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Neutronic analysis of mixed H_2O/D_2O moderated SMART reactor fuel assembly with varying fractions of D_2O during the fuel burnup

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Anahtar Kelimeler

SMART Reaktörü
Yanma Oranı
Spektrum
Yavaşlatıcı
Serpent

Graphical/Tabular Abstract (Grafik Özet)

The present work investigates how the mixed heavy/light water moderator affects the cycle length of a fuel assembly. / Bu çalışma, karışık ağır/hafif su yavaşlatıcısının yakıt demeti çevrim süresi üzerindeki etkisini araştırmaktadır.

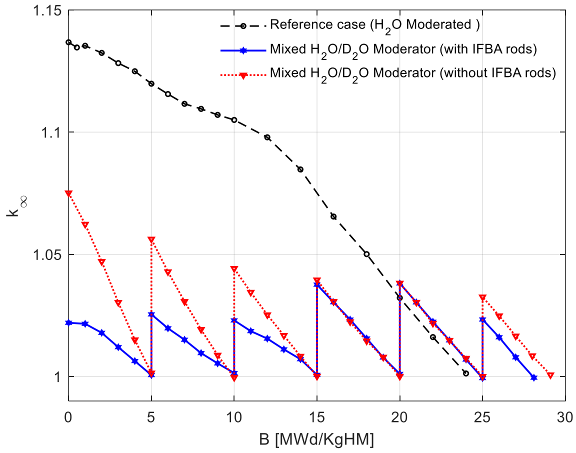

***Figure A**: Extension of the cycle length by changing the heavy water fraction during the burnup /**Şekil A:** Yanma sırasında ağır su oranını değiştirerek çevrim süresinin uzatılması.*

Highlights (Önemli noktalar)

- *The mixed heavy/light water mixture is considered as the moderator. / Karışık ağır/hafif su karışımı moderatör olarak dikkate alınır.*
- *The volume fraction of heavy water in each burnup-step is calculated. / Her yanma adımındaki ağır suyun hacim oranı hesaplanır.*
- *Fuel mass saving is calculated. / Yakıt kütlesi tasarrufu hesaplanır*

Aim (Amaç): *This work aims to investigate the effects of mixed heavy/light water moderator, with varying fractions of heavy water during the burnup, on the burnup performance of the SMART reactor's fuel assembly. / Bu çalışmada, yanma sırasında farklı oranlarda ağır su içeren karışık ağır/hafif su yavaşlatıcının, SMART reaktörünün yakıt demetinin yanma performansı üzerindeki etkileri araştırılmıştır.*

Originality (Özgünlük): *Calculation method of heavy water volume fraction in each burnup step is provided; and by implementing the proposed method on an assembly containing $(Th+U)O_2$ fuel, the UO_2 mass saving is calculated. / Her yanma adımındaki ağır su hacim oranının hesaplaması için gereken yöntem sağlanır; önerilen yöntemi $(Th+U)O_2$ yakıtı içeren bir düzenekte uygulayarak, UO_2 kütle tasarrufu hesaplanır*

Results (Bulgular): *The neutron spectrum is shifted to the resonance region at the beginning of the cycle wherease toward the end of the cycle, by increasing the light water fraction, the neutron spectrum becomes softer. / Nötron spektrumu çevrimin başlangıcında rezonans bölgesine kayarken, çevrimin sonuna doğru hafif su oranı arttıkça nötron spektrumu daha yumuşak (termal) hale gelir.*

Conclusion (Sonuç): *The cycle burnup is extended by almost 21%. Unlike the light water moderated reactors there is a highr conversion ratio at the beginning of the cycle. / Çevrimin yanma süresi neredeyse %21 oranında uzatılır. Hafif su yavşaltıcılı reaktörlerin aksine, çevrimin başlangıcında daha yüksek bir dönüşüm oranı görünmektedir.*

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Neutronic analysis of mixed H_2O/D_2O moderated SMART reactor fuel assembly with varying fractions of D_2O during the fuel burnup

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Abstract

The neutronic behavior of nuclear reactors is being investigated by considering different fuel, cladding, and neutron-moderating materials. In the present manuscript, two different assembly types of the Korean system-integrated modular advanced reactor with different enrichments and different numbers of integral fuel burnable absorber fuel rods are considered; and the effects of mixed heavy/light water moderator, with varying fractions of heavy water during the burnup, on the assembly cycle burnup are investigated. It is observed that, to extend the cycle burnup, it is required to use a higher fraction of D_2O at the beginning of the cycle whereas it reduces toward the end of the cycle. A higher fraction of heavy water causes the neutron spectrum to shift to the resonance region, resulting in a higher capture rate of the fertile materials. This, in turn, causes an increase in the conversion ratio. On the contrary, toward the end of the cycle, by increasing the light water fraction, the neutron spectrum becomes softer. This also causes an increase in the fission rate of fissile materials. Finally, a certain improvement in the cycle burnup is observed. Moreover, by implementing the proposed method on an assembly containing $(Th+U)O_2$ fuel, the UO_2 mass saving is calculated.

Yakıt yanma sırasında farklı oranlarda ağır su içeren karışık H_2O/D_2O yavaşlatıcılı SMART rektörünün yakıt demetinin nötronik analizi

Makale Bilgisi

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Öz

Nükleer reaktörlerin nötronik davranışı, farklı yakıt, zarf ve nötron yavaşlatıcı malzemeler dikkate alınarak araştırılmaktadır. Bu çalışmada, farklı zenginliklere ve farklı sayıda yanabilen soğurucu yakıt çubukları içeren Kore SMART reaktörüne ait iki farklı demet tipi ele alınmıştır; ve yanma sırasında değişen ağır su oranına sahip karışık ağır/hafif su yavaşlatıcısının demetin yanma çevrimi üzerindeki etkileri araştırılmıştır. Çevrimin yanma süresini uzatmak için çevrim başlangıcında daha yüksek bir D_2O oranının kullanılmasının gerektiği ve bu oranın çevrimin sonuna doğru azaldığı görülmektedir. Ağır suyun yüksek oranı, nötron spektrumunun rezonans bölgeye kaymasına neden olur ve bu da doğurgan maddelerin daha yüksek bir nötron yakalama hızıyla sonuçlanır. Bu da sonuç olarak dönüşüm oranında bir artışa neden olur. Aksine, çevrim sonuna doğru, hafif su oranının artmasıyla nötron spektrumu daha yumuşak hale gelir. Bu aynı zamanda fisil malzemelerin fisyon hızında da bir artışa neden olur. Son olarak çevrim yanma oranında belirli bir iyileşme gözlemleniyor. Ek olarak, önerilen yöntemi $(Th+U)O_2$ yakıtı içeren bir düzenekte uygulayarak, UO_2 kütle tasarrufu hesaplanır.

1. INTRODUCTION (GİRİŞ)

Fuel material used in nuclear reactors is a mixture of fissile and fertile materials known as Heavy Metal (HM). As the result of the fission event a certain value of thermal energy and some radiations (e.g., neutrons and gamma-rays) are released. Specific fuel burnup (or Burnup) is denoted by BU and defined as the total generated thermal energy due to the fission events per kg or metric ton of initial Heavy Metal (HM) loaded [1-4].

$$BU = \frac{\int_0^T P(t)dt}{m(HM)} = \frac{P_{av}.T}{m(HM)} \quad \left[\frac{MW-d}{kg(HM)}\right] \qquad (1)$$

Where $P(t)$ represent the generated thermal power at time t, m is the mass of the heavy metal, and

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Nomenclature	
y	Fission yield (number of fission fragment generated per fission event)
BU	Burnup
$P(t)$	Time-dependent thermal power
$m\ (HM)$	mass of heavy metal.
T	Time in day.
$\bar{\varphi}$	The average neutron flux
λ	Decay constant
σ_a	Microscopic absorption cross section
Σ_f	Macroscopic fission cross section
N	Atom number density
MOX	Mixed oxide fuel
UO_2	Uranium dioxide
ThO_2	Thorium dioxide
Pu	Plutonium
D_2O & H_2O	Heavy and light water.
Gd_2O_3	Gadolinium oxide
Abbreviations	
SMART	Korean system-integrated modular advanced reactor
BOC	Beginning of the cycle
EOC	The end of the cycle
IFBA	Integral fuel burnable absorber

$P_{av} = \int_0^T P(t)dt \,/\, \int_0^T dt$ is the average thermal power generated during the T days.

During the reactor operation, the fission fragments and non-fissioned fuel materials are irradiated by both neutron and gamma radiations and may undergo different types of induced reactions. These materials may experience different possible radioactive decay reactions as well. Fission fragment and their progenies (formed due to radioactive decay) are called fission products. By fuel material burnup both buildup and loss of fission products take place, in other words, fuel composition is changed over time. Some of the fission products such as $Xe-135$ and $Sm-149$ have considerable neutron capture cross sections and subsequently harm the neutron economy within the system and are known as burnable poisons. After a few hours of the nuclear reactor's startup, due to the buildup of these two burnable poisons, a sudden drop in system reactivity is observed. It should be noted that the densities of these two elements finally reach their equilibrium values. Fission products have a great effect on system reactivity, power distribution, delayed neutrons' parameters, and decay heat of fuel. Hence, the time rate of change of their atom numbers or concentrations should be taken into consideration [3,4].

If $N_i\ (t)$ be the number density of any material (either fissionable or fission products), the time rate of change of $N_i\ (t)$ is calculated as the production rate minus the loss rate. Production rate is the summation of productions due to: Fission, probable radioactive decay reactions, and probable radiation induced reactions. Loss rates also are due to both radioactive decay and radiation induced reactions [4-8].

$$\frac{dN_i(t)}{dt} = \sum_k y_{i_k}\Sigma_{f_k}\bar{\varphi} + \sum_j \left(\lambda_j + \sigma_{a_j}\bar{\varphi}\right)N_j(t) - \left(\lambda_i + \sigma_{a_i}\bar{\varphi}\right)N_i(t) \quad (2)$$

where y_{i_k} is the number of $i'th$ atom generated per fission of $k'th$ isotope, $\bar{\varphi}$ represents the average neutron flux within the system, $\lambda_j N_j(t)$ and $\sigma_{a_j}\bar{\varphi}N_j(t)$ are the production rates due to radioactive decay or neutron absorption of any $j'th$ isotope, respectively. Moreover, the considered material may experience a loss due to probable decay or neutron absorption reactions.

There are a huge number of elements within the fuel (due to fuel materials burning up, and other materials building up) and their numbers change as time passes. All of these come together to form a non-linear system of coupled equations, known as Bateman equations. To track the density of any isotopes within the fuel, it is required to solve the Bateman equations [8].

High majority of operating nuclear reactors are Light Water Reactors (LWR) which use water as both moderator and coolant [9]. The fission neutrons are generally fast with an average energy of $2\,MeV$. Since the microscopic fission cross section of thermal neutrons is higher than those of fast neutrons, the moderator is used to moderate the fission fast newborn neutrons to the thermal region through multiple scattering events. A good moderator must thermalize the fast neutron with a relatively small number of scatterings and also have a huge scattering cross section in comparison with capture [10-11]. Slowing-down power is defined as the production of average logarithmic energy loss and macroscopic scattering cross section. A greater slowing-down power means neutrons are more effectively moderated. This parameter cannot be used as a criterion for selecting the best moderator. Moderating ratio is defined as the slowing-down power divided by the macroscopic absorption cross section of the material. The best moderator, in turn, has a great moderating ratio. Light water has a greater slowing-down power in comparison with Heavy water, in contrast, heavy water is a better moderator than light water [10-12]. The neutron spectrum for a typical (Pressurized Water Reactor) PWR unit cell with H_2O and D_2O are plotted in Figure 1 and compared with each other. It is seen that due to the higher slowing-down power of H_2O, the spectrum can be considered thermal. Due to the higher fission cross section of $U-235$ at thermal energies, there is an excess reactivity at the Beginning of the Cycle (BOC), which is not desired from the fuel economy point of view.

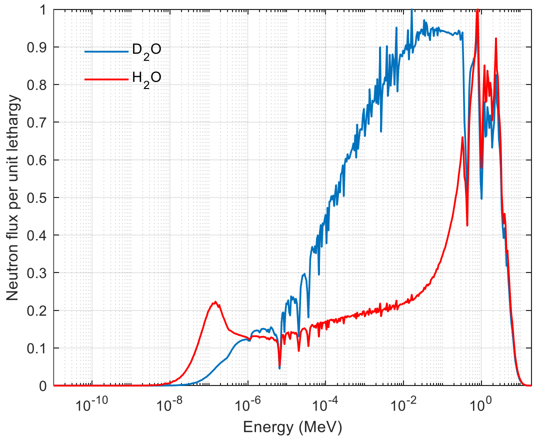


Figure 1. Neutron spectrum of 2D PWR pin-cell with two different coolants. (İki farklı soğutucuya sahip 2B PWR pin hücresinin nötron spektrumu.)

Fuel utilization and all changes in fuel composition during the reactor operation affect the nuclear power economy. It is also desired to get more thermal energy from a certain amount of loaded fuel, meaning that, it is longed to extend the cycle burnup (or cycle length). In the case of small modular and large-scale LWRs, this aim has been pursued by several authors in the literature by considering the different fresh fuel compositions, cladding materials, and moderator mixtures [13-26]. For instance, thorium and transuranic mixed fuels have been considered an alternative to UO_2 fuel elements in small modular reactors. It has also been shown that usage of the suggested fuel mixture doubles the cycle length and reduces the system's initial excess reactivity [13]. Prianka et. al. investigated the neutronic analysis of the SMART reactor considering the uranium-nitride and uranium di-silicide accident-tolerant fuels and compared them with UO_2 fuels. They also showed that using the uranium-nitride increases the cycle length more than the other evaluated fuels [15]. The neutronic properties of UO_2-BeO fuel with $FeCrAl$ and SiC claddings were investigated by Chen et. al. [18]. They reported that, in the case of the $UO_2-BeO-FeCrAl$ fuel-cladding system, the xenon and krypton production rates are lower. Semi-heavy water and H_2O coolants' effects on the neutronic parameter of a research reactor were investigated by G. Rahimi and et. al. [21]. According to the obtained results, they concluded that Semi-heavy water increases thermal neutron flux, improves the axial and radial power peaking factors, and hence can be used as a coolant instead of H_2O. Light water and mixed D_2O+H_2O moderators were also considered as candidate coolants for a civil marine reactor containing both solid and duplex fuel rods and their effects during the fuel burnup were investigated [22,23]. It was shown that the mixed coolant provides excellent core lifetimes comparable to those of high-enriched uranium military naval vessels while utilizing low-enriched uranium candidate fuels. Elzayat et. al. considered the H_2O/D_2O mixture as the candidate moderator for the UO_2-fueled VVER 1000 assembly and showed that the fuel utilization was improved by 60% [25]. Single-batch neutronic analysis of a D_2O+H_2O moderated SMR reactor with two different fissile loadings (i.e., 5% and 15%) was investigated by Lindley et. al. It was observed that natural uranium utilization experienced an increase between 39% and 47% in comparison to the reference benchmark [26].

In the present study, it is tried to investigate the effect of mixed heavy/light water on the cycle length, conversion ratio, and fuel depletion behaviors of two different assemblies of the SMART reactor. In line with this goal, by considering burnup steps of 5 $Mwd/kgHM$, in each burnup step the volume fraction of heavy water is calculated using the proposed methodology in Section 4. The simulation is also continued step by step using the restart feature of the Serpent Monte Carlo code. The results obtained are compared with reference H_2O moderated assemblies' results. In addition, the effect of the presence of the burnable absorbers using the suggested method is also assessed. In the last section, the method is applied to Th-fueled assembly. In which, by applying the suggested method the fraction of ThO_2 in $(U+Th)O_2$ fuel is such calculated that the cycle burnup becomes equal to the cycle burnup of the reference assembly, and then the UO_2 mass saving due to the mixed H_2O/D_2O moderator usage is calculated.

2. MONTE CARLO SIMULATION METHOD
(MONTE CARLO SİMÜLASYON YÖNTEMİ)

The Monte Carlo method is a stochastic simulation method and is extensively used to solve physical problems. The only requirement is to define probability distribution functions (PDFs) that describe stochastic processes of the physical system. In this method, neutron transport is monitored by random sampling of neutrons' path length, direction, energy, and probable collision type [27,28]. In the criticality problems, by considering a certain number of neutrons (histories), some randomly distributed neutrons with fission energy spectrum are generated in the first cycle. At the end of the cycle, the multiplication factor is tallied as shown in Equation 3. Subsequently, the neutron transport in the next cycle is monitored. Some of the initial cycles are ignored, these cycles are called passive cycles. The purpose of these passive cycles is to ensure that the distribution of neutrons released from fission converges to the fundamental mode. The remaining cycles are called the active cycles. The effective amplification factor is also calculated as the average of the amplification factors of these cycles [27-30].

$$\mathrm{K_{cycle}} = \frac{\text{Total number of neutrons generated in the } \mathrm{n^{th}} \text{ generation}}{\text{Total number of neutrons generated in the } \mathrm{(n-1)^{th}} \text{ generation}} \tag{3}$$

For each homogeneous subregion, the average neutron scalar flux is tallied by using either path-length or collision estimators. For instance, Equation 4 represents the either path-length estimation method [31]:

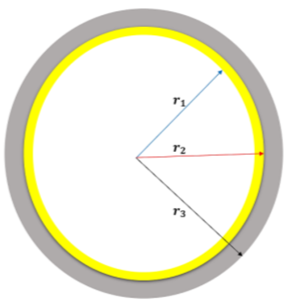
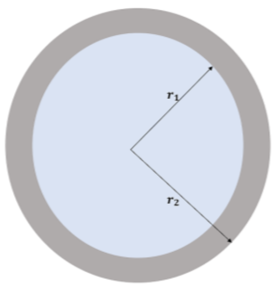
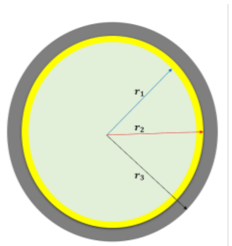
$$\bar{\varphi} = \frac{\text{The sum of the total path lengths traveled by neutrons}}{\text{Volume}} \tag{4}$$

The calculated flux is used in the calculation of the reaction rate and then used in the solution of Bateman equations.

3. REFERENCE PROBLEM DESCRIPTION
(REFERANS PROBLEM TANIMI)

For this study, the two different assemblies of the Korean SMART reactor are taken into account. The SMART reactor is a multi-purpose small modular reactor with 330 MW nominal thermal power and average specific power of 23.079 kW/kgU. In conjunction with electrical power generation, the generated thermal power can be used in the desalination of seawaters and process heat in industries. It contains 57 fuel assemblies with an assembly pitch of 21.504 cm. Each assembly, in turn, is designed as a 17-by-17 square lattice form. There are 264 fuel rods, one central instrumentational tube, and 24 control rod guide tubes in each assembly. The active height of the fuel rods is 200 cm and fuel rod pitch is equal to 1.2598 cm and the fuel enrichment is less than 5 $w/0$. Radial views and dimensions of fuel rods, empty guide tubes, and control rods are presented in Table 1 [13-14, 32-34].

Table 1. Radial views and dimensions of fuel rod, guide tube, and control rod. (Yakıt çubuğu, kılavuz tüpü ve kontrol çubuğunun radyal görünümleri ve boyutları.)

	 Fuel rod		 Guide tube		 Control rod	
Region	**Material**	**Dimensions (cm)**	**Material**	**Dimensions (cm)**	**Material**	**Dimensions (cm)**
1	Fuel	0.40960	water	0.56150	Ag-In-Cd	0.43305
2	He	0.41875	Zr-4	0.61200	He	0.43690
3	Zr-4	0.47500	-	-	SS-304	0.48380

To compensate for the BOC excess reactivity, some of the fuel rods comprise a mixture of Gd_2O_3 and UO_2; and are known as Integral Fuel Burnable Absorbers (IFBA) fuel rods. According to the number of IFBA rods and fuel enrichment, there are six types of assembly in the SMART core [13, 32]. In this manuscript, two different assembly types are considered to perform the assembly-level neutronic analysis. The schematics of the considered assemblies are depicted in Figures 2 and 3.

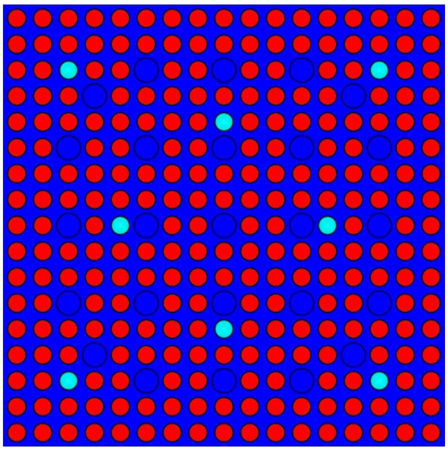


Figure 2. Assembly Type 1 (1. Tip demet)

Fuel Enrichment ($w/o\ U-235$): **2.82 w/0**
Number of IFBA rods: **8**
Gd_2O_3 content in IFBA rod (w/0): **8 w/0**

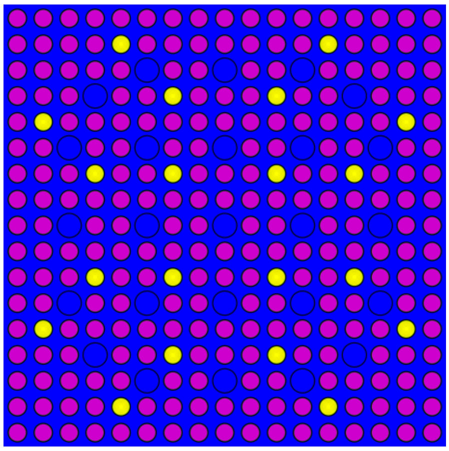


Figure 3. Assembly Type 2 (2. Tip demet)

Fuel Enrichment ($w/o\ U-235$): **4.88 w/0**
Number of IFBA rods: **20**
Gd_2O_3 content in IFBA rod (w/0): **8 w/0**

4. METHOD OF WORK (ÇALIŞMA YÖNTEMİ)

Burnup calculation of the considered reference study cases is performed by using Serpent 2.1.30 Monte Carlo code [35,36]. To reduce the computational time cost the geometry is prepared with 1/8 symmetry, and the 2D geometry is subjected to reflective boundary condition. That is, if a neutron of direction cosine μ exits the system at the boundary, a new neutron with direction cosine of $-\mu$ enters the system. Since, in this work, it is dealt with a single batch comparative study, the leakage effect on the system reactivity is not considered, that is, the cycle burnup is the burnup value corresponding to $k_{inf}=1$. In burnup calculation, each UO_2 fuel rod's pellet region is considered as a single depletion zone. To compensate self-shielding effect, each IFBA rod's pellet region is divided into ten annular (radial) depletion zones with equal volume. The volumes of depletion zones are also calculated using the -checkvolumes command. Variations of k_{inf} during burnup for the reference assemblies are shown in Figures 4 and 5.

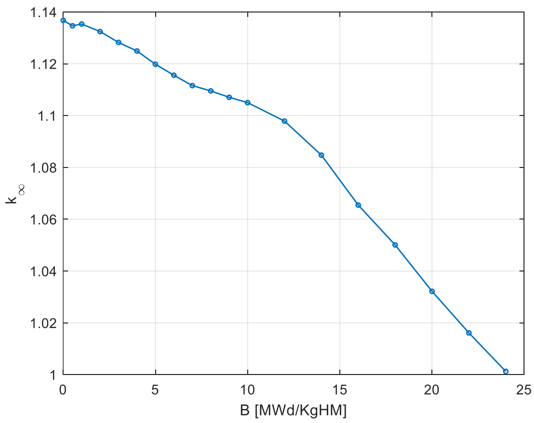


Figure 4. Infinite multiplication vs. burnup factor for assembly Type 1 (1.tip demet için sonsuz çoğalma faktörünün yanma oranıyla değişimi)

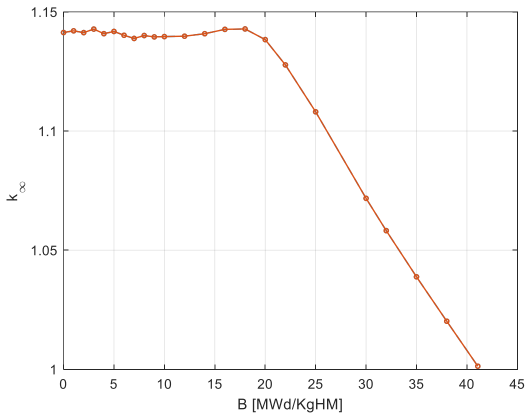


Figure 5. Infinite multiplication factor vs. burnup for assembly Type 2 (2.tip demet için sonsuz çoğalma faktörünün yanma oranıyla değişimi)

The cycle burnups corresponding to assembly types 1 and 2 are calculated as $24\,MWd/kgHM$ and $41.08\,MWd/kgHM$, respectively. It is should be noted that Xe-135 and Sm-149 equilibrium concentrations are set using the “set xenon 1” and “set samarium 1” commands. Monte Carlo simulation is performed by setting the fuel material and non-fuel materials’ (clad and coolant) temperatures to $900\,K$ and $600\,K$, respectively. Meaning that the system is operated at Hot Full Power (HFP) condition. In addition, neutron history, passive cycles, and active cycle numbers are set to 25000, 50, and 150, respectively. That is, the initial 50 cycles are devoted to the convergence of the flux to its fundamental mode, and the multiplication factor of the system is calculated as the average of the multiplication factors of the active cycles.

To investigate the effect of mixed heavy/light water moderator on the burnup analysis (cycle burnup), for different burnup steps with an increment of $5\,Mwd/kgHM$ the following are done:

In the first step, the burnup calculations are performed for different volume fractions of heavy water in the moderator (starting from 100 % with a 10% decrement). Using the obtained multiplication factors corresponding to each volume fraction, the optimum fraction of the heavy water to have a critical system at the end of the burnup step (here, $5\,MWD/kgHM$) is calculated using the linear interpolation method. For the calculated volume fraction, the burnup calculation is performed in the first step. By setting the "set inventory all" and "set rfw" restart file commands [35,37], the depleted fuel compositions at $5\,MWD/kgHM$ burnup step are written into a binary restart file and used in the following step calculations.

In the second step, using the depleted fuel compositions from the preceding simulation, for different volume fractions of D_2O (starting from the fraction calculated in the previous burnup step with a decrement of 10%) the burnup simulations are performed. Using the obtained results, the optimum volume fraction which satisfies the system criticality at $10\,MWD/kgHM$ is calculated. Similar to the previous step, the burnup simulation for the calculated fraction is done and depleted fuel compositions are stored to use in the next step calculation.

This procedure is pursued in the next steps. In the final step, the volume fraction of heavy water is set to zero and the burnup value corresponding to $k_{inf} = 1$ is calculated which is also the cycle burnup.

5. RESULTS AND DISCUSSION (BULGULAR VE TARTIŞMA)

In this section, the proposed simulation method is implemented on the considered reference assemblies, and the effect of usage of heavy water on the cycle length and its reasons are investigated. The effect of the presence of burnable poison materials (IFBA rods) is also analyzed. Finally, by implementing the developed simulation method on an assembly containing $(Th-U)O_2$ fuel material, the possibility of usage of Thorium is assessed.

5.1. Case1: Assembly Type 1 (Durum 1: 1. DEMET TİIPİ)

In this section, the proposed methodology is applied to assembly type 1. Furthermore, to investigate the effect of burnable absorber materials when using mixed D2O/H_2O moderator, IFBA rods in the reference setup are replaced with UO_2 rods and taken as another study case. Variations of multiplication factors during the burnup for three considered cases are plotted in Figure 6. In comparison with the $24\,MWD/kgHM$ cycle burnup of the H_2O moderated assembly, the cycle burnups of the assembly with mixed D2O/H_2O moderator in the presence and absence of the IFBA rods are calculated as $28.1\,MWd/kgHM$ and $29.1\,MWd/kgHM$, respectively. These, in turn, show a 17.08% and 21.25% improvement (extension) in cycle burnup. Volume fractions of the heavy water in the moderator are also presented in Figure 7. It is seen that this fraction goes down with the increase in burnup.

For the D_2O/H_2O moderated system containing IFBA fuel rods, burnup-dependent changes of Gd-155 and Gd-157 isotopes are shown in Figures 8 and 9, and compared with those of H_2O moderated assembly. It is seen that their number densities experience rapid decrease (due to their higher absorption cross section) and finally reach equilibrium with negligible atom number densities. This equilibrium value, in turn, is due to the low-yield production of the Gd-isotopes as the result of the fission event. As clearly seen in Figure 6, up to almost $15\,MWd/kgHM$ burnup step, the excess reactivities of the mixed D_2O/H_2O moderated system in the presence of IFBA rods are lower than those of the same system without IFBA rods. This is due to the rapid burnup (due to higher neutron absorption) of the Gd-155 and Gd-157 burnable absorber materials during the initial steps. However, after this burnup step, a negligible number of Gd isotopes remain at equilibrium conditions, and thus both systems have almost the same excess reactivity.

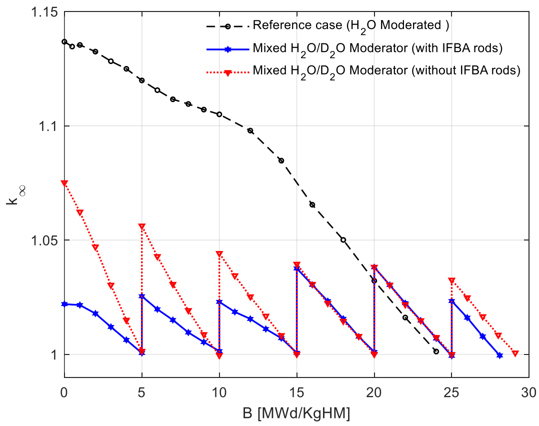


Figure 6. K_{inf} vs. burnup (yanma oranına karşın K_{inf})

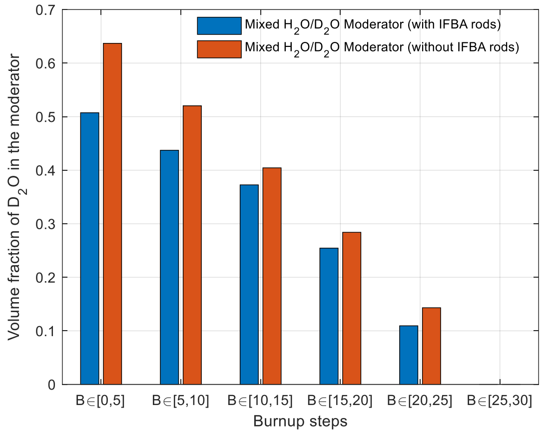


Figure 7. Burnup dependent heavy water volume fraction (Yanma Oranına bağlı ağır su oranı)

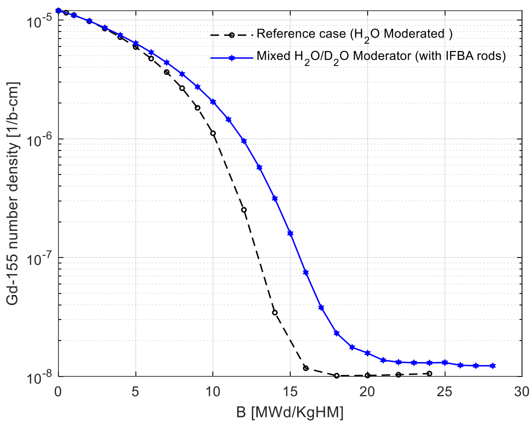


Figure 8. Burnup dependent variation of Gd-155 number density. (Gd-155 atom yoğunluğunun yanma oranına bağlı değişimi)

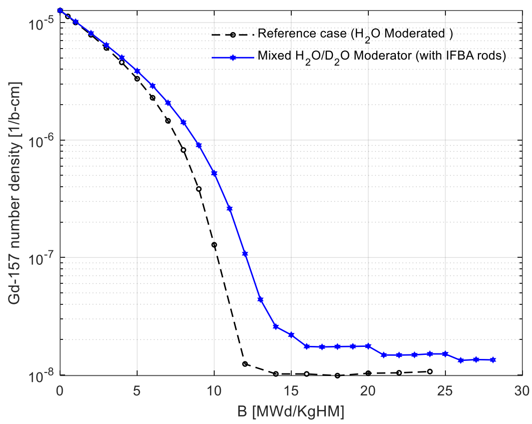


Figure 9. Burnup dependent variation of Gd-157 number density. (Gd-157 atom yoğunluğunun yanma oranına bağlı değişimi)

Figure 10 shows the change in the conversion ratio versus burnup for three cases. As seen in the figure, by increasing the D_2O fraction at the BOC, the conversion ratio experiences an increase, although the presence of burnable absorbers may reduce it slightly. In contrast, the conversion ratio of the H_2O moderated assembly is low at the BOC and goes up with burnup. It should also be noted that all burnup-dependent parameters are plotted up to the system's cycle burnup.

It can be concluded that the improvement in the cycle burnup originated from using a higher amount of heavy water during the first burnup steps. In other words, due to the lower slowing down power of D_2O, the neutron spectrum is hardened (shifted to the resonance region) at BOC. Subsequently, due to the higher capture cross section of the fertile ($U-238$) material in the resonance region, the high majority of excess neutrons are absorbed by these materials and cause to the generation of $Pu-239$ and subsequent ($Pu-241$) fissile materials. These result in an increase in the conversion ratio in comparison with the H_2O moderated system. Toward the End of the Cycle (EOC), by increasing the H_2O fraction neutron spectrum becomes softer (shifted to thermal region). Due to higher fission cross sections of fissile materials in the thermal region, the generated fissile materials during the first burnup steps have a considerable contribution to thermal power generation and subsequently result in the extension of cycle length. In the H_2O moderated assembly due to higher slowing down power the high majority of the BOC excess thermal neutrons are absorbed in the fuel material resulting in a lower conversion ratio and larger excess reactivity at the BOC.

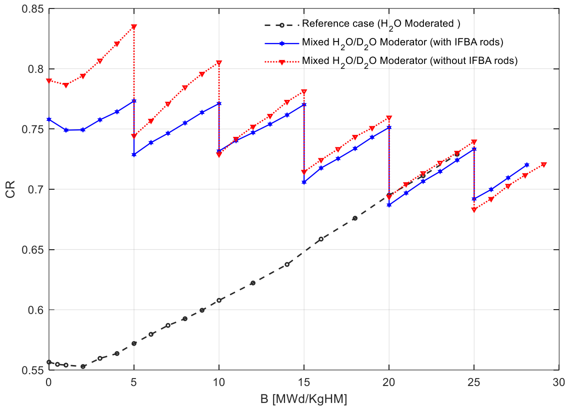


Figure 10. Conversion ratio vs. burnup. (Yanma oranına karşın dönüşüm oranı)

Atom density variations of $U-235$, $U-236$, $U-238$, and $Pu-239$ isotopes during burnup are plotted in Figures 11 through 14. It is clearly seen that, at any burnup point, due to the application of mixed H_2O/D2O moderators: $U-235$ and $U-238$ consumption rates experience a decrease and an increase, respectively. Moreover, the production rates of both $U-236$ and $Pu-239$ go up, although the increase rate of the $Pu-239$ is more considerable.

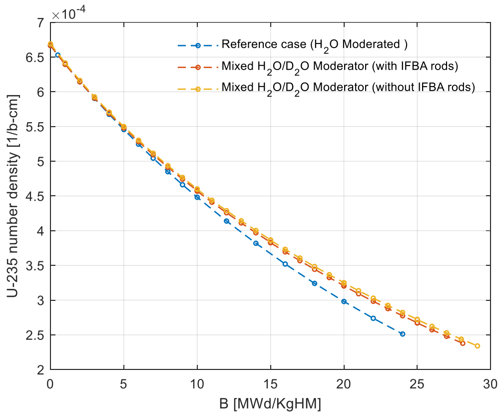


Figure 11. U-235 number density change with burnup. (U-235 atom yoğunluğunun yanma oranına bağlı değişimi)

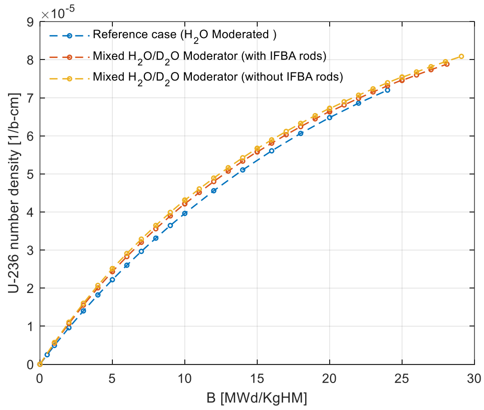


Figure 12. U-236 number density change with burnup. (U-236 atom yoğunluğunun yanma oranına bağlı değişimi)

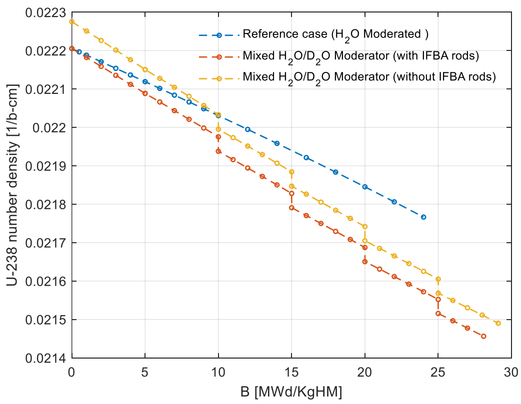


Figure 13. U-238 number density change with burnup. (U-238 atom yoğunluğunun yanma oranına bağlı değişimi)

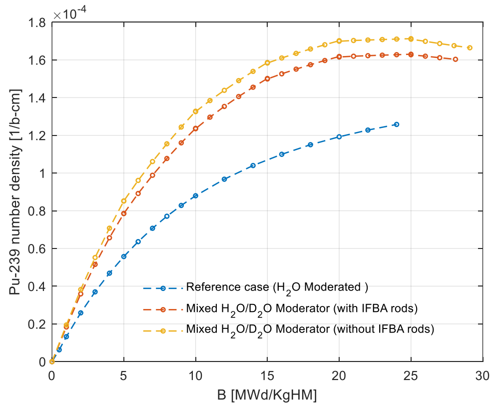


Figure 14. Pu-239 number density change with burnup. (Pu-239 atom yoğunluğunun yanma oranına bağlı değişimi)

5.2. Case2: Assembly Type 2 (Durum2: 2. DEMET TİPİ)

For assembly type 2, the infinite multiplication factor change with burnup is shown in Figure 15. The corresponding D_2O volume fractions are also plotted in Figure 16. The cycle burnup for the mixed moderated system in the presence and absence of the burnable absorbers are calculated as 47 MWd/kgU and 50.1 MWd/kgU, In comparison with

41.08 MWd/kgU cycle burnup of the H_2O moderated assembly, the cycle burnup is improved by 14.4% and 21.95%, respectively. Since there are more IFBA rods in this assembly type, the cycle burnup improvement rate is less than that of assembly type 1. But in the absence of burnable absorbers, in both assembly types, there is an improvement of about 21%. In addition, due to higher enrichment, BOC volume fractions of D_2O in this assembly type are calculated more than those of assembly type 1. Due to higher volume fractions, it is expected the system conversion ratio values are also greater than those of the assembly type 1, however conversion ratio is inversely proportional to the enrichment. As Figure 17 shows, there is a lower conversion ratio compared with assembly type 1.

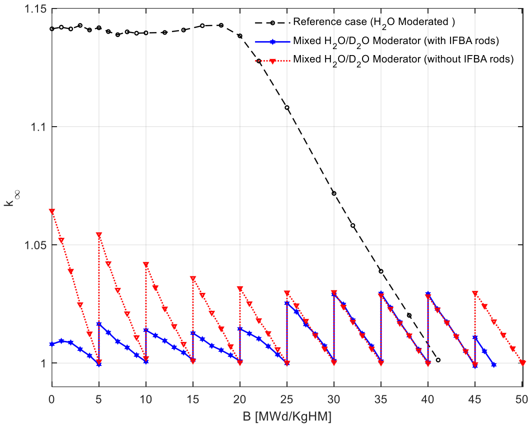


Figure 15. Infinite multiplication factor changes with burnup. (Sonsuz çoğalma faktörünün yanma oranıyla değişimi)

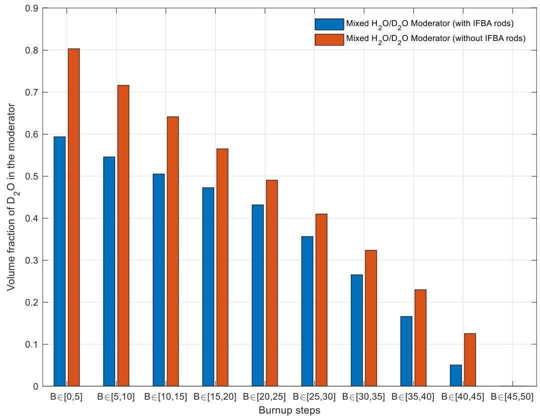


Figure 16. Heavy water volume fractions vs. burnup steps. (Yanma Oranına bağlı ağır su oranı)

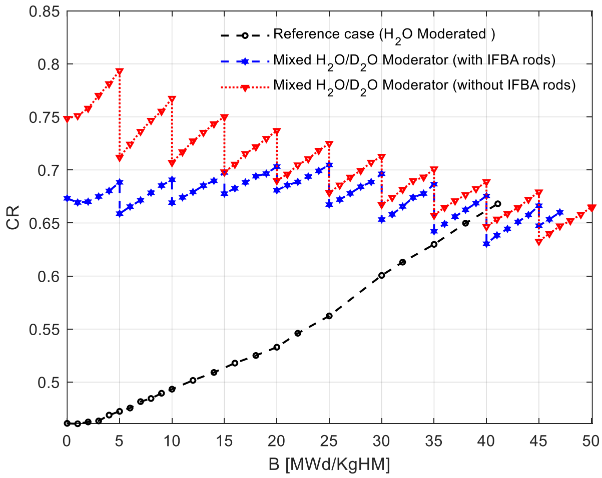


Figure 17. Conversion ratio variation with burnup. (Yanma oranına karşın dönüşüm oranı)

5.3. ASSESSMENT OF THE THORIUM EFFECTS (TORYUM ETKİLERİNİN DEĞERLENDİRİLMESİ)

$Th-232$ is a fertile material and by absorbing a neutron is finally converted to U-233 fissile material. As seen in the previous section, the usage of mixed D_2O/H_2O moderators with different volume fractions of D_2O causes an increase in the cycle burnup. In this test case, it is tried to use $(U+Th)O_2$ in assembly type 2 without considering any burnable poison. And calculate the required content of ThO_2 in fuel to have the same (41.08 MWd/kgU) cycle burnup with H_2O moderated reference assembly. To find the optimum ThO_2 content in fuel that satisfies the cycle burnup of 41.08 $MWd/kgHM$ the following is done: For different mass fractions of ThO_2, the cycle burnups of the assembly with a mixed moderator are calculated. By using the obtained values and applying the linear interpolation method the optimum ThO_2 content of the $(Th+U)O_2$ is calculated equal to 14.09$w/0$. As shown in Figure 18, our calculated value for ThO_2 content meets the cycle burnup criterion of 41.08 $MWd/kgHM$. In addition, by calculating the UO_2 mass in both H_2O

moderated reference assembly and $(Th+U)O_2$ fueled assembly with mixed D_2O/H_2O moderator, it is seen that there is about 40 $kg\ UO_2$ mass saving when using the mixed moderator in Th-fuel assembly (See the *Appendix A* for detailed calculation). This, in turn, is because of higher conversion ratios resulting from the presence of $Th-232$ and $U-238$ fertile materials (Figure 19). The volume fractions of heavy water are also calculated as 0.6323, 0. 5482, 0. 4701, 0. 3965, 0. 3154, 0. 2312, 0. 1350, 0. 0274, and 0.

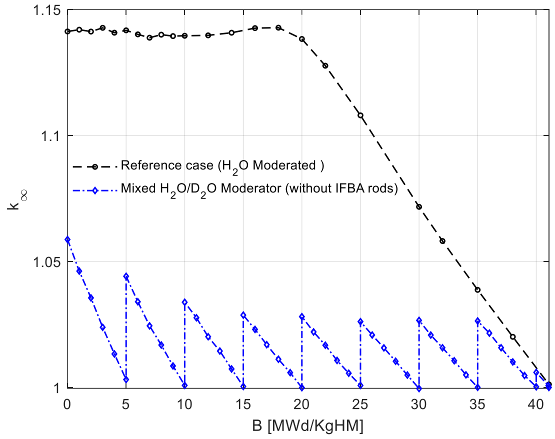


Figure 18. k_{inf} changes with burnup. (Yanma oranına karşın k_{inf})

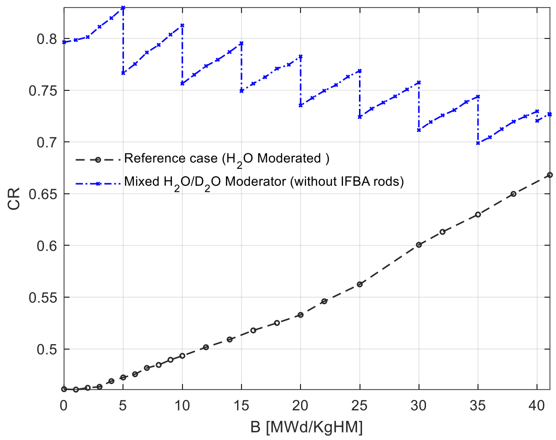


Figure 19. Conversion ratio vs. burnup. (Yanma oranına karşın dönüşüm oranı)

6. CONCLUSIONS (SONUÇLAR)

According to the IAEA report, more than 80 percent of the operating reactors are LWRs, which use UO_2 (with less than 5% enrichment) as fuel material. Due to the high consumption rate of uranium, the natural resource of this precious material decreases. Either the extracted plutonium from the spent fuel during the reprocessing process (in PuO_2 form) or thorium dioxide materials are suggested to reduce the excess uranium use. In addition, it is desired to get the maximum possible thermal energy from the loaded fuel, meaning that, for a certain amount of the fuel the maximum possible cycle burnup is wanted. In line with this goal, several fuel, cladding, and moderator materials are being suggested, and their effect on the neutronic behavior of the nuclear reactors is being investigated.

In the present work, it is investigated how the mixed heavy/light water moderator affects the cycle length of a fuel assembly. For this aim, for two different assemblies of the SMART reactor, the volume fraction of heavy water in each burnup step is calculated. It is observed that due to the high fraction of heavy water at the BOC, the neutron spectrum is shifted to the resonance region. Subsequently, the excess neutrons are absorbed by fertile material and this, in turn, increases the conversion ratio. Toward the EOC the spectrum is shifted to the thermal region. Due to the higher fission cross section of fissile materials in this region and also because of the higher production rate of the new fissile material during the BOC, cycle burnup is extended by almost 21%. Moreover, it is seen that the extension of cycle burnup of assemblies in the presence of burnable absorbers is less than that without burnable absorbers. Finally, the reference assembly is compared with an assembly fueled by $(Th+U)O_2$ and moderated with mixed D_2O/H_2O with the same cycle burnup; it is seen that there is an almost 40 $kgUO_2$ mass saving in comparison with the reference assembly.

APPENDIX A. UO_2 MASS SAVING DUE TO $(Th+U)O_2$ FUEL USAGE (($(Th+U)O_2$YAKIT KULLANIMI NEDENİYLE UO_2 KÜTLE TASARRUFU)

As mentioned earlier, H_2O moderated reference assembly contains 244 and 20 numbers of UO_2 and $(UO_2+Gd_2O_3)$ IFBA fuel rods, respectively. The physical density of the UO_2 and IFBA fuels are also taken equal to 10.286 g/cm^3 and 10.017g/cm^3, respectively. The active height and radius of the fuel regions are also given as $H=200\ cm$ and $R_f=0.4096\ cm$. Hereby, as shown in Equation A.1, the total mass of UO_2 is calculated as 284 kg .

$$m_{UO_2}=\left[244\left(\rho_{UO_2}\pi R_f^2H\right)+20\left(0.92\left(\rho_{IFBA}\pi R_f^2H\right)\right)\right]\times10^{-3}=283.996\ kg \qquad (A.1)$$

On the other side, in case of using $(Th+U)O_2$ fuel and mixed heavy/light water moderator, the required content of ThO_2 in fuel for 41.08 MWd/kgU cycle burnup is obtained as $14.09w/0$. By considering the physical density of ThO_2 as 9.970 g/cm^3 . The physical density of $(Th+U)O_2$ mixture is obtained equal to 10.241 g/cm^3. In this case, the total mass of UO_2 becomes equal to 244.844 kg.

$$m'_{UO_2} = \left[264\left((1-0.1409)\left(\rho_{(Th+U)O_2}\pi R_f^2 H\right)\right)\right] \times 10^{-3} = 244.844\,kg \qquad (A.2)$$

It is seen that there is an almost 40 $kgUO_2$ mass saving in comparison with the reference assembly.

DECLARATION OF ETHICAL STANDARDS (ETİK STANDARTLARIN BEYANI)

The author of this article declares that the materials and methods he used in his work do not require ethical committee approval and/or legal-specific permission.

AUTHORS' CONTRIBUTIONS (YAZARLARIN KATKILARI)

***Behram MELİKKENDLİ*:** Writing, Software, Methodology, Investigation, Formal analysis, Conceptualization.

CONFLICT OF INTEREST (ÇIKAR ÇATIŞMASI)

There is no conflict of interest in this study.

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