kg, neutron absorber materials cause the thermal neutron population to decrease more in the PWR assembly than in VVER-1200. When we look at the graphs of the scenarios containing boron and gadolinium, we see that the criticality difference between the two reactor assemblies increases and the criticality values decrease further in the scenarios containing boron. As it will be remembered, in IFBA rods, the neutron-absorbing material

ZrB₂ is placed on the fuel rod as a coating, while the neutron-absorbing material Gd₂O₃ used in IBA rods is placed directly into the fuel rod by mixing it with fuel. Let's consider the interaction of high-energetic neutrons that are formed as a result of fission in the assembly, with IBA and IFBA rods after they interact with water and reach thermal energy. When thermal neutrons interact with an IFBA rod, they interact with the neutron absorbing material ¹⁰B coating of the fuel rod, before the neutrons can interact with the fissionable material inside the fuel rod. As a result of this interaction, ¹⁰B absorbs thermal neutrons and reduces the thermal neutron population that can interact with ²³⁵U nuclei. When thermal energetic neutrons interact with IBA rods, it is easier for neutrons to penetrate inside the IBA rod, as there is no neutron absorber in the coating of the fuel rod, unlike IFBA rods. In IBA rods, since the neutron absorber material is mixed with the fuel in the fuel rod and its concentration is quite low compared to the fuel material, the thermal neutron entering the fuel rod is less likely to interact with the neutron absorber ²³⁵U nuclei instead of the uranium nuclei. Because of this low probability, ¹⁵⁷Gd cannot do as much work as ¹⁰B in reducing the thermal neutron population in the assembly, even though it has more thermal neutron absorption cross section. For this reason, assembly criticality decreases even more in scenarios with ¹⁰B. Looking at figures 3.2 (b) (c) and (d), it can be seen that the criticality values in the IFBA rod scenario are lower than in the IBA rod scenario. In addition, by looking at figures 3.2 (e) and (f), it can be seen how the criticality values decrease after ¹⁰B is added to the system in the hybrid fuel scenario created by adding IFBA rod to the IBA rod scenario. Therefore, the criticality differences between VVER-1200 and PWR have become even greater in scenarios with IFBA rods. Looking at the IFBA fuel and hybrid fuel scenarios, it can be seen that the difference between the effective neutron multiplication factor values of the two assemblies is even greater than in other scenarios.

In the hybrid fuel scenario with 9 wt% Gd₂O₃ enriched IBA rods, the total critical mass in the assembly will be even less due to the UO₂-Gd₂O₃ mixture, as in the previously interpreted IBA fuel scenarios. Since the scenario with the least critical

mass is 9 wt%, the system in which the thermal neutron population will be the least is in this scenario. Also since the thermal neutron population is low and the concentration of neutron absorber material with which thermal neutrons will interact is the highest in this scenario, this scenario has the lowest criticality levels compared to other scenarios. If the effective neutron multiplication for hybrid fuel scenario comparison is made from the figure 3.2 (e) and (f), while the VVER-1200 assembly has a criticality value of approximately 1.27 at 300 K in the 5 wt% scenario, this value has decreased to 1.26 in the 9 wt% scenario. While the PWR criticality value was approximately 1.25 at 300 K in the 5 wt% hybrid scenario, this value decreased to 1.24 in the 9 wt% hybrid scenario. According to the result obtained from these graphs, it has been observed that the assemblies have the lowest criticality values when the neutron absorber materials are used together and their concentrations are the highest when the assemblies start to operate.

3.2. Effective Neutron Multiplication Factor and Burn-Time

In this study, in addition to the variation of the effective neutron multiplication factor values of the PWR and VVER-1200 fuel assemblies with the temperature in the assembly, the variation with the assembly operation time was also observed. Assembly operation time can also be referred to as burn time, as it represents the time interval during which the fuel and neutron absorber materials in the assembly are burned. As it will be remembered from table 2.3, both reactor assemblies were operated for a total of 1400 days, divided into 21 time periods in total, and different power fractions were applied to the assemblies at each time interval. The figures show the effective neutron multiplication factor values of the PWR and VVER-1200 assemblies within the time periods shown in table 2.4 during the operation period of 1400 days.

Figure 3.7 (a) and 3.8 (a) show the change of effective neutron multiplication factor values of PWR and VVER-1200 assemblies with increasing burn time for the standard fuel scenario. The variation of effective neutron multiplication values with

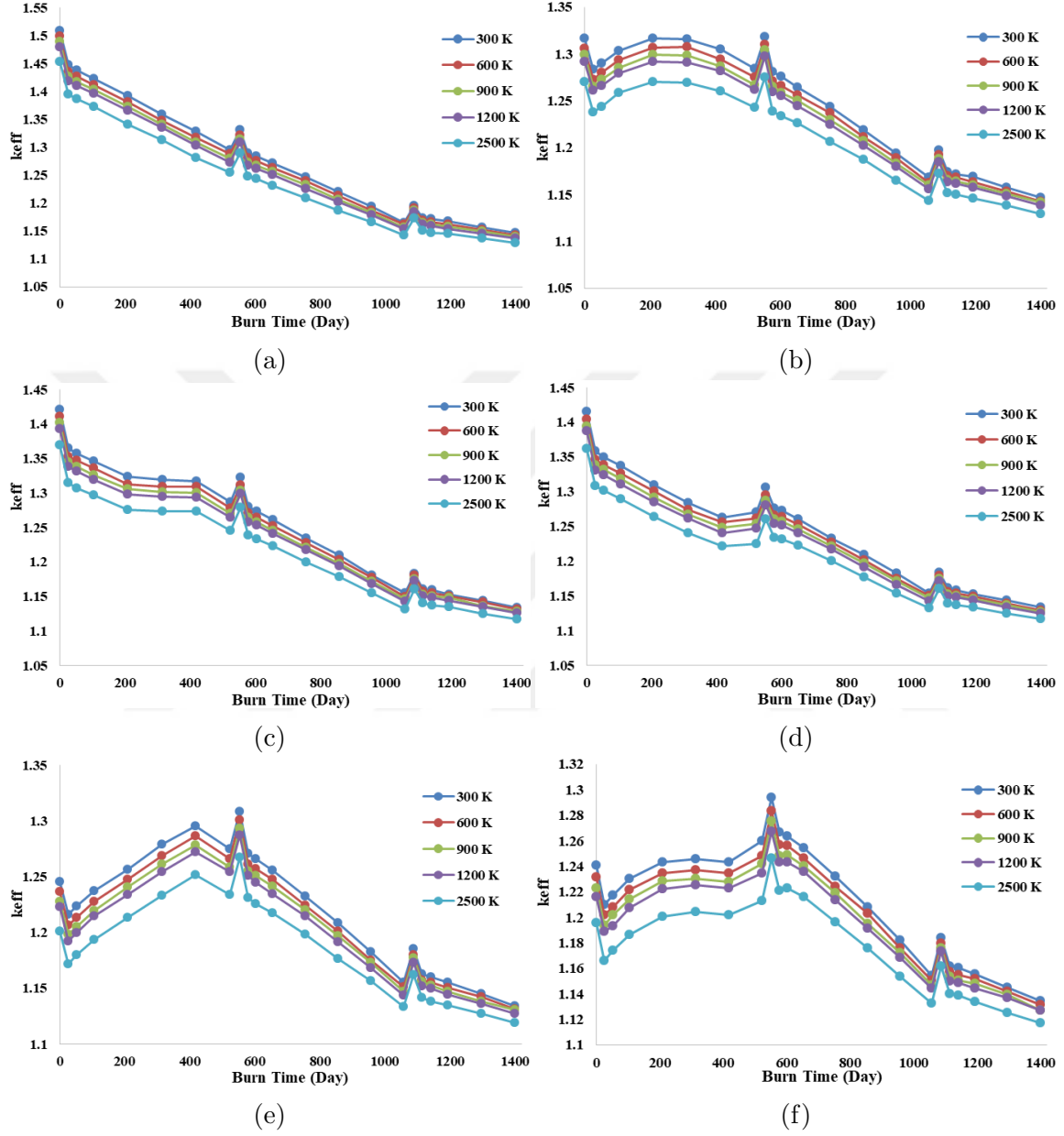


Figure 3.7 Change in Effective Neutron Multiplication Factor with Increasing Burn-Time and Temperature for PWR Assembly (a) for Standard Fuel scenario (b) for IFBA Fuel scenario (c) for 5 wt% Gd_2O_3 IBA Fuel scenario (d) for 9 wt% Gd_2O_3 IBA Fuel scenario (e) for 5 wt% Gd_2O_3 Hybrid Fuel scenario (f) for 9 wt% Gd_2O_3 Hybrid Fuel scenario

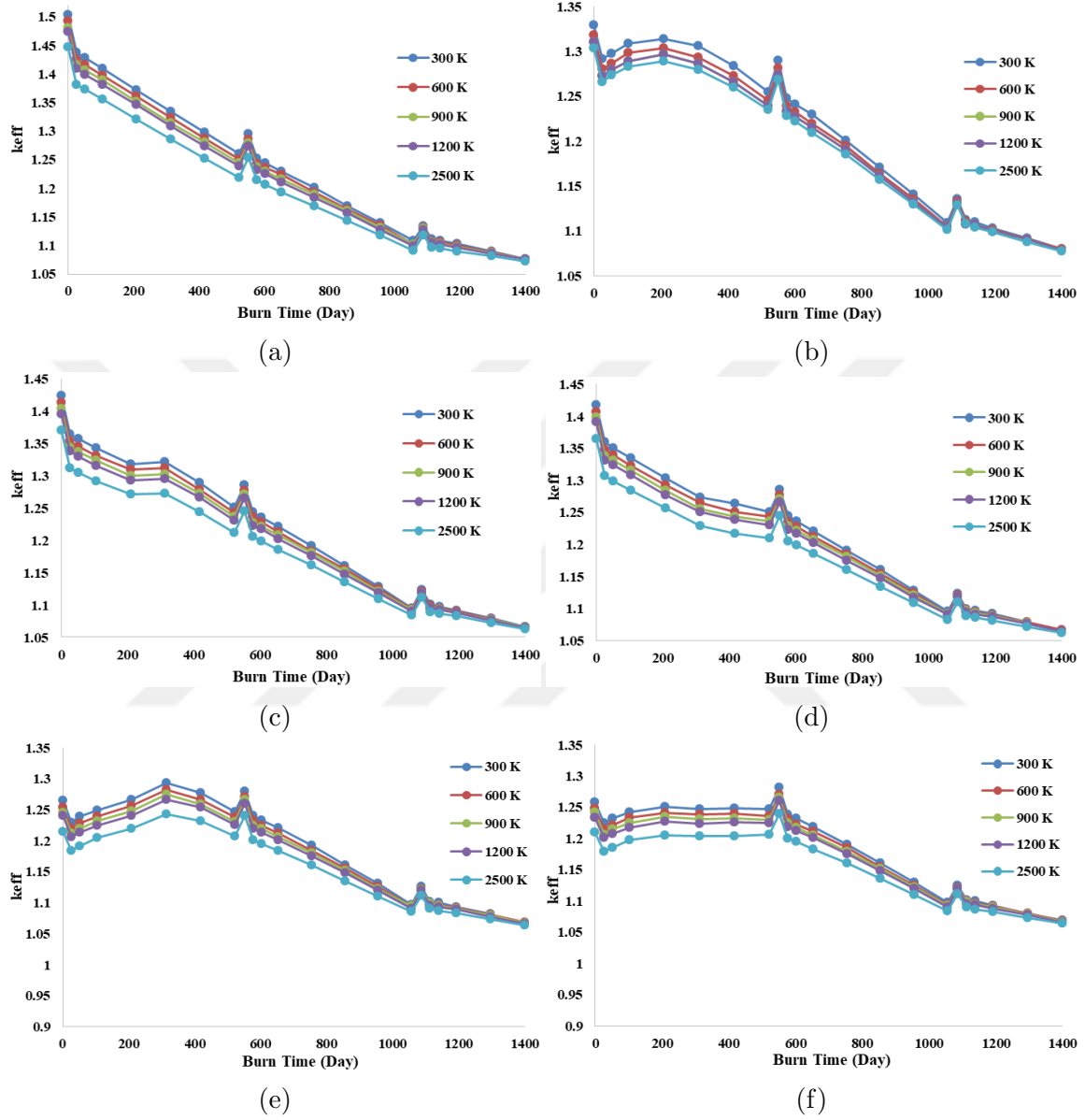


Figure 3.8 Change in Effective Neutron Multiplication Factor with Increasing Burn-Time and Temperature for VVER-1200 Assembly (a) for Standard Fuel scenario (b) for IFBA Fuel scenario (c) for 5 wt% Gd_2O_3 IBA Fuel scenario (d) for 9 wt% Gd_2O_3 IBA Fuel scenario (e) for 5 wt% Gd_2O_3 Hybrid Fuel scenario (f) for 9 wt% Gd_2O_3 Hybrid Fuel scenario

time is shown in the figures for assemblies at 300 K, 600 K, 900 K, 1200 K and 2500 K temperatures. As can be seen from the figures, the criticality values of both reactor assemblies decrease with increasing time. As soon as the reactor starts to operate, thermal neutrons interact with the target nuclei ^{235}U and the fission event occurs. As a result of the fission, the ^{235}U nuclei break apart to form other nuclei. For this reason, the amount of fissile element in the system decreases every time fission occurs in the system. The decrease in the fissile element ^{235}U causes the number of fissions and the number of thermal neutrons that can occur in the system to decrease. For this reason, the reactivity of the system gradually decreases and the criticality of the system decreases as the reactor operation time increases. Another reason is the poisons that are released as fission products in the system as the fission event takes place. The radioactive isotopes created by fission processes are referred to as fission products. Some of these isotopes function as neutron absorbers, lowering the reactor's total reactivity and neutron population. Reactor criticality decreases as the concentration of these fission products rises over time. As discussed in the introduction section, one of the most important of these poisons is ^{135}Xe . As the fission event in the system or assembly increases by increasing burn time or operation time, the probability of the emergence of this nucleus will increase, as well as causing the thermal neutron population in the system to decrease further. The decrease in the thermal neutron population will also reduce the reactivity by causing a decrease in the number of neutrons that will enable fission to occur by interacting with the fissile element. When looking at all the graphs showing the variation of the effective neutron multiplication factor value with increasing temperature for each scenario in figure 3.7 and figure 3.8, it can be seen that reactor criticality decreases with increasing temperature. If the effective neutron multiplication values are sorted for all temperatures from largest to smallest, they appear as 300 K, 600 K, 900 K, 1200 K and 2500 K. This shows us that in systems with these temperatures, the highest reactivity, namely fission, will occur in the system at 300 K, and the least reactivity will occur in the system at 2500 K. There are several reasons why the reactivity decreases as the system temperature increases. The first of these is what we call doppler broadening. As the temperature increases,

the atoms in the reactor core vibrate more vigorously, causing the neutron energies to spread out. This broadening of the neutron energy spectrum enhances the probability of neutron absorption by fuel nuclei, reducing the overall reactivity of the reactor. As will be remembered from Equation 1.5, reactor criticality varies inversely with the number of neutrons absorbed in the assembly. The increase in the number of neutrons absorbed by the fuel material or the neutron absorber in the assembly is one of the factors that reduces the criticality of the reactor. Considering that the concentration of ^{238}U in the fuel material is higher than the concentration of ^{235}U , it becomes more likely that these neutrons interact with ^{238}U instead of interacting with ^{235}U and making fission reaction, since the energy of the fission neutrons formed with the increasing temperature will also increase. For this reason, the probability of neutron absorption of the fuel material increases and this effect reduces the criticality of both PWR and VVER-1200 assemblies. Another reason for the decrease in criticality with increasing temperature is due to thermal expansion. Thermal expansion of the reactor materials, including the fuel, can also be brought on by increasing temperature inside the reactor assembly. Fuel density decreases as a result of this expansion. Since the mean free path of neutrons, which is the path that neutron travels between interactions and collisions, increases with decreasing fuel density, it is more likely that neutrons will leave the fuel before causing further fission reactions. As a result of the lower fuel density, the reactor assembly's reactivity diminishes with increasing temperature. Increasing temperature can also cause expansion in moderator material, water. Water slows down fast neutrons released by fission reactions and make them more likely to interact with ^{235}U to induce fission reaction. Density and neutron scattering cross section of the moderator material are temperature dependent. In water cooled reactors like PWR and VVER-1200, increase in temperature decreases the density of the water and reducing its capability of neutron slowing. As a result of the reduction of the neutron slowing capability of the water, the neutrons cannot be in the energy they need to create fission reaction by interacting ^{235}U , and this reduces the criticality of the system.

Figure 3.7 (b) and 3.8 (b) show the change of effective neutron multiplication factor

value with increasing burn time at all temperatures for the IFBA fuel scenario. As can be seen from the figures, the criticality of both reactor assemblies initially decreased, increased up to a certain point and started to decrease again. In the reactor shut down periods, the criticality value increased and decreased again as soon as the reactor started to operate. The reason for the initial decrease in criticality is the poison isotopes formed as a result of fission, just like in the standard fuel scenario, and in addition, the neutron absorber material ^{10}B in the assembly. The ^{10}B isotope in the ZrB_2 mixture, which is used as a coating on the fuel rods, interacts with the thermal neutrons reaching the fuel rod, absorbs the neutrons and causes a decrease in the number of fission reactions that will occur. As the isotope of ^{10}B interacting with thermal neutrons forms ^{11}B and decays to ^4_2He and ^7_3Li nuclei, the total amount of ^{10}B in the system decreases as the burn time increases. Figure 3.4 shows how the ^{10}B concentration in the system changes at 300 K temperature for the IFBA rod scenarios of PWR and VVER-1200 reactors. As it will be remembered from Equation 1.5, the criticality of the system decreases as the neutron absorption in the assembly increases. In this scenario, unlike the standard fuel scenario, there is a ^{10}B neutron absorber in the assembly. For this reason, it is seen that the PWR and VVER-1200 IFBA fuel criticality values are lower when compared to the standard fuel scenarios. Looking at the figures, it is seen that the reactor criticality increases from approximately the 50th to the 400th day of the operation time, after the 400th day it starts to decrease until the reactor is shut down, and decreases again after the reactor starts operating again. The reason for the increase in the criticality until the 400th day is due to the decrease in the total amount of ^{10}B in both reactor assemblies as the thermal neutron interaction increases. As the amount of ^{10}B decreases, the amount of neutron absorption in the system decreases, and as a result, the thermal neutron population begins to increase over time. As the ^{10}B concentration in the system decreases, neutron absorption decreases in each generation, which increases the thermal neutron population in each generation, resulting in an increase in the number of fissions occurring in the system. As can be seen from figure 3.4, the ^{10}B content in the two reactor assemblies came to an end in approximately 400 days and were completely depleted towards the 600th day. This explains why reactor

criticality values tend to decrease rather than increase at around day 400 in the IFBA scenario. When ^{10}B is completely depleted, since there is no neutron absorbing material other than fuel in the system, the criticality trend tends to decrease as time increases, just like in the standard fuel scenario. The reason for this is that, as explained in the standard fuel scenario, the fissile isotope ^{235}U burns with increasing operation time and this reduces the thermal neutron population that may occur in the system, causing a decrease in the reactor criticality and the poison elements released as a result of fission.

Figure 3.7 (c), 3.7 (d), 3.8 (c) and 3.8 (d) show the change of effective multiplication factor values with increasing burn time for 300 K, 600 K, 900 K, 1200 K and 2500 K temperatures for IBA fuel scenarios of PWR and VVER-1200 assemblies with enrichment rates of 5 wt% and 9 wt% Gd_2O_3 . As can be seen from the figures, the assembly criticality values are less than the standard fuel scenario and higher than the IFBA fuel scenario. The reason why it is less than the standard fuel scenario is due to the presence of neutron absorber isotope ^{157}Gd in the system. Thermal neutrons interacting with the IBA fuel interact with the neutron absorber material in the fuel rod and the thermal neutron population in the system decreases. While the only neutron absorber material in the system in the standard fuel scenario is the fuel material, in the IBA fuel scenario it is both the fuel material and the Gd_2O_3 material mixed with the fuel. Therefore, the total thermal neutron absorption in the system increases, which reduces the reactivity in the system, resulting in a lower criticality value. The reason why the assemblies are more critical in the IBA fuel scenario compared to the IFBA fuel scenario is that, as explained before, the neutron absorber material ZrB_2 in the IFBA fuel scenario is used as a coating outside the fuel rods, and the Gd_2O_3 neutron absorber material used in the IBA fuel scenario is directly mixed with the fuel material UO_2 . In addition, while the number of rods with ZrB_2 coating is 80 in the IFBA scenario, the number of rods with the UO_2 - Gd_2O_3 mixture is 8 in the IBA fuel scenario, and the Gd_2O_3 enrichments are quite low compared to the fuel composition in the rods. While the thermal neutrons formed in the assembly interact with the neutron absorber material before

interacting with the fuel in the IFBA fuel scenario, they interact with the fuel and Gd_2O_3 material in the IBA fuel scenario. While the thermal neutrons formed in the IFBA fuel scenario interact with the 80 IFBA rods in the assembly, they interact with the ZrB_2 coating before the fuel material, and the ^{10}B neutron absorber in the coating material reduces the thermal neutron population that can interact with the fuel material. In IBA rods, on the other hand, the number of thermal neutrons that can reach the fuel material is higher than that of IFBA rods, since they do not encounter any neutron absorbers before they reach to fuel material inside the fuel rods. Thermal neutrons interacting with the IBA rod interact either with the fuel material inside or with the Gd_2O_3 absorber material. The number of thermal neutrons absorbed as a result of the interaction of thermal neutrons with Gd_2O_3 nuclei is less than the number of thermal neutrons absorbed due to the interaction with ^{10}B , both because the number of IBA rods in the assembly is less than that of IFBA rods, and the Gd_2O_3 enrichment used in IBA rods is less compared to the fuel composition. Therefore, more reactivity occurs in the IBA rod scenario than in the IFBA rod. As a result, the system criticality is higher in the IBA rod scenario than in the IFBA rod. Looking at the figures, it can be seen that both PWR and VVER-1200 assemblies have higher criticality values in the IBA fuel scenario than in the IFBA fuel scenario. Considering the criticality values for the 5 wt% Gd_2O_3 scenarios during the 1400 day operation period in Figure 3.7 (c) and 3.8 (c), the criticality level of both reactors decreases at the beginning, increases slightly on the 400th day, and decreases towards the reactor shutdown time. After the reactor is shut down, the criticality level increases and decreases again as soon as it starts to operate again. From the moment the reactor started to operate, the poisons formed as a result of fission and the total fissile isotope ^{235}U concentration in the system decreases in each generation as a result of fission, and the criticality level decreases as the burn time increases with the effect of neutron absorber Gd_2O_3 in the system. As soon as the amount of ^{157}Gd in the system starts to run out, the only material with which thermal neutrons can interact is the fuel material and thermal neutron absorption in the system decreases, which increases the reactivity in the system by increasing the possibility of fission reaction. The reactor criticality therefore undergoes a slight

increase. When the ^{157}Gd is completely depleted, just like in the standard fuel scenario, the thermal neutrons interacting only with the fuel material cause the fission of the fissile isotope ^{235}U and decrease its concentration each generation, and therefore the criticality levels decrease. Looking at the figures 3.7 (d) and 3.8 (d) showing the criticality values for the 9 wt% Gd_2O_3 IBA scenarios, it is seen that the criticality values of both PWR and VVER-1200 assemblies decreased compared to the 5 wt% Gd_2O_3 IBA scenario. In this scenario, increasing the neutron absorber concentration from 5 wt% to 9 wt% increases the probability of interaction between thermal neutrons and Gd_2O_3 material. As a result, the decrease in the thermal neutron population in the system is greater, which reduces the number of fission that can occur and reduces the criticality value. When the criticality values of PWR and VVER-1200 assemblies are examined for different temperatures in IBA scenarios, it is seen that they have the highest criticality level at the lowest temperature 300 K, and the lowest criticality level at the highest temperature 2500 K. If the criticality values are compared for the temperatures from large to small, it is seen that they are 300 K, 600 K, 900 K, 1200 K and 2500 K, respectively. With each temperature increase, the thermal neutron population in the system will decrease, the thermal neutrons formed will interact with either the fuel material or Gd_2O_3 , and the increasing number of more energetic neutrons that cause the number of thermal neutrons to decrease will start to interact with ^{238}U and be absorbed by ^{238}U . Therefore, the number of fissions in the system decreases as the temperature increases and the system criticality decreases.

Figures 3.7 (e), 3.7 (f), 3.8 (e) and 3.8 (f) show the variation of the effective neutron multiplication factor values with burn time at all temperatures for the Hybrid Fuel scenarios with 5 wt% and 9 wt% Gd_2O_3 enrichments of PWR and VVER-1200 assemblies. Compared to other scenarios, it can be seen from the figures that the lowest criticality values occur in hybrid fuel scenarios. When compared among themselves, the scenario with the lowest criticality is the hybrid fuel scenario with 9 wt% Gd_2O_3 enrichment. This scenario is also the one with the lowest criticality among all scenarios. In the hybrid fuel scenario, the thermal neutron population

decreases the most in this scenario due to the presence of both 8 IBA rods and 80 IFBA rods in the assembly. In this scenario, thermal neutrons formed in the assembly will be absorbed by both ^{10}B and ^{157}Gd neutron absorbers, reducing the number of neutrons that can interact with ^{235}U . In addition, since the Gd_2O_3 material is mixed with the UO_2 fuel and placed in the assembly, the presence of Gd_2O_3 in the assembly will reduce the total concentration of the UO_2 fuel material, which will cause a decrease in the amount of ^{235}U , thus reducing the total number of thermal neutrons that may occur in the system. As it will be remembered from equation 1.5, since the criticality of the reactor changes inversely with the absorption of thermal neutrons, the lowest criticality values occur in this scenario where the number of thermal neutrons lost in the system is the highest. Considering the 5 wt% and 9 wt% hybrid scenarios, according to the criticality values formed during the 1400-day operation period, at the beginning of the reactor operation the criticality value decreases due to the neutron absorber poisons which are formed as fission products, ^{235}U burned as a result of fission, and IBA and IFBA rods placed in the assembly. As the ^{10}B and ^{157}Gd neutron absorbers in the system interact with thermal neutrons, their total concentration in the assembly will decrease, so thermal neutron absorption by neutron absorber materials decreases in each generation and the thermal neutrons that will be absorbed by the neutron absorbers begin to interact with the fuel and cause the number of fission occurring in each generation to increase. Therefore, as can be seen from the figures, the reactor criticality increases until the ^{10}B and ^{157}Gd concentrations are completely depleted. When the neutron absorbers are completely burned, the reactor criticality decreases as time progresses, because the total concentration of ^{235}U burned as a result of the interaction of thermal neutrons only with the fuel decreases in each generation. This decrease continues until the 520th day, which is the reactor shut down time, and the criticality increases after the reactor is shut down. As soon as the reactor is started to operate again, the criticality of the reactor decreases as time progresses with the absence of neutron absorber in the system, just like in the previous scenarios. In the hybrid scenario with 9 wt% Gd_2O_3 enrichment, the number of thermal neutrons that may occur decreased due to the increasing total amount of Gd_2O_3 and the decreasing critical

mass in the assembly, and the lowest criticality values were obtained in this scenario, since the thermal neutrons formed would be exposed to the neutron absorber the most in this scenario. When the variation of criticality values with temperature is examined, it is seen that the result obtained in the previous scenarios is also encountered in this scenario. As the temperature increases, the criticality values of both reactors decrease and it is seen that the highest criticality value is obtained at 300 K temperature and the lowest critical value is at 2500 K temperature.

In this study, the variation of the effective neutron multiplication factor value with increasing operation time was examined separately for the PWR and VVER-1200 assemblies at all temperatures, as well as examined together, and the criticality values of the two reactor assemblies in different scenarios were compared. As seen in Figures 3.7 and 3.8, all scenarios were simulated for temperatures of 300 K, 600 K, 900 K, 1200 K and 2500 K, and the criticality values of both reactors were obtained. In order to compare the criticality values of PWR and VVER-1200 assemblies in different scenarios, the reactor operating temperature of 600 K was chosen. Although we have criticality values for all temperatures, only 600 K temperature values are considered to compare the criticality of two reactors under standard fuel, IFBA fuel, IBA fuel and Hybrid fuel scenarios. The change of effective neutron multiplication factor values of PWR and VVER-1200 assemblies at 600 K temperature with increasing operation time can be seen in figure 3.9.

As can be seen in Figure 3.9 (a), when the criticality values of the PWR and VVER-1200 reactors are examined under the standard fuel scenario, it is seen that the criticality values of the PWR assembly are higher than the VVER-1200 during the 1400-day operating time. Although the critical mass values in the reactor assemblies are close to each other as the critical mass amounts in the reactor assemblies are almost the same when the reactors start operation, the PWR assembly has a higher criticality value at the beginning because it has more critical mass than VVER-1200 assembly. As the operation time increases, the difference between the criticality values of PWR and VVER-1200 starts to increase as the more fission reaction takes place. The criticality values of both reactors decrease as a result of the decrease

in the critical mass in the system due to the fission caused by the interaction of thermal neutrons with ^{235}U in each generation.

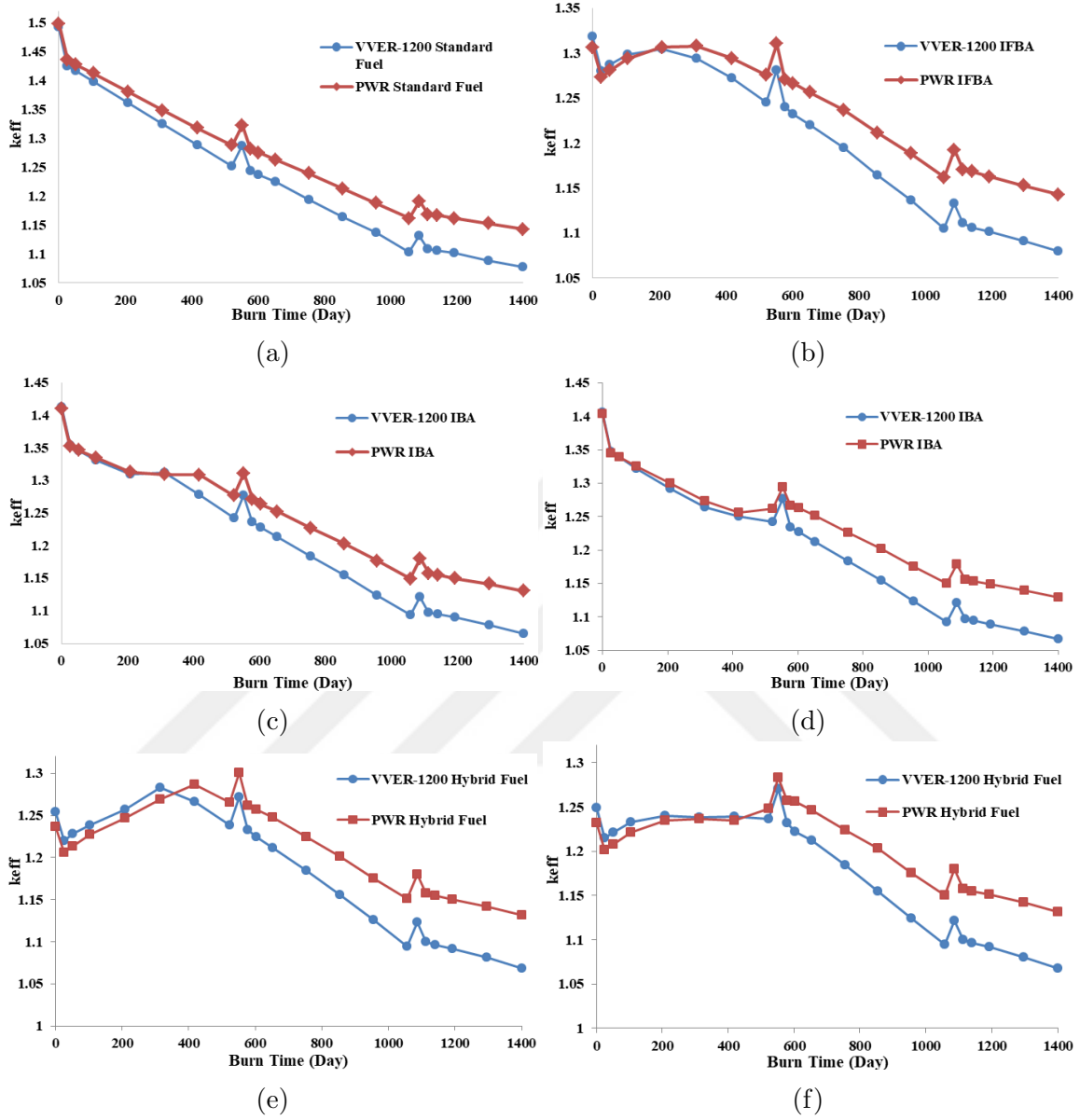


Figure 3.9 Change in Effective Neutron Multiplication Factor with Increasing Burn-Time for PWR and VVER-1200 Assemblies at 600 K Temperature (a) for Standard Fuel scenario (b) for IFBA Fuel scenario (c) for 5 wt% Gd_2O_3 IBA Fuel scenario (d) for 9 wt% Gd_2O_3 IBA Fuel scenario (e) for 5 wt% Gd_2O_3 Hybrid Fuel scenario (f) for 9 wt% Gd_2O_3 Hybrid Fuel scenario

Reactor criticality values at 600 K temperature for the IFBA fuel scenario are shown in Figure 3.9 (b). We chose to conduct a comparative analysis of the values observed at a temperature of 600 K due to its significance as the operational temperature of the reactors. Contrary to the standard fuel scenario, in this scenario, the VVER-

1200 assembly has a higher criticality value than the PWR assembly at the start of the reactor operation. This is because the total concentration of ^{10}B in the 80 IFBA rod coatings found in this scenario is higher in the PWR assembly. The thermal neutrons generated in the PWR assembly interact more with the ^{10}B , thus reducing the thermal neutron population more than the VVER-1200. As can be seen from the figure, the criticality decreases for a while after the start of the reactor operation time, due to fission product poisons formed in both reactors, burned ^{235}U nuclei and neutron absorber ^{10}B . As the amount of ^{10}B in the system decreases as time increases, the criticality of both assemblies begins to increase. Since the number of thermal neutrons absorbed in each generation will decrease with decreasing ^{10}B concentration and the fission number in the system will increase, the criticality values increase with the decrease of ^{10}B . After ^{10}B is completely depleted in both assemblies, the number of fissions in PWR assembly with more critical mass will be higher than VVER-1200 assembly since there is no neutron absorber effect to prevent the thermal neutrons in the system from interacting with the fuel material. That's why the critical value of PWR assembly begins to overpower the criticality value of VVER-1200 after the depletion of ^{10}B concentration in both systems.

Figure 3.9 (c) and 3.9 (d) show the change of effective neutron multiplication factor with increasing operation time under IBA fuel scenarios with 5 wt% and 9 wt% Gd_2O_3 enrichments at 600 K temperature of PWR and VVER-1200 assemblies. In this scenario, the trend of the criticality for both reactors is similar to the IFBA fuel scenario. Since the total concentration of Gd_2O_3 is higher in the PWR assembly than in the VVER-1200 assembly, it is seen that the effect of the neutron absorber on the thermal neutrons is more in the PWR assembly. Initially, the criticality value of the VVER-1200 assembly appears to be slightly higher than that of the PWR assembly, and this continues until the ^{157}Gd in the system is completely burned. While the thermal neutrons formed in the IFBA scenario interact with the ^{10}B absorber in the rod coating before interacting with the fuel material, there is no such effect in the IBA rod. The thermal neutrons interact with the $\text{UO}_2\text{-Gd}_2\text{O}_3$ mixture in any of the 8 IBA rods in the assembly, and if it can interact with the ^{235}U in the mixture,

the fission event takes place. If it interacts with the ^{157}Gd , the thermal neutron is absorbed by this nucleus. Therefore, the negative effect on the total reactivity in the system in the IBA rod scenario is not as much as in the IFBA rod scenario. With the effect of Gd_2O_3 absorber in the system, it is seen that the VVER-1200 assembly has more criticality than the PWR assembly from the moment the reactors start operating, and the criticality value of both reactors decreases for a while, and the criticality increases during the time when the ^{157}Gd nuclei in the system starts to burn completely. As the concentration of ^{157}Gd in the system begins to decrease, the number of thermal neutrons lost due to absorption in the system decreases in each generation, as in the IFBA scenario, and as a result, the number of fission in each generation begins to increase. It is seen that reactor criticality starts to increase with increasing fission number. When the neutron absorbing material is completely burned, there will be no effect in the system to prevent the thermal neutrons from interacting with the fuel material, so thermal neutrons interact with ^{235}U , causing the total critical mass in the system and reactor criticality to decrease in each generation. This decrease, which starts with the complete combustion of ^{157}Gd , starts on the 400th day in the 5 wt% Gd_2O_3 scenario, and on the 600th day in the 9 wt% Gd_2O_3 scenario. Since the total Gd_2O_3 concentration in the assembly is higher in the IBA scenario with 9 wt% Gd_2O_3 , it takes longer for Gd_2O_3 to be completely burned out of the system in this scenario.

Figures 3.9 (e) and 3.9 (f) show the variation of effective neutron multiplication values of PWR and VVER-1200 assemblies with increasing burn time for hybrid fuel scenarios at 600 K. Looking at the figures, it is seen that the effective neutron multiplication values change as in the IBA and IFBA fuel scenarios. This is an expected result as the hybrid fuel scenario is created as a result of combining these two scenarios. According to the effective neutron multiplication values when the reactors start operation, it is seen that the VVER-1200 assembly has more criticality than the PWR assembly in this scenario at the beginning of the operation. Because the concentration of ^{10}B in the coating of IFBA rods and the concentration of ^{157}Gd mixed with fuel in IBA rods are higher in the PWR assembly than in the

VVER-1200 assembly, the thermal neutrons generated in the PWR assembly are exposed to more absorbent material. Therefore, the thermal neutron population in the PWR assembly decreases more than in the VVER-1200 assembly. Looking at Figures 3.9 (e) and 3.9 (f), it is seen that the criticality of both reactor assemblies initially decreased, then increased over a period of time and then dropped again. The reason for the initial decrease is the neutron absorber poisons formed with the fission reactions that started to take place in the system, the decreasing critical mass and the IBA and IFBA rods in the system. The concentrations of neutron absorbing materials ^{10}B and ^{157}Gd in the system begin to decrease as they interact with thermal neutrons, and therefore the number of thermal neutrons absorbed by neutron absorbers in the system begins to decrease in each generation. With the decrease in the number of absorbed thermal neutrons, since the thermal neutron population in the system will start to increase in each generation, the number of reactions increases and the reactor criticality begins to increase as well. This situation continues until the ^{10}B and ^{157}Gd in the system are completely burned. As can be seen in Figure 3.9 (e), in the 5 wt% Gd_2O_3 hybrid scenario, the increase in the criticality of VVER-1200 assembly due to the depletion of neutron absorbers ends at approximately 300 days, and in the PWR assembly at approximately 420 days. Since both IBA and IFBA rods are present in the hybrid fuel scenario, the greatest loss of thermal neutron population occurs in this scenario. The generated thermal neutrons can interact with both IBA and IFBA rods, unlike the previous scenarios. This, in turn, will decrease the thermal neutron population the most in this scenario, as well as increase the depletion times of the ^{10}B and ^{157}Gd isotopes in the system. Therefore, when the hybrid fuel scenario is compared with the IFBA fuel scenario, it is seen that the increase in the criticality level takes longer. When we look at the 9 wt% Gd_2O_3 hybrid scenario in Figure 3.9 (f), it is seen that the increase in the criticality level takes longer than 5 wt% Gd_2O_3 hybrid scenario. The reason for this is the increased number of ^{157}Gd nuclei as a result of increasing the Gd_2O_3 concentration in IBA rods from 5 wt% to 9 wt%. Due to the increased number of ^{157}Gd nuclei, the neutron absorbers in this scenario will take longer to burn completely and therefore the increase in criticality level will take slightly longer than

in 5 wt. On days 522 and 1057, which are the time periods of reactor shutdown, an increase in criticality levels is observed due to the build up of ^{135}Xe when the reactor is shut down. From the moment the reactor starts operating after both shutdowns, criticalities begin to decrease as expected. Since the thermal neutrons will only interact with the fuel materials after the neutron absorber materials in the system are completely burned, the criticality value of the PWR assembly with a higher critical mass begins to overpower VVER-1200 and has a higher criticality value for the remaining operation time.

3.3. Accumulation of Actinides in PWR and VVER-1200 Assemblies

In this study, the variation of ^{235}U , ^{238}U , ^{239}Pu and ^{240}Pu isotopes in PWR and VVER-1200 assemblies with increasing operation time was investigated. Observing changes in the mass of specific isotopes, including ^{235}U , ^{238}U , and ^{239}Pu , holds significant importance in nuclear reactor operations. The mass change of ^{235}U is essential for sustaining a controlled chain reaction within the reactor. By monitoring the depletion of ^{235}U and the accumulation of fission by products, operators can evaluate the fuel's burnup. The production and consumption of ^{239}Pu are integral to nuclear reactor operations. Monitoring the mass change of ^{238}U facilitates the estimation of ^{239}Pu generation and consumption in the reactor. Reactor assemblies were modeled for standard fuel, IFBA fuel, IBA fuel and hybrid fuel scenarios and then simulated for 300 K, 600 K, 900 K, 1200 K and 2500 K temperatures. The effect of increasing assembly temperature on the mass changes of these isotopes in the system was also observed. Radioactive materials with atomic numbers between 89 and 103 are called actinide [35]. Accordingly, the changes of actinide masses according to different temperatures were compared. In addition, the standard fuel scenario for PWR and VVER-1200 assemblies and the varying actinide masses at 600 K system temperature are compared. In addition, the varying actinide masses for the standard fuel scenario at 600 K temperature were compared for the PWR and

VVER-1200 assemblies. We have these values for all scenarios and temperatures, and the results in all of them have the same trend with the results in the standard fuel scenario given in this thesis. Therefore, only the results obtained for the standard fuel scenario at 600 K are included in this study. Other results we have will be used in our future work.

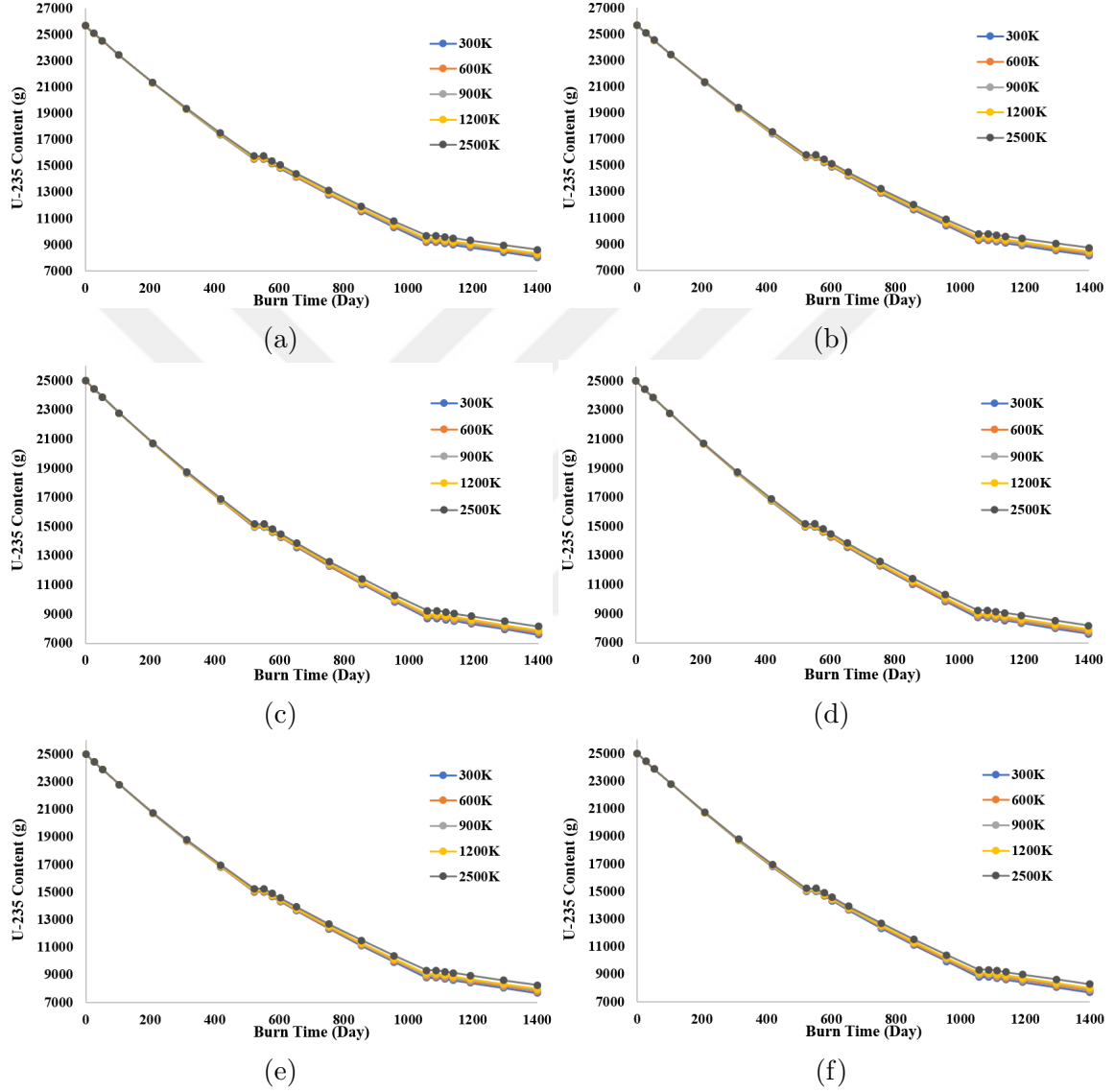


Figure 3.10 Comparison of change in ^{235}U mass with respect to burn time for PWR assemblies at different temperatures (a) for standard fuel scenario (b) for IFBA fuel scenario (c) for IBA fuel scenario with 5 wt% Gd₂O₃ (d) for IBA fuel scenario with 9 wt% Gd₂O₃ (e) for hybrid fuel scenario with 5 wt% Gd₂O₃ (f) for hybrid fuel scenario with 9 wt% Gd₂O₃

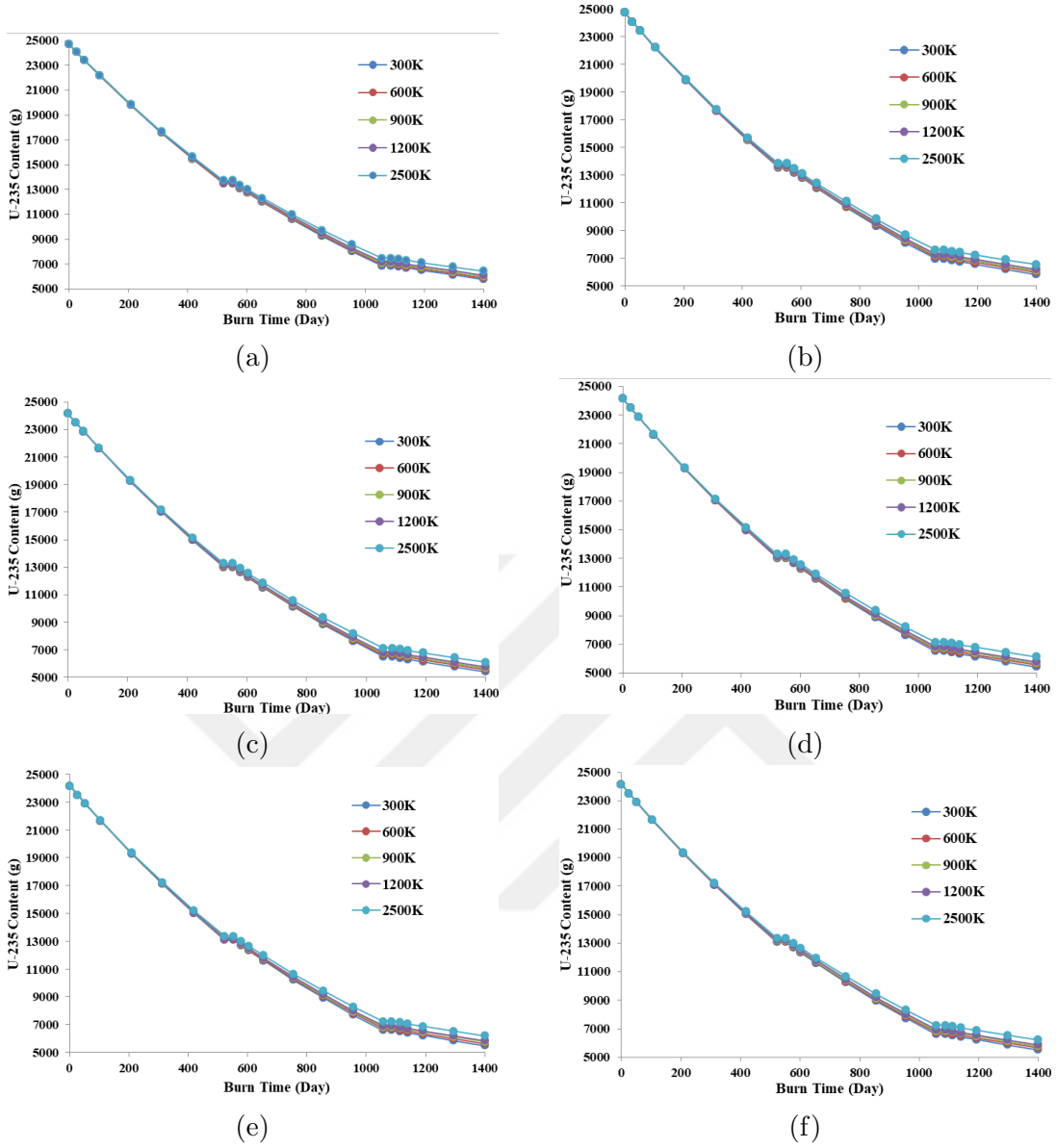


Figure 3.11 Comparison of change in ^{235}U mass with respect to burn time for VVER-1200 assemblies at different temperatures (a) for standard fuel scenario (b) for IFBA fuel scenario (c) for IBA fuel scenario with 5 wt% Gd_2O_3 (d) for IBA fuel scenario with 9 wt% Gd_2O_3 (e) for hybrid fuel scenario with 5 wt% Gd_2O_3 (f) for hybrid fuel scenario with 9 wt% Gd_2O_3

Figures 3.10 and 3.11 show the variation of ^{235}U mass in the system with increasing burn time for different scenarios of PWR and VVER-1200 assemblies at temperatures of 300 K, 600 K, 900 K, 1200 K and 2500 K, respectively. When the change in the mass of ^{235}U in both PWR and VVER-1200 assemblies is examined with increasing assembly temperature, it is seen that the mass change varies from temperature to temperature, and the increase in assembly temperature effects the burning rate

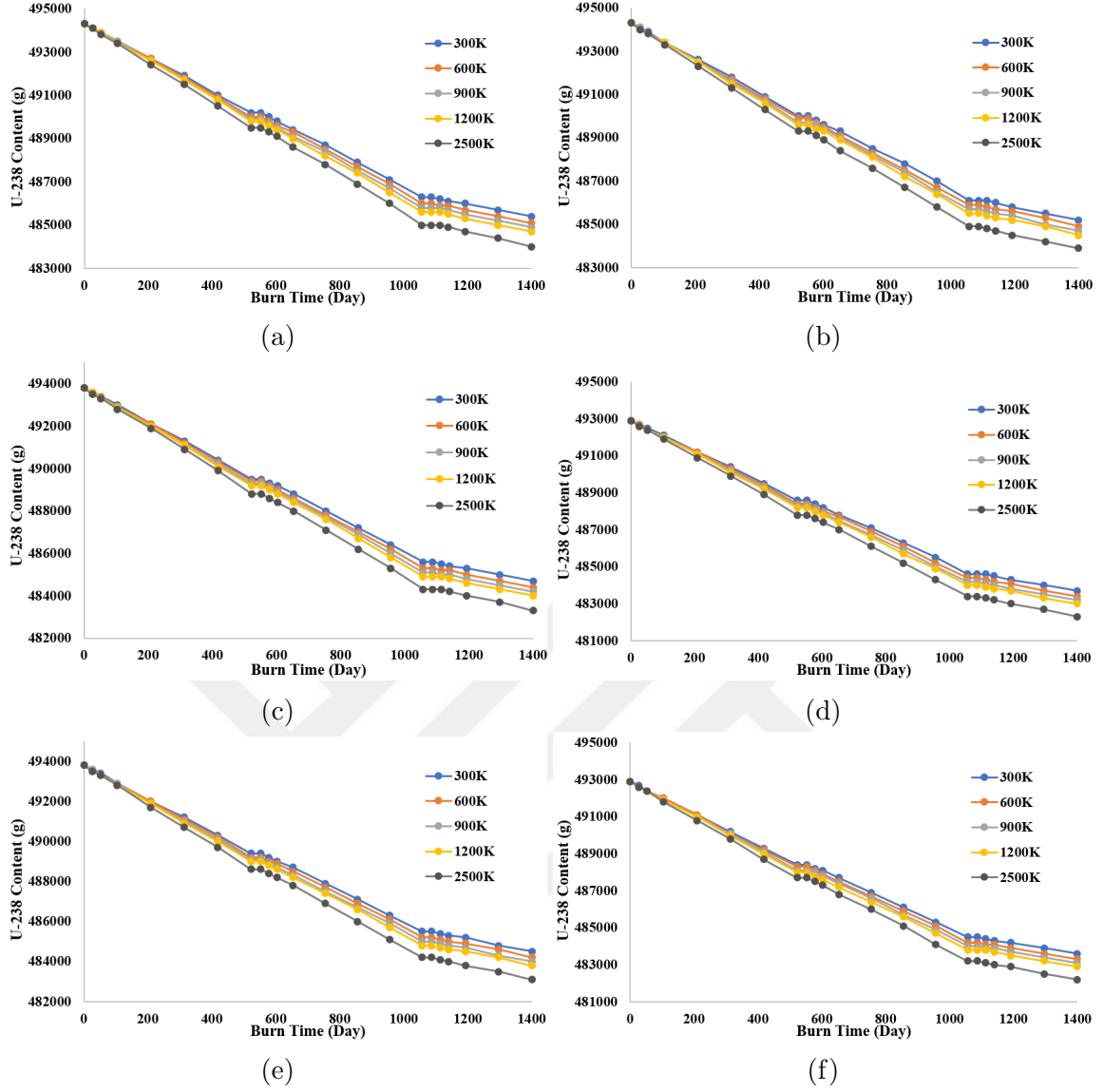


Figure 3.12 Comparison of change in ^{238}U mass with respect to burn time for PWR assemblies at different temperatures (a) for standard fuel scenario (b) for IFBA fuel scenario (c) for IBA fuel scenario with 5 wt% Gd_2O_3 (d) for IBA fuel scenario with 9 wt% Gd_2O_3 (e) for hybrid fuel scenario with 5 wt% Gd_2O_3 (f) for hybrid fuel scenario with 9 wt% Gd_2O_3

of ^{235}U in the system. When the ^{235}U masses of the assemblies after 1400 days of operation are compared for all scenarios by looking figures 3.10 and 3.11, it can be seen that the assemblies with the least ^{235}U mass are the assemblies with temperature of 300 K, and the assemblies with the highest temperature of 2500 K. If the remaining ^{235}U masses in the system are compared from small to large according to the assembly temperatures, it is seen that they are 300 K, 600 K, 900 K, 1200 K,

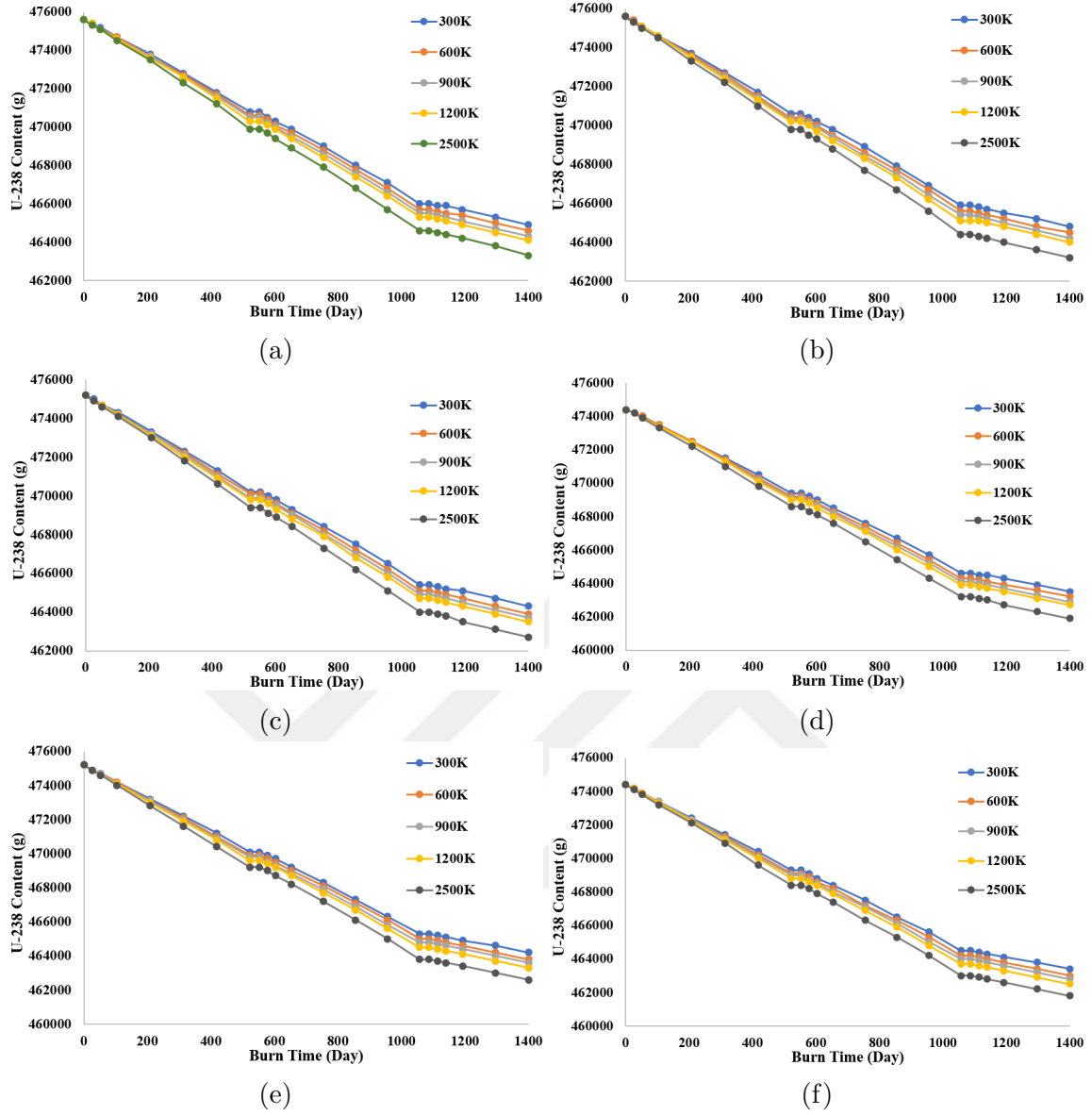


Figure 3.13 Comparison of change in ^{238}U mass with respect to burn time for VVER-1200 assemblies at different temperatures (a) for standard fuel scenario (b) for IFBA fuel scenario (c) for IBA fuel scenario with 5 wt% Gd₂O₃ (d) for IBA fuel scenario with 9 wt% Gd₂O₃ (e) for hybrid fuel scenario with 5 wt% Gd₂O₃ (f) for hybrid fuel scenario with 9 wt% Gd₂O₃

and 2500 K. According to the results that can be seen in the figures 3.10 and 3.11, as the temperature increases, the mass of ^{235}U burned in the reactor decreases. In the simulated systems, it is seen that ^{235}U is burned the most at the lowest temperature of 300 K, and therefore the least remaining mass of ^{235}U is in the assemblies with 300 K temperature. At the highest temperature of 2500 K, it is observed that ^{235}U is burned the least, so the maximum remaining mass of ^{235}U is in the assemblies with

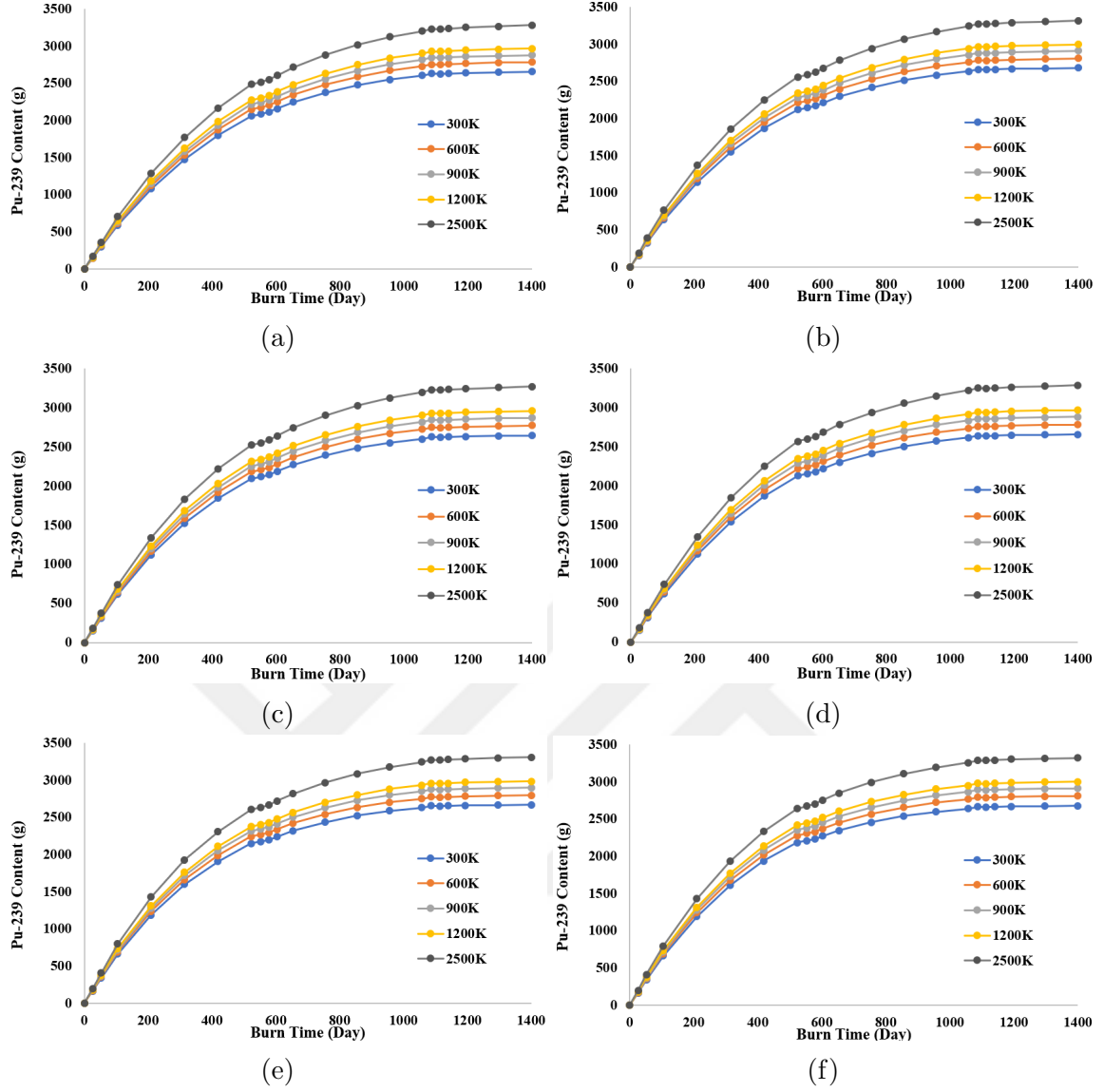


Figure 3.14 Comparison of change in ^{239}Pu mass with respect to burn time for PWR assemblies at different temperatures (a) for standard fuel scenario (b) for IFBA fuel scenario (c) for IBA fuel scenario with 5 wt% Gd_2O_3 (d) for IBA fuel scenario with 9 wt% Gd_2O_3 (e) for hybrid fuel scenario with 5 wt% Gd_2O_3 (f) for hybrid fuel scenario with 9 wt% Gd_2O_3

2500 K temperature. The reason for this can be explained as follows. As the thermal neutrons formed in the reactor assemblies interact with the ^{235}U nuclei and create a fission reaction, the number of ^{235}U nuclei in the system decreases with increasing burn time, and as a result, the total mass of ^{235}U in assembly begins to decrease. The reason why the mass of ^{235}U starts to burn less as the temperature increases is due to the decrease in the interaction between the neutron and ^{235}U nuclei in

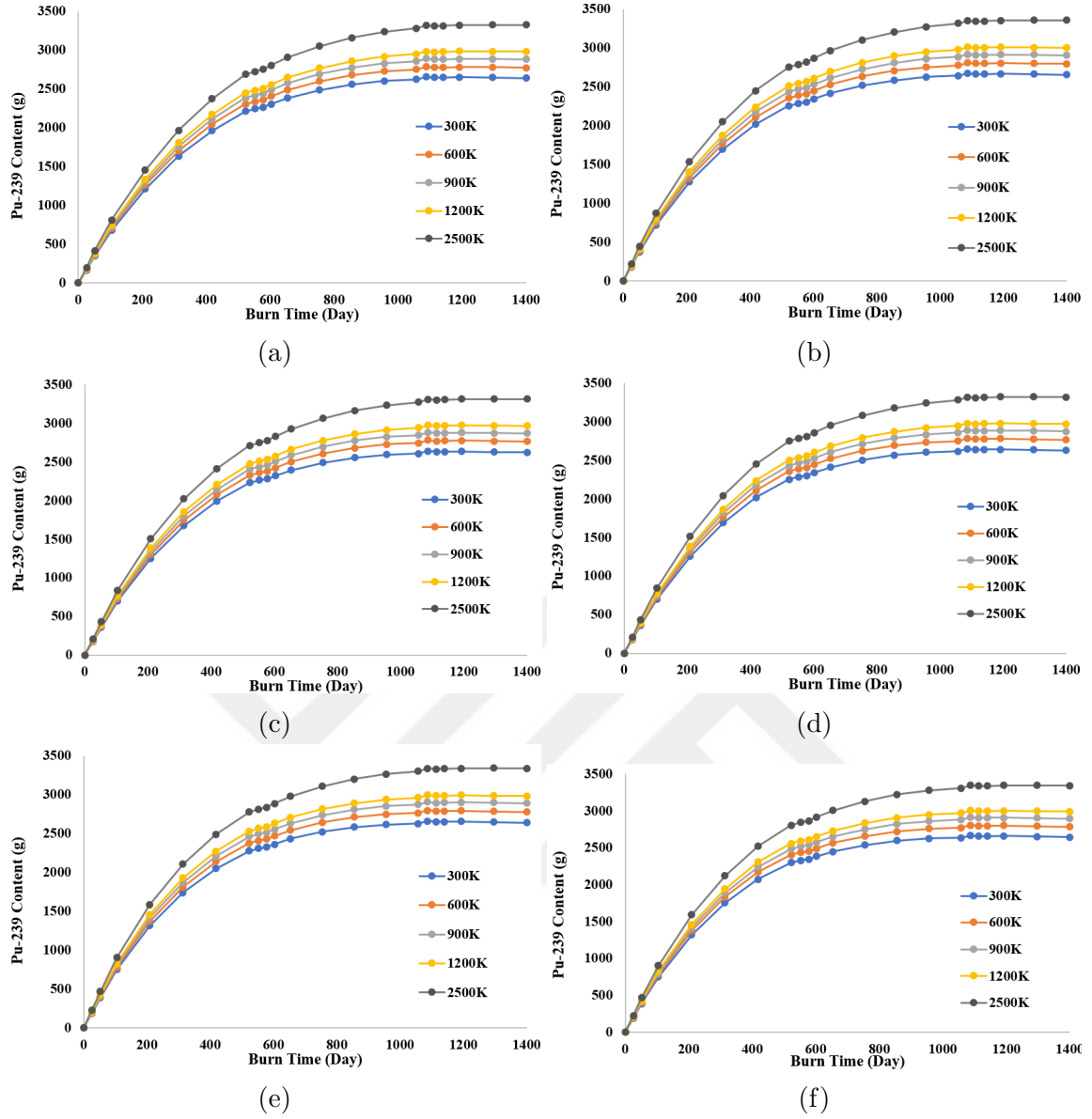


Figure 3.15 Comparison of change in ^{239}Pu mass with respect to burn time for VVER-1200 assemblies at different temperatures (a) for standard fuel scenario (b) for IFBA fuel scenario (c) for IBA fuel scenario with 5 wt% Gd_2O_3 (d) for IBA fuel scenario with 9 wt% Gd_2O_3 (e) for hybrid fuel scenario with 5 wt% Gd_2O_3 (f) for hybrid fuel scenario with 9 wt% Gd_2O_3

the system. As the system temperature increases, the water between the fuel rods, which allows high energetic neutrons formed as a result of fission to have thermal energy becomes less dense, and as the distance between the water molecules increases, the probability of interaction of fast neutrons and hydrogen atoms decreases. As a result, the thermal neutron population in the system decreases as the temperature increases, and the number of interactions with the ^{235}U nuclei in the system de-

creases. As a second reason, just like in water, thermal expansion occurs in the fuel material due to the increase in temperature, as a result of this, the mean free path of the neutron in the fuel material increases with the decreasing fuel density and the possibility of leaving the environment without interacting with the fuel material increases. For these reasons, the number of thermal neutrons that may be formed in the assembly may decrease with increasing temperature, the probability of the interaction between ^{235}U nuclei and formed thermal neutrons decrease, and the mass of ^{235}U burned in the system decreases as the temperature increases. Between the 522nd and 552nd days and between the 1057th and 1087th days, since both reactors were shut down, there is no change in the ^{235}U masses, and it can be seen that it has constant values between these time intervals.

Figures 3.12 and 3.13 show the variation of the mass of ^{238}U in the system with increasing burn time for different scenarios at 300 K, 600 K, 900 K, 1200 K and 2500 K temperatures of PWR and VVER-1200 assemblies, respectively. As can be seen from the figures, as the reactor operation time increases, the mass of ^{238}U decreases in all scenarios and at all temperatures, as it split into other nuclei as a result of high energetic neutron absorption. Looking at the variation of the mass of ^{238}U in the assemblies of both reactors with increasing temperature, it is observed that the total mass of ^{238}U in the assembly decreases as the temperature increases in all modeled and simulated scenarios and assemblies. If the mass of ^{238}U remaining in the assembly after 1400 days of operation is sorted from smallest to largest according to the assembly temperatures, it can be listed as 2500 K, 1200 K, 900 K, 600 K, 300 K. This situation is the opposite of the variation of the mass of ^{235}U with the increasing temperature. As the assembly temperature increases, the mass of ^{235}U remaining in the system as a result of the operation increases gradually, while the mass of ^{238}U decreases with increasing temperature. Since the water in the assembly undergoes thermal expansion with the increase in temperature and decreases in density, neutrons that are formed as a result of fission reach thermal energy more difficult and as a result they interact with ^{238}U instead of interacting with ^{235}U . This case leads ^{238}U to absorb more neutrons and form ^{239}U in the assembly. Since

the neutrons can have higher energies as the system temperature increases, the probability of interaction between the neutron and ^{238}U increases as the system temperature increases, and the mass of ^{238}U begins to decrease as the temperature increases. This situation was also observed as an event that reduced the criticality of the reactor. Between the 522nd and 552nd days and the 1057th and 1087th days when the reactor is shut down, the masses of ^{238}U take constant values since the fission reaction does not occur in the assemblies.

Figure 3.14 and 3.15 show the accumulation of ^{239}Pu mass in the system with increasing burn time for different scenarios of PWR and VVER-1200 assemblies at temperatures of 300 K, 600 K, 900 K, 1200 K, 2500 K, respectively. Based on the results can be seen from the figures, total mass of ^{239}Pu increases up to a point and then it gets saturated in all scenarios and assembly temperatures. The reason of the increase in the mass of the plutonium with the increasing burn time is the neutron capture of ^{238}U nuclei. Neutron capture, also known as breeding, is the process used in nuclear reactors to create ^{239}Pu . Fertile isotopes, like ^{238}U , can interact with neutrons released during fission reactions to transform them into ^{239}Pu . The reactor's ^{239}Pu concentration rises as a result of this breeding process. However, the subsequent fission reactions that ^{239}Pu goes through after being created result in the release of more neutrons. With the addition of these extra neutrons, ^{239}Pu is produced even more, starting a self-sustaining chain reaction. There are a number of factors that effect ^{239}Pu saturation over time as its concentration rises. The accumulation of other plutonium isotopes, particularly ^{240}Pu , is a significant factor. Compared to ^{239}Pu , ^{240}Pu exhibits a higher rate of spontaneous fission and neutron capture, which causes it to assemble in the reactor. As ^{240}Pu accumulates, it has an impact on the reactor's neutron economy and lowers the amount of neutrons available for producing more ^{239}Pu . Additionally, ^{239}Pu saturation may also be influenced by the buildup of fission products, such as different isotopes produced by the fission of uranium and plutonium. The amount of neutrons available for the production of ^{239}Pu is decreased by these fission products, which absorb them. As another reason, the buildup of other transuranic isotopes, such as ^{241}Am and ^{244}Cm ,

can effect the concentration of ^{239}Pu by competing with it for neutron absorption. Increase in the reactor temperature can also impact neutron leakage. Neutrons that escape the reactor assembly before causing fission reaction and neutron absorption reduce the overall effectiveness of plutonium production. As it was mentioned in introduction section, as the energy of the neutron increases, the non-leakage probability of the neutrons decrease. Higher temperatures may alter the neutron leakage pathways, potentially affecting the accumulation of ^{239}Pu . Neutrons that escape the reactor core before causing fission reactions reduce the overall effectiveness of plutonium production. Higher temperatures may alter the neutron leakage pathways, potentially affecting the accumulation of ^{239}Pu . Another result that can be seen figures is the accumulation of ^{239}Pu concentration in the assemblies of both PWR and VVER-1200 for all scenarios rises with increasing assembly temperature. If we compare the assemblies in terms of the accumulation of the ^{239}Pu concentration from large to small by looking the system temperatures, it can be listed as 2500 K, 1200 K, 900 K, 600 K and 300 K. The reason of increase in ^{239}Pu accumulation with increasing assembly temperature can be explained as follows. Increasing the temperature of a reactor assembly typically leads to an increase in neutron energy. High energetic neutrons have a greater probability of inducing neutron capture of ^{238}U nuclei, which can result in the production of more ^{239}Pu . Therefore, higher temperatures can promote the accumulation of ^{239}Pu through increased neutron-induced absorption rates. The temperature increase in the assemblies also effects neutron capture and absorption reactions in the reactor. At higher temperatures, the energy of the released neutrons as a result of the fission increases, enhancing the probability of capture by isotopes present in the reactor, including ^{239}Pu itself. These capture reactions reduce the number of available neutrons for the production of ^{239}Pu , potentially limiting its accumulation. The mass change of ^{235}U , ^{238}U , ^{239}Pu and ^{240}Pu isotopes of PWR and VVER-1200 assemblies at 600 K assembly temperature in the standard fuel scenario after 1400 days of operation time can be seen in figure 3.16.

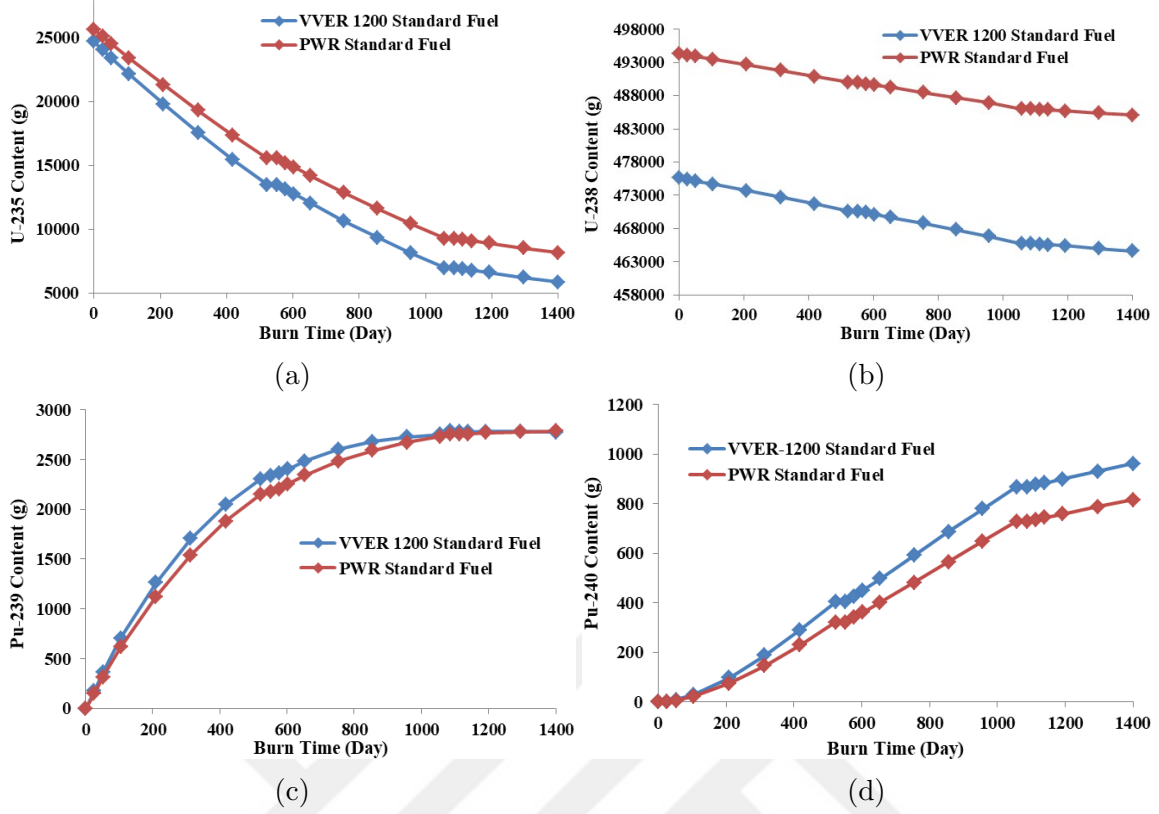


Figure 3.16 (a) Comparison of change in ^{235}U mass with respect to burn time in standard fuel scenario for PWR and VVER-1200 assemblies at 600 K temperature (b) Comparison of change in ^{238}U mass with respect to burn time in standard fuel scenario for PWR and VVER-1200 assemblies at 600 K temperature (c) Comparison of change in ^{239}Pu mass with respect to burn time in standard fuel scenario for PWR and VVER-1200 assemblies at 600 K temperature (d) Comparison of change in ^{240}Pu mass with respect to burn time in standard fuel scenario for PWR and VVER-1200 assemblies at 600 K temperature

In Figure 3.16 (a), the mass change of the ^{235}U isotope at 600 K assembly temperatures can be seen in the standard fuel scenario for PWR and VVER-1200. As can be seen, initially the total mass of ^{235}U in the PWR assembly is greater than in the VVER-1200 assembly. In both reactor assemblies, as the operation time increases, the mass of ^{235}U decreases due to the fission reactions. Since the reactor is shut down between the 522nd and 552nd days and the 1057th and 1087th days and the fission reaction does not occur between these time values, the mass of ^{235}U remains constant in the assemblies. The initial total mass of ^{235}U in the PWR assembly was 25690 grams and decreases to 8167 grams after 1400 days of operation. According to this result, the total mass of ^{235}U , which is reduced from the system as a result

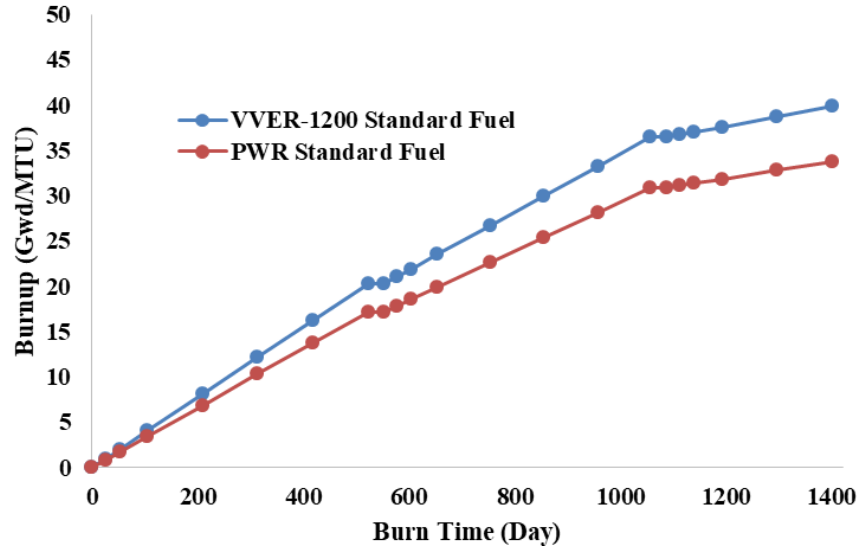
of fission in the PWR assembly, is 17523 grams. In the VVER-1200 assembly, while the initial total mass of ^{235}U is 24720 grams, this value decreased to 5883 grams after 1400 days of operation. Based on these results, it is investigated that the total mass of ^{235}U , which is reduced by fission from the VVER-1200 assembly, is 18837 grams. According to these results, it is seen that the total mass of ^{235}U burned in the VVER-1200 assembly is higher than the PWR assembly as a result of the 1400-day operation period. According to this result, it is one of our expectations that the burnup value of the VVER-1200 assembly is higher than that of the PWR assembly, and this expectation will be discussed in section 3.4.

In Figure 3.16 (b), the mass change of the ^{238}U isotope at 600 K assembly temperatures can be seen in the standard fuel scenario for PWR and VVER-1200 assemblies. As can be seen from the figure, the total mass of ^{238}U at the beginning of the operation time is greater in the PWR assembly than in the VVER-1200 assembly. The reason for this was discussed in section 3.3. In both reactor assemblies, the mass of ^{238}U decreases with increasing operation time, as the ^{238}U nuclei absorb the high-energy neutrons released as a result of fission and form other nuclei. While the initial mass of ^{238}U was 494300 grams in the PWR assembly, this value decreased to 485100 grams after 1400 days of operation. Looking at the mass changes, it was observed that the total mass of ^{238}U decreased after the operation in the PWR assembly is 9200 grams. In the VVER-1200 assembly, while the initial mass of ^{238}U is 475600 grams, this value decreased to 464600 grams after 1400 days of operation. Considering the mass changes, it was observed that the total ^{238}U mass decreased after the operation in the VVER-1200 assembly is 11000 grams. It can be seen that the ^{238}U mass reduction in the VVER-1200 assembly after operation is greater than in the PWR assembly. As a result of this, it is expected that the masses of ^{239}Pu nuclei, which are formed as a result of neutron absorption of ^{238}U , and ^{240}Pu nuclei, which are neutron absorption of ^{239}Pu nuclei, will be higher in the VVER-1200 assembly. Figures 3.16 (c) and 3.16 (d) show the masses of ^{239}Pu and ^{240}Pu isotopes formed in PWR and VVER-1200 assemblies for standard fuel scenario at 600 K assembly temperature. As it can be seen from the figures, since both nuclei

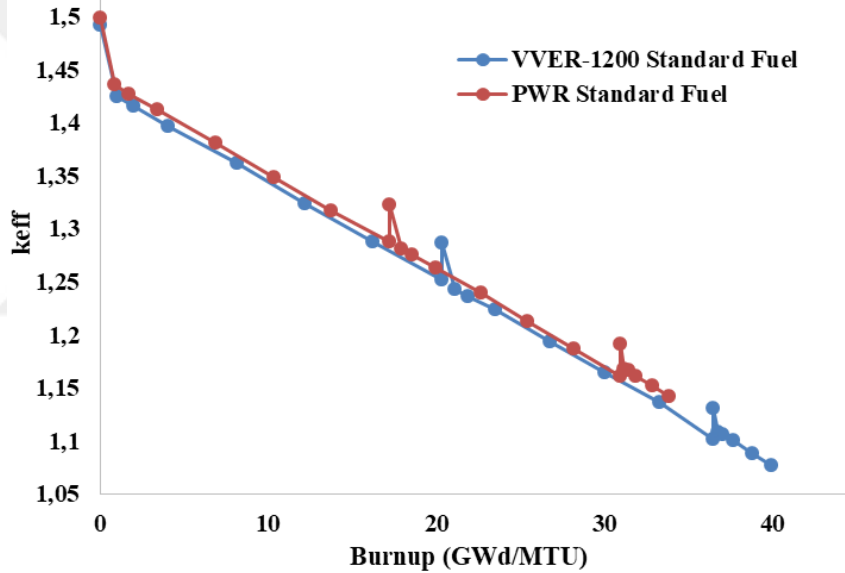
are not present in the assembly at the beginning of the operation time, their masses start from 0 grams and accumulate in the assembly over time with neutron-nuclei interactions. With the decreasing mass of ^{238}U , the mass of ^{239}Pu increases in both reactor assemblies, and the mass of ^{240}Pu also increases with the increasing mass of ^{239}Pu . Since the mass of ^{238}U burned in the VVER-1200 reactor is higher than that of the PWR assembly, it is observed that the mass of ^{239}Pu formed in the VVER-1200 assembly is higher than the PWR. In addition, more accumulation of ^{239}Pu nuclei in VVER-1200 assembly caused ^{240}Pu nuclei to build up more in VVER-1200 assembly than PWR.

3.4. Comparison of Burnup Values of PWR and VVER-1200 Assemblies

In this part, the burnup values of PWR and VVER-1200 assemblies during 1400 days of operation under standard fuel scenario and 600 K assembly temperature were compared. The reason to examine and compare the burnup values of the reactors is investigating the energy which can be obtained from both reactor assemblies. The investigation of reactor burnup offers valuable scientific insights into the operational behavior of nuclear fuel. Through the examination of the depletion of fissile material, researchers and experts are able to evaluate fuel utilization efficiency and enhance reactor performance. This understanding facilitates the determination of optimal fuel cycle durations, the design of fuel assemblies, and the advancement of innovative reactor technologies. Although we have burnup values for all scenarios and temperatures, only the standard fuel scenario at 600 K is compared in this thesis. Other results we have will be used in our future work. The burnup values of PWR and VVER-1200 assemblies during 1400 days of operation can be seen in figure 3.17 (a), and the change of effective neutron multiplication factor values with burnup can be seen in figure 3.17 (b).



(a)



(b)

Figure 3.17 (a) Comparison of burnup of PWR and VVER-1200 Assemblies for Standard Fuel Scenario at 600 K Assembly Temperature (b) Comparison of Change in Effective Multiplication Factor of PWR and VVER-1200 Assemblies with respect to Burnup for Standard Fuel Scenario at 600 K Assembly Temperature

Figure 3.2 (a) shows the burnup values of PWR and VVER-1200 assemblies during 1400 days of operation for the standard fuel scenario at 600 K assembly temperature. As can be seen from the figure 3.2 (a), as a result of the fission reactions taking place in the assemblies with the beginning of the operation, the burnup values increase over time. ^{235}U nuclei interacting with thermal neutrons perform fission

reaction, resulting in the release of other elements, neutrons and energy in the system. For this reason, as the fission reactions take place, the total mass of ^{235}U in the system decreases, with this decrease the amount of energy obtained from the fission increases and therefore the burnup value increases. For this reason, the burnup increase seen in figure 3.2 (a) is an expected result and therefore match with our expectations. Another result seen in Figure 3.2 (a) is that the burnup value of the VVER-1200 assembly is higher than that of the PWR assembly during the operation period. After 1400 days of operation, the burnup value of the VVER-1200 assembly is approximately 40 GWd/Mt, while the PWR assembly is approximately 34 GWd/Mt. In Section 3.3, the variation of ^{235}U mass amounts for both reactors at standard fuel scenario and 600 K assembly temperature was discussed. According to the result we found in section 3.3, it has been observed that the decrease in total mass of ^{235}U in the VVER-1200 assembly was higher than in the PWR assembly. The fact that the total mass of ^{235}U in the VVER-1200 assembly has decreased more than the PWR assembly indicates that more fission occurs in the VVER-1200 assembly, and accordingly, more energy is released in the VVER-1200 assembly. For this reason, when the changes in the mass of ^{235}U and the burnup values are examined, it has been observed that the two results are in agreement with each other. Figure 3.17 (b) shows the comparison of the effective neutron multiplication factor values of the PWR and VVER-1200 assemblies in standard fuel scenario at 600 K assembly temperature with increasing burnup. Upon analyzing the figure 3.17 (b), it is evident that the effective neutron multiplication factor values for both PWR and VVER-1200 reactors are distributed at distinct points, which do not intersect as the burnup increases. This disparity can be attributed to the varying burnup values between the two reactor assemblies during their operational period. While initially, the criticality values for both assemblies coincide, the changing burnup values and their corresponding criticality values evolve over burn time. By examining the peaks observed during reactor shutdown period between 522nd-552nd days, it can be noted that the PWR assembly has a burnup of approximately 18 GWd/Mt, whereas the VVER-1200 assembly has a burnup of about 20 GWd/Mt. These peaks were observed at the same point for both reactor assemblies in terms of the change

in effective neutron multiplication factor with burn time. However, in the graph, the effective neutron multiplication factor values do not overlap due to the differing burnup values at the time of shutdown for the two reactor assemblies. This disparity is not only evident at shutdown but also persists during burnup, which increases with burn time. By comparing the burnup values corresponding to the effective multiplication factor values of the VVER-1200 assembly and the PWR assembly, it can be concluded that the VVER-1200 assembly exhibits a higher burnup value than the PWR assembly. As can be seen from the figure 3.17 (b), it has been observed that the criticality values of both reactor assemblies decreased gradually as the burnup value increased and the mass of ^{235}U in the system decreased. Since the decreasing ^{235}U mass from the system causes a decrease in the total mass of fissile material in the system, the criticality values decreased in each generation. According to the results obtained, although the burnup value of the VVER-1200 assembly, in other words the energy released as a result of the interaction of thermal neutrons and ^{235}U nuclei, is higher than the PWR assembly, the criticality value of the VVER-1200 assembly in the standard fuel scenario is lower than the PWR assembly. We argue the reason for this result as follows. The thermal neutron and high-energetic neutron fluxes in the VVER-1200 assembly are more than in the PWR assembly, so even though the thermal neutrons cause more fission in the VVER-1200 assembly, the number of neutrons absorbed by the ^{238}U is also higher in the VVER-1200 reactor. For this reason, as it will be remembered from equation 1.5, since the number of neutrons absorbed by ^{238}U instead of fissile ^{235}U nuclei in the system increases, the criticality values of the VVER-1200 assembly becomes less than the PWR assembly. We think that the decrease in the total mass of ^{235}U obtained for the PWR and VVER-1200 assemblies in section 3.3 is one of the consequences of the thermal neutron flux in the VVER-1200 assembly being greater than that of the PWR assembly. In addition, another conclusion that we made is the decrease in the total mass of ^{238}U and the increase in the mass of ^{239}Pu found in section 3.3 for the PWR and VVER-1200 assemblies are also one of the results of the high energetic neutron flux in the VVER-1200 assembly is being greater than the PWR assembly. As it will be remembered from Section 3.3, when the ^{239}Pu and decreasing

^{238}U masses formed in the PWR and VVER-1200 assemblies at 600 K temperature in the standard fuel scenario were examined, it has been observed that there was a greater decrease in ^{238}U in VVER-1200, and accordingly a further increase in the ^{239}Pu mass. Accordingly, given that VVER-1200 has more reduction in total mass of ^{238}U and more accumulation in ^{239}Pu mass after 1400 days of operation, there must be more high energetic neutron flux in the VVER-1200 assembly. For this reason, the criticality value of the VVER-1200 assembly is less than the PWR assembly.

3.5. Average Burnup Value of TRIGA Mark II Reactor Fuel Element

Finally, in this part, the average burnup rate for the fuel element of the TRIGA MARK II reactor at Istanbul Technical University will be discussed. The fuel element is modeled and simulated according to the parameters mentioned in section 2.4. At the end of the total 50.25 day reactor operation period, the critical mass change is found by looking at the critical mass at the beginning and at the end of the operation, and the total grams of ^{235}U burned in the system is determined using the MCNP 6.1 program. Along with the critical mass, the amount of ^{239}Pu and neutron poison ^{135}Xe formed in the fuel element after the operation is also obtained. In addition to the ^{235}U burned in the system, the change in ^{238}U mass is also found. The variation of the mentioned isotopes can be seen in table 3.7.

Isotope	Total mass (g)	
	at the beginning	at the end
^{235}U	38	37.57
^{238}U	152.9	152.8
^{239}Pu	0	0.1267
^{135}Xe	0	1.089×10^{-4}

Table 3.7 Change in the Mass of Different Isotopes in TRIGA Mark II Fuel Element After the Operation

The change in isotope masses for the TRIGA Mark II fuel element at the start of operation and at the end of operation can be seen in table 3.1. The initial critical mass of the modeled fuel element is 38 grams, while the mass of ^{238}U is 152.9 grams. At the end of the operation period of 50.25 days, a decrease in the mass of uranium isotopes was expected due to the occurring neutron and uranium reactions inside the fuel element. The total mass of ^{235}U , which is called the critical mass, will decrease due to fission by interacting with the thermal neutrons, while ^{238}U will interact with high-energy neutrons formed as a result of fission that cannot reach the thermal energy level to form ^{239}Np , and ^{239}Np will beta decay to form ^{239}Pu . For this reason, it is expected that the total mass of ^{238}U in the system will decrease and the mass of ^{239}Pu will increase as a result of 50.25 days of operation. In addition, the mass of the ^{135}Xe , so called neutron poison and formed as fission product of ^{235}U , is expected to increase in the system. The mass of ^{135}Xe is expected to be 0 gram as no fission reaction takes place in the system before the reactor starts operating. As a result of the fission reactions occurring during the operation period, the mass of ^{135}Xe will also increase. As seen in Table 3.2, while the initial mass of ^{235}U was 38 grams, this mass decreased to 37.57 grams after 50.25 days of operation. From this result, it was determined that the decrease in the critical mass of the fuel element of ITU TRIGA Mark II fuel element is 0.43 grams. Looking at the change in the mass of ^{238}U , it is seen that it is quite low when compared to the change in the mass of ^{235}U . While the mass of ^{238}U was 152.9 grams at the beginning, it decreased to 152.8 grams after 50.25 days of operation. Due to this slight decrease in ^{238}U mass, the increase in ^{239}Pu mass is also low. Since there is no interaction between high energetic neutrons and ^{238}U at the beginning of the operation and the fuel material does not contain ^{239}Pu nuclei, the total mass of ^{239}Pu in the system is 0 grams. After 50.25 days of operation, the mass of ^{239}Pu is found to be 0.1267 grams. Due to the low interactions between high energetic neutrons and ^{238}U in the system, the mass of ^{238}U did not decrease much, and the total mass of ^{239}Pu formed after the operation is quite low. When the mass of ^{135}Xe formed in the fuel element as a result of 50.25 days of operation is examined, it is obtained as 1.089×10^{-4} grams. It can be said that the reason why the total mass of formed ^{135}Xe is quite low in the

fuel element where the mass of ^{235}U decreases by 0.47 grams, is due to the fact that the probability of formation of ^{135}Xe as a result of fission is very low, as mentioned in the introduction part.



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