



Original Article

Neutronic examination of the U–Mo accident tolerant fuel for VVER-1200 reactors

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ABSTRACT

In this study, we investigated the possibility of employing accident tolerant fuel (ATF) in VVER-1200/V491 assembly without gadolinium-containing fuel rods using the Monte Carlo code Serpent 1.1.7 with ENDF/B-VII cross-section library. The analysis involves assembly design with reflective boundary conditions. To compare the neutronic performances, U–5Mo, U–7.5Mo, U–10Mo, and U–15Mo fuels were chosen in addition to ordinary UO_2 fuel. The concentration of ^{135}Xe , ^{149}Sm , fissile and fertile isotopes with burnup, reactivity feedback with fuel temperature variation, and β_{eff} values were calculated. The results indicate that the fuel cycle length increases by 54.27% for U–5Mo, 32.6% for U–7.5Mo, and 13.8% for U–10Mo, while it decreases by 16.4% for U–15Mo fuel. Additionally, the effect of ^{95}Mo content in natural Mo was investigated by reducing the ^{95}Mo concentration. According to the results, each proposed fuel's fuel cycle length extended when the depletion ratio of ^{95}Mo increased. Additionally, the calculations for reactivity feedback guarantee safe operating conditions for all U–xMo fuels.

1. Introduction

Nuclear reactors are now used more frequently for peaceful reasons due to their reliability, sustainability, and effectiveness as a source of widespread electricity. As a result, research about safety and operational characteristics of nuclear reactors has risen, and these features have been improved [1–4]. Along with these improvements, nuclear fuel—one of the key elements of a nuclear reactor—has become safer and more cost-effective to handle any potential accidents to which the reactor may be exposed. Following the catastrophic events at Three Mile Island and Fukushima Daichi, research into accident-tolerant fuel (ATF), which may run in the reactor under challenging conditions instead of uranium dioxide, has risen [4,5]. The primary subjects of these ATF studies are cladding and fuel. FeCrAl and SiC have been proposed as new cladding materials as a consequence of cladding material investigations. Additionally, it has been suggested that coatings containing aluminum, niobium, chromium, and/or platinum be used to improve the corrosion resistance of current cladding materials, such as Zr alloy [6–9]. High uranium density and high thermal conductivity are the general characteristics of ATF fuel that are expected in studies focusing on its development. Some fuels with desired properties are namely U_3Si_2 , and UN. Neutronic analyzes were performed for potential ATF fuels, including UN, U_3Si_2 ([5,9–11], Sheng Li Chen et al., 2020, and [12]). Another ATF option is U–Mo metallic fuel, which

has a higher uranium density than UO_2 fuel and allows for a longer fuel cycle to be run at the same enrichment level [13]. U–Mo fuel was chosen by Department of Energy, Office of Nuclear Energy (DOE-NE) due to its potential to increase fuel utilization and be compatible with fuel assemblies using UO_2 [7]. Furthermore, irradiation studies in reactor fuel development programs have demonstrated that U–Mo alloys are the best match for the dispersed fuel phase in terms of irradiation behavior; however, fuel-matrix interaction causes extreme swelling behavior in specific irradiation circumstances [14]. On the other hand, monolithic U–Mo fuel provides highest density; however, corrosion interaction can occur between fuel and Al cladding [7,14]. Another negative characteristic of U–Mo is that it has a lower melting point than UO_2 , but since it has considerably higher thermal conductivity than UO_2 , a lower maximum fuel temperature will occur within the core [13,14].

Though it has been stated that U–Mo fuels have outstanding properties against radiation, molybdenum content is important in the fuel. There are several U–Mo fuel options with different molybdenum content are available [14]. The Reduced Enrichment for Research and Test Reactor (RERTR-1 and RERTR-2) studies showed that U–Mo fuels containing less Mo causes more fission gas driven swelling [15,16], and 7–10 wt% Mo are the best options [16–19]. U-7.65Mo alloy fuel in 7–10 wt% Mo range also selected to use in The Kilowatt Reactor Using Stirling Technology (KRUSTY) [20,21]. In order to investigate different Mo

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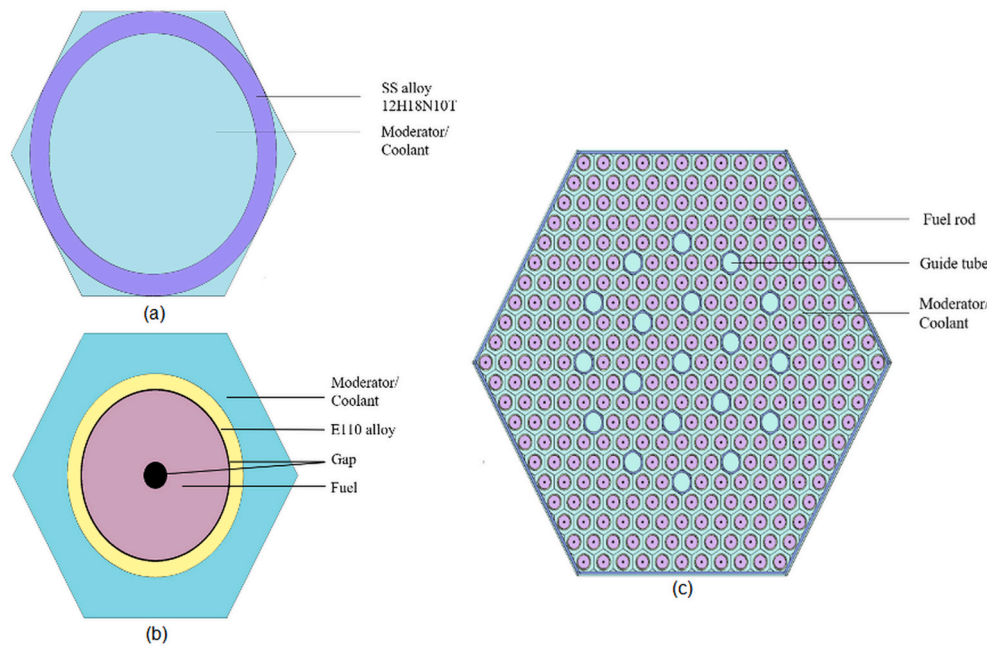


Fig. 1. Radial view of a) guide tube (upper left) b) fuel rod (lower left) c) VVER-1200 assembly (right).

content effect on neutronic behavior, U–Mo fuels with 5, 7.5, 10 and 15 wt% Mo that are close to 7–10 wt% Mo have been selected in this study.

When it comes to U–Mo fuels, there is another topic that needs to be examined: the ^{95}Mo isotope ratio. Some molybdenum isotopes, particularly ^{95}Mo , have a comparatively high parasitic thermal neutron absorption cross-section [1]. This causes fuel to have higher negative reactivity. According to ([1,22]; Chen, He and Yuan, 2020) decreasing the ^{95}Mo content in natural molybdenum can reduce the absorption rate and results in negative reactivity reduction in fuel. Therefore, the impact of the ^{95}Mo isotope for U–Mo compositions should be investigated to ascertain its effect on possible U–Mo fuels.

With the aim of reducing corrosion in U–Mo fuels, a corrosion-resistant layer diffused on the fuel surface has been proposed. In this layer, a corrosion-resistant coating material such as aluminum, niobium or chromium is diffused onto the fuel rod, and then a cladding material such as FeCrAl or Zircaloy material is used. Studies have shown that coating materials used in small amounts can reduce metal-water reactions, consequently corrosion [7]. In order to investigate the feasibility of U–Mo in VVER-1200 core, we should investigate the U–Mo fuel with the proposed multi-layer cladding model.

As it has been mentioned by (Chen, He and Yuan, 2020) and [22] that ^{95}Mo depleted fuel elongates the fuel cycle length in LWRs because of its higher thermal neutron capture cross-section. In this study, four proposed U–Mo fuels with different natural molybdenum composition were examined to investigate the effect on fuel cycle length for VVER-1200 assembly. Furthermore, ^{95}Mo compositions in proposed fuels changed to investigate the ^{95}Mo effect on fuel cycle length. Additionally, to effectively integrate the U–Mo metallic fuel into the VVER-1200 assembly, two fuel models incorporating niobium coating were applied to the U–Mo metallic fuel. One utilized FeCrAl cladding, while the other employed E110 alloy cladding. The utilization of (Nb) was chosen due to its substantial impact on the development of innovative cladding alloys[6,8,23]. Hence, Serpent 1.1.7, a continuous energy Monte-Carlo code created by the VTT Technical Research Centre of Finland, was used to examine these potential fuels [24]. The following procedures have been followed to demonstrate the results of U–Mo fuel compositions: (1) Serpent 1.1.7 with ENDF 7.0 cross section library used to compute the burnup of the VVER-1200 assembly using each proposed fuel, including UO_2 as a standard. Additionally, different U–Mo fuels with varying Mo compositions are used for multi-layer cladding model

Table 1

The VVER-1200 (V-491) assembly and material description (IAEA, 1995; Dien and Diep, 2017 [10]; Hafez et al., 2021)).

Parameter	Value
Reactor thermal output	3200 MWth
Power Plant Output	1200 MWe
Average fuel power density	36.80 kW/kgU
Coolant, Moderator	Purified water + boric acid
Fuel Assembly (FA)	
FA Shape	Hexagonal
Fuel assembly number in the core	163
FA flat-to-flat size	23.42
Gap between fuel assemblies	34 cm
Pitch between fuel elements	1.275 cm
Number of fuel elements in FA	312
Fuel Element (FE)	
Fuel Type	UO_2
Enrichment	Up to 5 wt % ^{235}U
FE shape	Rod
Height of fuel	375 cm
Diameter of fuel pellet	0.76 cm
Diameter of central hole	0.12 cm
VVER-1200 Fuel Clad Properties	
Material	E-110 alloy
Density	6.515 g/cm ³
Outer diameter	0.91 cm
Inner diameter	0.773 cm
Alternative Fuel Clad Material Density	FeCrAl: 7.10 g/cm ³
Clad thickness	0.046625 cm
Nb thickness	0.021875 cm
Control Rod Properties	
Number of guide channels in FA	18
Guide channel outer diameter	1.29 cm
Guide channel inner diameter	1.09 cm
Clad material	SS alloy 12H18N10T
Clad density	7.85 g/cm ³
Central tube	1
Outer diameter	1.29 cm
Inner diameter	1.09 cm

using FeCrAl and E110 alloy cladding materials to assess the impact of the proposed model on fuel cycle length. (2) Proposed U–Mo fuel compositions were used instead of UO_2 in the VVER-1200 assembly to

Table 2
Properties of proposed fuels[14,26].

Fuels	Density [g/cm ³]			Uranium enrichment [wt%]
	Uranium	The others	Total	
UO ₂	9.210	1.24	10.45	4.79
U-5Mo	17.10	0.90	18.00	
U-7.5Mo	16.30	1.30	17.60	
U-10Mo	15.50	1.70	17.20	
U-15Mo	14.10	2.50	16.60	

examine the fuel cycle length and change in the concentration of heavy metals that are important to evaluate the optimum fuel. (3) To assess the ⁹⁵Mo effect in fuel, the ⁹⁵Mo isotopic abundance in natural Mo in four of U–Mo fuel has been modified. (4) In view of fuel cycle length, ⁹⁵Mo content for U–Mo fuel has been assessed and compared with the conventional VVER-1200 assembly that is loaded with UO₂.

2. Materials and methods

2.1. VVER-1200 assembly description

The VVER-1200/V491 is a Generation III + type reactor which is developed by enhancing the VVER-1000 reactor characteristics. Purified water is used both as moderator to moderate the neutrons and coolant to absorb the heat produced by the UO₂ fuel. The VVER-1200 has more power and a longer lifespan (60 years) through improvements made to the VVER-1000. The reactor consists of 163 hexagonal fuel assemblies and is placed on a hexagon grid. Each assembly contains 312 fuel rods, 1 central tube and 18 guides tube[25] Water is contained in the geometry's guides and central tube. The schematic illustration (radial view) of the VVER-1200/V491 fuel assembly used in this study is shown in Fig. 1. Also, standard fuel assembly and alternative fuel cladding parameters are listed in Table 1.

To assess the effects of the multi-layered cladding material model, the proposed model was initially examined on the FeCrAl cladding material [7]. To enable a comparison with the existing protective material, the same model was also applied for the E110 alloy cladding material[23]. In this study, the thickness of niobium was selected to align with the average value within the recommended thickness range, while the cladding thickness was chosen to match the outer radius of a

Table 3
Thermal neutron absorption cross-sections for ¹⁶O and Mo isotopes[27].

Isotope	Thermal absorption cross-section (barn)
¹⁶ O	0.00019
⁹² Mo	0.02076
⁹⁴ Mo	0.34035
⁹⁵ Mo	13.5676
⁹⁶ Mo	0.59556
⁹⁷ Mo	2.19665
⁹⁸ Mo	0.13001
¹⁰⁰ Mo	0.19908

typical fuel rod.

In terms of strong irradiation performance and a sufficiently high intrinsic uranium density, the RERTR-1 and RERTR-2 studies showed that U–Mo fuels containing 7–10 w% Mo are the best options [16]. Therefore, in this study, U-5Mo, U-7.5Mo, U-10Mo, U-15Mo are selected to investigate the neutronic behavior in VVER-1200. Additionally, UO₂ fuel has also been taken into consideration in order to compare the VVER-1200 assembly with the proposed U–Mo fuel. The characteristics of discussed fuels are given in Table 2. The mentioned VVER-1200 assembly does not contain any gadolinium loaded fuel, and all fuel utilized in assemblies has the same ²³⁵U enrichment ratio of 4.79 wt% to study the Mo composition effect [26]. The VVER-1200 core was defined using the Monte Carlo code Serpent 1.1.7 with ENDF/B-VII cross-section library[27]. Calculations were performed with 10000 source neutrons, 500 active and 50 inactive cycles, resulting in a maximum statistical error of 35 pcm. To avoid neutron leakage, a reflective boundary condition is utilized at the assembly boundaries.

3. Results

3.1. Fuel burnup

To evaluate the status of the assembly by utilizing different fuels, the infinite multiplication factor (k_{∞}) has been considered. Based on the effective full power day (EFPD), Fig. 2 a shows the k_{∞} behavior for each of the five fuels. At the first look of Fig. 2 a, there are two noticeable results: the k_{∞} value at the beginning of the cycle (BOC), and the trend of U–Mo fuels. Though U–Mo fuels have a greater uranium density as given

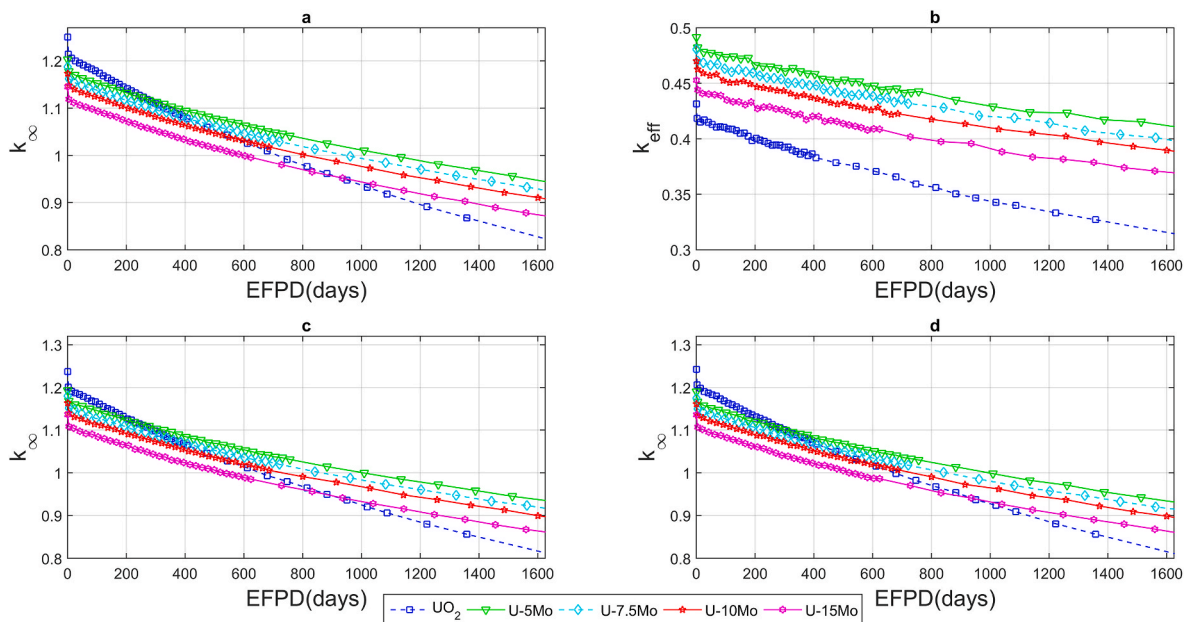


Fig. 2. Variation of a) k_{∞} in VVER-1200 assembly with reflective boundary condition, b) k_{eff} in VVER-1200 fuel assembly with vacuum boundary condition, c) k_{∞} for multi-layer cladding model with FeCrAl d) k_{∞} for multi-layer cladding model with E110 alloy for the proposed fuel assemblies with EFPD.

Table 4
U–Mo fuel properties[28].

Isotope	⁹² Mo	⁹⁴ Mo	⁹⁵ Mo	⁹⁶ Mo	⁹⁷ Mo	⁹⁸ Mo	¹⁰⁰ Mo
Abundance (%)	14.65	9.19	15.87	16.67	9.58	24.29	9.74
New Fuel Compositions							
U–Mo (10 wt%)	16.11	10.109	13.486	16.67	9.58	24.29	9.74
U–Mo (20 wt%)	17.58	11.028	11.102				
U–Mo (30 wt%)	19.04	11.947	8.718				
U–Mo (40 wt%)	20.51	12.866	6.334				
U–Mo (50 wt%)	21.97	13.785	3.95				

in Table 2, k_{∞} value for UO_2 is the highest at the BOC among all fuels. This is due to the absorption cross-section of ^{16}O in UO_2 fuel being significantly smaller than Mo isotopes in U–Mo fuels, as given in Table 3. The maximal fuel cycle lengths for UO_2 , U–5Mo, U–7.5Mo, U–10Mo, and U–15Mo are calculated to be 715, 1102, 947, 813, and 597 days, respectively. These values were calculated based on when k_{∞} value

reaches criticality value in burnup. According to the results, fuel cycle length increases by 54.27% for U–5Mo, 32.6% for U–7.5 and 13.8% for U–10Mo while it decreases by 16.4% for U–15Mo fuel at criticality. In Fig. 2 b, k_{eff} value for VVER-1200 assembly burnup has calculated using vacuum boundary conditions. Here, k_{eff} value is in subcritical level. Since the neutron leakage rate is effective here, UO_2 with the least density has the minimum k_{eff} value. In Fig. 2, sections c and d sequentially demonstrate the application of a multi-layered cladding model, where Nb serves as the coating material, and FeCrAl and E110 alloys are utilized as cladding. It has been evaluated that average cross-section values of the utilized isotopes in FeCrAl and E110 alloys are so closely aligned that the resulting values appear highly similar.

3.2. The effect of ^{95}Mo isotope

To investigate the ^{95}Mo depletion effect on fuel cycle length in proposed U–Mo fuels, five different ^{95}Mo depleted fuel compositions for each

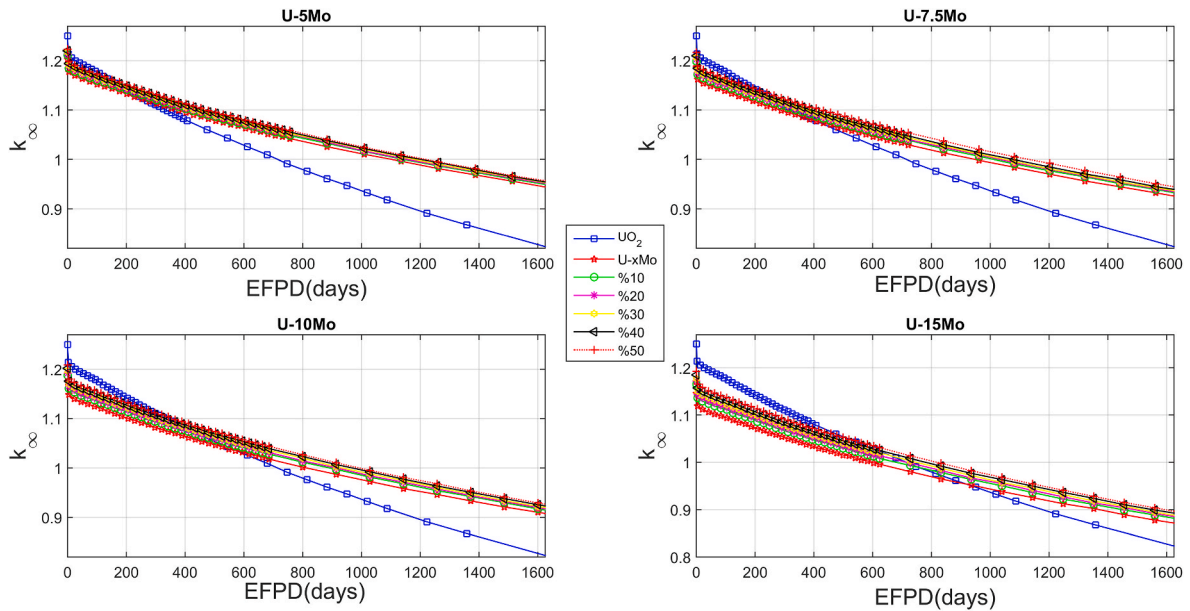


Fig. 3. Variation of effective multiplication factor for different depleted U–xMo fuel assemblies with EFPD.

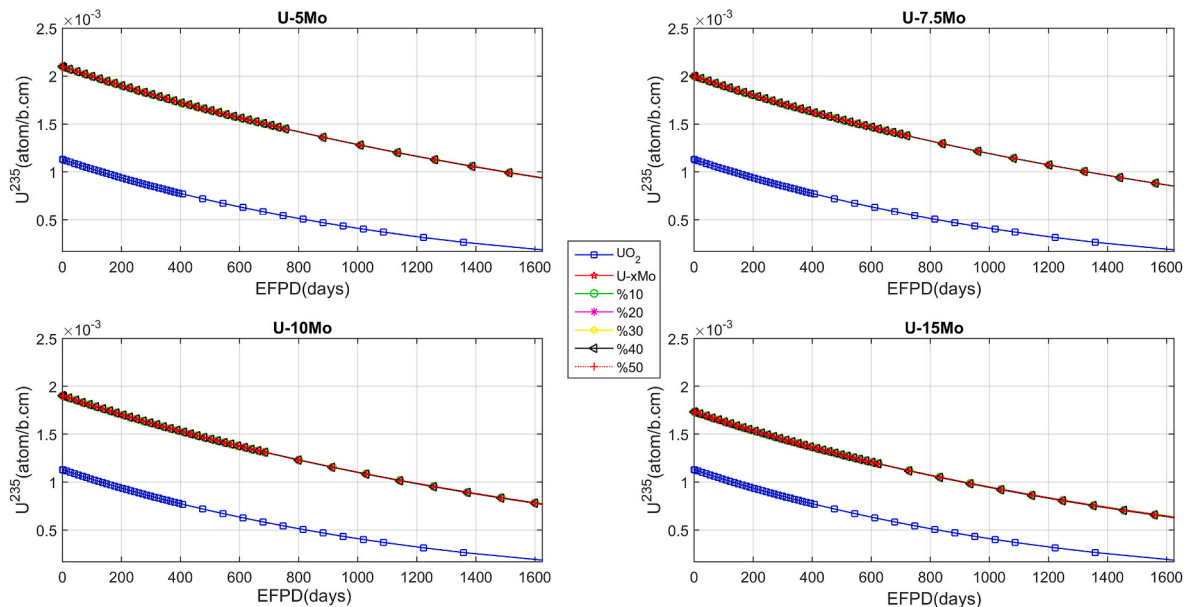


Fig. 4. Concentration variation of ^{235}U as a function of the depleted U–xMo fuel types.

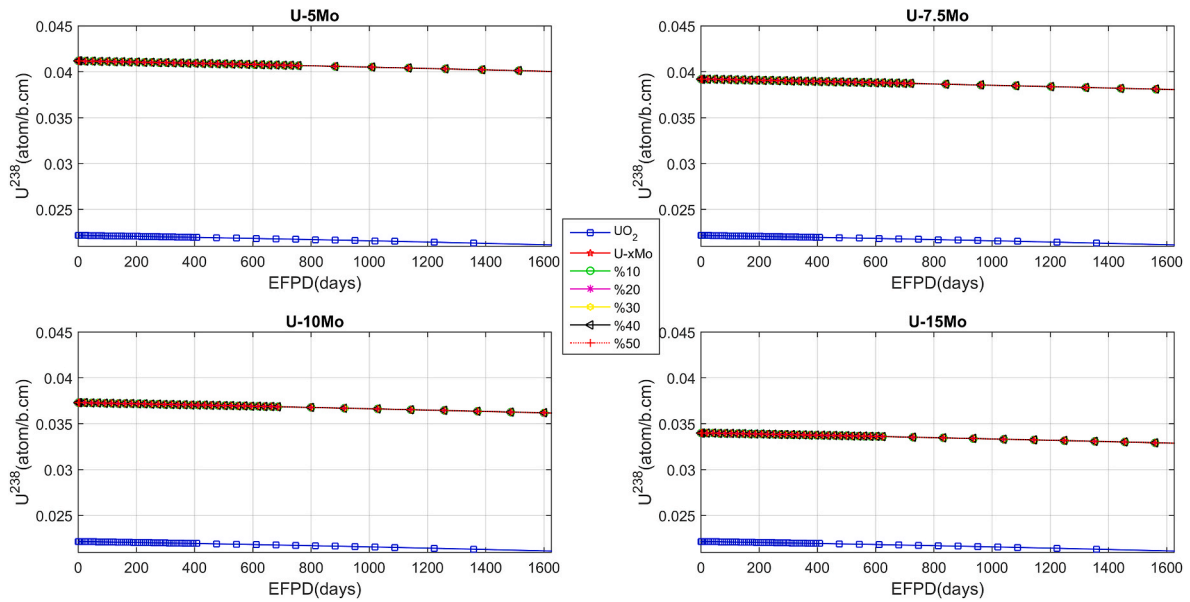


Fig. 5. Concentration variation of ^{238}U in the depleted U-xMo fuel types as a function of EFPD.

proposed fuel were considered. ^{95}Mo depletion is feasible in two ways: either increasing the proportion of heavier isotopes (^{96}Mo , ^{97}Mo , ^{98}Mo , ^{100}Mo) than ^{95}Mo or increasing the proportion of lighter isotopes (^{92}Mo , and ^{94}Mo) than ^{95}Mo while total mass of Mo is constant [22]. In this study, ^{95}Mo isotope depletion is obtained by increasing the lighter isotopes that have lower absorption cross-sections while decreasing ^{95}Mo isotope while total mass of Mo is constant. The content of ^{95}Mo is reduced by 10, 20, 30, 40 and 50 wt% and resulting new compositions are listed in Table 4.

The ^{95}Mo depletion effect on k_{∞} with EFPD is shown in Fig. 3 for all proposed and depleted fuels. These results indicate that as depletion ratio increases, fuel cycle length elongates for each proposed fuel. In the previous section, U-15Mo fuel cycle length (597 days) was found to be shorter than standard UO_2 fuel cycle length (715 days) due to the parasitic effect of ^{95}Mo . It has been concluded that in U-15Mo fuel with 20 wt% depleted Mo, fuel cycle length increases approximately 20 % and reaches the same fuel cycle length as UO_2 fuel.

Figs. 4 and 5 shows the ^{235}U and ^{238}U concentration as a function of EFPD for all investigated fuels. The composition of U-5Mo has a maximum atomic density of ^{235}U at the BOC. Since only molybdenum (5, 7.5, 10 and 15) content is changed for proposed ATFs, the production of ^{235}U and ^{238}U have similar trend as a function of burn-up.

Since proposed fuel types have ^{238}U as fertile material, it is important to investigate the ^{239}Pu production in fuels given in Fig. 6. Due to high amount of ^{238}U in U-5Mo fuel, ^{239}Pu production in this fuel higher among all proposed fuel. At the BOC, ^{239}Pu concentration increases rapidly in all fuels and increase rate slowly decreases since produced ^{239}Pu undergoes fission. As for all proposed U-Mo fuels, ^{239}Pu production does not reach equilibrium due to high ^{238}U concentration. For each U-Mo fuel, as ^{95}Mo depletion increases, ^{239}Pu production decreases due to net absorption cross-section decrease of neutron. Since U-15Mo fuel contains more Mo, this effect is more prominent among proposed U-Mo fuels.

It is crucial to consider some fission products such as ^{135}Xe and ^{149}Sm

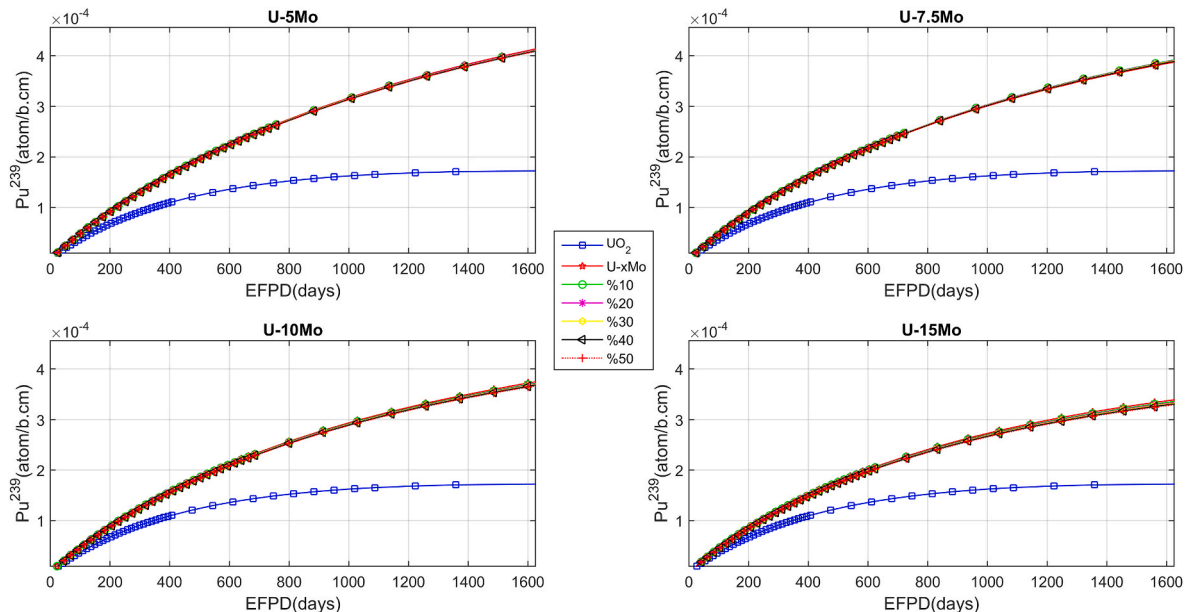


Fig. 6. Concentration variation ^{239}Pu in depleted U-xMo fuel types as a function of EFPD.

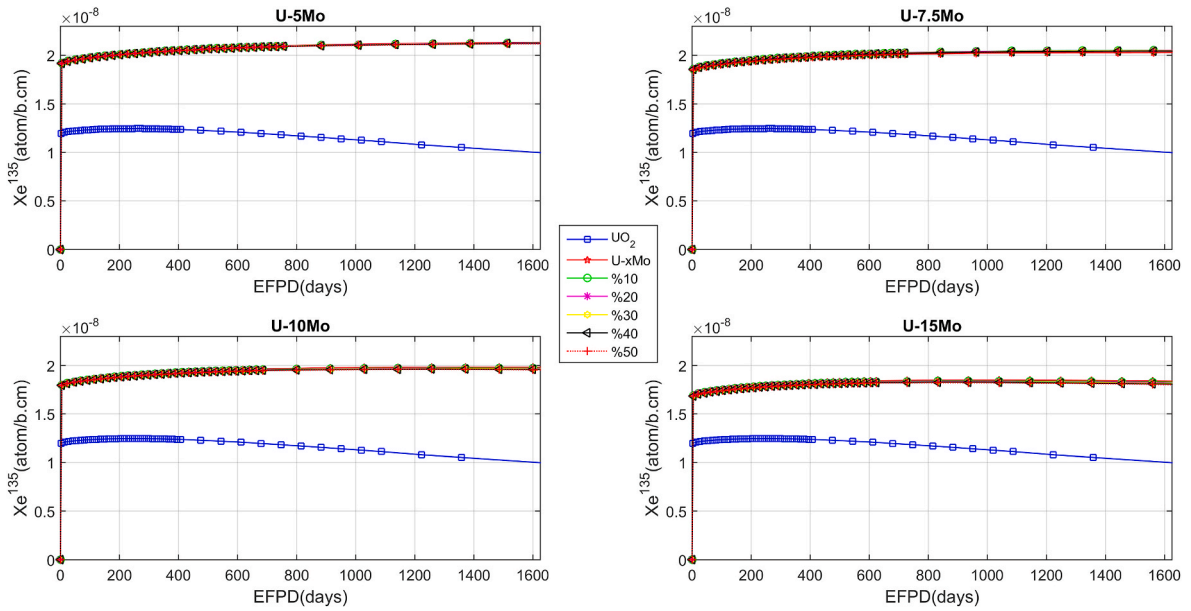


Fig. 7. Concentration variation of ^{135}Xe in depleted U-xMo fuel types as a function of EF.

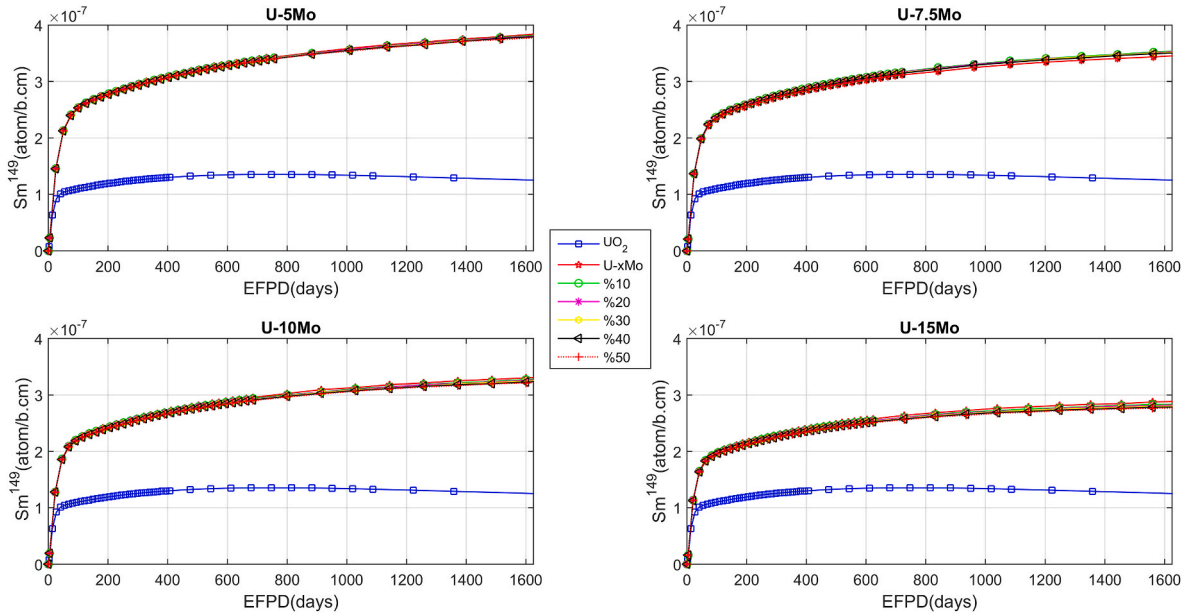


Fig. 8. Concentration variation of ^{149}Sm in depleted U-xMo fuel types as a function of EF.

during the burn-up of their large thermal neutron absorption cross-section. Figs. 7 and 8 shows variation of ^{135}Xe and ^{149}Sm concentration of with EF.

Due to its smallest ^{235}U density among proposed fuels, UO_2 fuel produces significantly lower ^{135}Xe than U-Mo fuels. Moreover, for UO_2 , ^{135}Xe concentration decreases after EF of approximately 200 days since ^{235}U concentration decreases beyond this point.

Unlike ^{135}Xe , ^{149}Sm can only be produced via the decay of fission product ^{149}Nd . Since ^{149}Sm is a stable isotope, it prolongs the time to reach maximum concentration as shown in Fig. 8 and can only be consumed by neutron capture. For UO_2 fuel, ^{149}Sm reaches maximum and its concentration decreases after this point (approximately 800 days) with burnup since ^{235}U concentration decreases with burnup. ^{149}Sm concentration increases continuously throughout burnup for U-Mo fuel.

3.3. Reactivity feedback

Reactivity feedback calculations are made for selected fuels in 300 K–1800 K temperature range. Fig. 9 displays the Doppler coefficients determined by Eq. (4) for U-Mo fuel compositions including 5, 7.5, 10, and 10 Mo. On the x-axis of this figure, temperatures are represented by the T_2 value that was utilized in the reactivity feedback calculations. Since the fuel temperature for Doppler coefficient values are estimated starting from 300 K with 300 K increments, the $T_2 - T_1$ value for this study is always equivalent to 300 K.

$$\alpha_F = \frac{1}{k_1 k_2} \times \frac{k_2 - k_1}{T_2 - T_1} \quad (4)$$

The reactivity feedback mechanisms in BOC and EOC appear to be different when we look at Fig. 9. In BOC, U-Mo fuels provide more negative reactivity to the system than UO_2 at low temperatures (~600 K–900 K). At moderate temperatures (900 K–1500 K), U-7.5Mo fuel has

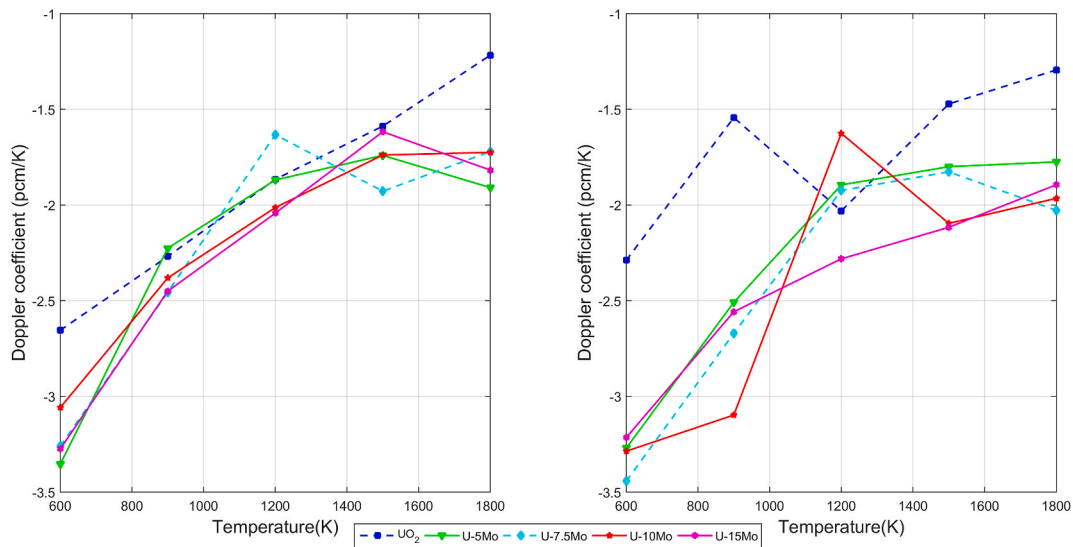


Fig. 9. Variation of Doppler coefficient at BOC (left) and EOC (right) with fuel temperature for all proposed U–Mo fuels.

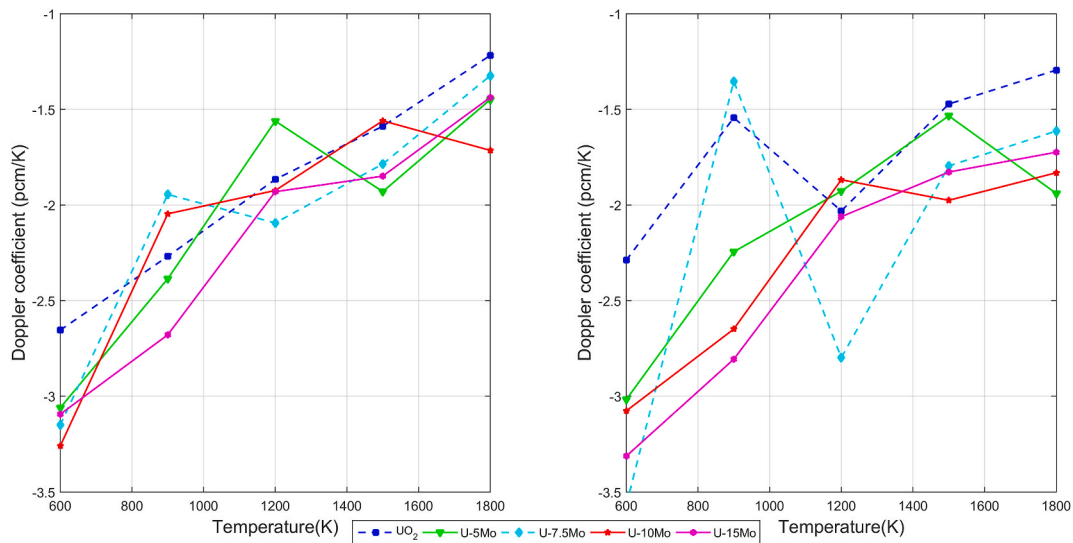


Fig. 10. Variation of Doppler coefficient at BOC (left) and EOC (right) with fuel temperature for all proposed U–Mo fuels with 50 wt% depleted ^{95}Mo .

less negative reactivity than UO_2 fuel at some temperature points, particularly around 1200 K. All of the U–Mo fuels that have been proposed demonstrate greater negative reactivity at high temperatures (1500 K–1800 K). In EOC, all U–Mo fuels, except those around 1200 K (only for U–10Mo), give more negative reactivity to the system. Here, the factors affecting the reactivity include the decrease in the neutron absorption cross section of Mo depending on the temperature and its consequent effect on the system as some positive reactivity as the temperature increases, and the change of the fuel moderator ratio according to the U–Mo composition.

Fig. 10 shows the reactivity feedback results obtained for the VVER-1200 fuel assembly using U–Mo fuels formed with depleted Mo. Although there are no significant variations in reactivity values in BOC, it can be noticed that U-7.5Mo fuel exhibits the most negative reactivity at temperatures about 1200 K in EOC. Accordingly, U–Mo fuels made with depleted Mo are appropriate and reliable for use in the VVER-1200 reactor since they also create negative reactivity during burn-up (typically more than UO_2).

$$\text{MTC} = \alpha_M = \frac{1}{k_1 k_2} \times \frac{k_2 - k_1}{T_2 - T_1} \quad (5)$$

One of the important parameters for reactor control is the β_{eff} value. When a new fuel is recommended, this parameter should also be examined. Therefore, in this study, β_{eff} values for the VVER-1200 fuel assembly were obtained for UO_2 standard fuel and U–Mo fuels throughout the burn-up process and they are presented in Table 5. The table shows that β_{eff} values for U–Mo ATF fuels are slightly higher than for UO_2 fuel, thus, this is a benefit of using ATF fuel in VVER-1200. This can be explained by the fact that there is significantly less prompt neutron-absorbing material for UO_2 fuel than for U–Mo fuel when fuel burnup starts. Since Mo absorbs some of the prompt neutrons, β_{eff} values for U–Mo fuels are higher than those of UO_2 fuel when the fuel starts to burn. However, as this effect diminishes toward the fuel EOC, β_{eff} values for all fuels reach comparable levels. Analyzing moderator temperature coefficient (MTC) is crucial as the reactor might encounter an increase in moderator temperature. In this context, MTC denotes the rate of reactivity change due to variations in moderator temperature. MTC values were estimated starting from 300 K to 600 K with 300 K increment at BOC. Here, the reactivity is influenced by several factors. The decrease in the neutron absorption cross-section and density of the moderator at higher temperatures increases the negative reactivity in the system. Here, due to higher neutron absorption in U–Mo fuels compared to UO_2

Table 5

Beta effective values with burnup and MTC at the beginning of the cycle.

Fuel	β_{eff} 1st	β_{eff} 2nd	β_{eff} 35th	MTC (294 K–600 K)
UO ₂	676.100	687.850	532.367	−11.8574
U–5Mo	701.095	703.466	550.308	−16.4631
U–7.5Mo	715.202	700.650	554.434	−17.3832
U–10Mo	702.426	703.594	551.440	−17.4485
U15Mo	697.861	692.035	556.581	−18.9513
U–5Mo (50 wt% dep.)	706.560	702.914	563.507	−16.0013
U–7.5Mo (50 wt% dep.)	693.235	733.008	549.925	−16.1976
U–10Mo (50 wt% dep.)	683.820	708.867	555.023	−16.3595
U–15Mo (50 wt% dep.)	701.183	698.094	560.761	−17.0361

fuel, the MTC is lower. When examining both U–Mo fuels and fuels containing 50% depleted Mo-95, an increase in Mo content has led to a decrease in the moderator temperature coefficient due to the corresponding increase in neutron absorption in the fuel. As we consider U–Mo fuels and 50% U–Mo fuels with the same Mo ratio for all proposed fuels, the reduction in Mo-95 content has resulted in a decrease in parasitic neutron absorption in the fuel, leading to a slight increase in the moderator temperature coefficient.

4. Conclusion

Neutronic evaluation of accident tolerant fuel U–Mo was performed for VVER-1200/V491 type assembly using the Monte Carlo method. The results were also obtained for standard UO₂ fuel. A comparison for the cycle length, the content of ¹³⁵Xe, ¹⁴⁹Sm, fissile and fertile isotopes with burnup, reactivity feedback with fuel temperature change and β_{eff} values were made.

This study examined the neutronic behavior of U–Mo fuels with various Mo contents (U–5Mo, U–7.5Mo, U–10Mo, and U–15Mo) in comparison to UO₂ fuel in the VVER-1200 core. According to the burn-up dependent k_{∞} results, U–5Mo fuel has the longest fuel cycle length among the proposed U–Mo fuels with natural Mo composition, while U–15Mo has the shortest fuel cycle length and ensures an even shorter fuel cycle length than UO₂ fuel. ⁹⁵Mo isotope amount in natural U–Mo fuel also examined to see its effect on fuel cycle length. As expected, while ⁹⁵Mo isotope depletion ratio increases fuel cycle length increases and maximum fuel cycle length evaluated when ⁹⁵Mo depletion rate is 50 wt% for all proposed fuels.

For all U–xMo fuels, the reactivity feedback calculations guarantee safe operating conditions. U–7.5Mo in BOC and U–10Mo in EOC provide slightly less negative reactivity to the system than conventional UO₂ fuel at moderate temperatures (about 1200 K). Regarding fuel Doppler coefficient, it may be said that U–Mo fuels are generally safer than UO₂ fuel. Additionally, in β_{eff} calculations, it was seen that all U–Mo fuels had greater β_{eff} values than UO₂. This can be shown as another advantage of these fuels in terms of reactor control.

Declaration of competing interest

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

Appendix A. Supplementary data

Supplementary data to this article can be found online at <https://doi.org/10.1016/j.net.2024.02.021>.

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