A STUDY ON IMPROVING THE PERFORMANCE OF A RESEARCH REACTOR'S EQUILIBRIUM CORE

by

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Utilizing low enriched uranium silicide fuel (U_3Si_2-Al) of existing uranium density (3.285 g/cm^3), different core configurations have been studied in search of an equilibrium core with an improved performance for the Pakistan Research Reactor-1. Furthermore, we have extended our analysis to the performance of higher density silicide fuels with a uranium density of 4.0 and 4.8 U g/cm^3. The criterion used in selecting the best performing core was that of "unit flux time cycle length per ^{235}U mass per cycle".
In order to analyze core performance by improving neutron moderation, utilizing higher-density fuel, the effect of the coolant channel width was also studied by reducing the number of plates in the standard/control fuel element. Calculations employing computer codes WIMSD/4 and CITATION were performed. A ten energy group structure for fission neutrons was used for the generation of microscopic cross-sections through WIMSD/4. To search the equilibrium core, two-dimensional core modelling was performed in CITATION. Performance indicators have shown that the higher-density uranium silicide-fuelled core (U density 4.8 g/cm^3) without any changes in standard/control fuel elements, comprising of 15 standard and 4 control fuel elements, is the best performing of all analysed cores.

Key words: research reactor, reactor fuel, MTR-PC26 package, WIMS/D4, CITATION

INTRODUCTION

Pakistan Research Reactor-1 (PARR-1) is a swimming pool-type material testing research reactor (MTR) with a parallelepiped core comprised of LEU (U_3Si_2-Al) fuel, containing 19.99% ^{235}U. Demineralized light water is used as coolant and moderator. One side of the parallelepiped core is reflected by graphite, *i. e.* the thermal column, while the opposite side is reflected by a blend of graphite reflector elements and light water. The bottom side is reflected by a combination of aluminum and water. The three remaining sides, *i.e.* the top and two lateral sides, are reflected by light water only. Fuel elements, control rods, graphite reflector elements, water boxes for the irradiation of samples and fission chambers with their guide tubes are assembled on a grid plate bearing 54 holes, arranged in a 9 6 array, with a lattice pitch of 81.0 mm × 77.1 mm. At PARR-1, five control rods (Ag-In-Cd alloy) are employed for the power-level control of the operating reactor and safe shutdown in normal or possible accidental circumstances. The PARR-1 core provides numerous irradiation facilities which include water boxes, a graphite thermal column, pneumatic rabbit tubes, beam port tubes and a dry gamma cell, bulk irradiation area, and hot cell. Main PARR-1 specifications are given in tab. 1[1].

In the current study, various research reactor models based on different fuel (U_3Si_2-Al) loadings and coolant gaps have been studied. The effect of uranium silicide fuel density variation was analyzed to propose an optimum fuel loading based on: thermal neutron flux at irradiation sites, cycle length, consumption of ^{235}U per cycle and the initial inventory of ^{235}U. A higher neutron flux at irradiation sites is always desirable in a research reactor for irradiation samples, basic research and isotope production. Minimum fissile material consumption per cycle length and longer cycle lengths are economical. Therefore, the criterion chosen for the selection of the best core performance was determined as "unit flux time cycle length per ^{235}U mass per cycle". An analysis was carried out by assuming the PARR-1 grid plate and no changes in the PARR-1 current system concerning structural/embedded piping systems for core cooling or primary and secondary pump systems, motors *etc*. At PARR-1, control rods should be at least 50% out of

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Table 1. Main specifications of PARR-1

Type	Swimming pool
Nominal core power [MW]	10
Lattice pitch [mm]	81.0 77.11
Fuel material and enrichment	U_3Si_2-Al (19.99 % by wt.%)
Cladding material	Aluminum
Coolant/moderator	Light water (H_2O)
Coolant flow rate [m^3h^{-1}]	950
Reflector	Light water and graphite
Fuel element description	Straight plate MTR type fuel element
^{235}U contents per fuel plate [g]	12.61
Control rods	Oval shaped 5 rods
Composition of control rods	80% Ag, 15% In, 5% Cd
Operational modes	Manual and automatic
Feedback coefficients:	
Doppler coefficient, %$\Delta k/k$ – °C	$-1.91\ 10^{-3}$
Moderator coefficient, %$\Delta k/k$ – °C	$-5.97\ 10^{-3}$
Void coefficient, %$\Delta k/k$ – °C Voids	–0.34
Irradiation sites:	Neutron flux [$cm^{-2}s^{-1}$]
Beam ports	$1.0 \cdot 10^{13}$
Thermal column	Depends on the depth in thermal column
Pneumatic rabbit	$\sim 3.0\ 10^{13}$

position in the critical core. The end-of-cycle excess reactivity of the equilibrium core should be 1.0% k/k, so as to account for 0.5% k/k of the sample load and 0.5% k/k for temperature effects at full power [2]. This criterion has also been considered during criticality calculations of the currently analyzed cores.

With higher-density fuel, the fuel-to-moderator ratio increases and the core which is otherwise critically-moderated becomes highly under-moderated [3]. Utilizing higher-density fuel, the effect of the coolant channel width was also studied by reducing the number of plates in the standard fuel element and control fuel element to check the performance by improving neutron moderation. By reducing the number of plates in fuel elements, the heat transfer area is decreased, limiting the steady-state power. Although the coolant flow per fuel plate is increased by reducing fuel plates in the element, this cannot counter the decreasing effect of the heat transfer area.

HEU fuel is the best option for a research reactor due to the higher burn-up, longer operating cycle and lesser radioactive waste [4]. Through the global threat reduction initiative (GTRI) and the reduced enrichment for research and test reactor (RERTR) Program, the international community has come together to minimize, and to the extent possible, eliminate the use of HEU in civil nuclear programs throughout the world. Therefore, LEU fuel is being considered worldwide as a design solution for future research reactors. The density of ^{235}U is much lower in LEU fuel than in HEU. The higher inventory of ^{238}U in LEU fuel is a source of prompt negative reactivity feedback, also affecting neutron economy and making the fuel requirement higher than that of HEU. To counter this effect, one option is to increase the number of fuel elements and plates per fuel element in the LEU-fuelled reactor core. The other one is to replace the low-density HEU fuel with LEU fuel of a higher density. Thermal neutron flux at irradiation sites and cycle length are improved by using higher density fuel. Both options have been analyzed in the current study. Enrichment reduction by simple substitution of lower-enriched uranium in existing fuel designs has the immediate effect of reducing core performance. Fuel burn-up capability decreases, while fuel costs increase. The burn-up potential can be matched to that of the unmodified reactor by increasing the ^{235}U content in the LEU core to an amount slightly over that of the HEU core, at the expense of a slight decrease in the in-core thermal flux-per-unit-power performance [5]; therefore, reactor power also needs to be upgraded within the limitations imposed by the thermal-hydraulic.

METHODOLOGY FOR ANALYZING THE EQUILIBRIUM CORE

A MTR-PC26 package was used for the generation of microscopic cross-sections of the different regions of the PARR-1 core. This package uses the WIMS/D4 [6], an upgraded version of the Winfrith improved multi-group scheme (WIMS) [7] computer code, along with an attendant code called BORGES [8]. The BORGES is used to read the output of WIMS/D4, as per instructions of the user, and then writes it in a form that is readily usable in the multi-dimensional, diffusion theory code, CITATION [9]. The WIMS/D4 code uses its own 69-group library and solves the neutron transport equation in one dimension with reflective boundary conditions.

Ten-energy group microscopic cross-sections were obtained. The energy group structure used is given in tab. 2 [2, 10-12]. All cross-sections were evaluated at 40 °C. The unit cell shown in fig. 1 [13] was selected for the generation of cross-sections and number densities of the fuel part of the standard and control fuel element. The half-unit cell shown in fig. 2, containing fuel meat, cladding, coolant and the extra region was used in WIMS/D4 to generate the cross-sections of the fuel element containing 23 fuel plates. The extra aluminum in the clad of the end fuel plate with the extra water in front of the end fuel plate have been equally distributed over 46 half-unit cells and were accommodated in the extra region. The side plates, the non-fuelled lateral portion of the fuel plates and the water in the non-fuel portion were modelled as structural materials in a separate unit cell. Control rods and control-follower regions were modelled in a separate unit cell. Microscopic

Table 2. Ten energy group structure for microscopic cross-sections

Energy group [eV]	Upper energy	WIMSD4 energy group	Average energy [eV]	Remarks
1	10 10^6	1-5	2.87 10^6	Above threshold fission of ^{238}U and no delayed production
2	0.821 10^6	6-7	4.98 10^5	Average energy of 2 to 6 delayed neutron groups produced
3	3.025 10^5	8-8	2.353 10^5	Average produced energy of first delayed neutron group
4	0.183 10^6	9-20	8.2 10^3	Fourth and fifth groups are of equal lethargy intervals in WIMSD groups from 9-33
5	367.262	21-33	20.551	
6	1.15	34-40	1.057	Group covers the resonance of ^{240}Pu *i. e.* at 1 eV
7	0.972	41-45	0.78	Group 7 separates the energy boundaries of group 6 and 8
8	0.625	46-55	0.296	This group covers the resonance of ^{239}Pu *i. e.* at 0.3 eV
9	0.14	56-60	0.0837	The ninth group is based on the maximum thermal neutron cut-off energy (0.14 eV (5 kT)
10	0.050	61-69	2.2 10^{-4}	Remaining thermal groups

cross-sections and number densities for the water reflector, graphite reflector and the thermal column lead and graphite regions, were also generated utilizing the WIMS/D4 code and BORGES. Similarly, half-unit cells shown in figs. 3 and 4 were employed to generate the cross-sections for the standard fuel element containing 22 fuel plates and 21 fuel plate, respectively.

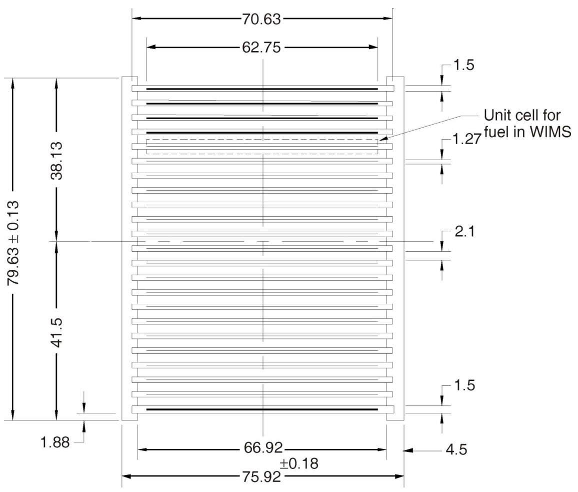

Figure 1. Standard fuel element of PARR-1 with unit cell configuration

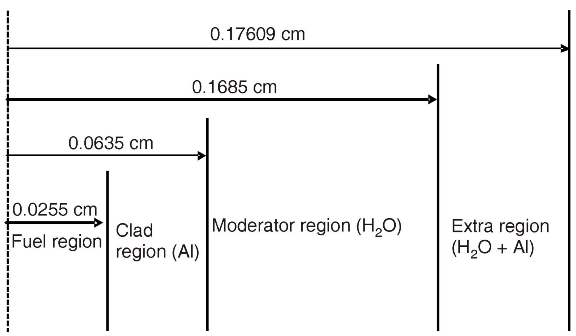

Figure 2. Half-unit cell configuration with a 23 plates-standard fuel element for WIMS/D4

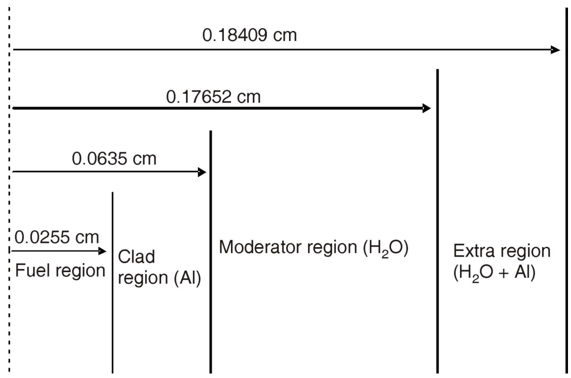

Figure 3. Half-unit cell configuration with a 22 plates-standard fuel element for WIMS/D4

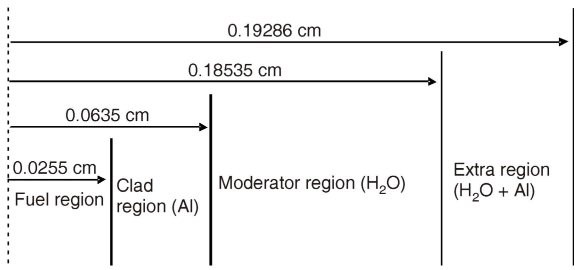

Figure 4. Half-unit cell configuration with a 21 plates/standard fuel element for WIMS/D4

In order to analyze equilibrium cores of different configurations, different fuel densities and a varying number of fuel plates listed in tab. 3 were used in WIMSD/4. The cores were modelled in x-y geometry (2-D) of CITATION, without a control absorber. In the third dimension, buckling was incorporated, with 8.0 cm reflector saving from top-to- bottom of the active part of the core [2]. For static, depletion and fuel management analysis, the diffusion theory based code, CITATION, was used. Research and power reactors are operated according to a certain fuel management scheme, based on reactor safety and economy. For a research reactor, the scheme should be optimized on peaking factors, cycle length and neutron flux at irradiation sites. Initially, fresh fuel elements with zero

Table 3. Results of the analyzed equilibrium cores

U density [gcm^{-3}]	4.8	4.8	4.8	4	4	4	3.285	3.285
No of standard fuel elements	15	15	15	17	17	17	24	29
No of control fuel elements	4	4	4	5	5	5	5	5
No of plates/fuel	21	22	23	21	22	23	23	23
Mass of ^{235}U loaded per cycle [kg]	1.36	1.44	1.51	1.14	1.20	1.26	1.32	1.32
Initial inventory of ^{235}U [kg]	6.61	6.96	7.31	6.33	6.66	7.00	7.78	9.23
Thermal flux at inner irradiation site [$ncm^{-2}s^{-1}$]	1.84E+14	1.83E+14	1.83E+14	1.59E+14	1.57E+14	1.56E+14	1.63E+14	1.39E+14
Full power days	35.5	38	41	24	27	30	35	47
Average burn-up at beginning of equilibrium core [%]	10.29	10.44	10.71	7.66	8.19	8.56	11.92	16.97
Maximum burn-up at discharge [%]	31.81	32.22	32.93	23.49	24.96	25.95	37.63	47.74
Average burn-up of discharged elements [%]	23.69	24.02	30.75	21.72	23.11	24.20	33.41	43.30
Unit flux.cycle length/ ^{235}U mass per cycle	4.80E+15	4.83E+15	4.97E+15	3.35E+15	3.54E+15	3.71E+15	4.30E+15	4.95E+15
Unit flux.cycle length/initial ^{235}U mass	9.90E+14	9.97E+14	1.02E+15	6.02E+14	6.36E+14	6.67E+14	7.31E+14	7.10E+14

fission products were taken for studying the equilibrium core. The cycle length of the equilibrium core was determined by burn-up analysis. Reshuffling schemes for all cores were selected so that a fresh element was first introduced to the core periphery and, after each cycle, moved towards the centre of the core in order to be, ultimately, discharged from it. The inbuilt convergence criteria for searching the equilibrium core in CITATION code is $9.999 \cdot 10^{-3}$ *i. e.* the maximum difference between the number densities of all isotopes of two successive cycles should be less than this number. At the beginning of cycle (BOC), the CITATION code checks the number densities of each isotope with previous cycle number densities for the effective multiplication factor and group neutron flux distribution, so as to test the equilibrium core conditions

$$e_n \quad \left|1 \quad \frac{N^{n}(r,0)}{N^{n\ 1}(r,0)}\right| \quad \varepsilon_n \tag{1}$$

$$e_k \quad \left|1 \quad \frac{k_{eff}^{n}(0)}{k_{eff}^{n\ 1}(0)}\right| \quad \varepsilon_k \tag{2}$$

$$e_\phi \quad \left|1 \quad \frac{\phi^{n}(r,0)}{\phi^{n\ 1}(r,0)}\right| \quad \varepsilon_\phi \tag{3}$$

where ε_n, ε_k, and ε_ϕ are the small numbers for testing equilibrium core conditions, while $N^{(n)}(r,\ 0), k_{eff}^{(n)}(0)$ and $\phi^n(r,\ 0)$ are the isotopic number densities, multiplication factor and flux at the n^{th} cycle at position (r) at BOC.

RESULTS AND DISCUSSION

Adopting the above mentioned criteria, equilibrium cores were studied for different configurations. The thermal flux at irradiation sites, fuel burn-up at the beginning of equilibrium cycle (BOEC), initial fissile inventory, cycle length (full-power days) and fissile consumption per cycle were determined and results summarized in tab. 3 so as to identify the best performing core.

From tab. 3 it is obvious that, while keeping the same size and reducing the number of plates per fuel element, the core size is reduced and that this is followed by an increase in the thermal flux at the irradiation site. Nevertheless, the decrement of fissile inventory reduces the cycle length and, hence, the average burn-up at discharge also decreases. The performance of high density fuel (U density 4.8 g/cm^3) with 15 standard fuel elements, 4 control fuel elements, 23 fuel plates per standard fuel element, and 13 fuel plates per control fuel element, is the best of the cores examined. The proposed core is shown in fig. 5, with water boxes at C(7) and C(4) positions. This is a compact and highly under-moderated core. It is reflected by graphite blocks from three sides, while the remaining one features a graphite thermal column, so as to minimize neutron leakage and flatten the flux profile.

The reshuffling scheme for analyzing the proposed equilibrium core is shown in tab. 4, based on the criterion of minimizing the peaking factors from a safety point of view. Starting with fresh fuel, the burn-up increases with each cycle number, as shown in fig. 6 and, hence, k_{eff} also decreases with burn-up, as shown in fig. 7, until the equilibrium core is established. When the equilibrium core is established, at each BOEC, the burn-up is the same and, hence, the multiplication factor for consecutive cycles remains the same. It is obvious from figs. 6 and 7 that a total of nine steps are sufficient for establishing the equilibrium. The 10^{th} step is BOEC.

The maximum discharge burn-up is 32.93%, while the average discharge burn-up is 30.745%. A total of 1511 gram of ^{235}U is loaded per cycle. The burn-up at BOEC is shown in fig. 8. Fresh elements are

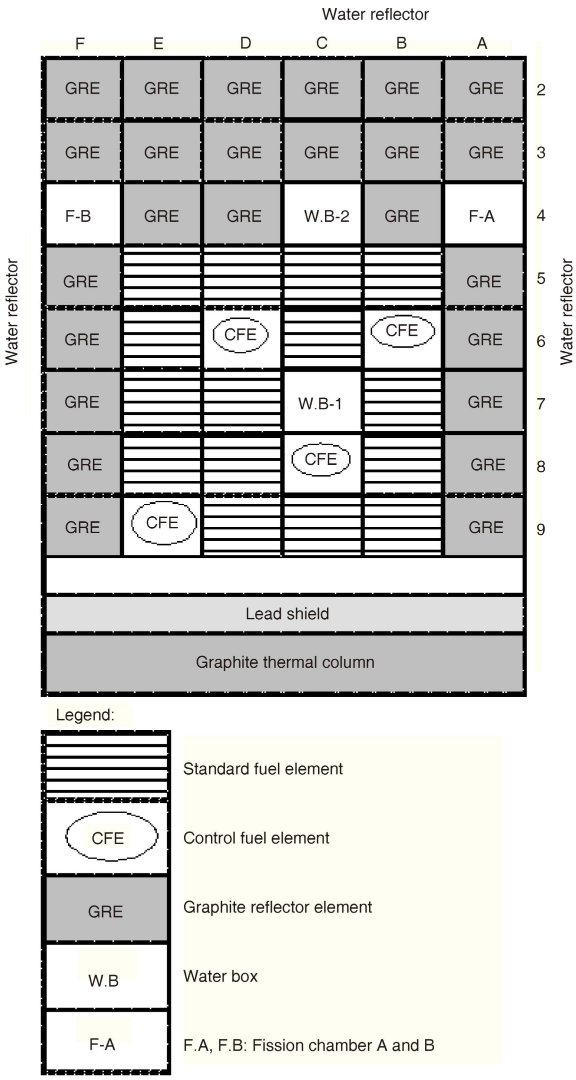

Figure 5. Proposed equilibrium core configuration

loaded at the core periphery, low-burned elements are also away from the core centre and high-burned elements are located almost at the very core centre, so as to minimize the peaking factor. Figure 9 shows the burn-up at end of equilibrium cycle (EOEC). Elements at C(8), C(6), D(7), and D(8) positions achieved the highest burn-up at EOEC. These elements should be discharged and the core reshuffled according to the

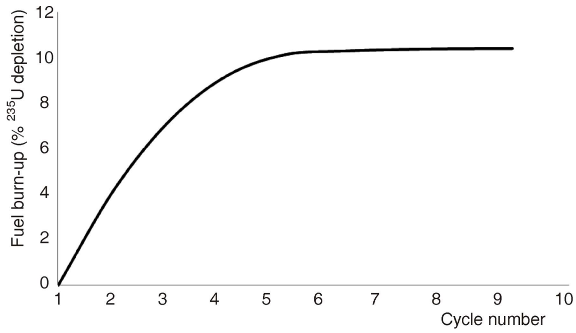

Figure 6. Fuel burn-up at BOC of each cycle

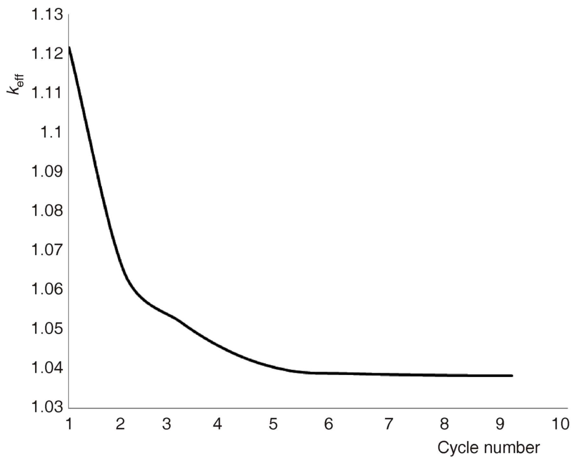

Figure 7. BOC multiplication factor with the increasing cycle number

scheme given in tab. 4 for the new BOEC. The burn-up of discharged elements is not very high but, due to insufficient reactivity, the core cannot be further operated after 41 full-power days.

With the increasing burn-up of the fuel, the initial loading of ^{235}U is depleted and ^{239}Pu and its higher isotopes are produced. Due to the internal conversion of ^{238}U, plutonium isotopes are also contributing to power-generation as the burn-up increases. Therefore, the composition of the discharged fuel is also important, due to plutonium isotopes. Table 5 shows the composition of the fuel in the core and discharged fuel elements at BOEC and EOEC.

Table 4. Reshuffling and refuelling scheme for analyzing the proposed equilibrium core

From core position	To core position	From core position	To core position
E(5) (Fresh)	D(5)	B(7)	E(8)
D(5)	E(6)	E(8)	C(6)
C(5)	B(7)	D(8)	Discharge
B(5) (Fresh)	C(5)	C(8)	Discharge
E(6)	E(7)	B(8)	C(9)
D(6)	C(8)	E(9) (Fresh)	B(6)
C(6)	Discharge	D(9)	D(8)
B(6)	D(6)	C(9)	D(9)
E(7)	D(7)	B(9) (Fresh)	B(8)
D(7)	Discharge		

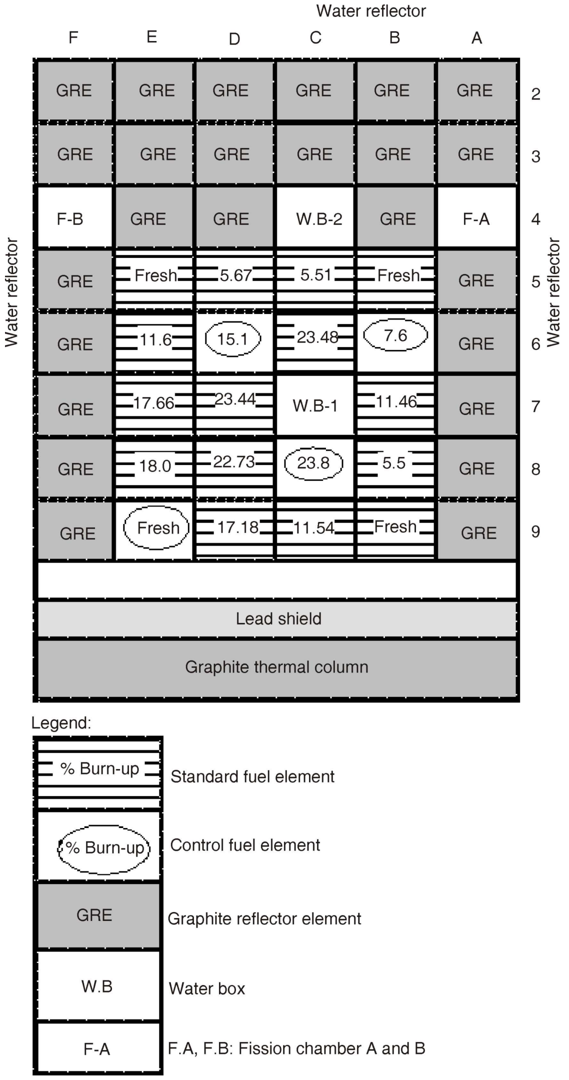

Figure 8. Burn-up (% age of ^{235}U) for the proposed core at BOEC

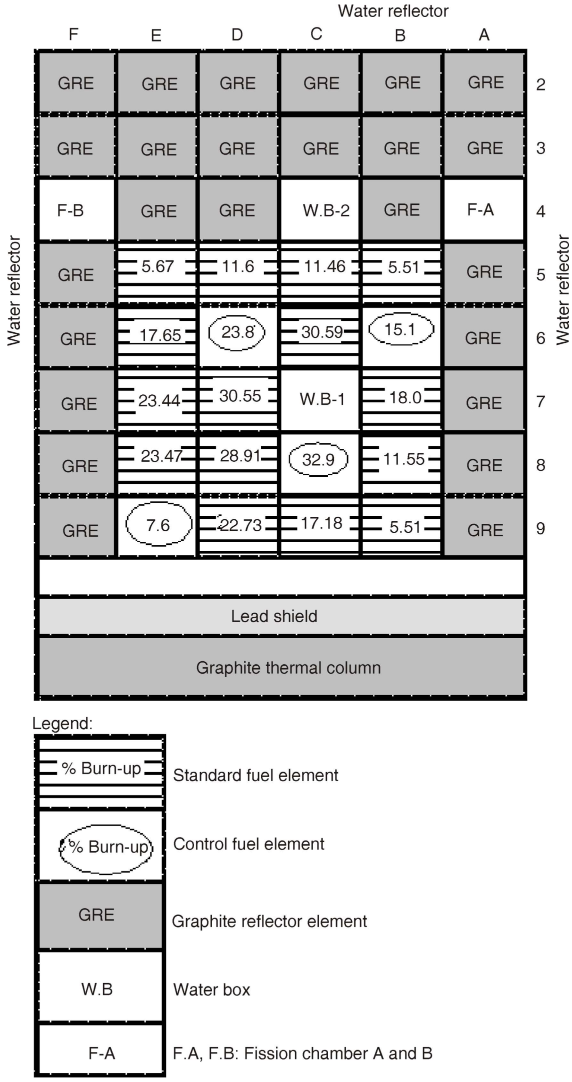

Figure 9. Burn-up (% age of ^{235}U) of the proposed core at EOEC

From tab. 5, it is clear that the ^{235}U depletion rate is high because of the high fission cross-section in the thermal region, while ^{238}U is depleted slowly, due to the fast fission and the internal conversion to plutonium isotopes. All plutonium isotopes are on the increase with burn-up.

Keeping the fixed core geometry for constant power generation, the flux and fissile concentration are inversely proportional to each other. When fissile contents are depleted, the flux increases, therefore the flux increases at EOEC. Moreover, with increasing burn-up, the fuel-to-moderator ratio decreases, leading to a soft neutron spectrum [2, 14]. Figure 10 shows the thermal neutron fluxes in the core, at different locations, including water box stands and control fuel elements at BOEC and EOEC.

The fast/thermal neutron flux averaged over each standard fuel element, control fuel element and water boxes for the proposed core are shown in figs. 11 and 12. The fast flux is increasing towards the core centre, while decreasing at the periphery, due to leakage. The thermal flux is at its maximum at C(7), *i. e.* the water box location for sample irradiation.

Table 5. Composition of fuel in the core and the discharged fuel element

Nuclide	Mass of nuclides [g] in core		Mass of nuclides [g] in the discharged fuel element	
	BOEC	EOEC	BOEC	EOEC
^{235}U	6467.46	6017.56	324.4397	294.74
^{238}U	29186.3	29139.0	1684.13	1680.0
^{239}Pu	66.86	96.33	7.88	9.38
^{240}Pu	4.68	8.48	0.75	1.19
^{241}Pu	0.89	1.94	0.18	0.36

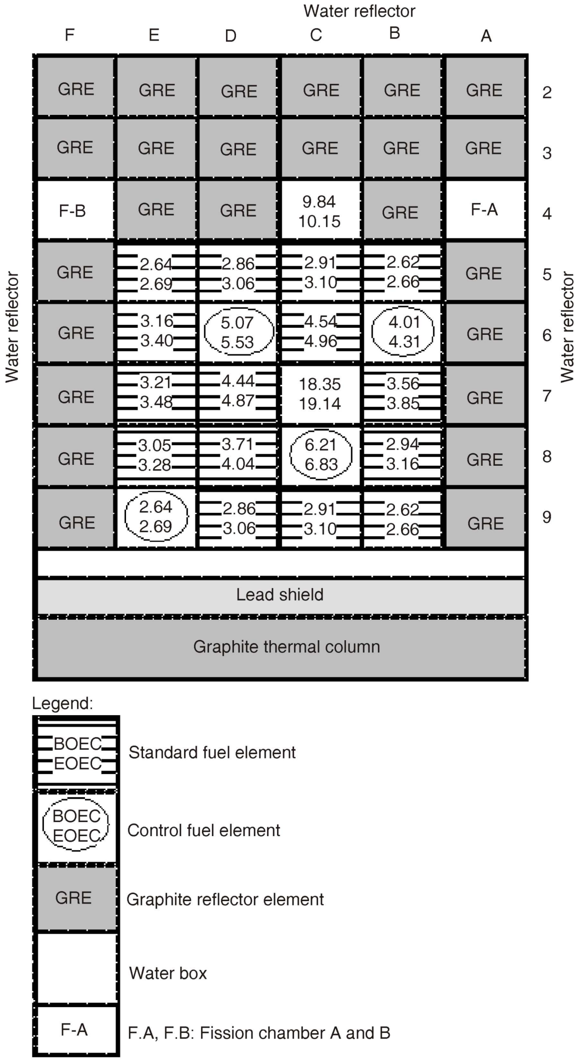

Figure 10. Thermal-neutron fluxes [10^{13} cm^{-2}s^{-1}] at BOEC (upper value) and EOEC (lowest value)

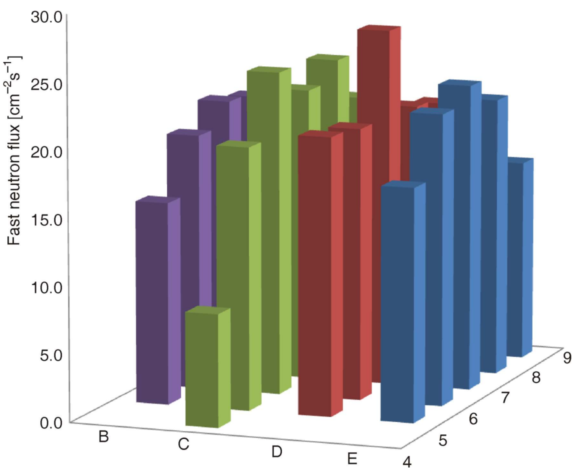

Figure 11. Fast neutron flux [10^{13} cm^{-2}s^{-1}] in the proposed core at BOEC

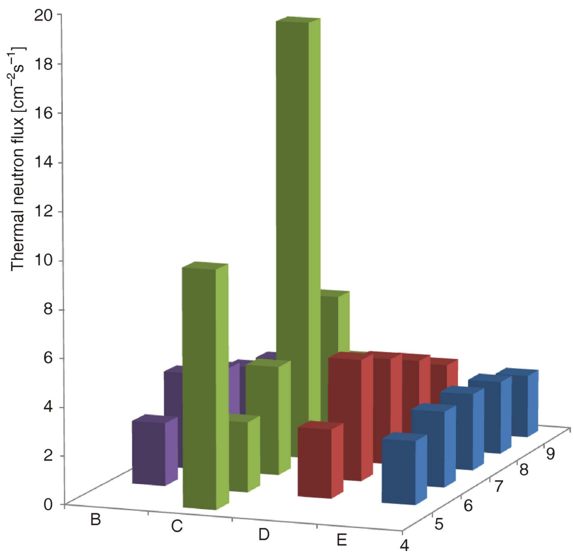

Figure 12. Thermal neutron flux [10^{13} cm^{-2}s^{-1}] in the proposed core at BOEC

CONCLUSIONS

With higher density fuel, the fuel-to-moderator ratio increases, the core otherwise criticality-moderated, becomes highly under-moderated. The performance of high-density fuel (U density 4.8 g/cm^3), with 15 standard fuel elements, 4 control fuel elements, 23 fuel plates per standard fuel element, and 13 fuel plates per control fuel element, has proven to be the most efficient of the cores examined. Our 2-D depletion analysis of the core reveals that it can be operated for 41 full-power days at 9 MW, continuously. Therefore, every standard fuel element will be irradiated for 205 full-power days, while every control fuel element will be for irradiated 164 full-power days. Although the discharge burn-up of the low-density fuel is higher, due to higher values, the thermal neutron flux is low at irradiation sites and, thus, the performance of these cores is inferior to those of the higher density fuels proposed.

AUTHOR CONTRIBUTIONS

Calculations were carried out by A. Muhammad under supervision and guidelines of M. Iqbal. T. Mahmood and M. Iqbal helped in modeling of reactor core. All authors discussed and analyzed the results. Manuscript was written by A. Muhammad and reviewed by T. Mahmood and M. Iqbal. Figures were prepared by A. Muhammad.

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Ата МУХАМАД, Масуд ИГБАЛ, Тајаб МАХМУД

УНАПРЕЂЕЊЕ МОГУЋНОСТИ РАВНОТЕЖНОГ ЈЕЗГРА ИСТРАЖИВАЧКОГ РЕАКТОРА

У циљу побољшања могућности равнотежног језгра Пакистанског истраживачког реактора-1, проучаване су различите конфигурације језгра са ниским обогаћењем силикатног горива (U_3Si_2-Al) густине уранијума од 3.285 g/cm^3. Потом је анализа проширена на силикатна горива веће густине уранијума – од 4.0 g/cm^3 и 4.8 g/cm^3. Избор језгра са најбољим особинама извршен је на основу критеријума да производ јединичног флукса и времена трајања циклуса, подељен масом ^{235}U, буде највећи.

У циљу анализирања особине језгра у погледу побољшања неутронског успоравања при коришћењу горива веће густине, проучаван је утицај ширине канала хладиоца свођењем броја плоча у стандардном/контролном горивном елементу. Обављени су прорачуни програмима WIMSD/4 и CITATION. За фисионе неутроне коришћена је десетогрупна структура за генерисање микроскопских пресека у програму WIMSD/4. Испитивање равнотежног језгра извршено је програмом CITATION коришћењем дводимензионалног модела језгра. На основу истражених перформанси показано је да силикатно гориво веће густине уранијума од 4.8 g/cm^3, без икакве промене стандардно/контролног горивног елемента сачињеног од 15 стандардних и 4 контролна елемента, има најбоље карактеристике од свих анализираних.

Кључне речи: истраживачки реактор, реакторско гориво, MTR-PC26, WIMS/D4, CITATION