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Research Article

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Determination of the neutron and gamma dose distribution due to the operation of vertical neutron beam lines

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HIGHLIGHTS

- Dose rates at different zones of reactor have been determined for different modes of neutron beamlines.
- The performance of the neutron radiography beam tube radiation protection system has been investigated.
- The average difference between the simulation results and experimental results was below 5%.
- More than 80% of cumulative dose rate was due to the gamma rays.

ABSTRACT

The Miniature Neutron Source Reactors (MNSRs) have high inherent safety, which makes them a suitable option for installation in urban areas such as hospitals and universities. In order to increase the radiation safety of this reactor there is no external beam tubes and only 10 irradiation sites near the reactor core are accessible through a pneumatic system. These irradiation sites are only suitable for irradiation of small samples and to develop reactor applications and irradiation of large samples, two external beam tubes were added outside the vessel of the Isfahan MNSR. These beam tubes transport neutrons from near the core to outside of the biological shield. In this study, the effect of these beam tubes on increasing the neutron and gamma dose rates in different zones of the reactor building was investigated and the dose distribution was determined through calculation and measurements. This study was conducted to investigate the radiation protection conditions at different operation conditions of these beam tubes and to ensure the radiation safety of the reactor operators and researchers. The results showed that when both beam tubes are operated simultaneously, the average total dose rate in the reactor hall, pneumatic room and corridors increases to 4.58 μSv.h^{-1}, 1.7 μSv.h^{-1} and 4.9 μSv.h^{-1}, respectively at the maximum power of reactor, i.e. 30 kW. The major part of the dose rate of neutrons and gamma rays distributed in the environment is caused by the neutron radiography channel, and more than 80% of the dose rate is related to gamma rays. On the other hand, the performance of the neutron radiography beam tube radiation protection system showed that even when the reactor is at its maximum power and these beam tubes are inactive, the total dose at the edges of the reactor pool is about 2% of the annual dose limit. This indicates that the radiation protection of the beam tube has a good performance in preventing the increase in the dose rate when the beam tube is deactivated.

KEYWORDS
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Dose rate
Beam tube
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1 Introduction

The operation of research reactors can be mainly considered for educational purposes as well as irradiation for various purposes. Irradiation in reactors is carried out either by irradiation in the reactor core or outside it. In-core irradiations are mainly performed for purposes such as neutron activation analysis (NAA), silicon doping, radioisotope production, and gemstone coloring. For some experiments and applications such as neutron radiography (NR), neutron diffraction, and boron neutron capture therapy (BNCT), and prompt gamma neutron activation analysis (PGNAA) irradiations are performed out-of-core via neutron beamlines (International Atomic Energy Agency, 2008; De Beer, 2015; Casali et al., 1995; Aswal et al., 2022; Jafari et al., 2025b). Neutron beamlines are usually equip-

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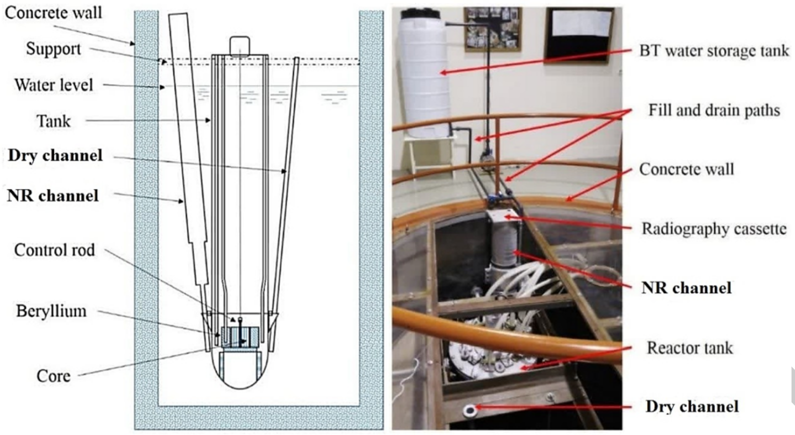

Figure 1: Schematic (Left) and of the Isfahan MNSR reactor and its two neutron channels (Mokhtari and Dastjerdi, 2023).

ment that transfer the neutron and gamma ray flux produced in the core to outside the biological shield of the reactor. The transfer of neutrons and gamma rays outside the biological shield of the reactor can increase the radiation dose in the reactor building, and therefore the design of the radiation facilities based on the beamlines must comply with the principles of radiation protection, and the radiation dose caused by their operation should be investigated (Jafari et al., 2025b; Svab et al., 2004; Dastjerdi et al., 2016; Laurent-Pettersson et al., 1992; Fujine et al., 1996).

Currently, there are 227 operational research reactors in the world, and nearly 120 of them produce neutron fluxes greater than 10^{12} n.cm^{-2}.s^{-1}. Most of these reactors are equipped with instruments and beam tubes to perform irradiation experiments inside and outside the reactor core. The miniature neutron source reactors (MNSR) are designed and built by closely modeling the SLOWPOKE reactors and are very similar in design and construction. There are 21 of this type of reactors in the world and 11 of them are operational, accounting for a total of nearly 10% of the world's operational reactors with a neutron flux greater than 10^{12} n.cm^{-2}.s^{-1} (https://nucleus.iaea.org/rrdb/home). These reactors did not have beam tubes for out of core irradiation experiments in their original design, and only the Royal Military College of Canada's SLOWPOKE reactor has an external beam tube installed for neutron radiography, and external beam tubes have been added to one of the MNSRs, i.e. the IHNI reactor. External beam tubes have also been added to the Isfahan MNSR between 2019 and 2023 as novel developments. The Isfahan MNSR is the only of these types of reactor in which two vertical neutron beamlines have been implemented (IAEA, 2025). These beam tubes include one small-sized beamline, i.e. dry channel (DC), and another large-sized beamline, i.e. neutron radiography (NR) beam tube, that transport neutrons from near the core to outside of the biological shield. The dry channel is mainly used for the possibility of conducting experiments such as neutron radiography, prompt gamma neutron activation analysis, and neutron attenuation coefficient measurements. The neutron radiography beamline is also used to perform experiments such as neutron radiography and the characterization and calibration of neutron detectors (Rahmati et al., 2023; Dastjerdi et al., 2019, 2023; Mokhtari et al., 2023; Bagherzadeh et al., 2023b,a; Asgari et al., 2023; Mokhtari and Dastjerdi, 2023). In this study, the effect of these beam tubes on increasing the neutron and gamma dose rates in the reactor building was investigated and the dose distribution was determined through calculation and measurements.

2 Methods and Materials

2.1 *Isfahan MNSR Reactor*

The MNSR is an optimized model of the Canadian SLOWPOKE reactor. The main advantage of these research reactors is their inherent safety. Due to this feature, this reactor is suitable for installation in urban areas including universities, research institutes and hospitals (Mokhtari et al., 2020). In this reactor, the core is located inside an aluminum tank. The reactor core is cooled by natural water convection and has an inherent safety due to the high negative feedback of the moderator temperature on the reactivity. The maximum power of Isfahan MNSR is 30 kW. Due to the high negative feedback of the moderator temperature, the reactor can only operate at maximum power for about 2.5 hours per day per week. The core of this reactor is a cylinder with the same diameter and height of 23 cm and contains 343 fuel pins consist of UAl4 with 90.2% of uranium 235 enrichment. The core is surrounded by an annular beryllium with 10 cm thickness. This beryllium optimizes fuel burnup by reflecting neutrons into the core and neutron production by $(n, 2n)$ and (g, n) reactions. In the original design of the MNSR reactor, this reactor has 10 irradiation sites near the core. Recently, two neutron beamlines were also added to this reactor (Rahmati et al., 2023; Dastjerdi et al., 2019, 2023; Mokhtari et al., 2023; Bagherzadeh et al., 2023b,a; Asgari et al., 2023; Mokhtari and Dastjerdi, 2023; Mokhtari et al., 2020; Asgari et al., 2024, 2025; Moslehi et al., 2022; Jafari et al., 2024a; Vatani et al., 2023; Ghasemi et al., 2023; Jafari et al., 2025a, 2024b; Gholamzadeh et al., 2024). The image of the Isfahan MNSR reactor is given in Fig. 1.

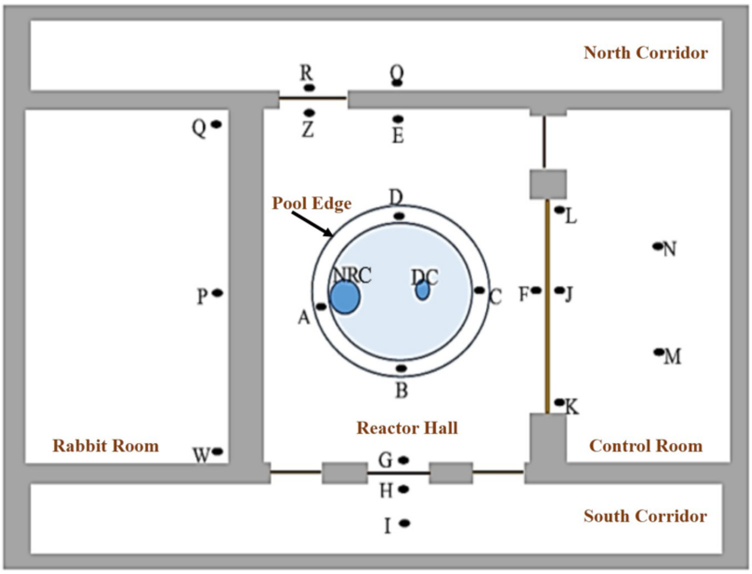

Figure 2: Plan of the Isfahan MNSR reactor building and the location of selected points for dosimetry.

2.2 Neutron radiography channel (Large size beam tube)

The NRC is a telescopic shape aluminum tube. It consists of three cylinders with 5.2, 18.8, and 24 cm diameters, and 65, 105, and 450 cm heights. The telescopic shape of the channel is designed to increase the diameter of the neutron beam at the outlet to makes it possible to perform NR of large samples. This channel is installed outside the reactor tank. For better access to the outlet, the channel is installed tangentially. The maximum thermal neutrons flux at the outlet of this channel is 2.96×10^5 n.cm^{-2}.s^{-1}. To increase radiation protection safety, a water injection and drainage system has been installed. To activate the channel, water is drained from the channel, and to deactivate it, the channel is filled with water. By draining and injecting water into the beam tube, the neutron beam is turned ON and OFF, respectively. Of course, when the reactor is ON and the beam tube is drained, the beam is considered ON, and the water draining and injection system acts like a beam shutter in a neutron radiography system. Figure 1 shows a schematic of NRC and DC positions (Mokhtari and Dastjerdi, 2023).

2.3 Isfahan MNSR Reactor Building

The plan of the Isfahan MNSR reactor building is shown in Fig. 2. In this figure, the location of the reactor hall, corridors, rabbit room, and control room can be seen.

20 points were selected for dosimetry in the reactor room and adjacent rooms (Fig. 2. For example, the seating positions of the reactor operators (M, N points) and control room window ledge to reactor hall (L, J, K points) are among these selected points. The distance between M and N points from the ground is 100 cm (position of reactor operator personnel), the distance between L, J, and K points from the ground is 73 cm (height of window ledge), and the distance between other points from the ground was selected to 10 cm (the height position of dosimeter instruments) in order to compare with measurements.

2.4 Dosimetry method

The neutron and gamma dose rates were determined at the selected points through calculations and measurements. Dose rate determination was performed in 4 different cases:

- First case: both channels were inactive (NRC and DC are OFF),
- Second case: only the DC was active (NRC is OFF and DC is ON),
- Third case: only the NRC was active (NRC is ON and DC is OFF),
- Fourth case: both channels were active (NRC and DC are ON).

In each case, the reactor was at its maximum power (30 kW). The calculations were performed through simulation with a validated Monte Carlo code (MCNPX) and ENDF libraries (Mokhtari et al., 2025). In defining the geometry of this code, all reactor components, including the core, tank, pool, beam tubes, as well as the corridors, hall, control, and rabbit rooms were precisely defined and the transport of neutrons and gamma rays from production in the reactor core to the desired points was simulated. The particle transport history was considered in such a way that the calculation results were obtained with the least errors, i.e. below 5%, using Russian roulette variance reduction method (Mokhtari et al., 2025). The Kcode with 180 active cycle by 6×10^5 particle per cycle was used as radiation source, the F5 and DF5 tallies were used for neutron and gamma dose calculation using ICRP flux-to-dose conversion coefficient. As calculations were performed per particle, a tally scaling factor of 2.38×10^{15} n.s^{-1} was used to convert values for 30 kW power of reactor. A schematic

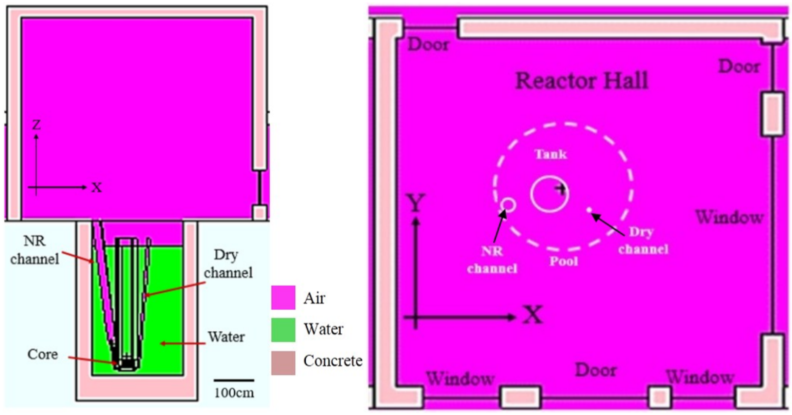

Figure 3: Schematics of the simulated model of the reactor, DC, and NRC (Gholamzadeh et al., 2024).

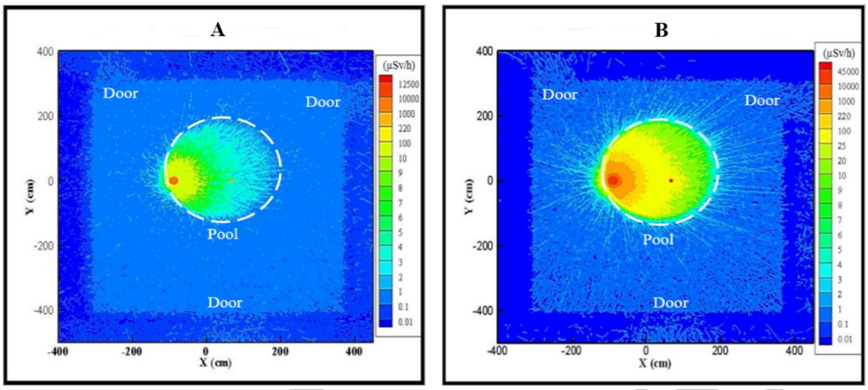

Figure 4: Distribution of (A) neutron dose rate and (B) gamma dose rate in the MNSR reactor hall for fourth case (Mokhtari et al., 2025).

of the simulation and dose rate distribution are shown in Figs. 3 and 4, respectively (Mokhtari et al., 2025).

NE Monitor NM2 dosimeter and TELETECTOR 6112 gamma dosimeter were used to measure the neutron and gamma dose rates, respectively (Tanner et al., 2006; Prince, 2012). The NE Monitor NM2 dosimeter is a neutron dosimeter consists of a BF3 neutron detector inside a cylindrical polyethylene as neutron monitor and responses a wide range of neutron energies from thermal (0.1 eV) to fast (10 MeV) neutrons which measure and display dose equivalent rate in μSv.h^{-1} to Sv.h^{-1} in a digital manner. The TELETECTOR 6112 is a gamma dosimeter consists of a Geiger-Muller detector that is able to measure and display gamma dose rates from several μSv.h^{-1} to Sv.h^{-1}. To measure the dose rate, dosimeters were placed at each mentioned points in section 2.3. (Fig. 2) for 5 minutes. Each measurement was repeated 10 times and the standard deviation of the results was considered as the measurement accuracy (below 10%).

3 Results and Discussion

The results of dose rate measurements in 160 different situations are presented in Tables 1 to 4. In each situation, the difference between the calculated and measured results is shown.

The data given in Tables 1 to 4 show that, the average difference between the simulation results and experimental results was below 5%. This means that the reactor and its building are simulated accurately.

The data given in Table 1 show that, the biological shields of DC and NRC are well designed. Point A is located to the right of the NRC outlet, and the total neutron and gamma dose rates there is about 0.19 μSv.h^{-1}. That is, if the reactor is at its maximum power and a person stands at point A for 8 hours a day, 5 days a week, and 50 weeks a year, his received dose will be about 0.38 mSv.h^{-1}, this dose is less than 2% of the annual dose limit.

The data given in Tables 1 to 2 show that, the majority of the dose rates in all parts of the reactor building is due to the NRC. Because, the outlet diameter of the NRC (25 cm) is 5 times bigger than the outlet diameter of the DC (5 cm), so it delivers more neutrons and gamma rays into the environment.

The data given in Table 4 show that, in 30 kW power and when both channels are active. In all parts of the reactor building, the majority of the dose rates is due to gamma rays. This could be due to the radioactive absorption reaction, i.e. (n, γ) reaction, of neutrons in the structural materials of the reactor as well as the reactor hall. For example, in the reactor operator seating area (M and N points), the average dose rate is about 4.58 μSv.h^{-1}, and more than 80% of it is due to gamma rays. The reactor room doors attenuate about 25% of the total

Table 1: Neutron and gamma dose rates when both channels are inactive, i.e. NRC and DC are OFF (first case). All dose rates are in μSv.h^{-1}.

Location of points	Neutron dose rate		Gamma dose rate	
	Experiment	Simulation	Experiment	Simulation
A	0.05 ± 7%	0.05 ± 5%	0.14 ± 7%	0.15 ± 5%
B	0.04 ± 7%	0.04 ± 5%	0.11 ± 7%	0.12 ± 5%
C	0.04 ± 7%	0.04 ± 5%	0.13 ± 7%	0.14 ± 5%
D	0.04 ± 7%	0.04 ± 5%	0.11 ± 7%	0.11 ± 5%
E	0.02 ± 7%	0.02 ± 5%	0.04 ± 7%	0.04 ± 5%
F	0.02 ± 7%	0.02 ± 5%	0.04 ± 8%	0.04 ± 5%
G	0.01 ± 8%	0.01 ± 5%	0.04 ± 7%	0.04 ± 5%
H	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%
I	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%
J	0.01 ± 8%	0.01 ± 5%	0.02 ± 6%	0.02 ± 5%
K	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%
L	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%
M	0.01 ± 8%	0.01 ± 5%	0.02 ± 5%	0.02 ± 5%
N	0.00 ± 8%	0.00 ± 5%	0.02 ± 5%	0.02 ± 5%
O	0.01 ± 7%	0.01 ± 5%	0.02 ± 5%	0.02 ± 5%
P	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%
Q	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%
W	0.01 ± 8%	0.01 ± 5%	0.02 ± 8%	0.02 ± 5%
Z	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%
R	0.01 ± 8%	0.01 ± 5%	0.02 ± 7%	0.02 ± 5%

Table 2: Neutron and gamma dose rates when only the DC is active, i.e. NRC is OFF and DC is ON (second case). All dose rates are in μSv.h^{-1}.

Location of points	Neutron dose rate		Gamma dose rate	
	Experiment	Simulation	Experiment	Simulation
A	2.69 ± 7%	2.70 ± 4%	3.65 ± 7%	3.70 ± 3%
B	2.75 ± 6%	2.80 ± 4%	3.85 ± 6%	3.90 ± 3%
C	3.06 ± 7%	3.10 ± 4%	4.03 ± 7%	4.33 ± 3%
D	2.69 ± 7%	2.76 ± 4%	3.79 ± 7%	3.84 ± 3%
E	1.47 ± 7%	1.50 ± 5%	2.26 ± 7%	2.30 ± 4%
F	1.18 ± 8%	1.20 ± 5%	2.35 ± 6%	2.40 ± 5%
G	1.01 ± 6%	1.00 ± 5%	1.21 ± 7%	1.23 ± 5%
H	0.79 ± 7%	0.79 ± 5%	0.9 ± 7%	0.90 ± 5%
I	0.3 ± 8%	0.30 ± 5%	0.73 ± 7%	0.73 ± 5%
J	0.84 ± 7%	0.80 ± 5%	0.95 ± 8%	1.00 ± 5%
K	0.74 ± 7%	0.74 ± 5%	0.89 ± 7%	0.90 ± 5%
L	0.75 ± 7%	0.75 ± 5%	0.87 ± 7%	0.87 ± 5%
M	0.2 ± 8%	0.20 ± 5%	0.68 ± 7%	0.70 ± 5%
N	0.2 ± 8%	0.20 ± 5%	0.72 ± 7%	0.72 ± 5%
O	0.15 ± 7%	0.15 ± 5%	0.48 ± 7%	0.50 ± 5%
P	0.11 ± 7%	0.10 ± 5%	0.39 ± 7%	0.40 ± 5%
Q	0.1 ± 8%	0.10 ± 5%	0.39 ± 7%	0.40 ± 5%
W	0.1 ± 8%	0.10 ± 5%	0.4 ± 6%	0.40 ± 5%
Z	0.3 ± 7%	0.31 ± 5%	0.88 ± 6%	0.89 ± 5%
R	0.2 ± 8%	0.20 ± 5%	0.64 ± 7%	0.65 ± 5%

neutron and gamma dose rates.

The total dose rates in the seating area of the reactor control room (K, J, and L points), in the rabbit room (Q, P, and W points), and behind the reactor room doors (H and R points) reaching 6.2 μSv.h^{-1}, 1.7 μSv.h^{-1}, and 4.9 μSv.h^{-1}, respectively.

To provide better access to the outlet of the NRC, this channel is tilted 5 degrees towards its left wall. In the space on the left side of the reactor room, the dose rate is very intense. At point A, the total dose rate is about 0.29 mSv.h^{-1}, this dose is about 12 times the dose rates at points B, C and D (the other edges of the reactor pool).

According to the dose limits recommended by ICRP, the limit on dose from occupational exposure is 20 mSv per year, averaged over defined periods of 5 years, with no single year exceeding 50 mSv (IAEA, 2014). To determine the annual dose to personnel from the simultaneous operation of these two neutron beams, i.e. realistic operations, it is assumed that the MNSR reactor is operating ("ON") at maximum power for an average of 100 days per year, for 2 hours per day, and that both beamlines are active for half of this time, i.e. 200 hour.year^{-1} (fourth case).

Table 3: Neutron and gamma dose rates when only the NRC is active, i.e. NRC is ON and DC is OFF (third case). All dose rates are in μSv.h^{-1}.

Location of points	Neutron dose rate		Gamma dose rate	
	Experiment	Simulation	Experiment	Simulation
A	66.00 ± 6%	68.00 ± 3%	205.66 ± 8%	210.00 ± 3%
B	5.86 ± 6%	5.80 ± 4%	19.90 ± 7%	20.30 ± 4%
C	5.15 ± 8%	5.21 ± 4%	19.43 ± 7%	19.65 ± 4%
D	5.54 ± 7%	5.65 ± 4%	19.95 ± 7%	20.00 ± 3%
E	3.65 ± 7%	3.72 ± 4%	6.01 ± 7%	6.00 ± 4%
F	2.48 ± 8%	2.50 ± 4%	5.18 ± 8%	5.30 ± 4%
G	1.70 ± 7%	1.71 ± 4%	3.74 ± 7%	3.80 ± 4%
H	1.45 ± 7%	1.44 ± 5%	2.95 ± 7%	3.00 ± 4%
I	1.20 ± 8%	1.20 ± 4%	2.77 ± 7%	2.78 ± 4%
J	1.28 ± 7%	1.28 ± 5%	3.65 ± 7%	3.70 ± 4%
K	1.13 ± 8%	1.10 ± 5%	3.72 ± 7%	3.68 ± 4%
L	0.98 ± 7%	1.00 ± 5%	3.60 ± 8%	3.60 ± 4%
M	0.69 ± 8%	0.70 ± 5%	2.98 ± 7%	3.00 ± 5%
N	0.73 ± 7%	0.73 ± 5%	3.12 ± 8%	3.10 ± 5%
O	0.4 ± 7%	0.41 ± 5%	1.90 ± 7%	1.90 ± 5%
P	0.32 ± 8%	0.30 ± 5%	0.87 ± 8%	0.90 ± 5%
Q	0.29 ± 7%	0.30 ± 5%	0.84 ± 7%	0.85 ± 5%
W	0.30 ± 7%	0.30 ± 5%	0.72 ± 7%	0.74 ± 5%
Z	1.34 ± 7%	1.35 ± 4%	3.51 ± 7%	3.51 ± 4%
R	0.93 ± 8%	0.93 ± 4%	2.66 ± 7%	2.70 ± 5%

Table 4: Neutron and gamma dose rates when both channels are active, i.e. NRC and DC are ON (fourth case). All dose rates are in μSv.h^{-1}.

Location of points	Neutron dose rate		Gamma dose rate	
	Experiment	Simulation	Experiment	Simulation
A	69.70 ± 7%	70 ± 3%	220.72 ± 7%	220.00 ± 3%
B	6.90 ± 7%	7 ± 3%	23.83 ± 7%	24.00 ± 3%
C	6.82 ± 7%	7.16 ± 3%	23.02 ± 8%	23.00 ± 3%
D	7.07 ± 7%	7.12 ± 3%	22.70 ± 7%	23.00 ± 3%
E	4.50 ± 7%	4.50 ± 4%	7.05 ± 7%	7.00 ± 3%
F	3.35 ± 7%	3.30 ± 4%	6.11 ± 7%	6.00 ± 3%
G	2.04 ± 7%	2.00 ± 4%	5.01 ± 7%	5.00 ± 3%
H	1.77 ± 7%	1.78 ± 4%	3.44 ± 7%	3.50 ± 4%
I	1.46 ± 7%	1.50 ± 4%	3.18 ± 8%	3.24 ± 4%
J	1.39 ± 7%	1.41 ± 5%	5.19 ± 8%	5.10 ± 4%
K	1.34 ± 7%	1.36 ± 4%	4.82 ± 7%	4.80 ± 4%
L	1.19 ± 7%	1.22 ± 4%	4.66 ± 7%	4.86 ± 4%
M	0.78 ± 8%	0.80 ± 5%	3.75 ± 7%	3.80 ± 5%
N	0.82 ± 8%	0.82 ± 5%	3.80 ± 7%	3.82 ± 5%
O	0.48 ± 8%	0.50 ± 5%	2.33 ± 7%	2.56 ± 5%
P	0.39 ± 8%	0.40 ± 5%	1.48 ± 8%	1.50 ± 5%
Q	0.39 ± 8%	0.40 ± 5%	1.30 ± 7%	1.30 ± 5%
W	0.39 ± 9%	0.40 ± 5%	1.19 ± 7%	1.20 ± 5%
Z	1.50 ± 7%	1.50 ± 5%	4.43 ± 7%	4.50 ± 5%
R	1.14 ± 7%	1.16 ± 5%	3.43 ±7%	3.50 ± 5%

Table 5 shows the total neutron and gamma dose in terms of mSv.year^{-1} for first and fourth case at the locations of dosimetry for 200 hours.

According to Table 5, only at point A (near the beam tube) and when the beam tube and dry channel are both active (fourth case), the received dose is higher than the recommended dose limit. The important point in this case is that there is no need for full-time personnel presence (200 hours) during simultaneous operation of both beamlines, i.e. fourth case, at point A and this is a highly exaggerated estimate of personnel presence.

Of course, it should be noted that a state was considered where the reactor is operating at its maximum power (30 kW), and at its maximum flux (10^{12} n.cm^{-2}.s^{-1}), while the reactor is not always operating at its maximum power. Also, the staff of the MNSR are not present in the reactor hall at all when the reactor is in operation. This point is considering as an operation regulation during the operation of neutron beamlines in reactor hall. The only places with high neutron and gamma doses are exactly above the beam tube and above the dry channel, but the operator is never placed above them, and long clamps are even used to place the sample for neutron radiography.

Table 5: Total annual dose for first and fourth case at the locations of dosimetry.

Location of Points	Total annual dose (mSv.year^{-1})	
	First case	Fourth case
A	0.020	28.8
B	0.016	3.1
C	0.018	3.016
D	0.016	3.012
E	0.007	1.15
F	0.007	0.93
G	0.006	0.7
H	0.004	0.528
I	0.004	0.474
J	0.004	0.651
K	0.004	0.616
L	0.004	0.608
M	0.004	0.46
N	0.004	0.464
O	0.004	0.306
P	0.004	0.19
Q	0.004	0.17
W	0.004	0.16
Z	0.004	0.6
R	0.004	0.466

4 Conclusions

One of the important features of the MNSR reactor is its high safety. Any changes in the reactor structure that affect on its safety must be carefully examined. DC and NRC transmit large amounts of neutrons and gamma rays outside the reactor's biological shield. These neutron channels have increased the average cumulative dose rates in the reactor building. But on the other hand, these neutron channels have significantly increased the research capability of the MNSR reactor. In this study, the neutron and gamma dose distribution in different zones of the Isfahan MNSR due to the operation of neutron beam lines have been determined. For this purpose, neutron and gamma dose rates were obtained at 20 different points, and in 4 different situations: 1- both channels were active, 2- both channels were inactive, 3- only the NRC was active, and 4- only the DC was active. In this work, 160 dose rate measurements were performed. The average difference between the simulation results and experimental results was below 5%. The results showed that when the reactor is at its maximum power, i.e. 30 kW, and these channels are inactive, the dose at the edges of the reactor pool is about 2% of the annual dose limit. When the reactor is at maximum power and both channels are active, the majority of the environment cumulative dose rate of neutrons and gamma rays is due to the NRC. More than 80% of cumulative dose rate is due to gamma rays. By following radiation safety requirements, personnel can be protected from radiation damage. Some of these requirements are: when the reactor is operating at high power and the beamlines are active, no one should enter the reactor hall except in accidents; the doors of the reactor room must remain closed and staff should avoid standing behind the doors and windows of the reactor hall as much as possible.

Conflict of Interest

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

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