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ORIGINAL PAPER

Source term estimation for the MARIA research reactor and model of atmospheric dispersion of radionuclides with dry deposition

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Abstract. Source term is the amount of radionuclide activity, measured in becquerels, released to the atmosphere from a nuclear reactor, together with the plume composition, over a specific period. It is the basis of radioprotection-related calculation. Usually, such computations are done using commercial codes; however, they are challenging to be used in the case of the MARIA reactor due to its unique construction. Consequently, there is a need to develop a method that will be able to deliver useful results despite the complicated geometry of the reactor site. Such an approach, based upon the Bateman balance equation, is presented in the article, together with the results of source term calculation for the MARIA reactor. Additionally, atmospheric dispersion of the radionuclides, analysed with the Gauss plume model with dry deposition, is presented.

Keywords: Research reactor • Radionuclide migration • Atmospheric dispersion • Radioactive cloud

Introduction

MARIA is a channels-in-pool type, water-cooled research reactor. It is located in the National Centre for Nuclear Research (NCBJ) in Otwock, Poland – roughly 25 km from Warszawa (the capital of Poland). The nominal power is 30 MWth, and the core contains 20+ fuel elements located in individually cooled pressure tubes called fuel channels. The channels are surrounded by beryllium blocks that act as moderators. All of the core elements are submerged in a 7-m-deep pool with light water [1].

When a hypothetical accident occurs, resulting in loss of fuel cladding integrity, radionuclides accumulated in the fuel elements during reactor operation are released. They start to migrate through the reactor's safety barriers, and after being weakened, are finally released into the atmosphere. The amount of radioactivity that is released is called "source term" and is the basis for the estimation of dose for the general public; it depends on the core inventory and the accident scenario; moreover, it is site specific and must be calculated for each research reactor separately [2].

Knowing the source term, the atmospheric dispersion of radionuclides can be calculated. It is modelled by the time-dependent advection-diffusion equation [3], which, when steady state conditions and overwhelming diffusion due to advection in the downwind direction are assumed, can be simplified to the Gauss plume model [4]. This approximation is widely used for radiological assessment purposes [5, 6].

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This article presents modelling of the source term created for the purpose of single fuel element failure due to loss-of-flow accident or large reactivity insertion as postulated initiating events. Loss-of-coolant accident (LOCA) might require a slightly different approach than that presented in this paper, as some of the reactor safety barriers are compromised during that event [7], and it is not discussed within the scope of this article. The calculated source term is combined with the atmospheric dispersion Gauss plume model that enables calculation of radionuclide concentration; eventually, the dose for the general population around the reactor is estimated.

Radionuclide migration routes through the safety barriers in the MARIA reactor

After single fuel element cladding failure, radionuclides that have been gathered in it start to be washed out to the loop of the channels. The majority of these settles on the filter or gathers in the pressurizer, but some leak through the seals to the reactor pool. From there, they migrate to the small volume of air above the reactor core, and they are released to the atmosphere through the ventilation system secured by filters. Additionally, some of them undergo sorption and desorption on the surfaces inside the confinement of the reactor. The scheme of radionuclide migration routes through the safety barriers of the MARIA reactor is presented in Fig. 1. Rectangles represent places where radionuclides gather, and arrows between them show the migration of radionuclides.

The reactor is scramed by the operator after an increase in the activity of the fuel element leak detection system (WNEP) or other previously obtained signals (e.g., power increase or flow decrease) [8]. The ventilation system is switched on in such a way

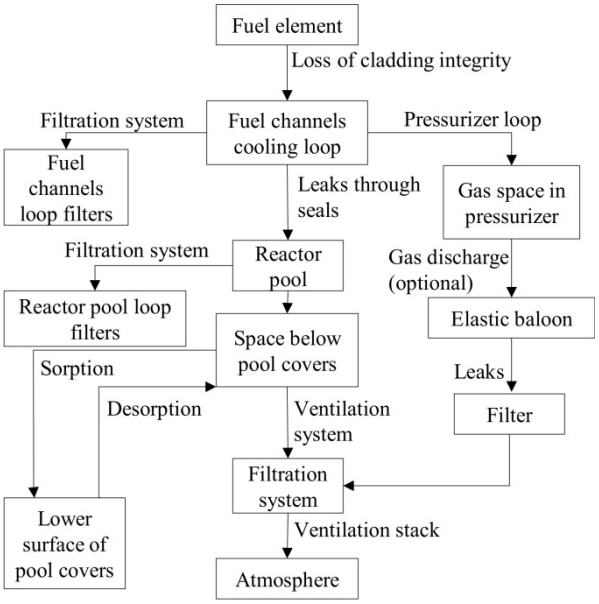

Fig. 1. Scheme of routes of radionuclide migration through the safety barriers (based on the MARIA reactor safety report [8]).

that the air is released into the atmosphere through the Vokes filtration system, the task of which is to weaken the amount of radionuclides in the atmosphere significantly.

The radionuclides in the filtration system are not further removed, but they undergo radioactive decay. The mixture in the gas space in the pressurizer can be discharged to the elastic balloon and later released through the intermediate filter to the ventilation system.

From the point of view of depleting the activity released to the atmosphere, the fuel channels and the pool, together with their filtration systems and ventilation filtration system, are essential elements protecting the general population from the effects of radiation exposure.

Mathematical model of radionuclide migration through the reactor's safety barriers

The mathematical model of activity in certain spots is based on the Bateman balance equations [9], individually for each of a number of radionuclides released from the nuclear fuel. The parameters used in the balance equations have been taken from the literature [10] or determined experimentally for the MARIA reactor [11]. The calculated activity is the input value for the calculations of atmospheric dispersion presented in the next section.

The mathematical equations for the activity balances are given below (Eqs. (1)–(10)). The definitions of the variables and constants that are used in the equations are given below each equation.

The activity balance in the cooling loops of the fuel channels is given by Eq. (1) [10], which combines the activity within the fuel element with the following: time constants of radioactive decay; radionuclide migration to the pressurizer and to the reactor pool; and the filtration efficiency:

$$\frac{dX_{\text{fcl}}}{dt} = Q \cdot \zeta \cdot f(t) - \left(\lambda + \lambda_{\text{press}} + \lambda_{pl} + \lambda_{\text{ffc}}\right) \cdot X_{\text{fcl}} \tag{1}$$

where X_{fcl} – activity in fuel channels' cooling loop [Bq], Q – the activity in the fuel element [Bq], ζ – release coefficient for each radionuclide [unitless], λ – time constant of radioactive decay [s^{-1}], λ_{press} – time constant of release to pressurizer gas space [s^{-1}], λ_{pl} – time constant of leakage to reactor pool [s^{-1}], λ_{ffc} – time constant of fuel channels' cooling loop filtration system [s^{-1}], and $f(t)$ is the time function of release of radionuclides, given by Eq. (2) [10]:

$$f(t) = \frac{1}{T_p} e^{-t/T_p} \tag{2}$$

where t – time [s], T_p – time constant of leaching of fission products from the fuel element [s].

Additionally, the function (2) is normalized to the value of "1", as described in Eq. (3) [10].

$$\int_0^\infty f(t)dt = 1 \tag{3}$$

The activity balance in the pressurizer combines the buildup of radionuclide activity in the pressurizer

gas space and the radioactive decay and is covered by Eq. (4) [10]:

$$\frac{dX_{\text{press}}}{dt} = \lambda_{\text{press}} + X_{\text{pcl}} - \lambda_{\text{ffc}} \cdot X_{\text{press}} \tag{4}$$

where X_{press} – activity in pressurizer gas space [Bq], X_{fcl} – activity in fuel channels' cooling loop [Bq], λ – time constant of radioactive decay [s^{-1}], λ_{press} – time constant of release to pressurizer gas space [s^{-1}], λ_{ffc} – time constant of fuel channels' cooling loop filtration [s^{-1}].

The activity balance in the fuel channels' cooling loop filtration system covers the radioactivity buildup in the filters and is given in Eq. (5) [10].

$$\frac{dX_{\text{ffc}}}{dt} = \lambda_{\text{ffc}} \cdot X_{\text{fcl}} - \lambda \cdot X_{\text{ffc}} \tag{5}$$

where X_{ffc} – activity in the fuel channels' filtration system [Bq], X_{fcl} – activity in fuel channels' cooling loop [Bq], λ – time constant of radioactive decay [s^{-1}], λ_{ffc} – time constant of fuel channels' cooling loop filtration system [s^{-1}].

The activity balance in the reactor pool combines the activity buildup due to the leakage from the fuel channels with the activity release to the air above the pool, radioactive decay, and efficiency of the filtration system and is covered by Eq. (6) [10].

$$\frac{dX_{\text{pool}}}{dt} = \lambda_{pl} \cdot X_{\text{fcl}} \cdot (1 - \kappa_w) - (\lambda + \lambda_{fp}) \cdot X_{\text{pool}} \tag{6}$$

where X_{pool} – activity in the reactor pool [Bq], X_{fcl} – activity in fuel channels' cooling loop [Bq], λ – time constant of radioactive decay [s^{-1}], λ_{pl} – time constant of leakage to reactor pool [s^{-1}], λ_{fp} – time constant of the pool cooling loop's filtration system [s^{-1}], κ_w – probability of the release of the nuclide from water to air [unitless].

The activity balance in the air above the reactor pool (below the covers) is the buildup due to the release from the reactor pool and depletion due to deposition and presence of the ventilation system, together with the radioactive decay, as described by Eq. (7) [10, 11].

$$\frac{dX_h}{dt} = \lambda_{pl} \cdot X_{\text{fcl}} \cdot \kappa_w - (\lambda + \lambda_{\text{dep}} + \lambda_v) \cdot X_h \tag{7}$$

where X_h – activity in the air above the reactor pool [Bq], X_{fcl} – activity in the fuel channels' cooling loop [Bq], κ_w – probability of the release of the nuclide from water to air [unitless], λ – time constant of radioactive decay [s^{-1}], λ_{pl} – time constant of leakage to reactor pool [s^{-1}], λ_{dep} – time constant of the deposition at the reactor pool cover's surface [s^{-1}], λ_v – time constant of the ventilation system [s^{-1}].

The activity balance at the bottom of the reactor covers the surface, combined with the deposition of radionuclides inside the reactor confinement and the radioactive decay, and is presented by Eq. (8) [10].

$$\frac{dX_{\text{dep}}}{dt} = \lambda_{\text{dep}} \cdot X_h - \lambda \cdot X_{\text{dep}} \tag{8}$$

where X_{dep} – activity at the reactor pool cover's surface [Bq], X_h – activity in the air above the reactor pool [Bq], λ – time constant of radioactive decay [s^{-1}], λ_{dep} – time constant of the deposition at the reactor pool cover's surface [s^{-1}].

The activity balance in the Vokes filtration system combines the filters' efficiency with activity in the air inside the reactor confinement; it is given by Eq. (9) [10].

$$\frac{dX_{fv}}{dt} = \lambda_v \cdot X_h \cdot \eta + \lambda_{\text{leb}} \cdot X_{eb} \cdot \eta(1 - \eta) - \lambda \cdot X_{fv} \tag{9}$$

where X_{fv} – activity in the Vokes filtration system [Bq], X_h – activity in the air above the reactor pool [Bq], X_{eb} – activity in the elastic balloon [Bq], λ – time constant of radioactive decay [s^{-1}], λ_v – time constant of the ventilation system [s^{-1}], λ_{leb} – time constant of the leakage from the elastic balloon [s^{-1}], η – filter's efficiency for each radioisotope [unitless].

The activity released to the atmosphere, or in other words, the "source term" is described by Eq. (10) [10]:

$$\frac{dX_{at}}{dt} = \lambda_v \cdot X_h \cdot (1 - \eta) + \lambda_{\text{leb}} \cdot X_{eb} \cdot (1 - \eta)^2 \tag{10}$$

where X_{at} – activity of the fission products released to the atmosphere [Bq], X_h – activity in the air above the reactor pool [Bq], X_{eb} – activity in the elastic balloon [Bq], λ_v – time constant of the ventilation system [s^{-1}], λ_{leb} – time constant of the leakage from the elastic balloon [s^{-1}], η – filter's efficiency for each radioisotope [unitless].

All the equations have the initial condition $X_i(t = 0) = 0$ as it is assumed that no radionuclides are present outside the fuel element before the loss of cladding integrity. This gives us the complete set of Eqs. (1)–(10), which enables the calculation of the source term. The parameters of radionuclides' transport, related to the presented equations, are presented in Table 1 [10].

Mathematical model of atmospheric dispersion

For calculation of the atmospheric dispersion, the Gauss plume model has been used, as described below. The concentrations were computed in the plume axis ($x = 0$) and at the ground level ($z = 0$). Plume is additionally depleted by radioactive decay and dry deposition of the radionuclides on the ground [12, 13]. The formula is given by Eq. (11) [12].

$$C_{(x,z=0)} = \frac{X_{at}}{2\pi \bar{u} \sigma_y \sigma_z} \cdot \exp\left(-\frac{y^2}{2\sigma_y^2}\right) \cdot \exp\left(-\frac{H^2}{2\sigma_z^2}\right) \cdot \Delta X_{at} \cdot f \tag{11}$$

where X_{at} – radionuclide concentration released to the atmosphere [Bq], $\bar{u}$ – mean wind velocity [m/s], σ_y – horizontal diffusion coefficient [m], σ_z – vertical diffusion coefficient [m], y – downwind distance from the emitter [m].

Here, f accounts for plume depletion due to radioactive decay [12].

$$f = \exp\left(-\lambda \frac{y}{\bar{u}}\right) \tag{12}$$

and ΔX_{at} is the so-called source depletion term, which accounts for dry deposition. It is given as in Eq. (13) [13]:

Table 1. Parameters of radionuclides' transport [10]

Nuclide	λ [s^{-1}]	x	κ_w	λ_{dep} [s^{-1}]	η
Br-83	8.1E-05	0.8	1.2E-02	3.6E-05	0.99
Kr-85m	3.1E-05	1	1.0E+00	0	0
Kr-85	2.1E-09	1	1.0E+00	0	0
Kr-87	1.5E-04	1	1.0E+00	0	0
Kr-88	6.8E-05	1	1.0E+00	0	0
Sr-89	1.6E-07	0.05	6.0E-02	3.6E-05	0.99
Sr-90	7.7E-10	0.05	6.0E-02	3.6E-05	0.99
Ru-103	2.0E-07	0.02	1.0E-04	3.6E-05	0.99
Ru-105	4.3E-05	0.02	1.0E-04	3.6E-05	0.99
Ru-106	2.2E-08	0.02	1.0E-04	3.6E-05	0.99
Te-131m	6.4E-06	0.1	1.0E-04	3.6E-05	0.99
Te-131	4.6E-04	0.1	1.0E-04	3.6E-05	0.99
Te-132	2.5E-06	0.1	1.0E-04	3.6E-05	0.99
Te-133m	2.1E-04	0.1	1.0E-04	3.6E-05	0.99
I-131	1.0E-06	0.8	1.2E-02	2.5E-04	0.96
I-132	8.1E-05	0.8	1.2E-02	2.5E-04	0.96
I-133	9.5E-06	0.8	1.2E-02	2.5E-04	0.96
I-134	2.2E-04	0.8	1.2E-02	2.5E-04	0.96
I-135	2.9E-05	0.8	1.2E-02	2.5E-04	0.96
Xe-131m	6.8E-07	1	1.0E+00	0	0
Xe-133m	3.6E-06	1	1.0E+00	0	0
Xe-133	1.5E-06	1	1.0E+00	0	0
Xe-135m	7.4E-04	1	1.0E+00	0	0
Xe-135	2.1E-05	1	1.0E+00	0	0
Cs-134	7.1E-08	0.5	3.0E-02	3.6E-05	0.99
Cs-137	7.3E-10	0.5	3.0E-02	3.6E-05	0.99

where λ [s^{-1}] – time constant of radioactive decay, λ_{dep} [s^{-1}] – time constant of the deposition at the reactor pool cover's surface, ξ [unitless]– release coefficient for each radionuclide, κ_w [unitless] – probability of the release of the nuclide from water to air, η [unitless] – filter efficiency for each radioisotope.

$$\Delta X_{at} = \exp\left\{-\sqrt{\frac{2}{\pi}\cdot\frac{V_T}{\bar{u}}}\cdot\frac{y}{\sigma_z}\exp\left(-\frac{H^2}{2\sigma_z^2}\right)\right\} \tag{13}$$

where V_T – dry deposition velocity [m/s], y – downwind distance from the emitter [m], $\bar{u}$ – mean wind velocity [m/s], σ_z – vertical diffusion coefficient [m]. H is the effective height of the emitter (stack) and is calculated with the Holland formula [11], as shown in Eq. (14) [14].

Where [14]

$$H = h + \Delta h \tag{14}$$

h – emitter height [m], Δh – plume rise [m], calculated as in Eqs. (15) and (16) [14], depending on the relation between the wind velocity and the exhaust gas velocity.

$$\Delta h = 0 \text{ for } v \le 0.5u_h \tag{15}$$

$$\Delta h = \frac{1.5\cdot v\cdot d}{u_h}\cdot\frac{v-u_h}{0.5u_h} \text{ for } v > 0.5u_h \tag{16}$$

where v – velocity of exhaust gas at the emitter outlet [m/s], u_h – wind velocity at the emitter outlet height [m/s].

The mean wind velocity $\bar{u}$ [m/s] used in Eqs. (11)–(13) is given as in Eqs. (17) and (18), depending on the height of the emitter [14]:

Table 2. Parameters for the calculation of coefficients of diffusion [14]

Pasquill stability class	A	D	F
m	0.080	0.270	0.440
a	0.888	0.818	0.756
b	1.284	0.822	0.551

A – most unstable, D – most probable, F – most stable.

$$\bar{u} = u_h \text{ for } h = H \tag{17}$$

$$\bar{u} = \frac{u_a}{(H-h)\cdot(1+\mathrm{m})\cdot 14^m} \text{ for } H \le 300\mathrm{m} \text{ and } h \ne H \tag{18}$$

where u_h – wind velocity at the emitter outlet height [m/s], h – emitter height [m], H – the effective height of the emitter [m], m – calculation factor [unitless] from Table 2.

Finally, horizontal and vertical coefficients of diffusion are calculated according to Eqs. (19)–(22) [14], with the parameters given in Table 2.

$$\sigma_z = \beta\cdot x^{\mathrm{b}} \tag{19}$$

$$\beta = 0.38\mathrm{m}^{1.3}\cdot\left[8.7-\ln\left(\frac{H}{z_0}\right)\right] \tag{20}$$

$$\sigma_y = \alpha\cdot x^{\mathrm{a}} \tag{21}$$

$$\alpha = 0.088\cdot\left[6\mathrm{m}^{-0.3}+1-\ln\left(\frac{H}{z_0}\right)\right] \tag{22}$$

where H – effective height of the emitter [m], z_0 – roughness parameter [m]; m, a, b – constants depending on Pasquill stability class, taken from Table 2.

The activity concentrations of the radionuclides deposited on the ground surface are calculated as in Eq. (23) [12].

$$C_{gr} = V_T\cdot C_{(x,z=0)} \tag{23}$$

where $C_{(x,z=0)}$ – activity concentration in the plume centerline at the ground level [Bq/m^3], V_T – dry deposition velocity [m/s] (0 for noble gases and 1000 m/d for all the other radionuclides [12]).

Activities of the radionuclides in the air on the ground level and the activity on the ground due to the dry deposition were calculated. The calculation concluded with the estimation of the doses from exposure to the external gamma field and inhalation of some of the isotopes.

Doses from exposure to the atmospheric radioactive plume

The external dose from immersion in the atmospheric discharge of the radionuclides is calculated as in Eq. (24) [12]:

$$E_{im} = DF_{im}\cdot C_{(x,z=0)} \tag{24}$$

where DF_{im} – effective dose coefficient for immersion [Sv/s per Bq/m^3], $C_{(x,z=0)}$ – radionuclide concentra-

tion in the plume centerline on the ground level [Bq/m^3].

Internal dose due to inhalation of iodine isotopes is calculated as shown in Eq. (25) [12]:

$$E_{\text{inh}} = DF_{\text{inh}} \cdot R_{\text{inh}} \cdot C_{(x,z=0)} \tag{25}$$

where DF_{inh} – inhalation dose coefficient [Sv/Bq], R_{inh} – the inhalation rate [m^3/s], $C_{(x,z=0)}$ – radionuclide concentration in the plume centerline on the ground level [Bq/m^3].

External dose from the ground deposits is calculated as in Eq. (26) [12]:

$$E_{\text{dep}} = DF_{gr} \cdot C_{gr} \cdot \int_0^T \exp(-\lambda_E t)\,dt = DF_{gr} \cdot C_{gr} \cdot \left[\frac{1-\exp(-\lambda_E t)}{\lambda}\right] \tag{26}$$

where DF_{gr} – dose coefficient for exposure to ground deposits [Sv/s per Bq/m^2], C_{gr} – activity concentration on the ground surface [Bq/m^2], λ_E – combination of the time constant of radioactive decay and the constant of washout of radionuclides from the ground [12], t – time [s], T – time of exposition to the radionuclides [s].

Source term calculation

For the calculation of the source term, the following scenario has been assumed:

1. After the loss of cladding integrity, the reactor is SCRAMed.
2. Five minutes after SCRAM, the main circulation pumps are switched to auxiliary pumps to limit the pressure in the fuel channels' cooling loop.
3. Ten minutes after SCRAM, ventilation flow is limited from 11000 to 6000 m^3/h.
4. Thirty minutes after SCRAM, all pumps are switched off.

According to Pytel *et al.* [8], this scenario can be considered to include the typical operator's actions that should be taken in the MARIA reactor after a nuclear accident.

It has been assumed that a single fuel element, which is the source of the radionuclides, is operated at a power of 1.8 MWth and had a burnup of 70 MWd (after eight reactor operation cycles) and total activity of $3.1 \cdot 10^4$ TBq. The activities of the radionuclides contained in the fuel element are presented in Table 3.

Using the mathematical model that combines Eqs. (1)–(10) with the assumptions of the accident scenario given above, calculation of the source term given in Table 3 is conducted by creating a computer using a MATLAB based code. The total activity that was released to the atmosphere through the reactor stack equals 3.1 TBq and contains mainly noble gases (Xe-133m, Kr-88, Xe-131m, Xe-135m, and Kr-87), which account for >98% of activity in the plume. From the radioprotection point of view, it is essential to mention that they do not contribute to the internal dose from the inhaled radionuclides.

Table 3. Activity contained in the fuel element with burnup of 70 MWd, operating continuously at 1.8 MWth power level [8]

Nuclide	Activity [Bq]	Nuclide	Activity [Bq]
Br-83	2.8E+14	Te-133m	2.8E+15
Kr-85m	6.8E+14	I-131	1.1E+15
Kr-85	1.3E+12	I-132	1.9E+15
Kr-87	1.4E+15	I-133	3.5E+15
Kr-88	2.0E+15	I-134	4.3E+15
Sr-89	9.7E+14	I-135	3.4E+15
Sr-90	8.4E+12	Xe-131m	8.1E+12
Ru-103	7.4E+14	Xe-133m	6.4E+13
Ru-105	5.6E+14	Xe-133	2.4E+15
Ru-106	1.7E+13	Xe-135m	1.0E+15
Te-131m	2.3E+14	Xe-135	2.1E+14
Te-131	1.4E+15	Cs-134	1.7E+12
Te-132	1.8E+15	Cs-137	8.5E+12
Total	3.1E+16		

Table 4. Activity of radionuclides released to the atmosphere (source term)

Nuclide	Activity [Bq]	Nuclide	Activity [Bq]
Br-83	1.2E+07	Te-133m	2.2E+04
Kr-85m	6.5E+08	I-131	3.0E+06
Kr-85	3.3E+09	I-132	1.6E+08
Kr-87	3.9E+11	I-133	1.45E+06
Kr-88	7.1E+11	I-134	2.1E+08
Sr-89	4.7E+07	I-135	3.9E+07
Sr-90	5.8E+07	Xe-131m	7.1E+11
Ru-103	3.1E+04	Xe-133m	1.1E+12
Ru-105	1.2E+01	Xe-133	3.2E+09
Ru-106	6.2E+03	Xe-135m	6.2E+11
Te-131m	6.4E+04	Xe-135	2.5E+10
Te-131	6.8E+04	Cs-134	1.4E+07
Te-132	3.4E+04	Cs-137	6.8E+07
Total	3.6E+12		

The source term can be used for calculation of atmospheric dispersion, which is necessary for estimation of the accurate dose; the steps are described in the next section of the article. Table 4 presents the activities and composition of the plume of radionuclides released to the atmosphere.

Doses for the general public

For calculation of the doses for the general population, it has been assumed that the representative of the critical group (a child <1-year-old) is standing still during the time when the plume is passing by. Calculation of 1-year-dose from the radioactive deposits assumes that this person is outside of the building for the whole year. Additionally, the representative of the critical group is not shielded by any kinds of constructions (i.e., house walls). Doses are calculated in the cloud axis at the ground level, as presented in the section "Mathematical model of atmospheric dispersion" of this article.

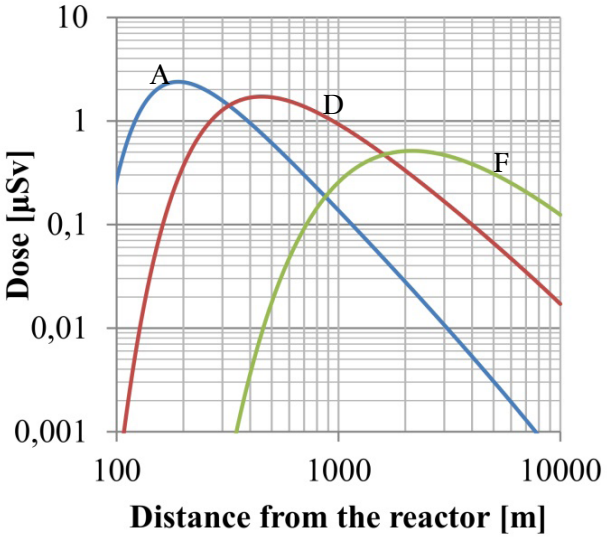

Fig. 2. External dose from the immersion of the atmospheric discharge. A – most unstable, D – most probable, F – most stable.

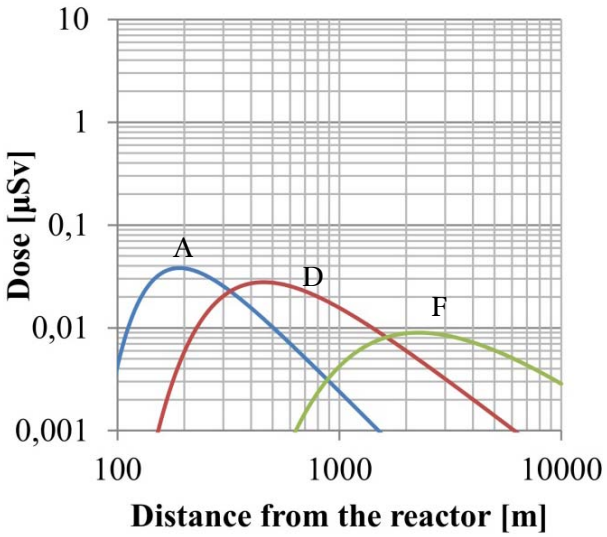

Fig. 3. Internal dose due to radionuclide inhalation. A – most unstable, D – most probable, F – most stable.

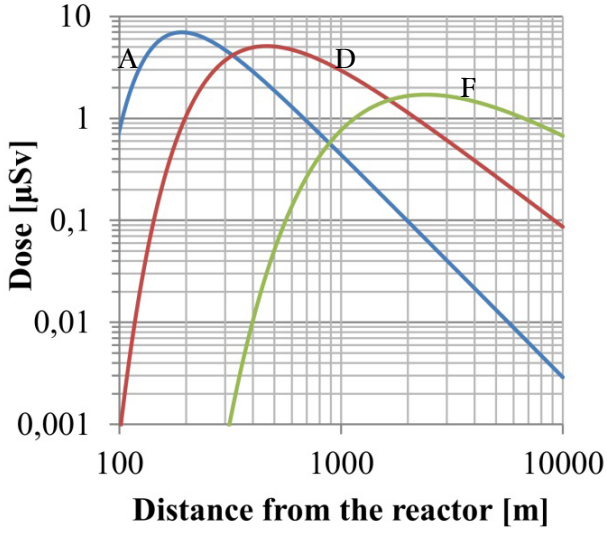

Fig. 4. External, yearly dose from the ground deposits. A – most unstable, D – most probable, F – most stable.

Three atmospheric situations have been assumed – most unstable, most probable, and most stable (A, D, and F Pasquill stability classes, respectively) [15]. For each category, the lowest possible wind speed of $u = 1$ m/s was taken into account.

For the calculation of the sufficient emitter height, the following parameter values have been assumed: emitter height $h = 60$ m, the gas velocity at the emitter outlet $v = 3.3$ m/s, emitter diameter $d = 2$ m, surface roughness $z_0 = 0.85$ m [8, 16]. The inhalation rate of the infant has been assumed to equal $R_{\text{inh}} = 4.44 \cdot 10^{-5}$ m^3/s [12]. The values of the dose coefficients for the radionuclides present in the plume, i.e., DF_{im}, and DF_{gr}, have been taken from Eckerman and Ryman [17], and the DF_{inh} values come from the Regulation of 18 January 2005 of the Council of Ministers on ionizing radiation dose limits [18].

Figures 2–4 present the dose from immersion, the dose from inhalation, and yearly dose from the ground deposits, respectively.

Discussion and conclusions

The source term of the MARIA reactor has been calculated, as follows: from the stack to the atmosphere, 3.6 TBq of radionuclides are discharged, containing mainly Xe-133m (30.7%), Kr-88 (20.0%), Xe-131m (19.8%), Xe-135m (17.5%), and Kr-87 (11.0%). This amount, compared with the initial $3.1 \cdot 10^4$ TBq that is contained within the fuel element, means that the safety barriers of the MARIA reactor reduce the activity of the radionuclides by a factor of 10^4.

The total maximum dose for a representative of the critical group (a child <1-year-old) from among the general population, considering all three routes of exposure, calculated for the whole calendar year after the accident is 2.5 µSv at the distance roughly 200 m from the reactor stack and 95% of this comes from the immersion in the cloud of radionuclides.

The dose calculation presented above has been conducted under the pessimistic approach, assuming that representatives of the groups exposed to radiation are standing still in one place for the whole time of the simulation. The hot spot present 200 m from the reactor stack is within the fenced area of the National Centre for Nuclear Research, accessible only for the employees working for 8 h/d, limiting the dose by a factor of three (not accounted for in the dose calculation).

The calculated dose is not only lower than the emergency limits described in Regulation of 27 April 2004 of the Council of Ministers on intervention levels for various intervention measures and criteria for cancelling intervention measures [19] (10 mSv for staying inside the buildings) but also way <1 mSv, which is the yearly limit for the general population under normal operating conditions of the reactor [18]. The dose from ingestion of crops, milk, and meat has not been calculated due to the lack of agricultural production in the areas around the reactor [8].

All of the remarks noted above state that the MARIA reactor has an effective system of reduction of the radionuclides produced in the core, which

actually protects the general public from radiation hazard in the case of a nuclear accident.

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