BORE-HOLE DISPOSAL OF SEALED RADIATION SOURCES: SAFETY ANALYSIS

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ABSTRACT

Shallow ground bore-hole repositories (BHR) are used for storage and disposal of spent sealed radiation sources (SRS). A typical shallow ground BHR has a 200 l stainless steel underground vessel in a concrete steel enforced well as SRS reservoir. The BHR were designed for disposal of sources with total radioactivity up to 50 kg-eqv. of radium per one repository. In the middle of 80-s the BHR disposal technology was corrected by addition of a supplementary SRS immobilisation technological process [1]. Immobilisation procedure is carried out directly in the underground vessel by applying metal matrices (mainly lead). SRS with total radioactivity higher than one million Ci are disposed of in Russian Federation, practically all these SRS being immobilised in metal matrices at regional disposal facilities. Herein safety analysis of SRS disposal is carried out. Probabilistic calculations taken into account some data uncertainties and variability. They showed that practically there is no any release of short lived radionuclides into environment during about 1000 years. This is completely due to very low corrosion rate of lead matrix. Various models were applied for more detail numeric simulation of temperature and radiation fields and transport of radionuclides. Super-conservative scenarios were chosen for these models. The worst case comprises both breaching of all engineer barriers and flooding of disposal site plus eventual sources partly out of matrix. Maximum dose was found to be not higher than 55 – 75 μSv/year for population. Even this super-conservative estimation gives data much below than safety criterion 10 mSv/year.

INTRODUCTION

Practice of operation of shallow ground BHR for SRS and inspection of their status have shown the possibility of accelerated corrosion of engineering barriers in ionisation radiation fields [2]. To increase the safety of BHR disposal a technological scheme providing immobilisation of SRS in metallic matrix directly in the BHR was developed. This scheme permitted to use the conventional scheme of SRS disposal. Immobilisation of SRS provides an additional barrier with a metal layer between the sources and environment, which increases the disposal safety [3].

After the decay of radionuclides will occur (this will require about 500 - 1000 years) the metallic block with SRS will be removed and melted for matrix material reutilization.

To estimate the efficiency of SRS immobilisation technology in BHR one should perform the evaluation of long term safety of the repository on the basis of the most possible scenarios of events.

ANALYSIS

For modelling we used a number of computer codes which comprise the models for each scenarios of possible release of nuclides from the disposal site and migration into environment. We considered many possible scenarios, among them the most important:

- Lateral spreading of activity;
- Migration of activity in perched water table due to flow into repository;
- Migration of activity in perched water table due to flow into repository and heat convection caused by heating of repository;
- Migration of activity due to inhomogeneity of filtration factor when flooding repository;
- Migration of activity due to heat convection caused by heating of repository when flooding it;
- Migration of activity due to capillary uplift of ground water.

In order to assess the safety we adopted the conservative model of event scenarios. We assumed (despite this is not the case) that the SRS embedded in lead can contact with stainless steel BHR walls, which can be destroyed due to physical-chemical reactions.

The calculation of population dose for evaluating the safety was made with GENII code [4] on the basis of recommendations given in 26, 30 and 60 publications ICRP and Hanford models of Environmental.

PARAMETERS

The modelling process suggests that engineering barriers made of concrete, carbon steel and stainless steel cannot be corroded and destroyed earlier than 77 years. After this time corrosion of matrix and sources cases begins. The corrosion rate of matrix is $5x10^{-6}$ m per year. The corrosion is accompanied by the migration of radionuclides into underground waters. SRS cases are made of stainless steel. We assumed that SRS contain radionuclides soluble salts. We assumed also that the distribution factor for metal oxides is 5, which mean only 20 % of activity release into environment. The radionuclide inventory was taken as 90% of Co-60 and 10% of Cs-137. The total radioactivity immediately after disposal is 180.000 Ci.

The heat conductivity of host rocks is 2,8 W/(m^2 ^{0}C) and is used in calculations for all rocks and engineering barriers. Heat conductivity of lead matrix is 35 W/(m^2 ^{0}C) [1,2].

The volume heat capacity of host rock, matrix and barriers are supposed constant and equal to 1200 kJ/(m^3 ^{0}C). We assume that the water content of rock is equal to common porosity, e.g. 0,3.

The distribution coefficient for surface loamy soils was assumed 200. For Moscow moraine clay the distribution coefficient is 2000, for moraine sediments is 530. The distribution coefficient for concrete corrosion products and soil filling of bore-hole was taken 1,5. The Co-60 produces complex anions, which practically are not sorbed in the soil. Therefore we considered two cases. In the first case the distribution coefficient K_d of unfixed Co-60 was zero, whereas in the second case the distribution coefficient K_d of fixed Co-60 was 20. Last data are used for all rocks and engineer barriers.

The filtration coefficient for surface loamy soil was taken from conservative data as 0,1 m/day, for moraines sediments - as 1 m/day, for broken soil and products of corrosion - as 0,5 m/day. The lead matrix is assumed impenetrable. For Moscow moraine clay the filtration factor is 10^{-4} m/day.

We assume that in the moment of time when all barriers are destroyed (after 77 years) the vessel of BHR is filled by broken soil and crashed concrete.

We suppose that perched water table level locates above the Moscow moraine clay on 0,5 m and the infiltration flux is $3x10^{-5}$ m/day. The perched water table was supposed to exist constantly. Accordingly with [5] for lateral spreading the longitudinal dispersion was taken as 50 m and the transverse dispersion as 10 m at the distance 2 km. To calculate the vertical migration in the case of BHR flooding the longitudinal dispersion was taken as 0.3 m and the transverse dispersion as 0.1 m. The porosity of soil was 0.3, whereas the active porosity 0.2.

RESULTS

The results of calculations show that the temperature in the repository is 160 ^{0}C for the moment when the total radioactivity is 180.000 Ci. This temperature does not exceed the maximum permitted value.

Accordingly to hydro-geological data the slope of the underground water level in water horizon on the disposal side is about $10^{-4.}$ Groundwater in between moraine's horizon has hydrostatic pressure 22 m. Thus radionuclides can not be directly transferred with infiltration stream in water level horizon and will migrate in horizontal direction in clay with a very low velocity. However the results show that zero contents of nuclides will be observed at the distance 2 km from repositories. This is the typical distance from nearest water source. From this results we can conclude that the lateral spreading of activity is not hazardous.

We analysed the scenario of radionuclides release from matrix into water level horizon by perched water table and horizontal transfer of nuclides in perched water table. We assumed that a layer of loamy soils locates 0,5 m above matrix and on distance 10 m from BHR. Thereafter we assumed that the filtration stream should flow into the area of destroyed concrete barrier and leach the radionuclides. Along the stream and in the place of products of barrier degradation the filtration coefficient is constant and equal to 0,5 m/day.

We modelled mathematical the situation when the stream first flows down and then up resulting into leaching radionuclides into perched water table followed by migration into horizontal direction. The major part of leached radioactivity is located near repositories

(Fig. 1). As one can see from the Table only data for unfixed Co-60 can be compared with the limits determined by regulatory document NRB-96.

Table I. Predicted concentrations of nuclides and doses for scenario of radionuclides release into water level horizon through perched water table.

Nuclide	Maximum concentration, Bq/l	Maximum dose, Sv/y	Maximum time, y
Co-60(K_d=0)	4,1	5,7x10^{-5}	89
Co-60(K_d=20)	7,4x10^{-8}	1,1x10^{-12}	144
Cs-137	1,1x10^{-4}	7,3x10^{-9}	465

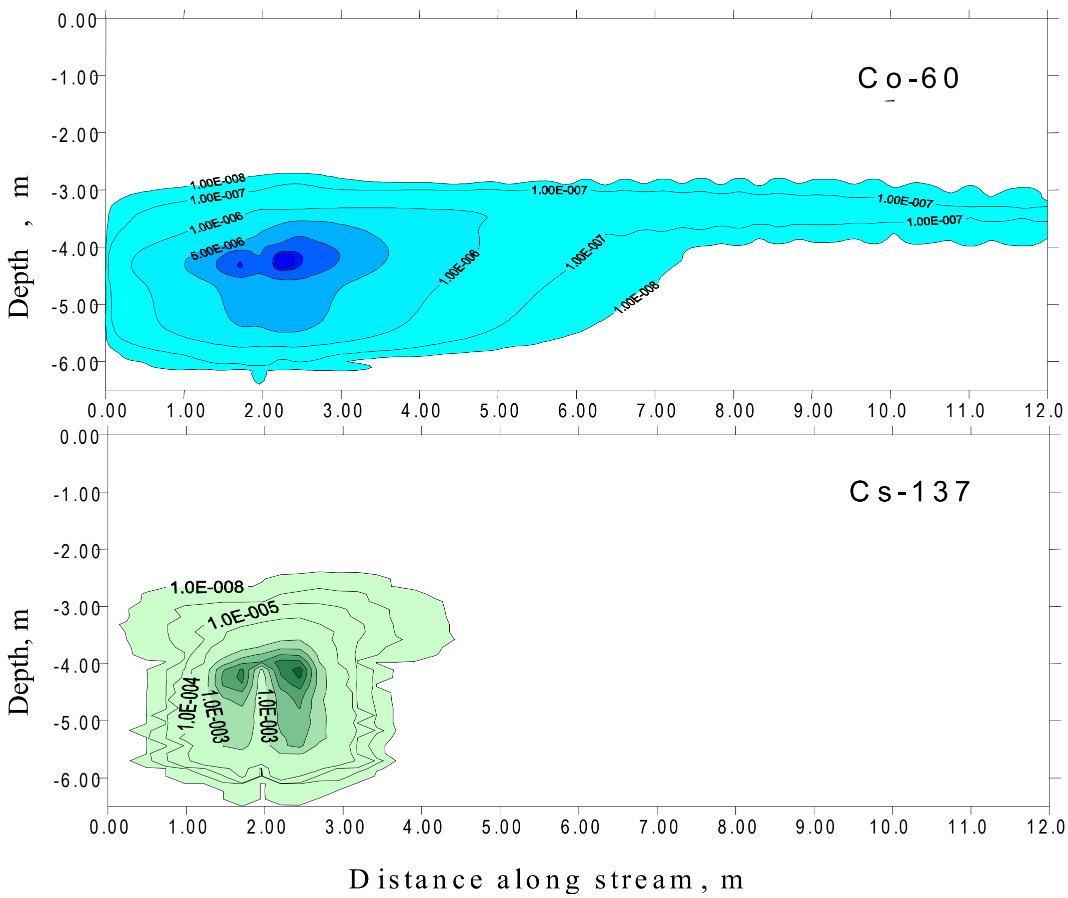

Fig. 1 after 100 years for the case of flowing of infiltration stream in BHR.

In order to simulate the migration of activity in perched water table taking into account heat convection caused by the heating of BHR due to radioactive decay, we used the assumptions of the previous scenario. In the area of repository we supposed the vertical upraise of ground water level. We modelled mathematical situation when the stream first flows down and then upward. This results into the radionuclides leaching by perched water table and migration into horizontal direction. We can conclude that the filtration inhomogeneities can play a more significant role in transportation of radionuclides with perched water table than convection circulation due to matrix heating (Fig 2, Table II).

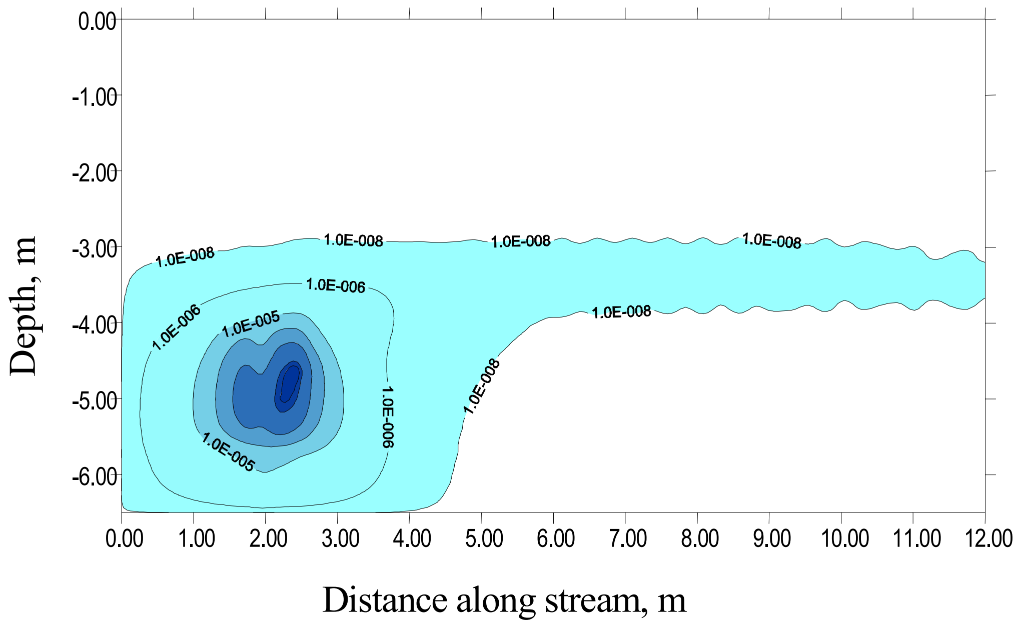

Fig. 2 Groundwater specific radioactivity (Ci/m^3) after 100 years taking into account convection caused by heating of repositories due to radioactive decay.

Table II. Predicted concentration of radionuclides and doses for scenario with migration of radionuclides in perched water table due to convection caused by heating of repository.

Nuclide	Maximum concentration, Bq/l	Maximum dose, Sv/y	Maximum time, y
Co-60(K_d=0)	5,4	7,5x10^{-5}	90
Co-60(K_d=20)	3,2x10^{-8}	4,5x10^{-12}	150
Cs-137	5,0x10^{-5}	3,3x10^{-9}	460

We analysed the scenario of possible migration of radionuclides in the case of full flooding of BHR. The flooding is possible if confining stratum (water-resistant clay) locates on the low depth. We assumed that the concrete barrier is destroyed not completely and in the centre of BHR there is a confining stratum. Only in this case there is a possibility for descending and ascending streams to achieve the metal block with SRS.

One can see (Table III) that the repository flooding results in nonessential release of radioactivity.

Table III. Predicted concentration of radionuclides and doses for scenario with radionuclides migration when flooding repository.

Nuclide	Maximum concentration, Bq/l	Maximum dose, Sv/y	Maximum time, y
Co-60 (K_d=0)	$4,6x10^{-5}$	$6,4x10^{-10}$	106
Co-60 (K_d=20)	$1,9x10^{-23}$	$2,7x10^{-28}$	204
Cs-137	$5,6x10^{-4}$	$3,7x10^{-8}$	220

We analysed the scenario of possible radionuclides migration due to heat convection caused by heating of repository when flooding it. The conditions are the same as in the previous scenario, except the gradient of water level and the filtration coefficient is constant and equal to 0,5 m/day.

One can see (Table IV) that BHR heating can not cause essential migration of radionuclides to ground surface.

Table IV. Predicted concentration of radionuclides and doses for scenario with radionuclides migration due to heat convection caused by heating of repository.

Nuclide	Maximum concentration, Bq/l	Maximum dose, Sv/y	Maximum time, y
Co-60 (K_d=0)	$4,5x10^{-7}$	$6,3x10^{-12}$	115
Co-60 (K_d=20)	$2,6x10^{-30}$	$3,6x10^{-35}$	152
Cs-137	$2,8x10^{-9}$	$1,9x10^{-13}$	249

On the base of the empirical model of aeration zone [6] we considered the possible radionuclides migration with ground water capillary uplift.

We assumed that the BHR locates in clay. The space between the matrix and walls of BHR is filled with the sandstone soil and has the thickness 0,3 m. The BHR hypothetically is filled by water on 1 m and the filtration stream through clay is absent.

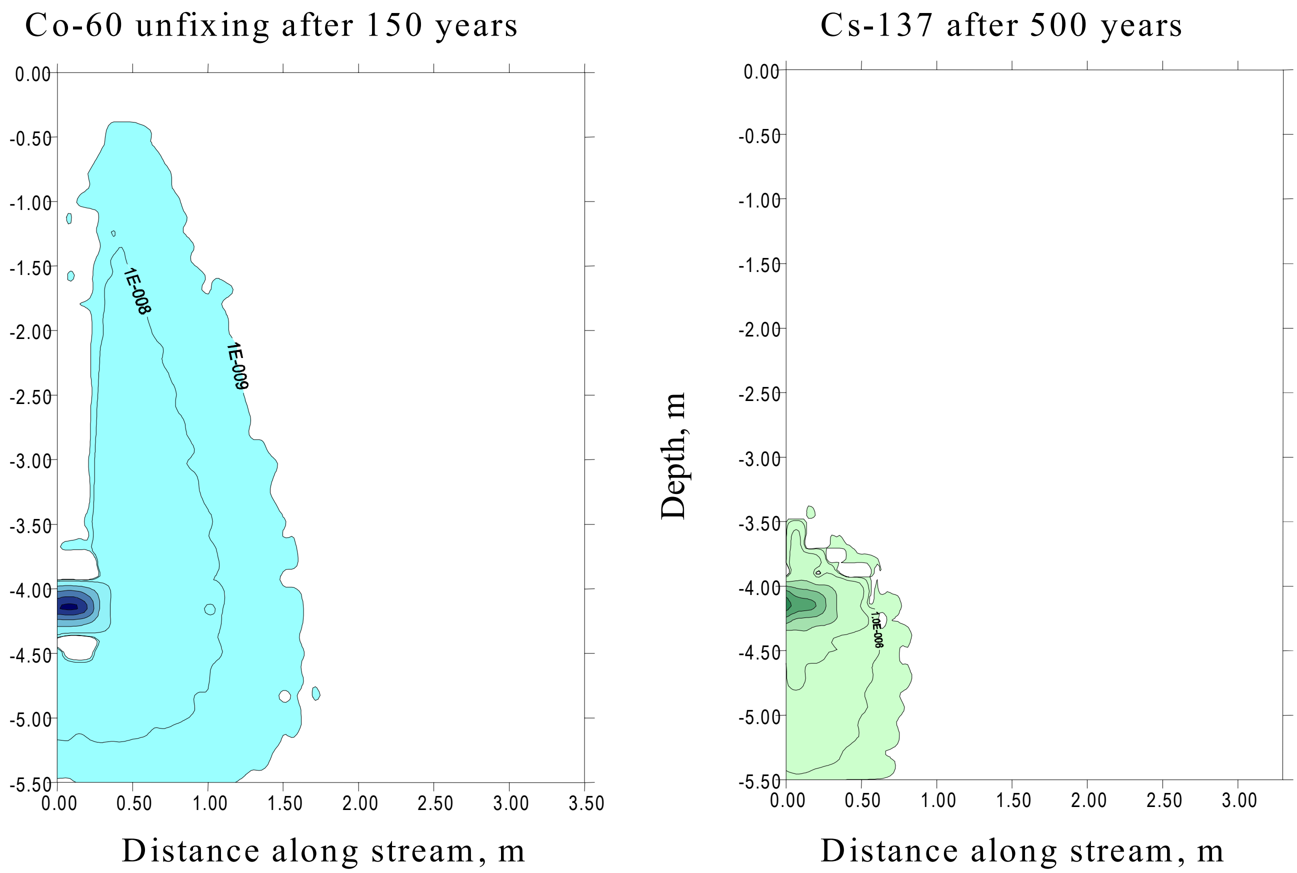

Fig. 3 Groundwater specific radioactivity (Ci/m^3) due to the groundwater capillary uplift.

The results of calculations (Table V) show that only the content of unfixed Co-60 can be observed at the upper limit. From the spread of Cs-137 and unfixed Co-60 (Fig. 3) we see that Co-60 rise with groundwater capillary uplift while Cs-137 remain in site of location of matrix with SRS. It indicates that groundwater capillary uplift can not cause any essential surface water and soil contamination.

Table V. Predicted concentration of radionuclides and doses for scenario with groundwater capillary uplift.

Nuclide	Maximum concentration, Bq/l	Maximum dose, Sv/y	Maximum time, y
Co-60 (K_d=0)	$1,5x10^{-2}$	$2,lx10^{-7}$	147

CONCLUSION

The results of calculations show that the maximum dose for populations not exceed (5,5 - 7,5)$x10^{-5}$ Sv/y even in the case of full destruction of engineered barriers and complete flooding of BHR. It indicates the high degree of safety of SRS disposal in BHR when sources are immobilised into a metal (lead) matrix. Predicted data comply the safety requirements (10^{-4} Sv/y) used at present time in Russian Federation and confirmed safe operation of repositories for population.

Due to its chemical properties the lead forms practically only insoluble compounds with groundwater anions. Heterogeneous character of exchange reaction provides fixation of lead corrosion products near the site of repository location. In addition, as been marked above, after a certain period of time (500-1000 years) the metallic matrix with sources will be removed and melted for material utilisation. This is an additional mean to protect the environment from the potential contamination by lead corrosion products.

LITERATURE

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