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RADIATION DOSE EVALUATION FOR HYPOTHETICAL ACCIDENT WITH TRANSPORT PACKAGE CONTAINING IRIDIUM-192 SOURCE

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ABSTRACT

The aim of this paper is to evaluate dose rates for a hypothetical accident with transport package containing Iridium-192 source and to design additional shielding necessary for the safe unloading of the container, assuming that during the unloading process the whole contents of a radioactive source is unshielded and that the operation is going to take place at the site where a working area exists in the vicinity of the unloading location. Based on the calculated radiation dose rates, a single arrangement of the additional concrete shields necessary for reduction of the gamma dose rates to the permitted level is proposed. The proposed solution is optimal considering safety on one hand and costs on the other.

1 INTRODUCTION

In the last couple of decades there has been an impressive increase in scientific and industrial application of radioactive materials. Such an extensive and widely spread usage of radioactive materials demands safe transportation of radioactive materials from the production site to the application location, as well as quick and effective response in a case of an unexpected transportation event. According to USA Department of Energy, radioactive materials transportation events can be classified into three types: transportation accidents, handling accidents, and other occurrences.

The aim of this paper is to evaluate dose rates for a hypothetical accident with transport package containing Iridium-192 source, and to propose a single arrangement of additional shielding necessary for reduction of the gamma dose rates to the permitted level.

In our analysis we have assumed the following scenario for the hypothetical accident. Upon arrival of the radioactive material on the designated location, an increase of gamma radiation dose rate has been detected on the package surface. The package consists of a container (phenolic foam), outer shielding pot (depleted uranium), and inner shielding pot (tungsten). Radioactive material is Iridium-192 with activity of 370 TBq (10000 Ci) packed in a form of solid metal pellets in three capsules. The main assumptions of the analysis are that all three capsules are accidentally opened and that the pellets are crushed, so that the normal unloading procedure can not be followed. The goal of the analysis was to design additional shielding necessary for the safe unloading of the container, assuming that during the unloading process the whole contents of a radioactive source is unshielded and that the operation is going to take place at the site where the working area exists in the vicinity of the unloading location.

QAD-CGGP code has been used as a computational tool for gamma dose rates investigation. QAD-CGGP is a member of a well-known family of point-kernel codes, developed by Los Alamos Scientific Laboratory, and is routinely used for engineering calculations of gamma ray penetration through various shield configurations. CGGP version comprises two important additions: combinatorial geometry (CG) and geometric progression (GP) fitting function for the determination of gamma-ray buildup factor.

2 CALCULATIONAL MODEL

In our analysis we have assumed that upon arrival of the radioactive material on the designated location, an increase of gamma radiation dose rate has been detected on the package surface.

The package consists of a container (phenolic foam), outer shielding pot (depleted uranium), and inner shielding pot (tungsten). The structure of the transport package is depicted in Figure 1. Radioactive material is Iridium-192 ($T_{1/2}$=74 days) with activity of 370 TBq (10000 Ci) packed in a form of solid metal pellets in three capsules (1 cm in diameter and 3.8 cm in height), which are placed inside the inner shielding pot.

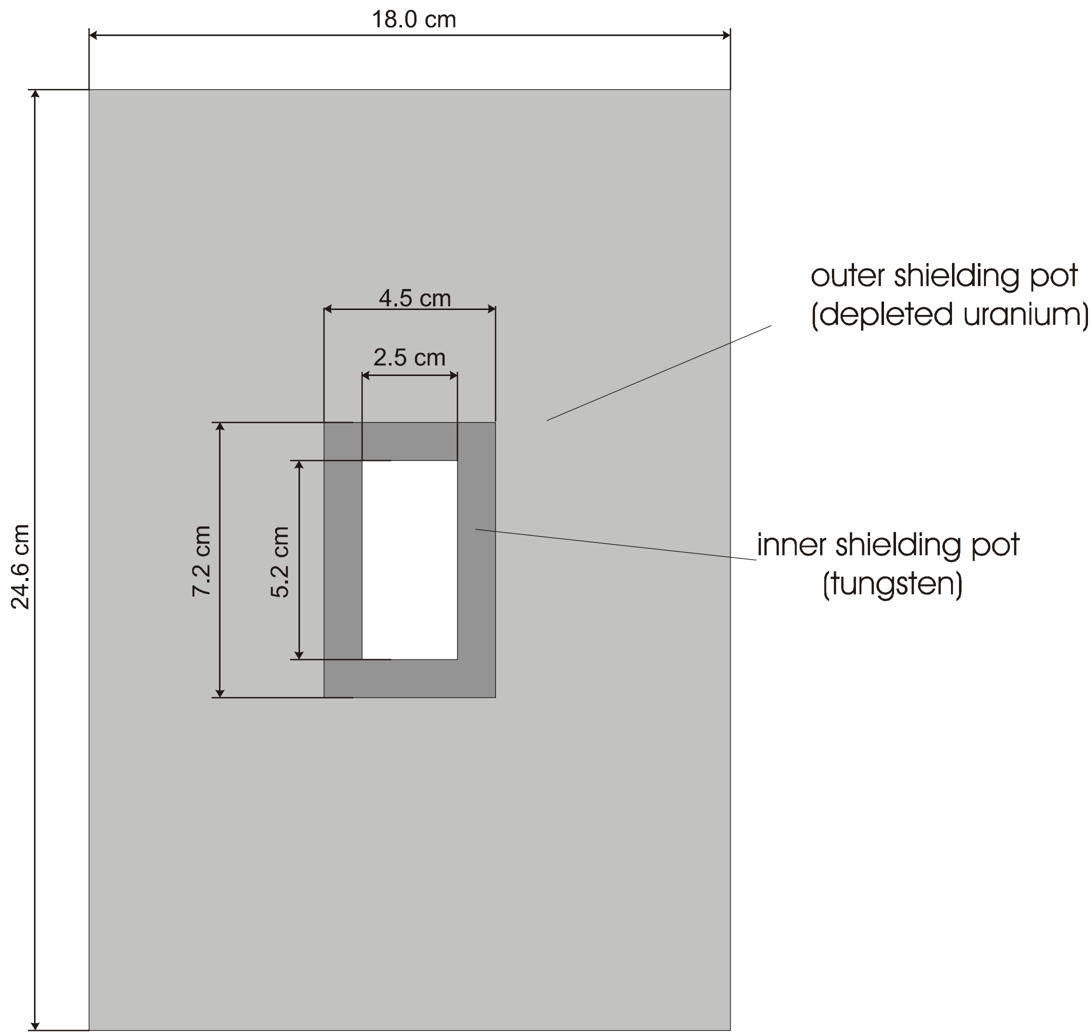

Figure 1: Inner and outer shielding pot of the radioactive material transport package

Since the goal of this analysis was to design additional shielding necessary for safe unloading of the radioactive material, worst case scenario has been analyzed. We assumed that the integrity of all protective barriers (capsules, inner pot, and outer pot) are violated and that the only way to unload the radioactive material from the damaged package is to turn the package upside down using the robot, and release the Iridium-192 pellets on the surface of the working plate. The formed pile of radioactive material has been modeled, based on our engineering judgment, as a cylinder, 17 mm in diameter and 38 mm in height, thus preserving the volume of the Iridium-192. Decay of Iridium-192 produces approximately two gamma rays per decay with a complex energy spectrum. To simplify the analysis reduction and regrouping of the original energy source spectrum has been performed, resulting in an 8 – energy group spectrum given in Table 1.

Table 1: Iridium-192 gamma spectrum

Group number	Mean energy (MeV)	Intensity (gamma/decay)
1	1.3780	$1.3000 \cdot 10^{-5}$
2	1.0620	$5.3500 \cdot 10^{-4}$
3	0.8845	$2.8390 \cdot 10^{-3}$
4	0.6076	$1.3350 \cdot 10^{-1}$
5	0.4782	$5.6533 \cdot 10^{-1}$
6	0.3148	1.1354
7	0.2858	$3.2616 \cdot 10^{-1}$
8	0.1384	$1.9100 \cdot 10^{-3}$
	Total	2.1657

The layout of the shielded building where the unloading operation would take place is depicted in Figure 2. The building wall is 20 cm thick and it is made out of ordinary concrete. It was assumed that the building is situated inside the area where a normal every day activity takes place, therefore dictating the radiation dose rates outside the building to be below permitted levels. The surrounding buildings are assumed to be in the vicinity of the shielded building (objects B, and B' on Figure 2).

Additional shielding (objects MS, AW1, AW2 and Panel 1 on Figure 2) necessary for reduction of the gamma dose rate below the limits imposed by regulations (10 CFR 20 for unrestricted area) is assumed to be achieved by ordinary concrete blocks of fixed dimensions: 40 cm × 40 cm × 19.68 cm (15.75″ × 15.75″ × 7.75″). The composition of the concrete used for building walls as well as for concrete blocks is given in Table 2.

Table 2: Ordinary concrete composition

Element	Atomic number	Density (g/cm^3)
Oxygen	8	1.2236
Silicon	14	0.7751
Calcium	20	0.1012
Aluminum	13	0.0782
Sodium	11	0.0667
Iron	26	0.0322
Hydrogen	1	0.0230

Preliminary analyses have been performed to establish the optimum position of the source inside the working area (S1 to S4 on Figure 2). To maximize the distance reduction factor, position S4 has been selected, which places the source on approximately 1 m distance from the additional shielding wall.

Locations on which gamma dose rates were calculated are marked as detector locations D1 to D15 (Figure 2). The locations were selected to provide an adequate view on the radiation dose rate distribution in and around the shielded building where the unloading operation would take place. Although the model that was used was a 3D model, all the detectors were placed in the same plane as the source, which represents conservative approach.

Gamma dose rate calculations have been performed by QAD-CGGP code. QAD-CGGP is a point-kernel code for calculating gamma-ray penetration through various shield configurations defined by combinatorial geometry (CG) specifications. Geometric progression (GP) fitting function is used for the gamma-ray buildup factor determination.

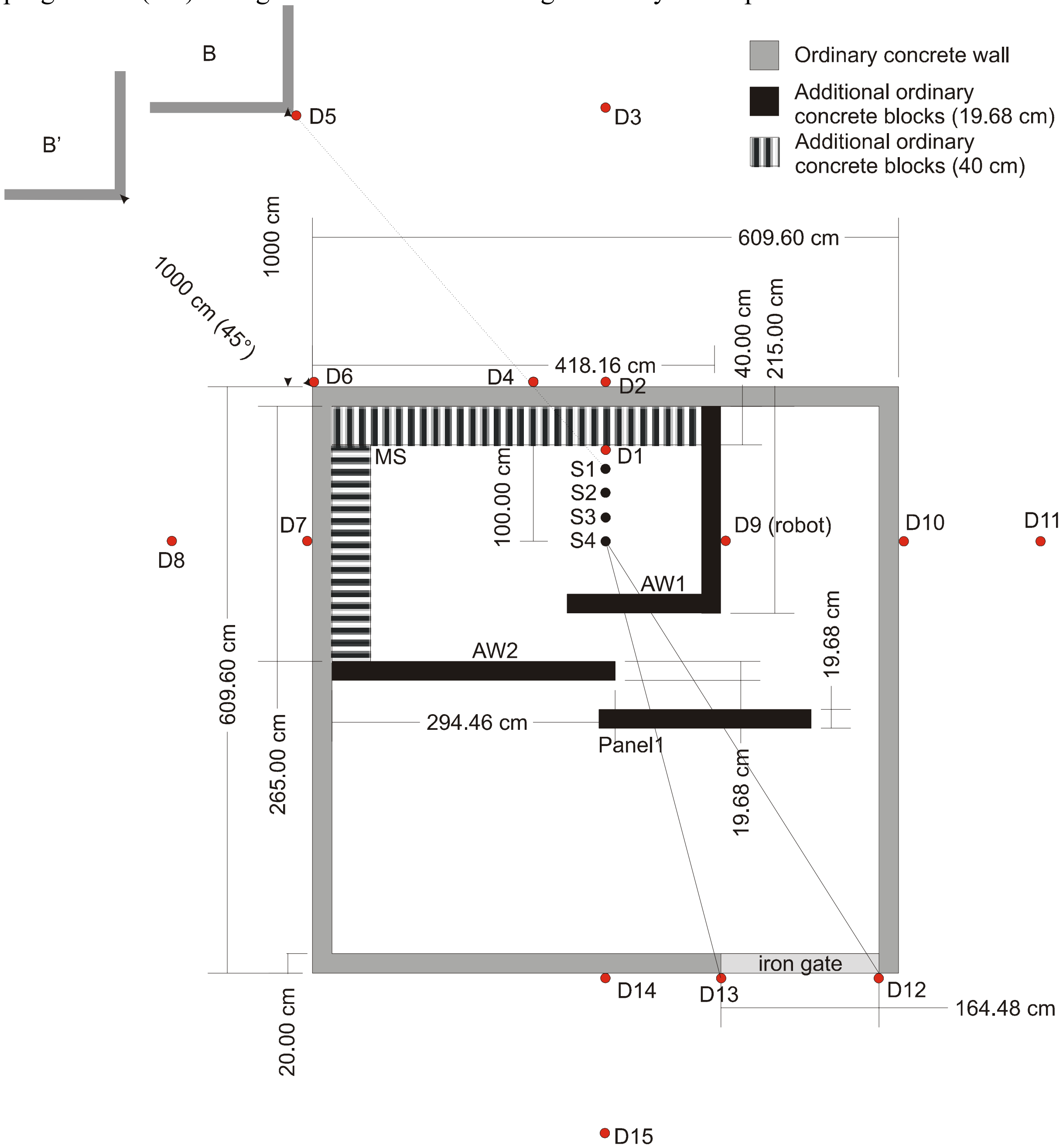

Figure 2: Shielded building layout

3 RESULTS OF THE CALCULATION

A simplified preliminary gamma dose rate hand calculation has been performed for two detector locations, D1 and D2, varying shield thickness. A point isotropic source has been assumed and the point-kernel calculation has been performed for the three most prominent energy groups (group numbers 5, 6, and 7 from Table 1) while the contribution of groups 1 to 4 and group 8 has been added to group 5 and group 7, respectively. For the additional shield thickness of 40 cm (detector location D2) the calculated gamma dose rate is $1.4235 \cdot 10^{-3}$ Sv/hr, and for the detector location D1 (no shield; 1 m distance from the source) the calculated gamma dose rate is $3.177 \cdot 10^{1}$ Sv/hr.

Based on our previous experience and preliminary simplified calculations the additional shielding arrangement and shield thickness as depicted in Figure 2, has been proposed.

Results of the QAD-CGGP calculation for the described model are given in Table 3.

Table 3: Gamma dose rates calculated by QAD-CGGP code

Detector	Distance from the source (m)	Dose rate (Sv/hr)
D1	1.00	$2.8571 \cdot 10^{1}$
D2	1.60	$1.1882 \cdot 10^{-3}$
D3	12.00	$1.9197 \cdot 10^{-5}$
D4	1.62	$1.0411 \cdot 10^{-3}$
D5	12.05	$1.4980 \cdot 10^{-5}$
D6	3.44	$3.0026 \cdot 10^{-10}$
D7	3.05	$3.2848 \cdot 10^{-4}$
D8	10.00	$1.7910 \cdot 10^{-5}$
D9	1.14	$1.5249 \cdot 10^{0}$
D10	3.04	$9.3800 \cdot 10^{-3}$
D11	62.00	$1.9972 \cdot 10^{-5}$
D12	7.20	$1.9269 \cdot 10^{-5}$
D13	4.78	$1.5972 \cdot 10^{-4}$
D14	4.68	$1.6769 \cdot 10^{-4}$
D15	9.00	$1.8583 \cdot 10^{-5}$

The main goal of the analysis was to design additional shielding necessary for minimizing the human exposure to radioactive material limiting the radiation dose rates As Low As Reasonably Achievable (ALARA principle). Additional shields placed as depicted in Figure 2 result in dose rates at the surface of the shielded building (detectors D2, D7 and D10), which are tolerable if the area is designated as a secure area. A described situation would require additional barriers around the shielded building, probably in a form of a fence.

Buildings B and B' are outside the zone with radiation dose rates higher than 0.02 mSv/hr (detectors D5 and D6). Therefore the working area in the vicinity of the shielded building can be designated as unrestricted area, which is the most important requirement for the design of the additional shielding.

Locations inside the shielded building (detectors D1 and D9) are exposed to very high radiation. Definite structure of the additional shields would have to be defined in correspondence with technical data (radiation exposure limit) of the robot that would be used for unloading operation.

The dose rates directly above the source, inside and outside of the shielded building, were not part of the analysis.

4 CONCLUSION

The aim of this paper is to evaluate dose rates for a hypothetical accident with transport package containing Iridium-192 source and to design additional shielding necessary for the safe unloading of the container. The main goal is to minimize human exposure to radioactive material limiting the radiation dose rates As Low As Reasonably Achievable (ALARA principle).

The proposed structure of additional shields, as depicted in Figure 2, results in optimal and satisfactory radiation dose rates outside the shielded building. The design of the additional shields allows safe unloading and treatment of the radioactive material inside the shielded building, as well as normal activity in the buildings that are in the vicinity of the shielded building. Although the usage of thicker shields would reduce the dose rates and minimize the restricted area around the building, it would also minimize the working area inside the shielded building and increase the costs.

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