Nuclear Engineering and Technology 50 (2018) 1037–1042

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

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

Original Article

Neutronic assessment of BDBA scenario at the end of Isfahan MNSR core life

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ARTICLE INFO

Article history:
Received 8 January 2018
Received in revised form
18 May 2018
Accepted 3 June 2018
Available online 7 June 2018

Keywords:
BDBA
Induced reactivity calculation
Regulating rod worth
MNSR
Neutronic assessment

ABSTRACT

The present study aims to assess the excess induced reactivity in a Miniature Neutron Source Reactor (MNSR) for a Beyond Design Basis Accident (BDBA) scenario. The BDBA scenario as defined in the Safety Analysis Report (SAR) of the reactor involves sticking of the control rod and filling of the inner and outer irradiation sites with water. At the end of the MNSR core life, 10.95 cm of Beryllium is added to the top of the core as a reflector which affects some neutronic parameters such as effective delayed neutrons fraction (β_{eff}), the reactivity worth of inner and outer irradiation sites that are filled with water and the reactivity worth of the control rod. Given those influences and changes, new neutronic calculations are required to be able to demonstrate the reactor safety. Therefore, a validated MCNPX model is used to calculate all neutronic parameters at the end of the reactor core life. The calculations show that the induced reactivity in the BDBA scenario increases at the end of core life to 7.90 ± 0.01 mk which is significantly higher than the induced reactivity of 6.80 mk given in the SAR of MNSR for the same scenario but at the beginning of the core's life. Also this value is 3.90 mk higher than the maximum allowable operational limit (i.e. 4.00 mk).

1. Introduction

To ensure the safety of nuclear research reactors throughout their lives, from siting to decommissioning, and to prevent any probable accident resulted from their malfunction, the board of governors of the International Atomic Energy Agency (IAEA) chose to follow the Code of Conduct on the Safety of Research Reactors on March 8, 2004. According to the code of Conduct, it is recommended that operating organization undertakes the periodic safety examination of the reactors at appropriate intervals throughout the reactor's life. That is to say organization investigates the reactors' modifications, changes in their utilization and significant experimental activities as well as the manner their aging is managed [1]. In addition, Specific Safety Requirements (SSR-3) of IAEA Safety Standards reflect that the safety analysis report of research reactors shall be periodically updated over the research reactor's operating lifetime [2].

The Miniature Neutron Source Reactor (MNSR) is a 30 kW research reactor which designed by China based on the Canadian HEU SLOWPOKE-2 reactor. This reactor can be applied for neutron activation analysis (NAA), neutron radiography, boron neutron capture therapy (BNCT), radioisotope production, and training [3–7]. The safety analysis report (SAR) is the only published document on Beyond Design Basis Accident (BDBA (analysis in MNSR research reactor. However, several studies have examined the neutronic parameters of MNSR [8–11]. The effect of three different reflector materials on the core excess reactivity, i.e. graphite, beryllium (Be) and heavy water, in the MNSR have been studied by M. Albarhoum. The results also shows the reactivity worth of 1.9743 mk and 1.0575 mk for filling the inner irradiation site (IIS) and the outer irradiation site (OIS) with water, respectively [12]. Iqbal et al. have investigated some key parameters including the effective delayed neutron fraction (7.71 mk), the excess reactivity (3.96 mk), the worth of control rod (6.5 mk), and the reactivity temperature coefficient (0.132 mk/°C) in Pakistan MNSR when the first 1.5 mm top Be reflector was added [13]. In another work, Khattab and Sulieman have applied MCNP-4C code to evaluate some neutronic parameters of MNSR for fresh fuel including the core excess reactivity (3.97 mk), the control rod worth (6.54 mk), the worth of three reactivity regulating rods (1.2 mk), the worth of top beryllium shim plates (20.99 mk), and the effective delayed neutron fraction (7.54 mk). Furthermore, they have

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https://doi.org/10.1016/j.net.2018.06.001

calculated the thermal and fast neutron fluxes in Syrian MNSR's IIS and OIS [14,15]. Nawaz et al. have also studied the MNSR core life neutronic parameters for HEU core and have compared the results with LEU core parameters [16]. Finally, the burn up calculation of Isfahan MNSR, 15 years after the first operation, have been performed using IRBURN code system by Feghhi et al. [17].

To simulate the BDBA scenario and calculate the thermohydraulic response of the reactor, the first step is to evaluate the maximum value of induced excess reactivity which could insert into the core in the scenario. Maximum induced reactivity given in SAR of Isfahan MNSR in the BDBA scenario is 6.80 mk which was calculated at beginning of operational life (BOL) of MNSR core [18,19]. The SAR defined BDBA as a scenario resulted from two design basis accidents: 1) sticking the control rod and 2) filling all IIS and OIS with water. The SAR results confirmed that the MNSR is safe for 6.80 mk of reactivity insertion accident.

To compensate the loss of fuel reactivity as a result of fuel burnup, Be plates are gradually added to the top of the reactor core during its life time [11]. This increases the core's height and changes the neutron flux distribution and consequently affects most of core neutronic parameters. Neither the SAR nor the other studies have assessed the changes occurred in the core geometry as well as the effect such changes may have on the core physics parameters in the BDBA scenario.

This study investigates the neutronic parameters and calculates the induced reactivity in the BDBA scenario at the end of operational life (EOL) of MNSR core, i.e. when all Be reflectors are added to the top Be tray. To achieve this goal, first, a detailed 3D model of the reactor is modeled using the MCNPX Monte Carlo code and after the validation of the model using the available experimental data, the verified MCNPX model is then applied to calculate the induced reactivity into the core.

The applied scenario in this work is the same as the one investigated by the SAR. In both cases, the control rod stuck and the filling of the irradiation sites are postulated. This incident can occur as a result of an earthquake when the control rod gets out of the core [3]. The difference between this study and the work done by SAR is in the time of the scenario, i.e. the BOL in SAR and the EOL in this paper. In addition to the features enumerated for the previously described BDBA scenario (i.e. filling all IISs and OISs with water and the control rod stuck), the considered scenario in this paper has another characteristic that is filling all inner and outer irradiation air ducts and all regulating tubes with water.

2. MNSR description

The Miniature Neutron Source Reactor (MNSR) is a compact tank-in-pool type research reactor that uses high-enriched uranium (HEU) alloy in form of UAl_4 as fuel, Be as reflector and demineralized light water as moderator and coolant. This reactor is an under-moderated reactor with only one control rod that is made up of cadmium to regulate the power. MNSR is a safe reactor due to its characteristics such as: 1) an excess reactivity less than half of the effective delayed neutrons fraction ($\rho_{ex} \leq 0.5\ \beta_{eff}$), 2) a high negative temperature coefficient of reactivity, and 3) the natural circulation of fluid [20].

The Isfahan MNSR core is composed of 343 UAl4-Al fuel rods with Al cladding, 4 tie rods and 7 dummy rods that are arranged in ten concentric circles. The schematic of fuel rod structure is shown in Fig. 1. The fuel assembly is surrounded by annular Be reflector, in and out of which there are five IISs and five OISs, that are utilized for NAA. The reactor configuration includes one to four Regulating Rod (RR), fission chamber and thermocouples.

The amount of excess reactivity of one fuel rod varies depending on the position of the fuel rod. It is 0.81mk and 0.72 mk in the tenth and ninth circles, respectively. Typically, the core consists of a specific number of fuel rods based on the initial calculations and one to four Stainless Steel (S-S) RRs are also added to control the excess reactivity higher than 4 mk. In Isfahan MNSR, the use of 343 fuel elements and one RR gives the cold core excess reactivity of 3.86 mk [3].

Three types of Be reflectors are used in MNSR core including the annular Be, the bottom plate Be and the top Be shims. Considering the ^{9}Be isotope (n, 2n) and (γ, n) reactions, MNSR is designed to have the possibility of adding top Be shims to increase the excess reactivity, and thus extending the lifetime of the core. The top Be shims will be gradually added to the top of the reactor core in the beryllium tray (see Fig. 2). For this purpose, several top Be shims of

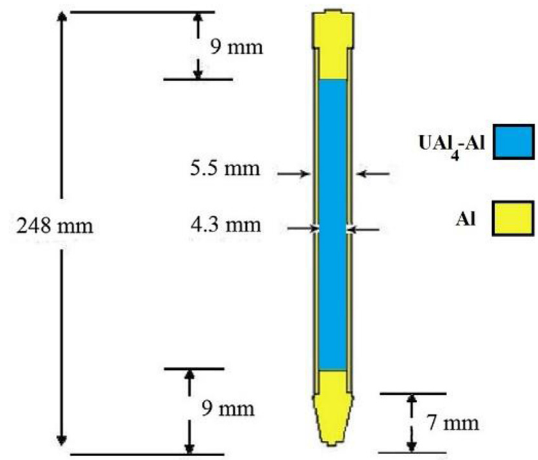

Fig. 1. Schematic of Isfahan MNSR fuel rod.

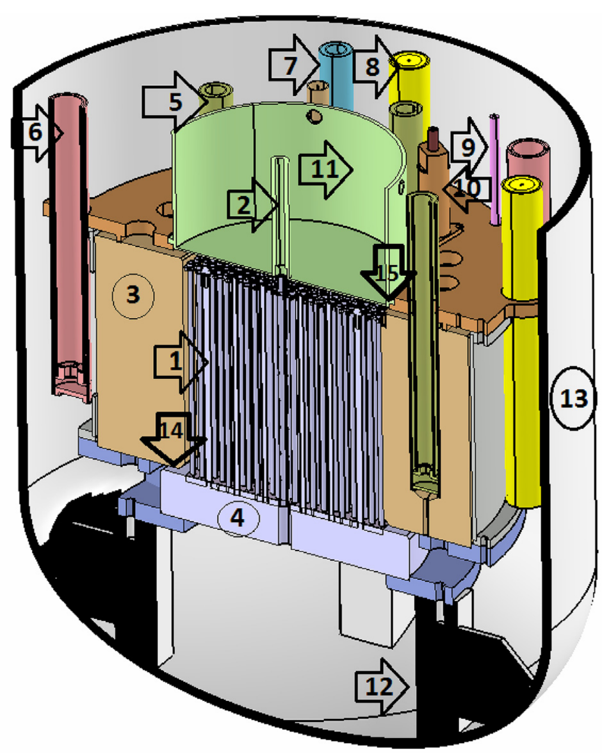

Fig. 2. Schematic view of the longitudinal section of MNSR: 1-Fuel Elements 2- Control Rod 3-Annular Be 4- Bottom Be 5- IIS 6- Small OIS 7- RR 8- Big OIS 9-Fission Chamber 10- Thermocouple 11- Be Tray 12-Supporting Legs 13- Vessel 14- Inlet orifice 15- Outlet orifice.

each thickness of 1.5, 3, 6 and 12 mm are available and a combination of them are applied to achieve the desired Be thickness. The equivalent reactivity worth for unit thickness decreases as the thickness of the Be shims increases and finally reaches to its saturated value. The saturated value of top Be shims with 10.95 cm thickness (at EOL) is 18.6 mk [3] which can affect the axial flux shape.

The MNSR core is cooled through natural circulation of water that leads to the removal of fission heat. The cooling water is sucked up through the inlet orifice between the annular Be and the bottom Be and flows out of the core through the outlet orifice between the top Be tray and the annular Be. The longitudinal section of MNSR reactor is illustrated in Fig. 2 and the general characteristics of Isfahan MNSR are presented in Table 1.

As previously discussed, filling all inner and outer irradiation air ducts and all regulating tubes with water is a feature of BDBA scenario. The irradiation sites (IISs and OISs) are composed of two coaxial cylinders provided in a typical MNSR. These cylinders are joined together at the bottom of the site in a manner that the air (or water) can flow in both of them simultaneously. The inner cylinder is the irradiation tube and the outer one is the air duct. For a certain height, the volume of the air duct is more than the irradiation tube.

3. MCNPX model and its validation

The MCNPX code is used for the development of a 3D model of Isfahan MNSR for neutronic calculations. All calculations are carried out with the KCODE card which is inserted in the input file with 20 inactive cycles out of the total 160 cycles and one million histories per cycle. The special S(α, β) neutron slowing down feature of the MCNPX code leads to the treatment of thermal scattering for hydrogen in light water and Be. Table 1 presents the data used in developing the MCNPX model of the MNSR reactor. For all performed Monte Carlo calculations, the statistical error is below 0.9% (≈0.01). Furthermore, the neutronic calculations are done at the initial temperature of 15.8 °C SAR states that the reactor has the maximum excess reactivity. Fig. 3 illustrates the MCNPX model of the MNSR reactor at the EOL.

In the MCNPX model the burn card is applied to study the variation of reactor neutronic parameters during operating life and to obtain the new fuel composition at EOL. To find the EOL fuel composition, the top Be shims are added one-by-one in MCNPX model. In each step when the cold excess reactivity decreases to 2.8–3 mk, another top Be plate is added and depleted fuel is used as new fuel composition for next depletion calculation.

The validation of the MNSR MCNPX model is performed using the available experimental and SAR data for the fresh and burned fuels. Fig. 4 shows the variation in the reactivity worth of top Be reflector versus its thickness at BOL. As it can be seen, there is a good agreement between the MCNPX model results and experimental and SAR data. Also, Table 2 compares the results of MCNPX model to experimental and/or SAR data at BOL.

To validate the MNSR MCNPX model for burned fuel, the experimental data obtained from Isfahan MNSR after the first

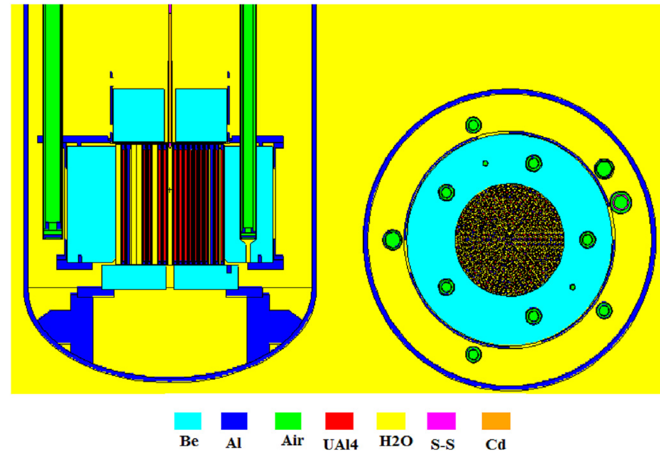

Fig. 3. The MCNPX model of Isfahan MNSR at the EOL.

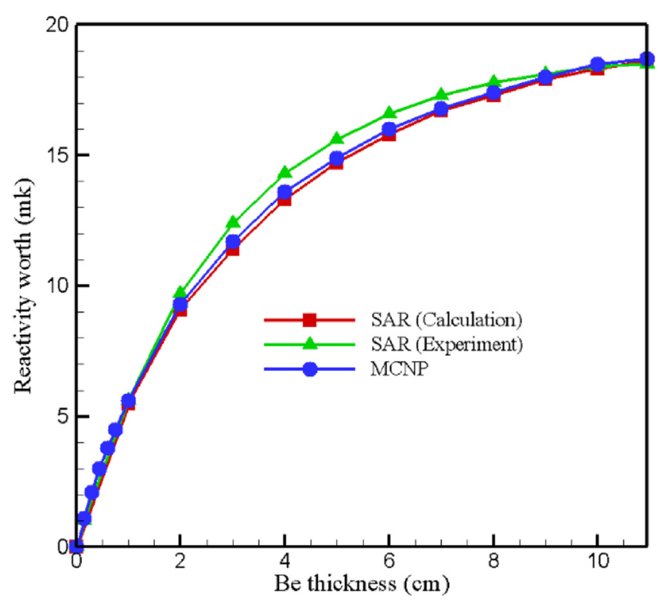

Fig. 4. Calculated reactivity worth of the top Be reflector using the MCNPX model and its comparison with SAR data.

Table 1
Brief description of Isfahan MNSR characteristics [20].

Parameters	Description
Reactor (type/shape/thermal power)	Tank-in-pool/cylindrical/30 kW
Materials (fuel/clad/moderator/reflector)	UAl4 dispersed in Al/Al/light H_2O/Be
Diameter of fuel (meat/clad)	4.3/0.6 (mm)
Fuel (enrichment/density/total U-235 Weight)	90.2 (%)/3.456 (g cm^{-3})/998 (g)
Number of (fuel rod/dummy rod/tie rod)	343/7/4
Number of (thermocouple/fission chamber)	2/2
Number of small IIS/small OIS	5/3
Number of large OIS	2
Number of RR	1–4
Thickness of reflector (annular/bottom)	100/51(mm)
Thickness of orifice (inlet/outlet)	6/7.5 (mm)
Control rod (material/diameter/S-S clad Thickness) Cooling system	Cadmium/3.9/0.5 (mm) Natural circulation

Table 2
Comparison of the results of MCNPX model with the experimental and/or SAR data at BOL.

Parameter	MCNPX	SAR	EXP	Diff. (max. in %)
Control rod reactivity worth (mk)	7.01	6.80	-	3.08
Shut down margin (mk)	3.35	> 2.0	-	-
Pool water reactivity worth (mk)	0.54	-	0.47	14. 89
Regulating rod reactivity worth (mk)	0.47	0.40	0.48	17.50
Prompt neutron life time (s)	6.69E-5	8.12E-5	-	17.61
Effective delayed neutron fraction (mk)	7.80	8.08	-	3.46

adding top Be shims in 2006 [21] are compared with the MCNPX model results. In first adding top Be shims, Be plate with 1.5 mm thickness was added to the top of the reactor to compensate the lost excess reactivity [21]. The effect of adding Be plate with 1.5 mm thickness to the top of the reactor core on k_{eff} is assessed and the results are compared with experimental data in a cold state (15.8 °C) (see Table 3). As shown the MCNPX model results are in line with the experimental data obtained after and before the first adding top Be shim [21].

4. Results

Since the MNSR reactor is capable of using 1 to 4 reactivity regulating rods, the BDBA scenario is investigated for MNSR reactors with the least and highest number of reactivity RRs. The amount of excess reactivity resulting for the BDBA scenario is compared in two different times at the BOL and at the EOL. In this section, a comparison is made between parameters such as the effective delayed neutron fraction, the reactivity worth of the control rod, and the thermal neutron flux in the radial and axial direction at the BOL and EOL.

4.1. Calculation of the induced reactivity

4.1.1. Case I

In case I, the MNSR core with one RR configuration was analyzed at EOL. The results obtained from the calculation of the induced reactivity at BOL in the BDBA scenario were compared with those obtained at the EOL for the same scenario in Table 4. In the first row of this table, the worth of the control rod has been compared. The value of the control rod in these two modes, given the uncertainty caused by the calculations (0.9%), does not show any significant changes. The comparison of the results in rows 2 to 7 indicates an increase in the reactivity worth of penetrated water in the canals at the EOL relative to the core with fresh fuel. These significant changes, as will be described in Section 4–3 (i.e. the spatial distribution of neutron flux), are due to an increase in the neutron flux in the axial direction after the addition of beryllium to the top of core. By adding the beryllium to the top of the core, the moderating ratio will increase. The increase in flux consequently leads to an increase in the reactivity worth of water entering the tubes. The data given at 9th row indicates that the reactivity caused by filling of all irradiation sites and reactivity regulating tubes has increased by 13.37 %((EOL-BOL)/BOL %) at the EOL compared to the BOL. The excess reactivity of the core in this reactor must always be between 3.5 and 4 mk [3]. Assume that the accident occurs when the reactor is operating with the maximum allowable reactivity value (4 mk), therefor, this reactivity should be added to the ninth row of Table 4 to obtain the total induced reactivity in the BDBA scenario (tenth row).

Table 3
Comparison of the experimental data with MCNPX results at the first adding top Be shim.

	Cold excess reactivity (mk)	
	Isfahan MNSR experiment [21]	MCNPX model
Before adding Be	2.93± 0.12	2.88± 0.01
After adding 1.5 mm Be	3.80±0.16	3.99± 0.01

Table 4
Comparison of the reactivity worth at BOL and EOL for one-RR MNSR.

No	Parameter	Reactivity worth		
		BOL	EOL	Diff (%)
1	Control Rod	7.01	6.94	-0.99
2	Filling 5 Inner Irradiation Tubes with water	1.80	1.98	+10.00
3	Filling 5 Inner Irradiation Air ducts with water	2.04	2.19	+7.35
4	Filling 5 IISs (Tubes + Air ducts) with water	2.39	3.01	+25.94
5	Filling 5 Outer Irradiation Tubes with water	0.68	0.71	+4.41
6	Filling 5 Outer Irradiation Air ducts with water	0.36	0.38	+5.55
7	Filling 5 OISs(Tubes + Air ducts) with water	0.88	1.03	+17.04
8	Filling tube around RR with water	0.36	0.34	-5.55
9	Filling all IIS, OIS and RR with water	3.44	3.90	+13.37
10	Total induced reactivity in the BDBA	7.44	7.90	+6.18

4.1.2. Case II

In case II, the MNSR core with four RRs configuration was analyzed at EOL. Maximum number of designed RRs is four in a typical MNSR core. The difference between the MNSR reactor with 4 reactivity regulating rod (case II) and one reactivity regulating rod (case I) is in the number of fuel rods. Using 345 fuel rods and 4 reactivity regulating rods, the cold state reactivity of the core was calculated to be 4.01mk. Other parameters of this reactor are same as the parameters of the Isfahan MNSR core with 343 fuel rods and one reactivity regulating rod (see Figs. 1 and 2 and Table 1).

To calculate the maximum induced reactivity in the proposed reactor core, the MNSR was simulated with four RRs at the EOL. The results obtained from the calculation of the induced reactivity at BOL in the BDBA scenario were compared with those obtained at the EOL for the same scenario in Table 5.

The worth reactivity changes in a MNSR with four reactivity RRs at the end of the core's life relative to fresh fuel are similar to the changes in the MNSR with one RR and the reactivity of the water

Table 5
Comparison of the reactivity worth at BOL and EOL for the four-RR MNSR.

No	Parameter	Reactivity worth		
		BOL	EOL	Diff (%)
1	Control Rod	6.96	6.99	+0.43
2	Filling 5 Inner Irradiation Tubes with water	1.78	1.93	+8.42
3	Filling 5 Inner Irradiation Air ducts with water	1.99	2.21	+11.05
4	Filling 5 IISs (Tubes + Air ducts) with water	2.41	2.93	+21.57
5	Filling 5 Outer Irradiation Tubes with water	0.65	0.70	+7.69
6	Filling 5 Outer Irradiation Air ducts with water	0.36	0.37	+2.77
7	Filling 5 OISs(Tubes + Air ducts) with water	0.85	0.99	+16.47
8	Filling 4 tubes around RR with water	1.11	1.05	-5.40
9	Filling all IIS, OIS and RR with water	3.75	4.33	+15.46
10	Total induced reactivity in the BDBA	7.75	8.33	+7.48

entering all irradiation sites at EOL is increased compared to BOL. This due to the higher moderating ratio of beryllium compared to water. Ninth row data indicates that the reactivity resulting from the filling of all irradiation sites and reactivity control tubes increased by 15.46% at the EOL compared to the BOL. As mentioned the excess reactivity of the core in this reactor should always be between 3.5 and 4 mk, assuming that the BDBA accident occurs when the reactor is in operation with the maximum allowable reactivity (4 mk), this amount of reactivity should be added to the ninth row of Table 5 to obtain the total of the induced reactivity in the BDBA scenario (tenth row).

4.2. Effective delayed neutron fraction (β_{eff})

Delayed neutron fraction is calculated for both fresh and burned fuels with application of TOTNU card in MCNPX input file [22]. The results show that the effective delayed neutron fraction β_{eff} and the delayed neutron fraction β are increased at the EOL (see Table 6). The findings also reveal that the increase in β_{eff} is more than that of β at the EOL. The difference can be attributed to various fuel compositions and the increase in the core height.

4.3. The spatial distribution of the neutron flux

After adding the top Be reflector, more neutrons are reflected on the reactor core because of the greater moderating ratio of Be (($\xi\Sigma_s/\Sigma_a$) = 143) in comparison with that of the light water (($\xi\Sigma_s/\Sigma_a$) = 71) [23]. Moreover, due to (n, 2n) and (γ, n) reactions in Be, the thermal neutron flux increases in the upper part of the reactor core. According to the mentioned reasons, the induced reactivity rises after the insertion of all top Be at the EOL. Fig. 5 compares the axial distribution of the thermal neutron flux in the IIS before and after the addition of top Be reflector (i.e. at BOL with EOL).

Addition of the top Be does not influence the spatial distribution of the radial neutron flux. Therefore, the fluxes are coincident in the core, in and out of the annular Be at the BOL and EOL (see Fig. 6).

5. Conclusion

The SAR provided on the MNSR neglected the changes that might occur in the core geometry and the effect it may have on the core physics parameters in the BDBA scenario at EOL. Therefore, in this study, the induced reactivity was investigated for a BDBA scenario at the EOL of MNSR to assess the safety of the reactor. In the SAR of the MNSR, the BDBA scenario includes sticking of the control rod and filling of the irradiation sites with water at the BOL. In this case, a maximum reactivity of 6.80 mk is released in the core.

Through the lifespan of the reactor's core, the geometry of the core changes gradually by inserting the beryllium on the top of the core. The results of this paper indicated an increase in the thermal neutron flux magnitude in the upper parts of the core after the insertion of the top beryllium (Fig. 5). The increase in the flux in the upper sections of the irradiation sites increased the reactivity worth of water penetrating in the irradiation sites. The results showed that the BDBA scenario at the EOL of Isfahan MNSR prompted 7.90 mk of reactivity into the core which presented the growth of 6.18% compared to the same scenario at BOL which was analyzed in SAR.

Table 6
Comparison of the β and β_{eff} at BOL and EOL of MNSR.

β_{eff}(mk)		β(mk)	
BOL	EOL	BOL	EOL
7.80	8.36	6.31	6.71

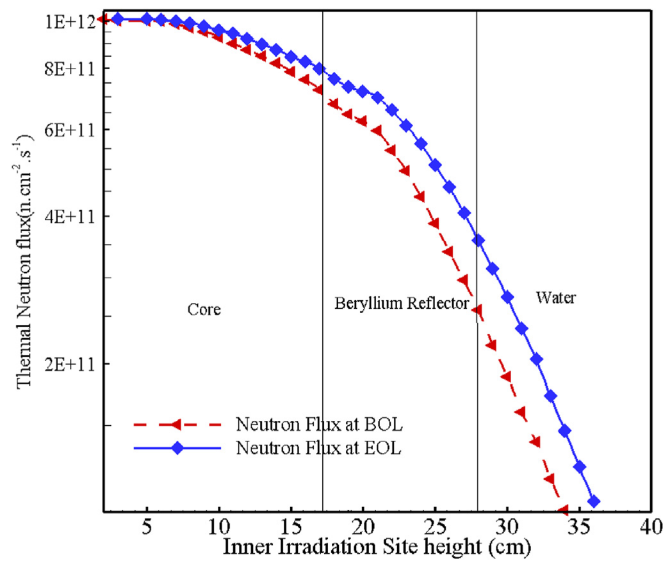

Fig. 5. Comparison of the axial neutron flux in IIS at BOL with the EOL.

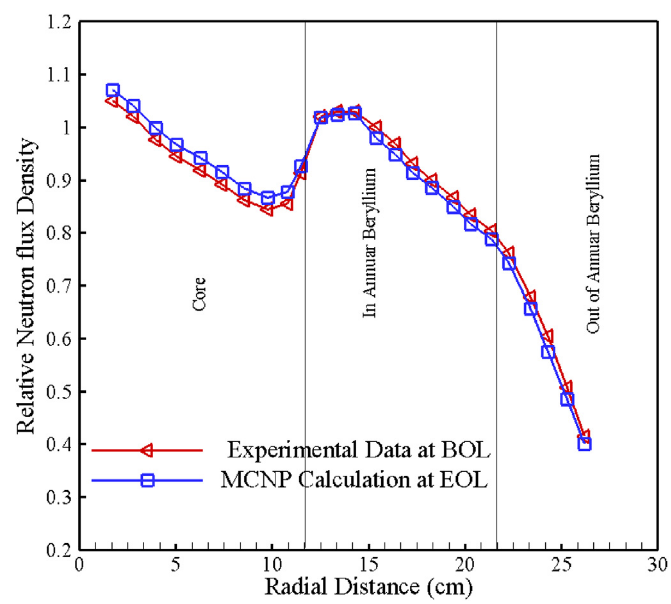

Fig. 6. Comparison of the radial thermal neutron flux in MNSR center plane (z = 0).

The calculation of changes in the delayed neutrons fraction in the two modes of fresh and burned fuel was another result of this paper. The results illustrated that, at the EOL, the effective delayed neutrons fraction is increased by 7.17%. This means that if the BDBA scenario occurs at the EOL, a reactivity of 0.945 $ is induced into the Isfahan MNSR core.

For the MNSR core configuration with four RRs, the same conclusion can be made. The results revealed that the BDBA scenario at the EOL for four-RR MNSR induced 8.33mk of reactivity into the core which presented the growth of 7.48% compared to the same scenario at BOL. Also, the BDBA scenario at the EOL induces reactivity of 0.996 $ into the four-RR MNSR core.

The results of the current study can be used for the calculation of thermohydraulic response of the core in the BDBA scenario which should be examined in a thermohydraulic-neutronic coupling calculation framework. This subject is under further investigation by the authors using Fluent software and applying user-defined functions (UDF) for coupling purposes to investigate whether or not the inherent safety of the reactor will be guaranteed and how the reactor responds to this amount of reactivity. The comprehensive results will be published in future works.

Acknowledgment

We would like to use this opportunity to acknowledge those who helped us conduct this work, especially the personnel and management of Isfahan MNSR department.

References

[1] Code of Conduct on the Safety and Security of Radioactive Sources, IAEA/CODEOC/2004, IAEA, Vienna, 2004.
[2] IAEA Safety of Research Reactors, Specific Safety Requirements SSR-3, 2016. Vienna.
[3] G. Chengzhan, Z. Xianfa, The Iranian MNSR Safety Analysis Report (SAR), China Institute of Atomic Energy, 1992 internal report).
[4] J. Mokhtari, F. Faghihi, J. Khorsandi, Design and optimization of the new LEU MNSR for neutron radiography using thermal column to upgrade thermal flux, Progress Nuclear Energy 100 (2017) 221–232.
[5] J. Mokhtari, F. Faghihi, J. Khorsandi, K. Hadad, Conceptual design study of the low power and LEU medical reactor for BNCT using in-tank fission converter to increase epithermal flux, Progress in Nuclear Energy 95 (2017) 70–77.
[6] M.A. Hosseini, M. Ahmadi, Miniature Neutron Source Reactors in medical research: achievements and challenges, Journal of Radioanalyt. Nucl. Chem. 314 (3) (2017) 1497–1504.
[7] A. Golabian, M.A. Hosseini, M. Ahmadi, B. Soleimani, M. Rezvanifard, The feasibility study of 177Lu production in Miniature Neutron Source Reactors using a multi-stage approach in Isfahan, Iran, Appl. Rad. Isotopes 131 (2018) 62–66.
[8] I. Khamis, K. Khattab, Neutronics-design modification of the Syrian miniature neutron source reactor, Progress Nuclear Energy 36 (2) (2000) 91–96.
[9] S. Waqar, S.M. Mirza, N.M. Mirza, T. Asad, A comparative neutronic study of the standard HEU core and various potential LEU alternatives for a typical MNSR system, Nuclear Eng. Design 238 (9) (2008) 2302–2307.
[10] K. Khattab, Measurement of the fast neutron flux in the MNSR inner irradiation site, Applied Rad. Isotopes 65 (1) (2007) 46–49.
[11] S.A. Agbo, Y.A. Ahmed, I.O.B. Ewa, Y. Jibrin, Analysis of Nigeria research reactor-1 thermal power calibration methods, Nuclear Engineering and Technology 48 (3) (2016) 673–683.
[12] M. Albarhoum, Reactivity cost for different top reflector materials in miniature neutron source reactors, Progress in Nuclear Energy 58 (2012) 39–44.
[13] M. Iqbal, M. Abdullah, S. Pervez, Parametric tests and measurements after shimming of a Beryllium reflector in a miniature neutron source reactor (MNSR), Annals Nuclear Energy 29 (13) (2002) 1609–1624.
[14] K. Khattab, I. Sulieman, Monte Carlo simulation of core physics parameters of the Syrian MNSR reactor, Annals of Nuclear Energy 38 (5) (2011) 1211–1213.
[15] K. Khattab, I. Sulieman, Calculations of the thermal and fast neutron fluxes in the Syrian miniature neutron source reactor using the MCNP-4C code, Applied Radiation and Isotopes 67 (4) (2009) 535–538.
[16] A. Nawaz, S.M. Mirza, N.M. Mirza, M. Sohail, Analysis of core life-time and neutronic parameters for HEU and potential LEU/MEU fuels in a typical MNSR, Annals of Nuclear Energy 47 (2012) 46–52.
[17] S.A.H. Feghhi, S. Jafarikia, F. Abtin, Miniature neutron source reactor burn up calculations using IRBURN code system, Annals of Nuclear Energy 47 (2012) 242–248.
[18] G. Chengzhan, MNSR Accident (Event) Analysis, China Institute of Atomic Energy, 1992 (internal report).
[19] G. Ke, Z. Sun, F. Shen, T. Liu, Y. Li, Y. Zhou, The study of physics and thermal characteristics for in-hospital neutron irradiator (IHNI), Applied Radiation and Isotopes 67 (7) (2009) S234–S237.
[20] G. Jilin, General Description of Miniature Neutron Source Reactor, China Institute of Atomic Energy, 1992 internal report).
[21] J. Ebadati, I. Shahabi, M. Rezvanifard, January). Calculation and experiment of adding top beryllium shims for Iran MNSR, in: 14th International Conference on Nuclear Engineering, American Society of Mechanical Engineers, 2006, pp. 311–314.
[22] Monte Carlo N-particle Transport Code System Manual, Los Alamos National Laboratory, New Mexico, April 2000.
[23] James J. Duderstadt, Louis J. Hamilton, Nuclear Reactor Analysis, John Wiley&: Sons, Inc, New York/London, 1976.