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Neutronic feasibility study for neutron flux upgrading of Tehran research reactor

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HIGHLIGHTS

- The main goal is reaching to the average thermal neutron flux of the order of 1.5×10^{14} #.cm^{-2}.s^{-1}.
- Combining the TRR power upgrading with the compact core configuration is the main idea of this study.
- The results showed that TRR can be upgraded to 8.5 MW and the thermal flux larger than 1.5×10^{14} can be achievable.

ABSTRACT

The present work is concerned on neutron flux increasing in Tehran Research Reactor (TRR). TRR is a 5 MW pool-type research reactor with plate type fuels in which the light water is used as both the coolant and moderator. The main goal of this paper is reaching to the average thermal neutron flux of the order of 10^{14} #.cm^{-2}.s^{-1} in the central irradiation box. Combination of the TRR power upgrading with the compact core can enable us to reach a neutron flux higher than 1.5×10^{14} #.cm^{-2}.s^{-1} without violating the neutronic and thermal-hydraulic safety criteria. The compact core, with 19 and 5 standard and control fuel elements respectively, is used as a base for the neutronic analyses. Compact core with 26 fuel assemblies fulfilled all neutronic and operation criteria. Considering thermal hydraulic aspect from previous study lets TRR to be upgraded to 8.5 MW, resulting in neutron thermal flux greater than 1.5×10^{14}.

KEYWORDS
TRR
Neutronic Analyses
Power upgrading
MTR
CITVAP code

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Abbreviations

APPF	Axial Power Peaking Factor
ARO	All Rods Out
CFE	Control Fuel Element
FE	Fuel Assembly
C & C	Cold & Clean
HFP	Hot Full Power
HFPX	Hot Full Power with Xe
SDM	Shutdown Margin
SDM-1	Stuck Rod Shutdown Margin
IR Box	Irradiation Box
LEU	Low Enriched Uranium
MTR	Material Test Reactor
RCR	Reactivity Consumption Rate
PPF	Power Peaking Factor
RPPF	Radial Power Peaking Factor
SAR	Safety Analysis Report
SFE	Standard Fuel Element
SRF	Safety Reactivity Factor
SSR	Shim Safety Rod
TRR	Tehran Research Reactor
EOC	End Of Cycle

1 Introduction

Research reactors are sophisticated devices for basic and applied research in the fields of particle and nuclear physics, radiochemistry, activation analysis, materials sciences, nuclear power and nuclear medicine. These reactors also enable the testing of various types of nuclear fuel and the study of radiation resistance of new materials. Operators of many research reactors have found that their facilities are not being utilized as fully as they might desire. This can generally be attributed to a complex multitude of reasons. Most notably, many existing reactors are no longer capable of performing innovative research. Furthermore, neutron intensities at many facilities are lower than others neutron sources. Some reactors lack precise and determined direction following the fulfilment of their designed mission (IAEA, 2014). Some of the existing research reactors have been upgraded and renovated based

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on the above reasons and their owners decisions (Tőzsér, 2009; Israr et al., 2009).

Tehran Research Reactor (TRR) is a 5MW pool-type light water research reactor. TRR became critical, using HEU fuel that was more than 90% enriched in U-235, in 1967 with hot cells for the production of medical isotopes. In 1987, Argentina's Applied Research Institute converted the reactor core to run on LEU instead of HEU. During of 5 decades of TRR operation, various plans have been proposed to upgrade the power of the TRR to 10 MW, but none of them have not been implemented. The main reasons for TRR upgrading are the degradation of many parts of the TRR and increasing the volume of radioisotope production (Farhadi and Khakshournia, 2008). Refurbishment or increasing the applications of research reactor cannot be the goal and justification of increasing the power of a research reactor like TRR. Having a specific goal based on feasibility study can be the most important requirement in power upgrading of TRR. Increasing the volume of current radioisotope production, creating new applications such as nuclear fuel testing and materials, producing new industrial and medical radioisotopes can be considered as the result of power upgrading. In order to achieve this goal, there are two consecutive basic steps:

- The first step, is to increase the neutron flux in the irradiation sites by increasing the thermal power of the reactor without changing the core configuration.

- The second stage, includes increasing the neutron flux in the irradiation sites by changing the core configuration and compacting the core without increasing the thermal power of the reactor.

The results of a neutronic study, to explore the possibility of the TRR power upgrading from 5 to 10 MW with minimum changes in the primary cooling circuit, showed that from the neutronic aspect, there is no major limitations for the operation of the reactor at the 10 MW power level. The neutronic analysis was carried out for a fresh core with 22 SFE and 5 CFE under normal operating conditions (Afshar and Shahidi, 2002).

The results of a complete study from the neutronic and thermos-hydraulic point of views have been presented in the reference (Farhadi and Khakshournia, 2008). This study investigated the possibility of raising power of the TRR from the 5 MW to a higher level without violating the original thermal-hydraulic safety criteria. Different reactor powers (5 to 10 MW) and different core coolant flow rates (500 to 921 $m^3.h^{-1}$) are investigated. It was shown that, for the core configuration with 27 FEs (22 SFE + 5 CFE), 7.5 MW is achievable safely by gradually opening the butterfly control valve until the desired coolant flow rate is reached (800 $m^3.h^{-1}$) (Farhadi and Khakshournia, 2008).

Replacement of the TRR graphite reflector with heavy water, beryllium and beryllium oxide showed that the aforementioned replacement cannot noticeably increase the thermal neutron flux (Gholamzadeh et al., 2019). Compacting the core configuration as a method to increase thermal neutron flux of TRR core has been studied in order to provide desired neutronic condition to perform domestic fuel testing. TRR compact core configuration with 24 FEs was proposed for fuel test purposes with satisfying the neutronic and thermal-hydraulic safety criteria according to FSAR and OLCs of TRR (Arshi et al., 2021).

The main goal of this study is reaching to the average thermal neutron flux of the order of 1.5×10^{14} $\#.cm^{-2}.s^{-1}$ instead of only power upgrading. Combining the TRR power upgrading, with the minimum changes in the primary cooling system, with the compact core configuration is the main idea and methodology of this study which enable us to reach a neutron flux higher than 10^{14} $\#.cm^{-2}.s^{-1}$. In this work unlike previous power upgrading, which uses fresh fuel, burned-up fuel is used in the compact form of core configuration. The use of burned-up fuel reduces the maximum power peaking factor and increases the amount of power enhancement based on neutronic safety criteria. This paper investigates only neutronic aspects of thermal neutron flux upgrading of TRR.

2 Procedure

2.1 Description of TRR

Tehran Research Reactor is a 5 MW pool-type research reactor with heterogeneous solid fuels in which the light water is used as both the coolant and moderator. Tehran Research Reactor use U_3O_8-Al MTR type fuel. The reactor core is composed of SFEs and CFEs. which are made of 19 and 14 fuel plates, respectively. The cross-sectional view of low enriched uranium SFE and CFE are given in Figs. 1-a and 1-b, respectively (Report, 1989). Other details of LEU fuel assemblies and core parameters are given in TRR- Safety Analysis Reports (SAR) (Report, 2001). The main design data are given in Table 1.

2.2 Neutronic Design Criteria

The neutronic criteria that must be fulfilled for every core configuration are described as follows (Report, 1989):

- Minimum shutdown margin must be 50% of the excess reactivity.

- Minimum shutdown margin in absolute value must be equal to or greater than 3000 pcm.

- The Reactor must be sub-critical with the shutdown margin of at least 500 pcm with any of the safety rods 100% withdrawn.

- Reactivity worth of RR must be less than effective delayed neutron fraction.

- The power peaking factor and the number of fuel elements on the core must stay within the limits set in the thermal hydraulic analysis.

2.3 Simulation methodology

The MTR_PC package has been developed by INVAP S.E in order to perform neutronic, thermal hydraulic and shielding calculations of MTR-type reactor for personal

Table 1: Main Characteristics of Tehran Research Reactor.

Parameters	Values
Fuel elements:	
U-235 per SFE	290 g
U-235 per CFE	214 g
U per fuel plate	76 g
Meat:	
Enriched U_3O_8	20% in weight of U-235
U density	2.9617 g.cm^{-3}
Meat density	4.76 g.cm^{-3}
Void fraction	10.0%
Weight percentage	U-235 12.45%, U-238 49.78%, O 11.18%
Aluminum Meat	Purity 99.6% Density: 2.7 g.cm^{-3}
Frame and covers	Aluminum 6061 Density= 2.7 g.cm^{-3}
Shim and safety rods absorber	Ag-In-Cd Alloy (80, 15, 5% in weight respectively) Density: 10.17 g.cm^{-3}
Control rods' Cladding Material	AISI-316/L stainless steel Density: 7.95 g.cm^{-3}
Gap between absorber and clad	He (1 atm. pressure)
Regulating rod	AISI-316/L stainless steel Density: 7.95 g.cm^{-3}
Gird plate	Grid array X-Y Pitch: 7.71 × 8.1 cm
Grid plate material Grid z thickness	AL-1100, 12.7 cm, 54 holes, diameter: 6.19 cm, Max: 6.17 cm
Grid passing holes	Min: 40 holes, diameter: 2.222 cm with a reduction to 1.9053 cm
Reflectors	Water/Graphite

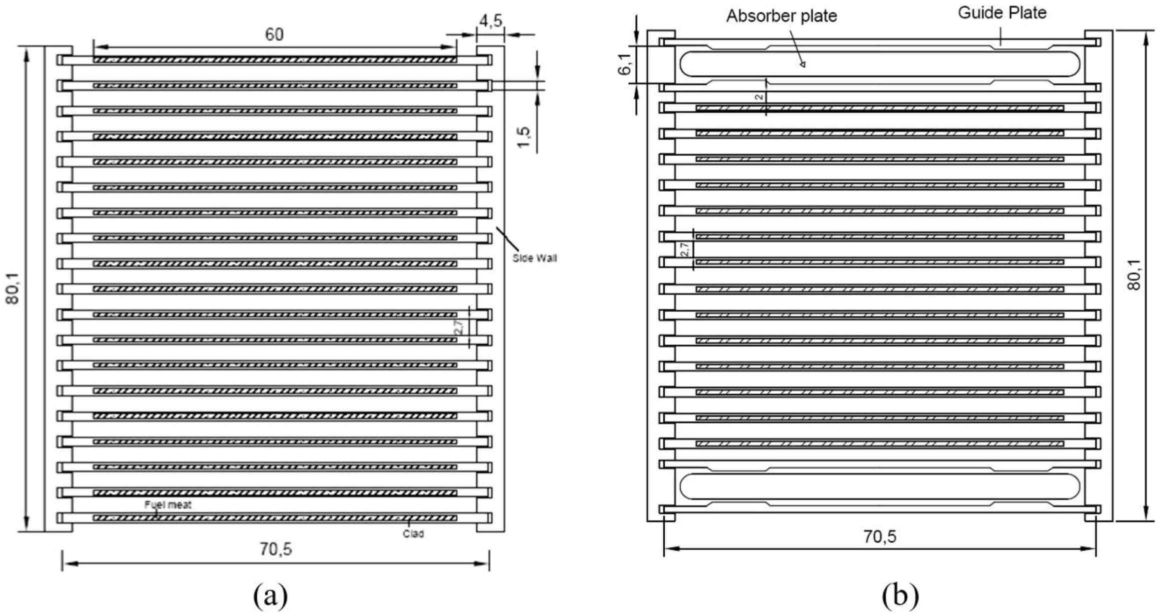

Figure 1: The cross-sectional view of the TRR fuels: a) SFE, b) CFE (all dimensions in cm).

computers (PC). In this research, WIMS-D4 (Askew et al., 1966), POS_WIMS, HXS and CITVAP (Villarino and Lecot, 1993) neutronic codes of MTR_PC package are used to calculate neutronic core parameters of TRR and TRR mixed-core. WIMSD employed for macroscopic cross-section generation, which provides nuclear cross-sections in the form of 69-energygroup structure. POS_WIMS is a post processor program of WIMS code used to condenses and homogenizes macroscopic cross section from WIMS output. CITVAP code is a new version of the CITATION-II code. It solves one, two or three-dimensional multi-group diffusion equation in rectangular or cylindrical geometries. HXS (Handle Cross-Section) program makes the connection between cell calculation and core calculation. Core calculations are performed with the CITVAP diffusion code, in X-Y-Z geometry, using the three-group energy structure according to Table 2 .This energy structure agrees with the 5-45-69 partition of the 69 groups WIMS library. WIMSD code was run with applying DSN and PERSEUS options to carry out required macroscopic cross-section for different states (C&C, HFP and HFPX) in each zone. Due to the proximity of DSN answers to the reference, SEQUENCE2 was used in WIMS calculations.

Table 2: Energy groups used for macroscopic cross-section generation by WIMSD.

Energy group	Energy range	Remarks
1	0.821 to 10.00 MeV	Fast
2	0.625 to 0.821 MeV	Epithermal
3	< 0.625 eV	Thermal

Table 3: Neutronic Parameters of the First Core of TRR.

Neutronic Parameters	Calculated	SAR
Excess Reactivity *(pcm) (could state)	6997	6916
Xenon equilibrium(5MW)	3010	3150
Total Worth of Safety Rods	21460	19457
Worth of RR	535	550
Radial	1.65	1.66
Axial	1.3	1.3
Total PPF	2.2	2.1 to 2.7
SRF	2.79	2.81

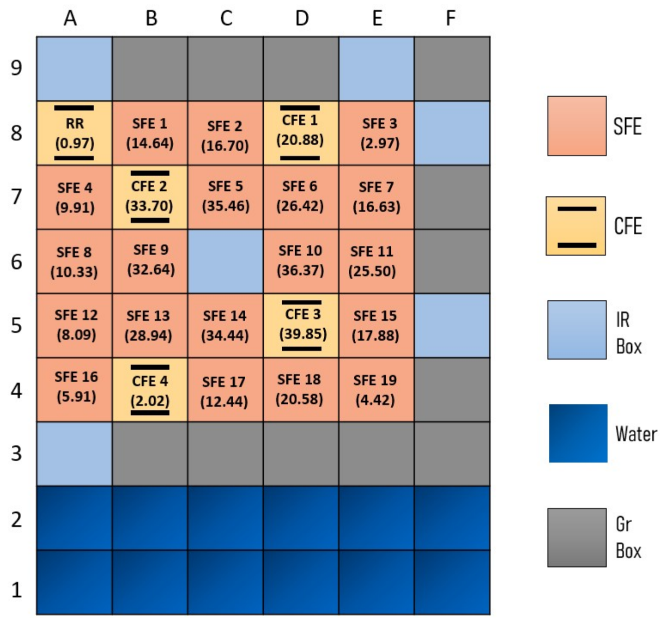

Figure 2: Compact core configuration containing 24 fuel assemblies.

2.4 *Validation of simulation methodology*

In order to validate the simulation methodology, the first core of TRR was simulated. This core contains 14 SFE, 5 CFE and water as reflector. The core configuration and specification have been given in the reference documents (Report, 2001). The neutronic parameters for the first operating core were calculated and compared with the value of SAR parameters in Table 3. Comparing the results shows the good agreement between the calculated and the SAR values.

3 Results and Discussion

The compact core configuration with 24 FEs is given in Fig. 2 as the reference core, which was operated to perform the domestic fuel test. The fuel management strategy of TRR is out to in. In this scheme, one fresh fuel loads in periphery of the core and the irradiated fuel is shuffled in toward the inner zone, while the fuel in the central zone is withdrawn from the core. This fuel management strategy reduces PPF and allows the power to be upgraded to higher power from neutronic aspect. The neutronic parameters of the compact core have been reported completely in the reference (Arshi et al., 2021) with MCNP code in the 5 MW. New calculation is done by CITVAP diffusion code in the 10 MW and the results are shown in Table 4 for three states C&C, HFP and HFPX. By comparing the excess reactivity of HFP and HFPX states with CC state the temperature and neutronic poisons (Xenon & Samarium) effects are calculated about -302 and -4205 pcm in 10 MW respectively. The cycle length of the core is calculated according to the reactivity consumption rate (6.7 pcm.MW^{-1}.d^{-1}) and EOC excess reactivity parameters. EOC excess reactivity is a reserve reactivity to compensate the negative reactivity effect of some tests and experiments that considered about 1500 pcm TRR. Although the compact core with upgraded power (10 MW) meets all neutronic safety criteria, but it is not practical core due to zero cycle length. Total PPF is obtained from multiplying APPF in RPPF in ARO state with taking account of the control rods insertion effect on PPF. The control rods insertion effect on total PPF is 1.15 in TRR (Report, 2001).

The purposed core configurations are made by adding one or two fresh fuel assemblies to the compact core. Figure 3 shows two configurations of 25 FEs core. Table 5 shows the neutronic parameters of 25 FEs configurations in three states. In the CC state, first two configurations a and b are compared together then the configuration a (due to more excess reactivity) is selected to perform neutronic analyses for HFP and HFPX states. The cycle length of the selected core configurations is 4 days, which is not best for operation.

To increase the cycle length of the core, one else fresh fuel is added. Figure 4 shows two core configurations with 26 FEs. Table 6 shows the results of neutronic parameters of the relevant arrangements. Adding any fresh fuel at the side of the core, increases about 1200 pcm excess reactivity in HFPX core state. This growth in excess reactivity increases the core cycle length to more than 15 days. In the C&C state, two configurations a and b are compared together and configuration b is rejected due to the violation of stuck rod criteria. so, configuration a is suggested as the final upgraded core configuration and passed all neutronic criteria.

3.1 *Thermal hydraulic consideration*

In this article power upgrading has been investigated only in neutronic aspect. Since the most important limitation of power upgrading is applied from the thermal-hydraulic aspect of the core, so thermal-hydraulic considerations must be taken into account. For this purpose, the results of previous studies have been used. Thermal-hydraulic studies of TRR power upgrade have been carried out

Table 4: Neutronic parameters of TRR compact core with 24 FEs.

Neutronic Parameters	C&C	HFP	HFPX	Safety Criteria
Effective Multiplication Factor	1.05307	1.04973	1.00534	-
Core Excess Reactivity (pcm)	5039.2	4737.2	531.4	-
Absolut SDM (pcm)	9376.1	9859.2	14459.5	> 3000
Absolut SDM-1 (pcm)	3418.7	3817.7	8277.3	> 500
APPF	1.3	1.3	1.3	-
RPPF	1.772	1.752	1.767	-
Total PPF*	2.65	2.62	2.64	¡ 3.0
Integral Worth of SSRs	14415.3	14596.9	14991	-
SRF	2.86	-	-	> 1.5
RCR (MW*day)	-	-	6.7	-
Cycle Length (full power day)	-	0	0	-

*Total PPF= APPF × RPPF × 1.15

Table 5: Neutronic Parameters of core configurations with 25 fuel assemblies.

Neutronic Parameters	C&C		HFP	HFPX	Safety Criteria
	25a	25b	25a	25b	
Effective Multiplication Factor	1.06588	1.06444	1.06262	1.01789	-
Core Excess Reactivity (pcm)	6180.8	6053.8	5892.9	1758.0	-
Absolut SDM (pcm)	7889.1	7850.1	8350.2	12857.8	> 3000
Absolut SDM-1 (pcm)	2093.7	1922.8	2476.5	6857.3	> 1000
APPF	1.3	1.3	1.3	1.3	-
RPPF	1.783	1.780	1.763	1.773	-
Total PPF	2.67	2.66	2.64	2.65	< 3.0
Integral Worth of SSRs	14700	13903.9	14243.1	14615.8	-
SRF	2.27	2.29	-	-	> 1.5
RCR (MW*day)	-	-	-	6.5	-
Cycle Length (full power day)	-	-	-	4	-

Table 6: Neutronic Parameters of core configurations with 26 fuel assemblies.

Neutronic Parameters	C&C		HFP	HFPX	Safety Criteria
	26a	26b	26a	26b	
Effective Multiplication Factor	1.07683	1.07405	1.07377	1.02883	-
Core Excess Reactivity (pcm)	7134.9	6894.8	6870.4	2802.2	-
Absolut SDM (pcm)	6261.2	6132.2	6685.5	11097.8	> 3000
Absolut SDM-1 (pcm)	683.9	315.7	1030.6	5321.7	> 500
APPF	1.3	1.3	1.3	1.3	-
RPPF	1.786	1.787	1.767	1.781	-
Total PPF	2.671	2.672	2.642	2.663	< 3.0
Integral Worth of SSRs	13396.1	13027	13555.9	13900	-
SRF	1.87	1.88	1.97	4.96	> 1.5
RCR (MW*day)	-	-	-	6.2	-
Cycle Length (full power day)	-	-	-	21	-

for different configurations, PPFs and cooling flow rates (Farhadi and Khakshournia, 2008). The pressure of 1.7 bar above the core of the TRR provides the sufficient pressure drop of the fluid passing through the core for configurations higher than 18 fuel assemblies. Experimental results showed the maximum cooling flow rate of 800 $m^3.h^{-1}$ possible with the minimum changes in the cooling circuit of TRR. Table 7 of reference (Farhadi and Khakshournia, 2008) shows the results related to the maximum upgraded power of the configuration 27 FEs for different PPFs in mass flow rate 800. Maximum PPF calculated is 2.7. With the conservative assumption of 2.8 for PPF in an arrangement of 26, the maximum upgradeable power is equal to 8.5 MW.

Table 8 shows the neutronic calculation of 26 FEs core with upgraded power of 8.5 MW. Neutron flux in three thermal, epi-thermal and fast groups in the central and surrounding irradiation channels are shown in Table 9. The results show the average thermal flux in the central channel is higher than 1.5×10^{14} so, the goal of this study is achievable.

4 Conclusions

In order to provide desired neutronic condition to achieve some new applications and fuel irradiation tests in TRR,

Table 7: Upgrading power (MW) for various flow rates and total peaking factors (Farhadi and Khakshournia, 2008).

Total P.P.F	Flow rate ($m^3.h^{-1}$)							
	700	750	775	800	850	875	900	921
2.4	9	9.3	9.6	9.9	10.2	10.5	10.8	11.1
2.5	8.6	8.9	9.2	9.5	9.8	10.1	10.4	10.7
2.6	8.3	8.6	8.9	9.1	9.4	9.7	10	10.2
2.7	8	8.3	8.5	8.8	9	9.3	9.6	9.9
2.8	7.7	8	8.2	8.5	8.7	9	9.3	9.5
2.9	7.4	7.7	7.9	8.2	8.4	8.7	8.9	9.2
3	7.2	7.4	7.7	7.9	8.2	8.4	8.6	8.9
3.1	7	7.2	7.4	7.7	7.9	8.1	8.4	8.6
3.2	6.7	7	7.2	7.4	7.6	7.9	8.1	8.3
3.3	6.5	6.8	7	7.2	7.4	7.6	7.8	8.1
3.4	6.3	6.6	6.8	7	7.2	7.4	7.6	7.8
3.5	6.2	6.4	6.6	6.8	7	7.2	7.4	7.6

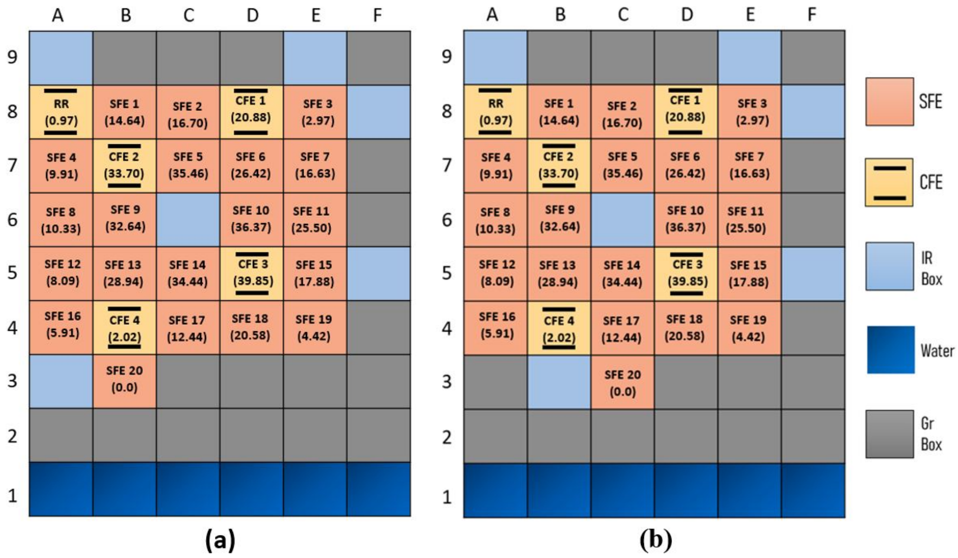

Figure 3: Compact core configuration with 25 fuel assemblies (20 SFE and 5 CFE).

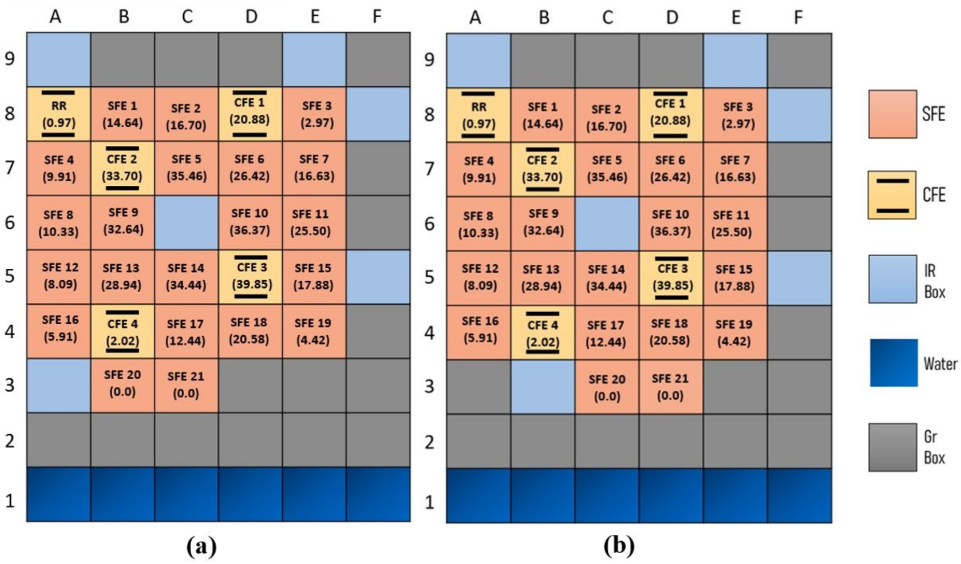

Figure 4: Compact core configuration with 26 fuel assemblies (21 SFE and 5 CFE).

thermal neutron flux of TRR core configurations must be increased. In this regard, compacting the core configuration, along with power upgrading can be solution to increase thermal neutron flux of TRR core and conse-

Table 8: Neutronic Parameters of core configuration with 26 fuel assemblies.

Neutronic Parameters		C&C	HFPX	Safety Criteria
Effective Multiplication Factor		1.07683	1.02854	-
Core Excess Reactivity (pcm)		7134.8	2775.2	-
Absolut SDM (pcm)		6261.3	11131.3	> 3000
Absolut SDM-1 (pcm)		683.9	5355.8	> 500
APPF		1.3	1.3	-
RPPF		1.786	1.779	-
Total PPF		2.671	2.661	< 3.0
SSRs worth (pcm)	SSR1	1318.2	1371.5	
	SSR2	2208.5	2288.3	
	SSR3	3359	3485.8	
	SSR4	2735.3	2861	
	RR	704	727.9	$< \beta_{\text{eff}}$
Integral Worth of SSRs		13396.1	13906.5	-
SRF		1.87	5.01	> 1.5
RCR (MW*day)		-	5.3	-
Cycle Length (full power day)		-	24	-

Table 9: Neutron flux in irradiation channels.

	Irradiation neutron flux ($\times 10^{13}$ n.cm^{-2}.s^{-1})					
	A9	E9	F8	C6	F5	A3
State	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal
C&C	0.72 1.51 4.37	0.85 1.75 4.61	0.93 1.90 4.83	4.48 8.40 15.30	1.55 3.22 7.62	1.78 3.40 7.47
HFPX	0.79 1.65 4.80	0.93 1.92 5.06	1.02 2.08 5.29	4.80 9.04 16.69	1.68 3.49 8.30	1.94 3.70 8.17

quently, increases the volume and variety of radioisotope production and other new applications. This research paper showed that the compact core with 26 fuel assemblies fulfilled all neutronic and operation criteria. Considering thermal hydraulic aspect from previous study, and the results of various core configurations showed that TRR can be upgraded to 8.5 MW and consequently the thermal flux larger than 1.5×10^{14} can be achievable.

Conflict of Interest

The authors declare no potential conflict of interest regarding the publication of this work.

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