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حساب الوسطاء النترونية و آمان الحرجية للمفاعل MTR –22 MW المحمل بالوقود الأساسي U_3O_8 – Al باستخدام الكود MCNP-4C2

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□ ملخّص □

استخدم الكود MCNP-4C2 لنمذجة مفاعل اختبار المواد MTR – 22 MW المحمل بالوقود الأساسيU_3O8-Al بنسبة إغناء U^{235} % 19.75. الوسطاء النترونية و آمان الحرجية **مثل:** معامل التضاعف الفعال k_{eff}، و زيادة تفاعلية القلب ρ_{ex}، و وثوقية صفائح التحكم (CPsW)، و هامش الإغلاق (SM)، و هامش الإغلاق مع صفيحة تحكم خارج قلب المفاعل ((SM with SCPF or (SM-1)، و معامل آمان التفاعلية (RSF)، و تدفق النترونات الحرارية في المصيدة النترونية المركزية (CNT)، و في صندوق التشعيع المركزي (CIB) و في عاكس البيريليوم Be للمفاعل MTR-22MW تم حسابها باستخدام الكود MCNP-4C2.

قورنت القيم المحسوبة لكل من الوسطاء النترونية و آمان الحرجية مع القيم التجريبية و قيم التصميم للمفاعل فوجد تطابق جيد. توضح هذه النتائج مدى التوافق الدقيق ما بين نظرية النقل العشوائية أي طريقة مونتي كارلو المبني على أساسها الكود MCNP-4C2 و القياسات التجريبية و القيم التصميم و المتعلقة بمفاعل البحث MTR-22M .

الكلمات المفتاحية: المفاعل MTR – 22 MW، الوسطاء و النترونية و آمان الحرجية و الكود MCNP-4C2.

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Calculation of the Criticality Safety and Neutronic Parameters of the MTR -22 MW Reactor fueled by the U_3O_8 - Al Original Fuel Using the MCNP-4C2 Code

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□ABSTRACT □

The MCNP-4C2 code has been applied to simulate the Material Testing Reactor (MTR – 22 MW), fueled by the U_3O_8-Al with 19.75 % enrich uranium – 235 (19.75 % U^{235}). The critically safety and neutronic parameters such as: Effectivity Multiplication Factor (k_{eff}), Excess Core Reactivity (ρ_{ex}), Control Plates Worth (CPsW), Shutdown Margin (SM), Shutdown Margin with Single (first) Control Plate Failure (SM with SCPF or (SM-1)), Reactivity Safety Factor (RSF), thermal neutron flux in the Central Neutronic Trap (CNT), in the Central Irradiation Box (CIB) and in the Be reflector of the MTR-22MW reactor were evaluated by using the MCNP-4C2 transport code.

The calculated values of the critically safety and neutronic parameters were compared with experimental and design values of the reactor and good agreements was found. These results illustrate the close agreement between stochastic transport theory (Monte Carlo method - MCNP-4C2 code) and the experimental measurements and design values conducted of the MTR -22 MW research reactor.

Key words: MTR – 22 MW reactor, critically safety and neutronic parameters, MCNP-4C2 code.

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1. Introduction

1.1 Material Testing Reactor MTR – 22 MW.

The MTR– 22 MW is a Material Testing Reactor (MTR). It was commissioned in 1997. It is an open pool research reactor which uses low enriched MTR fuel elements (19.7% enrichment). It is cooled and moderated with light water and reflected by beryllium. The reactor power is 22 MW with high thermal neutron flux irradiation positions ($>10^{14}$ n/cm^2.s). The MTR −22 MW core consists of 29 positions for fuel elements, where three distinct types of fuel elements are used in the MTR −22 MW which are: 7 Standard Fuel Elements (SFE), 8 Fuel Element Type 1 (FE Type 1) and 14 Fuel Elements Type 2 (FE Type 2). The composition and general characteristics of the Fuel Material (FM), FE, Fuel Plate (FP), absorber material, active zone dimensions and water gap between plates of the MTR – 22 MW reactor for the U_3O_8−Al original fuel are given in Table 1, Table 2 and Table 3. The reactor is controlled by 6 control plates made of Ag−In−Cd alloy (See Table 2) [1], [2], [3], [4], [5].

Table 1: Composition of the U_3O_8−Al original fuel used in the MTR – 22 MW research reactor.

Parameter	Wright 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	25.91	60.504	49.730
^{16}O	11.263	6.004	7.642
Density (g/cm^3)	4.802	3.299	3.655

Table 2: General characteristics of the fuel material, fuel element, absorber material of the MTR – 22 MW research reactor fueled by the U_3O_8–Al original fuel.

Parameter	Fuel material	
	U_3O_8–Al original fuel	
Fuel meat	U_3O_8–Al	
Enrichment ^{235}U (%)	19.7	
Total core loading ^{235}U (g)	6944.5	
Density of fuel meat (g/cm^3)	See Table 1	
Cladding material	Al-6061	
Fuel element		
Number of the FES	29	
Number of fuel plates in the FE	19	
Dimensions of the fuel piece (cm) (length x width x thickness)	80 x 6.4 x 0.07	
Absorber material		
Composition	Ag–In–Cd alloy	
	Wright percentage %	
	Ag	15
	In	80
	Cd	5
Density (g/cm^3)	10.18	

Table 3: General characteristics of the active zone dimensions and the water gap between plates of the MTR – 22 MW core fueled by U_3O_8–Al original fuel.

Active zone dimensions	
Parameter	One piece
Active length (cm)	80
Clad length (cm)	80
External section of fuel element (cm^2)	8x8
Section in grid to house the fuel element (cm^2)	8.1x8.1
Plate thickness (cm)	0.15
Meat thickness (cm)	0.07
Meat width (cm)	6.40
Side plate thickness (cm)	0.50
Side plate width (cm)	8.00
External distance between frames (cm)	8.00
Internal distance between frames (cm)	7
Water gap between plates	
of single fuel element (cm)	0.27
of different fuel element (cm)	0.39

1. 2 The main problem of this research

The main problem of this research is simulating the MTR – 22 MW research reactor using the MCNP-4C2 code and determination the criticality safety and neutronic parameters of the MTR – 22 MW research rector fueled by U_3O_8-Al original fuel.

1. 3 The importance of this research

The importance of this research is using the Monte Carlo method (MCNP4-C2 code) in neutronic and criticality calculations in medium-power research reactors, this helps in:

1. Increasing scientific experience and knowledge in neutronic and criticality calculations of medium-power research reactors.
2. Determining the criticality safety and neutronic parameters of the MTR – 22 MW research reactor before and after making any modification in the structure of the reactor core (such as: replacing a depleted fuel rods with new fuel, replacing the control plates and design neutronic channels for various scientific applications…etc.) to save the safe operation conditions, which are repotted in the safety report of the MTR – 22 MW reactor and which are taken in the design of the MTR- 22 MW reactor.

1. 4 Importance of the MCNP-4C2 code in the simulation of the MTR – 22 MW reactor

The MCNP-4C2 code is a popular, versatile multipurpose Monte Carlo particle transport code used worldwide. It has the capability to model and treat different geometries in 3-D, and also simulate the transport behavior of different particles and using a continuous energy cross section treatment as opposed to a multi-group approach thereby eliminating the errors in formulating few group cross sections. Additionally, MCNP-4C2 has the ability to treat complex nuclear interaction processes [6]. Therefore, in this work, the MCNP-4C2 code was used to:

- simulate the medium-power MTR – 22 MW research reactor.
- evaluate the critically safety parameters such as: Effectivity Multiplication Factor (k_{eff}), Excess Core Reactivity (ρ_{ex}), Control Plates Worth (CPsW), Shutdown Margin (SM), Shutdown Margin with single (first) control plate failure (SM-1) and Reactivity Safety Factor (RSF).
- determine the neutron spectrum in the Central Neutronic Trap (CNT), in the Central Irradiation Box (CIB) and in the Be reflector of the MTR-22MW reactor.

2. Simulation the MTR 22 MW research reactor by using the MCNP-4C2 code

The medium- power MTR -22 MW research reactor was simulated by using MCNP-4C2 code with three-dimensional detail to reduce possible systematic errors due to inexact geometry simulation. Therefore, this model of the medium power MTR -22 MW research reactor represents in detail all components of the core with literally no physical approximation. In the simulation of the medium power MTR -22 MW research reactor, the nuclear data for the fissile and the non-fissile materials such as:

- the fuel meat, fuel clad, coolant and moderator water, control rod and the reflector were taken from the ENDF/B-VI.1 nuclear data library.
- the thermal particle scattering $S(\alpha,\beta)$ was applied to treat the thermal scattering in both beryllium and hydrogen of the coolant and moderated water.

In this Simulation, the CNT was taken and located near the center of the reactor core. In the simulation the following specification of the MTR – 22 MW reactor were used:

1. the fuel composition shown in Table 1,
2. U_3O_8 as fuel material dispersed in an Al matrix,
3. a plate type fuel element,
4. an Al fuel clad,
5. an Al frame for the fuel element with U_3O_8-Al original fuel,
6. a de-mineralized light water as coolant.

The 3-D Monte Carlo MCNP-4C2 plot of the MTR – 22 MW reactor core configuration is shown in Figure 1 and Figure 2. In addition, The FE, EP used in the MTR- 22 MW reactor is shown in Figure 3, Figure 4 and Figure 5.

In the simulation of the MTR – 22 MW reactor core, the dimensions 90 (cm) wide × 110 (cm) length and 95 (cm) height of the reactor core, as well as the general characteristics mentioned in the Table 1, Table 2 and Table 3 were taken.

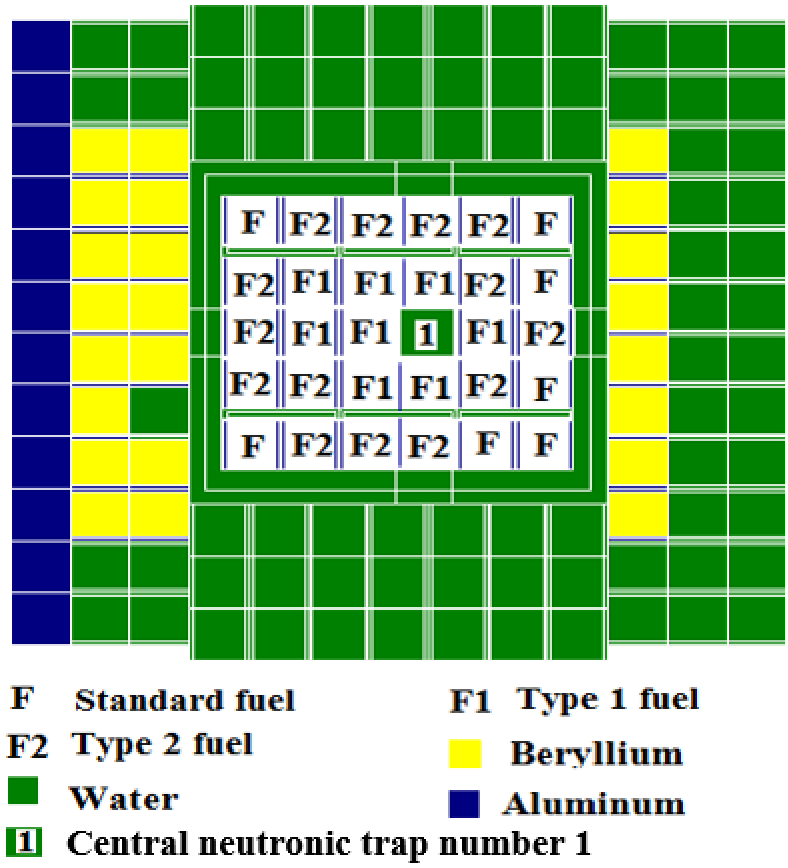


Figure 1: A schematic representation of the 1/98 MTR – 22 MW reactor core in the plane X–Y by using the MCNP-4C2 code

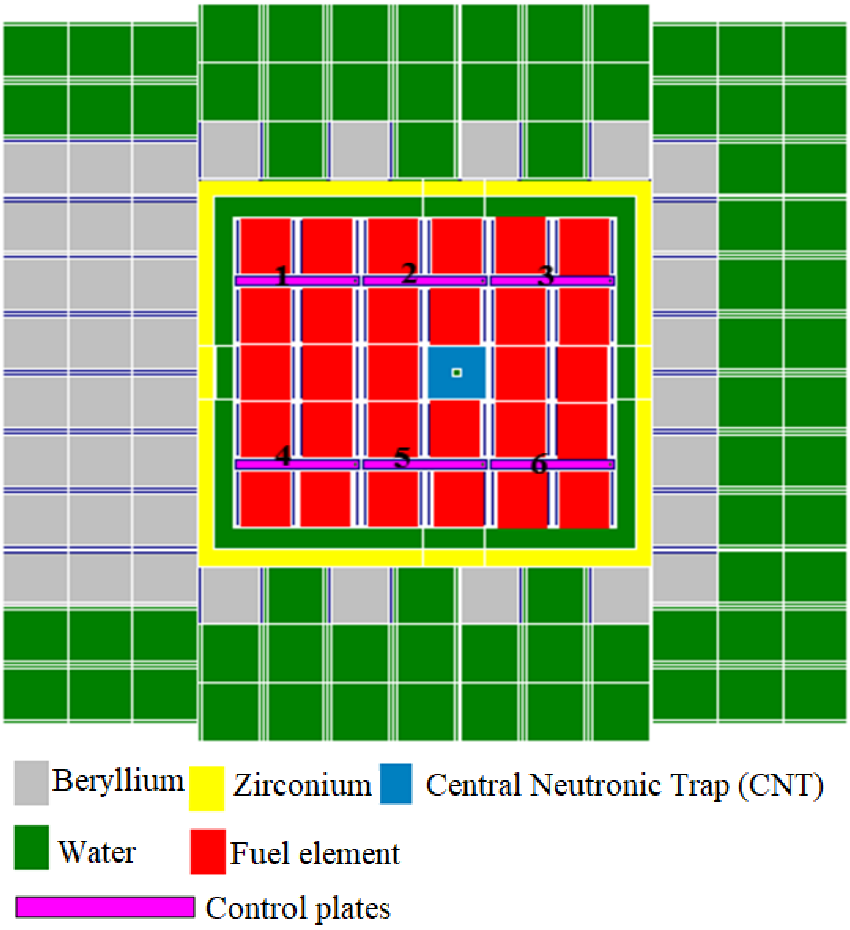
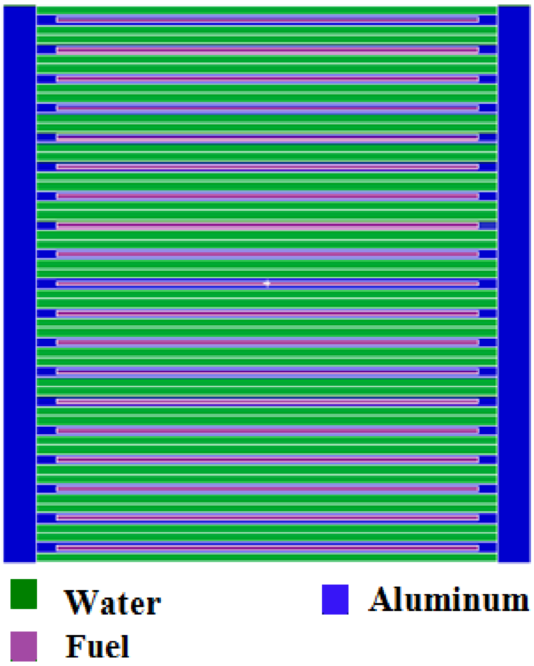


Figure 2: A schematic horizontal cross section of the MTR -22 MW reactor core coming from the MCNP4C code with six control plates.



F igure 3: A cross section of the FE used in the MTR −22 MW reactor core by using the MCNP4C code

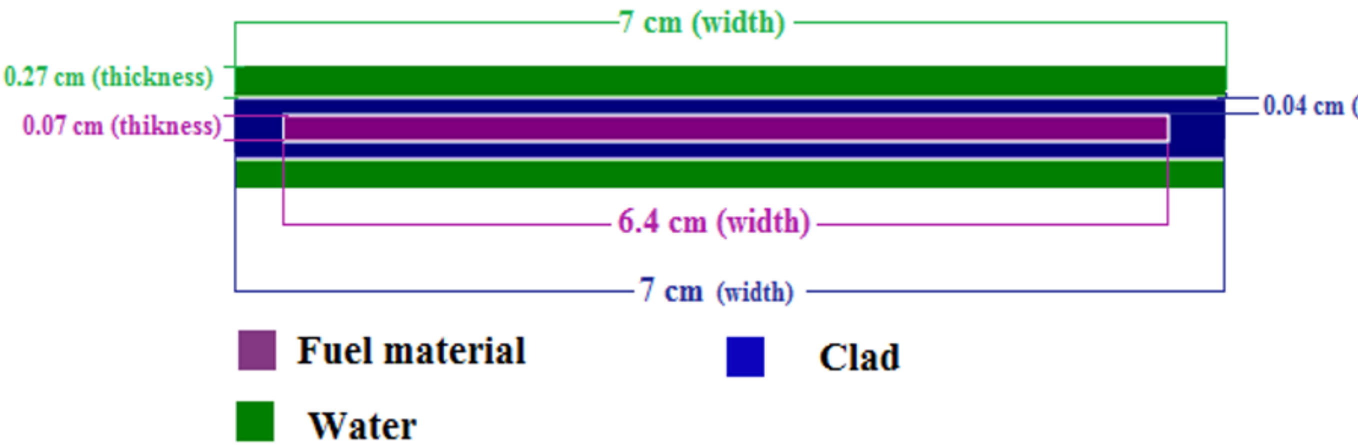


Figure 4: A cross section of the FP used in the MTR −22 MW reactor core by using the MCNP4C code

3. Calculation of the safety criticality parameters of the MTR – 22 MW reactor

The criticality safety parameters of the MTR – 22 MW reactor are:

- the effectivity multiplication factor (k_{eff}),
- the excess core reactivity (ρ_{ex}),
- the Shutdown Margin (SM) of the control plates,

- the SM of the control plates with Single (first) Plate Failure (SM with SCPF),

- the Control Plates Worth (CPsW),

- the Reactivity Safety Factor (RSF) with RSF meaning total control rod worth/core excess reactivity,

The criticality safety parameters were calculated by the KCODE criticality source card [6], and using all fuel elements as fission source points, where the fission source is located in the middle of each FE. The criticality calculations were done for a clean fresh core (zero burn-up).

The effective multiplication factor k_{eff} was estimated by running the input file of the MTR – 22 MW reactor by the MCNP-4C2 code with All CPsW withdrawn from the reactor core. The excess core reactivity ρ_{ex} was calculated using the following equation [7], [8].

$$\rho_{ex} = (k_{eff} - 1)/ k_{eff} \quad (1)$$

Where, the effective multiplication factor k_{eff} is defined as the ratio of the number of fission or fission neutrons in the generation divided by the number of fissions or fission neutrons in the preceding generation [6]. In the equation form,

$$k_{eff} = \frac{Number\ of\ fissions\ in\ one\ generation\ (i+1)}{Number\ of\ fissions\ in\ preceding\ generation\ (i)}$$

The formula of k_{eff} comes directly from the time-integrated Boltzmann transport equation without external source. k_{eff} can be written as follow [6]:

$$k_{eff} = \frac{\rho_a \int_V \int_0^\infty \int_E \int_\Omega v\sigma_f \Phi dVdtdEd\Omega}{\int_V \int_0^\infty \int_E \int_\Omega \nabla J dVdtdEd\,\Omega + \rho_a \int_V \int_0^\infty \int_E \int_\Omega \upsilon(\sigma_f + \sigma_c + \sigma_m)\Phi dVdtdEd\Omega}$$

This equation is derived from integro-different neutron transport equation with energy (E), flux (Φ), direction of neutron (Ω), reactor volume (V), neutron velocity (v), time (t) and neutron current flux (J), material or atom density (ρ_a) by meant of σ_c, σ_f and σ_m are microscopic cross section for capture (n, pn) fission and multiplicity (n, xn), respectively.

The effective multiplication factor k_{eff} is computed in MCNP-4C2 based on the calculation of three different estimators a collision-based k_{eff}^{C}, an absorption-based k_{eff}^{A} and a track length based k_{eff}^{TL} [6].

The control plates worth was obtained using the relation [7], [8]:

$$Control\ Plates\ Worth\ (CPsW) = \frac{k_{eff}^{out} - k_{eff}^{in}}{k_{eff}^{out} \times k_{eff}^{in}} \qquad (2)$$

Where:

$k_{eff}^{out} = k_{eff}$ − is the effective multiplication factor with control rod is fully withdrawn from reactor core and equal to k_{eff}.

$k_{eff}^{in} = k_{in}$ − is the effective multiplication factor with control rod is fully inserted inside reactor core, this value was calculated by using MCNP−4C2 code with same conditions mentioned used to estimate the k_{eff}.

The SM of the control plates was calculated using the equation [7], [8]:

$$SM = CPsW - \rho_{ex} \qquad (3)$$

Whereas, the SM and SM with SCPF is the negative reactivity, and they are calculated when the control plates are fully inserted and when the control plates are fully inserted but failing a single plate, respectively.

The obtained results of the 1/98 MTR – 22 MW reactor core (See Figure 1) fueled by the U_3O_8−Al original (the first digit is a correlative number and the last two digits are the year, then 1/98 is the first core in 1998 [2] and the current MTR – 22 MW reactor core (See Figure 2) fueled by U_3O_8−Al original fuel (Figure 4) are given in Table 4 and Table 5, respectively.

The calculated values of each control plate worth of the MTR – 22 MW reactor core fueled by the U_3O_8−Al original fuel and with Ag−In−Cd alloy as absorber material in the control plates are shown in Table 6 and Table 7 with the design values of these parameters.

The control plates worth values were calculated using equation (2) and with same conditions which are used to evaluate the k_{eff} and k_{in} with one control plate is failure. In this case, the $k_{in}$is calculated for five control plates.

Table 4: Measured and MCNP4C results of the effective multiplication factor k_{eff} and k_{in}, core excess reactivity ρ_{ex}, the SM and the SM with SCPF of the 1998 MTR – 22 MW core fueled by the U_3O_8–Al original fuel.

Core 1/98 of the MTR – 22 MW fueled by U_3O_8–Al original fuel (Figure 1)			
Fuel material used in the fuel plate consisting of the one plate (Figure 4)			
Parameter	Measured[(a)]	Calculated values using MCNP – 4C2 code	Relatively error %
Effective multiplication factor k_{eff}	–	1.07208 ± 0.00043	–
Effective multiplication factor k_{in}	–	0.89375 ± 0.00041	–
Core excess reactivity ρ_{ex} ($)	9.1	8.964 ± 0.049	1.494
SM ($)	15.2	15.849 ± 0.029	4.269
SM with SCPF ($)	8.7	8.929 ± 0.029	2.263

(a) M

easured values of the core 1/98 were taken from the references [2], [4], [5].

Table 5: MCNP– 4C2 results of the core excess of reactivity ρ_{ex}, the SM and SM with SCPF, the CPsW and the RSF of the current MTR– 22 MW reactor core fueled by the U_3O_8–Al original fuel.

Current MTR – 22 MW reactor core fueled by the U_3O_8–Al original fuel (Figure 2)	
Parameter	Calculated values using MCNP–4C2
Core excess reactivity ρ_{ex} ($)	9.376 ± 0.007
SM ($)	15.141 ± 0.036
SM with SCPF ($)	8.429 ± 0.036
CPsW ($)	24.670 ± 0.043
RSF	2.631± 0.004

Table 6: Criticality safety parameters of each control plate of the MTR – 22 MW reactor core fueled by the U_3O_8–Al original fuel and with Ag–In–Cd alloy as absorber material in the control plates.

Parameter	Design value for U_3O_8–Al original fuel (pcm)[(4)]	Calculated values of the U_3O_8–Al original fuel by using MCNP–4C2	Differences %	Design value of safety criteria (pcm)[(4)]
ρ_{ex} at BOC[(1)]	8220	8219.07 ± 5.30	0.011	> 3000
CPsW[(2)]	14120	14223.22 ± 17.35	0.73	> 1000
CPsW[(3)] #1	10390	10540.01 ± 12.10	1.44	>1000
CPsW #2	11030	10991.23 ± 19.27	0.35	>1000
CPsW #3	11050	11064.29 ± 13.21	0.13	>1000
CPsW #4	10810	10766.12 ± 17.24	0.40	>1000
CPsW #5	11330	10987.22 ± 19.04	3.02	>1000
CPsW #6	11290	11297.84 ± 11.24	0.07	>1000

(1) ρ_{ex} at BOC– is the reactivity at Beginning Of fuel Cycle,

(2) (CPsW) – is the Control Plates Worth,

(3) CPsW#1 – is the Control Plates Worth without first control plate (See Figure 1)

(4) A. M. El. Messiry and Moustafa M. Aziz, 2000. Seventh Conference of Nuclear Science and Applications 6 – 10 February, Safety Assessment of ETRR-2 Reactivity Criteria, Cairo, Egypt.

Table 7: of the RSF, SM and SM with SCPF (SM-1) of the MTR- 22 MW reactor core fueled by the U_3O_8-Al original fuel and Ag-In-Cd alloy as absorber material in the control plates.

Statement	Design value for U_3O_8-Al original fuel (pcm)(1)	Calculated values of the U_3O_8-Al original fuel by using MCNP-4C2	Relatively error %
RSF >1	1.7177	1.7259 ± 0.0031	0.48
SM > 3000 pcm	5900	5978.23 ± 13.25	1.33
SM -1 > 1000 pcm	2170	2195.12 ± 7.19	1.16

(1)A. M. El. Messiry and Moustafa M. Aziz, 2000. Seventh Conference of Nuclear Science and Applications 6 – 10 February, Safety Assessment of ETRR-2 Reactivity Criteria, Cairo, Egypt

4. Calculation of the neutronics parameters

The neutronics parameters of the MTR – 22 MW reactor include:

1. The Average Thermal Neutron Flux (ATNF) in the CNT which is located in site 1 as shown in Figure 2 and Figure 5,
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 calculate the neutronics parameters of the MTR – 22 MW core fueled by the U_3O_8-Al original fuel, the F_4 tally, the FS, the SD and the FM cards in the MCNP-4C2 code were used in the input file of the MTR – 22 MW reactor, and then the input file was run by the MCNP-4C2 code. The F_4 tally, the FS, the SD and FM cards are used as follows:

F4: 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 MTR – 22 MW reactor defined by equation (4) to give the normalized flux in the correct unit of neutrons/cm^2.s (See manual MCNP−4C2 code, [6]).

$$C = \frac{P(watt).\tilde{\nu}}{E(MeV)} \cdot \frac{1\, joule/sec}{watt} \cdot \frac{1\, MeV}{1.602\ \times 10^{-13}\,(joules)} \qquad (4)$$

Where:

P(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 MCNP−4C2 output file),

$E(MeV)$ − is the released energy per fission (the value of the *E (MeV)* is listed in the MCNP−4C2 output file).

The calculated values by using the MCNP−4C2 code of the ATNF in the CNT and in the Be reflector, and the TNF in the CIB are tabulated in Table 8 for the current MTR – 22 MW reactor core fueled by the U_3O_8−Al original fuel.

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.

Table 8: Reference and MCNP4C2 results of the neutron flux of the current MTR −22 MW core fueled by U_3O_8−Al original fuel.

Current MTR – 22 MW reactor core fueled by the U_3O_8−Al original fuel (Figure 2)			
Parameter	Reference value	Calculated value using MCNP−4C2	Relatively error %
TNF in the CIB (n/cm^2.s)×10^{14}	4.230$^{(a)}$	4.215 ± 0.017	0.354
ATNF in the CNT (n/cm^2.s) ×10^{14}	2.700$^{(b)}$	2.665 ± 0.007	1.296
ATNF in the Be reflector (n/cm^2.s) ×10^{14}	1.000$^{(b)}$	0.988 ± 0.014	1.200

(a) Reference value was taken from reference [5].

(b) Reference values were taken from the references (www.etrr2-aea.org.eg/Data Sheet about MTR-22MW.html and Main Core Data of MTR -22 MW Reactor.html; http://archive.is/Mnf73).

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5. Results and discussion

Table 4 shows that the maximum difference between the measured and the calculated values using the MCNP4C2 code of the core excess reactivity ρ_{ex}, the SM and the SM with SCPF of the 1/98 MTR -22 MW reactor core (See Figure 1) fueled by the U_3O_8-Al original fuel is 4.27%.

From Table 5 it can be seen also that the calculated values for the core excess reactivity ρ_{ex}, the SM and the SM with SCPF for the current MTR -22 MW reactor core (See Figure 2) differ from those for the 1/98 MTR – 22 MW reactor core (See Figure 1) by 4.39, 4.67 and 5.60% for the excess core reactivity ρ_{ex}, the SM and the SM with SCPF, respectively. This difference is due to the change in the structure of the MTR -22 MW reactor core where the Be cubes were added around the current MTR -22 MW reactor core (See Figure 1 and Figure 2).

From Table 6 the maximum difference between the design value of the control plates worth and the calculated values of the MTR – 22 MW reactor fueled by the U_3O_8-Al original fuel is 3.02 %.

From Table 7 it can be seen another good agreement between the design value of the RSF, SM and SM with SCPF (SM-1) of the MTR- 22 MW reactor core fueled by the U_3O_8-Al original fuel and Ag-In-Cd alloy as absorber material in the control plates and the calculated values for the same parametrs is 0.48, 1.33 and 1.16 %, respectively.

Additionally, Table 8 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 MTR 22 MW reactor core (See Figure 2) fueled by the U_3O_8-Al original and the same calculated values is 1.20%.

At finally, the good agreement between the calculated values and the design, measured and reference values of the above-mentioned parameters confirm the accuracy of the 3-D Monte Carlo model of the MTR – 22 MW reactor using MCNP-4C2 code.

Conclusion

Stochastic Monte Carlo methods have been applied to accurately develop a 3-D model of the MTR – 22 MW reactor using the versatile MCNP - 4C2 particle transport code. Within limits of errors caused by approximations in material balance and dimensions of some components of the MTR – 22 MW reactor core, the Monte Carlo model has successfully simulated some important reactor physics and criticality safety design parameters. The calculated results of the critically safety and neutronic parameters are in very good agreement with experimental measurements and design values obtained during the facility's zero power criticality and on-site cold tests. This work thus provides further evidence that Monte Carlo methods are accurate in modeling complicated geometries and also capable of simulating physics and engineering design parameters of nuclear reactors such as the MTR – 22 MW reactor.

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