Journal of Radiation and Nuclear Applications
An International Journal

http://dx.doi.org/10.18576/jrna/090207

Justification of Physical Characteristic of Mixed Oxide (MOX) in VVER-1200 Reactor

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Received: 2 Feb. 2024, Revised: 22 March. 2024, Accepted: 1 April. 2024.

Published online: 1 May 2024

Abstract: The increasing burnup in a Mixed Oxide (MOX) fuel has necessitated a revisit of the strategic approaches of the fuel cycle development in Light Water Reactors (LWRs) using the MOX fuel option. This paper therefore studied the improvements in the MOX fuel utilization in LWRs. Different trends of the fuel cycle development were considered by the type of reactor used. The research is based on the experiences accumulated on the typical standard enriched uranium fuels in LWRs, as well as the lessons learned from the plutonium utilization in VVER reactors. Using the reactor-grade plutonium and depleted uranium, the partial reloading method was used and the model simulated in GETERA-93 from the beginning of the campaign and at the end of the campaign, considering the multiplication factor of both natural uranium fuel and that of reactor-grade plutonium at different enrichments. Safety parameters such as Fuel Temperature Coefficients (FTC), Moderator Temperature Coefficients (MTC), Moderator Density Coefficients (MDC), and Reactivity were calculated. From the values obtained, it was observed that these parameters show full agreement with the reactor safety requirements which gives the model validity.
Keywords: MOX fuel, Enrichment, Utilization.

1 Introduction

Mixed-oxide fuel has been widely utilized in the nuclear field since the 1980s, although it is still less widely used than traditional conventional uranium dioxide (UO2) fuels.[1]

Mixed oxide fuel refers to nuclear fuels consisting of UO_2 and PuO_2. MOX fuel was initially designed for use in liquid metal fast breeder reactors and in light water reactors (LWRs) when reprocessing and recycling of the used fuel is adopted. The UO_2 content of MOX may be natural, enriched, or depleted uranium, depending on the application of MOX fuel. In general, MOX fuel contains between 3% and 5% PuO_2 blended with 95–97% natural or depleted.

The Russian efforts are focused on using WG MOX in both fast (BN-600) and light-water reactors (VVER-1000); consequently, the certification of computational codes, the design of MOX-fuel assemblies, and the core configurations are crucial. Valuable benchmarking efforts have been made by OECD/NEA experts to advance the process of code certification.[1] Benchmarks are used to validate computer codes and test nuclear data libraries. Also, they play as an educational tool for engineers to be capable of performing neutronic calculations for nuclear reactors. The Russian Federation and the United States have studied the modifications that may be required to use MOX fuel in nuclear reactors.[2] Our work outlines the fuel composition changes that occur in an operational VVER-1200 reactor and the significance of mixed oxide fuel that lies in its ability to easily increase burn-up and fissile concentration by adding a small amount of plutonium while enriching uranium to higher levels of U-235 is comparatively more costly.

2 VVER-1200 Reactor

The VVER reactor is a Russian-designed pressurized light water-cooled and moderated reactor. It was created before the 1970s and has been updated regularly. It now refers to a wide range of reactor designs, ranging from generation I reactors to the latest generation III+ reactors. VVER technology was unquestionably important in the development and growth of the nuclear power sector in Russia and across the world. [3]

In 1964. The first VVER nuclear reactor was installed at the Novovoronezh nuclear power station. The VVER-210 and VVER-365 power reactors were the first two power reactors built in this location (the numbers originally referred to electrical output). The successful commissioning

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and operation of these early reactors paved the path for more efficient reactors to be built. The VVER-440 was the first VVER to be built in a serial format. For better heat retention [4]

The VVER-440 model V-230 used six primary coolant loops, each with a horizontal steam generator, to improve heat transfer. While working together, they were able to have a large volume of coolant. It also contained isolation valves, which allowed one or two of the plant's six coolant loops to be repaired while it was still running. The model's principal flaws were the lack of emergency core-cooling systems or supplementary feedwater systems, as well as the accident localization system's layout, which operated as a reactor containment. [3]

The VVER-1200 is the most recent construction variant, and it is an upgrade of the VVER-1000 with a higher power output of approximately 1200 MWe (gross), improved plant performance,

and more passive safety features. [3] One of the key aims of the VVER-1200 design was to lower costs while boosting safety without modifying the nuclear steam supply system's core architecture. Additional passive safety mechanisms were built to control mishaps that happened outside of the design foundation, and thermal power was boosted to 3200 MW. [6]

The NPP-2006 project aims to improve customer appeal for dependability, maneuverability, and maintainability while also increasing safety, economic competitiveness, and consumer appeal for reliability, maneuverability, and maintainability. Primary and secondary circuit parameters have improved. Among the VVER-1200's putative safety benefits over other reactors are a passive decay heat removal system, a passive containment cooling system, and a passive hydrogen removal system.[6]

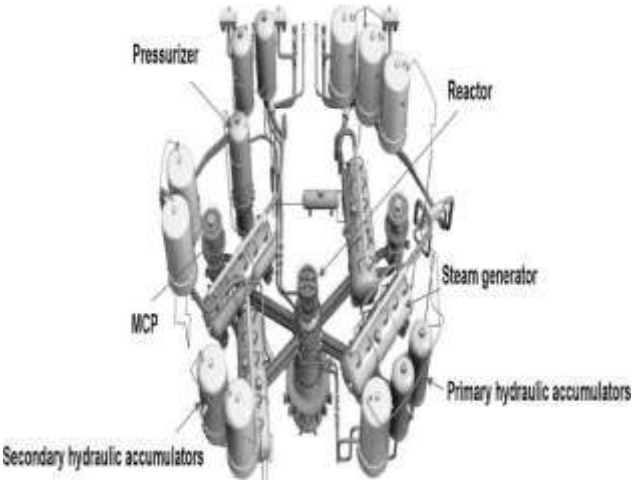

Fig. 1: Main Components of the VVER-1200 reactor Plant.

Reactor Core and Fuel Assembly

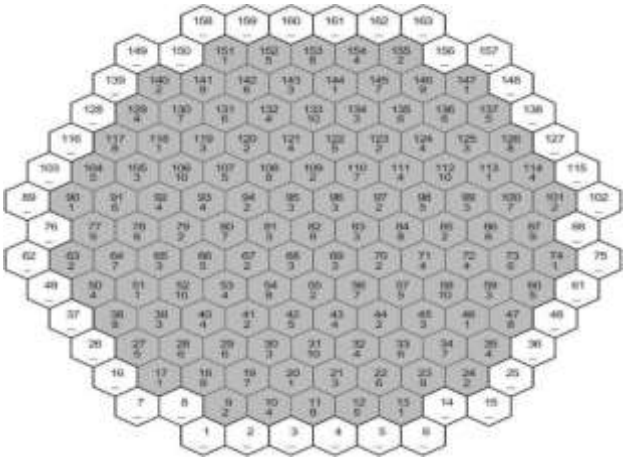

Fig. 2 : The core configuration of the VVER-1200 reactor

Recycling Normal spent nuclear fuel

Mining and milling, conversion, enrichment, and fuel manufacturing are all steps in the process of producing uranium nuclear fuel for use in nuclear reactors. These processes make up the front end of the nuclear fuel cycle, often known as the open nuclear fuel cycle. After 3-4 years of producing energy with uranium in a reactor, the spent fuel may undergo a variety of operations before being disposed of, including temporary storage, reprocessing, and recycling. All of these phases are referred to as the back end of the fuel cycle, or closed nuclear fuel cycle. [15]

The fissionable ^{235}U level has been reduced to less than 1% in spent nuclear fuel, which includes roughly 96% of its original uranium. Around 3% of the spent fuel is waste, with the remaining 1% made up of plutonium (Pu) generated while the fuel was in the reactor. The neutron transmutation of ^{238}U produces a large amount of plutonium in commercial LWRs.

Reprocessing removes uranium and plutonium from waste products by chopping up the fuel rods and soaking them in acid to separate the individual components (and from the fuel assembly cladding). It permits uranium and plutonium to be recycled into new fuel and produces significantly less waste (compared with treating all used fuel as waste). The remaining 3% of high-level radioactive waste can be held as a liquid and then solidified and disposed of in this manner.[16]

Table 1. Isotopic compositions of recycled plutonium

Isotopes	Pu238	Pu239	Pu240	Pu241	Pu242
Share, wt%	2.30	56.35	21.42	13.21	6.72

Table 2. Nuclear properties of main fissile materials

Parameter	^{239}Pu	^{235}U
Thermal absorption cross-section [barns]	1015	678
Thermal fission cross-section [barns]	741	577
The average number of neutrons per fission	2.86	2.42
Energy per fission [MeV]	198.5	192.9
Delayed neutron fraction	0.0021	0.0065

The following are some of the issues with using MOX instead of UO2 as fuel:

- The presence of plutonium hardens the neutron spectrum, resulting in a shorter neutron lifetime and a lower delayed neutron fraction, perhaps allowing the reactor to reach criticality quicker. [22]
- Because many plutonium isotopes have bigger absorption cross-sections than uranium isotopes, plutonium fuel has a bigger one-group cross-section than uranium fuel.
- The actinides' fission-to-capture ratios at thermal energy are reduced as a result of their harsher neutron spectra and increased one-group absorption rates (mostly of ^{238}U, ^{240}Pu, and ^{242}Pu). Actinides, which produce energetic alpha particles, accumulate to a greater extent in a MOX fuel, resulting in higher radioactive decay heat removal requirements.

The higher thermal fission and absorption cross-sections of ^{239}Pu in MOX fuel have two key effects:

- The pins at the MOX/UO_2 interfaces have a lot of power peaking, and they're a pain to work with.
- The heat flux of MOX assemblies is smaller than that of LEU assemblies.

The first impact, power peaking, is minimized by proper assembly design and is caused mostly by plutonium's larger fission cross-section. The pins are usually positioned in concentric rings with varying plutonium loadings in most MOX assemblies, which decreases the peaking effect. The second of the aforementioned flaws deserves a more in-depth explanation. MOX assemblies have a much lower thermal neutron flux than UO_2 assemblies, as well as a marginally lower fast neutron flux. The fast-to-thermal ratio of MOX is about double that of UO_2. Table contains information relating to power peaking.[20]

Table 3. Fast and thermal neutron flux in UO2 and MOX fuel assemblies

Parameter	MOX fuel assembly	UO_2 fuel assembly
Fast neutron flux (> 0.625 eV)	8.08·1013	8.31*1013
Thermal neutron flux (< 0.625 eV)	4.67*1012	9.95*1012
Fast/thermal neutron flux ratio	17.29	8.35

These second effect facts produce a one-of-a-kind phenomenon. One of the most important is the lowering in the reactivity of neutron-absorbing materials. which is used to offset the burnup of the fuel and the burnable absorber, is reduced because it is a thermal absorber. The gadolinium enrichment must be raised since MOX loading decreases the shutdown margin and scram efficiency.

To address this issue, the gadolinium isotope's concentration or gadolinium enrichment must be increased. Increasing the concentration can make it difficult to adjust the gadolinium levels and increase the amount of water and chemicals that must be processed; on the other hand, enriching the gadolinium, which can solve these operational issues, is very expensive and may necessitate plant modifications to recover the gadolinium.[23]

In the event of a reactor scram, the number of control rods is crucial in defining the shutdown margins as well as the reactivity insertion rates. As a result, because MOX loading lowers shutdown margin and scram efficiency, gadolinium enrichment must be increased.

Higher absorption cross-sections of the principal plutonium isotopes in the epithermal energy spectrum resulted in the following consequences:

- The dangers of xenon poisoning and xenon transient-induced power distribution oscillations are decreased.
- Control clusters are inefficient, and control worthy are often lower.

Soluble gadolinium has a substantially lower value.

The reactor pressure vessel fluence

As a result of the harder spectrum and increased fast neutron flux, the pressure vessel's fast fluence has the potential to increase. A higher fluence can lead to increased embrittlement. The tables below show that proper design is needed to remove the risk of the pressure vessel's excessive fluence. [4]

Table 4. The surveillance assembly's fast neutron flux results

	Lower level		Upper level	
	UO_2	MOX	UO_2	MOX
Maximum	287	285	104	104
Minimum	164	163	64.6	64.2

Table 5. Fast neutron flux results at the pressure vessel.

	Inner surface	Quarter of thickness	Outer surface
	UO_2	UO_2	UO_2
Maximum	31.1	22.7	3.90

Minimum	**16.4**	**12.3**	**2.67**

Noted: The neutron flux in the tables given in (109n/cm2s)
The neutron flux of the MOX core is slightly higher than that of the UO2 core. Both forms of fuel had nearly identical neutron fluxes in the specimens. The careful design of the core loading patterns resulted in negligible fluctuations in neutron flux on the pressure vessel. The MOX core's outer ring is made up of uranium assemblies and the highest burnt assemblies. Although this ring is closest to the pressure vessel and has the most impact on it, the assemblies that occupy it are identical for both types of fuel assemblies. All of these facts would exert a noticeable impact on the amount of MOX contained in a reactor core.

3 Methodologies

Using GETERA-93 simulation for uranium cycle

The lattice of cell cycles of uranium dioxide is used as input for GETERA. The prolongation of the burnup cycle is found in the simulation. The first burnup cycle time must not be more than 20 days related to ^{35}Xe which increases the reactivity of the reactor. The campaign time is four times the time cycle which is the time for spent fuel. To determine the duration of the one fuel cycle time, it is necessary to perform a calculation without reloading. It should be remembered that the condition for the end of the cycle from the neutron-physical point of view is the equality K∞ =1.03. After the simulation, we got the burnup cycle time. In this investigation, we got a cycle burnup time (Tc) of 344 days. The campaign time is 1376 days. After four cycles of burn-up of the UO2 fuel, it will be discharged from the reactor as spent fuel. Then its plutonium composition is used to fabricate MOX fuel by closing the fuel cycle. The MOX fuel is made from plutonium from spent nuclear fuel and depleted uranium. The enrichment of depleted uranium is 1%. The plutonium isotopic composition or Pu vector in discharged UO2 fuel after four cycles of burn-up is given in the table below.

Table 6. Plutonium Isotopes Composition after four cycles of burn-up

Isotopes	**Pu238**	**Pu239**	**Pu240**	**Pu241**	**Pu242**
Share, wt%	**2.30**	**56.35**	**21.42**	**13.21**	**6.72**

.
The neutron infinite multiplication factor K∞ is the main value characterizing the change in the neutron flux. The efficiency of the reactor is determined by the possibility of obtaining maximum energy production per unit mass of loaded nuclear fuel. This efficiency is characterized by fuel burn-up. The maximum burnup obtained is 15.483 MW*day/kg for one cycle of time and 62.566 MW*day/kg for the core after four cycles of burn-up.

Analysis of Results

The modeling findings for VVER-1200 reactors that use mixed-oxide (MOX) fuel containing reactor-grade plutonium and depleted uranium oxide will be discussed in this chapter. We will discuss, compare, and analyze the behavior of the multiplication factor (K) and selected isotopes in UO2 and MOX fuel; investigate the influence of MOX fuel on the delayed neutrons fraction in the reactor; assess changes in safety parameters; and observe plutonium vector degradation in MOX fuel.

Table 7. Calculation Conditions for spent fuel properties.

Fuel Type and Enrichment	**UO2. MOX & 4.69 wt%**
Burn-up	**40 – 60 MWd/kg**

Loading of uranium fuel

The nuclear density (ρ) or nuclear concentration of the corresponding type of nuclei in the medium in (number/barn·cm) is calculated from the formula

$$\rho = x \cdot \gamma \cdot NaM$$

Where,
x - enrichment of uranium
γ - mass density in *g/cm3*
Na - Avogadro number
M - the molar mass of the isotope in *g/mol*.

For simulating our model, we calculated the isotopic concentrations of fuel, moderator, and cladding materials by using formula and values from Appendix A. We got the nuclei concentration for uranium fuel and water. The average mass density of the UO2 fuel pellet is 9.015 g/cm3 considering the central hole and gas gap. The water has a mass density of 0.75 g/cm3. The cladding material is Э110 alloy which contains 99% zirconium and 1% niobium. The average temperatures for neutron calculation for VVER-1200 are given in the table
Isotopic concentration of UO_2 fuel

Table 8. Isotopic concentration of Uranium Oxide fuel.

	Fuel	**Cladding material**	**Coolant**
Average temperature(K)	**1000**	**600**	**579**

Table 9. The average temperature for neutron calculation.

	Fuel			Water	
Isotope	**^{235}U**	**^{238}U**	**O**	***H***	***O***
Concentration (number/barn.cm)	**9.9235*10^{-4}**	**1.9125*10^{-2}**	**4.0213*10^{-2}**	**5.0273*10^{-2}**	**2.4128*10^{-2}**

Comparison and analysis of the results

In this section, we compare results from GETERA files. For better analysis. Comparing the uranium oxide with 4.4% enrichment and mixed oxide fuel and see the difference and the importance of each.

Comparing Uranium Oxide and Mixed Oxide Fuel

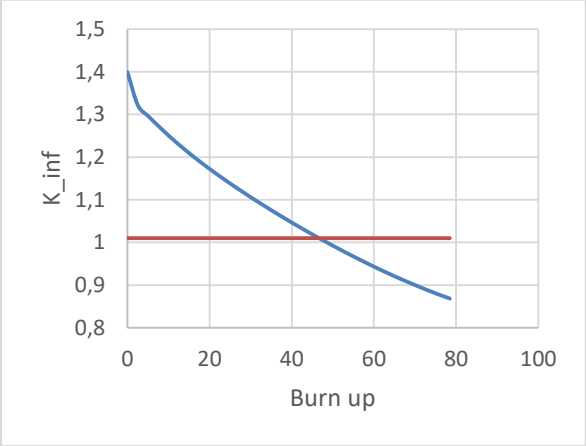

Figure 3

This graph shows the uranium oxide with 4.69% enrichment, the highest burnup achieved here is 40MWd/kgt with the multiplication factor value of 1.4.(K_∞ = 1.4).

Table 10: Composition of Depleted Uranium and Plutonium for MOX fuel

	235 U	238 U	238 Pu	239 Pu	240 Pu	241Pu	242 Pu	Total
Concentration	**1.59 E-04**	**1.44 E-02**	**2.31 E-06**	**8.88 E-04**	**5.15 E-05**	**1.57 E-04**	**8.81 E-05**	**1.58 E-02**
Percentage %	**1.01 E+00**	**9.15 E+01**	**1.46 E-02**	**5.63 E+00**	**3.27 E-01**	**9.93 E-01**	**5.59 E-01**	**1.00 E+02**

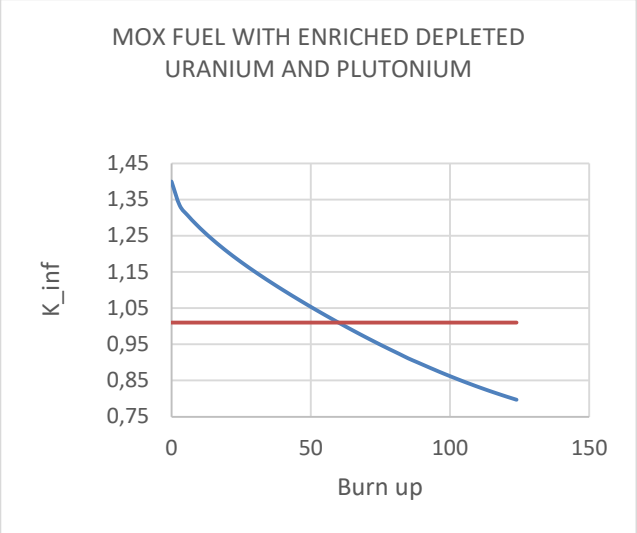

Figure – 4

The major advantage of MOX fuel is that its allows you to improve burn-up from 40 to 60 MWd/kg, using the initial enrichment, and from this we know that the more we increase the enrichment the more the burn-up.

Reactivity Coefficients and Effects

When the reactor is in operation the fuel amount decreases continuously. So, to maintain the criticality for a long time, excess fuel is loaded in the core. This excess fuel results in excess positive reactivity at the beginning of the cycle. Thus, negative reactivity is introduced to the core. Materials having high neutron absorption cross-section that convert into relatively low absorption cross-section material after neutron absorption is called burnable absorbers.

The introduced negative reactivity for controlling the criticality of the core is a prime concern that can be done by moving control rods with neutron-absorbing materials. A negative reactivity coefficient for fuel or coolant allows the reactor's power to be controlled throughout possible increases, which is critical for the reactor's safety. In reactors using depleted fuel, the presence of a sufficiently significant negative power reactivity coefficient safeguards the core from damage in the event of uncontrolled rises in the neutron flux in potential reactivity accidents.

Reactivity feedback are changes in reactivity caused by changes in any operational parameters, such as fuel temperature, coolant temperature, coolant density, and so on, and are characterized by reactivity coefficients. Because reactivity feedbacks affect the reactor's stability, this is a critical aspect of reactor design. Assuming that the reactor's initial state is critical, we have:

$$\alpha_i = \frac{\partial \rho}{\partial x_i} = \left|\rho = 1 - \frac{1}{K_{eff}}\right| = \frac{1}{K_{eff}} \frac{\partial K_{eff}}{\partial x_i} = \frac{1}{K_\infty \cdot P}\left[\frac{\partial K_\infty}{\partial x_i} \cdot P + K_\infty \frac{\partial P}{\partial x_i} = \frac{1}{K_\infty}\frac{\partial K_\infty}{\partial x_i} + \frac{1}{P}\frac{\partial P}{\partial x_i}\right. \quad (4.1)$$

The infinite multiplication factor is only a fitting representation for a reactor that is infinitely large because it assumes that no neutrons leak out of the reactor. Because the neutron life cycle has to include those neutrons that leak out of the core, a full description of the neutron life cycle must be in a functional reactor. To factor in leakage and absorption, the effective multiplication factor (K_{eff}) is used. This effective multiplication factor is defined as the ratio of the total number of neutrons created during one fission cycle to the number of neutrons lost through absorption and leakage during the preceding fission cycle. [36] And we have

$$K_{eff} = K_{\infty} \cdot P = \frac{K_{\infty}}{1+M^2B^2}$$

(4.2)

Where, αi – reactivity coefficient; xi – operating parameter; P – leakage escape probability; M^2 – migration area of the neutron; B^2 – geometrical buckling of the reactor; K_{eff} – effective multiplication factor; K_{∞} - infinite multiplication factor

Geometric buckling (B^2) is a parameter of neutron leakage and depends on the shape or geometry of the reactor. For a cylindrical shaped reactor with radius R and height H, we can write:

$$B^2 = \left(\frac{\pi}{H}\right)^2 + \left(\frac{2.405}{R}\right)^2 \quad (4.3)$$

For VVER-1200. H = 3.75 m and R = 1.58 m;

$$B^2 = \left(\frac{\pi}{375}\right)^2 + \left(\frac{2.405}{158}\right)^2 = 3.01878 * 10^{-4}\ \frac{1}{cm^2} \quad (4.4)$$

Migration length squared (M^2) is the square of the average distance between the neutron's birth point mainly fast neutrons and the point where the neutron becomes thermal. [37] It is the summation of Fermi-age and diffusion length squared of thermal neutrons.

$$M^2 = \tau + {L_{th}}^2 \quad (4.5)$$

$$\tau = \frac{D_1}{\Sigma_a^1+\Sigma_s^{1\to2}} \quad (4.6)$$

$${L_{th}}^2 = \frac{D_2}{\Sigma_a^2} \quad (4.7)$$

Where, τ – Fermi-age

L_{th}^2– diffusion area

D_1, D_2 – Diffusion coefficient for the first and second group of neutron energy, respectively.

Σ_a^1, Σ_a^2 – absorption macroscopic cross-section of the first and second group of neutron energy.

$\Sigma_s^{1\to2}$ – Scattering cross-section from the first group to the second group.

The energy ranges in the first and second groups are 10.5 MeV - 2.15 eV and 2.15 eV - 0 eV respectively.

We can simulate the GETERA by changing the temperature of fuel and coolant as well as the mass density of the coolant. After simulation, we can get the values of the macroscopic parameters D_1, D_2, Σ_a^1, Σ_a^2 and $\Sigma_s^{1\to2}$ from the GETERA output file.

Fuel temperature coefficient (α_f)

The fuel temperature coefficient is the change in reactivity due to a change in fuel temperature. The fuel temperature coefficient (α_f) is known as Fuel Temperature Coefficient or Doppler Coefficient. The mathematical expression of this coefficient is given:

$$\alpha_f = \frac{\partial\rho}{\partial T_f} = \frac{1}{K_{\infty}}\frac{\partial K_{\infty}}{\partial T_f} + \frac{1}{P}\frac{\partial P}{\partial T_f}$$

(4.8)

The fuel temperature coefficient (α_f) does not depend on leakage escape probability P, as a result, we can ignore the second part of the equation (4.5) and get:

$$\alpha_f = \frac{1}{K_{\infty}}\frac{\partial K_{\infty}}{\partial T_f} = \frac{1}{K_{\infty}}\frac{\Delta K_{\infty}}{\Delta T_f}$$

(4.9)

Coolant temperature coefficient (α_c)

Coolant temperature coefficient (α_c) is the change in reactivity per degree change in moderator/coolant temperature. A negative temperature coefficient is a desirable feature because it has the self-regulatory effect of control of reactivity. The coolant temperature coefficient (α_c) can be written as:

$\alpha c = \partial\rho/\partial Tc$

As we know that if the coolant temperature changes, the density of the coolant also changes. Thus the reactivity ρ is a function of two variables:

$\rho = f[T_c, \gamma(T_c)]$.

Hence,

$$\alpha_c = \frac{\partial\rho}{\partial T_c} = \frac{\partial\rho}{\partial T_c}\Big| + \frac{\partial\rho}{\partial\gamma_c}\Big|\frac{\partial\gamma_c}{\partial T_c} \quad (5.0)$$

γ = const. T_c = const.

Where,

$\partial\gamma_c/\partial T_c$ = const. = $-1.73*10^{-2}$ g/(cm^3·K)

$\partial\rho/\partial\gamma_c = \alpha\gamma$; Coolant density coefficient at the constant coolant temperature

$\partial\rho/\partial T_c$ = Coolant temperature coefficient at constant coolant density

Now, we can calculate and analyze reactivity coefficients associated with fuel temperature, coolant temperature, and density change at end of the cycle (EOC) at full power of the reactor. The boric acid is used as a burnable absorber for the compensation of reactivity reserve for fuel burnable, and equalization of energy release in the core and optimizes fuel utilization when added in the coolant at the beginning of the cycle (BOC) for safety in light water reactors. So, at the beginning of the cycle (BOC), we can analyze those coefficients using boric acid. For comparison purposes, we

used uranium fuel and the first case of variant three (four-cycle UO_2 and three-cycle MOX fuel).

Table 11: Safety Assessment Parameter Fuel Temperature Coefficient Af.

α_f (K^{-1})	UO_2 fuel	MOX fuel
BOC (1100 K)	-4.30E-05	-3.90E-05
MDC(1050K)	-4.44E-05	-3.68E-05
EOC (900 K)	-4.95E-05	-4.43E-05

Safety assessment parameter

This section includes the calculated results of the reactivity coefficients (α_f, α_c) for both UO_2 and MOX fuel. Also, the reactivity assessment for both fuel assemblies will be presented

Table 12: COOLANT TEMPERATURE COEFFICIENT α_c

α_c (K^{-1})	UO_2 fuel	MOX fuel
BOC (629 K)	-4.07E-04	-4.49E-04
MDC(604K)	-8.60E-04	-9.1E-04
EOC (529 K)	-4.07E-04	-4.57E-04

From the listed parameters we can see that all the coefficients agree with the criteria for safe reactor operations. For example, the fuel temperature coefficients are both negative for each case which secured one condition for safe operation.

A negative fuel temperature coefficient is considered to be the most important parameter than a negative moderator temperature coefficient. In the case of reactivity-initiated accidents, the fuel temperature coefficient of reactivity would be the most important effect in the compensation of inserted positive reactivity. The duration for heat being transferred to the moderator is generally measured in seconds whereas the fuel temperature coefficient is effective almost instantaneously. From our results, we can observe that the fuel temperature coefficients, as well as the moderator temperature coefficient, are both negative which ensures a safe operation of the reactor core. As for the reactivity, we see that reactivity at BOC is less than the cold state. It is because at cold state, ^{135}Xe concentration becomes zero and the boric acid burns out which is a usual situation.

4 Conclusions

In summary, Using reactor grade plutonium mixed with depleted uranium fuel is very important because the fissile concentration of the fuel and hence the burn-up can be increased easily by adding a bit more plutonium, whereas enriching uranium to higher levels of U-235 is relatively expensive. The most important is that mox allows you to improve burn-up from 40 to 60 MWday/kg, keeping the same initial excess of reactivity. One of the main parameters is the safety assessment parameter we can observe that the fuel temperature coefficients, as well as the moderator temperature coefficient, are both negative which ensures a safe operation of the reactor core. The reactivity coefficients were determined for every situation using the two group macroscopic cross-sections from GETERA-93. All the coefficients shows promising agreement to the safety requirements. unlike uranium oxide, the fuel cycle cost still decreases with fuel burn-up above the 60 MWd/kg. If the current reprocessing option is pursued, it is expected that global capacity for spent fuel storage will be sufficient for discharged volume over the next decade.

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