RADIOLOGICAL RISK ASSESSMENT OF WORKERS FOR RADIOACTIVE LIQUID EFFLUENTS TRANSFER

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Abstract. The paper aims to assess the radiological risk of workers involved in liquid waste transfer operations resulted from decommissioning process of a nuclear reactor, considering both normal and abnormal risk scenarios. The sources for liquid effluents are an underground buffer tank (source 1) and four spent fuel storage ponds (source 2). In normal working condition is calculated the effective dose based on the activity concentrations measured for liquid effluents and specific dose coefficients. The total personal equivalent dose Hp(10) received by a worker from source 1 is about 13.16 μSv and for source 2 is 185.04 μSv, 198.20 μSv from both of them (67 transfer operations). In parallel the dose ambient equivalent H*(10) measurements was performed for source 2 using an environment thermo-luminescent system, resulting in a value of 3.8 μSv in good_agreement with evaluation from activity, considering both operational quantities as being equivalent. In abnormal situation the workers are exposed during clean-up operations. Supposing that a single incident of contamination occurs, beta dose rates are highest for accidental discharges of liquid from the pond 3 - 0.70 μSv/h - and underground buffer tank - 0.16 μSv/h - due to the concentration of activity 3.21E+08 Bq and 7.29E+07 respectively.

Key words: radiological risk, worker, activity concentration, equivalent effective dose, nuclear reactor decommissioning

1. INTRODUCTION

The VVR-S nuclear research reactor (NR) from IFIN-HH Bucharest-Magurele, Romania operated between 1957 and 1997 without any incident was permanently shut-down in 2002. Currently it is in the third decommissioning stage. (Tuca et. all) presented the operation history and decommissioning process of reactor [1].

The first source (1) of liquid effluents is the underground buffer tank where are stored those effluents (34.7 cubic meters) resulted from decommissioning activities (e.g. primary circuit, the nuclear spent fuel cooling pond (CP) of reactor).

The nuclear spent fuel was stored away from reactor in deionized water in the four ponds of a Spent Nuclear Fuel Storage (SNFS) facility. This is considered the second source (2) of radioactive liquid effluents (about 29 cubic meters). The ponds must be drained in order to reuse them as interim waste store for graphite and aluminum.

The effluents are transferred from the ponds or the buffer tank, using a peristaltic pump, into the 1.2 cubic meters stainless steel containers and then are transported to the Treatment Plant for Radioactive Waste (TPRW). Each transport contains 1.0 cubic meter of radioactive liquid effluents and is accompanied by the paperwork regarding the activity inventory of radionuclides.

The gamma spectrometry and gross beta analyses performed before the liquid transfer had emphasized the existing of a significant concentration of activities for both sources [2]. The radiological inventory consists of the following radionuclides: ^{60}Co, ^{134}Cs, ^{137}Cs, ^{108m}Ag, ^{54}Mn, ^{235}U, ^{238}U (source 1) and ^{60}Co, ^{134}Cs, ^{137}Cs, ^{108m}Ag, ^{54}Mn, ^{152}Eu, ^{154}Eu, ^{241}Am (source 2).

The transfer operation of liquid effluents is divided in two phases: the transfer from pond into container and the container transport at treatment plant. The worker risk assessment is done taking into account the normal and abnormal working conditions.

The effective dose for normal working conditions (the liquid transfer from pond into container is calculated based on activity inventory [2]. For the 1-st phase of operation are performed parallel to monitoring with thermo-luminescent dosimeters (TLD). During the real transport operation, will be done the effective measurement by TLD. For the second phase of operation (container transport at treatment plant) the external and internal irradiation of worker is assessed by radiological characteristics of both sources and by internal contamination measurements.

2. METHODS FOR RISK ASSESSMENT

The method used to determine the effective dose received by a worker, during an operation, involves determination of the activity concentration of each radionuclide by gamma-ray spectrometric and gross beta activity measurements for water samples taken from areas of interest [2].

2.1. Risk assessment for normal operation

For a normal working operation (one hour) the external irradiation risk of a worker was evaluated by exposure, air kerma and effective dose rate calculation. Thus, due to the significant values of activity concentration for both sources, dose ambient equivalent measurements with environment TLD were performed for the liquid transfer from pond in container. The calculated values of effective dose received by a worker for the entire process (the liquid transfer into container and its transport to treatment plant) are then compared with the TLD measured values.

2.1.1. Exposure assessment

The worker exposure assessment for this geometry (see Fig. 1) was based on volume source model [3]:

i) the container (material stainless steel; dimensions: 1m diameter and 1.1 m high; 2.5 mm tightness; container volume 1 m^3); ii) the liquid volume of container is split into five sub-volumes (0.2 m^3 each); iii) the activity of each sub-volume is concentrated into a point in its middle.

The exposure at P is calculated for the „point sources” at 1, 2 and 3 sub-volumes and because of symmetry the exposures for 4 and 5 sub-volumes are the same as those of 1 and 2. The total exposure is obtained by addition of individual exposures. For this purpose it is calculated: the distance between each point source and P (the worker position) using (1) and (2) equations; the angular thickness of water and stainless steel to be penetrated and then the attenuation of flux by each of these thickness, and the buildup flux too.

$$x = \sqrt{z^2 + (2l)^2} \qquad (1)$$

$$y = \sqrt{z^2 + (l)^2} \qquad (2)$$

where:
x - distance from 1 or 5 sub volume centres at point P;
y- distance from 2 or 4 sub volume centres at point P;
z - distance from 3 sub volume centre at point P;
l – distance between two sub volumes centres (l=h/5 and h - container height).

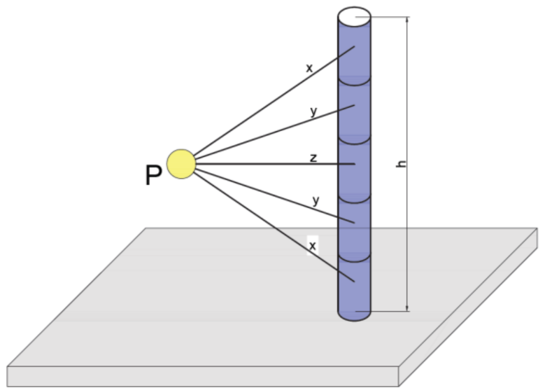

Fig.1 - Volume source modeled as five “point” sources producing exposure at P

The flux calculation involve the estimation of photon fields under poor geometry conditions, using the good geometry conditions and then adjusting the results in order to take into account the buildup of scattered photons, as follows:
i) the total attenuation of the beam in good geometry is determined by calculating the change in intensity for energy/absorber combination as:

$$I(x) = I_0 \exp(-\mu x) \qquad (3)$$

where:
I(x) – the unscattered intensity;
I_0- the attenuated intensity at P by air;
μ- the linear absorption coefficient.

ii) the unscatered intensity is multiplied by the buildup factor for the particular photon energy/absorber combination

$$I_b(x) = BI_0 \exp(-\mu x) \quad (4)$$

$$B=1+\mu x \quad (5)$$

where:
μx - number of mean free path in the absorber material;
B= (1 + μx)- buildup factor.

The exposure rate is calculated with equation (6):

$$Exp = \frac{0.533 \times A \times f \times E}{r^2} \quad (6)$$

where:
A – subvolume activity ;
f - emission yield of each radiation from one radionuclide;
E- photon energy emitted by radionuclide “i” in (0.07 ÷ 3) MeV range;
r- distance from source sub-volume to exposure point P.

2.1.2. Air kerma

Air kerma calculation is done according to equation (7) and (8):

$$K_{air} = \frac{87.64 \times 10^{-4} Exp}{1-g} \quad (7)$$

$$g = \frac{E_{tr} - E_{ab}}{E_{tr}} \quad (8)$$

where:
K_{air} - air kerma;
g - fraction of energy lost by radiative processes;
E_{tr} - average energy transferred to electrons in the medium;
E_{ab} - average energy absorbed in the medium.

2.1.3. Effective dose

The effective dose received by a worker for the transfer operation (one hour - the loading/unloading of container 20 minutes and the transport 40 minutes) from all of five sub-volumes is calculated according equation (9), considering gamma ray absorption into the air, water and the container wall.

The estimated values are conservative in general because they tend to overestimate the dose but the calculation method is simplified and associated errors are small.

$$E_{eff} = C_k \times K_{air} \qquad (9)$$

where:

C_k - weighted mean of AP (antero-posterior) and PA (postero-anterior) coefficients from table A.2 [4].

2.2 Risk assessment for abnormal operation

During the transport operation, the container could fall accidentally from the forklift and the effluents spread into the environment. The uncontrolled discharging of 1 m^3 liquid effluents determines surface contamination with beta hot particles from radionuclides: ^{60}Co, ^{134}Cs, ^{137}Cs, ^{152}Eu and ^{154}Eu.

The worker could be irradiated by internal and external exposure during clean-up operations. There also exists the risk of soil and vegetation contamination in the vicinity and consequently the situation must be analyzed immediately and punctually. The internal irradiation of worker is assessed by radiological characteristics of both sources and by internal contamination measurements according to Romanian legislation provisions [5].

2.2.1. External exposure

The external exposure assessment is made by calculation of skin and shallow dose from beta emitters -was performed assuming that the worker is placed in the middle of a uniformly contaminated surface having a 100 cm diameter.

The dose rate calculation was considering the number of particles crossing the environment per unit area (usually tissue), their range (energy dependent) and the energy deposition fraction per unit mass - beta absorption coefficient, μ_β.

The beta absorbtion coefficient from tissue, $\mu_{\beta,\ tissue}$ [3] was calculated using equation (10).

$$\mu_{\beta,tissue} = 18.6(E_{\beta\max} - 0.036)^{-1.37} \qquad (10)$$

For solid surfaces, the particle flow is determined from its contamination level (Bq/cm^2), a geometry factor (usually 0.5 since half of beta particles would be directed into the surface), and a backscatter factor, to account for beta particles that penetrate into the surface but are scattered back out to increase the beta flux reaching a receptor near the surface. On the other side, when beta particles are absorbed in tissue which has a low atomic number, less than 1 % of interactions are producing bremsstrahlung and many of those that do are likely to escape from tissue because their probability of interaction is low. Due to the hot particle small size and higher radioactivity, the hot particles could affect a small area of tissue which receives sufficient energy deposed before the detection or replacing of particles [3].

The dose rate is calculated taking in consideration that the contaminated surface contains a lot of beta point sources emitting in all directions and its average energy is distributed relatively uniform into the hemispheric volume having a R_β radius, along the tissue mass. Thus, the beta dose rate to tissue, without correction for absorption in dead layer of skin from a hot particle of activity A (Bq), one-half of which impinges of the skin, was calculated using equations (11) and (12) [3]:

$$\dot{D}_{\beta}(x) = \frac{0.51\bar{E}_{\beta} \times A}{R_{\beta}^{3}} \quad (11)$$

where:
A -hot particle activity;
R_β - range of beta particle;
E_β - average energy of each beta particle.

$$R_{\beta} = 0.412E^{1.265-0.095E} \quad (12)$$

The shallow dose Hp (0.07) was calculated with equation (13), taking into account the mass air attenuation and tissue mass thickness [3].

$$\dot{D}_{shallow} = \dot{D}_{\beta}(x) \times \exp(-\mu_{\beta,tissue}(\rho x)_{tissue}) \quad (13)$$

2.2.2. Internal exposure

In abnormal operation (accident) the worker may be exposed to internal radiation dose by inhalation or ingestion of radionuclides, by their entering through a wound or by absorption into the skin. It was assumed that clean-up operation is about one hour and incorporation was due to inhalation of evaporated activity concentrations equal to one thousandth of the volume spilled. Thus, the committed effective dose was calculated using the equation (14) and values are presented in table 5.

$$H_{50,T} = e_{inh} \times A \quad (14)$$

where:
e_{inh} - effective dose coefficient for inhalation ($SvBq^{-1}$) [4];
A - activity of inhaled radionuclide (Bq).

3. EXPERIMENTAL DATA

3.1. The normal operation

3.1.1 The dose ambient equivalent H*(10)

The assessment was done supposing that the workers are externally exposed to gamma radiations at each transfer operation of liquid effluents from ponds to container. Thus, it was measured the dose ambient equivalent H*(10) using the environment thermo-luminescent system (TLD) for source 2. It wasn't possible to perform measurements for source 1 due to the fact that the buffer tank is underground. For a complete monitoring, the dosimeters were placed on the aluminum rack of each pond for nuclear spent fuel assemblies. These were distributed as uniform as possible on the entire surface of the pond, for three days. The measured values are presented in table 1, where: D_L- dosimeter placed on the left side of pond; Dc- dosimeter placed on the middle of pond; D_R- dosimeter placed on the right side of pond.

As expected, the ambient dose equivalent values, measured with the TLD system are consistent with those doses calculated starting from activity concentrations of the identified and measured radionuclide by gamma spectrometry. For all ponds the distribution of ambient dose equivalent is uneven, what requires that the highest value of each pond radionuclide activity must be considered for worker risk assessment. For ponds 1, 2 and 4, the maximum values range between (80 ÷ 800) nSv/h and are significantly higher for pond 3, up to an order of magnitude, regardless of dosimeters position; the maximum value is (4387 ± 259) nSv/h. The irregular distribution of measured values in each pond could be caused by TLDs exposure time and radionuclides concentration into the pond.

Table 1
The dose ambient equivalent H*(10) for source 2

Dosimeter	**$H^*(10)$ (nSv/h)**			
	Pond 1	**Pond 2**	**Pond 3**	**Pond 4**
D_L	60±4	173±11	1684±120	305±8
D_C	80±7	445±25	2595±158	800±21
D_R	44±5	250±14	4387±259	425±19
Mean values	61±5	289±17	2887±179	510±16

3.1.2. Gamma dose

The penetrant dose rate, received by a worker sitting at 1 m distance far the container wall and the skin dose were assessed for the container transfer operation from ponds to the treatment plant. It was considered that the exposure time is 1 hour (loading and unloading of container 10 min each of them and 40 min for container transport to treatment plant).

The conversion coefficients for photons C_K (Sv/Gy) are determined as weighted mean of AP (antero –posterior) and PA (postero-anterior) external radiation exposure. The results are presented in the table 2 and the values are compared with direct measurements performed with environment thermo-luminescent systems (TLD) (see table 1).

Moreover the external evaluation of the dose received by a worker for the transfer operations of radioactive liquid effluents takes into consideration that it is necessary to do thirty-five transports from source 1- buffer tanks - and about eight transports for each pond of source 2 – SNFS (32 transports).

The total personal equivalent dose Hp(10) received by a worker from source 1 is about 13.16 μSv and for source 2 is 185.04 μSv, 198.20 μSv from both of them (67 transfer operations). Those values are much lower than equivalent dose limits 20 mSv per year for profesional exposure as it is mentioned in national radioprotection norms [2].

It can be concluded that in normal conditions, the transfer of containers with liquid radioactive waste from IFIN-HH nuclear reactor to the waste treatment plant, does not involve a significant risk for external irradiation of workers.

From the comparative analysis of dose values obtained by direct measurement (TLD) for source 2 and calculation (from activity concentrations) (see table 3), it may deduce that for a worker who performs a transfer operation (1 hour) - the dose for each pond is in a good agreement with direct measurements. The differences between measured and calculated values for ponds 2 and 3 should be determined by TLD dosimeter emplacement and exposure time.

3.2. Abnormal operation

In abnormal operation (accident), the workers can be exposed both externally and internally during clean-up operations. So, both exposures were assessed. The external beta dose rate in tissue and dead layer due to the hot particles, was calculated using equations (8)÷(10) (see table 4), assuming that the surface is uniformly contaminated and the worker is located in middle of the surface (a circle with 1 m diameter).

3.2.1. The external dose assessment

The results show that the external beta dose rate are highest for accidental discharges of water from the pond 3 almost 0.70 (μSv/h) and 0.16 (μSv/h) for buffer tank due to the higher concentration of activity of effluents. Ponds 2 and 4 have comparable values and the lowest value was obtained for pond 1 where the concentration of activity is lowest.

3.2.2. The internal dose assessment

The committed effective dose was calculated using the equation (14) and the values are presented in table 5. The results shown that the most important internal exposure is due to the liquid waste transfer from pond 3 – 30 mSv - and pond 4 at about 23 mSv, that refelects the over exposure of a worker. As consecquences the exposure time must be limited for all personal involved.

Table 2
Air Kerma and effective dose for one transport in normal conditions

Radio nuclide	**SOURCE 1**		**SOURCE 2**							
			Pond 1		**Pond 2**		**Pond 3**		**Pond 4**	
	K (µGy)	**Eeff (µSv)**	**K (µGy)**	**Eeff (µSv)**	**K (µGy)**	**Eeff (µSv)**	**K (µGy)**	**Eeff (µSv)**	**K (µGy)**	**Eeff (µSv)**
^{60}Co	3.87E-01	3.52E-01	2.24E-02	2.04E-02	2.25E-02	2.03E-02	1.89E-01	1.72E-01	5.72E-02	5.20E-02
^{134}Cs	8.22E-03	7.40E-03	1.66E-03	1.49E-03	2.37E-03	2.13E-03	1.22E-03	1.10E-03	2.84E-03	2.56E-03
^{137}Cs	1.47E-02	1.32E-02	2.01E-02	1.81E-02	7.33E-02	6.59E-02	5.28E+00	4.74E+00	5.66E-01	5.09E-01
^{108m}Ag	1.09E-03	9.78E-04	8.31E-03	7.45E-03	2.93E-03	2.63E-03	7.44E-02	6.68E-02	1.62E-02	1.45E-02
^{152}Eu	-	-	-	-	4.27E-03	3.83E-03	2.60E-02	2.34E-02	2.72E-02	2.44E-02
^{154}Eu	-	-	-	-	8.23E-03	7.41E-03	1.06E-02	9.51E-03	1.15E-02	1.03E-02
^{54}Mn	3.41E-04	3.07E-04	-	-	-	-	-	-	-	-
^{235}U	2.31E-03	2.08E-03	-	-	-	-	-	-	-	-
^{238}U	2.69E-02	5.17E-04	-	-	-	-	-	-	-	-
^{241}Am	-	-	-	-	-				1.50E-03	7.65E-04
Total	4.41E-01	3.76E-01	5.25E-02	4.74E-02	1.14E-01	1.02E-01	5.58E+00	5.02E+00	6.82E-01	6.13E-01

Table 3
Comparison between calculated and measured effective dose values for source 2

	H*(10) (μSv)	**Ec (μSv)**	**Ec/H*(10)**
pond 1	6.10E-02	4.74E-02	0.78
pond 2	2.89E-01	1.02E-01	0.35
pond 3	2.89E+00	5.02E+00	1.73
pond 4	5.10E-01	6.13E-01	1.20

Table 4
The external beta dose rate for abnormal operation

	SOURCE 1 BUFFER TANK		**SOURCE 2 SNFS**							
			Pond 1		**Pond 2**		**Pond 3**		**Pond 4**	
HOT PARTICLES	$\dot{D}_{\beta,tissue}$ (μSv/h)	$\dot{D}_{\beta,shallow}$ (μSv/h)	$\dot{D}_{\beta,tissue}$ (μSv/h)	$\dot{D}_{\beta,shallow}$ (μSv/h)	$\dot{D}_{\beta,tissue}$ (μSv/h)	$\dot{D}_{\beta,shallow}$ (μSv/h)	$\dot{D}_{\beta,tissue}$ (μSv/h)	$\dot{D}_{\beta,shallow}$ (μSv/h)	$\dot{D}_{\beta,tissue}$ (μSv/h)	$\dot{D}_{\beta,shallow}$ (μSv/h)
^{60}Co	2.37E-02	1.14E-02	1.90E-05	9.10E-06	5.44E-03	2.60E-03	1.16E-02	5.55E-03	3.50E-03	1.67E-03
^{134}Cs	6.79E-05	5.29E-05	1.90E-07	1.48E-07	2.72E-06	2.12E-06	1.00E-05	7.83E-06	3.26E-06	2.54E-06
^{137}Cs	7.54E-03	5.27E-03	1.43E-05	1.00E-05	3.77E-04	2.63E-04	2.71E-01	1.89E-01	4.03E-03	2.82E-03
^{152}Eu	1.27E-05	1.01E-05	-	-	2.00E-06	1.59E-06	8.79E-05	6.98E-05	1.27E-05	1.01E-05
^{154}Eu	1.15E-05	8.49E-06	-	-	8.29E-06	6.10E-06	7.64E-05	5.62E-05	1.15E-05	8.49E-06
^{90}Sr	6.40E-02	4.61E-02	2.36E-06	1.70E-06	9.39E-05	6.77E-05	1.18E-01	8.52E-02	4.45E-04	3.21E-04
Total	9.54E-02	6.28E-02	3.59E-05	2.10E-05	5.92E-03	2.94E-03	4.01E-01	2.80E-01	8.00E-03	4.83E-03

Table 5.
Commited effective dose $H_{50,T}$ by internal exposure for abnormal operation.

	$H_{50,T}$ [μSv]				
	Source 1	**Source 2**			
Radionuclides		**pond 1**	**pond 2**	**pond 3**	**pond 4**
^{60}Co	2.30E+01	1.84E-01	5.29E+00	1.13E+01	3.40E+00
^{90}Sr	3.54E+02	1.31E-01	5.19E+00	6.54E+02	2.46E+00
^{134}Cs	4.76E-01	1.33E-02	1.90E-02	7.03E-02	2.28E-02
^{137}Cs	1.59E+01	3.02E-01	7.93E-01	5.69E+02	8.48E+00
^{108m}Ag	1.11E-01	1.18E-01	4.16E-02	7.63E+00	2.30E-01
^{152}Eu	-	-	4.34E+02	1.91E+04	2.77E+03
^{154}Eu	-	-	1.02E+03	9.42E+03	1.42E+03
^{54}Mn	3.48E-03	-	-	-	-
^{235}U	1.92E+01	-	-	-	-
^{238}U	6.18E+02	-	-	-	-
^{241}Am	-	-	-	-	1.95E+04
Total	1.03E+03	7.48E-01	1.47E+03	2.98E+04	2,37E+04

5. CONCLUSIONS

The assessed exposure of workers that perform the transfer operations of the radioactive liquid wastes was evaluated by calculation and direct measurement of the ambient dose rate using TLDs. The calculation use Martin's model based on the radionuclide content and some assumptions regarding volume source for both scenarios normal irradiation and accidental irradiation due to effluent spilling.

The external beta dose rate received by an worker which perform clean-up activities in abnormal operation, accidental irradiation due to effluent spilling, is significant for those pond 3 and buffer tank transfer process.

The ccomparative analysis of dose values obtained by direct measurement (TLD) and calculation for source 2 show that the dose for each pond it's in a good agreement with direct measurements.

The containers transfer operations to treatment plant do not involve a significant risk for external irradiation of workers as they may be received only 1 % of the dose limit allowed by national radioprotection requirements [6].

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