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ESTIMATION OF THE RESEARCH LIGHT WATER REACTOR RELEASE INFLUENCE ON THE POPULATION EXPOSURE

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Приводятся предварительные результаты расчетов доз облучения населения при нормальной эксплуатации и в случае гипотетической проектной аварии исследовательского водо-водяного реактора. Мощность реактора принята равной 10 МВт (т). Высота трубы равна 100 м. Предполагается, что площадка реактора расположена в 15 км от г. Исфахан (Иран). Оценены радионуклидные составы выбросов при нормальной эксплуатации реактора и в случае аварии. Коллективные дозы для наиболее крупных населенных пунктов в окрестностях реактора лежат в диапазоне от $2.2 . 10^{-6}$ чел.Зв/год (г. Фалаварджан) до $1.5 . 10^{-4}$ чел.Зв/год (г. Исфахан). Средняя индивидуальная доза облучения в данном случае оказалась равной примерно $1.0 . 10^{-11}$ Зв/год, что не превышает 0.0005% дозы внешнего облучения за счет естественных радионуклидов и «глобальных» радиоактивных выпадений. Годовые выпадения ^{137}Cs составляют 0.02-0.2 мБк/м^2 и не превышают 10^{-7} доли запаса радиоцезия в почве, накопленного после атмосферных ядерных испытаний.

INTRODUCTION

In this paper a preliminary calculational estimation of the possible influence of a light water research reactor release on the population and environment in the vicinity of the reactor site, which is supposed to be located near Isfahan city (Iran), has been done. A uranium fuelled reactor with light water as moderator and coolant is considered. The reactor is supplied with the containment (the concrete protective envelope with the inner stainless steel lining). Its power is adopted equal to 10 MW (th). The calculations of the atmospheric transport of the radionuclides released from this reactor are performed using Gaussian atmospheric transport model [1, 2]. For these calculations the dose coefficients from [3] are used. The annual collective and average individual irradiation doses for the people living in the big cities near this reactor site are calculated. The environment radioactive contamination levels connected with the radioactive fall-out are estimated also.

ESTIMATED RADIOACTIVE RELEASES DURING NORMAL REACTOR OPERATION

In the calculation procedure of radioactive releases during normal reactor operation the data on radioactive releases of the light water power reactors [4, 5] has been used. The assumption of radioactive equilibrium in the reactor core has been adopted. So the number of the different radionuclides formed in the core is supposed to be equal to the number of the decaying radionuclides. The possible accumulation and release of the different radionuclides for this calculation were estimated from the data for a typical light water power reactor taking into account the difference of the thermal power of these reactors.

Table 1 lists the nuclides, which may be released from this reactor during normal operation. The estimations of the fission product activities and of the possibility of releasing of these products into the environment demonstrates that the main contributions to the released activity and to the population irradiation are due to the following radionuclides:

• the fission products ^{133}Xe, ^{135}Xe, ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I, ^{134}Cs, ^{137}Cs, ^{89}Sr, ^{90}Sr;

• the nuclides formed by neutron capture from stainless steel corrosion products ^{51}Cr, ^{54}Mn, ^{58},Co, ^{60}Co.

The release values presented in Table 1 correspond to the daily leakage equal to 1% of the protective pillow gas and 5 l of the water from the first cooling loop.

Table 1

Annual release of different radionuclides

Nuclide	$T_{1/2}$, hr	Release, GBq/yr
Xe-133	125.9	38
Xe-135	9.08	38
Kr-85m	4.48	4.8
Kr-87	1.27	12.8
Kr-88	2.84	22
I-131	193.0	0.006
I-132	2.3	0.019
I-133	20.8	0.01
I-134	0.876	0.4
I-135	6.61	0.019
Cr-51	665.5	0.008
Mn-54	7496	0.00004
Co-58	1699	0.018
Co-60	46183	0.04
Sr-89	1212	0.0007
Sr-90	254916	0.0007
Cs-134	18063	0.0005
Cs-137	264324	0.00095

ESTIMATED RELEASE FOR THE HYPOTHETICAL ACCIDENT

The main features of the design accident were adopted in accordance to USA Regulatory Guide 1.183 [6].

For the estimation of the release of radionuclides to the environment in an accidental case, the following hypothetical accident scenario is considered

• loss of envelope sealing (clad failure) for 50% of the reactor fuel elements;

• release into the containment of all fission products contained in the broken envelopes;

• coolant and moderator are evaporated and the inner containment space (6.10^4 m^3 approximately) is filled with the saturated water vapor;

• 20% of the containment contents is instantly released into the atmosphere.

The reasons and the events leading to this scenario are not discussed here.

The temperature inside the containment after this accident is adopted equal to 30^oC and the saturated vapor density is equal to 0.04 kg/m^3 (see [7]). So the total quantity of the water vapor inside the containment is approximately equal to 2.5 t.

The release of the radionuclides for this hypothetical accident is presented in Table 2.

Table 2

Release of radionuclides for the hypothetical accident

Nuclide	$T_{1/2}$, hr	Release activity, TBq
I-131	193.0	2.9
Xe-133	125.9	18
Xe-135	9.08	1

POPULATION EXPOSURE TYPES RESULTING FROM RADIOACTIVE RELEASES

In the process of population exposure estimation, the real geographical population distribution around the reactor site and its age composition are taken into account. It is supposed that radionuclide intake via the food chain (alimentary path) is negligible because of the small fraction of the local food manufacturing and consumption. Therefore for the normal reactor operation and accidental condition the following doses are calculated:

- annual dose of external total-body irradiation from the radionuclides contained in the atmosphere (from the radioactive cloud);
- annual equivalent doses for the internal irradiation from the radionuclides inhaled with the atmospheric air;
- annual dose of external total-body irradiation from the radionuclides contained in the radioactive fall-out on the soil surface.

Here only the annual dose values are calculated. However, for the obtaining of pessimistic estimation, during the calculation of the radioactive fall-out level for the first year of the reactor operation the equilibrium activities of all nuclides, which can be accumulated in the reactor, were used.

The doses from the radionuclides existing in the atmosphere after resuspending from the soil by the wind are neglected. The possible dose decreasing connected with the radionuclide migration in the soil depth is not considered too.

The dose coefficients from [1,3] are used in the calculations. Since the preliminary dose estimation demonstrates that the equivalent dose values are relatively low, therefore detailed calculations of the equivalent doses for selected organs of people from different age groups were not performed and only the values of the total equivalent doses were calculated.

During estimation of the population irradiation dose for normal reactor operation, the source is assumed as constant and reactor power will not be changed with time. The radioactive cloud is considered as semi-infinite irradiation source. The radionuclide migration in soil depth is adopted as negligible and infinite plane source is used for calculation of the irradiation source related to radioactive fall-out.

CALCULATION MODEL, MAIN FORMULA AND PARAMETERS

The concentration Cn,i (i-th radionuclide air concentration near soil surface, Bq/m^3) and An,i (i-th radionuclide surface density, Bq/m^2) necessary for the calculation of population exposure dose are calculated using Gaussian model of the atmospheric transport (for detail description see [1,2]). The influence of meteorological conditions on the radionuclide atmospheric dispersion has been accounted using the Smith-Hosker parameterization [1]. In the calculations of the population irradiation during the normal reactor operation the data on the occurrence of the Pasquill atmospheric stability classes and the wind directions were used for the meteorological post located near the supposed reactor site. The depletion of the radioactive cloud connected with the dry deposition and wash-out by the precipitation and radioactive decay during the transport period from the source to the point where the dose value is calculated is token into account [1].

In the estimations of the accidental irradiation the calculations were performed for the atmospheric stability class D (neutral stability).

The external irradiation equivalent dose rate from i-th radionuclide is calculated by the formula:

$$D_{ex.ai} = K_e * C_{n.i} * B_{a\gamma i},$$

where K_e – shielding factor accounting the decreasing of the external irradiation dose by the building walls where the persons live. This factor is adopted as equal to 0.7 for all calculations; $C_{n.i}$ – near-surface concentration of i-th radionuclide activity during the cloud transport above the given point, Bq/m^3 (averaged for 10-min period); $B_{a\gamma i}$ - corresponding dose coefficient reducing the activity value (the air concentration, soil radioactive contamination density) to the equivalent dose value.

Then the equivalent doses calculated for the separate radionuclides are summed

$$D_{ex.a} = \sum_i D_{ex.ai}.$$

The equivalent dose rate related to inhaling of the i-th radionuclide for the person from the j-th age group is calculated with the formula

$$D_{ing.aci} = C_{n.i} * B_{ai} * v_j,$$

where v_j – annual volume of inhaled air for person from j-th age group; B_{ai} – dose coefficient.

Then the equivalent doses calculated for the separate radionuclides are summed.

The external irradiation dose rate connected with the radionuclide deposited on the soil surface is calculated with the formula

$$D_{ex.fi} = K_e * A_{n.i} * B_{s\gamma I},$$

where $B_{s\gamma i}$ – corresponding dose coefficient; $A_{n.i}$ – the soil surface contamination density for i-th radionuclide.

Then the equivalent doses calculated for the separate radionuclides are summed.

The collective annual dose is calculated using the real population distribution around the supposed reactor site.

The analogous formula is used during the calculation of the irradiation dose connected with the accidental radioactive release. The integration period (the transport time of the radioactive cloud above the given point) is adopted equal to 0.3 hr (the approximate transport period for the largest inhabitant point near reactor site – Isfahan City). The time period for the external irradiation dose calculation is adopted equal to 1 year.

The “dry deposition velocity” values [1] are adopted equal to zero for all the gases, 2 cm/s for iodine radioisotopes and 1 cm/s for other radionuclides.

Calculation results. Population irradiation dose values

Figures 1 and 2 and Tables 3-6 demonstrate some of the calculation results obtained in the process of the population irradiation estimation.

It should be noted that the annual dose of the external irradiation from the natural radionuclides and from the so-called “global” radioactive fall-out after the atmospheric nuclear tests is equal to 1.3 mSv/yr.

The data presented on the Fig.2 shows that during normal reactor operation the yearly ^{137}Cs activity deposited to the soil surface is equal to 0.02-0.2 mBq/m^2. Now the “global” radioactive soil contamination level is equal to 1500-2000 Bq/m^2. Therefore the additional radioactive environment contamination connected with this reactor operation can be accounted as negligible.

Table 3

Average individual and collective doses for Isfahan City

Irradiation type	Age group, years					
	1	1-2	2-7	7-12	12-17	>17
Cloud, external irradiation	0.16e-10	0.16e-10	0.16e-10	0.16e-10	0.16e-10	0.16e-10
Inhalation, internal irradiation	0.69e-12	0.31e-11	0.12e-10	0.12e-10	0.13e-10	0.16e-10
Fall-out, external irradiation	0.78e-7	0.78e-10	0.78e-10	0.78e-10	0.78e-10	0.78e-10
Equivalent dose rate	0.95e-10	0.95e-10	1.1e-9	1.1e-9	1.1e-9	1.2e-9
Population	24200	66000	330000	302700	302700	605400

Collective dose rate: 1.51e-4 Man*Sv/yr
Average individual dose rate: 0.9e-10 Sv/yr

Table 4

Average individual and collective doses for Falavarjan City

Irradiation type	Age group, years					
	1	1-2	2-7	7-12	12-17	>17
Cloud, external irradiation	0.27e-11	0.27e-11	0.27e-11	0.27e-11	0.27e-11	0.27e-11
Inhalation, internal irradiation	0.12e-12	0.52e-12	0.19e-11	0.22e-11	0.22e-11	0.27e-11
Fall-out, external irradiation	0.16e-10	0.16e-10	0.16e-10	0.16e-10	0.16e-10	0.16e-10
Equivalent dose rate	0.16e-10	0.16e-10	0.18e-10	0.18e-10	0.18e-10	0.18e-10
Population	1800	4930	24650	22700	22700	45400

Collective dose rate: 2.2e-6 Man*Sv/yr
Average individual dose rate: 1.8e-11 Sv/yr

Table 5

Average individual and collective doses for Najafabad City

Irradiation type	Age group, years					
	1	1-2	2-7	7-12	12-17	>17
Cloud, external irradiation	0.12e-11	0.12e-11	0.12e-11	0.12e-11	0.12e-11	0.12e-11
Inhalation, internal irradiation	0.53e-13	0.23e-12	0.85e-12	0.82e-12	0.96e-12	0.12e-11
Fall-out, external irradiation	0.64e-11	0.64e-11	0.64e-11	0.64e-11	0.64e-11	0.64e-11
Equivalent dose rate	0.77e-11	0.78e-11	0.85e-11	0.84e-11	0.86e-11	0.88e-11
Population	4700	12800	64000	58700	58700	117400

Collective dose rate: 2.7e-6 Man*Sv/yr
Average individual dose rate: 0.9e-11 Sv/yr

Table 6

Average individual and collective doses for Khomeynishahr City

Irradiation type	Age group, years					
	1	1-2	2-7	7-12	12-17	>17
Cloud, external irradiation	0.50e-11	0.50e-11	0.50e-11	0.50e-11	0.50e-11	0.50e-11
Inhalation, internal irradiation	0.22e-12	0.97e-12	0.36e-11	0.35e-11	0.41e-11	0.50e-11
Fall-out, external irradiation	0.27e-10	0.27e-10	0.27e-10	0.27e-10	0.27e-10	0.27e-10
Equivalent dose rate	0.27e-10	0.33e-10	0.31e-10	0.31e-10	0.31e-10	0.32e-10
Population	3700	10000	60000	45800	45800	91600

Collective dose rate: 7.6e-6 Man*Sv/yr
Average individual dose rate: 3.2e-11 Sv/yr

After the hypothetical accident the radioactive releases do not lead to any significant population exposure and environment radioactive contamination.

CONCLUSION

Reviewing of the estimations performed concerning to the possible population exposure and radioactive environment contamination during the normal operation of the light water research reactor near Isfahan city allow us to do the following conclusions:

- The collective dose values for the largest cities located near this reactor are in the range from 2.2e-6 Man*Sv/yr (Falavarjan city) to 1.5e-4 Man*Sv/yr (Isfahan city). The average annual individual dose values for this region are equal to 1.0e-11 Sv/yr and are not higher than 0.0005% of the external irradiation dose connected with the natural radionuclides and with the “global” radioactive environment contamination. These dose values are much lower for example than the values adopted in Russian radiation safety rules [8];

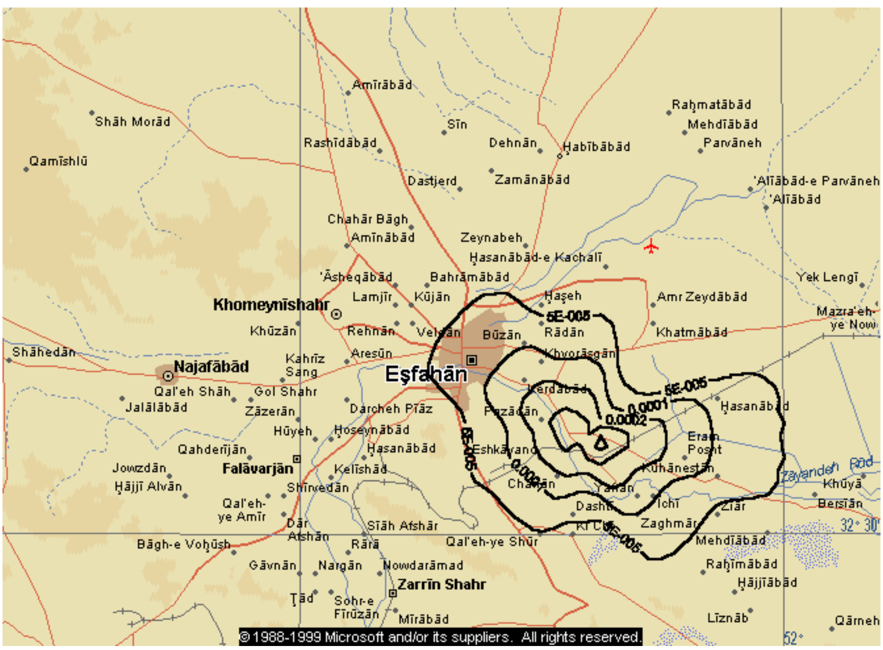

Fig. 1. Air concentration of total gamma-activity (averaged for one-year period, Bq/m^3)

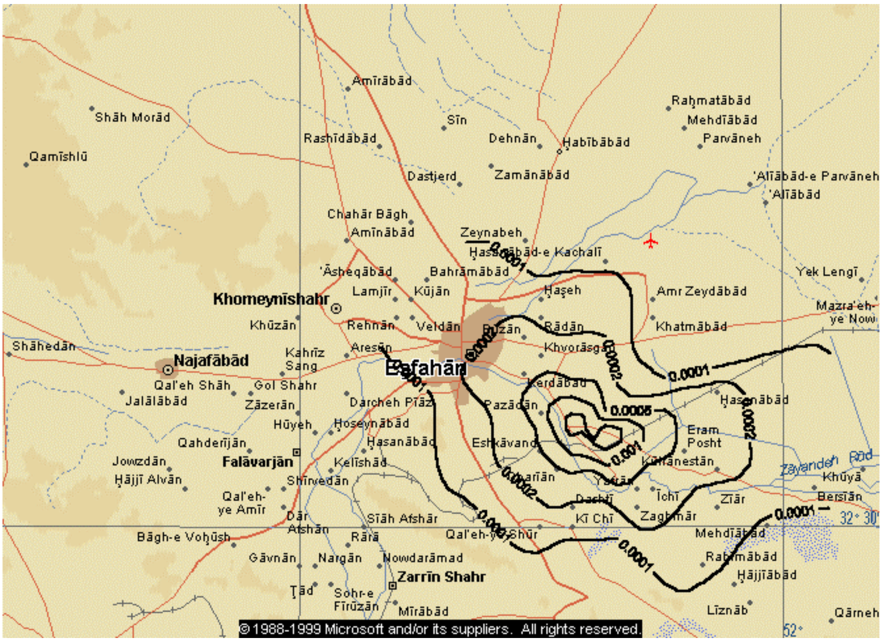

Fig. 2. One year deposition density of Cs-137 (Bq/sq.m*y)

- The normal reactor operation does not lead to the significant additional radioactive environment contamination in the vicinity of this reactor. So annual ^{137}Cs fallout is in the range of 0.02-0.2 mBq/m^2, which is 10^{-7}-th fraction of the accumulated dose in the soil activity after the atmospheric nuclear tests performed before 1962;
- The population irradiation dose values after hypothetical accident are negligible. The concentration of the elevated air radionuclide connected with the radioactive cloud transport can exist in the inhabitant points in the reactor vicinity during 20-30 min.

In spite of the relatively low population irradiation dose values in the reactor vicinity, It is recommended to perform a more detailed calculation for different unfavorable irradiation consequences (e. g. the malformations of the different types).

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of fine - film multilayer structures (hundred thousands of layers) which consist of fissile and structural materials. The conversion efficiency is up to 30 %.

In the paper consideration is given to a cell - converter of uranium fission fragments energy into electric energy, based on which the EGE has been designed. Besides, one of the options of «cold» reactor - converter is described as well as some calculational and experimental studies are indicated. They are required for the further development of design and for perfection of methods for calculation of proposed reactor -converter parameters.

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Procedure of Sensitivity and Uncertainty Estimation for the box Model of Pollution Transport by Sea\ A.N. Ershov, D.A. Kamaev, O.V. Shershakov; Editorial board of Journal "Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetica" (Communications of Higher Schools. Nuclear Power Engineering) - Obninsk, 2001. - 7 pages, 6 illustrations.- References, 6 titles.

The present work is devoted to elaboration of the procedure of sensitivity and uncertainty estimation for the box model of pollution transport by sea. For sensitivity estimation of the model relatively to small perturbations of input data and parameters, the transition to conjugated system of equations have been used. The problem of uncertainty estimation of modeling, caused by inexact knowledge of model parameters and input data. This problem removes to laborious problem of global optimization. In respect to the box model such approach is inapplicable, because of large dimension of parameters space. There is shown the procedure of finding conservative estimates of uncertainty in the work, based on interval mathematics. There are demonstrated the results of computations, illustrating the efficiency of suggested procedure of uncertainty estimation.

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The Modeling of the Radionuclide Transportation in Reservoirs Located in the Head Part of the East Ural Radioactive Trace\ P.M.Stukalov, A.I.Smagin; Editorial board of Journal "Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetica" (Communications of Higher Schools. Nuclear Power Engineering) - Obninsk, 2001. - 8 pages, 3 illustrations.- References, 7 titles.

It is presented the results of modeling for the radioactive contamination dynamics of the reservoirs located in the East Ural trace head part. The satisfactory comparison of experimental data and calculated results is shown.

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Estimation of the research light water reactor release influence on the population exposure\ M. Moniri, V.E. Cherkashin; Editorial board of Journal "Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetica" (Communications of Higher Schools. Nuclear Power Engineering) - Obninsk, 2001. - 7 pages, 2 illustrations, 6 tables.- References, 8 titles.

The preliminary results of the calculation of the population irradiation dose values are presented during the normal operation and the hypothetical design accident of the light water research reactor. The thermal reactor power was adopted equal to 10 MW. The stack height is equal to 100 m. The supposed reactor site is located near Isfahan city (Iran). The radionuclide composition of the reactor for the different situations are estimated. The collective dose values for the largest cities located near this reactor are in the range from $2.17.10^{-6}$ man*Sv/yr (Falavarjan city) to $1.45.10^{-4}$ man*Sv/yr (Isfahan city). The average annual individual dose value for this region is equal to $1.0.10^{-11}$ Sv/yr approximately and it is not higher than 0.0005% of the external irradiation dose connected with the natural radionuclides and with the "global" radioactive environment contamination. Annual ^{137}Cs fallout is in the range of 0.02-0.2 mBq/m^2, which is equal to 10^{-7}-th fraction of the accumulated soil activity after the atmospheric nuclear tests.

УДК 621.039.51

On Some Modifications of the Point Reactor Kinetics Equations\ B.D.Abramov; Editorial board of Journal "Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetica" (Communications of Higher