Management of Radioactive Devices at the NCSR “Demokritos”

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Abstract

Radioactive gauges are used throughout the world in medicine, industry, agriculture, research and education. Furthermore, other radioactive devices like smoke detectors and lighting rods which were used widely in the past are withdrawn during the last years. The use of gauging systems with radioactive sources in industry is common. Fill level gauges are often used in mineral, chemical and industrial processing plants at filling lines as well as at hoppers and thickness gauges are usually used by cigarette and paper manufactures.

The risks posed by radioactive sources vary widely, depending on such factors as the radionuclides used, the physical and chemical form of the radioactive material as well as the activity. Leakage from the sources may give rise to contamination of the environment and subsequent intake of radioactive materials into the human body. Usually, after the useful life, the radioactive devices are returned to the suppliers if this is foreseen by a back-end contract. In some cases this is not accomplishable and the spent radioactive device needs to be stored in a dedicated interim storage facility for decay or until arrangement for the final solution (i.e. export for recycling or disposal).

In the present work the management of spent or orphan sources at the facilities of the National Centre of Scientific Research “Demokritos” (NCSR“D”) is presented. The management of the radioactive devices includes dismantling of the devices, recovery of the radioactive material and then packaging in appropriate containers for interim storage or transport. The procedure is carried out under a radiation protection program which concerns the occupationally exposed personnel, the public and the environment in accordance with the national and European regulations.

Keywords: radioactive devices, radioactive sources, smoke detectors, withdraw, export, recycling

1. Introduction

At the facilities of the Radioactive Material Management Laboratory (RMML) of the NCSR “D”, radioactive devices are managed and temporarily stored. The facilities are located inside a 2 m height metal lattice fence and are protected by a physical security system. The facilities comprise the radioactive waste characterization laboratory and compartments for management and safekeeping of radioactive devices and sources.

The management of the devices may include dismantling for recovering of the radioactive material, packaging in appropriate containers for interim storage or transport and safekeeping for efficient time for decay or until export for recycling or disposal. The work is carried out under a radiation protection program and is described by the approved work plan.

The risks posed by the management of these devices depend on the radionuclide used, the activity and the physical and chemical form of the radioactive material. In Table 1 examples of the wide range of radionuclides and activities in radioactive devices are presented. Usually, a radioactive device provides shielding for the radioactive source. A ‘shutter’, when is opened, allows a beam of radiation from the source to be directed towards the examined material. The devices that containing gamma emitting radioactive sources like Cs-137, Co-60, are efficiently shielded by metals of high density, such as lead and tungsten for attenuation of the radiation emitted by the source. Nevertheless, there is always the danger of high dose rates from the devices when the shutter is left opened. In general the radioactive material in the devices is sealed but in case of breaches or leakage the sources may give rise to contamination. Measurements of dose rate and contamination and in some cases radiological characterization of the radioactive source need to be performed in order to ensure radiological safety of occupationally exposed personnel, the public and the environment.

Table 1 Examples of type of radioactive devices

Type	Radionuclide	Activity (Ci)
Thickness gauges	Kr-85 Sr-90 Am-241	1 1×10^{-1} 6×10^{-1}
Fill level gauges	Am-241 Cs-137 Co-60	6×10^{-2} 6×10^{-2} 2.4×10^{-2}
Density gauges Moisture detectors Lightning Preventers Smoke detectors	Cs-137 Am-241/Be Am-241 Am-241	1×10^{-2} 5×10^{-2} 1×10^{-3} 1×10^{-6}

2. Methods and Procedures

During dismantling of devices, special radiation protection measures should be taken. The disassembling works are performed in accordance to IAEA safety guides [1-3] and complies with EU [4] and national radiation protection regulations [5].

For the safe management of radioactive devices, the radiological condition of the source should be determined and radiological characterization should be carried out when there isn't any certificate for the particular source. The characteristics of the sources are taken into account for the selection of the appropriate storage conditions, i.e. acceptance criterion, type of package, as well as for the radiation protection of the personnel.

2.1. Radiological condition and characterization of radioactive sources

Smear samples are taken from the liable for contamination points of the source and measured by a portable contamination monitor. If contamination is detected, the dismantling of the device should be performed inside a glove box with ventilation system. In case of no contamination the dismantling is carried out on a workbench with local ventilation.

In case of an Am-241, Cs-137 or Co-60 orphan or without certificate source, the non-destructive gamma spectrometry technique is used for determination of the source activity. The measurement is performed by positioning the detector active centre at a sufficient distance, usually some meters from the geometrical centre of the source and vertically to the source main axis of symmetry. The spectrometer, an Expluranium gamma ray spectrometer GR-130, is equipped with a 38 mm in diameter, 57 mm in length NaI(Tl) scintillation detector of 7% resolution for the Cs-137 peak at 662 keV. The active centre of the detector is considered at a depth of 40 mm from the detector surface. The GR-130 detector was modelled as a cylinder of NaI. The model geometries included a point source inside lead shielding of certain thickness. The models were validated against experimental measurements performed by using point sources. To obtain the energy distribution of pulses created within the detector volume, the MCNP pulse height tally (F8) was used. Counted pulses correspond to the total energy deposited in the detector by each photon at a specified energy range equal to the peak integration area. The detector efficiencies for 60 keV, 662 keV and 1332 keV photons that correspond to Am-241, Cs-137, and Co-60 respectively were predicted. A relative error of less than 6 % was achieved in all predicted efficiencies.

2.2. Acceptance criteria for storage in the NCSR "D" facility

The total external dose that a worker receives inside the storage facility for 50 h of work per year (12 h inspection and 36 h for placement of new sources) should not exceed the one half of the annual occupational dose constraint of 6 mSv [5]. So the dose rate at any point inside the facility shouldn't be higher than 3 mSv/50h = 60 μSv/h. Based on this assumption, the maximum acceptable dose rate of the devices or containers with sources arisen from the dismantling of the devices, was determined. A point source or equivalently a line source was considered at the geometrical centre or on the main axis of symmetry of each device shielding / container. Cobalt point sources of the same dose rate were considered. Then, for the radioactive devices or containers configuration inside the storage facility, the dose rates were calculated analytically as well as by Monte Carlo simulation using the MCNPX code.

The acceptable contact dose rate is a function of the distance from the geometrical centre or equivalently from the main axis of symmetry of the device shielding / container (Figure 1).

In case of higher contact dose rate than the given at Figure 1, shielding is necessary. In the case of a device / container with radius lower than 15 cm even if the contact dose acceptance criterion is satisfied, the dose on the surface may exceed the 60 μSv/h. In this case the radioactive item should be put and stabilized inside a larger container [6].

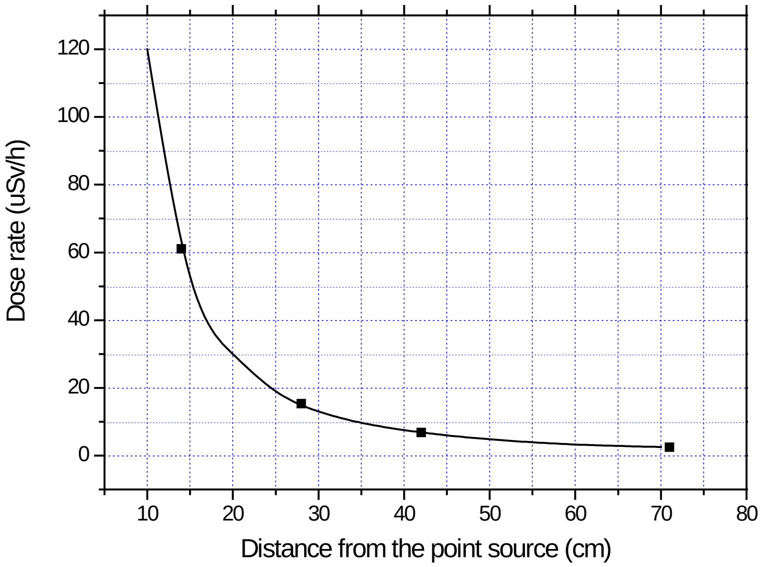

Figure 1. The acceptable dose rate at several distances from the point source. Analytical results are presented by the solid line and MCNP simulation results by the points.

2.3. Radiation protection measures for dismantling

For the workers who deal with the management of the radioactive materials, an annual occupational dose constraint is set to 6 mSv [5]. Additional for the disassembling of radioactive devices, a maximum acceptable dose rate on the hands was calculated in accordance with the annual occupational dose constrains. Particularly, any worker is considered to work 100 hours per year for disassembling of devices. The acceptable total effective dose from this activity is 1 mSv or equivalently 25 mSv equivalent doses to hands. Therefore the maximum acceptable dose rate on hands is calculated as 25 mSv/100 h=250 μSv/h. The dose rate is measured with portable dose rates meters. Furthermore personal dosimeter (TLD, EPD) are used during the work. If the dose rates is up to the maximum limit, the source from the radioactive device is removed as the work plan describe, put in a plastic bag and then into an appropriate containers. Sources of different radionuclide are put in separate containers. An identification label with code is fastened to each container in order to have under the code, all the necessary information. The containers are kept inside the interim storage under supervision for decay or until export for recycling or disposal.

In case of leakage on the source, the other materials of the device should be examined for contamination and are decontaminated by strippable coating if necessary. For release of the materials, special measurements for verification of clearance should be performed [7]. The applied practice in Greece for release of materials from regulatory control is the compliance with the general clearance levels in Bq/g that are mentioned in the Greek Radiation Protection Regulation [5]. These limits are in accordance with the EU publication RP 122 (Part I) [8]. The adopted strategy in EU for unconditional removal of materials from regulatory control are either compliance with the mass specific general clearance levels mentioned in RP122 and simultaneously with the surface specific clearance levels for direct reuse given in RP89 [9]. It is noted that generally the mass specific clearance criterion is followed when the surface specific clearance criterion is satisfied.

2.4. Export for recycling or disposal

For transport of radioactive materials, national regulations as well as international modal regulations, relevant international agreements and recommendations should be taken into account. Before any shipment special certificates and licensees should be issued. In all cases approval of shipment by the competent authority is required. The ANNEX I- Shipment of sealed sources between the member states of the European Community (1493/93) should also be completed by the country of acceptance. Regarding the packages of the radioactive materials, these should be suitable and meet the requirements of the activity limits and material restrictions for the transport [ ]. Most frequency Type A, Type B and excepted packages are suitable for transport of radioactive materials arisen from dismantling of these devices. Proper marking and labelling of packages in accordance with the Transport Regulations is also required. This includes the proper definition and application of the correct transport index.

3. Conclusion

The Radioactive Materials Management Laboratory of the National Centre for Scientific Research Demokritos performed safe and cost effective management of spent/ orhan radioactive devices and sources. The works are performed in accordance to international safety guides and complies with EU and national radiation protection regulations.

The devices are dismantled and stored for decay or until export for recycling or disposal at facilities abroad. In case the radioactive material can decay in an acceptable time, usually in some years to few decades, it is kept in the interim storage under the appropriate conditions. When the necessary time is expired, the material is released after verification of clearance by special measurements.

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