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Dose Distribution of Radioxenon Due to a Hypothetical Accident of TRIGA Research Reactor in Bangladesh

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ABSTRACT

Radiological dose distribution owing to the deposition of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, and ^{138}Xe on ground and immersion considering a postulated accident of TRIGA Mark-II research reactor has been assessed. The radiological dose distribution has been carried out in various directions with the help of Gaussian Diffusion Model. Local meteorological data such as average wind speed, frequency, etc. has been collected and evaluated for various directions around the reactor site. For all the dominant directions, the maximum dose values due to ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{138}Xe and the total (^{131m}Xe + ^{133m}Xe + ^{133}Xe + ^{135m}Xe + ^{135}Xe + ^{138}Xe) were observed within the limit 3.03E-7–1.23E-4 µSv/h, 1.01E-5–4.09E-3 µSv/h, 0.0003–0.14 µSv/h, 2.29E-5–9.26E-3 µSv/h, 0.002 – 1.111 µSv/h, 1.11E-5–4.55E-3 µSv/h, and 0.003–1.269 µSv/h, respectively. Dose distribution was found to be dominant due to immersion and the contribution was 87.55 %. There is shortage of data regarding the release of radioxenon in the atmosphere during nuclear accident especially in the case of TRIGA type research reactor. This paper is the first such detailed study on atmospheric release of radioxenon and its dose distribution for a full power- reactor and the consequences towards the environment and public health. The result can be applied to develop the radiological protective measures and to prepare an emergency response plan for the TRIGA reactor site.

INTRODUCTION

The research reactor of Bangladesh at Atomic Energy Research Establishment (AERE), Savar, Dhaka started its first operation on 14th September 1986. Since then the reactor has been operated at different power levels for various research and development activities, manpower training, and production of radioisotopes [1]. Bangladesh Atomic Energy Regulatory Authority (BAERA)'s regulations necessitate that the nuclear installations embark on all proper safety measures to protect the surrounding environment as well as the health and safety of the public and to identify, control, and monitor radioactive release to the local environment [2]. A nuclear reactor may cause an uncontrolled release of radioactivity to the environment and thus causes a significant radiological hazard to human beings or animals. Hence radiological dose assessment due to nuclear accident is important because nuclear and radiological safety consideration is important for nuclear reactor [3]. During accidental conditions, emergency response is the most important issue of the emergency management of the nuclear reactor [4]. In case of a radiological accident, studies of atmospheric dispersion and assessment of radiological dose are crucial for regulators to prepare emergency response plans [5-7]. Analysis of radiological safety for postulated accidents allows a dominant input for safety assessment of nuclear reactors considering the safety of occupational workers and the general public [8-19]. The TRIGA research reactor was licensed on the condition that there will be no significant hazard or undue radiation effect on human health and safety as the reactor becomes a source of intense ionizing radiation during its operations [20-23].

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In this study, an assessment has been performed of dose distribution of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, and ^{138}Xe caused by ground deposition and immersion due to a postulated accident of the TRIGA research reactor. In the accidental scenario, it was assumed to happen at the effective stack height, and the radionuclides released from the core was presumed to mix within the reactor bay region and released to the atmosphere from the containment building through the normal ventilation system at an effective stack height. Simulations of accident scenarios are important to assess the possible hazards in case of an accident at any nuclear facility. Rahman et al. 2003 [24] studied the dose distribution in various environmental media due to deposition of ^{131}I and ^{137}Cs in soil at the AERE campus. Sulaiman et al. 2019 [25] determined the core inventory for PUSPATI TRIGA Reactor (RTP) under a hypothetical severe accident by assuming the reactor to be operated continuously for 365 days at full power (1 MW). But no dose distribution was studied. Due to the shortage of data, the present study focused on the release of radioxenon in the atmosphere during nuclear accident of TRIGA reactor.

Because of the non-interacting properties of noble gas (radioxenon), the air concentration activity and dose due to immersion as a function of time and distance are very important data to assessing safety analysis, possible release and the licensing purposes. Moreover, this data is also important for Comprehensive Nuclear-Test-Ban Treaty Organization (CTBTO). Such data is not so readily available in paper that we have studied so far. In our present research work we determined the core inventory of radioxenon by assuming 15 days reactor operation at full power (3 MW) and dose distribution of radioxenon due to its atmospheric dispersion was also studied. Instantly after an accident occurs, the radiological impact may cause on the public in and around the reactor site due to the dose received owing to penetration of gamma rays in the body or by inhalation or irradiation of skin by the Beta radiation. In this study, the radiological dose calculations were performed using the Gaussian Plume Model (GPM) and only the release of radioactive Xenon was evaluated for which the release fraction was assumed to be 100 %. The raw weather data such as wind speed and frequency at various directions was collected from the Bangladesh Meteorological Department (BMD), Dhaka for the TRIGA reactor site and analyzed. Thereby, this work is the first such detailed study on the source term atmospheric release of radioxenon and its dose distribution for a full power-operated BTRR and the consequences towards the environment and public health. The obtained data from the study should be useful for ensuring radiological or environmental safety and for establishing the radiological safety measures in and around the reactor area.

METHODOLOGY

BAEC TRIGA research reactor

The Bangladesh Atomic Energy Commission (BAEC) TRIGA Research Reactor (BTRR) is a pool-type research reactor (cylindrical-shaped), cooled by light water. The fuel elements of the reactor is uranium-zirconium hydride arranged in a circular grid array. The composition of the fuel is 20 % (wt) ranium enriched to 19.7 % (the amount of ^{235}U isotope is 19.7 %), irconium hydride (ZrH1.6), and burnable poison rbium (^{167}Er). The core is located close to the base of the water-filled tank, which is enclosed by a concrete bio-shield [26]. The core of the reactor consists of 100 fuel elements, six control rods, 18 graphite dummy elements, a neutron source, a Dry Central Thimble (DCT), and a pneumatic transfer system irradiation terminus, [27-28]. Fig. 1 and 2 shows the TRIGA reactor shield structure and core structure, respectively. The reactor is accommodated in a hall having dimension of 23.5 m x 20.12 m, with a height of 17.4 m. The reactor hall volume is 8202.65 m^3. The cross-sectional view of the TRIGA reactor is shown in Fig.3. The BAERA issued license to operate the reactor at a steady-state maximum thermal capacity of 3 MW. In addition, the reactor can also be operated at peak power of about 852 MW at pulsed mode [29].

Fig. 1. The TRIGA reactor shield structure [26].

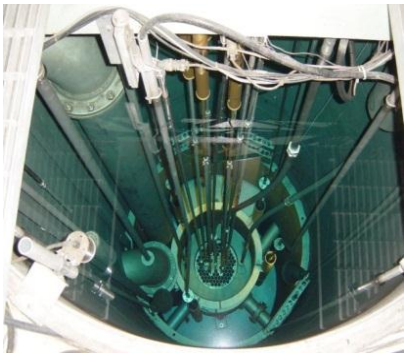

Fig. 2. The TRIGA reactor core structure [26].

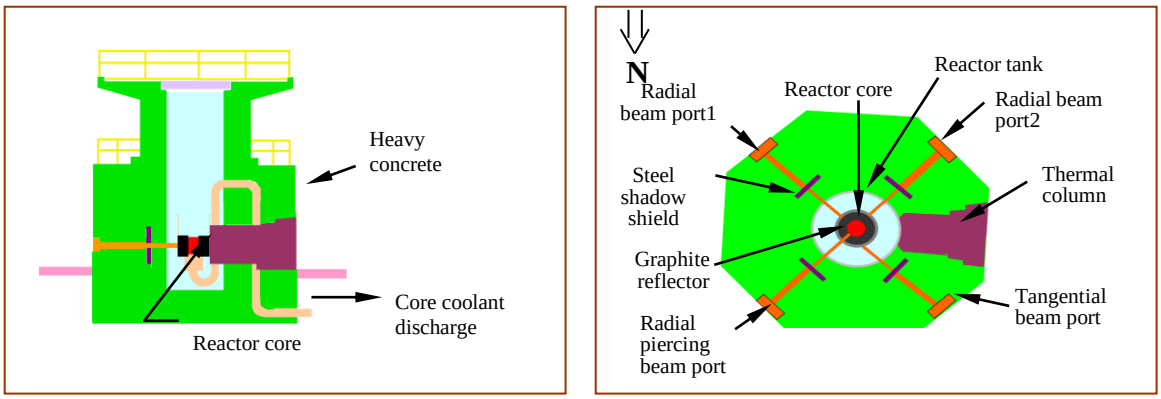

Fig. 3. Cross-sectional view of the TRIGA reactor [26].

Definition of source-term and calculation

For nuclear reactors, the term radiological source-term is used to express the released amount of radionuclides to the reactor containment. The amount of activity $A_i(t)$ caused by radionuclides (isotope i) can be determined by using Eq. (1) and accordingly the release rate can also be determined by using Eq. (2). The equation of activity $A_i(t)$ of an element (isotope) i is below [30].

$$A_i(t) = 0.82\gamma P\left(1 - e^{-\lambda_i T}\right) \times e^{-\lambda_i (t-T)} \quad (1)$$

The overall activity Q (τ) of an isotope i which is released over time τ is calculated using the equation given as follow:

$$Q_i(\tau) = F_P F_B A_i(t) \frac{\lambda_l}{\lambda_l + \lambda_r}\left[1 - e^{-(\lambda_l + \lambda_r)\tau}\right] \quad (2)$$

Where, t period (sec), γ fission product yield, λ_i the decay constant (sec^{-1}), T irradiation time, P thermal power (megawatts), F_P the release fraction from fuel to containment, F_B the airborne element ready to be released to the atmosphere from the containment building, λ_l the leak rate parameter (sec^{-1}), and λ_r the radioactive decay constant (sec^{-1}).

In this calculations we have made several assumptions: a) It was assumed that the reactor was in critical at 3 MW (t) thermal power (full power) and the reactor was operated continuously at 3 MW (full power) for 15 days; b) Just 2 hours after occurrence of the accident, the radioisotopes were assumed to be released into the atmosphere from the reactor stack; c) The considered radionuclides for dose assessment was ^{131m}Xe ($T_{1/2}$=11.93 days), ^{133m}Xe ($T_{1/2}$=2.19 day), ^{133}Xe ($T_{1/2}$=5.24 day), ^{135m}Xe ($T_{1/2}$=15.29 month), ^{135}Xe ($T_{1/2}$=9.14 hour) and ^{138}Xe ($T_{1/2}$=14.08 min); d) The fraction of release for radio-xenon was assumed 100 % and the leak rate parameter (λ_1) was considered 1.157×10^{-7} sec^{-1}. i.e., 1 %/day [31]. The calculated result for fission product inventory at maximum thermal power level 3 MW (t) by using Eq. 1 and 2 are shown in Table 1.

Table 1. Evaluated inventory of fission product for continuous operation of 15 days at full power (3 MW).

Radionuclide	Half-life ($T_{1/2}$)	Fission yield	Total activity in the core (Bq)	Released rate (Bq/sec)	Referenced activity (Bq), Sulaiman et al., 2019 [25]
^{131m}Xe	11.93 days	0.000313	1.65×10^{13}	1.91×10^{6}	9.73×10^{12}
^{133m}Xe	2.19 days	0.00183	1.60×10^{14}	1.85×10^{7}	5.96×10^{13}
^{133}Xe	5.24 days	0.066	5.03×10^{15}	5.82×10^{8}	2.03×10^{15}
^{135m}Xe	15.29 months	0.0122	2.48×10^{13}	2.88×10^{6}	3.45×10^{14}
^{135}Xe	9.14 hours	0.0661	4.76×10^{15}	5.76×10^{8}	1.07×10^{15}
^{138}Xe	14.08 minutes	0.0539	1.33×10^{13}	4.96×10^{5}	1.87×10^{15}

The atmospheric dispersion model

For the calculation of atmospheric dispersion, the Gaussian Plume Model (GPM) was applied, and this distribution in both lateral and vertical directions can be written as Eqs. (3,4) [32].

$$\chi(x,y,z) = \frac{Q_i}{2\pi.\sigma_y.\sigma_z.u_a}.\exp\left(\frac{-y^2}{2\sigma_y^2}\right)$$

$$\left\{\exp\left[-\frac{(Z-H)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(Z+H)^2}{2\sigma_z^2}\right]\right\} \quad (3)$$

$$\chi(x,y,z) = \frac{Q_i}{\pi.\sigma_y.\sigma_z.u_a}.\exp\left(\frac{-y^2}{2\sigma_y^2}\right)\left\{\exp\left[-\frac{H_{eff}^{\ 2}}{2\sigma_z^2}\right]\right\} \quad (4)$$

Here, χ(x,y,z) means the concentration of radionuclides at point (x,y,z) (Bq/m^3), Q_i means release rate (Bq/s), u_a means wind velocity (m/s) at the actual height H(m) of the reactor stack, σ_y and σ_z means the dispersion parameters (m) for lateral and vertical directions, respectively and H means height of the release (m).

The effective height of the stack can be expressed as follow in Eq. (5) [33]:

$$H_{eff} = H + D\left(\frac{v}{u}\right)^{1.4}\left(1+\frac{\Delta T}{T}\right) \quad (5)$$

Here, D means the diameter (m) of the stack outlet, v means the exit velocity of the effluent (m/s). For the BAEC TRIGA research reactor, the temperature difference, ΔT, is assumed to be zero due to the active operation of the air circulation system of the reactor facility.

Now, applying the following relationship [32,34], at effective stack height the wind velocity is converted into an average wind velocity as in Eq. (6):

$$u = u_z\left(\frac{H_{eff}}{z}\right)^m \quad (6)$$

Here, u_z means the ground level wind velocity at a height z =10 m and '*m*' means the coefficient of wind reliant on the primary surface and diffusion class.

Local meteorological parameters

Since there is no meteorological station at the TRIGA reactor site, the raw meteorological data was collected from the Bangladesh Meteorological Department (BMD), Dhaka, which is situated approximately at a distance of 40 km from the TRIGA reactor site. Important meteorological parameters such as wind frequency and wind speed were analyzed at the Nuclear Energy Division (NED), Institute of Energy Science (IES) at AERE for different directions. The data incorporates wind speed and frequency throughout the previous 10 years, for example (2009-2018). Fig. 4 and 5 show the percentage of average wind frequencies and average wind velocity for various directions, respectively. From Fig. 4, it is seen that the wind flow is dominant in the South direction around the reactor site (about 16.23 %). The wind velocity data measured by the BMD at 10 m height was then converted to 32.36 m height, i.e., effective stack height (Fig. 5) of the reactor. To consider the stability class surrounds the reactor area Pasquill-Gifford stability classification was considered in the Gaussian plume equation. The meteorological data throughout the previous 10 years (2009-2018) was utilized to ascertain the rates of frequency and speeds in 16 cardinal directions, shown in Fig. 5.

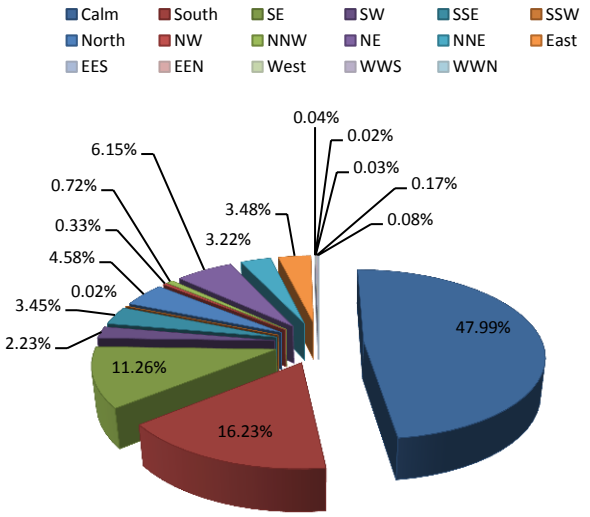

Fig. 4. The percentage of average wind frequencies for various directions.

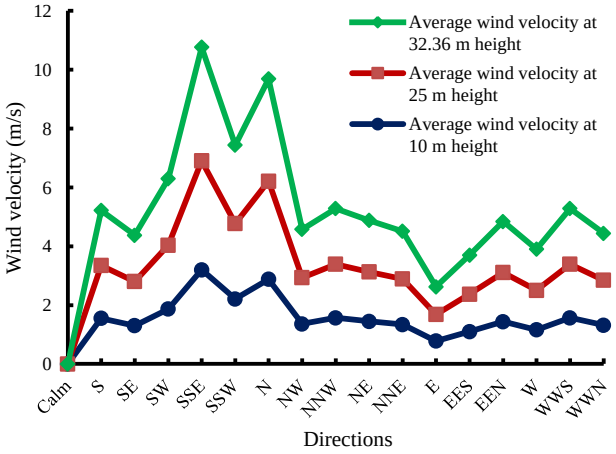

Fig. 5. Average wind velocity (percentage) at various height for different directions.

Air concentration and radiological dose calculations

For this case, with the subsequent simplifying assumptions, the sector averaged form of the GPM can be applied:

a) An individual wind direction with a frequency for calculation of each air concentration,
b) For each direction, a single long-term average wind velocity, and
c) A impartial atmospheric stability class (Pasquill-Gifford stability class A) [31].

Thus, the air concentration of a radionuclide was evaluated using the following in Eq. (7) [31]:

$$C_A = \frac{P_P F Q_i}{u_a}\exp\left(-\lambda_i \frac{x}{u_a}\right) \quad (7)$$

The Gaussian diffusion factor F can be evaluated by applying the 22.5° sector averaged form of the GPM for a defined value of H_{eff} as follows in Eq. (8):

$$F = \frac{16}{\sqrt{2\pi^3}} \times \frac{\exp\left[-\left(\frac{H_{eff}^{\ 2}}{2\sigma_z^2}\right)\right]}{x\sigma_z} \quad (8)$$

Here, σ_z denotes the diffusion parameter (m) in vertical direction.

It is considered that the area is covered with woodlands, fields, and little towns. The value σ_z can be estimated using the formula in Eq. (9) [31]:

$$\sigma_z = E \cdot x^G \quad (9)$$

Here, the two parameters *E* and *G* depend on the effective stack height and stability class, where *x* indicates the distance in downwind.

The radiological doses in various environmental pathways rely on the concentration distribution of radioactive materials of those pathways. The concentration calculation methodologies with the help of GPM are presented elsewhere [35]. The annual effective dose in the discharge plume due to immersion E_{im} can be given by in Eq. (10) [31]:

$$E_{im} = C_A DF_{im} Q_f \quad (10)$$

The annual effective dose in case of ground deposition E_{gr} can be evaluated as in Eq. (11) [36]:

$$E_{gr} = C_{gr}.DF_{gr}.Q_f \quad (11)$$

The total dose can be obtained by adding the annual effective dose due to immersion and ground deposition in Eq. (12):

$$E_{total} = E_{im} + E_{gr} \quad (12)$$

RESULTS AND DISCUSSION

To study the radiological dose within the TRIGA reactor site, a mathematical code was developed in this work using MathCAD software valid only for radioisotopes of xenon, such as ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, and ^{138}Xe. The code was developed to solve the mathematical expressions which consist of two parts. The first part was employed to calculate source-term and the second part to calculate concentration as well as doses in various environmental pathways. The input parameters used for the calculations were measured for the TRIGA reactor area. Thus, an attempt was made to investigate the contribution of dose due to the deposition of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, and ^{138}Xe on ground and immersion assuming a postulated accident of the TRIGA research reactor. One of the main focuses of the present study was also to establish a correlation between total effective dose and the air concentration from the pathways of immersion and ground deposition.

Table 2 gives the relationship between the total dose rate and air concentration. It is found from the table that the constant of proportionality for ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{138}Xe in all the directions are almost equal. From the table, correlation between total dose for various pathways and air concentration for radioxenon can be expressed as:

E_{total} = 1.17E-8 × CA for ^{131m}Xe (13)
E_{total} = 4.02E-8 × CA for ^{133m}Xe (14)
E_{total} = 4.38E-8 × CA for ^{133}Xe (15)
E_{total} = 5.84E-7 × CA for ^{135m}Xe (16)
E_{total} = 3.50E-7 × CA for ^{135}Xe (17)
E_{total} = 1.72E-6 × CA for ^{138}Xe (18)

Table 2. The relationship between air concentration and total dose from immersion.

Directions	E_{total} for ^{131m}Xe	E_{total} for ^{133m}Xe	E_{total} for ^{133}Xe	E_{total} for ^{135m}Xe	E_{total} for ^{135}Xe	E_{total} for ^{138}Xe
N	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A
NE	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A
E	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A
SE	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A
S	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A
SW	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A
W	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A
NW	1.17E-8 × C_A	4.02E-8 × C_A	4.38E-8 × C_A	5.84E-7 × C_A	3.50E-7 × C_A	1.72E-6 × C_A

* E_{total} = Total dose, C_A= Air concentration

Total dose contribution due to radio-xenon for immersion and ground deposition pathways

Total dose due to ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{138}Xe, and total (^{131m}Xe + ^{133m}Xe +^{133}Xe +^{135m}Xe+^{135}Xe+^{138}Xe) were evaluated after adding the contribution of two pathways, such as external doses due to immersion and ground deposition. The total dose rate for the radionuclides of ^{131m}Xe, ^{133}Xe, ^{133m}Xe, and ^{135}Xe as a function of time (in days) for eight directions are shown in Figs. 6, 7, 8, and 9, respectively. The total dose rate for ^{135m}Xe and ^{138}Xe as a function of time (in years and in hours) for eight directions are presented in Fig. 10 and Fig. 11, respectively. The result reveals that dose rates are maximum at time t = 0 and ^{138}Xe disappear after 10 hours, ^{135}Xe after 20 days, ^{133m}Xe after 92 days, ^{133}Xe after 250 days, ^{131m}Xe after 440 days in all the directions, and for ^{135m}Xe the dose rate disappears after 55 years in all the directions. Highest dose rate for different pathways at t = 0, were found in S-direction which are 1.23E-4 µSv/h, 4.09E-3 µSv/h, 0.14 µSv/h, 9.26E-3 µSv/h, 1.111 µSv/h, 4.55E-3 µSv/h and 1.269 µSv/h for ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{138}Xe and total (^{131m}Xe +^{133m}Xe +^{133}Xe +^{135m}Xe +^{135}Xe +^{138}Xe), respectively.

It is apparent that total dose from immersion is the only governing provider in total dose to the human for radionuclides of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{138}Xe and total (^{131m}Xe + ^{133m}Xe + ^{133}Xe + ^{135m}Xe + ^{135}Xe + ^{138}Xe). It was found that the contribution of dose from ground is zero in total dose for radionuclide of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{138}Xe and (^{131m}Xe + ^{133m}Xe + ^{133}Xe + ^{135m}Xe + ^{135}Xe + ^{138}Xe). As the total deposition co-efficient, V_t (=V_d + V_w), is assumed to be zero for non-reactive gases, the total ground deposition rate is zero, which leads the ground deposition density (C_{gr}) to zero, and as a result, there is no contribution of dose from ground deposition.

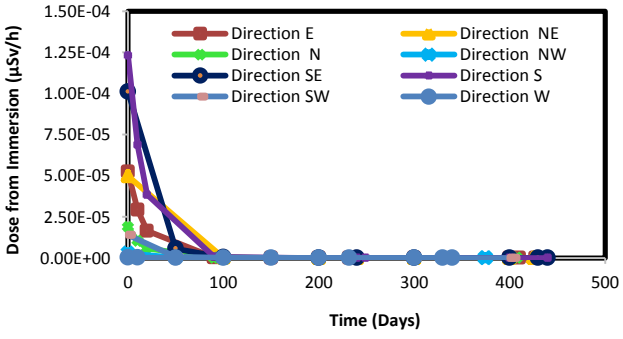

Fig. 6. Total dose rate of ^{131m}Xe intended for the directions of E, NE, N, NW, SE, S, SW, and W.

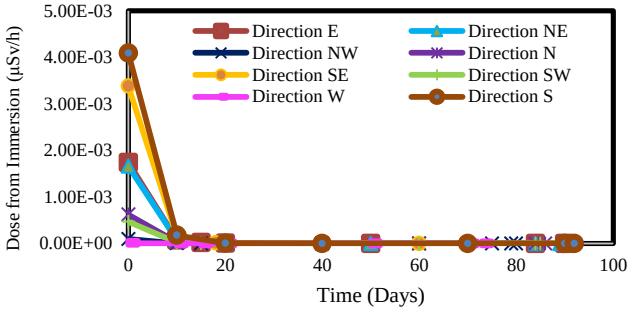

Fig. 7. Total dose rate of ^{133m}Xe intended for the directions of E, NE, N, NW, SE, S, SW, and W.

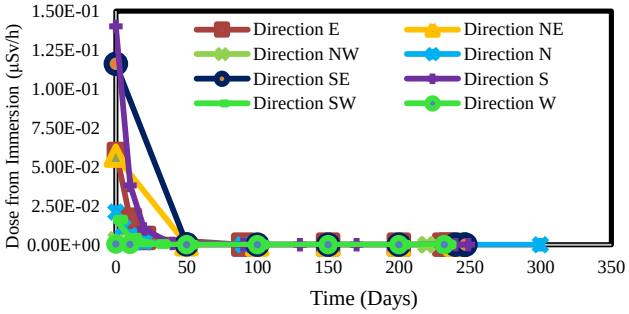

Fig. 8. Total dose rate of ^{133}Xe intended for the directions of E, NE, N, NW, SE, S, SW, and W.

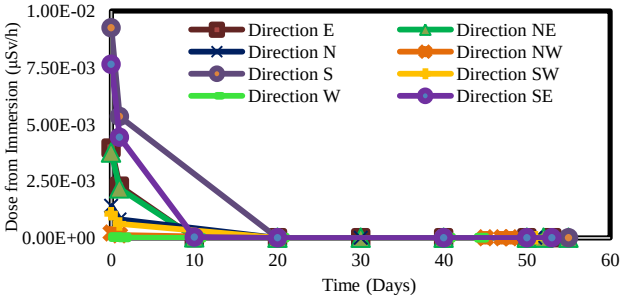

Fig. 9. Total dose rate of ^{135m}Xe intended for the directions of E, NE, N, NW, SE, S, SW, and W.

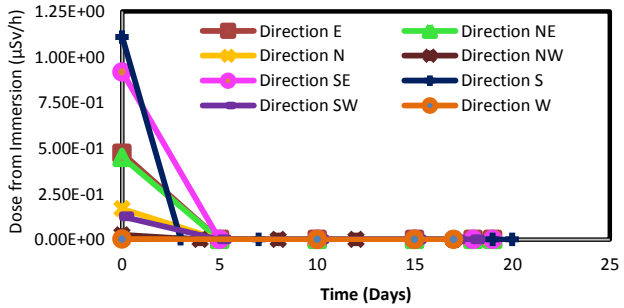

Fig. 10. Total dose rate of ^{135}Xe intended for the directions of E, NE, N, NW, SE, S, SW, and W.

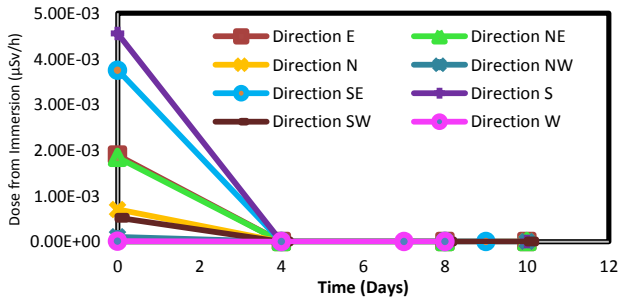

Fig. 11. Total dose rate of ^{138}Xe intended for the directions of E, NE, N, NW, SE, S, SW, and W.

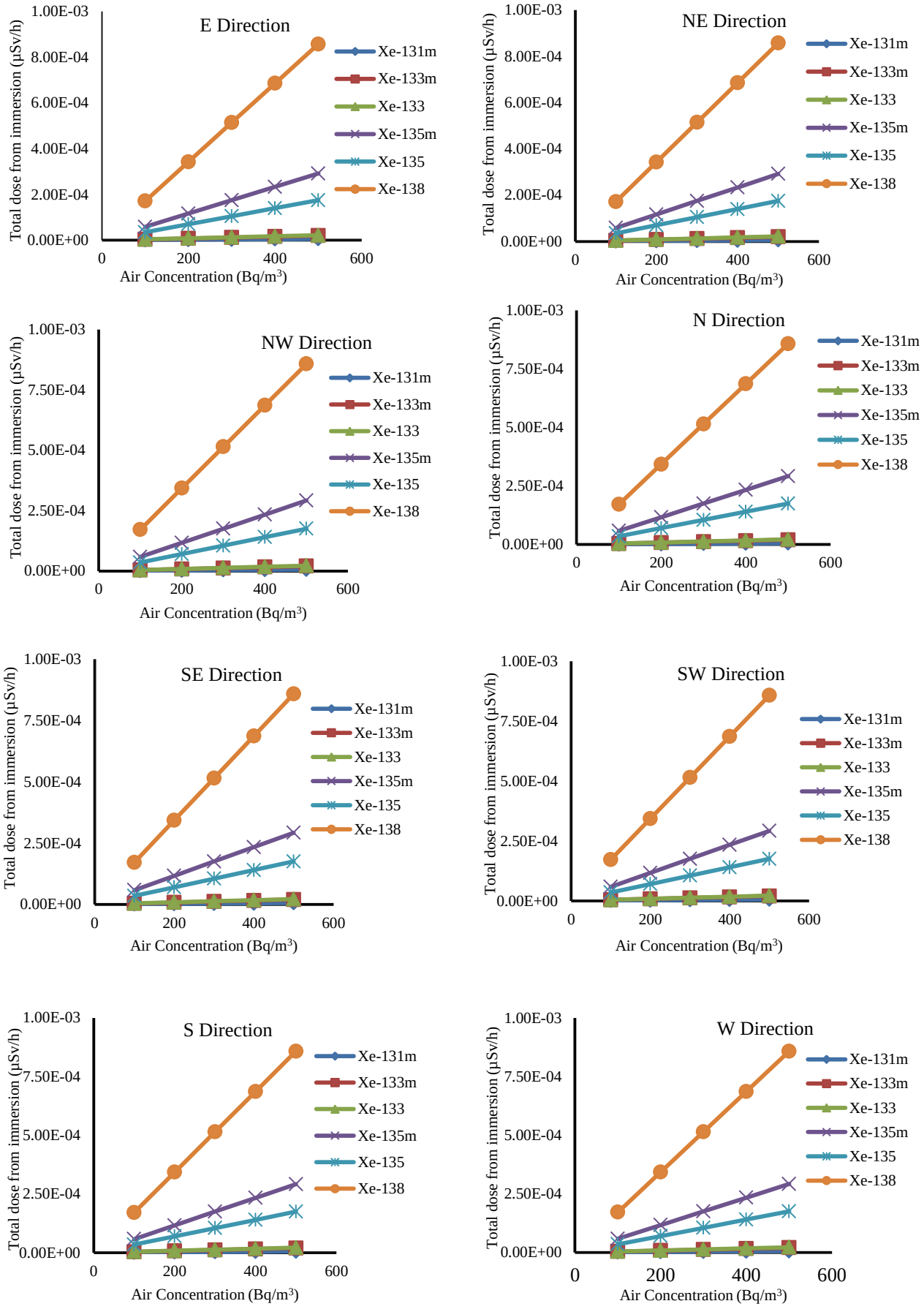

Fig. 12. Total dose rate of radioxenon from immersion as a function of air concentration (Bq/m^3) in different pathways for various directions of E, NE, N, NW, SE, S, SW and W.

Air concentration and dose evaluation for different pathways

In this study, an effort was made to establish a relationship between air concentration and the total dose for different pathways. It is obvious that with the occurrence of an accident in a nuclear reactor facility, the ejection of a substfantial amount of radionuclides through the stack to the environment should be occurred and then radioactive materials would be mixed in the air.

This concentrated radioactive materials may then expose to the people in and around the reactor facility through different pathways. Concerning this, it is crucial to understand the relationship between the concentration of radio-xenon in air and the total dose for various pathways. In this study air concentration of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{138}Xe considering eight directions with different downwind distance was evaluated. For every air concentration total dose values for various pathways was also evaluated. Fig. 12 represents air concentration and the total dose for radioisotopes of xenon for eight different directions. It is observed from this figure that the total doses are directly proportional to the air concentration.

In the present work, a study was performed to estimate radiological dose due to the deposition of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, and ^{138}Xe on the ground and immersion considering a hypothetical accident of TRIGA Mark-II research reactor at AERE, Savar. Another focus of the current study was to establish a correlation between air concentration and the total dose of different pathways. Since radionuclides released during accidental conditions are health hazardous, atmospheric dispersion and radiological dose calculations for the accidental releases of radioactive materials are required for ensuring safety of a nuclear research reactor like TRIGA type reactor. In this analysis, noble gas radioxenon was selected as they do not chemically combine with other material. As a result, they cannot be contained and will not be washed down. Hence atmospheric dispersion and dose calculations of radioxenon are useful to detect nuclear accident/nuclear test severity.

CONCLUSION

From the obtained results, the contribution of dose from the immersion of ^{135}Xe was found to be higher (87.55 %) than that of other isotopes of xenon. The contribution of dose from the immersion of ^{133}Xe was also found to be high (11.04 %). Contribution of dose from immersion of ^{135m}Xe, ^{138}Xe, ^{133m}Xe, ^{131m}Xe was found to be 0.73 %, 0.36 %, 0.32 %, 0.01 %, respectively. The maximum total dose (^{131m}Xe + ^{133m}Xe + ^{133}Xe + ^{135m}Xe + ^{135}Xe + ^{138}Xe) was found within the range of 0.003 – 1.269 µSv/h, respectively for all the dominant directions. The contribution of dose from the ground of ^{131m}Xe, ^{133m}Xe, ^{133}Xe, ^{135m}Xe, ^{135}Xe, and ^{138}Xe was found to be zero. Finally, it was found that the external dose from the immersion was the only contributor to the radiological dose to the human being due to radioxenon. It was found that radioactivity and doses were significantly higher in the accidental cases than in normal conditions. Since the doses were found higher due to severe accidental case, immediate radiological protective measures must be taken based on the emergency response plan of the national regulatory authority. The results of the dose distribution can be applied as regulatory requirements, such as establishing the radiological protective measures and preparing an emergency response plan for TRIGA research reactor site.

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AUTHOR CONTRIBUTION

All authors contributed to the concept, formulate and implementation of the research work. Authors also contributed to the calculation and analysis of the results and to the writing of the manuscript.

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