Assessment of Thyroid Radiation Doses Due To Various Iodine Radionuclides Released From Triga Mark II Research Reactor Accident

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***Abstract*— In the present work, the whole body external dose due to various iodine radionuclides for hypothetical reactor accident is calculated. Aiming to get the dose, calculations have also been done for core inventory, source term, and atmospheric dispersion factor. In this work, Pasquill-Gifford stability classification has been employed to consider stability class around the TRIGA reactor site. Six weather categories, designated from A to F are defined in order of increasing atmospheric stability and the site is found to be "B" according to the stability classification. A computer program named 'RaDARRA' has been developed in the present work for the calculation. The objective of this study is to assess thyroid doses due to various iodine radionuclides such as, ^{131}I, ^{132}I, ^{133}I, ^{134}I and ^{135}I at different distance from the reactor site for hypothetical research reactor accident. The largest contributors to the thyroid doses are ^{133}I and ^{131}I.**

***Index Terms*— dispersion , external, Iodine, radionuclides, reactor, research, whole body dose**

I. INTRODUCTION

For over fifty years research reactors have been a key component in the development of nuclear science and technology. Their contribution to nuclear power, basic science, materials development, radioisotope production for medicine and industry, and education and training of scientists and engineers, are well documented. There are 651 research reactors that have been built, of which 284 are operational in 58 countries (85 in 40 developing countries), 258 are shut down and 109 have been decommissioned as in April 2008 [1]. They have had a remarkable safety record and have amassed more reactor-years of operation than power reactors. Dozens of different designs have been built, often for special purposes, using a large variety of different fuel types and fuel geometry. Although over 60% of operational reactors were commissioned over thirty years ago, in many cases reactors have been refurbished one or more times so that the key components are very much younger. Of course, in many instances spent fuel has been accumulating for the lifetime of the reactor and some of it is badly degraded.

The potential impact of harmful releases into the environment from reactors has received growing attention after accidental releases, particularly the Chernobyl nuclear accident in 1986 in the former USSR [2]. During nuclear accidents, radioactive materials such as fission and activated products are released into the surrounding environment. Afterwards they migrate through various environmental media through which they finally expose the human and non-human biota. Release fractions of the isotopes present in the fuel can vary under severe accident conditions from virtually 0% for some elements up to 100% for the noble gases. Off-site radiological consequences will be dominated by the cesium isotopes and the halogens - in particular the iodine isotopes. These isotopes all together constitute 10–20% of the total fission product inventory, and are expected to have high release fractions under severe accident conditions. The radiological consequences of a severe hypothetical accident scenario are calculated and compared with radiological acceptance criteria for the purposes of emergency planning. The hypothetical accident is selected to represent an upper bound risk to the surrounding population. This is also used to assist in planning emergency arrangements. As has been observed the use of such an accident is also the basis for estimating the adequacy of emergency planning for any research reactor site.

In the past a few years or so there has been increasing interest, both nationally and internationally, in the development and use of severity scales to describe the impact of incidents and accidents at nuclear installations[3-7]. Reactor accident studies have been conducted in the developing countries for the analysis of the risk from severe accidents, which includes the environmental impact statement for each reactor constructed during the last decades. Bangladesh owns a 3 MW TRIGA Mark-II research reactor established at Atomic Energy Research Establishment (AERE) campus, Savar, Dhaka. But no such study has yet been conducted for it in the country till now. On this plea the present work has been undertaken in order to find out the safety zone around the reactor site [8]. The aim of the present study is assessment of thyroid dose at different ranges and directions from the reactor site, radiological impact assessment for the environment due to the release of iodine radionuclides to it during the accident. This study can ultimately be used to determine the radiological protective measures, which might be required to ensure the safety of the people living in the vicinity of the reactor site.

II. MATHEMATICAL FORMULATION

The 'source term' referring to the magnitude, composition and timing of the release of radioactivity to the environment [3] is often given as a fraction of the radionuclide inventory $q_i(t)$ (given in Ci) in the core. The total activity of isotope i released over time t, Q (t), is obtained from the following equation [4]:

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$Q_i(\tau) = F_P F_B\, q_i\, [\lambda_\ell / (\lambda_\ell + \lambda_r)]\,[1 - \exp\{-(\lambda_\ell + \lambda_r)\,\tau\}]$ (1)

where F_P = fractional release from fuel to building, F_B = fraction remaining airborne and available to be released from the building to the atmosphere, q_i = quantity of isotope i in Curies, in the reactor core at time of accident, λ_ℓ = leak rate parameter in the unit of sec^{-1}, and λ_r = radioactive decay constant. An approximate formula giving activity $q_i(t)$ of an isotope i at time t after the start of irradiation (t=0) whose fission yield is γ and its decay constant is λ irradiate for time period T in P (megawatts of thermal power) can be written [4] as:

$$q_i = 0.84\,\gamma_i\, P_o [1 - \exp(-\lambda_r T_o)] \qquad (2)$$

in which q_i = the amount of isotope i in Curies contained in the fuel after T_o, P_O = the fuel power level, γ_i = the ^{235}U fission yield of isotope i, λ_r = the radiological decay constant for the isotope, and, T_o = the time interval during which the fuel has been at power P_o.

Gaussian plume model is widely applied to calculate atmospheric dispersion. This model assumes that a Gaussian distribution in both lateral and vertical directions can be described in case of the dispersion of radioactive materials released to the atmosphere which can be expressed by [5]

$$\chi(x,y,z) = \frac{Q}{2\pi.\sigma_y.\sigma_z.u_x} .\exp[\ \frac{-y^2}{2\sigma^2{}_y}\ ] \exp[-\frac{(z+h)^2}{2\sigma_z{}^2}] + \exp[-\frac{(z-h)^2}{2\sigma_z{}^2}] \qquad (3)$$

Here $\chi(x,y,z)$ is the atmospheric concentrations at point (x,y,z) (Bq/m^3), Q is the uniform release rate of radioactive materials from the stack (Bq/s), u_x is the wind speed at the height of the stack (m/s), h is the effective stack height (m), and σ_y, σ_z are the lateral and vertical dispersion parameters (m).

For any radial distance x from the release point, the maximum ground level (z=0) time-integrated concentration directly downwind occurs beneath the center line (y=0) of the cloud. The value there, useful as a conservative estimate for the time-integrated concentration at off-centerline positions, is

$$\chi = \frac{Q}{\pi\sigma_y\sigma_z u} \exp\left[-\frac{1}{2}\left(\frac{H}{\sigma_z}\right)^2\right] \text{Bq-sec/m}^3$$

Therefore the atmospheric dispersion factor (= χ / Q) is

$$\chi / Q = \frac{1}{\pi\sigma_y\sigma_z u} \exp\left[-\frac{1}{2}\left(\frac{H}{\sigma_z}\right)^2\right] \text{sec/m}^3 \qquad (4)$$

The calculation of dose estimates may be split into two categories, an internal (inhalation) dose and an external (immersion) dose. The thyroid dose from isotope i to organ k which is the interest of the present work may be expressed as

$$D_i^{Thy} = \chi/Q\ (t) \times Q_i(t) \times DCF_i^{Thy} \qquad (5)$$

In this case $\chi/Q(t)$ is the atmospheric diffusion factor (s/m^3), Q_i (t) is the inventory of isotope i released over time t (Ci), DCF_i^{Thy} is the external dose conversion factor for organ k and isotope i (rem/ci). The total dose to organ k or to whole body is then the sum over all isotopes in the inventory released in the time interval t. The diffusion factor and the inventory released may vary with time.

III. Results and Discussion

In the present work, radiation dose from various radionuclides due to a hypothetical reactor accident is calculated. Aiming to get the radiation dose, calculations have also been done for finding out core inventory, source term, ground level atmospheric concentration, and thyroid dose.

A. The Core Inventory

The inventory of fission products and other radionuclides can be estimated by adopting Eqn. (2). A computer program named 'RaDARRA' (Radiological Dose Assessment for Research Reactor Accident) has been developed in the present work for the necessary calculations. After verification of the RaDARRA code, the core inventory of the RR was calculated for 600 MWD for continuous operation of the reactor. We have considered the accident to occur after running the reactor for 200 days, which is equivalent to 600 MWD (3 MW x 200 days). Core inventory values for 2 MW, 2.5 MW and 3 MW power (Table 1) show that the values are increased due to increase of power of a reactor.

Table 1. Core inventory values for various radionuclides

Radionuclides	Half-life(days)	Core inventory, q_i (Bq) $\times 10^{13}$				
		2 MW (200 days)	2.5 MW (200 days)	3 MW		
				100 days	200 days	300 days
^{131}I	8.04	174	218	261	261	261
^{132}I	0.09	255	319	382	382	382
^{133}I	0.86	423	528	634	634	634
^{134}I	0.03	448	559	671	671	671
^{135}I	0.27	398	497	597	597	597

The table shows that the highest value of the core inventory is for ^{134}I, followed serially by those for $^{133}I > ^{135}I > ^{132}I > ^{131}I$ for all cases of varying irradiation time. Due to the variation of irradiation time even from 100 days to 300 days, there is no remarkable change in the values of core inventory of the radionuclides ^{131}I, ^{132}I, ^{133}I, ^{134}I and ^{135}I. A careful watch at the table will make clear about this difference – the first group is having low values of half life; whereas the other ones have long half lives.

B. *Source Term, $Q_i(t)$*

The total activity of isotope i released over time t, is obtained from Eqn. 1. The leakage rates depend strongly on system design and containment or reactor building design. In addition, the leak fraction of a given radionuclide depends on its chemical form. The noble gases, krypton and xenon, will be free to escape completely; solid, non-volatile fission products will remain in place. Iodine is volatile, and it is normally assumed that a significant fraction of it will escape. An assumption must be made regarding the release of radionuclides to the containment. One common set of assumption is that 100% of the noble gases, 50% of the halogens, and 1% of the solids in inventory are released into the containment, and to assume that only 50% of the halogens remain available for release to the atmosphere. This assumption has been made in the present work as well. A leakage rate to the atmosphere of 1%/day is assumed for most of the sample cases. The variation in leakage rate with the change in pressure within the building is not included. The inventory of fission product of various radionuclides required for the calculation of source strength for 200 days irradiation time and 3 MW power is obtained from Table 2. These source strength values are relevant for 30 days after the accident. The source strength is calculated first considering 30 days after the hypothetical reactor accident has taken place. These calculations have been done for the radionuclides of interest, e.g., ^{131}I, ^{132}I, ^{133}I, ^{134}I, and ^{135}I (Table 3).

Table 2. Source strength values for the radionuclides of interest

Radionuclides	Half life	Source strength, Q_i (Bq)$\times 10^{11}$					
		0-2 h	3-8 h	9-24 h	2-4 d	5-30 d	Total for 30 days
^{131}I	8.04	5.39	16.1	42.1	170	624	857.59
^{132}I	0.1	5.98	11	13.1	13.2	13.2	56.48
^{133}I	0.87	12.7	35.9	81.6	178	196	504.19
^{134}I	0.04	7	8.74	8.82	8.82	8.82	42.2
^{135}I	0.28	11.2	27.7	48.16	59	58.96	204.9

C. *Atmospheric Dispersion Factors*

Dispersion factors, (χ/Q) are determined based on actual values of release height, wind velocity, air stability, and the distance from release to receptor locations. Calculation of atmospheric dispersion factor has been made for B stability for which the actual wind speed of all the 16 cardinal directions has been considered. Distance has been varied from 50 m to 1000 m; and variation from 100 m to 1000 m has been maintained at a slab of 100 m. The cardinal directions were all standard ones, e.g., E meaning east, EEN (East East North) etc. Other directions are EES, N, NE, NNE, NNW, NW, S, SE, SSE, SSW, SW, W, WWN, WWS. The calculated results of atmospheric dispersion factor values finally show the following salient features:

The dispersion factor (=χ/Q) is found to be inversely proportional to the wind velocity (Fig. 1). The maximum value of χ/Q is found to occur along the north (N) direction whereas the minimum value is along north north east (NNE). At 1000 m distance, the magnitude is not significant. It is found to lie within 0.02×10^{-4} to 0.03×10^{-4} s/m^3. The ratio of maximum to minimum value of the χ/Q is 1.56 at a distance of 50 m. This ratio is supposed to be valid at all the distances, but due to the effect of approximation it has changed a bit towards the higher distance values.

D. *Conversion Factors for the Calculation of Dose*

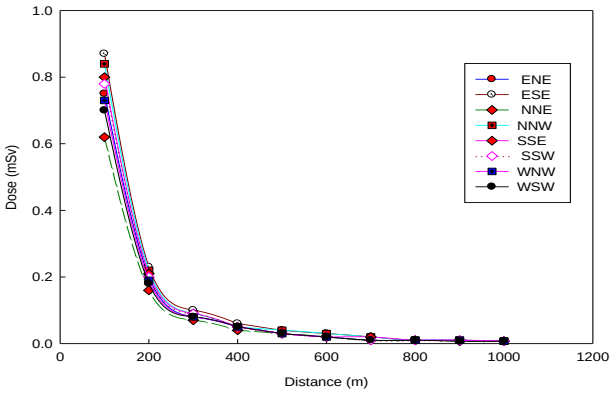
Necessary dose conversion factors for the calculation are given in Table 3. Other parameters necessary for the calculation are duly considered in the work. Out of them the value of wind speed for B stability class is based on the country's meteorological observation. The release height value of 25 m is a realistic datum. Dose conversion factor values for whole body external cases have been taken from literature [6].

Table 3. Dose conversion factors for the radionuclides of interest.

Type of radionuclides	External (whole body) (Sv.m^3/Bq.s)
^{131}I	4.46×10^{-14}
^{132}I	2.14×10^{-13}
^{133}I	8.42×10^{-14}
^{134}I	2.32×10^{-13}
^{135}I	1.57×10^{-13}

E. *Dose Values*

The whole body external dose can be estimated by using Eqn. 5. In the present work, the whole body external dose is calculated considering a gap of 30 days after the accident for 16 cardinal directions due to the downwind distances of 50 to 1000 m.



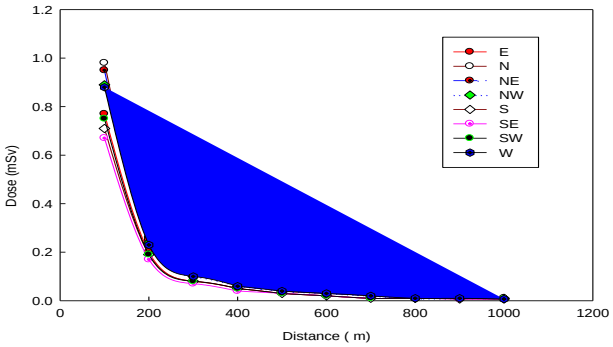
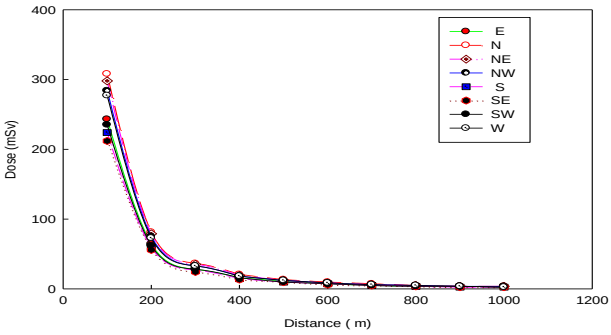
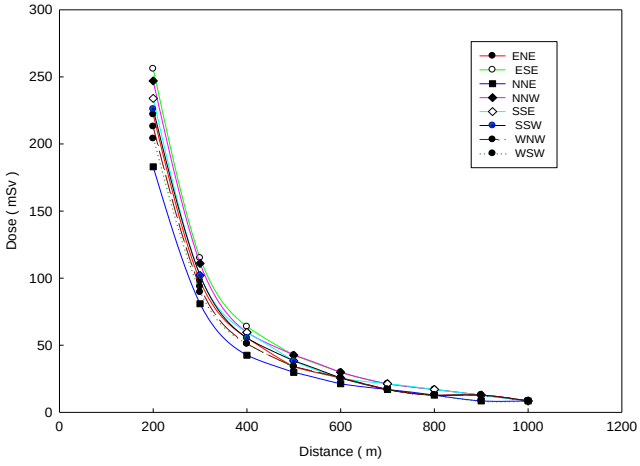


Fig. 1 Variation of dose due to ^{131}I with distance from the reactor along the 16 cardinal directions.





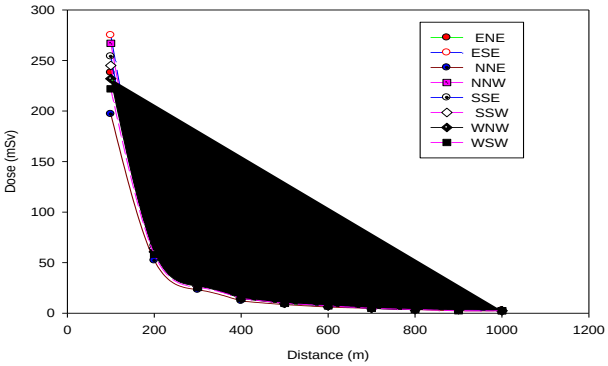


Fig. 2. Variation of dose due to ^{132}I with distance from the reactor along the 16 cardinal directions

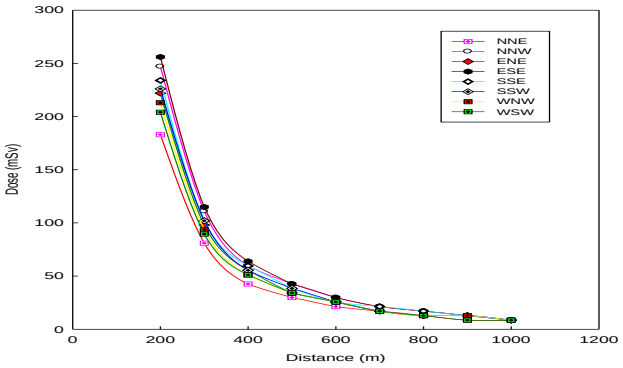
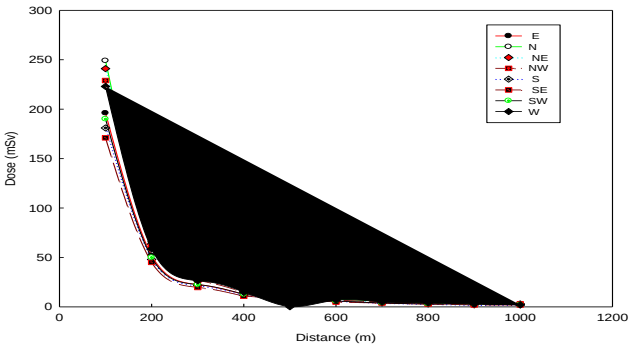


Fig. 3. Variation of dose due to ^{133}I with distance from the reactor along the 16 cardinal directions.



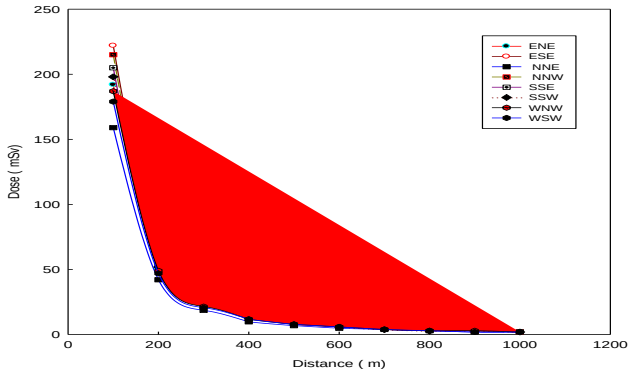


Fig. 4. Variation of dose due to ^{134}I with distance from the reactor along the 16 cardinal directions.

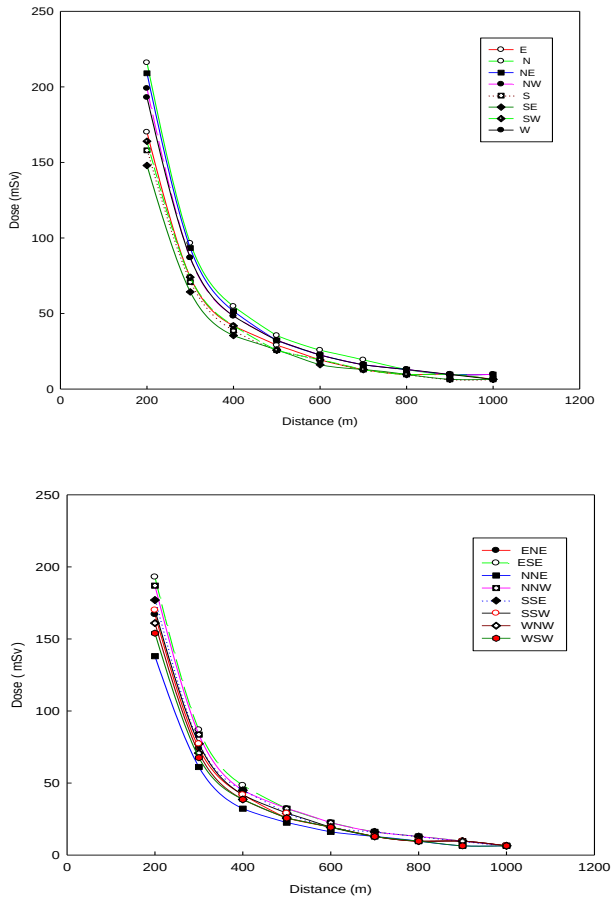


Fig. 5. Variation of dose due to ^{135}I with distance from the reactor along the 16 cardinal directions.

The calculated results for ^{131}I are shown in Fig.1 A significant variation has been observed amongst the results in different directions, maximum values are found always towards N and minimum towards NNE, the values varying by a factor of 1.56. External whole body dose for ^{131}I varies according to the following serial

N > NE > NW > W > ESE > NNW > SSE > SSW > E > ENE > SW > WNW > S > WSW > SE > NNE.

This serial of variation has also been observed in case of all other radionuclides studied presently. The maximum value of the external whole body dose is 1.56 times higher than the minimum one of magnitude 1.77 mSv happening along NNE. The main reason of the variation is the meteorological cause of air speed. Along N the average speed is 3 m/s against 4.69 m/s along NNE. Results of variation of the whole body external dose for ^{131}I also show that the doses decrease with the increase of distance from the reactor. This result is obvious. For all the radionuclides the dose versus distance curve (Fig.1) shows that the variation obeys approximately the inverse square law. The same pattern of variation is observed for all the 16 cardinal directions.

The dose value due to ^{131}I at 50m is 2.75 mSv, which is higher than the limit of public exposure (1 mSv/yr). Due to the same radionuclide the dose values at 100 m and 1000 m distances are 0.87 mSv and 0.01 mSv respectively. These values are below the annual exposure limit. So people remain safe from hazard due to this radionuclide after 100 m distance from the reactor site. The whole body external dose due to ^{132}I is shown in Fig. 2. The dose value due to this radionuclide is less than that of ^{131}I. The values remaining within the 50 m to 1000 m distance slab and along specified directions vary from 0.873 mSv to 0.003 mSv, and thus are insignificant. So remaining even at 50 m distance, people are out of danger. The dose value due to the radionuclide ^{133}I (Fig. 3) is higher than that due to the other two radionuclides. The dose value measured at 50 m distance is 3.06 mSv; and that at 100 m distance is 1.08 mSv. The latter one is very near to the annual limit of public exposure. The dose values arising from ^{134}I radionuclide as shown in Fig. 4 are comparatively less than that from other iodine isotopes. The dose value at 50 m distance is 0.705 mSv and at 1000 m distance is 0.002 mSv. There will be hardly any effect of radiation to the people within this range due to the isotope. Fig. 5 shows that the dose value due to ^{135}I is insignificant at 1000 m distance apart. The dose value at 50 m distance is 2.32 mSv, which crosses the safety limit. But from 100 m to 1000 m distances from the reactor no remarkable dose value is observed.

3.5 Total Dose at Various Site Boundaries

After calculating the doses from various iodine radionuclides at different distances from the reactor now the individual contributions are added to get the total dose. Though the calculation was done for 16 cardinal directions from 50 m to 1000m distances but it was found that the doses were maximum towards north direction. Total dose values calculated this way are shown in Table 4 considering the north direction and for 30 days only after the reactor accident. Doses from 3 MW research reactor are compared in the table with guide line doses for 500 m site boundary from 10 CFR 100 [7].

Table 4. Thyroid total doses at different site boundary distances from 3 MW research reactor

Site boundaries in meter	Total dose (mSv) (30 days)	Total dose (mSv) (2 hr)	Guideline doses (mSv) for 500 m site boundary from 10 CFR 100 [6]
50	9.76	0.43	250
100	2.47	0.15	
500	0.144	0.007	
1000	0.064	0.002	

The present value of thyroid dose is 0.144 mSv (after 30 days). It may be mentioned here that the value of 250 mSv for whole body external dose was once considered the maximum once-in-a-lifetime doses for radiation workers. As noted in 10 CFR 100, these values should not be considered acceptable limits for doses to the public under accident conditions but may be used as reference values for evaluations of reactor sites. In the work doses are also calculated separately for 2 hr duration for whole body external dose. The calculated values in this work (Table 4) are well below the guideline dose as prescribed by the USNRC at 500 m site boundary. It may be mentioned here that no such guideline are formulated for the perspective of Bangladesh. A regulatory body should be formed to formulate such guideline dose for various site boundaries for the RR of our country.

IV. CONCLUSION

A software RaDARRA has been developed in Visual Basic language to assess the external and internal radiation doses following an accidental release of radionuclides to the atmosphere. The software is user interactive and asks for source parameters, meteorological parameters and distances of interest. This software also evaluates the production of fission products as well as the transport of these fission products through all possible pathways. The modified atmospheric diffusion model adopted from Gaussian model is embedded in the software to provide a rapid calculation of the radioactive effluent dispersion and the dose levels in the EPZ (emergency planning zone). The program is expected to be useful not only in a radiation exposure but also for routine dispersion calculations. The output of software RaDARRA is available on both screen and the line printer. The program is useful for planning actions in an off-site radiological emergency.

The doses for release after 200 full power days reactor operation at a 500 m site boundary are given at 2 hrs and 30 days. The largest contributors to the thyroid external doses are ^{133}I and ^{131}I. This is far below the public exposure dose limit (1 mSv/yr).

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