

Re-fueling of the MOX Caramel Fuel mixed with ^{237}Np Oxide in the MTR -22MW Reactor

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Abstract

In this paper, the re-fueling MTR-22 MW reactor by MOX (UO_2 & PuO_2) Caramel Fuel (CF) mixed with ^{237}Np oxide as a minor actinide was investigated. The obtained results of the criticality and neutronic parameters showed that the re-fueling MTR 22 MW reactor by the MOX CF mixed with ^{237}Np oxide does not have negative effect on the criticality and the neutronic parameters of the reactor. Re-fueling the MTR-22 MW reactor by this fuel, leads to reduce the ^{235}U loaded mass in the reactor core to 33 % and contributes to enhance the proliferation resistance of the fissile material and a bit burn of the plutonium isotopes in the MTR-22MW reactor.

Key words: MTR-22 MW reactor, MOX caramel fuel, minor actinides, criticality and .neutronic parameters, MCNP4C code

تحميل الوقود المختلط MOX على شكل قطع صغيرة و الممزوج بأوكسيد النيبتونيوم NpO_2 في مفاعل البحث MTR-22 MW

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الملخص

بحث في هذه الورقة تحميل المفاعل MTR-22 MW بالوقود المختلط MOX ($UO_2&PuO_2$) على شكل قطع صغيرة و الممزوج بأوكسيد النيبتونيوم NpO_2 كأكتينيد ثانوي قابل للإستحراق. و أظهرت نتائج حساب الوسطاء النترونية و الحرجية للمفاعل MTR-22 MW و المحمل بالوقود المختلط MOX على شكل قطع صغيرة و الممزوج بأوكسيد النيبتونيوم NpO_2 إمكانية تشغيله دون تأثير سلبي على هذه الوسطاء و أنّ تحميل المفاعل MTR-22 MW بهذا الوقود، يخفض كتلة اليورانيوم ^{235}U المحملة في المفاعل حتى % 33 و هذا يساهم في تعزيز مبدأ مقاومة الوفرة من المواد المشعة و حرق جزء صغير من نظائر البلوتونيوم في المفاعل MTR-22 MW.

الكلمات المفتاحية: المفاعل MTR-22 MW، الوقود المختلط MOX على شكل قطع صغيرة، الاكتينيديات الثانوية، الوسطاء الحرجية و النترونية و الكود MCNP4C.

Introduction

Energy generation by nuclear reactors entails production of plutonium and radioactive waste such as: fission products and Minor Actinides (MAs). To utilize this plutonium and to minimize the long-term radiotoxic wastes re-cycling of plutonium and MAs in the Light water reactors (LWRs) becomes a beneficial option. The major benefit of beginning plutonium and MAs re-cycling in the existing thermal reactors is that it would reduce the residual toxicity and the subsequent life time of the final waste. Also, it provides an effective burning of fissile plutonium and avoids its accumulation in the spent fuel stockpiles and so decreasing the risk of diversion. This contributes to enhance the proliferation resistance, and improve the fuel cycle performance [1], [2], [3], [4], [5].

All published studies up to now are investigating re-cycling of plutonium and MAs in LWRs and Fast Reactors (FRs) without mentioning Research Reactors (RRs) having medium or high power such as: MTR reactors with 22 MW and 50 MW or other reactors [1], [2], [6].

The main problem of this research

The radioactive properties and the toxicity of the MAs make them very harmful to human beings and the environment. Therefore, many studies to re-cycle the MAs in the fuel cycle of LWRs and fast reactors have been performed [2], [3], [4], [5], [7], [8]. The results of these studies have showed the following:

- The enrichment with ^{235}U and the required amount of the fuel decrease significantly with increasing the number of fuel assemblies charged with plutonium and Mas such as Neptunium,
- Re-cycling of plutonium and MAs together with discharging uranium can reduce the radio-toxicity of discharged heavy metal (HM) waste to become less than that of the loaded uranium, and to reduce the spent fuel for storage and improve the proliferation resistance of the radioactive materials,
- Re-cycling MAs and UO_2 fuel together can serve as burnable absorber to reduce the initial excess reactivity.

Since re-cycling RRs by MOX mixed with MAs seems not to be investigated, therefore, it is very important to investigate the re-fueled and effect of the MAs such as Plutonium and Neptunium on the neutronic parameters of the RRs. This helps to improve the life of the fuel cycle and re-design the RRs using fuels with MAs and using closed fuel cycle in the future; this leads the burn the MAs resulting from these reactors and remove their risks.

The importance of this research

The major benefit of beginning minor actinide recycling in existing thermal reactors and in the RRs having medium power is that it would remove the long-lived radiotoxic nuclides from the fission product waste, helps in saving energy and enhancing the proliferation resistance of radioactive materials and serving as a burnable absorber to improve the fuel cycle performance. In addition, the advantage of re-cycling MOX fuel with MAs in RRS is to help in:

- reducing the volume of the spent fuel for storage and enhancing the proliferation resistance, especially the length of the fuel cycle in these reactors is short in comparison with that of LWRs which means burning more plutonium and MAs,

- contributing to extend the spreading of RRs which use MOX fuel with minor actinides or original fuel with some fuel elements of MOX fuel mixed with minor actinides,
- increasing plutonium isotopes utilization,
- reducing the enrichment of the fuel with ^{235}U loading in the reactor core,
- transmuting to less hazardous and possibly more useful forms.
- using a closed fuel cycle for RRs in the future.

Therefore, this paper will investigate the re-fueling of the MTR-22 MW ((the Egyptian Second Research Reactor (ETRR-2) as example) reactor by the MOX (UO_2 & PuO_2) CF mixed with ^{237}Np oxide without changing the dimensions of the fuel material, the Fuel Plate (FP) and the Fuel Element (FE), the control plates and the reactor core or any component of the reactor. The modified ETRR-2 core should maintain the already available facilities, the neutronics parameters as the U_3O_8 -Al original fuel. The criticality and neutronics parameters of the ETRR-2 core fueled by the U_3O_8 -Al original fuel and re-fueled by the MOX CF fuel mixed with ^{237}Np oxide were estimated using the MCNP4C code [9].

Methodology

.1 The ETRR-2 reactor

The ETRR-2 core consists of 30 position, 29 position for the FEs (Standard F, Fuel type1 and Fuel type 2, See Table 1) and one position for the Central Neutronic Trap (CNT). Each FE has 19 FPs separated from each other by a 0.27 cm coolant channel. The reactor uses U_3O_8 -Al fuel with 19.70% ^{235}U enrichment. The reactor power is 22 MW with high thermal neutron flux in the CNT ($>10^{14}$ n/cm².s). The active zone of FP dimensions is 80 cm length, 6.4 cm width and 0.07 cm thickness. The main specifications of the fuel material, the FP, the FE, the absorber material, the active zone dimensions and the water gap between plates are given in Table 1 and Table 2 [10], [11], respectively.

Table 1: Composition of the fuel in a typical ETRR-2 core.

Parameter	Weight percentage %		
	SFE	FE Type 1	FE Type 2
^{235}U	12.377	6.598	8.398
^{238}U	50.450	26.894	34.230
^{27}Al	٢٥,٩١	60.504	49.730
^{16}O	11.263	6.004	7.642
Density (g/cm ³)	٤,٨٠٢	3.299	3.655

Table 2: Specifications of the fuel material, fuel element, absorber material, active zone dimensions and water gap between plates of the ETRR-2 core for the U_3O_8 -Al OFEs and the U_3O_8 -Al OFEs mixed with ^{239}Pu , ^{241}Am and ^{243}Am actinides.

Parameter	Fuel material			
	U_3O_8 -Al original fuel		MOX (UO_2 & PuO_2) mixed with NpO_2	
Fuel meat	U_3O_8 -Al		MOX + NpO_2	
Weight percentage (%)	100		97.2 (UO_2) + 2.7 (PuO_2) + 0.1 NpO_2	
Enrichment ^{235}U (%)	19.70		2.9	
Initial mass of ^{235}U loading in the reactor core (g)	6944.5		5197.16	
Density of fuel (g/cm^3)	See Table 1		10.25	
Fuel element				
Number of the fuel materials	29		29	
Number of fuel plates in the fuel material	19		19	
Dimensions of the fuel plate (cm) (length x width x thickness)	80 x 6.40 x 0.07		80 x 6.40 x 0.07	
Absorber material				
Composition	Ag - In - Cd		B_4C	
	Weight percentage (%)		Weight percentage (%)	
	Ag	80	^{10}B	64
	In	15	^{11}B	16
	Cd	5	C	20
Density (g/cm^3)	10.18		2.02	
Active zone dimensions				
Active length (cm)	80		80	
Clad length (cm)	80		80	
External section of fuel element (cm^2)	8x8		8x8	
Section in grid to house the fuel element (cm^2)	8.1 x 8.1		8.1 x 8.1	
Plate thickness (cm)	0.10		0.10	
Meat thickness (cm)	0.07		0.07	
Meat width (cm)	6.40		6.20	
Side plate thickness (cm)	0.50		0.50	
Side plate width (cm)	8.00		8.00	
External distance between frames (cm)	8.00		8.00	
Internal distance between frames (cm)	7		7	
Cladding material	Al- 6061		Zircaloy-4	
Water gap between plates				
of single fuel element (cm)	0.27		0.27	
of different fuel element (cm)	0.39		0.39	

The reactor is used: to perform neutron activation analysis, radioisotope production (e.g., ^{14}C , ^{32}S , ^{51}Cr , ^{60}Co , ^{89}Sr , ^{153}Sm , ^{169}Yb , ^{170}Tm , ^{131}I , ^{125}I , ^{32}P , ^{192}Ir and ^{99}Mo) and other scientific applications. The ETRR-2 reactor is cooled and moderated with light water, reflected by beryllium and controlled by 6 plates made of a Ag-In-Cd alloy [10], [11], [12], [13], [14].

2 Simulation of the ETRR-2 reactor using MCNP4C code

The ETRR-2 reactor was simulated using the MCNP4C code. The cross-sections of the ETRR-2 core which comes out from the MCNP4C code are shown in Figure 1 and Figure 2, where Figure 1 and Figure 2 represent the 1998/1 core and the current ETRR-2 core, respectively [13], [15].

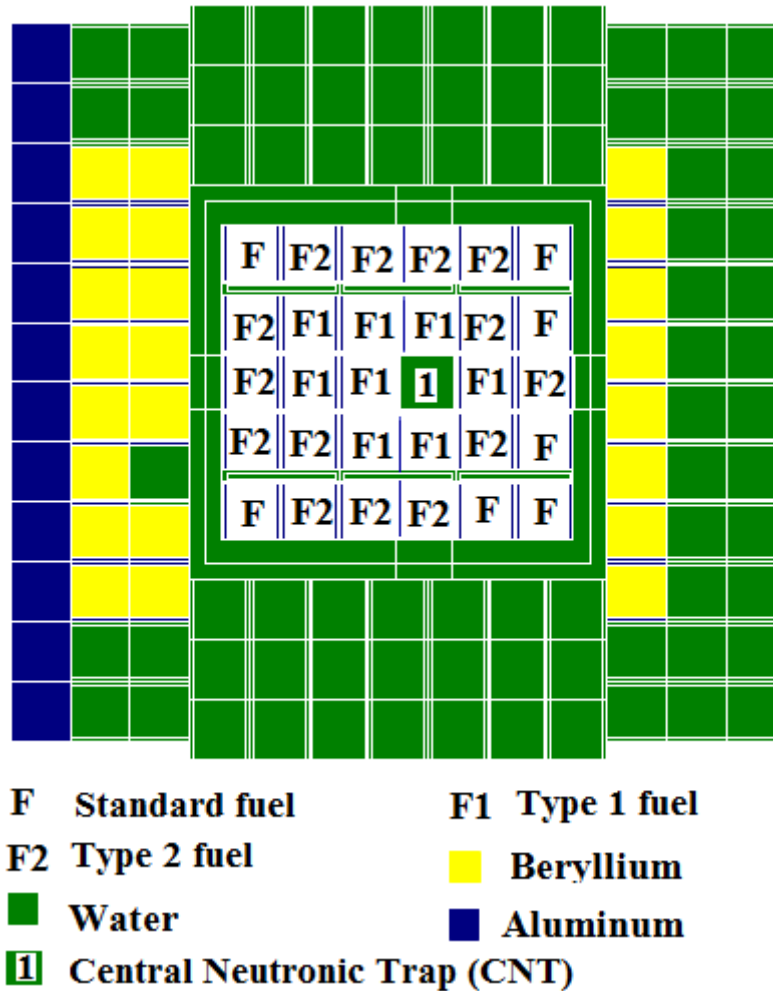


Figure 1: A cross section of the 1998 ETRR-2 core in the plane X–Y with CNT using the MCNP4C code.

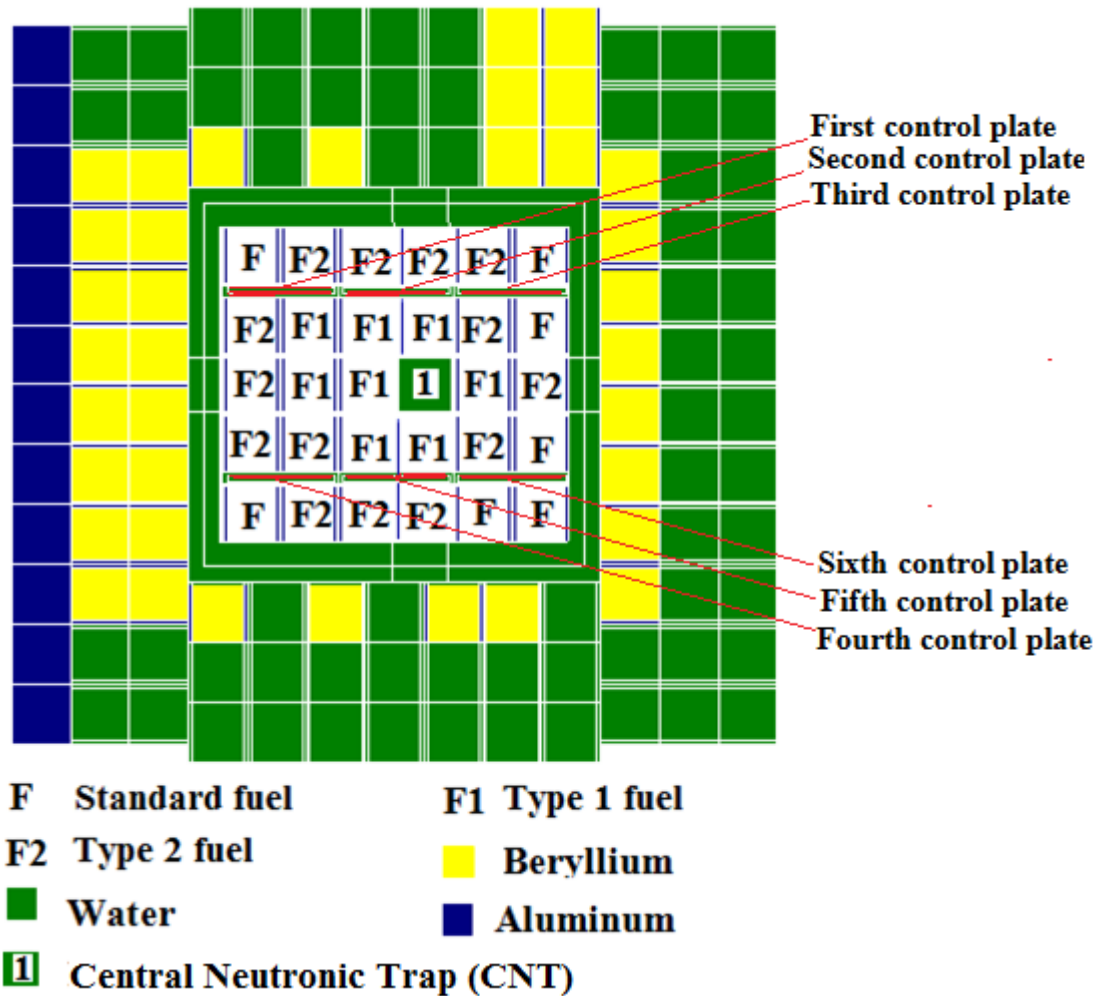


Figure 2: A cross section of the current ETRR-2 core in the plane X-Y with CNT using the MCNP4C code.

The cross-sections of the fuel material in the FPs consisting of the FEs in the ETRR-2 core using the MCNP4C code are shown in Figure 3, Figure 4 and Figure 5, where:

- Figure 3 shows the fuel material consisting of one piece with 0.07 cm thickness, 80 cm length and 6.4 cm width for the U_3O_8 -Al original fuel. The fuel material is covered with two Al-6061 plates to form the FPs for the U_3O_8 -Al original fuel,

- Figure 4 shows the fuel material for the MOX (UO_2 & PuO_2) CF mixed with ^{237}Np oxide as a minor actinide. In this case the fuel material in the FPs was divided into eighty small pieces, each piece has dimensions: 2 cm (length) x 3.2cm (width) with a 0.07 cm thickness. Also, the fuel material is covered with two zircaloy-4 plates to form the FPs for the MOX CF mixed with ^{237}Np oxide.

- Figure 5 presents a cross section of the fuel element used in the ETRR-2 reactor using the MCNP4C2 code.

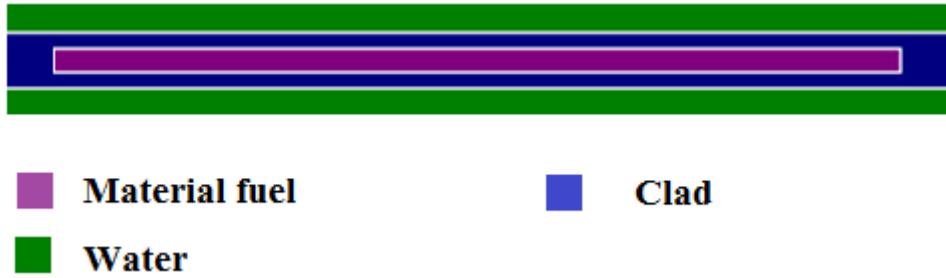


Figure 3: A cross section of the FP used in the ETRR-2 reactor fueled by the U_3O_8 -Al original fuel using the MCNP4C code.

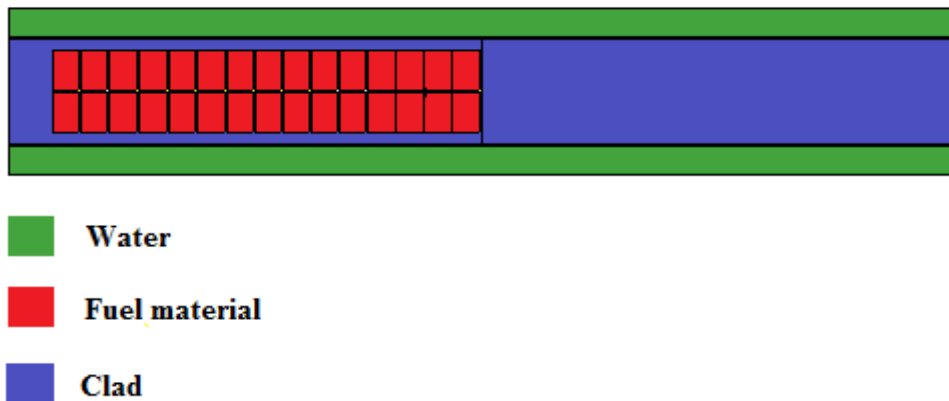


Figure 4: Schematic representation of the FP used in the ETRR-2 reactor re-fueled by the MOX CF mixed with ^{237}Np oxide using the MCNP4C code.

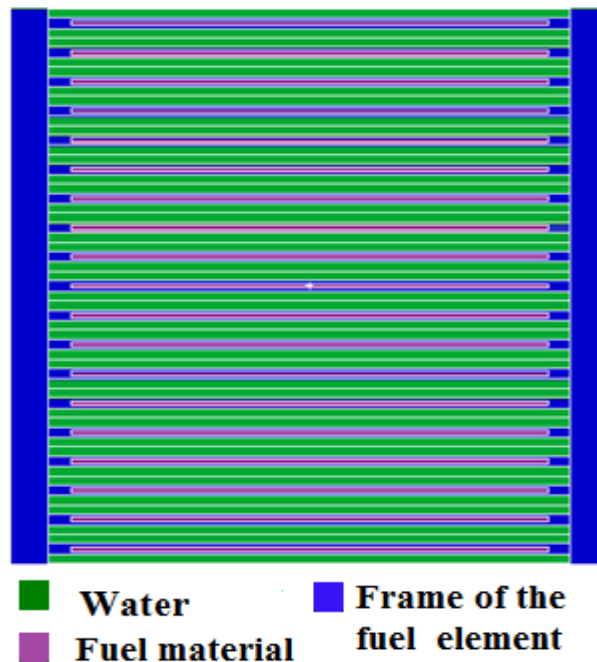


Figure 5: A schematic horizontal cross section of the fuel element used in the ETRR-2 reactor fueled by the U_3O_8 -Al original fuel, and re-fueled by the MOX CF mixed with ^{237}Np oxide using the MCNP4C2 code.

The main properties of the U_3O_8 -Al original fuel and the MOX CF are given in Table 2. The UO_2 CF was used in some French research reactors such as OSIRIS reactor [16], [17]. The composition of the PuO_2 [6] and ^{235}Np oxide is listed in Table 3 and Table 4, respectively.

Table 3: Composition of the plutonium isotopes.

Parameter	Percentage %
^{238}Pu	1.81
^{239}Pu	59.14
^{240}Pu	22.96
^{241}Pu	12.13
^{242}Pu	3.96

Table 4: Composition of the ^{237}Np oxide.

Parameter	Weight percentage %
^{237}Np	88,104
^{16}O	11,896

The MCNP4C model of the ETRR-2 core was used to estimate the neutronics parameters of the ETRR-2 core before and after replacing its U_3O_8 -Al original fuel by the MOX CF mixed with ^{237}Np oxide.

To evaluate these parameters:

1. the MCNP4C code was run for three hundred million neutron histories (10^6 particles and 300 criticality cycles) with an initial criticality k_{eff} guess of 1 and thirty passive cycles), and using all FEs as fission source points, where the fission source is located in the middle of each FE.
2. the ENDF/B-VI as a nuclear data source for the fissile and the non-fissile materials, and the thermal particle scattering $S(\alpha, \beta)$ to treat the thermal scattering in both beryllium reflector and hydrogen of the moderated water were used,
3. the following conversion factor was used as described in the MCNP4C manual and presented herein to get the neutron source strength of the reactor which is used in the calculation of the neutronics parameters of the ETRR-2 reactor [9]:

$$C = \frac{P(\text{watt}) \cdot \tilde{\nu}}{E(\text{MeV})} \cdot \frac{1 \text{ joule/sec}}{\text{watt}} \cdot \frac{1 \text{ MeV}}{1.602 \times 10^{-13} (\text{joules})} \quad (1)$$

Where:

$P(\text{watt})$ - is the steady state power of the reactor (22 MW),

$\tilde{\nu}$ - is the average number of neutrons released per fission (the value of the $\tilde{\nu}$ is listed in the MCNP4C output file),

$E(\text{MeV})$ - is the released energy per fission.

3. Calculation of the criticality and neutronics parameters of the ETRR-2 reactor for the ETRR-2 core fueled by the U_3O_8 -Al original fuel and re-fueled by the MOX CF mixed with ^{237}Np oxide using the MCNP4C code.

3.1 Calculation of the criticality parameters

The KCODE criticality source card [9] was used in the input file of the ETRR-2 core to calculate criticality parameters for the ETRR-2 core fueled by the U_3O_8 -Al

original fuel, and re-fueled by the MOX CF mixed with ²³⁷Np oxide. The criticality parameters include the following parameters:

1. The effective multiplication factor (k_{eff}) and the corresponding core excess reactivity (ρ) using the following equation [18]:

$$\rho = (k_{eff} - 1)/k_{eff} \quad (2)$$

2. The Shutdown Margin (SM) of the control plates,
3. The SM of the control plates with Single Failure (SM with SF),
4. The Control Rod Worth (CRW), where the CRW is defined by the following formula:

$$CRW = (k_{out} - k_{in})/k_{out} \cdot k_{in} \quad (3)$$

Where:

k_{out} - is the calculated value of the effective multiplication factor when All Control Plates Out (ACPO) are fully out the ETRR-2 core,

k_{in} - is the calculated value of the effective multiplication factor when the all control plates are inserted in the ETRR-2 core,

5. The Reactivity Safety Factor (RSF), where the RSF is defined as a ratio of the CRW to the core excess reactivity,

Table 5 shows the measured and MCNP4C results of the core excess reactivity, the SM and the SM with SF of the 1998 ETRR-2 core (See Figure 1) fueled by U₃O₈-Al original fuel, whereas, Table 6 shows the MCNP4C results of the same parameters and the CRW, and the SRF of the current ETRR-2 core (See Figure 2) fueled by the U₃O₈-Al original and re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide.

Table 5: Measured and MCNP4C results of the core excess reactivity, the SM and the SM with SF of the 1998 ETRR-2 core fueled by U₃O₈-Al original.

Core 1/98 of the ETRR-2 fueled by U ₃ O ₈ -Al original fuel with ONT (Figure 1)		
Fuel material in the fuel plate consisting of the one plate		
Parameter	Measured ^(a)	Calculated values using MCNP4C
Core excess reactivity ρ (\$)	9.1	8.964 ± 0.049
SM (\$)	15.2	15.849 ± 0.029
SM with SPF (\$)	8.7	8.929 ± 0.029

values of the core 1/98 were taken from the references [12], [13], [14].

Table 6: MCNP4C results of the core excess of reactivity, the SM and SM with SF, the CRW and the RSF of the current ETRR-2 core fueled by the U₃O₈-Al original fuel and re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide.

Current ETRR-2 core fueled by the U ₃ O ₈ -Al original fuel (Figure 2)	
Parameter	Calculated values using MCNP4C
Core excess reactivity ρ (\$)	9.376 ± 0.007
SM (\$)	15.141 ± 0.036
SM with SF (\$)	8.429 ± 0.036
CRW (\$)	24.670 ± 0.043
RSF	2.631 ± 0.004
Current ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷ Np oxide (Figure 2)	
Core excess reactivity ρ (\$)	9.026 ± 0.007
SM (\$)	15.024 ± 0.038
SM with SF (\$)	8.069 ± 0.038
CRW (\$)	24.050 ± 0.042
RSF	2.665 ± 0.004

In the calculation of the criticality parameters of the modified ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide, the Ag-In-Cd alloy used as absorber material in the control plates for the ETRR-2 core fueled by the U_3O_8 -Al original fuel was changed to the B_4C material without changing the dimensions nor the position. The compositions of the Ag-In-Cd alloy and the B_4C absorbers is listed in Table 2.

3.2 Calculation of the neutronics parameters

The neutronics parameters of the ETRR-2 reactor include:

1. The Average Thermal Neutron Flux (ATNF) in the CNT which is located in site 1 as shown in Figure 2,
2. The TNF in the Central Irradiation Box (CIB) located in the center CNT and used to produce ^{60}Co for medical and scientific applications,
3. The Average TNF (ATNF) in the Be reflector.

To maintain the scientific applications of the ETRR-2 reactor, the ATNF in the CNT and in the Be reflector, and the TNF in the CIB of the modified ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide should have the same order of the ATNF in the CNT and in the Be reflector, and TNF in the CIB for the U_3O_8 -Al original fuel.

To calculate the neutronics parameters of the ETRR-2 core fueled by the U_3O_8 -Al original and re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide, the F_4 tally, the FS, the SD and the FM cards in the MCNP4C code were used in the input file of the ETRR-2 reactor, and then the input file was run by the MCNP4C code. The F_4 tally, the FS, the SD and FM cards are used as follows:

- F4:n - This card allows to estimate the track-length of the neutron flux in the desired cell.
- FS - This card allows to subdivide a cell or a surface into segments for tallying purposes.
- SD - This card allows to divide a volume or area into segments for tallying purposes.
- E - Energy bins in MeV.

The FM card was written in the input file as follows:

FM C,

Where:

C - is the source strength of the ETRR-2 reactor defined by equation (1) to give the normalized flux in the correct unit of neutrons/cm².s (See manual MCNP4C code [9]).

The calculated values using the MCNP4C2 code of the ATNF in the CNT and in the Be reflector, and the TNF in the CIB are tabulated in Table 7 and Table 8 for the current ETRR-2 core fueled by the U_3O_8 -Al original fuel and re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide for both the ACPO and the criticality cases.

Table 7: Reference and MCNP4C results of the neutron flux of the current ETRR-2 core fueled by U_3O_8 -Al original fuel.

Current ETRR-2 core fueled by the U_3O_8 -Al original fuel (Figure 2)		
Parameter	Reference value	Calculated value using MCNP4C for the ACPO case
TNF in the CIB (n/cm ² .s)×10 ¹⁴	4.230 ^(a)	4.215 ± 0.017
ATNF in the CNT (n/cm ² .s) ×10 ¹⁴	2.700 ^(b)	2.700 ± 0.007
ATNF in the Be reflector (n/cm ² .s) ×10 ¹⁴	1.000 ^(b)	0.988 ± 0.014

- (a) Reference value
was taken from reference [14].
- (b) Reference
values were taken from the references [20].

Table 8: MCNP4C results of the neutron flux of the ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide for the ACPO and criticality cases.

Current ETRR-2 core re-fueled by the MOX CF with 0.1% ²³⁷ Np oxide (Figure 2)		
Parameter	Calculated values for the ACPO case	Calculated values for the criticality case
TNF in the CIB (n/cm ² .s)×10 ¹⁴	4.153 ± 0.019	3.817 ± 0.020
ATNF in the CNT (n/cm ² .s) ×10 ¹⁴	2.600 ± 0.008	2.843 ± 0.008
ATNF in the Be reflector (n/cm ² .s)	0.995 ± 0.015	1.020 ± 0.015

The neutronics calculations were performed using three energy groups as: <0.625 eV for thermal neutrons, (0.625 eV to 5.53keV) for epithermal neutrons and up to 20 MeV for fast neutrons.

Results and discussion

Table 5 shows that the maximum difference between the measured and the calculated values using the MCNP4C code of the core excess reactivity, the SM and the SM with SF of the 1998/1 ETRR-2 core (See Figure 1) fueled by the U₃O₈-Al original fuel is 4.27%. Additionally, Table 7 shows that the maximum difference between the reference values of the TNF in the CIB, the ATNF in the CNT and the ATNF in the Be reflector of the current ETRR-2 core (See Figure 2) fueled by the U₃O₈-Al original fuel and the same calculated values is 1.20%. The good agreement between the calculated values, the measured and reference values of the above mentioned parameters will be used as reference to bring reliability to the obtained results of the ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide.

From Table 6 the following can be seen:

1. The calculated values using the MCNP4C code of the core excess reactivity, the SM and the SM with SF of the current ETRR-2 core (See Figure 2) fueled by the U₃O₈-Al original fuel differ from the same parameters of the 1998/1 ETRR-2 core by 4.39%, 4.47% and 5.6% for the core excess reactivity, the SM and the SM with SF, respectively. These errors are probably due to the change in the structure of the ETRR-2 core where the Be cubes were added around the current ETRR-2 core (See Figure 1 and Figure 2), where the Be cubes increase the value of the core excess reactivity as a result of reflecting neutrons into the reactor core.
2. The calculated values using the MCNP4C code of the core excess reactivity, the SM and the SM with SF, the CRW and the RSF of the current ETRR-2 core (See Figure 2) re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide differ from the same parameters of the current ETRR-2 core fueled by the U₃O₈-Al original fuel by 1.29% for the MOX CF mixed with 0.1% ²³⁷Np oxide.

As a result, re-fueling ETRR-2 reactor by the MOX CF mixed with ²³⁷Np oxide as a minor actinide does not have negative effect on the criticality parameters of the ETRR-2 reactor.

3.1 For the ACPO case (or $k_{eff} = 1.07633 \pm 0.00039$ and $k_{eff} = 1.07261 \pm 0.00042$ for the MOX CF mixed with 0.1% ²³⁷Np oxide, respectively).

The calculated values of the TNF in the CIB, the ATNF in the CNT and in the Be reflector of the modified ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide differ by 1.82%, 3.70% and 0.50% from the reference values of the TNF in the CIB, the ATNF in the NT and the ATNF in the Be reflector of the ETRR-2 core fueled by the U_3O_8 -Al original fuel (See Table 7 and Table 8). This result indicates that the re-fueling ETRR-2 core by the MOX CF mixed with 0.1% ^{237}Np oxide as a burnable actinide does not have negative effects on the neutronic parameters of the ETRR-2 reactor.

This result leads to say that all the scientific applications which were available in the ETRR-2 reactor fueled by the U_3O_8 -Al original fuel are still available for the ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide.

3.2 For criticality case (or $k_{eff} = 1.00032 \pm 0.00041$ and $k_{eff} = 1.00007 \pm 0.00045$ for the MOX CF mixed with 0.1% ^{237}Np oxide.

a. The TNF in the CIB is reduced by 8.09%, whereas the ATNF in the CNT is increased by 9.34% in comparison with the ACPO case for the modified ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide.

b. The TNF in the CIB and the ATNF in the CNT of the modified ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ^{237}Np oxide is higher than the 2.0×10^{14} n/cm².s. This value is sufficient to produce the radioisotopes such as: ^{14}C , ^{32}S , ^{51}Cr , ^{60}Co , ^{89}Sr , ^{153}Sm , ^{169}Yb , ^{170}Tm , ^{131}I , ^{125}I , ^{32}P , ^{192}Ir and ^{99}Mo [7], [15], [19].

This result leads to say that the reactor can be run with MOX CF mixed with 0.1% ^{237}Np oxide as a fuel without any negative effect on the scientific applications of the reactor.

Re-fueling current ETRR-2 core by the MOX CF mixed with 0.1% ^{237}Np oxide contributes to:

1. Reduce the ^{235}U loaded mass in the modified ETRR-2 core by 33 % in comparison with ETRR-2 core fueled by the U_3O_8 -Al original fuel.
2. Burn the plutonium isotopes and neptunium which helps in enhancing the proliferation resistance of the fissile materials,
3. Re-design research reactors having medium or high power fueled by MOX CF mixed with MAs,
4. encourage researchers to initiate further studies in this field to improve the life of the fuel cycle in research reactors and reach to a closed cycle in the future.

Re-fueling current ETRR-2 core by the MOX CF mixed with 0.1% ^{237}Np oxide and using the Ag -In-Cd alloy as a control plates reduces the values of the SM, SM with SF, CRW and the RSF in comparison with the reference values of the U_3O_8 -Al original fuel as shown in Table 11.

Table 9: Calculated values of the SM, SM with SF, CRW and the RSF of the current ETRR-2 core re-fueled by MOX CF mixed with 0.1% ^{237}Np oxide controlled by the Ag-In-Cd plates.

Parameter	Current ETRR-2 core re-fueled by the MOX CF with 0.1% ^{237}Np oxide (Figure 2)
SM (\$)	7.211 ± 0.018
SM with SF (\$)	2.764 ± 0.018
CRW (\$)	16.237 ± 0.027
RSF	1.799 ± 0.003

Table 9 shows that the differences between the calculated values of the SM, SM with SF, CRW and the RSF of the current ETRR-2 core re-fueled by the MOX CF mixed with

0.1% ²³⁷Np oxide, and the same values of the U₃O₈-Al original fuel are the 52.37%, 67.20%, 34.18% and 31.62%. Therefore, the absorber material Ag -In-Cd alloy was changed to B₄C for the ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide without changing neither the dimensions nor the position. The B₄C material seems to be more effective than the Ag-In-Cd alloy for the ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide. The neutronics calculations showed that the thermal, the epithermal and the fast neutron fluxes inside the control plates (See Figure 2) reduce by about 27.60%, 35.65% and 8.93% for the ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide, and with B₄C material for control plates in comparison with Ag-In-Cd alloy as shown in Table 10.

Table 10: Calculated values of the neutron flux in the control plates for the current ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷Np oxide, and with the Ag-In-Cd alloy and the B₄C as absorbing material for the control plates.

Parameter	Current ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷ Np oxide for the Ag-In-Cd alloy as a control plates	Current ETRR-2 core re-fueled by the MOX CF mixed with 0.1% ²³⁷ Np oxide for the B ₄ C material as a control plates
First control plate (See Figure 2)		
Thermal neutron (n/cm ² .s) x10 ¹³	3.270 ± 0.014	2.405 ± 0.016
Epithermal neutron (n/cm ² .s) x10 ¹³	5.330 ± 0.011	3.469 ± 0.012
Fast neutron (n/cm ² .s) x10 ¹⁴	1.047 ± 0.008	0.956 ± 0.004
Second control plate (See Figure 2)		
Thermal neutron (n/cm ² .s) x10 ¹³	3.534 ± 0.013	2.504 ± 0.015
Epithermal neutron (n/cm ² .s) x10 ¹³	6.896 ± 0.009	4.410 ± 0.010
Fast neutron (n/cm ² .s) x10 ¹⁴	1.322 ± 0.007	1.198 ± 0.007
Third control plate (See Figure 2)		
Thermal neutron (n/cm ² .s) x10 ¹³	2.860 ± 0.014	2.137 ± 0.016
Epithermal neutron (n/cm ² .s) x10 ¹³	4.876 ± 0.011	3.104 ± 0.013
Fast neutron (n/cm ² .s) x10 ¹⁴	0.968 ± 0.008	0.884 ± 0.009
Fourth control plate (See Figure 2)		
Thermal neutron (n/cm ² .s) x10 ¹³	3.202 ± 0.014	2.349 ± 0.016
Epithermal neutron (n/cm ² .s) x10 ¹³	5.270 ± 0.011	3.477 ± 0.012
Fast neutron (n/cm ² .s) x10 ¹⁴	1.042 ± 0.008	0.963 ± 0.008
Fifth control plate (See Figure 2)		
Thermal neutron (n/cm ² .s) x10 ¹³	3.456 ± 0.013	2.439 ± 0.016
Epithermal neutron (n/cm ² .s) x10 ¹³	6.824 ± 0.009	4.289 ± 0.010
Fast neutron (n/cm ² .s) x10 ¹⁴	1.328 ± 0.007	1.199 ± 0.007
Sixth control plate (See Figure 2)		
Thermal neutron (n/cm ² .s) x10 ¹³	2.923 ± 0.014	2.085 ± 0.016
Epithermal neutron (n/cm ² .s) x10 ¹³	4.836 ± 0.011	3.121 ± 0.013
Fast neutron (n/cm ² .s) x10 ¹⁴	0.953 ± 0.008	0.862 ± 0.009

The dispersion of the neutron flux inside B₄C is due to absorption cross section of the B₄C material, where the B₄C material has high absorption cross section in the thermal region in comparison with Ag-In-Cd alloy as shown in Table 11. Where the material constants were calculated using the WIMS-D4 code for the three neutronic groups as: (0.0 to 0.625) eV, (0.625 eV to 5.53 KeV) epithermal and (5.53 KeV to 10 MeV) fast neutron.

Table 14: Material constants of the control plates.

Type	Ag-In-Cd alloy		
	Fast neutron	Epithermal neutron	Thermal neutron
D	1.23611E+00	5.68857E-01	3.67896E-02
Σ_a	1.05526E-02	9.85794E-02	8.99976E+00
B ₄ C material			
D	1.06905+00	1.93132E-01	2.81699E-03
Σ_a	5.91670E-02	1.17206E+00	1.17767E+02

Conclusion

The MOX CF mixed with ^{237}Np oxide was proposed as a fuel in the MTR-22 MW. The neutronic analysis was performed by the MCNP4C code. The calculation of the criticality and neutronics parameters showed a good agreement with the reference values. Re-fueling MTR 22 MW reactor by the MOX CF mixed with 0.1% ^{237}Np oxide as a minor actinide leads to reduce the ^{235}U mass loaded in the reactor core by 33 % and contributes to enhance the proliferation resistance of the fissile material and a bit to burn plutonium isotopes in the MTR-22MW reactor.

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