Shield assessment of the radioactive waste storage facility

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Alpha-contaminated solid radioactive waste is generated in several plants in the front-end and back-end of the nuclear fuel cycle. Typical plants producing such waste include MOX fuel fabrication plants, fuel reprocessing plants and waste management facilities. The main contaminants of concern are uranium, plutonium and the other actinides. The disposal of such waste needs to be carried out in accordance with regulatory guidelines which require that alpha-contaminated solid waste consignments at levels > 4000 Bq/g may not be disposed of in the near-surface disposal facilities. It is recommended that such waste should be stored temporarily in storage facilities from where they can be retrieved at a later date for ultimate disposal. The storage and disposal of alpha-contaminated waste is occasionally complicated by the co-existence of high radiation fields from the β-γ sources present in the waste. Typical examples include $^{137+134}$Cs and $^{60}$Co on zircaloy hulls. This requires that the alpha waste carrying high external fields be stored in shielded vaults. Since the analysis involves for the most part shielding of large number of sources and their complex geometries, an elaborate point Kernel method based computer code QAD CG is used. Sources in the drums containing Alpha Waste are estimated on the basis of maximum contact dose rate on the drum which is 6960 mGy/h. The gamma emitting nuclide shall be assumed to be represented by $^{137}$Cs to convert the given dose rate in to the activity. A selection of 1000 mm as the side wall and top slab concrete thickness is found to be acceptable. The maximum lateral dose rate is 0.131 µGy/h, and maximum top slab dose rate is 0.651 µGy/h.

Keywords: Reactor, Radiation, Safety, Fuel, Uranium, QADCG

1 Introduction

Alpha-contaminated solid radioactive waste is generated in several plants in the front-end and back-end of the nuclear fuel cycle. Typical plants producing such waste include MOX fuel fabrication plants, fuel reprocessing plants and waste management facilities. The main contaminant elements of concern are uranium, plutonium and the minor actinides. The disposal of such waste needs to be carried out in accordance with regulatory guidelines which require that alpha-contaminated solid waste consignments at levels > 4000 Bq/g may not be disposed of in the near-surface disposal facilities. It is recommended that such waste should be stored temporarily in above-ground building-storage facilities from where they can be retrieved at a later date for ultimate disposal.

The storage and disposal of alpha-contaminated waste are occasionally complicated by the co-existence of high radiation fields from the β-γ sources present in the waste. Typical examples include $^{137+134}$Cs and $^{60}$Co on zircaloy hulls. This requires that the alpha waste carrying high external fields be stored in shielded vaults.

In view of the expanding Indian power production program, the amounts of alpha wastes are steadily increasing and it is imperative that additional storage facilities be designed at all nuclear sites to store both low β-γ and high β-γ wastes. This paper examines the radiation shielding requirements for a typical facility for storing alpha wastes carrying high β-γ dose rates.

2 Description of the Facility

The facility is intended to be an interim storage of alpha-active solid waste. It can store both low-β-γ and high β-γ wastes. Wastes carrying <1740 µGy/h are stored in drums which in turn are enclosed in mild steel containers, 4 drums to a container. The other waste is stored in drums in a shielded underground vault of the building. The typical facility building is a two-storey RCC frame structure with RCC raft foundation. The underground vault is about 5 m below grade. The ground floor is designed to provide services, such as vehicle entry, EOT operation area, personnel access facilities, etc. The handling of shielded containers is carried out with a 25t EOT crane. Access to the vault is made from the ground floor. The first floor of the structure is designed to store low β-γ active waste in steel containers; this floor is serviced by its own 5t crane. The overall dimension of the building is 38m long × 21 m wide and 24 m high above the raft level.
The capacity of the typical facility is as follows:

- Low $\beta$-$\gamma$ (200L drums, 576 mm $\phi$, 870mm height): 600 drums
- High $\beta$-$\gamma$ (200L drums): 72 drums
- High $\beta$-$\gamma$ (500L drums, 800mm $\phi$, 1000 mm height): 96 drums
- Odd consignments (e.g. glove boxes): 200 sq. m.

The underground vault is defined by the raft slab, the side walls and the top slab. The latter is punctured by a larger number of positions, normally closed by shield plugs, wherein high-active drums can be lowered. The side wall and the top slab are designed to provide adequate radiation shielding for access to the hall. The typical UG vault has $1 \times 7$ empty locations, $15 \times 8$ locations for 500 L drums and $6 \times 4$ locations for 200 L drums. At each position, the drums may be stacked one above the other in three tiers. The empty positions can accept oversized drums to hold any damaged active drums. (Fig. 1). The underground vault also has a hot cell for the repackaging of a defective container.

### 3 Procedure of Shield Assessment

The walls of the UG vault to be assessed for shielding are: the top cover, the side walls and the walls of the enclosure containing hot cell. The design information on the contents of the drums is scanty. Only the maximum contact gamma dose rate on the surface of the drum$^1$ is given as 6960 mGy/h. This must be converted into activity by making appropriate assumption. The following guidelines are followed:

- The acceptable dose at the shield surface shall be $\leq 1$ $\mu$Gy/h;
- The gamma emitting nuclide shall be assumed to be represented by $^{137}$Cs;
- The packed density of the drum shall be assumed to be 0.4 g/ml.
- All drums are full and the stack of three drums forms a continuous column.

Using cylindrical geometry and standard drum dimensions, the following Table 1 is obtained:

This shows that a maximum value of contained activity in a single drum is 14.8 TBq $^{137}$Cs equivalent, which occurs for the 500 L drum. The same activity is assumed for both sizes of drums.

### 4 Calculation of Dose rate using QAD-CG Code

Dose rate at any detector location outside the vault is calculated by using the QAD-CGGP code$^{5,6}$. This is a FORTRAN 77 code for calculation of dose rate due to a volume source in arbitrary complex geometry of shield configuration. It uses the point-kernel method, in which the original source volume is divided into a large number of small volumes of voxels. From each voxel, the optical path traversed in the source/shield material to the detector point is estimated and suitable corrections are made for the scattered contribution to obtain the dose rate based on the basic point source formula. The dose rates from all voxels are summed up to give the dose rate due to the entire volume of the source. Several build-up factor estimation techniques are optionally available, e.g. the GP method and Kapo’s method. The advanced versions of the code can take a limited number of multiple identical sources. The dose conversion factor$^7$ is taken from ICRP publication 74.

<table>
<thead>
<tr>
<th>Detector Position</th>
<th>Contact dose rate, mGy/h-Bq</th>
<th>$^{137}$Cs activity (TBq) for dose rate of 6960 mGy/h</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top, on axis</td>
<td>5.10E-10 9.20E-10 1.36E+01 7.57E+00</td>
<td></td>
</tr>
<tr>
<td>Side, at mid-height</td>
<td>4.70E-10 9.70E-10 1.48E+01 7.17E+00</td>
<td></td>
</tr>
</tbody>
</table>

Table 1 — Calculation of maximum gamma activity in drum
5 Application of QAD-CG to UG Vault:

The geometry being modeled in QAD-CG code is shown in Fig. 1. As indicated, there are 8×15 positions for the 500L drums and 6×4 positions for the 200L drums. Each position carries 3 tiers of drums and the sources carried in all drums are identical. Two wall thicknesses are tested, 900 mm and 1000 mm. With the above assumptions, the side wall is assessed for shielding effectiveness at the positions A and B opposite the 500 L drum array and at mid-height, with the position A directly opposite a stack, and the position B staggered. These values are found to be nearly identical, so in this paper only values for position ‘A’ are given. (Tables 2 and 3). A similar assessment is made at positions D and C opposite the 200L drums.

The top slab is assessed for axially penetrating radiation at points F and G for the 500 L drum array and H and J for the 200 L drum array. Here all four values are given in Tables 2 and 3.

The internal walls of the hot cell (repackaging facility for defective drums) located in one corner of the UG vault have also been assessed. The walls interfacing with the drum area are designed to the same thickness as the outer walls. However, the internal wall separating the drum handling area from the operator area needs to be designed. The dose rate for 900 mm internal wall thickness is given in Table 4.

6 Results

The results of shield calculation at different locations (Detector Location) are summarized in the Tables 2 and 3. These locations are shown in Fig. 1.

7 Discussion and Conclusions

A perusal of result Table 2 shows that for the side wall and top slab, a thickness of 900 mm concrete violates the dose criterion of 1 µGy/h. The dose rate for lateral position A is found to be 1.183 µGy/h, and that for position D 2.04 µGy/h. The dose rate at top slab position F is 0.931 µGy/h and for positions H & J, 3.33 µGy/h and 3.715 µGy/h.

A selection of 1000 mm as the side wall and top slab concrete thickness is found to be acceptable (Table 3). The maximum lateral dose rate is 0.131 µGy/h and maximum top slab dose rate is 0.651 µGy/h.

For the external walls of the repackaging cell, a thickness of 1000 mm should be adequate. However, for the internal wall, a thickness of 900 mm is sufficient.
The following wall/slab thicknesses are recommended from the viewpoint of shielding:

- a side wall thickness of 1000 mm for UG cell
- a top slab thickness of 1000 mm for top slab
- an external wall thickness of 1000 mm for the repackaging hot cell
- an internal wall thickness of 900 mm for the hot cell.

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References

6 QAD-CGGP, Fast Neutron and Gamma Penetration in Shields with Combinatorial Geometry, CCC-0493/01.
7 ICRP publication 74: Conversion Coefficients for use in Radiological Protection against External Radiation, 1st Edition.