Nuclear Engineering and Technology 50 (2018) 1017–1023

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original Article

Neutronic analysis of control rod effect on safety parameters in Tehran Research Reactor

Mina Torabi [a], A. Lashkari [b,*], Seyed Farhad Masoudi [a], Somayeh Bagheri [b]

[a] K.N. Toosi University of Technology, Faculty of Science Physics Department, Iran
[b] Research School of Reactor and Nuclear Safety, Nuclear Science and Technology Research Institute (NSTRI), Atomic Energy Organization of Iran (AEOI), Tehran, 14399-51113, Iran

ARTICLE INFO

Article history:
Received 10 February 2018
Received in revised form
6 May 2018
Accepted 28 May 2018
Available online 1 June 2018

Keywords:
Neutronic parameter
Power peaking factor
Moderator temperature coefficient
Fuel temperature coefficient
Kinetic parameters

ABSTRACT

The measurement and calculation of neutronic parameters in nuclear research reactors has an important influence on control and safety of the nuclear reactor. The power peaking factors, reactivity coefficients and kinetic parameters are the most important neutronic parameter for determining the state of the reactor. The position of the control shim safety rods in the core configuration affects these parameters. The main purpose of this work is to use the MTR_PC package to evaluate the effect of the partially insertion of the control rod on the neutronic parameters at the operating core of the Tehran Research Reactor. The simulation results show that by increasing the insertion of control rods (bank) in the core, the absolute values of power peaking factor, reactivity coefficients and effective delayed neutron fraction increased and only prompt neutron life time decreased. . In addition, the results show that the changes of moderator temperature coefficients value versus the control rods positions are very significant. The average value of moderator temperature coefficients increase about 98% in the range of 0–70% insertion of control rods.

1. Introduction

Research reactors play an important role in the development of nuclear science and technology. They provide a proper source of neutron for research, testing and analysis. They are also used for various applications in the fields of nuclear engineering, nuclear physics, radiochemistry, materials sciences, nuclear medicine, agriculture etc. Requests of research reactors fall into four wide-ranging categories: human resource development, irradiations, extracted beam work and testing [1]. In all of these applications, research reactors must be operated in a safe condition. Several activities related to normal operation involve safety evaluation. Generally, any activity or modification that influence neutronic, thermal-hydraulic and mechanical properties of the reactor should be supported by safety analyses. IAEA Safety Standards Series No: NS-R-4 establishes safety requirements for the utilization and modification of research reactors [2]. Safety Series No: SS 35-G2[1] provides guidance on the safety categorization of modification and utilization projects and the associated approval routes. In principle, safety parameters are divided in two neutronic and thermal-hydraulic category, which connected to them from PPF. To ensure the safe operation of reactor in both steady state and transient situation, all safety parameters must be analysed in any modification.

Many parameters such as core configuration, fuel type and burn up,etc., will have an effect on the safety parameters. During one cycle of research reactor operation, the positions of control rods change from BOC to EOC. Any nuclear reactor such as TRR at the BOC must contain more fuel than required to reach critical mass. This excess reactivity is necessary for overcoming temperature effects, fission-product build-up, and fuel depletion. As control rods are withdrawn and inserted, reactivity in the core is changed. The relation between reactivity and control rods positions are defined by six factor formula [3]. Pressurized Water Reactors (PWRs) use a combination of control rods and chemical shim (boron) for reactor control. The chemical shim consists of boric acid dissolved in the reactor coolant system and used for slowly changing the core reactivity and to ensure the reactor is adequately shut down.

* Corresponding author.
E-mail addresses: torabi_m@email.kntu.ac.ir (M. Torabi), alashkari@aeoi.org.ir (A. Lashkari).

[1] Safety Series No, 35-G2, INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1994.

https://doi.org/10.1016/j.net.2018.05.008

Nomenclature

TRR	Tehran Research Reactor
PPF	Power peaking Factor
LEU	Low Enriched Uranium
SFE	Standard Fuel Element
CFE	Control Fuel Element
OLC	Operational Limit Condition
LEU-CFE	Low Enriched Uranium- Control Fuel Element
BOC	Begin Of Cycle
EOC	End Of Cycle
MTC	Moderator Temperature Coefficient
FTC	Fuel Temperature Coefficient
HCF	Hot Channel Factor
SAR	Safety Analysis Report
GR.B	Graphite Box
E.B	Empty Box
SR	Shim Safety Rod
RR	Regulating Rod
ρ	Reactivity
$\alpha_{T,f}$	Fuel temperature coefficient
α_m	Moderator temperature coefficient
$\alpha_{T,m}$	Moderator temperature only
$\alpha_{D,m}$	Moderator density
α_V	Void reactivity coefficient
β_{eff}	Effective delayed neutron fraction
ι	Prompt neutron life time

Control rods are essentially fully withdrawn at full power. Operators use the control rods mainly for control of fast-changing reactivity transients, power changes, and reactor trips. The role of control rods in the research reactors are different from the power reactors. Usually research reactors do not have a chemical shim control system. Control rods in the research reactors can be designed and used for coarse and fine control, or fast shutdowns and also are employed to compensate for short term reactivity effects due to the fission product poisons, etc. Therefore the change in the position of control rods during one cycle of operation may be significant.

This work describes the effect of control rods on neutronic parameters and their relationship to core safety. If control rods are inserted too far into the core, power production at the top of core will be suppressed, resulting in increasing power production at bottom of core. Higher power at bottom of core could result in abnormally high fuel temperatures and fuel damage. High peak power results in high fission product concentrations in that location. The presence of control rods results in neutron flux profiles that have higher peaks and valleys than occur when the rods are fully withdrawn. In this article, the partially inserted of control rods in 3D geometry core configuration of TRR are simulated. The main purpose of this work is the evaluation of the effect of control rods movement on the, reactivity coefficients, kinetic and PPF safety-related parameters in TRR. The quantities of all parameters are treated in one of the TRR operating cores. In a special, case axial PPF simulation results are compared with experimental measurement. The methodology used to calculate the kinetic parameters is different from the one applied for neutronic and reactivity feedback coefficients [4,5].

1.1. Description of TRR

The TRR is pool type, heterogeneous, solid fuel, light water moderated research reactor, in which the light water is also used for cooling, shielding and reflecting. The reactor has been designed and licensed to operate at maximum thermal power level of 5 MW with forced cooling mode. The reactor core assembly has been located in two-section pool and may be operated in either of two sections of the pool. One of the sections contains experimental facilities such as beam tubes, a rabbit system, and thermal column. The other section is an open area for bulk irradiation studies. Diagram of TRR pool are given in Fig. 1 The major components of TRR are the pool (including embedment and accessories), bridge and support structure, core, cooling system, control and instrumentation, ventilation system, and the experimental facilities. Other details of the reactor description and core parameters are given in TRR-Safety Analysis Reports (SAR[2]). Elements of the reactor core are arranged in a 9 by 6 grid plate structure. The core configuration of the reference core and the burn-up of the fuel elements (in percent of the initial value of ^{235}U) at the BOC are given in Fig. 2.

1.2. TRR reactivity control system

TRR is controlled by the positioning of four shim safety rods made of neutron absorbing materials including Silver (Ag), Indium (In) and Cadmium (Cd) alloy (80%, 15%, 5% respectively) and one stainless steel regulating rod within the core lattice. The material selected as control rod absorber should have good absorption cross-section for neutrons and long lifetime as an absorber (not burn out rapidly). Silver-indium-cadmium rods are excellent neutron absorbers over a large energy range. The silver-indium-cadmium rods absorb essentially all neutrons from thermal energy to approximately 50 eV (SAR). Fork type assembly is used for both of safety and regulating rods. This assembly is composed of two reactivity control plates (Fig. 3). In any core configuration, the reactivity worth of the shim safety rods is sufficient to keep the core in a deep critical state in normal operating conditions as well as abnormal occurrences such as stuck rod. The drive mechanism moves the neutron absorbing materials in specified speed as it is determined in the OLCs. The drop time of the fork type absorber rods are always less than 700 ms to warranty the fast shut down of the reactor. For each core lattice configuration, the calibration of the shim safety rod worth is independently performed to ensure safety.

2. Methodology

The MTR_PC package has been developed by INVAP (Argentina) to perform neutronic, thermal hydraulic and shielding calculations of MTR-type reactors. In this research, WIMSD-5B [6] POS_WIMS, HXS, BORGES, and CITVAP v.3.1 [7], neutronic part of MTR_PC package are used to calculate kinetic and neutronic parameters of TRR reference core. The WIMSD ENDF/B-IV library was employed for the generation of macroscopic cross-section, which provides nuclear cross-sections in the form of a 69-energygroup structure. POS_WIMS is a post processor program of WIMS code used to condense and homogenizes macroscopic cross section for CITVAP from the WIMS output for neutronic and reactivity feedback coefficient calculations. The HXS program (Handle Cross-Section) makes the connection between cell and core calculations. The

[2] AEOI, 2001. Safety Analysis Report for the Tehran Research Reactor (LEU), Tehran-Iran.

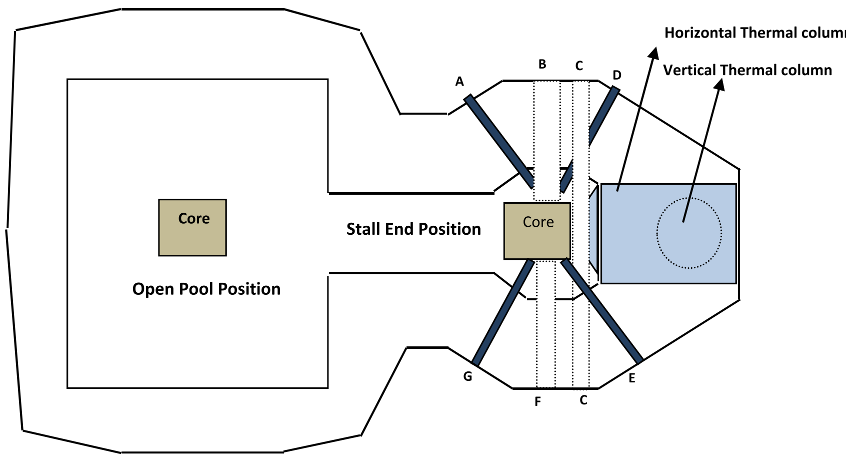

Fig. 1. Schematic view of TRR pool and facility.

A	B	C	D	E	F	
E.B	GR	GR	GR	E.B	GR	9
SFE 8.15	RR CFE_	SFE 28.44	SFE 28.94	SFE 18.26	SFE 11.45	8
SFE 24.51	SFE 36.09	SFE 46.53	SFE 52.53	SR2 CFE 1.10	SFE 28.55	7
SFE 21.37	SR1 CFE 36.07	SFE 50.24	E.B	SFE 44.07	SFE 28.55	6
SFE 31.66	SFE 31.36	SFE 42.08	SFE 55.62	SR3 CFE 58.03	SFE 15.70	5
SFE 4.03	SFE 22.59	SR4 CFE 48.71	SFE 54.00	SFE 39.15	SFE 3.62	4
E.B	SFE 13.07	SFE 23.99	SFE 37.84	SFE 3.79	E.B	3
GR	E.B	E.B	GR	GR	GR	2
GR	GR	GR	GR	GR	GR	1

SFE: STANDARD FUEL ELEMENT **CFE**: CONTROL FUEL ELEMENT
GR-BOX: GRAPHITE BOX **E.B**: EMPTY BOX
SR: SHIM SAFETY ROD **RR**: REGULATING ROD

Fig. 2. TRR 61 core configurations.

Fig. 3. View of fork type control rod assembly.

BORGES code prepares microscopic cross section libraries for CITVAP from the WIMS output for kinetic parameters calculations. This code homogenizes and condenses microscopic cross section in any region and energy group structures. Energy group structures for the calculation of neutronic and kinetic parameters are given in Table 1. Five and twelve energy group structures were used to calculate the cross section of SFEs and CFEs in WIMS code, respectively. Core calculations are performed with the CITVAP code using the three-group energy structure according to Table 1. This energy structure agrees with the 5-45-69 partition of the 69 groups WIMS library. The methodology and the interfaces between neutronic parts of MTR_PC is shown in Fig. 4. The simulation methodology used in this paper, has been validated in my previous paper [8]. The neutronic parameters for the first operating core of TRR were calculated and compared with the values of SAR parameters. This core contains 14 SFE, 5 CFE and water as reflector. Comparison of the results shows good agreement between the calculated and the SAR values.

Table 1
Energy group structures used in the calculations.

Energy range	Energy group		
	12 groups	5 groups	3 groups
1	10–0.821 MeV	10–0.821 MeV	10–0.821 MeV
2	0.821–0.00553 MeV	0.821–0.00553 MeV	821000–0.625 eV
3	5530–367.262 eV	5530–0.625 eV	0.625–0.00001 eV
4	367.262–48.052 eV	0.625–0.08 eV	
5	48.052–15.968 eV	0.08–0.00001 eV	
6	15.968–4.00 eV		
7	4.00–2.10 eV		
8	210–1.123 eV		
9	1.123–0.625 eV		
10	0.625–0.280 eV		
11	0.280–0.080 eV		
12	0.080–0.00001 eV		

3. Result and discussion

3.1. The effect of control rods position on PPF

The power peaking factor (PPF) is defined as the highest power density divided by the average power density in the reactor core. Since core power distribution is proportional to thermal neutron flux distribution, the ratio of maximum thermal neutron flux to average (Φ_{max}/Φ_{ave}) often referred to PPF. PPFs greater than 1.0 indicates that core flux profile is peaked. High PPF would indicate that high local power densities exist in reactor core. The hot channel or total PPF is the combination of axial and radial PPFs which are used to ensure that no localized power peaking could result in damage to the fuel. Fig. 5 shows the axial PPF versus distance from top of the active zone in different present of control rods presence in the 61th core configuration of TRR. Axial PPF distributions in 0 and 90 % (percentage values define as the ratio of inserted length of absorber to the total length of absorber) are

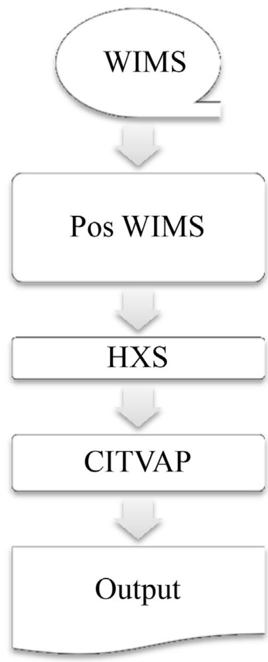

Fig. 4. Diagram of the MTR_PC calculations.

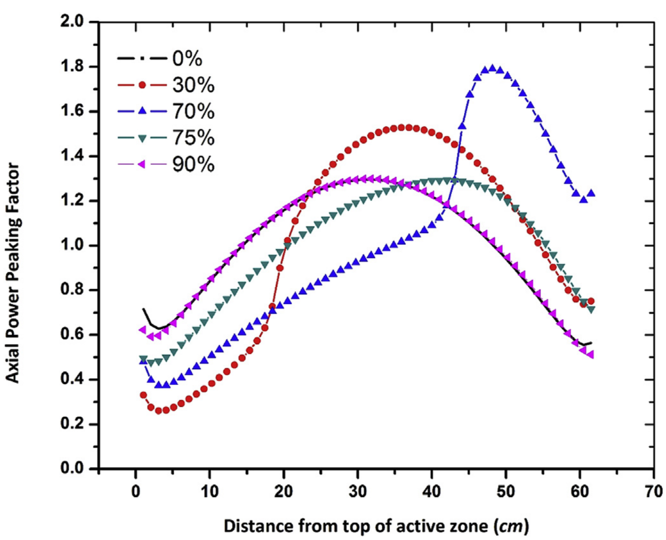

Fig. 5. Axial PPF of the reference core in different control rod positions.

similar and distribution in 70% has the greatest asymmetry. In CITVAP code, we gradually inserted a bank of control rods into the core and calculated PPFs in each step. Fig. 6 shows the variation of PPFs versus rods position. The behaviour of the axial PPF with respect to the rod position is almost opposite of the radial PPF behaviour from 0- 70 percent of the rod position. However, after 70%, both have a same increasing trend. Results show that the total PPF increases by the rod positions and reaches a maximum value at 45% until 70% rod positions. The maximum value of total PPF when considering rods effect is less than 3 which is used as the conservative value of total PPF in the SAR. In all of the simulations, power of TRR is assumed constant about 5 MW. In the both of simulation and practical, the inserting of control rods in the core change the flux distribution in the both radial and axial direction. Neutron flux is depressed in the region of the core where control rods are present and is greater in regions where there are no control rods (where control rods have been withdrawn). This subject causes more Non-uniformity of neutron flux in axial direction. On the other, the radial neutron flux become uniform. Therefore movement of control rods (0–70 %) causes increment in axial PPF and decrement in radial PPF. The rate of this rise and fall in PPFs is such that the total PPF is approximately constant in duration 45–75 % rods positions. When the control rods are near the bottom of the core (i.e. 75–100 %), the neutron flux peak will shift back to the core mid plane. Since the fully inserted rods are a uniformly distributed poison (in the vertical dimension), the axial flux distribution will return to its original shape.

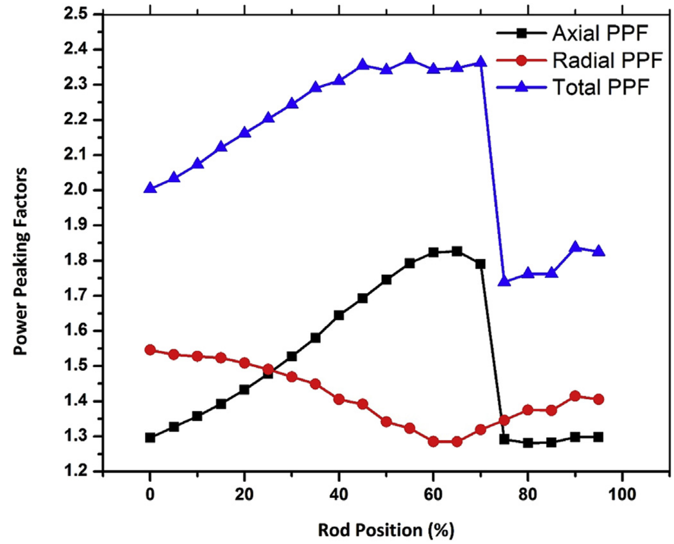

Fig. 6. Variation of PPFs vs. rod positions.

Fig. 7 shows a comparison between simulation and experimental axial PPF in the reference core 61. The positions of rods in the reference core are shown in Table 2. The experimental results are according to flux measurement in the reference core 61 [8]. Since the flux mapping has been measured only in the axial direction, could only determine the Axial PPF as a result. According to the thermal neutron flux mapping in the positions of D6 (central

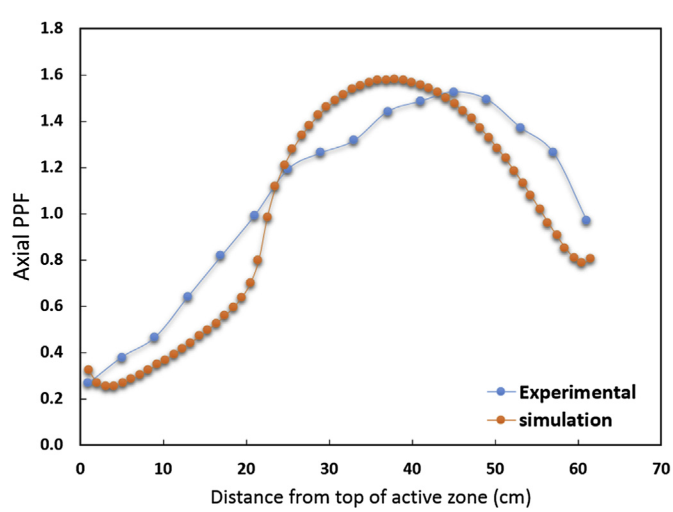

Fig. 7. Experimental and simulation value of the axial PPF.

Table 2
SRs position in the reference core No: 61.

SR	SRs % in
RR	50
SR1	35
SR2	35
SR3	35
SR4	35
Excess Reactivity (pcm)	3300

irradiations box of TRR), the maximum axial PPF is calculated to be approximately 1.5. According to the results shown in Fig. 6, the axial PPF predicted by the simulation for the 35% rod positions is approximately 1.6. Although the experimental value (1.5) is somewhat than simulation value (1.6), the discrepancy is considered justifiable by the authors. Part of this difference could be the smaller result of the calculation and measurement errors but the major difference may be traced back to the ununiformed burn-up of fuel assemblies due to control rod effects. Movement of control rods in the short time have not significant effects on the axial burn up of fuels. However, in each cycle, at BOC control rods are not fully out of the core. For example in my case study 35% of the control rods inserted in the core. From the BOC to the EOC of each core configuration, the control rods are gradually going up and this process takes a long time (about 30 days). This subject repeat in each cycle during the operation years, as a result the uniformity of the axial burn up is changed. Gharib has published quantitative analysis of the control rods effects on axial fuel burn-up in MTR research reactor in the literature [9].

3.2. The effect of control rods on reactivity feedback

Reactivity feedback is the portion of reactivity change arising from the effect of energy production, which includes temperature and void coefficients of reactivity. Temperature coefficient of reactivity due to fuel, coolant, and moderator component of a reactor core are defined as the change in reactivity per unit change in average temperature of that component. If $\alpha_{T,j}$ represents the temperature reactivity coefficient of a component j and α_V represents the void reactivity coefficient then they can be written as:

$$\alpha_{T,j} = \frac{\partial \rho}{\partial T_j}(\mathrm{pcm}/^\circ\mathrm{C}) \tag{1}$$

$$\alpha_V = \frac{\partial \rho}{\partial(\%\ \mathrm{Voids})}(\mathrm{pcm}/\%\ \mathrm{Voids}) \tag{2}$$

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \times 10^5(\mathrm{pcm}) \tag{3}$$

Where K_{eff} is effective multiplication factor corresponding to average temperature T of the core component j.

To study the effects of temperature (temperature and density corresponding to temperature) of the fuel and moderator on excess reactivity, the values of average temperatures are varied in steps of 20 °C around the reactor operating temperature range of the components. The temperature of the other core components were kept unchanged. The macroscopic cross-section sets for each component regions are generated using WIMSD at each step and reactivity is calculated by using the CITVAP code. The value of reactivity feedback coefficients, in each temperature (and in each void percent) are calculated by reactivity changes due to temperature and void perturbations.

The reactivity feedback coefficients of different control rod positions in the reference core configuration 61 are calculated and the results are compared in Table 3. The results of the fuel and moderator temperature feedbacks and void reactivity feedback are analysed separately. All coefficients are calculated at the BOC.

3.2.1. Moderator temperature coefficient (α_m)

The reactivity change per degree change in moderator temperature is the Moderator Temperature Coefficient (MTC) of reactivity. Units for the moderator temperature coefficient are pcm/°C. MTC is primarily a function of the moderator-to-fuel ratio and density of the moderator. The moderator-to-fuel ratio is the ratio of the number of moderator nuclei within the volume of a reactor core to the number of fuel nuclei. The reactor designer adjusts the amount of moderator with the fuel in the core (Nm/Nu ratio) to an optimum value to ensure a negative MTC throughout core life. MTR type reactor are designed with an under moderated condition that provides a negative MTC, since a negative MTC is desirable. A negative MTC results in a decrease in power when a power increase causes the moderator temperature to rise. Changes in the Nm/Nu ratio affect the thermal utilization factor and the resonance escape probability, which in turn affect the k_{eff} and reactivity or more precisely the MTC. As the moderator, temperature increases the ratio of the moderating atoms (molecules of water) decreases because of the thermal expansion of water. This, in turn, causes a hardening of neutron spectrum in the reactor core resulting in higher resonance absorption and decreases the Resonance escape probability. Also decreasing density of the moderator causes that neutrons stay at a higher energy for a longer period, which increases the probability of non-fission capture of these neutrons. This process is one of processes, which determine the MTC. With decreasing the Nm/Nu ratio due to moderator temperature, thermal utilization factor increases. The value of the thermal utilization

Table 3
Reactivity coefficients in 61 core for positions of 0%,35%,50% and 70%.

Inserted rods position (%)		0	35	50	70
FTC (20–340 °C) dR/dT [pcm/C]		−1.65 to −1.14	−1.71 to −1.19	−1.68 to −1.23	−1.69 to −1.24
Average value		−1.39	−1.45	−1.45	−1.46
MTC dR/dT [pcm/C] (20–114 °C)		−13.65 to −26.19	−14.55 to −30.86	−16.70 to −33.52	−21.31 to −36.98
Temperature only		−8.89 to −8.7	−10.25 to −9.62	−11.56 to −10.79	−14.19 to −13.17
Average value		−8.79	−9.93	−11.17	−13.68
Density only		−4.76 to −7.12	−4.3 to −21.24	−5.14 to −22.73	−7.12 to −23.81
Average value		−5.94	−12.77	−13.93	−15.46
Void reactivity feedback (0 to 40%)		−189 to −6.12	−189.57 to −544.07	−206.09 to−577.43	−325 to −976
Kinetic parameters	β_{eff} (pcm)	763.31	766.67	770.10	776.44
	ι(μs)	57.280	57.10	57.025	56.695

factor is given by the ratio of the number of thermal neutrons absorbed in the fuel (all nuclides) to the number of thermal neutrons absorbed in all the material that makes up the core.

In the under moderated region, a decrease in the Nm/Nu results in a decrease in k_{eff}, equivalent to negative reactivity.

Moderator temperature was changed in the range of 20–114 °C and densities were changed in accordance with temperature variations. MTC values are shown in Table 3 in the mentioned temperature range. Fig. 8 shows the changes of MTC (%) versus the control rods positions. The absolute average value of $\alpha_{T,m}$ increases about 56% in the range of zero to 70% inserted of control rods. The variation of $\alpha_{p,m}$ not only in the moderator temperature range is significant but also the average value of this parameter grows very rapidly with increasing the presence of control rods. The average of the total MTC ($\alpha_{T,m} + \alpha_{p,m}$) has a 98% increment in 70% control rods inserted.

The reason for this effect is that control rods act as leakage boundaries. When density of moderator decreases with rods partially inserted, the number of Hydrogen atoms per unit volume decreases, and therefore neutrons will travel further between collisions and have an increased chance of leaking out of the reactor. Consequently, this leads to lower reactivity and increase in the neutron migration length. It is then more likely for a thermal & epithermal neutron to be captured by a control rod, causing the thermal non-leakage probability to decrease, and adding negative reactivity. When the moderator temperature is raised only with partially-inserted rods, the neutron spectrum tends to hardening. The shifting of the average neutron energy to higher values results in the control rods worth to increase. Since the rods contain material that has a high cross-section for absorption of epithermal neutrons. Therefore, MTC becomes more negative with control rods inserted. If having rods inserted makes MTC more negative, rods withdrawn makes MTC more positive. Another reason is that by inserting control rods in the core the ratio of Nm/Nu decreases and the reactor core become more under-moderated, which results more negative MTC.

3.2.2. Fuel temperature coefficient (FTC)

The MTC provides an inherent safety feature for research reactors; The fuel temperature coefficient (FTC) is just as much an inherent safety feature in that it results in negative reactivity on a power/fuel temperature increase and, as an added benefit, It is the first and fast acting against power rising. The FTC is the change in reactivity per degree change in fuel temperature (Δk/k/°C). The FTC feedback also responds more quickly to an increasing power transient than MTC, since reactor power causes an immediate increase in fuel temperature. The moderator lags due to the time for the transfer of heat from the fuel to the moderator. This is also true for decreasing power (fuel temperature decrease). In an accident involving a large positive reactivity insertion, The MTC cannot slow the reactor power rise for several seconds due to the delay in the moderator temperature change, whereas the FTC starts adding negative reactivity immediately. Another name applied to the FTC is the Doppler reactivity coefficient, often shortened to Doppler. This coefficient was named after the Doppler Effect or Doppler broadening of the U-238 and Pu-240 resonance peaks. The Doppler broadening phenomenon occurs when the fuel temperature increases and causes the target nucleus to have more energy. As a result, the relative energy between the target nucleus and the incident neutron changes and the acceptable neutron energy band that the nucleus will absorb a neutron widen. However, the dominant effect is that the nucleus will absorb a broader band of neutrons (off-peak neutrons). This effect plays a dominant role in low enriched cores, since there is much more U-238 in the core relative to highly enriched cores.

The fuel temperature is varied in the range from 20 to 340 °C in steps of 20 °C. The FTC values are shown in Table 3 in the mentioned temperature range. The fuel reactivity coefficient decreases with an increase in fuel temperature. At low fuel temperatures, the resonance absorption peaks for uranium-238 and plutonium-240 are very narrow, and only a small fraction of the neutrons passing through the resonance energy spectrum are absorbed. Thermal neutron energy is relatively low at low fuel temperatures, and a sizeable fraction of the neutrons is absorbed in the fuel by uranium-235. A small increase in fuel temperature causes a significant increase in the number of neutrons absorbed in the fuel by uranium-238 and plutonium-240 in the resonance range.

Additionally, uranium-235 absorbs a slightly lower number of thermal neutrons due to the slightly higher energy thermal neutrons. At high fuel temperatures, the resonance absorption peaks for uranium-238 are broad, and a large fraction of the neutrons slowed down in the core are captured in the resonance range. A small increase in temperature results in a small fractional increase in the number of neutrons resonantly absorbed, and a small decrease in the number of thermal neutrons absorbed in fuel by uranium-235. This results in the effect from the Doppler coefficient being smaller at higher fuel temperatures (smaller change).

By increasing the control rod insertion in the core, $|\alpha_{T,F}|$ increases, but not significant. The percentage of the average changes is 5% lower with respect to the situation in which all control rods are withdrawn out of core. By inserting control rods in the core, some of moderator replaced with absorber materials and thermal utilization factor decreases and therefore the neutron spectrum tends to hardening. Slowing down length and time for neutrons increases when the thermal utilization factor decreases. Changes in resonance absorption peaks (Doppler) will now be more significant since neutrons are spending longer periods in the resonance energy range. This means that the Doppler coefficient (FTC) is more negative at high inserted control rods value. The amount influence of these changes on the reactor behaviour characterized by sensitive analyses by PARET/ANL code. PARET/ANL one of the thermo-hydraulic codes of MTR_PC package that used to transient analyses such as reactivity insertion and loss of flow in TRR [10].

3.2.3. Void reactivity coefficient (α_v)

The void coefficient is the change in reactivity per percent change in void volume (Δk/k/percent void). Voiding may occur in

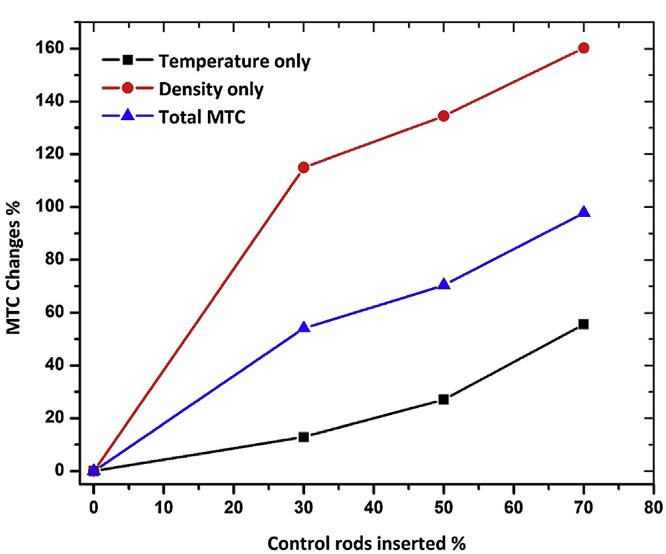

Fig. 8. Control rod position effects on MTC.

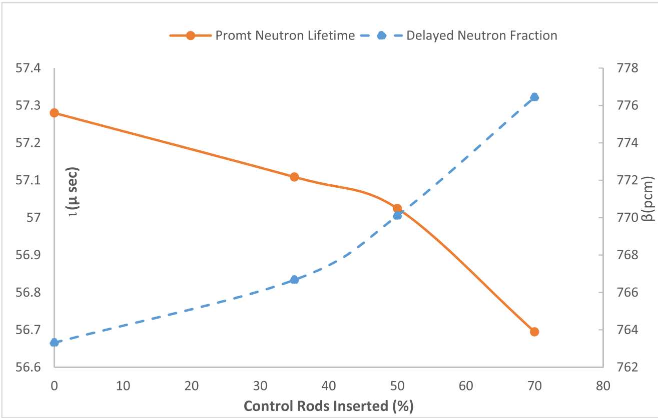

Fig. 9. Kinetic parameters versus control rods position (%).

research reactors when power increases to higher levels. The void displaces the moderator from the coolant channels within the core. This reduces the moderator-to-fuel ratio, and in an under moderated core, results in a negative reactivity. Bulk boiling of the moderator/coolant does not occur in a research reactor. Voids have the effect of reducing the moderator density in the area of the void. The result is similar to an increase in moderator/coolant temperature that lowers moderator density. The value of αV for different percentages of void from 0 % to 40 % is calculated with the CITVAP code in a four control rods inserted state according to Table 3. Macroscopic cross sections are evaluated by WIMS code for moderator and coolant regions for different void fractions, by changing the density of water accordingly. The minimum and maximum values of α_V are shown in Table 3. It is evident from Table 3 that absolute value of αv increases with void percentage in each state. The main reason for this increment is the loss in water density, due to void, which reduces the neutron thermalization and increases the 238U resonant absorption, thus the core reactivity decreases. The average value of αv increased with increasing the control rods positions in the TRR core due to the reason mentioned in the MTC discussion in Section 3.2.

3.3. The effect of control rods on kinetic parameters

In the MTR_PC computer code, first-order perturbation theory is used to calculate kinetic parameters.

Fig. 9 show that the increasing of the control rods position in the reference core results in a little reduction in ι and an increase in β_{eff}. Although any increase or decrease of these parameters can be justified, but in practice it can be concluded that these changes do not have any major effect on the dynamic behaviour of the reactor core. The reason for these changes can be explained by recalling the shifting of the average neutron velocity to higher values, which is the result of increased the control rod insertion. Therefore, the contribution of thermal fission decreases and fast fission increases. With increasing fast fissions in the reactor core, the value of ι decreased and β_{eff} increased due to increasing ^{238}U contribution in β_{eff} [5].

4. Conclusion

The main objective of this work is to calculate the effects of the control rods on neutronic parameters.

Gradually inserting of the control rods bank into the core result in an increase in the axial PPF and a decrease in the radial PPF until 70% rod position. After the 70% rod position both PPF have a same increasing trend. The total PPF increased with rod positions and reached to maximum value in 45% until the 70% rod positions. In the axial PPF case, comparison between simulation and experimental axial PPF in the reference core 61 in 35% rods position show a good agreement between the results.

The changes of MTC value versus the control rods positions was very significant. The average values of $\alpha_{T,m}$, $\alpha_{p,m}$ and total MTC increase about 56%, 160% and 98% in the range of zero to 70% inserted of control rods respectively. Control rod insertion results in a decreased thermal neutron utilization, which results in a large negative MTC at the higher inserted present. The average value of α_v increases with increased the control rods positions in the TRR core due to the reason discussed in Section 3.2.

Results show that the increasing of the control rods position in the reference core results in little reduction in the prompt neutron lifetime and an increase in effective delayed neutron fraction. The reason for these changes can be traced back to the shifting of the average neutron velocity to higher values, which results from the increase in the control rods insertion.

Appendix A. Supplementary data

Supplementary data related to this article can be found at https://doi.org/10.1016/j.net.2018.05.008.

References

[1] IAEA-TECDOC-1234, The Applications of Research Reactors, IAEA, Viana, 2001.
[2] IAEA, Safety of Research Reactors. Iaea Safety Standards Series, 2005. NS-R-4.
[3] J.R. Lamarsh, A.J. Baratta, Introduction to Nuclear Engineering, 2001.
[4] A. Lashkari, et al., Neutronic analysis for Tehran research reactor mixed-core, Progr. Nucl. Energy 60 (2012) 31–37.
[5] A. Lashkari, H. Khalafi, H. Kazeminejad, Effective delayed neutron fraction and prompt neutron lifetime of Tehran research reactor mixed-core, Annal. Nucl. Energy 55 (2013) 265–271.
[6] T. Kulikowska, WIMSD-5B: a Neutronic Code for Standard Lattice Physics Analysis, Distributed by NEA Data Bank, Saclay, France, 1996.
[7] E.V.a.C. Lecot, Neutronic Calculation Code CITVAP 3.1.
[8] A. Lashkari, et al., Experimental study of neutronic parameters in Tehran research reactor mixed-core, Progr. Nucl. Energy 83 (2015) 398–405.
[9] M. Gharib, M. Arkani, A. Hossnirokh, Design and application of MTR fuel assemblies in new proposed inverted mode, Nucl. Eng. Des. 240 (10) (2010) 2981–2987.
[10] W.L. Woodruff, R.S. Smith, ANL/RERTR/TM-16 ANL/RERTR/TM-16, 2001.