Estimation of Thyroid Dose From the Miniature Neutron Source Reactor

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ABSTRACT

A computer code programmed in Pascal for a simple and fundamental calculation model has been developed and used for the estimation of radiological dose to the thyroid arising from a hypothetical failure of the whole core. The code was validated using the International Atomic Energy Agency (IAEA) Generic 10MW Reactor. Results obtained are comparable to those obtained by other workers. Calculations based on non-site specific data for Miniature Neutron Source Reactor (MNSR) gave very conservative results. The calculated dose to the thyroid (the limiting dose) of a receptacle (person) at 500m, 1000m and 5000m from the nuclear reactor were found to be very low, suggesting that the MNSR is safe even in the event of total core meltdown.

Keywords Fission, radiological dose, radioactive, radionuclides, activity, fuel element

INTRODUCTION

Reactor accident evaluations, which form part of Safety Analysis Report (SAR) for research reactors, are based on the assumption that a hypothetical accident results in the release of some portion of the inventory of radioactive materials to the atmosphere.

The doses and their relative significance strongly depend on the kind of accident considered, the fission products inventory of the core, the extent of fuel failure and hence the amount of radionuclides released to the atmosphere, the distribution of release rate, the meteorological conditions of the reactor region etc.

The accidental air borne releases of radionuclides contribute to the following radiological doses. External whole body dose (gamma) due to submersion in the exhaust air plume, external whole body (gamma) dose due to the activity deposited on the ground, internal irradiation originating from radionuclides inhaled with the air, resulting in both critical organ and whole body doses, external β-radiation from the exhaust air plume (skin dose mainly) and internal irradiation due to consumption of contaminated food [1].

An important element from human biology standpoint is iodine, which concentrates in the thyroid gland. During its stay in the thyroid, the radioactivity of each iodine isotope can result in deposition of energy which can cause serious damage to the gland.

This paper presents a model based on U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guides [2] for estimating radiological doses from hypothetical accidents in research and test reactors and the calculated results form part of the discussion.

THEORY AND CALCULATIONAL MODEL

General

In general, evaluating the dose D to an organ or individual is a two step process.

1. evaluate the activity released from the containment,
2. evaluate the dose arising from the atmospheric transport of the activity released.

Step 1 can be broken into a number of stages:

i. the core contains an inventory $q_i(t_0)$[Bq] i =1, 2, 3, 4.....n nuclides at t_0, the time of the accident.
ii. A portion of the core f_c is damaged.

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iii. Activity is released from the damaged portion of the core. A fraction f_{FEi} for each nuclide will be released from damaged fuel elements.

iv. The released activity may pass through several barriers, each of which may contribute to the retention, and decrease the available activity for eventual release. These release fractions will be denoted by $f_{B1}, f_{B2}, \ldots\ldots\ldots f_{Bn}$ etc.

v. Finally a fraction f_{CB}, will be the activity remaining airborne and available to be released from the building to the environment. The environmental source term will become for each nuclide[1];

$$Q_i = f_{CBi}.(f_{B1i}.f_{B2i}, \ldots\ldots\ldots f_{Bni}).f_{FEi}.f_C.q_i(t_0) \text{ [Bq]} \quad (1)$$

where $q_i(t_0)$ is the core inventory of nuclide i.

If data on $q_i(t_0)$ is not available from the reactor manufacturer or the fuel supplier then it can be calculated as follows [3]

$$\frac{dN_i}{dt} = \frac{\Gamma_i}{\omega} - \lambda_i N_i(t) \quad (2)$$

where $N_i(t)$ is the number of atoms of the ith fission product at time t.

Γ_i is the cumulative yields (takes account of short lived precursors) of the fission product

ω is the thermal energy produced per fission

P is the thermal power of the reactor

λ_i is the decay constant

With P constant in the time interval (0, t), Equation 2 integrates to

$$N_i(t) = \frac{\Gamma_i P(1 - e^{-\lambda_i t})}{\omega \lambda_i} \quad (3)$$

The activity $q_i(t)$ is $\lambda_i N_i(t)$. Substituting this in Equation 3 gives

$$q_i(t) = \frac{\Gamma_i P(1 - e^{-\lambda_i t})}{\omega} \quad (4$$

With ω = 200Mev/fission =

$$200*1.6*10^{-13} \text{J/fissi}$$

$$= 3.2*10^{11} \text{watts.sec/fission}$$

$$q_i(t) = 3.13*10^{4}*\Gamma_i*P*(1 - e^{-\lambda_i t})[T.Bq] \quad (5)$$

$$= 0.84*\Gamma_i*P*(1 - e^{-\lambda_i t})\text{Ci} \quad (6)$$

where P is in watts.

Fuel Element Release Fractions

Significant factors influencing f_{FE} are;
Physical shape/structure of fuel elements, ie., is gas release possible? The extent of fuel element melting and chemical and physical properties of fission product nuclei, eg; volatility etc.

The USNRC Regulatory Guides 1.3 and 1.4 require the following assumptions if no other information is available [2].

Primary Groups	f_{FE}
Inert Gases	100%
Volatiles	25% {91% I_2, 4% CH_3I, 5% particulates}
Non-Volatiles	1%

Source Terms and Leakage Rates

Assuming that there are no barriers other than the containment/reactor building, then the total activity of isotope i released over time τ
$Q_i(\tau)$, is obtained from the following equation [4]

$$Q_i(\tau) = f_c*f_{FE}*f_{CB}*q_i(t)*\frac{\lambda_L}{\lambda_i + \lambda_r}[1 - e^{-(\lambda_i + \lambda_r)\tau}] \quad (7)$$

where

f_C = fraction of core that melts,
f_{FE} = fractional release from fuel to containment building,
f_{CB} = fraction remaining airborne and available to be released from the building to the atmosphere
$q_i(t)$ = quantity of isotope i in reactor core at the time of accident, Ci,
λ_L = leakrate parameter, sec^{-1},
λ_r = radioactive decay constant, sec^{-1}.

Dose Calculation

The calculation of dose estimates is split into an internal (inhalation) dose and an external (immersion) dose. The internal dose to organ k from isotope i is expressed [5];

$$D_i^k = \frac{X}{Q_{(\phi)}}*Q_i(t)*BR(t)*DCF_i^k \quad (8)$$

and the external dose from isotope i may given by

$$D_i^{Ex} = \frac{X}{Q_{(\phi)}}*Q_i(t)*DCF_i^{Ex} \quad (9)$$

where

$\frac{X}{Q_{(\phi)}}$ is the atmospheric diffusion factor, s/m^3,

$Q_i(t)$ is the inventory of isotope i released over time t, Ci,

$BR(t)$ is the breathing rate for the receptor during the time t, m^3/sec,

DCF_i^k is the dose conversion factor for organ k,

DCF_i^{Ext} is the dose coversion factor for whole body.

Calculation of Atmospheric Diffusion Factor

If X is the concentration of some effluent as function of space and time, then X is determined by the time dependent diffusion equation as follows[3];

$$k\nabla^2 X = \frac{\partial X}{\partial t} \qquad (10)$$

in which K is the diffusion coefficient (cm^3)
If a point source located at x = y = z = 0 at t = 0 emits an isotropic puff containing a total of Q units of effluent into an infinite stationary atmosphere, it can be shown that the solution to Equation 10 which satisfies all the boundary conditions, is[3]

$$X_{(r,t)} = \frac{Q}{(4\pi Kt)^{3/2}} * e^{-r^2/4Kt} \qquad (11)$$

where r is the distance from the origin.
If the atmosphere is not isotropic then the solution is expressed in the form

$$X_{(x,y,z,t)} = \frac{Q}{(4\pi t)^{3/2}(K_x K_y K_z)^{1/2}}$$

$$\exp[-1/4t(\frac{x^2}{K_x} + \frac{y^2}{K_y} + \frac{z^2}{K_z})] \qquad (12)$$

where K_x, K_y, and K_z, are the diffusion coefficients in the x, y, z directions respectively.
Further derivation yields the following final expression[3];

$$\frac{X}{Q_{(t)}} = \frac{1}{\pi v \sigma_y \sigma_z} \exp(\frac{-h^2}{2\sigma_z^2}) \qquad (13)$$

where

$\frac{X}{Q_{(t)}}$ = dilution factor or atmospheric diffusion factor[3]

σ_y = horizonal dispersion coefficient,
σ_z = vertical dispersion coefficient
h = An altitude h where effluents (radiation) are emitted.

Breathing Rate Data

For the calculation of inhalation doses the breathing rate of the receptor during the time of exposure must be specified. The breathing rate data for man in these analysis are taken from USNRC Regulatory Guides 1.3 and 1.4[2].

The first eight hours of the exposure are taken to be an active period with a breathing rate of $3.47*10^{-4} m^3/s$. The interval from eight to twenty-four (8-24) hours is considered to be a resting period with a rate of $1.75*10^{-4} m^3/s$. For time periods greater than one day a breathing rate of $2.32*10^{-4} m^3/s$ is used [5].

Dose Conversion Factors

During its stay in the thyroid, the radioactivity of each iodine isotope can result in damage to the thyroid gland. This effect measured in rem, is estimated by using dose conversion factor (DCF). DCF data are available from a number of sources for external whole body immersion and for internal whole body, bone, lung, thyroid etc.

COMPUTER PROGRAMME

A computer code, Core3 which computes dose estimates for thyroid has been developed in this work.

A Pascal Function Q_i in the programme uses Equation 6 to calculate the core inventory of the isotopes and Function Activity calculates the following term in Equation 7;

$$\frac{\lambda_1}{(\lambda_1 + \lambda_r)}[1 - e^{-(\lambda_1 + \lambda_r)t}]$$

in Equation 7. The atmospheric dispersion factors (X/Q), are read from datafiles. It was assumed in the computation that 100% of noble gases, 25% of halogens, and 1% of all others are available for release from the containment building, and a leakage rate from the reactor building is 1% day [5].

RESULTS

Table 1 shows the activity (source term), cumulative activity, dose and cumulative dose of I-131 at 500m for the MNSR over various time intervals. Similar results were obtained for the remaining iodine isotopes (I-132, I-133, I-134, I-135) that contribute to the thyroid dose. It is seen from the table that after 30 days only 1.146Ci of the I-131 inventory have been released to the atmosphere. The half-live of I-131 is 8.07 days. This means that there are about four (4) half-lives before the thirty days. These results indicate that there is a substantial reduction in the activity released.

Table 1. Activities and Doses of I-131 from MNSR at 500m

Release Time	Activity(Ci)	Cumulative Activity(Ci)	Dose (rem)	Cumulative Dose (rem)
0 - 2 h	0.136	0.136	0.141	0.141
2 - 8 h	0.405	0.541	0.421	0.562
8 - 24 h	1.058	1.599	0.132	0.694
1 - 4 d	4.270	5.870	0.245	0.940
4 - 30 d	15.618	21.487	0.209	1.148

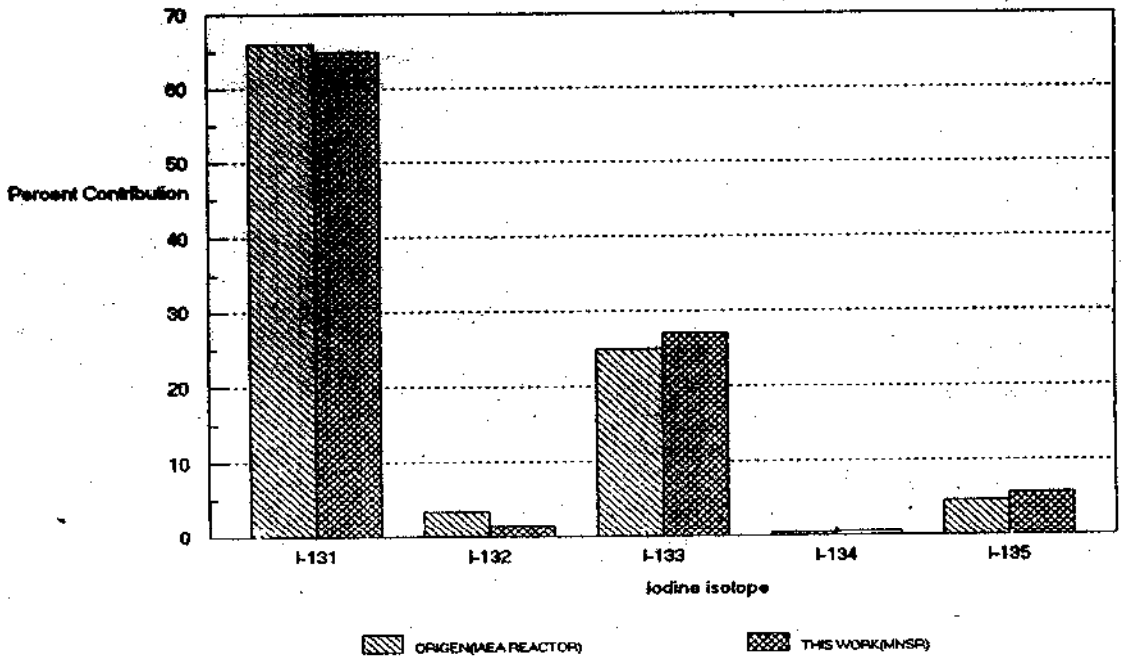

Fig. 1 Percent Iodine Isotopic Contributions toThyroid Dose

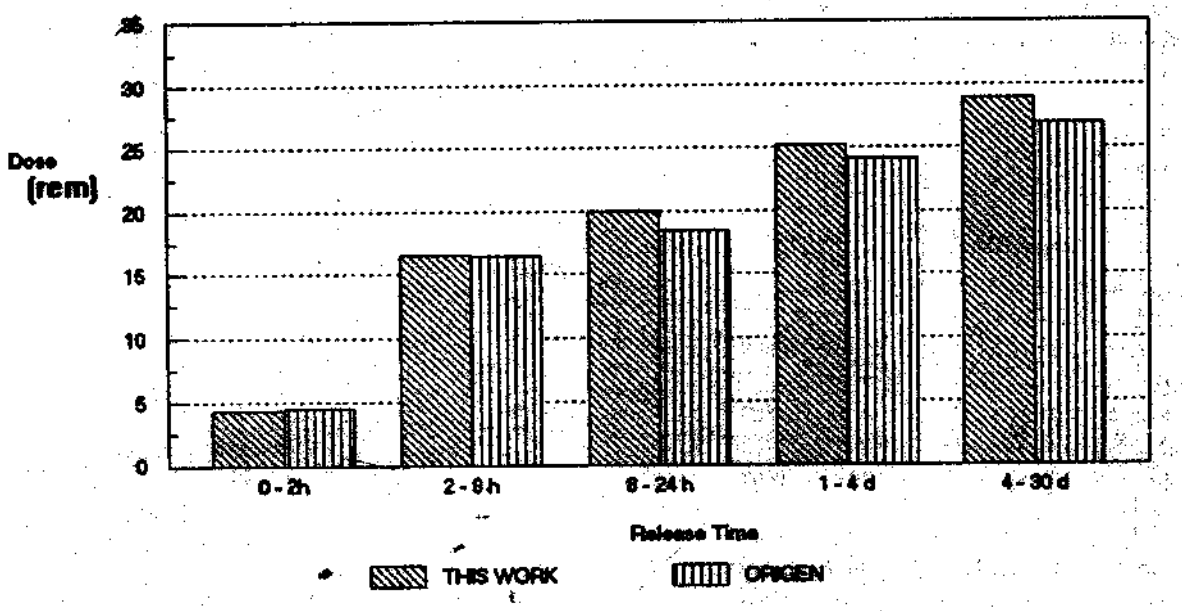

Fig. 2 Total Cumulative Doses for Iodine Isotopes (IAEA Reactor)

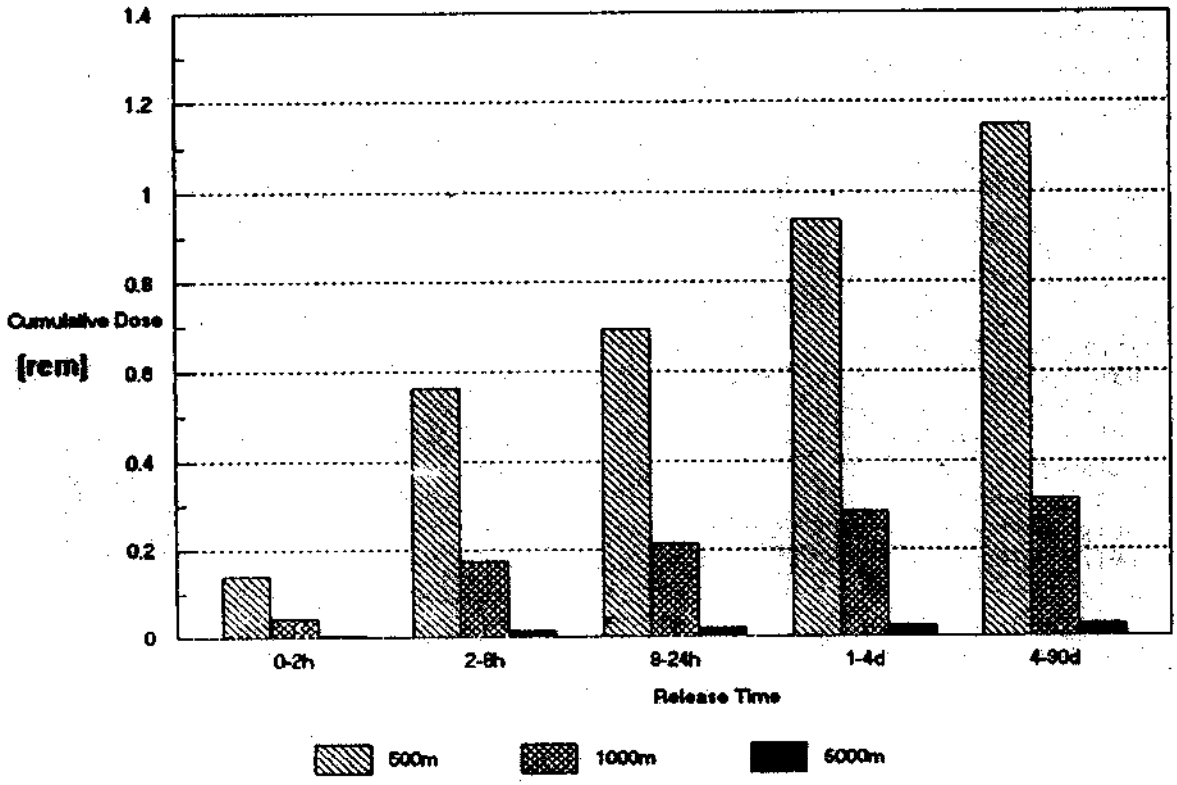

Fig.3 Variation of Dose with Release Time and Distance

The per cent isotopic contributions of the iodine isotopes as calculated by the ORIGEN-79[7] code are given in Figure 1. These results are compared with the per cent contributions of the iodines from the MNSR estimated with the code developed in this work. In both cases, the figure shows that I-131 contributes more than 60% of the thyroid dose. This is followed by I-133 with over 20%. The remaining three isotopes contribute less than 20% of the dose. The isotopic dose conversion factors (DCF) of I-131[5] plus the relatively large amount of release make it the most significant isotope with respect to thyroid dose.

Figure 1 also shows that the quantities of the isotopes released from the two reactors are comparable. This is because the fissile materials (U-235) used in both reactors are the same.

The total cumulative doses for the isotopes determined in this work and by ORIGEN-79 for the IAEA reactor are depicted in Fig. 2. The similarity of the results show that CORE3 code developed by the authors of this study give results which are comparable to those of other workers.

The iodine released in the time interval 0-8 hours is more than 50% of the total iodine released in 30 days (Fig. 3). This is due to the fact that most of the isotopes have short half-lives and after few hours, they decay faster than they are released from the reactor building.

Figure 3 also shows that the dose is significantly reduced at large distances. This is due to turbulent diffusion [3].

CONCLUSION

From the results the most important isotopes are I-131 and I-133 contributing respectively about 66% and 27%. The total cumulative dose after 2 hours and 5 days of release are 0.141rem and 1.148rem respectively. Both of these values are well below the 300rem guideline dose for the thyroid [5].

The worst possible accident that can occur in a research reactor is the whole core melting, and in this work, it is this scenario that has been considered. However, the doses computed can be scaled down depending on the fraction of core that is damaged. The failure of even part of fuel element is possible. For the MNSR, such a failure would release very insignificant activities to the atmosphere.

It was assumed in this work that the reactor core attained the highest power level at the end of 100 full power days (100 FPD) of irradiation. Different irradiation histories for example, 5-day, 8 hours per day for some number of days can be considered. An irradiation history like this will lead to even lower doses.

The data used were selected to give very conservative results. If site specific conditions were used, lower activity and does values may have been obtained.

The results show that the Miniature Neutron Source Reactor (MNSR) is very safe even in an accident situation and this is mainly due to its low power of 27kW.

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