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Activation Products In Medical Linear Accelerators

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Introduction

It is a well-known fact that medical linear accelerators, when operating above a certain energy, will produce activation products, also sometimes called "induced activity". This has been shown in many experiments, and can easily be proven by every linac operator, placing a dose rate meter or an electronic personal dosemeter (EPD) near the beam output window after a high energy field. The effect is most pronounced after prolonged irradiations, e.g. during water phantom checks, and this is the situation where it first became really apparent to the authors.

The effect is known for decades and has found its way into legislation, e.g. by demanding an increased air exchange rate in treatment rooms by the authorities in order to reduce inhalation doses from activation products in air. On the other hand, information about the radiation doses caused by activation products is far from complete. Medical linacs operate at different electron and photon energies, dose rates and duty cycles, are constructed by different manufacturers using varying concepts and materials and have different ages and service histories. Previous publications (e.g. *Ewen 1987*, *Rawlinson 2002*, *Perrin 2003*) concentrate predominantly on one or few of these many aspects. A comparative dosimetric study including machines from different manufacturers has not been published yet.

The aim of the present study was

- to deepen the knowledge on activation processes and products in medical linacs and to identify important processes and factors,
- to compare different brands of machines in their activation behaviour, including measurement and calculation of personal doses and
- if possible, to give recommendations how to avoid excess doses if necessary and possible

Background

Activation reactions

In the energy range covered by medical linear accelerators, two major processes are involved in the generation of activation products. The first one is the (γ,n) reaction, also called nuclear photo effect. It involves the emission of a neutron from a nucleus hit by a photon of an energy that exceeds the nucleon binding energy of that nucleus. Obviously, photons are needed to produce this effect. In the linac case they are produced by Bremsstrahlung created from the electrons – either on purpose in the target in photon mode, or as an unwanted side effect in

electron mode. The mean nucleon binding energy is about 8 MeV, depending on the mass of the nucleus – thus the complete absence of activation products in low energy linacs.

The (γ,n) reaction will produce a nucleus situated one field to the left of the parent nucleus in the table of isotopes, i.e. of the same element with the number of neutrons reduced by one. In most cases the daughter isotope will be a e^+ emitter or decay by electron capture.

The second process involved is the (n, γ) reaction, also called neutron capture, i.e. the absorption of a neutron by a nucleus, followed by the emission of the excess energy in form of a photon. Here, neutrons are needed, which in the linac case are provided by previous (γ,n) reactions. The nucleus produced is located right from the parent nucleus in the TOI, and will be a e^- emitter in most cases.

Experimental

Accelerators involved in the study

Four high energy accelerators, one from each manufacturer present in German radiotherapy centers, were investigated. The basic machine data are given in *Tab. 1*.

Manufacturer	Elekta/Philips	General Electric (GE)	Siemens	Varian
Type	SL-18	Saturne 42F	Primus	Clinac 2100 C/D
High photon energy (MeV)	15	18	15	15
Age (years)	12	12	8	2
Main refit/ time since (y)	MLC, target (1)	-	Target (1)	-
Mean workload (MU/day)	15000	12000	13000	9000
Max. Output (Mu/min)	400	200	300	400

Tab. 1: Main technical data of the accelerators involved in the study

In situ gamma spectroscopy

The main part of the experimental work consisted in the usage of a portable high resolution gamma spectrometer (see *Fig 1*), consisting of a high purity germanium detector (Canberra Industries, USA, 10% relative efficiency) and battery-driven spectrometer hardware (Inspector 2000, Canberra Industries, USA). This equipment is normally used to monitor environmental radioactivity in the field, but proved suitable for the indended use. The sensitive volume of the detector was placed at the isocenter of the linac under investigation as quickly as possible after the termination of a 400MU high energy photon beam delivered at the maximum output rate. The field size during and after irradiation was set to 15 x 15cm, assumed to be an average treatment field size. Spectra were collected during 2 hours and for

varying times: 1 min in the first part of the experiment, increasing to 15 min in the later part. These experiments were done on Friday afternoons directly after the end of the weekly clinical routine. Two days later, one or several spectra were collected in the same geometry and for longer times, in order to identify long-lived radioisotopes. The accelerators were left in the stand-by state between the two measurement cycles.

Spectrum analysis

Spectra were analyzed in a half-automatized way using the Genie 2000 software package (Canberra Industries, USA), using pre-defined isotope libraries, which were enlarged in the course of the work due to newly identified isotopes. An efficiency calibration of the detector had been obtained using ^{152}Eu and ^{133}Ba point sources, located radially at 1 m distance from the center of the detector. The obtained curve is displayed in *Fig. 1*.

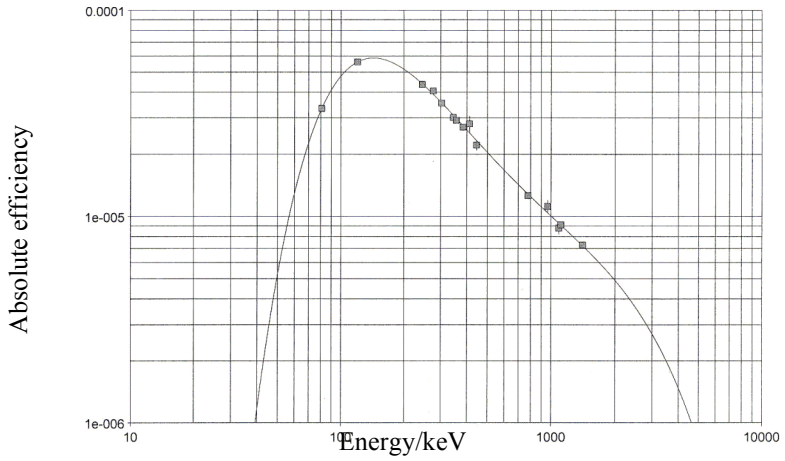

Fig. 1: Efficiency calibration curve of the gamma detector

Dose rate measurements

Gamma dose rate measurements were performed in parallel to gamma spectroscopy, using a standard portable dose rate meter with data logging capabilities (LB123, EG&G Berthold, Germany). The detector was placed adjacent to the spectrometer crystal, within the 15 x 15 cm field as outlined by the accelerator light field. The measurements were started synchronously with recording of the spectra, and the instrument was set to record a dose rate value every 30 seconds. *Fig. 2* displays the complete experimental setup in one of the sites.

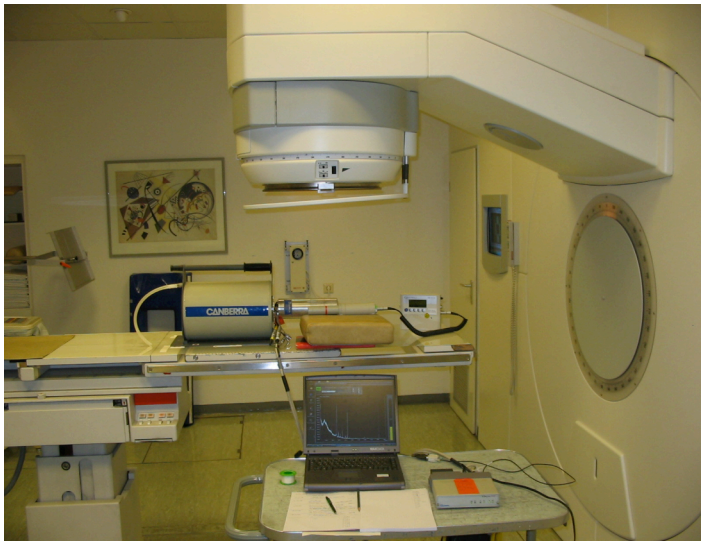

Fig. 2: Complete experimental setup, showing gamma detector (left) and dose rate meter (right) on the treatment couch top, plus spectrometer hardware in front.

"Apparent activity" and dose rate calculation

Using the efficiency calibration described above, a quantity A_{ap} we denominated "apparent activity" could be determined for each identified isotope according to *Eq. 1*:

Eq.1:
$$A_{ap}(\Delta t = 0) = \exp(\lambda \Delta t)\frac{n}{t_{live} f \varepsilon},$$

with A_{ap} = "apparent activity" as defined below, λ = radioactive decay constant of the isotope, Δt = time delay between time of beam off and spectrum acquisition, n = net peak area, t_{live} = measurement time, f = branching ratio of the observed line and ε = peak efficiency of the detector at the energy of the observed line.

A_{ap} gives the value of the activity of an isotope assuming that all nuclei are concentrated at 1m distance from the detector, i.e. at SID. We assume in this way that all induced activity is concentrated in the accelerator target. Whilst this is not true in all cases, this quantity is useful for comparison between different machines under similar geometric situations.

In the next step, gamma dose rates were calculated from the obtained activity data, using the gamma dose rate constant Γ for each isotope, either obtained from the literature or calculated by the method described in *(Krieger 2002)*. We assume that these dose rates are representative for the absorbed dose rate at the isocenter of the accelerator and use them later on for an estimate of the radiation dose received by the medical and technical staff working near the treatment head of the machine.

Results

Data resulting from spectroscopic measurements are summarized in *Tab. 2*. A_{ap} data from different Δt were used to calculate a weighted mean. Data for ^{62}Cu (a positron emitter with a very weak gamma line) were obtained by time series analysis of the 511 keV annihilation

peak in the spectra. The data were then used to calculate the sum of the dose rate contributions from all isotopes.

In order to determine the effect of time, dose rate sum values were calculated for different time delays. Results are given in *Tab. 3* for all four sites and allow the estimation of long-time effects.

Time after beam-off	Elekta Dose rate µSv/h	GE Dose rate µSv/h	Siemens Dose rate µSv/h	Varian Dose rate µSv/h
0 min	1.05	1.50	0.78	3.16
1 “	0.95	1.26	0.63	2.55
5 “	0.69	0.74	0.36	1.34
10 “	0.56	0.56	0.25	0.85
30 “	0.43	0.46	0.18	0.54
60 “	0.38	0.42	0.17	0.46
120 “	0.34	0.36	0.16	0.41
1 day	0.14	0.136	0.087	0.15
7 days	0.04	0.061	0.015	0.018
1 month	0.03	0.052	0.007	0.012
1 year	0.01	0.025	0.004	0.0036
10 years	0.0007	0.0033	0.0007	0.0008

Tab. 3: Calculated total dose rates for all involved sites.

Dose rate measurements

Measured dose rates (after subtraction of the average environmental background of 0.1µSv/h) were compared to calculated dose rates; the measured values were systematically higher by about 20 to 100% in all experiments (see *Fig. 3*). The effect is most pronounced for short delay times. Several reasons can responsible for this discrepancy:

- the detector of the dose rate meter is also sensitive to beta particles. A shielding with 1 cm of water-equivalent material reduced the dose rate by about 40%. The same effect was found by Ewen *(Ewen 1987)*;
- the calibration of the instrument (certified to +/- 30% between 30 keV and 1.2 MeV) contains a large uncertainty and does not cover the full range of photon energies encountered in this study;
- one or several short-lived isotopes may have been missed in the early spectra
- long-lived positron emitters without or with weak gamma lines may have been missed in the process of nuclide identification;
- the background in the treatment room could be higher than average.

On the other hand, the comparison proves that our calculated dose rates give realistic values, probably biased towards a slight under-estimation.

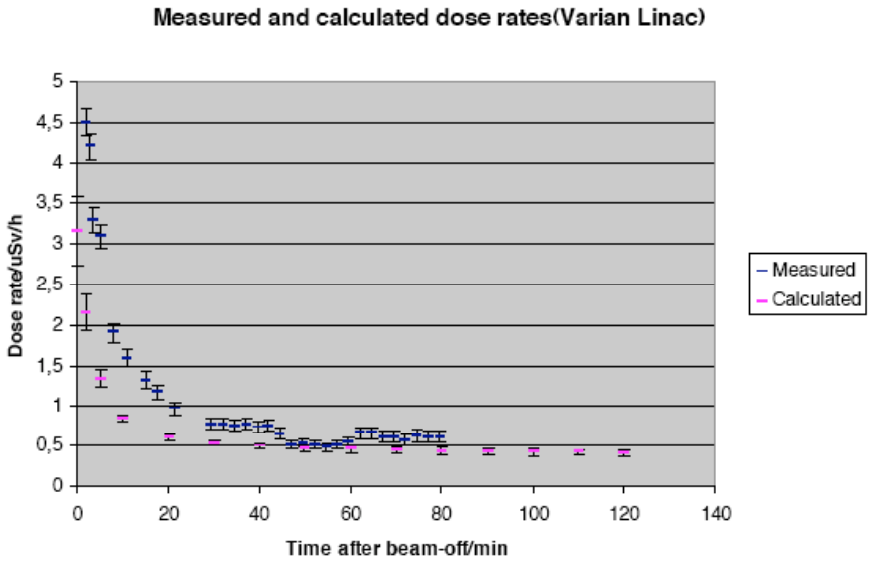

Fig. 3: Comparison of measured and calculated dose rates for one of the sites.

Doses for the staff

The obtained total gamma dose rates allow the calculation of annual doses for the staff assuming realistic workflow scenarios. This has been done for RTTs using the following equation:

Eq. 2:
$$D_{RTT} = 220\,\frac{d}{yr}\quad n\quad \int_{2\,\mathrm{min}}^{7\,\mathrm{min}} DR_{meas}\,d(\Delta t)$$

with 220 = number of working days per year, n = number of patients treated in an 8h shift and DR_{meas} = measured dose rate. As the workflow of an RTT is repetitive and can be described quite well, the mathematical simplification should be not too severe.

For service technicians and physicists the situation is less easy to describe. A rough estimate has been made using the following equation:

Eq. 3:
$$D_{tech} = 220\,\frac{d}{yr}\quad 8\,\frac{h}{d}\quad DR_{meas}(\Delta t = 1h)$$

with the same definitions as in *Eq. 2*. The working time of 8 hours per day might be an over-estimate, but on the other hand, the affected persons often work at the accelerator after prolonged beam-on periods and with less time delay. In any case, here the obtained annual doses cannot be more than an estimate. On the other hand, our data allow closer estimates for well-defined situations

The results are shown in *Tab. 4*. It should be noted that the data do not justify the assumption that occupational doses resulting from induced activity are systematically higher in acceleators from specific manufacturers compared to others (see discussion).

	Annual Dose/mSv			
	Elekta	GE	Siemens	Varian
Clinical staff	1.68	2.3	0.62	2.53
Technical staff	1.06	1.41	0.35	1.06

Tab. 4: Calculated annual doses for hospital staff for the four machines.

Discussion

We assume that all important isotopes have been identified with the possible exception of very short-lived ($T_{1/2} < 1$min) or pure beta-emitting nuclides. The calculated dose rates at isocenter are realistic and in satisfactory agreement with measurement, given the uncertainties of both calculation and measurement. Calculated annual doses for the staff will vary with the assumed scenario. The large range of possible radiation burdens for the technical staff make the use of EPD with alarm function advisable for these persons.

The comparison between the four machines shows that the important isotopes are almost identical. On the other hand, no clear distinction could be made according to manufacturer, age or photon energy. Whether the choice of materials used in the machines and in the treatment room can have an influence is an open question, work in this area is in progress.

Conclusions

In situ gamma spectroscopy proves to be a suitable method for investigations of radiation protection problems in medical accelerators, and the results obtained in this study can be used for dose calculations based on measured apparent activities and calculated dose rate constants. Calculated annual doses for clinical staff are below annual limits for radiation workers, but not negligible. The annual dose for RTTs could be reduced by delaying staff entrance to the treatment room after each high energy photon beam. Technical staff and physicists would profit from the use of an EPD with alarm function in order to recognize situations of increased radiation levels.

References

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Isotope	$T_{1/2}$	Decay modes	Γ mSv m^2 h^{-1} GBq^{-1}	Elekta		General Electric		Siemens		Varian	
				A_{ap} MBq	Dose rate µSv/h	A_{ap} MBq	Dose rate µSv/h	A_{ap} MBq	Dose rate µSv/h	A_{ap} MBq	Dose rate µSv/h
^{24}Na	15.0 h	e^-	0.429	0.052	0.022	0.060	0.026	0.008	0.034	0.036	0.0156
^{28}Al	2.3 m	e^-	0.222	1.85	**0.41**	4.3	**0.95**	2.30	**0.51**	9.8	**2.17**
^{51}Cr	27.7 d	e^+	0.0042	0.124	0.0005	0.40	0.0017	n.d.		n.d.	
^{54}Mn	312.3 d	e^+	0.11	0.073	0.008	0.13	0.015	0.024	0.0026	0.0058	0.0006
^{56}Mn	2.6 h	e^-	0.23	0.63	**0.145**	1.25	**0.29**	0.131	**0.030**	0.69	**0.160**
^{57}Co	271.8 d	e^+	0.0133	0.049	0.0006	0.144	0.0019	0.0132	0.0002	0.012	0.0002
^{57}Ni	36.0 h	e^+	0.255	0.085	0.022	0.35	**0.090**	0.260	**0.066**	0.144	0.037
^{58}Co	70.9 d	e^+	0.129	0.0113	0.0015	0.037	0.0048	0.005	0.0006	0.025	0.0032
^{59}Fe	45.1 d	e^-	0.147	0.007	0.001	0.019	0.0028	0.005	0.0007	0.0065	0.001
^{60}Co	5.3 y	e^-	0.307	0.009	0.0028	0.040	0.0122	0.0079	0.0024	0.0097	0.003
^{62}Cu	9.7 m	e^+	0.151	1.50	**0.226**	0.54	**0.082**	0.59	**0.090**	3.21	**0.48**
^{64}Cu	12.7 h	e^-, e^+	0.029	3.78	**0.110**	n.d.		0.76	0.022	5.4	**0.160**
^{65}Zn	244.3 d	e^+	0.073	0.006	0.0004	0.013	0.0009	0.015	0.0011	0.0127	0.0009
^{82}Br	35.5 h	e^-	0.343	0.059	0.0203	0.009	0.0031	0.091	**0.031**	0.105	0.0360
^{99}Mo	66.0 h	e^-	0.034	0.008	0.0003	n.d.		0.008	0.0003	n.d.	
^{122}Sb	2.7 d	e^-, e^+	0.069	0.088	0.0061	0.040	0.0028	0.0089	0.0006	0.081	0.0056
^{124}Sb	60.2 d	e^-	0.26	0.0318	0.008	0.0116	0.003	0.0014	0.0004	0.024	0.0061
^{184}Re	38.0 d	e^+	0.129	0.077	0.01	n.d.		n.d.		n.d.	
^{187}W	23.7 h	e^-	0.073	0.72	0.053	0.24	0.018	0.134	0.0098	1.09	0.080
^{196}Au	6.2 d	e^-, e^+	0.071	0.044	0.003	n.d.		0.079	0.0056	0.005	0.0004
^{203}Pb	51.9 h	e^+	0.045	n.d.		0.021	0.001	n.d.		n.d.	

Tab. 2: Identified isotopes, their dosimetric properties, apparent activities obtained from the spectra and the resulting dose rate at isocenter at the time of termination of the beam, for all four machines. The isotopes of highest dosimetric importance are highlighted. Errors resulting from counting statistics are omitted for clarity and are below 10% in most cases.