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Original Research Paper

Evaluating the Radiation Risk of Ionization Smoke Detector by MCNPX code; A Radioactive Contaminated Product

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ABSTRACT

Ionization smoke sensors are the best smoke sensors; however, the small amount of radioactive source they include is no longer desirable since it makes recycling more complicated where the people encountered with radioactive waste. In this study, a conventional ionization type smoke sensor which contained americium-241 radioactive source modelled by Monte Carlo MCNPX code. The absorbed dose rate is calculated by MCNPX code where the human tissue meddled as a soft tissue material. The dosimetry results are comparable with the experimental safety margins. It concluded that if the ionization smoke sensors positioned at a right distance from the human body, so the radiation risk and the exposure hazard can be lower than the exposure level of the background radiation.

INTRODUCTION

Besides background radiation in our radioactive planet, we receive some radiation exposure during medical diagnosis and medical treatments. However, indoor radioactive breathing due to radon gas remained a problem (Hutchison 1997). The human uses products containing radioactive materials without even knowing it (Mokhtari et al. 2013). Since smoke is one of the first indicators of fire, therefore smoke sensor/ alarm being an important safety product. Two types of smoke sensor are commonly available. One type uses a photoelectric sensor to detect the variation in the level of the source light due to smoke. This type is expensive to buy and install, and is less effective in some conditions. Another type contains a little radioactive material to detect the smoke in ionized medium. The ionic smoke sensor is the most popular and the best one, because it is cheap and cover an extended range of fire conditions (Mokhtari et al. 2013). They respond more quickly to flaming fires and are sensitive to a smaller quantity of smoke than optical sensors. However, the small amount of radioactive material they include makes the recycling procedure more complicated. But, it is determined that the average annual radiation dose received from the smoke sensor is between 0.9 and 5 μrem (Hutchison 1997). The OECD nuclear energy authority has estimated an external dose of 160 nSv from Am 241 to the inhabitants of a protected house (OECD NEA 1977).

Some green people try to condemn the use of this type of smoke sensor. Mokhtari et al. (2013) proposed a smoke sensor that the radioactive source of conventional ionic sensors has been replaced with an electrostatic system. This system produces a corona discharge to generate the needed ions for detection. The proposed sensor had shown better sensitivity rather than conventional ionic and optical smoke sensors (Mokhtari et al. 2013). In this study, to evaluate the safety aspects of radioactive smoke sensor, the absorbed dose rate in human tissue in different ranges from a radioactive type smoke sensor has been calculated using a Monte Carlo code. Then dosimetry results can be compared with radiation safety standards.

The structure of radioactive smoke sensor: The radioactive smoke sensor is composed of a source of ionization for generating charges. Although radium isotope was initially used in smoke sensors, but americium-241 is currently used in most ionization-type smoke sensors. Alpha particles generate ions by the ionization of air particles. Charges can slowly drift between electrodes polarized under low voltage which has a steady electric current to flow in the drift chamber (known as ionization chamber). This movement generates a measurable continuous electric current. When smoke particles enter the ionization medium they decrease the measured current which activates the alarm. This is illustrated in Fig. 1 (Mokhtari et al. 2013).

The hazards of americium 241: From ICRP Publication 68 (ICRP 1994), the occupational annual limit on intake (ALI) by inhalation for an aerosol with an activity median aerodynamic diameter of 1μm is 500 Bq. The ingestion ALI

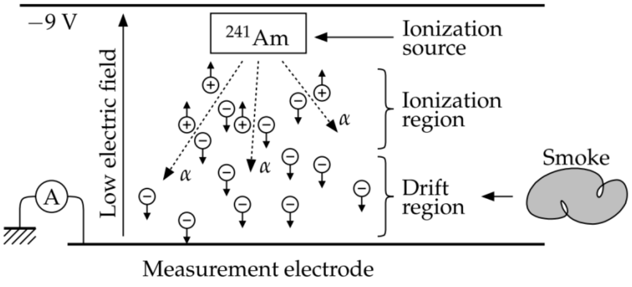


Fig 1. Principle of ionic smoke sensors. The ion source is made up of an americium radioactive element (Mokhtari et al. 2013).

is 100 kBq. Thus, a typical smoke sensor with an americium source of 33.3 kBq contains about 67 times the ALI by inhalation and about one third the ALI by ingestion for an adult worker. The chemical form of Am 241 sources used in smoke sensors is the oxide form. The radionuclide remains in a gold matrix, sintered at high temperature, then sealed between a silver backing and a thin gold or gold alloy cover (Singh et al. 1992). They are robust and can survive quite severe treatment without damage. The risk of accidental intake of Am-241 by inhalation and ingestion in normal form and fire incinerated form can be found extended in Carter's work (Carter 1996).

MATERIALS AND METHODS

Calculating absorbed dose rate of a typical radioactive smoke sensor: ARIAK™ ASD-2 radioactive smoke sensor which contains Am-241 is modelled by MCNPX Monte Carlo code. This sensor in the new design, which is known as ASD-24, can be found on the manufacture's website (http://ariak.co.ir/). Table 1 shows the specification of ASD-2 modelled sensor. The isolated components of the ASD-2 smoke sensor are shown in Fig. 2.

Modelling radioactive smoke sensor by MCNPX: MCNPX™ is a general purpose Monte Carlo radiation transport code. MCNPX is a formal extension of MCNP that tracks nearly all particles at nearly all. A main application of the code is radiation protection and the dosimetry purposes. Santos et al. (2009) used MCNP4C code as an alternative method to obtain the information about absorbed dose in a specific blood volume. In this study, MCNPX 2.6 used to model the radioactive smoke sensor to calculate absorbed dose rate of the Am-241 source. ENDF/B-VI library used to read the photo-atomic data. The details of geometrical modelling of the radioactive smoke sensor, surrounded by standard air, is shown in Fig. 3.

Table 1: Characteristic of ASD-2 radioactive smoke sensor.

Supply voltage	16-28 VDC
Radioactive material	Am-241, 0.9ìCi, 33.3 kBq
Response time	< 5 s
Standby current	50 ìA at 24 VDC
Alarm current	50 mA at 24 VDC
Connection	2 lines
Dimension	74×74 mm (78×82 mm with base)
Weight	140 g

The Am-241 radioactive source emits two types of radiation, alpha radiation and gamma radiation. The alpha radiation ionizes the air inside the chamber. It cannot move through the air for more than a few centimetres and will be stopped by a sheet of paper (Javidkia et al. 2011). So, alpha radiation confined within the chamber of the sensor. Low intensity gamma radiation of the source has penetrating ability and should be shielded by the sensor housing. However, its intensity decreases with increasing distance from the source. Source card definition for three main peaks of gamma radiation with a discrete energy source, which included Am-241 energy peaks, provided on the input deck of the MCNPX.

SDEF ERG = D1 PAR=2 POS xyz

SI1 L 0.06 0.018 0.014 ...(1)

SP1 0.36 0.18 0.14

The "F4" (cell fluence) tally used in this study. This tally gives a quantity in units of particles/cm^2 per source particle. In order to convert these units to units of dose, we will use the "Fm" (tally multiplier) card. Two basic approaches are useful for converting from fluence quantities to units of dose. One option is to fold in one or more fluence to dose conversion function. The other option is to use a heating number method. Both approaches are valid for photon dose, but the use of conversion functions is recommended for neutron dose equivalent, ambient dose equivalent, and effective dose. In the heating number method, MCNP calculates absorbed dose on the basis of the kinetic energy released per unit mass^2 approximation, which assumes that kinetic energy transferred to charged particles, is locally deposited (Lazarine 2006). The tally in the data card entry of the input deck can be written as:

F4: P soft tissue cell No.

FM4 0.000000525 3 -5 -6 ...(2)

0.000000525 factors convert each tally result from Ci to Gy/s (Lazarine 2006) and obtained as following equation.

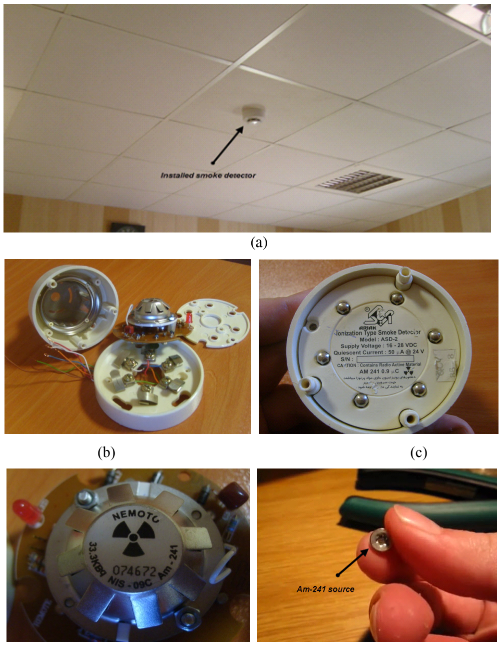
$0.9\ \mu\text{Ci} \times (3.7 \times 10^{10}\ \text{decays/s/Ci})$

$\times\ (1\ \text{photon/decay})$

$\times\ (1.5787 \times 10^{-11}\ \text{Gy/source photon})$

$= 0.00000525\ \text{Gy/s}$

1.5787×10^{-11} is the dose conversion constant of soft tissue (Lazarine 2006).



(a)

(b) (c)

(d) (e)

Fig. 2: Radioactive smoke sensor dismantled (By courtesy of ARIAK. Co); (a) installed sensor in office, (b) dismantled sensor, (c) base, (d) ionized medium, (e) Am-241 tiny disk source prepared by Nemotu Co.

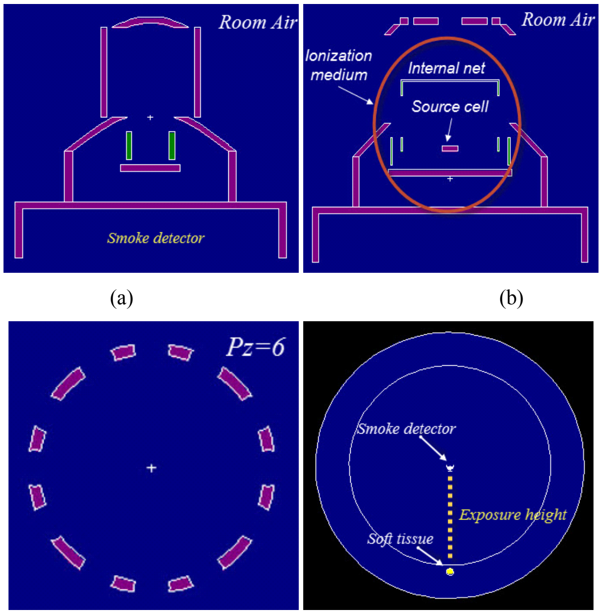
For this card, “3” represents a constant based on the material and the number of the source particles of interest for a particular problem. The next entry on the multiplier card should be the material in which the dose is to be calculated. This should be indicated using a material card number that you defined in the data section of the input deck. The “-5”entry on this card indicates that each score should be multiplied by the total microscopic cross section, $\sigma_T(E)$, at the energy of the scoring track. The “-6” entry indicates that each score should be multiplied by H(E) at the energy of the scoring track. Addition to source and tally definition for proper simulation a soft tissue sphere cell (like human body materials) was positioned in a range of distances from the sensor. Table 2 shows the details of soft tissue material (Lazarine 2006).

RESULTS AND DISCUSSION

The ionization smoke sensor modelled by MCNPX code and the effect of the enclosed radioactive source was studied at different distances. Table 3 shows the absorbed dose rate in the range of 10 cm to 400 cm.

In computation, relative errors less than 6% with 1.00×10^7 histories were achieved for F4 tally. However, the errors can be minimized by $1/\sqrt{nps}$ where nps are the histories in the MCNP code (Hashemi-Tilehnoee & Hadad 2009).

According to the work of Hutchison et al. (1997), the average annual radiation dose received from household use of smoke sensors is between 0.9 and 5 µrem or 0.285×10^{-15} to 1.585×10^{-15} Gy/s. These reported measured radiation absorbed dose rates are comparable to calculated dosimetry values given in Table 3. The calculated values are lower than the level of the background radiation.



(a) (b)

(c) (d)

Fig. 3: Smoke sensor modeling by MCNP; (a) geometry, (b) main cells, (c) upper side crosses view, (d) exposure measuring height.

(a) (b)

Fig. 4: Radiation survey over a radioactive ionized smoke sensor with 0.9µCi americium radioactive material; a) CPM of background radiation, b) CPM of Am-241 contained in smoke sensor.

In addition, the OECD nuclear energy agency reported that absorbed dose rate due to the exposure of the radioactive smoke sensor is about 160 nSv/year or 5.00×10^{-15} Gy/s (OECD NEA 1977). By comparing the absorbed dose rate measured value by the OECD and calculated values using MCNPX code concluded that the exposure of a radioactive smoke sensor at a distance greater that 2 meters is safe. So the MCNPX code has a good capability for such an application in radiation dosimetry. It is notable that the tiny source inside the sensor might actually overwhelm the pancake Geiger counter. Fig. 4 shows the pancake probe that is lo-

Table 2: Components of soft tissue card material.

Material	H-1	C-12	N-14	O-16	Na-23	Mg	Si	P-31	S-32	Cl	K	Ca	Fe	Zn	Rb-85	Zr
Density (g/cm^3)	0.10454	0.22663	0.02490	0.63525	0.00112	0.00013	0.00030	0.00134	0.00204	0.00133	0.00208	0.00024	0.00005	0.00003	0.00001	0.00001

Table 3: Absorbed dose rate in soft tissue.

Distance (cm)	10	50	100	150	200	250	300	350	400
Dose (Gy/s)	912.0E-15	75.0E-15	21.1E-15	9.80E-15	5.87E-15	3.82E-15	2.71E-15	2.04E-15	1.63E-15
Relative error	0.0023	0.0079	0.0149	0.0219	0.0283	0.0352	0.0421	0.0488	0.0546

cated over the most radioactive area of the smoke sensor casing. Although some radiation passes through the sensor casing, but the radiation that is measured in counts per minutes (CPM) is comparable to normal background radiation.

CONCLUSION

The radioactive smoke sensor MCNPX dosimetry result is comparable results with ICRP and NEA standards. It is obvious that the radiation safety legislators recommend the safe length between human body and the installed smoke sensor. The exposure detected from a radioactive smoke sensor is low and has no destructive effect on the human body. However, the inhalation or exposing at distances less than 2 m is not suggested or guaranteed. This can be a reason that ionic sensors are no longer authorized in most developed countries. The MCNPX code results have a good agreement with experimental data. So it can be used as a model dosimeter for dosimetry purposes. In addition, because of the radioactive material that is included for generating the ions, the collection and recycling is more conservative (Mokhtari et al. 2013) and it should be treated as a low-level radioactive waste. Moreover, the problem of the accidental intake of radioactive source should be considered too. Smoke detection by this type of smoke sensors can help to prevent low density fire progression, save many lives and reduce the industrial cost of fires. However, the people should be aware of the use of that low-level radioactive product.

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