q

Proteus®***ONE*** **–** Think big, scale smart

Environmental Impact report

June 2024

www.iba-worldwide.com

Protect, enhance and save lives

iba

TABLE OF CONTENTS

1. Introduction

Protontherapy is a technique which allow to treat the patient with proton beam with a very low energy dispersion in cells. The Proteus®ONE system is composed of :

- a Super-Conducting Synchro-Cyclotron (S2C2), an accelerator producing a 230 MeV protons beam,
- a compact gantry (CGTR) ended by
- a Pencil Beam Scanning (PBS) nozzle allowing patient treatment with protons beams of energy between 230 and 70 MeV.
- The energy selection is performed using a passive energy degrader and an Energy Selection System (ESS) to ensure the correct beam energy.

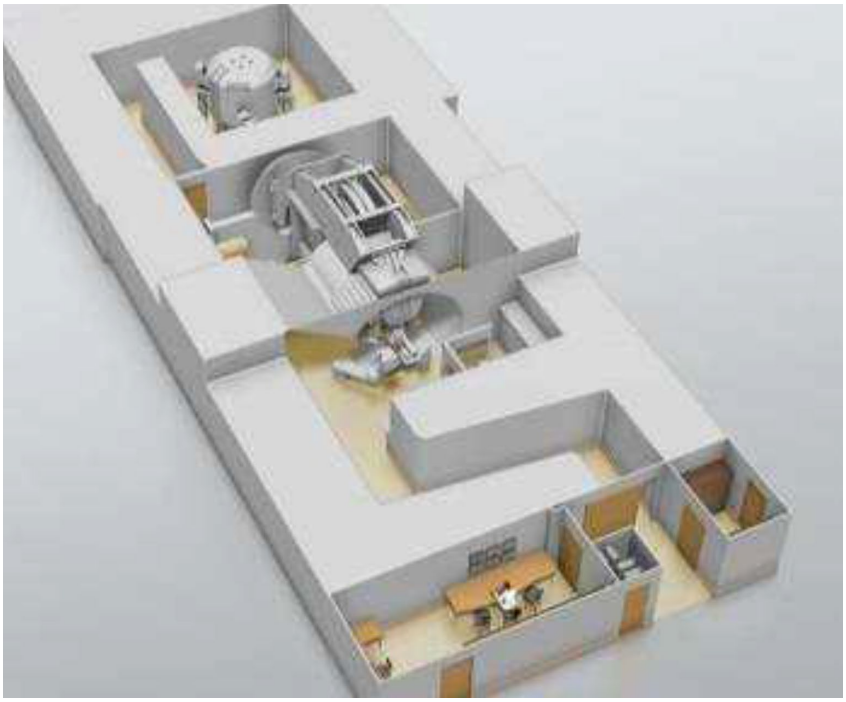


Figure 1. The *Proteus®ONE* facility.

During the use of the installation, different radiation sources occurs along the all beam line, starting from the beam production process and ending in the patient. These secondary radiation sources are due to beam losses during the beam transport and involve the use of different thick and thin target materials, from water up to Tantalum, and proton beam energies from 70 MeV up to 230 MeV. These radiation sources (neutrons and photons) have an impact on the surrounding environment as described in [1] :

1) Generation of large radiation doses around the neutron sources, requiring the construction of a suitable biological shielding around all the elements of the Proton Therapy (PT) equipment.
2) Generation of radioactive isotopes inside the installation, which have impact inside and outside the PT center. These radioactive products include :
    a. Equipment components made of metallic parts;
    b. Concrete shielding;

c. Cyclotron and beam line elements cooling water;

d. Air activation inside the cyclotron vault and the treatment room;

e. Earth activation.

In chapter 2, the various radiation sources encountered in a Proton Therapy (PT) center will be described. The Proteus®ONE shielding design is detailed in chapter 3, before the review, in chapter 4 of the different activations ad impacts for the air, environment, equipment and vault.

The conclusion will summarize all impacts on the environment and public.

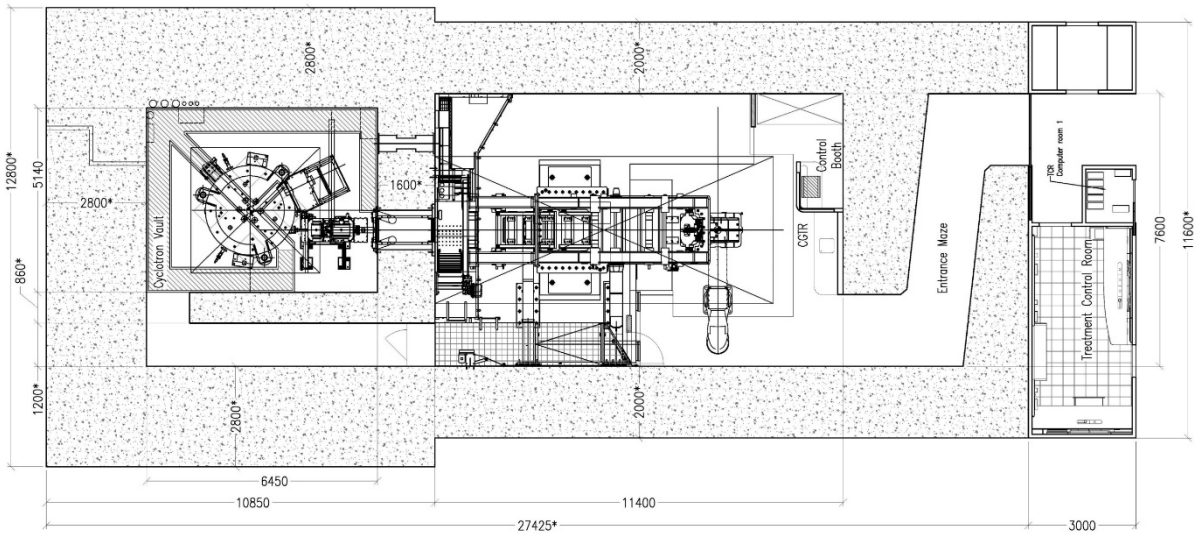


Figure 2 Schema of the installation

2. Radiation Sources

To estimate the environmental impact of a PT center, it is of paramount importance to estimate the different radiation sources along the all beam line. This chapter describes these different beam losses from the accelerator through the gantry up to the patient.

2.1. S2C2 cyclotron

The IBA accelerator used in the planned PT Center has a fix energy protons beam of 230 MeV. The following figure represents a general view of the S2C2 cyclotron.

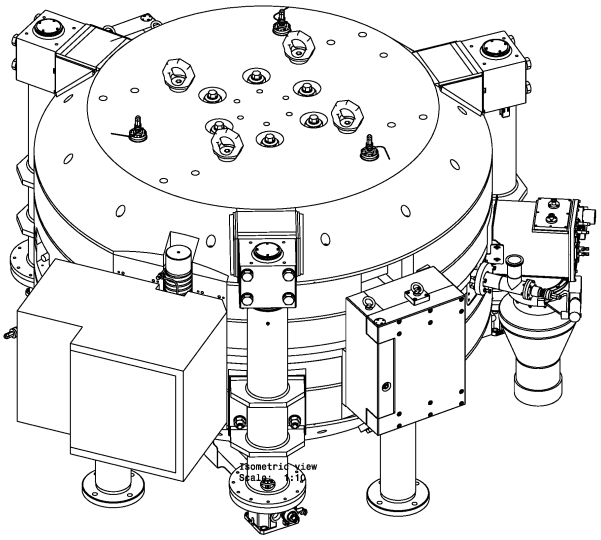


Figure 3. General view of the S2C2 cyclotron.

Beam losses occur during the acceleration process as well as during beam extraction from the cyclotron :

- Some losses occur close to the ion source, where protons have a very low energy below 10 MeV. Thanks to the low proton energy and the shielding of the cyclotron, these losses can be neglected.
- Beam losses occur close to the extraction radius, due to the de-focalized protons in the vertical plane and hitting the poles. These losses are uniformly distributed along the S2C2 circumference. At this place, a beam energy of 230 MeV needs to be considered.
- Finally, beam losses occur during the extraction stage, with protons at the maximal energy of 230 MeV. These losses are distributed at the pre-septum and the septum.

All details can be found in ref [IBA 6].

2.2. Compact Gantry (CGTR)

The beams extracted from the S2C2 have a fixed energy close to 230 MeV. The CGTR is then used to modulate the beam energy and limit the emittance before sending the beams to the patient.

A global view of the CGTR is presented in Figure 4. The main components interacting with the beam are:

- Quadrupoles Q2, Q3 used to (i) shape the beam extracted from the S2C2 and (ii) focus the beam on the degrader ;
- The energy degrader used to degrade the beam energy (from 230 up to 70 MeV) ;
- The collimator placed just after the degrader and used to intercept protons transmitted by the degrader with a large polar angle with respect to the initial beam trajectory ;
- Two quadrupoles Q1G, Q2G located inside the wall separating the S2C2 and the CGTR vaults. They are used to re-focus the beam after energy degradation ;
- Divergence and momentum slits are used to further limit the beam spread (in angle and momentum) to obtain a beam with the required kinematic properties.

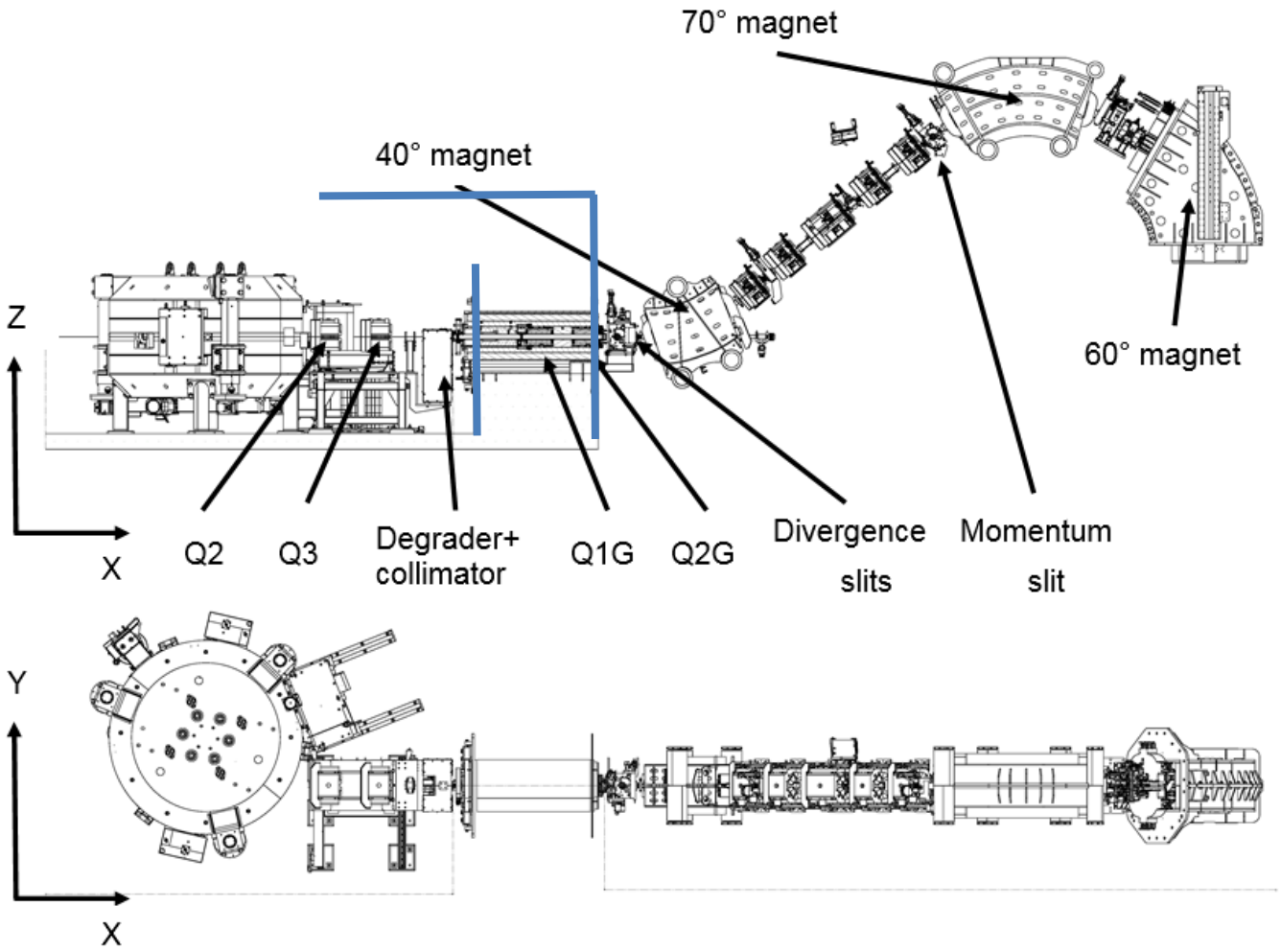


Figure 4. Global view of the S2C2 and CGTR.

The other components of the CGTR are dipole magnets needed to bend the beam (40°, 70° and 60° magnets) and quadrupole magnets to focus it. The fraction of beam hitting these elements is negligible and they will not be considered in this first environmental impact.

2.2.1. Quadrupoles Q2, Q3

Two extraction quadrupoles are located at the S2C2 exit, in the cyclotron vault. Due to the large emittance of the extracted beams, these two quadrupoles will intercept part of the beam at the maximal energy of 230 MeV.

2.2.2. Degrader

As shown in Figures 5 and 6, the energy degrader is constituted of a wheel rotating in front of the beam and equipped with blocks of variable thicknesses. Three types of materials are used for these blocks :

- Blocks 1 and 2 are made of Beryllium (Be) with a density of 1.85 g/cm^3 ;
- Blocks 3 to 8 are made of high-density graphite (C) with a density of 1.7 g/cm^3 ;
- Block 9 is made of aluminum with density of 2.7 g/cm^3.

The Be blocks are used for the smallest transmitted energies, below 130 MeV, in order to reduce the multiple scattering compared to graphite. The Al block is limited to the largest transmitted beam energies, typically above 220 MeV.

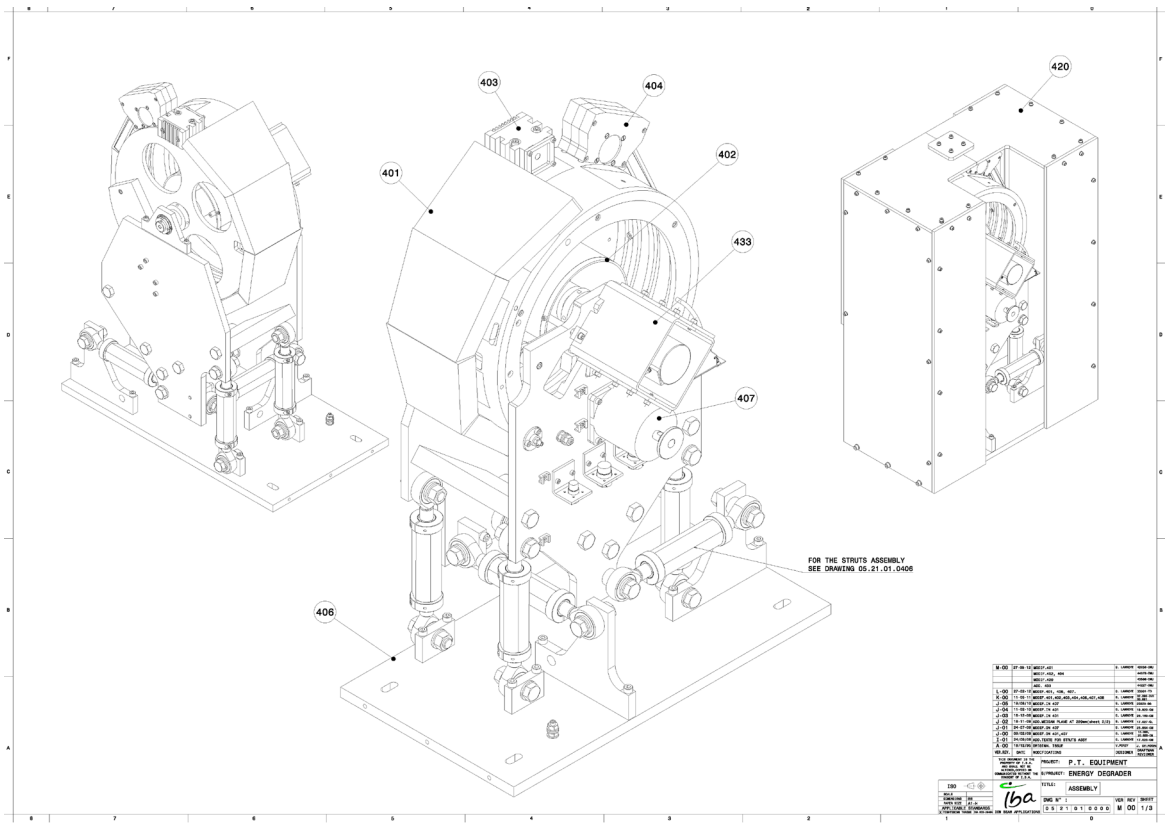


Figure 5. Global view of the energy degrader.

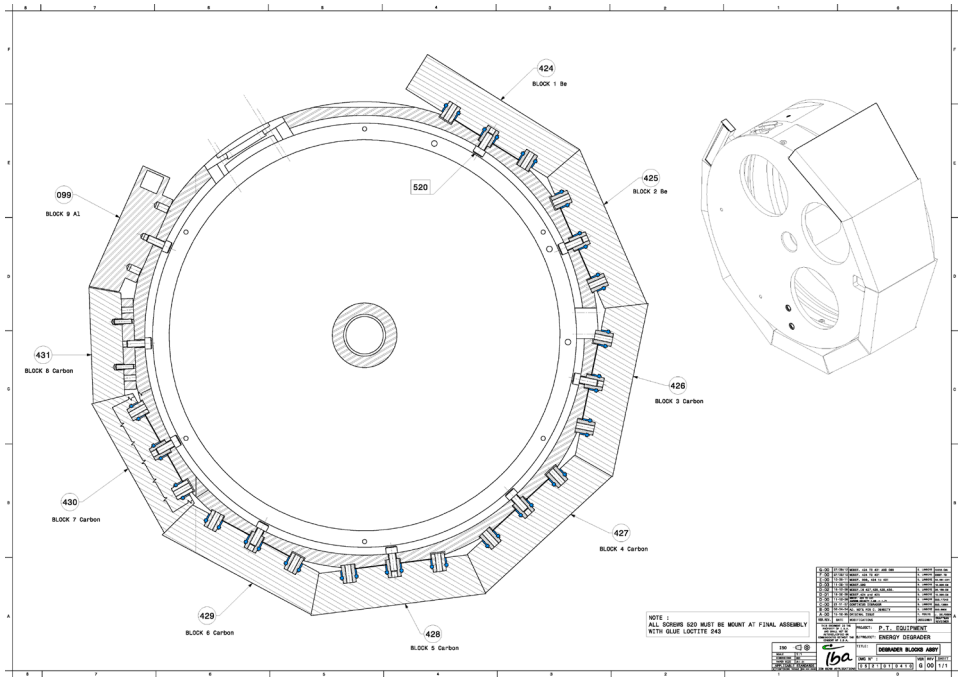


Figure 6. Details of the Be, C and Al blocks of the energy degrader.

2.2.3. Collimator

A collimator is placed right after the degrader to reduce the beam angular divergence. A global view of the collimator assembly is shown in the Figure 7. The collimator itself consists of a cylinder made of Tantalum with a 10 mm diameter hole in the middle.

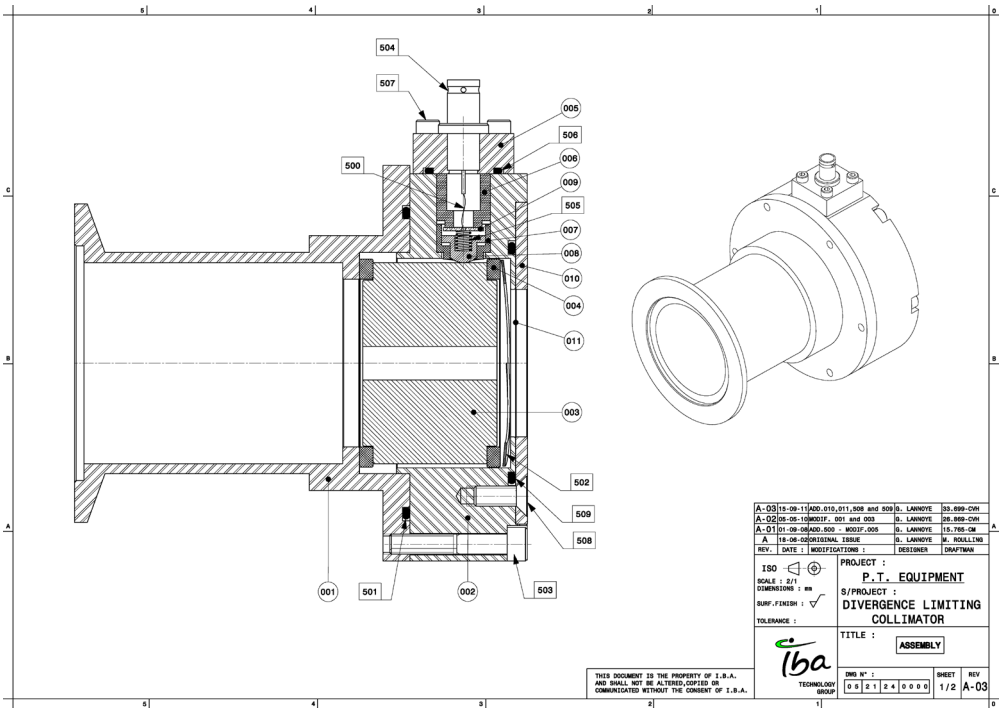


Figure 7. Global view of the divergence limiting collimator.

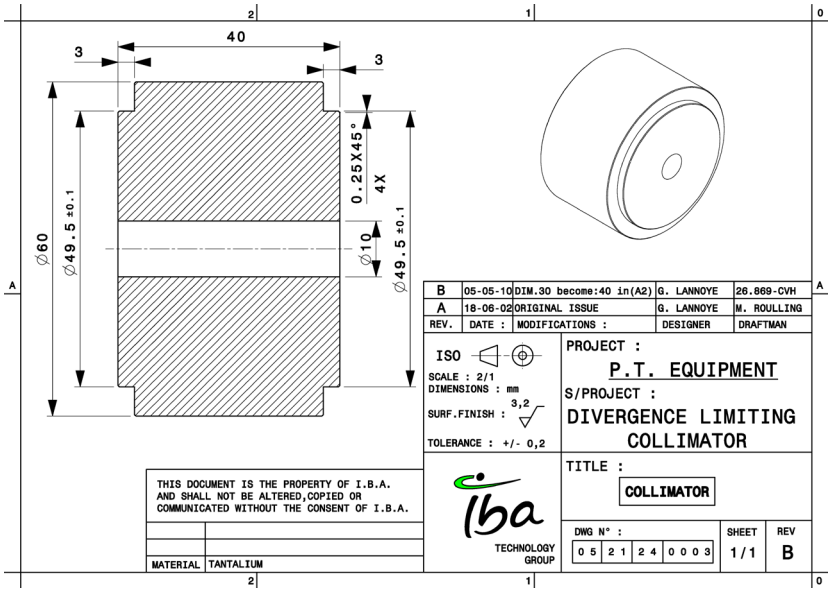


Figure 8. Detail of the collimator Tantalum.

2.2.4. Q1G and Q2G

Two quadrupoles are located, in the wall, just after the degrader. Due to the large emittance of the extracted beams, these two quadrupoles will intercept part of the beam up to the maximal energy of 230 MeV.

2.2.5. Divergence slits

Thereafter, the divergence limiting slits are placed to further reduce the beam divergence.

As shown in the Figure 9, these slits are made of two sets of “jaws”, limiting the beam divergence either in the horizontal or the vertical planes. In practice, protons are absorbed in Nickel (Ni) blocks with a thickness of 6 cm.

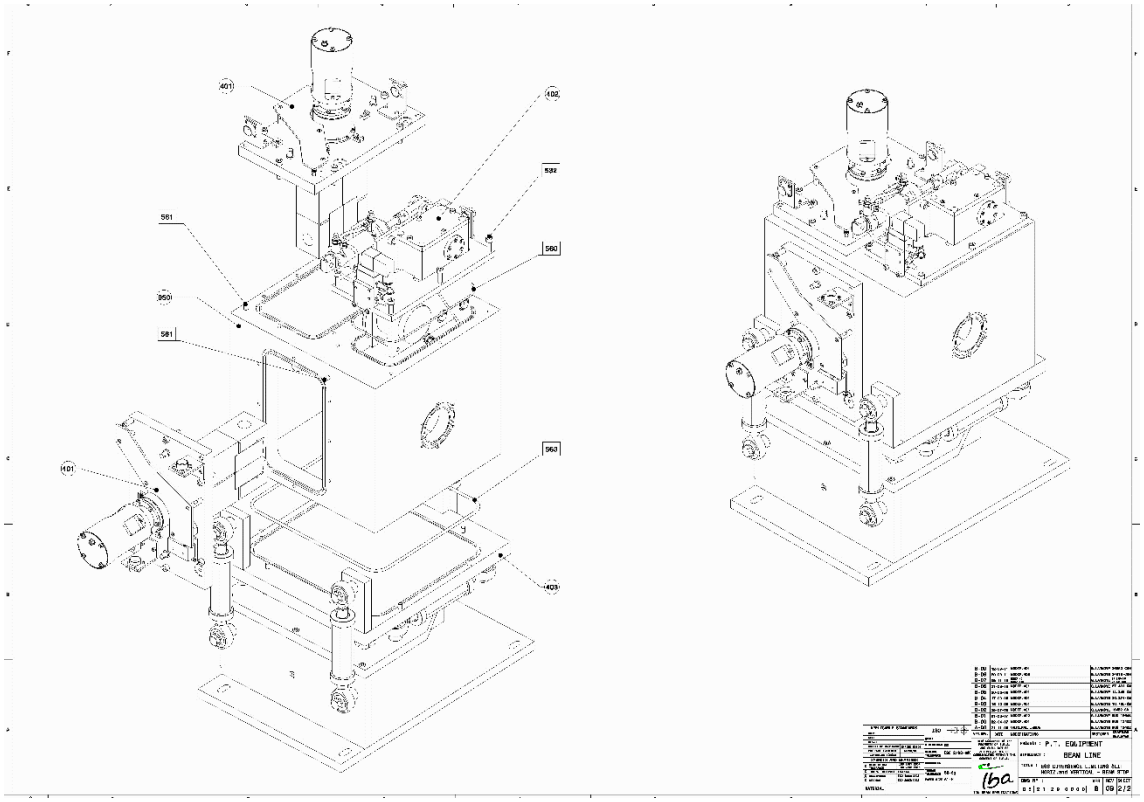


Figure 9. Global view of the divergence limiting slits.

2.2.6. Momentum slit

An additional slit is located before the 70° dipole magnet. This slit allows a reduction of the beam momentum spread by absorbing protons transmitted with a momentum far from the average momentum.

The momentum limiting slit has a very similar design to the divergence limiting slits but with only one pair of jaws aligned along the vertical axis. Again, the protons are stopped in the Ni blocks.

3. Radiation Shielding Design

3.1 Requirement of Local Authorities

To design the needed shielding surrounding the PT installation, it is essential to know how those installation will be used during its expected lifetime (+\- 20). This usage scenario allows the determination of the beam workloads delivered at the isocenter on a yearly basis. The Patient Model is described in section 3.2.2. Once the beam workload is known, the amounts of beam losses at the different locations along the beamline can be estimated. These results will also be used to determine the activation of the different elements of the installation (equipment + building) as well as the environmental impact.

In addition, most of regulations requires that the maximal dose for public area (around the installation) do not exceed 1 mSv/y, and this in the worst case condition of the system utilization. In order to take into account the multiple sources encountered by the public in a year, the annual dose constraint used as reference for this study is 0.4 mSv/y.

As basis for this environmental impact study, the worse conditions leading to maximum exposure in Clinical Mode and in Service Mode are used by IBA. For example, the shielding has been calculated assuming a maximal beam current at the nozzle exit of 5 nA in service mode with an proton energy of 230 MeV. This value cannot be overridden in any case and is limited by the system current readout feedback.

3.2 Shielding Calculation

3.2.1. Methodology

The Proteus®ONE shielding design is based upon the use of the MCNPX Monte Carlo radiation transport code. This code developed by Los Alamos National Laboratory (USA) is the reference transport code for nuclear reactor and accelerator shielding design.

A conservative patient case mix is established (see 3.2.2.) based upon IBA clinical experience in Proton Therapy in many different countries. It defines all the assumptions and shielding needed to respect the annual radiation doses received and dose rates outside the vault.

An inventory is made of all possible radiation sources encountered inside the Proteus®ONE system (see previous chap 2.). It includes beam losses occurring inside the S2C2 cyclotron and along the beamline, including the final beam absorber (the patient).

A detailed modelling of the Proteus®ONE vault and the major pieces of equipment is introduced in MCNPX. Given the significant attenuation of secondary neutrons and photons in the metallic elements, the complete and correct description of the cyclotron and the compact gantry magnets is of vital importance to obtain an accurate dose evaluation outside the vault.

The ambient equivalent dose rate H*(10) is computed using fluence-to-dose conversion coefficients defined for neutrons and photons by the International Commission on Radiological Protection (ICRP) [2]. This operational quantity gives a reasonable estimation of the shielding and is commonly used to assess compliance with the legal limits used in most countries. The neutron and photon contributions are summed together and the total equivalent dose is compared to the legal limits.

Shielding requirements are established based upon a limit of 400 µSv for the annual equivalent dose H*(10) everywhere outside the vault, including above the roof. This limit is a factor 2.5 below the common legal limit of 1 mSv/y for public area in order to take into account uncertainties on different simulation parameters such as the concrete atomic composition or the choice of nuclear models needed to describe the proton and neutron interactions with matter.

As the shielding design is entirely based on Monte Carlo predictions, it is important to validate these predictions with measurements. Several tests campaigns have been conducted at the WPE Proteus®PLUS facility located in Essen, Germany. As both types of systems use the same beam of 230 MeV protons, these measurements are directly applicable to the Proteus®ONE system. All measurements (see Figure 10) were performed using a wide-energy range neutron detector Wendi-2 from Thermo Scientific, which can detect neutrons up to 5 GeV (25 meV – 5 GeV, 0.01 – 100 mSv/h).

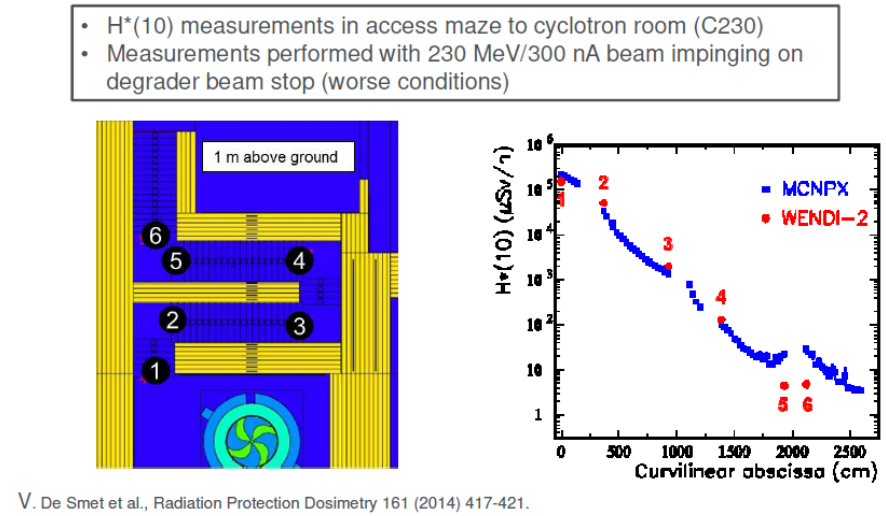


Figure 10. H*(10) measurements and MCNPX predictions around Cyclotron room

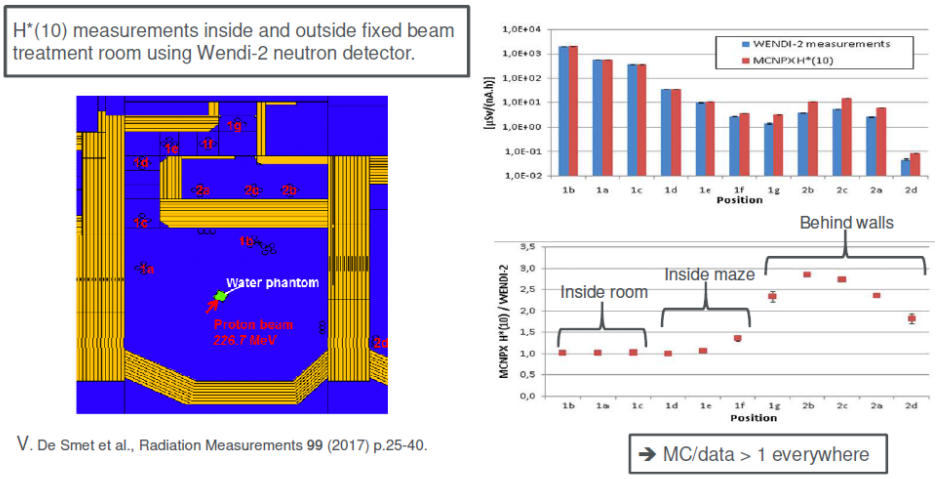


Figure 11. Positions of H*(10) measurements inside maze and outside Gantry Treatment Room for Pencil Beam Scanning Mode

These results confirm, that the use of MCNPX does not lead to an underestimation of the neutron dose rates outside the PT shielding in comparison with the on-site measurements. Monte Carlo simulation is thus a useful tool to determine the shielding requirements around a PT system.

More details may be found in the references [3] and [4].

3.2.2. Patient Model

Ideally, the patient model needs to be conservative and should to anticipate all type of treatments who can be performed during the all lifetime of the facility. In correlation, the shielding placed at the construction should also meet all safety requirements for the all life of the PT System. Therefore, this patient model should be envisaged as a maximal treatment scenario for the next 20 or 30 years.

As this patient model mix can vary from one customer to another, IBA is using a "patient model" based upon its previous experience with the *Proteus®PLUS* facilities in the different installation in Europe, Asia and North America.

We consider the following usage for a *Proteus®ONE* system:

- Clinical operations = 4,800 hours/year
  - 16 hours/day (2 shifts)
  - 6 days/week
  - 50 weeks/year
- # Patients treated per year = 435
- # Fractions of 2 Gy delivered per year = 16,660
- QA activities performed every morning to verify system with low-, medium- and high-range beams.

The yearly activity is divided into 4 types of indications corresponding to major groups of tumors :

- Head & Neck tumors ;
- Lung tumors ;
- Pelvic sarcoma ;
- Pediatric (Central Nervous System only).

The irradiation parameters corresponding to these different indications are described in the following Table 1, together with the parameters used for the morning QA checks. The tumor volumes are based on average values obtained from the book "Practical Radiotherapy Planning" by J. Dobbs et al. [5].

Given the therapeutic doses delivered per patient for each indication, including the doses used for morning QA, and using a total dose per patient (66 Gy - 86 Gy), the total doses obtained are thereafter multiplied by the number of patients expected for each indication, in order to obtain the total annual doses delivered for each treatment field. The total assumptions are that a total of 33,324 Gy are deposited at the isocenter for the patient treatments and 3,600 Gy for the morning QA activities.

Indication	Max Range (g/cm²)	Min Range (g/cm²)	Field size (cm²)	Modulation (g/cm²)	Tumor Volume (cm³)
Head & Neck	18	8	7.7 x 7.7	10	600
	12	2	7.7 x 7.7	10	
Lung Tumor	20	8	7.9 x 7.9	13	810
	17	4	7.9 x 7.9	13	
Pelvic Sarcoma	32	20	10.1 x 10.1	12	1218
	27	15	10.1 x 10.1	12	
Pediatric (CNS)	18	8	18.6 x 18.6	10	3460
	12	2	18.6 x 18.6	10	
QA – low range	14	4	10 x 10	10	1000
QA – medium range	23	13	10 x 10	10	1000
QA – high range	32	22	10 x 10	10	1000

Table 1. Irradiation parameters for the different clinical indications and morning QA irradiations

The clinical parameters were then transposed into physical parameters usable into our Monte Carlo simulations. Each treatment field was described in the RayStation Treatment Planning System (TPS) from RaySearch. The TPS then provides a list of beam energies and weights to obtain the required Spread-Out Bragg Peaks (SOBP) corresponding to each field of treatment.

Next, each SOBP is simulated in a water phantom using the code MCNPX 2.7.0. These simulations are needed to establish the relationship between the deposited dose in the tumor volume and the beam workload, i.e. the number of incident protons. MCNPX results are in good agreement with expectations, exhibiting a rather flat SOBP in the tumor region. These plateaus are fitted to a constant value to determine the average dose deposition obtained in the tumor volume per unit of beam workload, expressed in Gy/nC.

Once these averaged dose values are computed, it becomes trivial to determine the beam workloads (# nC) needed to deposit 1 Gy in the tumor volume for each indication. These values range between 6.3 nC for the smallest tumor volumes up to 42.1 nC for the largest ones. This quantity W (1Gy) is directly proportional to the field size and can be adapted to other field sizes if needed.

Finally, the yearly doses at isocenter determined for each indication can be multiplied by W(1Gy) to obtain the yearly workloads W(iso). They are expressed in nA.h, with 1 nA.h = 3,600 nC. The total annual workloads calculated is thus of 136.42 nA.h.

This approach is in accordance with the recommendation given in [6], which says in essence that the discharge shall be “assessed on the basis of estimated throughput”.

3.3. Results

The annual doses are computed for the standard ProteusONE layout, without taking into account the presence of additional “ground shielding” outside the vault walls. The complete results are described in IBA document M-ID 64063 [IBA 7]. Table 2 shows the maximal ambient dose equivalent values (H*(10)) obtained at different locations. As stated in section 3.1, all annual doses remain below 400 µSv.

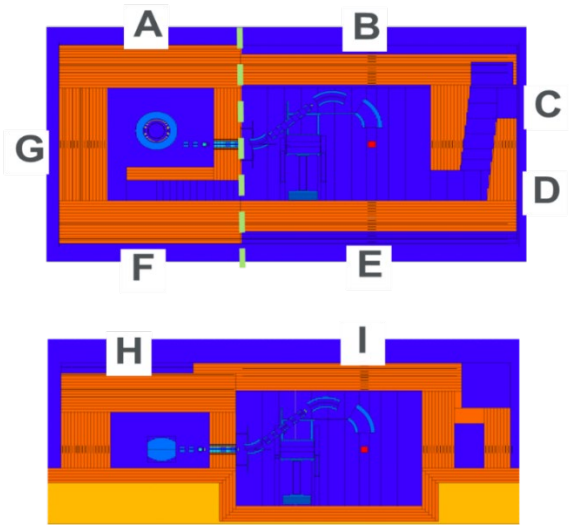


Figure 12. Point of interest in the dose calculation – see following table.

Location	Barrier thickness (m)	Max. H*(10) (µSv/year)
A	2.8	392.2
B	2.0	148.5
C	1.8 to 2.4	275.5
D	1.4 to 2.0	108.0
E	2.0	325.2
F	2.8	347.9
G	2.8	397.8
H	2.5	390.2
I	1.9	233.4
J	2.1	356.6

Table 2. Maximal annual H*(10) doses obtained at the different locations outside the *Proteus®ONE* vault.

Conclusion

Impact of gamma and neutrons dose rates during the working life of the installation are :

Impact of gamma and neutron dose rates (§ 3)	
Public	< 400 µSv/y
Radiation Worker (in PT Center)	< 20 mSv/year averaged over 5 years
Building	-
Environment	-

4. Environmental Impacts

At the time of preparation of this Environmental Impact Assessment, 7 installations of the same model ProteusONE® are operational in USA, Europe and Japan (see Figure 2.a)

The use of a PT center can have various environmental impacts ([6] chap 1.5 and 3.12). Those impacts must be calculated, compared with local and international regulations and appreciated at their real level.

The environmental impacts are dealing with external and internal exposition. The first one is related to dose rate around the PT Center and the second one is link with the activation products generated during the working life of the installation.

Because of the intense protons and neutrons fields generated around the PT installation, various stable elements can be activated. Depending of the element, the produced radioisotopes can have short or long lifetimes.

In [1], the IAEA addresses the following activation aspects:

- Air circulating inside the cyclotron room and the treatment rooms ;
- Cooling water used in the cyclotron and the beam line elements during operation and after decommissioning) ;
- Ground surrounding the facility and beneath the cyclotron room.

This report further considers

- Concrete (during operation and after decommissioning) ;
- Metallic parts (after decommissioning)

For this reason, at the end of the use of the PT Center, the installation must be dismantled with precautions.

The beam workload (cfr. patient model) delivered at the isocenter on a yearly basis is the base for the estimation of all activations process not solely in the equipment and building but also for air activation and potential soil and cooling water activations.

4.1 During the working life

4.1.1 Activation Products - Equipment's

The interaction of neutrons with material will lead to the production of new isotopes according to (n, gamma) reactions. Theses isotopes can be stable and have thus no effect at all. Nevertheless, there are also many radioisotopes who can be produced by this neutron capture.

The table 3 present major reactions on metallic compounds :

Radionuclide	Possible Reaction	Cross-section	Half life	Abundance (%)
^{134}Cs	$^{133}Cs\ (n,\gamma)\ ^{134}Cs$	29 barn	2.06 year	100
^{60}Co	$^{59}Co\ (n,\gamma)\ ^{60}Co$	37 barn	5.3 year	100
^{60}Co	$^{60}Ni\ (n,p)\ ^{60}Co$		5.3 year	26
^{59}Fe	$^{58}Fe\ (n,\gamma)\ ^{59}Fe$	1.15 barn	44 days	0.3
^{65}Zn	$^{64}Zn\ (n,\gamma)\ ^{65}Zn$	0.78 barn	244 days	49
^{54}Mn	$^{55}Mn\ (n,2n)\ ^{54}Mn$	910 mbarn at $E_n = 18MeV$	312 days	100
^{54}Mn	$^{54}Fe\ (n,p)\ ^{54}Mn$	590 mbarn at $E_n = 10MeV$	312 days	6
^{108}Ag	$^{107}Ag(n,\gamma)^{108}Ag,\ ^{108m}Ag$	36 barn	127 year	52
^{110}Ag	$^{109}Ag(n,\gamma)^{110}Ag,\ ^{110m}Ag$	91 barn	249 days	48
^{123}Sn	$^{122}Sn(n,\gamma)^{123}Sn$	0.15 barn	129days	4.6
^{125}Sn	$^{124}Sn(n,\gamma)^{125}Sn$	0.13 barn	9 days	5.6
^{22}Na	$^{23}Na\ (n,2n)^{22}Na$	17 mbarn at $E_n = 15MeV$	2.6 year	100
^{22}Na	$^{27}Al\ (n,2p4n)^{22}Na$	10 mbarn at $E_n = 25MeV$	2.6 year	100

Table 3. Possible reactions on metal [7]

Moreover, copper is also part of the installation. Copper is naturally composed of two stable isotopes : ^{63}Cu and ^{65}Cu.The neutron and protons capture will lead to following reactions :

Radionuclide	Possible Reaction	Reaction product	Half life	Abundance (%)
^{63}Cu	(n,γ)	^{64}Cu	12.7 hours	69.17
	(n,α)	^{60}Co	5.3 years	
^{65}Cu	(n,2n)	^{64}Cu	12.7 hours	30.83
	(n,p)	^{65}Ni	2.5 hours	
	(p,n)	^{65}Zn	244 days	

Table 4. Interactions of Cu with n fluxes

Other metallic parts in the cyclotron and the installation (beam line, structural pieces, …) may also be activated by direct impact of the proton beam or by neutrons fluxes. The following elements will be formed in all metallic pieces :

Radionuclide	Half-life
^{60}Co	5.3 years
^{57}Co	271 days
^{58}Co	70 days
^{54}Mn	312 days
^{65}Zn	244 days
^{22}Na	2.6 years

Table 5. Radionuclides produced in metallic parts

The radioisotopes interest produced in metallic parts to be considered are: ^{22}Na, ^{57}Co, ^{60}Co, ^{65}Zn and ^{54}Mn, due to their long half-life. The other isotopes have either a short half-life or the parent isotope has a relatively small cross section.

The dose rate generated by these radioisotopes must be considered in the vault only. This is the role of the RPO (Radio Protection Officer) on site, during the life of the installation (application of the ALARA principle). The impact of these dose rates on the public who can be near the installation does not needs to be considered, due to the presence of shielding material use to for higher energy beams and higher dose rates.

Conclusion

Impact of activated equipment during the working life of the installation, on:

Impact of activated equipment during the working life (§ 4.1.1)	
Public	-
Radiation Workers (in PT Center)	-
Building	-
Environment	-

4.1.2 Activation Products - Concrete

Inside the cyclotron vault, the production of long-lived isotopes by neutron induced reactions on concrete can result from two different mechanisms :

1. With low-energy or thermal neutrons, neutron capture processes (n,γ) are responsible for the production of long-lived isotopes such as ^{152}Eu, ^{154}Eu, ^{134}Cs or ^{60}Co. Their properties are summarized in the table 5.
2. Nuclear reactions can also occur with high-energy neutrons hitting matter. These spallation processes are essentially responsible for the production of ^{22}Na (see Table).

Radionuclide	Possible Reaction	Cross section	Half life	Abundance (%)[b]
^{152}Eu	^{151}Eu (n,γ) ^{152}Eu	9198 barn	13.33 years	48
^{154}Eu	^{153}Eu (n,γ) ^{154}Eu	312 barn	8.8 years	52
^{134}Cs	^{133}Cs (n,γ) ^{134}Cs	29 barn	2.06 years	100
^{134}Cs	^{134}Ba (n,p) ^{134}Cs	9 mbarn at E_n = 16 MeV	2.06 years	2
^{60}Co	^{59}Co (n,γ) ^{60}Co	37 barn	5.3 years	100
^{46}Sc	^{45}Sc (n,γ) ^{46}Sc	27 barn	83 days	100
^{133}Ba[a]	^{132}Ba (n,γ) ^{133}Ba	7 barn	10.5 years	0,1
^{133}Ba[a]	^{133}Cs (p, n) ^{133}Ba	0,16 mbarn at E_n = 12 MeV	10.5 years	100
^{54}Mn	^{55}Mn (n,2n) ^{54}Mn	910 mbarn at E_n = 18 MeV	312 days	100
^{54}Mn	^{54}Fe (n,p) ^{54}Mn	590 mbarn at E_n = 10 MeV	312 days	6
^{22}Na	^{23}Na (n,2n) ^{22}Na	40 mbarn at E_n = 15 MeV	2.6 years	100
^{22}Na	^{27}Al (n,2p4n) ^{22}Na	10 mbarn at E_n = 25 MeV	2.6 years	100
^{137}Cs	^{136}Ba (n,γ) ^{137m}Ba → ^{137}Cs	0.4 barn	30 years	8
^{137}Cs	^{137}Ba (n,p) ^{137}Cs	3.7 mbarn at E_n = 16 MeV	30 years	11

[a] only detected where "barite" concrete is used
[b] "Abundance relates to the abundance of target nuclide in the natural element

Table 6. Possible reactions on concrete [7]

Even if the abundance of ^{151}Eu (stable isotope and naturally occurring) in concrete is low, this element presents a large capture cross section for thermal neutrons and generates ^{152}Eu via (n,γ) reactions. This element is the most constraining one with a half-life of 13.3 years. All other elements present in the concrete has either a low cross section or are short lived.

While decommissioning the vault (see § 4.2), one has also to consider reinforcement rod activation. Three isotopes must be considered: ^{60}Co, ^{55}Fe and ^{59}Fe. While ^{55}Fe doesn't present a major problem due to its short half-life (44 days) it is not the case for ^{60}Co (5,3 years) and ^{55}Fe (2,7 years). A decommissioning plan is being presented.

The activation of standard concrete has been the subject of a detailed analysis described in the IBA report [IBA1].

To potentially reduce the concrete activation, IBA has developed a special concrete called "low-activation concrete" that will be presented in Section 4.2.2 [IBA2].

Conclusion

Impact of activated equipment during the working life of the installation is :

Impact of activated concrete during the working life (§ 4.1.2)	
Public	-
Radiation Workers (in PT Center)	-
Building	-
Environment	-

4.1.3 Activation Products - Air

Secondary neutrons produced inside the S2C2 and treatment vaults can interact with the air atoms and can produce unstable isotopes. The neutron-induced air activation can be divided into two groups :

- The fast neutrons interacting with ^{16}O and ^{14}N (stables and naturally occurring in air) atoms can generate various unstable isotopes by so-called spallation reactions ;
- The capture of thermal neutrons by ^{40}Ar (stable and naturally occurring in air) present in air leads to the production of ^{41}Ar.

These two processes are treated separately in the following sections.

The gap between the nozzle and the patient is small, limiting the ozone production. This phenomenon can thus be neglected.

4.1.3.1 Spallation Products

The interactions of secondary neutrons with the ^{16}O (23.2 % in dry air [8]) and ^{14}N (75.5 %) atoms present in air lead to the production of a few unstable isotopes. The most frequently produced isotopes with a half-life larger than 1 min are: ^{3}H, ^{7}B, ^{11}C, ^{13}N, ^{14}O and ^{15}O [ref. 9 and 10].

Isotopes	From ^{16}O	From ^{14}N
^{3}H	$^{16}O(n,X)^{3}H$	$^{14}N(n,X)^{3}H$
^{7}Be	$^{16}O(n,X)^{7}Be$	$^{14}N(n,X)^{7}Be$
^{11}C	$^{16}O(n,X)^{11}C$	$^{14}N(n,X)^{11}C$
^{13}N	$^{16}O(n,X)^{13}N$	$^{14}N(n,X)^{13}N$
^{15}O	$^{16}O(n,X)^{15}O$	

Table 7. Isotopes produced from ^{16}O and ^{14}N

The production yield R_i for isotope i is given by the relationship [11] :

$$R_i = 100 \times Q_{HE} \times N_A \times A_i^{-1} \times f_w^i \times \rho_{ai} \times \sigma_i \times L \qquad (4.1)$$

where:

- Q_{HE} = fast neutron emission yield (above production threshold for isotope i).
- N_A = Avogadro number
- f_w^i = mass fraction of the parent element i in air
- A_i = atomic weight for the parent element i
- ρ_{air} = air density
- σ_i = production cross section for isotope i
- L = Neutron paths lengths in air (in meters)

Isotope	$T_{1/2}$	λ (s^{-1})	Parent	s (mb)
^{3}H	12.32 years	$1.8\ 10^{-9}$	N	30
			O	30
^{7}Be	53.22 days	$1.51\ 10^{-7}$	N	10
			O	5
			Ar	0.6
^{11}C	20.33 min	$5.68\ 10^{-4}$	N	10
			O	0.7
			Ar	0.7
^{13}N	9.96 min	$1.16\ 10^{-3}$	N	10
			O	9
			Ar	0.8
^{14}O	1.18 min	$9.79\ 10^{-3}$	O	1
			Ar	0.06
^{15}O	2.04 min	$5.7\ 10^{-3}$	O	40

Table 8. Half live, parent and Cross Sections for Neutron induced radionuclides

In the absence of air renewal, the saturated activity would be equal to R_i. However, the air is regularly renewed so that most produced isotopes never reach saturation. This air renewal is taken into account by replacing the isotope decay constant λ by an effective decay constant $\lambda' = \lambda + r$

where r represents the number of air renewal per unit of time [10]. Then, the saturated activity becomes :

$$A'_{sat} = A_{sat} \times \lambda / \lambda' = A_{sat} \times \lambda / (\lambda + r) \qquad (4.2)$$

Considering an air renewal every 10 minutes, the constant r is equal to 1/600 s^{-1}. The air renewal will be measured and recorded yearly

4.1.3.2. ^{41}Ar Production

To compute the reaction rate $^{40}Ar(n_{th},\gamma)^{41}Ar$ (1.8 h), calculations are based on the procedure described in publication NCRP-144 [11]:

$$R = F_{th} \times N_A \times A^{-1} \times f_w \times \rho_{air} \times \sigma \qquad (4.3)$$

where:

- N_A = Avogadro number
- A = atomic weight for Argon-40
- f_w = mass fraction of Ar in air = 0.013
- ρ_{air} = air density
- σ= cross section = 660 mb
- F_{th} = thermal neutron flux inside the room

The thermal flux is given by the relationship [11]:

$$F_{th} = 1.25\ Q_F / S \qquad (4.4)$$

where Q_F is the total neutron yield (n/s) and S the inner surface of the room

4.1.3.3. Determination of neutron fluxes

As seen from relationships (4.1) and (4.3), the determination of isotopes production rates relies on an estimation of the neutron fluxes.

The total annual neutron fluences (in 1/cm²) were computed considering the IBA patient model and the different radiation sources present inside the installation. Then, an average neutron flux was computed by dividing the total fluence by 1.728 10^7 seconds, corresponding to the 4,800 hour/year operating time. This is in line with IAEA [6], who recommends estimating the amount of produced isotopes using clinical standard operation.

4.1.3.4. Results

The neutron fluxes used are the averaged neutron fluxes obtained in S2C2 and in the treatment room. Once the neutron fluxes are known, one can apply the relationships (4.1) and (4.3) to determine the production rates of the various isotopes. For the spallation isotopes, the product QHE x σ_i is replaced by the integral $\int f(E_n).\sigma_i(E_n).dE_n$ using the differential cross sections $\sigma_i(E_n)$ given in [9 and 10].

Finally, the saturated activities are obtained by multiplying the production rates R_i by the constants $\lambda_i / (\lambda_i + r)$.

The saturated activities obtained for the various isotopes produced inside the S2C2 and treatment rooms are presented in the two following tables.

Nuclide	Parent	Production yield (atoms/m³.s)	Saturated Activity (Bq/m³)	Total Activity (Bq/m³)
^{3}H	N	$7.03\ 10^{3}$	$7.60\ 10^{-3}$	$8.27\ 10^{-3}$
	O	$6.19\ 10^{2}$	$6.68\ 10^{-4}$	
^{7}Be	N	$6.12\ 10^{2}$	$5.54\ 10^{-2}$	$7.04\ 10^{-2}$
	O	$1.65\ 10^{2}$	$1.49\ 10^{-2}$	
^{11}C	N	$8.99\ 10^{2}$	$2.29\ 10^{2}$	$3.80\ 10^{2}$
	O	$5.97\ 10^{2}$	$1.52\ 10^{2}$	
^{13}N	N	$5.16\ 10^{3}$	$2.12\ 10^{3}$	$2.16\ 10^{3}$
	O	$1.04\ 10^{2}$	$4.25\ 10^{1}$	
^{15}O	O	$2.07\ 10^{3}$	$1.60\ 10^{3}$	$1.60\ 10^{3}$

Table 9. Saturated activities of the isotopes produced inside the S2C2 room.

For H-3, the produced activity is of the same level than the naturally occurring Tritium (6 - 61.10^{-3} Bq/m³).

Nuclide	Parent	Production yield (atoms/m³.s)	Saturated Activity (Bq/m³)	Total Activity (Bq/m³)
^{3}H	N	$2.23\ 10^{1}$	$2.41\ 10^{-5}$	$2.59\ 10^{-5}$
	O	$1.66\ 10^{0}$	$1.79\ 10^{-6}$	
^{7}Be	N	$1.43\ 10^{0}$	$1.29\ 10^{-4}$	$1.64\ 10^{-4}$
	O	$3.81\ 10^{-1}$	$3.45\ 10^{-5}$	
^{11}C	N	$1.95\ 10^{0}$	$4.95\ 10^{-1}$	$8.69\ 10^{-1}$
	O	$1.47\ 10^{0}$	$3.75\ 10^{-1}$	
^{13}N	N	$1.46\ 10^{1}$	$6.00\ 10^{0}$	$6.09\ 10^{0}$
	O	$2.10\ 10^{-1}$	$8.63\ 10^{-2}$	
^{15}O	O	$5.14\ 10^{0}$	$3.98\ 10^{0}$	$3.98\ 10^{0}$

Table 10. Saturated activities of the isotopes produced inside the treatment room.

^{41}Ar	Production Yield (atoms/s.cm³)	Saturated Activity (Bq/m³)
S2C2	$1.50\ 10^{-2}$	$8.91\ 10^{2}$
CGTR	$9.01\ 10^{-6}$	$5.35\ 10^{-1}$

Table 11. Saturated activities for ^{41}Ar in S2C2 room and in Gantry Room.

The derived saturation activities for the different unstable nuclides need to be compared to the release limits imposed by the legislation.

Moreover, the presence of people (from public) in the CGTR is not a regular situation due to the fact that the CGTR is a restricted area only accessible to workers (from PT Center) and to patients.

We can compare our results to the recent 'Ordonnance sur la radioprotection' (ORaP) published in April 2017 by the Swiss Confederation [12]. The appendix 3 of this publication provides a comprehensive list of radioprotection data for all the major nuclides, resulting from a compilation of IAEA safety guides and ICRP publications. IAEA [12] suggests, that the discharge limit can either be expressed as dose/year (Sv/a) or discharge quantity (Bq/m³).

Annexe 3
(art. 2, al. 1, let. j, l et m, ainsi que 194, al. 3)

Données pour la radioprotection opérationnelle, limites de libération, limites d'autorisation et valeurs directrices

Les explications concernant les différentes colonnes et les notes de bas de page sont données sous le tableau.

Nucléide	Période	Mode de désintégration / rayonnement	Grandeurs d'appréciation					Limite de libération		Limite d'autorisation	Valeurs directrices		Nucléide de filiation instable
			e_{inh} Sv/Bq	e_{ing} Sv/Bq	h_{10} (mSv/h)/ GBq à 1 m de distance	$h_{0,07}$ (mSv/h)/ GBq à 10 cm de distance	$h_{c,0,07}$ (mSv/h)/ (kBq/cm²)	LL Bq/g		LA Bq	CA Bq/m³	CS Bq/ cm²	
1	2	3	4	5	6	7	8	9		10	11	12	13
H-3, OBT	12.32 a	β⁻	4.10 E-11	4.20 E-11	<0.001	<1	<0.1	1.E+02		1.00 E+08	2.00 E+05	1000	
H-3, HTO		β⁻	1.80 E-11	1.80 E-11	<0.001	<1	<0.1	1.E+02		3.00 E+08	5.00 E+05	1000	
H-3, gaz [7]		β⁻	1.80 E-15		<0.001	<1	<0.1			3.00 E+12	5.00 E+09		
Be-7	53.22 d	ec / ph	4.60 E-11	2.80 E-11	0.008	<1	0.1	1.E+01		1.00 E+08	2.00 E+05	100	
Be-10	1.51 E6 a	β⁻	1.90 E-08	1.10 E-09	<0.001	2000	1.6	1.E+02		3.00 E+05	4.00 E+02	3	
C-11	20.39 min	ec, β⁺/ ph	3.20 E-12	2.40 E-11	0.160	1000	1.7	1.E+01	[1]	7.00E+07	7.00 E+04 [3]	3	
C-11 monoxyde			1.2 E-12							7.00E+07	7.00 E+04 [3]		
C-11 dioxyde			2.2 E-12							7.00E+07	7.00 E+04 [3]		
C-14	5.70 E3 a	β⁻	5.80 E-10	5.80 E-10	<0.001	200	0.3	1.E+00		9.00E+06	1.00 E+04	30	
C-14 monoxyde			8.00 E-13							6.00E+09	1.00 E+07		
C-14 dioxyde			6.50 E-12							8.00E+08	1.00 E+06		
N-13	9.965 min	ec, β⁺/ ph			0.160	1000	1.7	1.E+02	[1]	7.00E+07	7.00 E+04 [3]	3	
O-15	122.24 s	ec, β⁺/ ph			0.161	1000	1.7	1.E+02	[1]	7.00E+07	7.00 E+04 [3]	3	

Nucléide	Période	Mode de désintégration / rayonnement	Grandeurs d'appréciation					Limite de libération		Limite d'autorisation	Valeurs directrices		Nucléide de filiation instable
			e_{inh} Sv/Bq	e_{ing} Sv/Bq	h_{10} (mSv/h)/ GBq à 1 m de distance	$h_{0,07}$ (mSv/h)/ GBq à 10 cm de distance	$h_{c,0,07}$ (mSv/h)/ (kBq/cm²)	LL Bq/g		LA Bq	CA Bq/m³	CS Bq/ cm²	
1	2	3	4	5	6	7	8	9		10	11	12	13
Si-31	157.3 min	β⁻/ ph	1.10 E-10	1.60 E-10	<0.001	1000	1.6	1.E+03		5.00E+07	8.00 E+04	3	
Si-32	132 a	β⁻	5.50 E-08	5.60 E-10	<0.001	500	0.6	1.E+02	[2]	9.00E+04	2.00 E+02	10	→ P-32
P-30	2.498 min	ec, β⁺/ ph			0.371	900	1.7					3	
P-32	14.263 d	β⁻	2.90E-09	2.40E-09	<0.001	1000	1.6	1.E+03		2.00E+06	3.00E+03	3	
P-33	25.34 d	β⁻	1.30E-09	2.40E-10	<0.001	700	0.8	1.E+03		4.00E+06	6.00E+03	10	
S-35 (inorg.)	87.51 d	β⁻	1.10E-09	1.90E-10	<0.001	200	0.3	1.E+02		5.00E+06	8.00E+03	30	
S-35 (org.)	87.51 d	β⁻	1.20E-10	7.70E-10	<0.001	200	0.3	1.E+02		4.00E+07	7.00E+04	30	
Cl-36	3.01 E5 a	β⁻, ec, β⁺/ ph	5.10E-09	9.30E-10	<0.001	1000	1.5	1.E+00		1.00E+06	2.00E+03	3	
Cl-38	37.24 min	β⁻/ ph	7.30E-11	1.20E-10	1.551	1000	1.8	1.E+01	[1]	4.00E+07	4.00E+04 [3]	3	
Cl-39	55.6 min	β⁻/ ph	7.60E-11	8.50E-11	0.241	1000	1.7	1.E+01	[1]	7.00E+07	1.00E+05	3	→ Ar-39
Ar-37	35.04 d	ec / ph			<0.001	<1	<0.1			6.00E+13	6.00E+10		
Ar-39	269 a	β⁻			<0.001	2000	1.5			6.00E+09	6.00E+06 [4]		
Ar-41	109.61 min	β⁻/ ph			0.188	1000	1.7			5.00E+07	5.00E+04		
K-38	7.636 min	ec, β⁺/ ph			0.480	1000	1.8					3	

Table 12. Swiss Confederation 2017, Ordonnance sur Radioprotection, authorization discharge limits and conversion factors

In this table, two quantities are of interest :

- e_{inh} is the effective engaged dose coefficient for inhaled particulates (5 µm) in Sv/Bq.
- CA (20 mSv) is the activity concentration resulting in an effective engaged dose of 20 mSv for workers who inhalate this air for 40 hours/week, 50 weeks/year (2,400 m³/y). Release limits are defined in the Swiss Guideline [12]. This case is an extremely worst condition where we consider that the worker will stay at the exit of the chimney and breath the 'whole' ejected activity.

We can derive a limit CA (1 mSv) corresponding to an effective engaged dose of 1 mSv/y for a person from the public, assuming a presence of 4,800 h/year (the annual operating time for the PT center) :

$$CA\ (1\ mSv) = CA\ (20\ mSv) / (20 \times 2.4)$$

The factor 2.4 is to take into account the occupation factor for the public (4,800 h/y instead of 2,000 h/y). This is equivalent to a member of the public being next to the point of discharge the whole year 24h/day.

The CA (1 mSv) limit can be directly compared to our saturated activities as they are expressed in Bq/m³. The 'dilution factor' due to the ejected activities in the air by the chimney is not yet taken into account.

Room	Nuclide	T1/2	Saturated Activity (Bq/m³)	e_{inh} (Sv/Bq)	Yearly Dose (µSv/y)	CA-20 mSv (Bq/m³)	CA-1 mSv (Bq/m³)
CGTR	H-3	12.32 y	$2.59\ 10^{-5}$	$1.8\ 10^{-11}$	$2.69\ 10^{-6}$	$5\ 10^{9}$	$1.04\ 10^{8}$
	Be-7	53.22 d	$1.64\ 10^{-4}$	$4.6\ 10^{-11}$	$4.35\ 10^{-5}$	$5\ 10^{5}$	$1.04\ 10^{4}$
	C-11	20.39 min	$8.69\ 10^{-1}$	$3.2\ 10^{-12}$	$1.60\ 10^{-2}$	$2\ 10^{5}$	$4.17\ 10^{3}$
	N-13	9.965 min	6.09		NA	$7\ 10^{4}$	$1.46\ 10^{3}$
	O-15	122.24 sec	3.98		NA	$7\ 10^{4}$	$1.46\ 10^{3}$
	Ar-41	109.61 min	$5.35\ 10^{-1}$		NA	$5\ 10^{4}$	$1.04\ 10^{3}$
S2C2	H-3	12.32 y	$8.27\ 10^{-3}$	$1.8\ 10^{-11}$	$8.57\ 10^{-4}$	$5\ 10^{9}$	$1.04\ 10^{8}$
	Be-7	53.22 d	$7.04\ 10^{-2}$	$4.6\ 10^{-11}$	$1.87\ 10^{-2}$	$5\ 10^{5}$	$1.04\ 10^{4}$
	C-11	20.39 min	$3.80\ 10^{2}$	$3.2\ 10^{-12}$	7.	$2\ 10^{5}$	$4.17\ 10^{3}$
	N-13	9.965 min	$2.16\ 10^{3}$		NA	$7\ 10^{4}$	$1.46\ 10^{3}$
	O-15	122.24 sec	$1.60\ 10^{3}$		NA	$7\ 10^{4}$	$1.46\ 10^{3}$
	Ar-41	109.61 min	$8.91\ 10^{2}$		NA	$5\ 10^{4}$	$1.04\ 10^{3}$

Table 13. Volumetric activity to reach the CA-20 mSv and CA-1 mSv

In the CGTR, the effective dose resulting from the long-lived isotopes (^{3}H, ^{7}Be, ^{11}C) is negligible (max 10^{-2} µSv/y to be compared with the public limit of 10^{3} µSv/y).

For the other isotopes (^{13}N, ^{15}O, ^{41}Ar) the largest dose is due to immersion inside the released air and we can compare the released activities to the CA-1 mSv/y limit. As the released activities are almost 200 times smaller than the CA-1 mSv/y limit, we can assume that the air released from the CGTR does not represent any hazard for the surrounding population. These calculations don't take into account yet any of the dilution factor.

In the S2C2 room, the effective dose resulting from the long-lived isotopes (^{3}H, ^{7}Be, ^{11}C) is small, with a maximal value of 7 µSv/y. It is important to note, that this room is only accessible to IBA Service Personnel. If the room has to be enterred, time shall be planned for sufficient air renewal.

The saturated activities obtained for the 3 other, short lived isotopes ^{13}N, ^{15}O, ^{41}Ar are close to the CA-1 mSv values (A_{sat}/Ca-1mSv : 1,5, 1.1, 0.85 respectively) leading to a total annual dose of around 3.45 mSv. They would be applicable only if a member of the public would stand the whole year at the point of discharge.

In reality, member of the public would be at receptor points at some distance from the discharge point and with a certain air dilution. IAEA Safety Report [13] lists for various scenarios (dispersion in displacement zone, cavity zone, wake zone) the dispersion of the activated particles.

As an exemple, following simulation are based on a worst case configuration. The air exhaust is located at the rooftop ambulance car port and a guard house is located just next to it.

For dispersion within the dispersion/wake zone the decrease in activity concentration follows approximately a ≈ $1/x^2$ relation, close to the discharge point (if x > 3 x diam. of discharge tube).

A reduction factor based on the average annual wind direction, the wind speed and height of the closest building are given in [13].

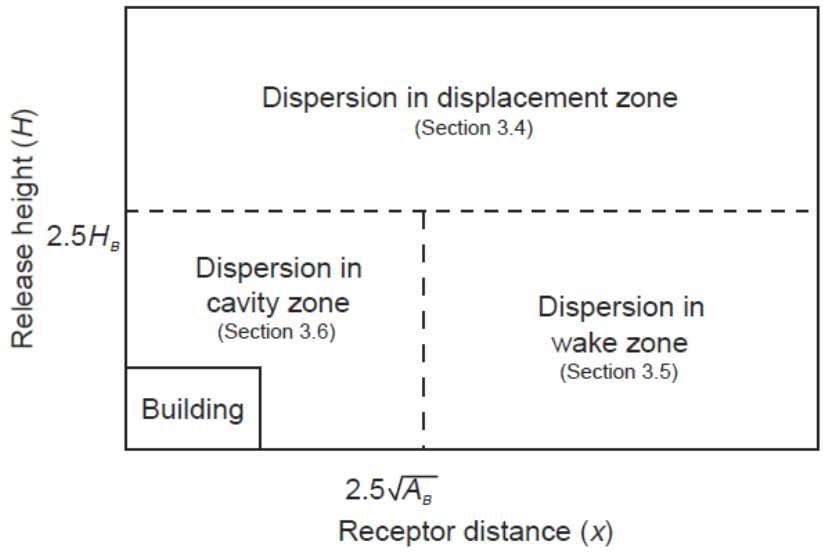


Figure 13. Relationship between release height and receptor distance for determination of the dispersion model to be used [13]

As an exemple, following simulation are based on a worst case configuration. The air exhaust is located at the rooftop ambulance car port and a guard house is located just next to it.

Using average data, activity concentration levels are also dispersed by a factor of > 10^3 using a dispersion due to Inverse Square Law and the average % of time the wind is blowing in the respective direction for points that are > 10 m away from the point of discharge

A more detailed « screening » model can be applied using reference [13]. Here models to calculate the Air Activity Concentration for various geometries are proposed.

"The screening models contained in this report are expected to be particularly useful for assessing the radiological impact of discharges from small scale facilities, for example hospitals or research laboratories." (ref [13]

In table 9 – 11 above, the "Saturated Activities" A_{Sat} (Bq/m^3) for all relevant isotopes are given for the Cyclotron Room and the Gantry Room. Using the volume of both rooms (190 m^3, 650 m^3) and the time for one cycle of air exchange (600 s) , we can calculate the annual average discharge rate Qi, as per Ref [13], *IAEA Safety Report Series 19, Generic Models for Use in Assessing the Impact of Discharges of Radioactive Substances to the Environment, 2001*.

With Qi, the Air Activity Concentration according to reference [13], CA^{IAEA} can be calculated for various location around the point of discharge. These values again can be compared to CA-1 mSv according to Ordonnance sur la radioprotection' (ORaP) published in April 2017 by the Swiss Confederation [12].

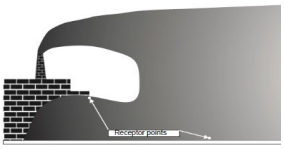
By definition, CA-1mSv is the Activity Concentration leading to 1 mSv effective dose to a person standing the whole year 24 h/day in the respective discharge zones as per figure 13.

Cavity Zone *IAEA Safety Report 2001, 3.6*

Receptor Point Rooftop Ambulance Bay

Source and Receptor same building surface

$$C_A = \frac{B_0 Q_i}{u_a x^2}$$



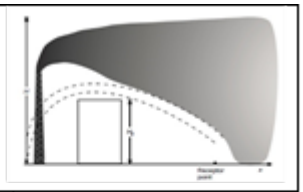
B_o	unitless	30
u_a	m/s	4
x	m	10
Hb	m	30
K	m	1
Pp	%	2

	S2C2		GTRY		CA_{total}
	Q_i Bq/s	C_A Bq/m^3	Q_i Bq/s	C_A Bq/m^3	C_A Bq/m^3
3-H	0.0	0.00	0.00	0.00	**0.00**
7-Be	0.0	0.00	0.00	0.00	**0.00**
11-C	114.0	8.55	0.94	0.07	**8.62**
13-N	648.0	48.60	6.60	0.49	**49.09**
15-O	480.0	36.00	4.31	0.32	**36.32**
41-Ar	267.3	20.05	0.58	0.04	**20.09**
				Sum_{cav1}	**114.13**

Wake Zone *IAEA Safety Report 2001, 3.5*

Receptor Point Guard House, Hospital

$$C_A = \frac{P_p B Q_i}{u_a}$$



B	m^{-2}	0.003
u_a	m/s	5
x	m	N/A
Hb	m	N/A
F	m^{-2}	N/A
Pp	%	11

	S2C2		GTRY		CA_{total}
	Q_i Bq/s	C_A Bq/m^3	Q_i Bq/s	C_A Bq/m^3	C_A Bq/m^3
3-H	0.0	0.00	0.00	0.00	**0.00**
7-Be	0.0	0.00	0.00	0.00	**0.00**
11-C	114.0	0.75	0.94	0.01	**0.76**
13-N	648.0	4.28	6.60	0.04	**4.32**
15-O	480.0	3.17	4.31	0.03	**3.20**
41-Ar	267.3	1.76	0.58	0.00	**1.77**
				11 Sum_{Wake}	**10.04**

Above tables corresponds to the area inside the hospital compound close to the point of discharge and to the nearest area at the Guard House of MHA premises.

We can calculate the effective engaged dose for the worst-case scenario of a person standing 24 h a day for 365 days a year in the Cavity Zone on the rooftop of the ambulance bay next to the hospital building. Weather Data from http://www.weather.gov.sg/climate-climate-of-singapore/ are being used.

It can immediately be seen, that the areas at the guard house of MHA and hospital area are receiving more then a magnitude less of effective engaged dose compared to the worst-case scenario.

Swiss Ordonnance CA-1 mSv/a, IBA, Bq/m^3	**CA-1 mSv, Bq/m^3 (**)**	**CA_{IAEA}Cavity Zone same building surface**	**Factor CA-1-mSv/CA_{IAEA}**	**Resulting Dose to public, micro-Sv/y**
Activity concentration leading to 1 mSv effective engaged dose				
H-3	**104000000**	0.00	552664180120.70	**0.00**
Be-7	**10400**	0.00	6516852.17	**0.00**
C-11	**4170**	8.62	483.72	**2.07**
N-13	**1460**	49.09	29.74	**33.63**
O-15	**1460**	36.32	40.19	**24.88**
Ar-41	**1040**	20.09	51.76	**19.32**
(**) Swiss Ordonnance de Radioprotection 2017			**Total**	**79.89**

We calculate the "worst case scenario" effective dose for a person standing 24 h/day and 365 days a year on the roof of the ambulance bay next to the hospital building to be estimated 80 micro-Sv/y.

It can thus be concluded, that the exposure to the general public due to the operation of the Proton Therapy Center is well below the published natural background, even for the worst-case scenario.

Conclusion

Summary of the impact on the doses due to air activation during the working life of the installation, on :

Impact on the dose due to air activation during the working life (§ 4.2.3)	
Public	<3.45 µSv/y – 80 µSv/y (for worst-case)
Radiation Workers (in PT Center)	< 5 µSv/year
Building	-
Environment	-

The isotopes with the highest activation level (^{13}N, ^{15}O) have a half-life time of only 10 min and 2 min respectively. The typical times between two successive treatments are more than 12 min and the center only operates 14h/day. Hence no build-up of radioactive concentration levels on flower and fauna has to be expected The ^{3}H, with a half-life-time of 12.2 years, is calculated to contribute to the discharged equivalent dose in a magnitude of 10^{-4} µSv/year and can be neglected.

4.1.4 Activation products - Cooling Water

The cooling water used for the cyclotron and the magnets of the beamline is circulated inside a closed-loop circuit. The radionuclides that can be produced by neutron induced interactions in cooling water are the following : ^{3}H, ^{7}Be, ^{10}Be, ^{10}C, ^{11}C, ^{14}C, ^{13}N, ^{14}O, ^{15}O [1]. These nuclides have lifetimes ranging from 19 seconds (^{10}C) up to 1.5 10^{6} years (^{10}Be).

This water is nevertheless exposed to intense neutron fluxes and there is a potential risk of accumulating long-lived isotopes such as ^{3}H. Tritium is not considered as a very dangerous isotope (CL = 100 in Council directive 2013/59/EURATOM [14]). However, it can mix with normal water to make tritiated water (T_2O) that becomes a radiation hazard if ingested.

To assess the potential presence of the long-lived isotopes ^{3}H and ^{7}Be, samples of cooling water coming from the ProteusONE system installed at Nice were measured by an accredited laboratory in France. The results presented in Table 14 show that, after one year of operation, the concentrations of these two isotopes remain below the detection limit of 0.008 Bq/g and 0.006 Bq/g for the ^{3}H and ^{7}Be isotopes, respectively. There is thus no evidence of production of those long-lived isotopes in cooling water.

Isotope	Method	Detection Limit (DL)	Result	Clearance Level (IAEA)	Result / CL
^{3}H	Scintillation counting (ISO 9698)	8 Bq/L	< DL	100 Bq/g	$< 8\ 10^{-5}$
^{7}Be	Gamma spectrometry (ISO 10703)	6 Bq/L	< DL	10 Bq/g	$< 6\ 10^{-4}$

Table 14 : Results of ^{3}H and ^{7}Be activity measurements in cooling water at Nice [16].

Conclusion

Impact of activated water during the working life of the installation, on :

Impact due to water activation during the working life (§ 4.1.4)	
Public	- never in contact
Radiation Workers (in PT Center)	-
Building	-
Environment	-

4.2 After the working life - decommissioning

4.2.1 Equipment's decommissioning

Decommissioning will be performed by trained people. The decommissioning team will consist at least of:

- 1 project manager (procedures, risk analysis, ALRA study, reporting, site-works follow up)
- 1 Radiation Safety Officer (RSO) (radiation protection safety, radiological control of parts, release of parts, waste management)
- 1 RSO assistant
- 2 decommissioning workers (accelerator and system dismantling)
- Specialized sub-contractor (building deconstruction, founder)

(e.g. CEN-SCK/Mol/Belgium, Tecnobel/Belgium…..)

During the decommissioning phase, the activated material will be treated separately from the non-activated material and must be considered as nuclear waste.

For MENH it is planned to ship metallic parts to a Founder for re-use in Nuclear Industry and let the activated concrete decay until it has reached the international limit of Clearance Level (see below). A special "Low Activation Concrete" is used to eliminate the need for long time storage. Please also refer to 4.2.2

These specific activities are compared to Clearance Levels (CL) defined by IAEA [15]. If the ratio A/CL remains smaller than 1, the material can be considered as non-nuclear waste and thus treated as standard waste.

When several unstable isotopes are produced inside the material, it is the sum over all produced isotopes ($\sum A_i / CL_i$) that must remain smaller than 1. That sum is called the Clearance Index (CI).

The CL values change from one isotope to another, depending upon their lifetime and decay modes. The table hereunder gives IAEA clearance levels for decommissioning isotopes of interest:

Nuclide	CL (Bq/g)	Nuclide	CL (Bq/g)	Nuclide	CL (Bq/g)
^{3}H	100	^{54}Mn	0.1	^{134}Cs	0.1
^{7}Be	10	^{55}Fe	1000	^{140}Ba	1
^{22}Na	0.1	^{59}Fe	1	^{152}Eu	0.1
^{45}Ca	100	^{60}Co	0.1	^{154}Eu	0.1
^{46}Sc	0.1	^{64}Cu	100		
^{48}V	1	^{65}Zn	0.1		

Table 15. The IAEA recommendations for CL values have been adopted by the European Council.

Additionally, a good sorting and segregation of activated wastes (material sorting) needs to be performed at the time of decommissioning, (decommissioning team)

For metallic parts, an unconditioned recycling (founders i.e. “Energy Solution” – USA - for the steel) is to be preferred. Not least of all due to economic considerations.

Founders accept different metal types like stainless steel, carbon steel, galvanized steel, with specific activities as large as 40 Bq/g for ^{60}Co (taking the whole transport weight into account). These metallic parts are then melted and reused in the nuclear industry.

The total waste volume to be sent for melting can be estimated at 60 Tons.

A cooling down period must be taken into account for these wastes. Taking the center ramp down and the needed time to submit the final decommissioning plan and receive the official authorization will allow to free release the activated parts that have a short half-life time (IBA experience from Belgium suggest this takes at least 2 years)

Conclusion

After the use of the PT Center, the activated metallic parts will be shipped to a specialized founder or stored for activity decrease

Impact from metallic activation after the working life (§ 4.2.1)	
Public	-
Radiation Workers (in PT Center)	-
Building	To be treated in function of activity
Environment	-

4.2.2 Concrete decommissioning

Concrete activation is only to be considered in the cyclotron room and not in the treatment room due to the lower beam current.

To reduce the amount of active waste produced inside the S2C2 vault, IBA has launched a vigorous research program focusing on the development of low-activation concrete (LAC). The goal is to carefully select the compounds used to prepare the concrete in order to eliminate almost completely the few elements responsible for the production of long-lived isotopes by neutron capture (Eu, Co and Cs), while preserving the same physical properties as standard concrete shielding / structure.

Based on the results obtained from individual concrete compounds, IBA mandated the Scientific and Technical Center for Construction (CSTC in Limelette, Belgium) to formulate various concrete compositions. They propose two possible formulations :

- Concrete type EI made of limestone aggregates (1914 kg) combined with white cement from CBR (260 kg) – concentration in concrete of 0.023 ppm for Eu, and well below 1 ppm for Co and Cs;

- Concrete type S1 made of limestone aggregates (1815 kg) together with Secar 71 aluminous cement from Kerneos (400 kg) - concentration in concrete of 0.015 ppm for Eu and 0.25 and 0.01 ppm for Co and Cs respectively.

Component	^{nat}Eu (ppm)	^{59}Co (ppm)	^{133}Cs (ppm)
Standard Concrete	1.08	21.9	3.21
LAC type EI	0.023	< 1	< 1
LAC type S1	0.0081	0.25	0.01

Table 16. Concrete composition

Note: Above values are only average over many samples and can vary depending on the location and depth where the material is taken from. **In general it shall be tried to achieve a concentration of ≤ 0.1 ppm of Europium.**

Assuming that the Proteus®ONE system will be used for 20 years, we can compute the time evolution of the activity for each produced isotope, taking also into account their decay with time. We then obtain the specific activity A_i for each type of isotope i at the facility end-of-life, as a function of location inside the vault and depth value. Finally, the sum of A_i / CL_i, called the clearance index, is computed and compared to the nuclear waste limit $\Sigma\ A_i / CL_i = 1$.

A batch of simulations using the same inputs for the standard concrete as well for LAC EI concrete was performed and the $\Sigma\ A_i / CL_i = 1$ thicknesses have been estimated. Hereunder are the needed decommissioning layer thicknesses for the cyclotron vault using standard concrete and the low activation alternative (LAC EI) :

Wall	Standard Concrete	LAC EI
West	40 cm	0 cm
East	20 cm	0 cm
South	40 cm	0 cm
North	120 cm	50 cm
Maze	50 cm	0 cm
Floor	50 cm	0 cm
Roof	50 cm	0 cm
Total Volume	**88.1 m³**	**2.0 m³**

Table 17. Needed decommissioning layer in function of the concrete composition.

For most of the walls surrounding the S2C2, one observes a striking decrease of the Clearance Index that is, for LAC EI concrete, well below the limit of 1 even in the first 10 cm of concrete. That is demonstrated in the following figure showing the CI values in West wall using three different types of concrete. Thanks to the use of LAC, the production of low level activated waste vanishes for this wall. The same results are obtained in for the East and South walls, maze, floor and roof.

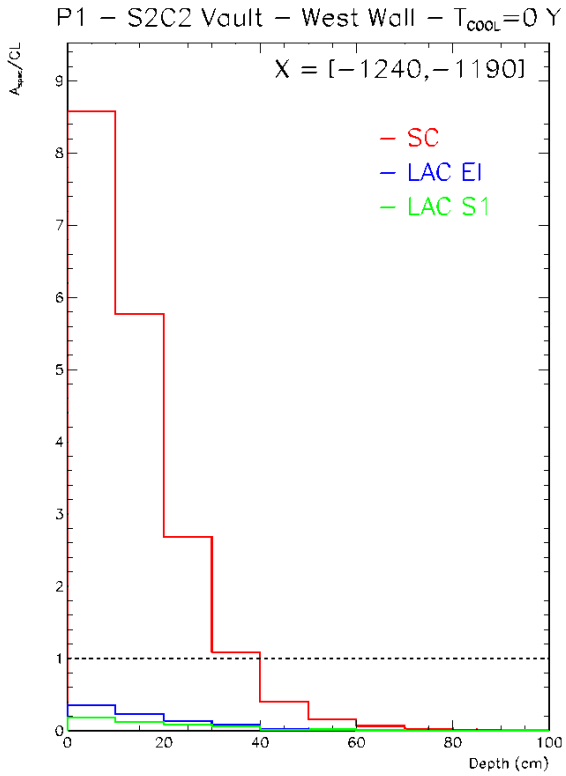


Figure 14. Clearance Index evolution inside West wall using different types of concrete.

The situation is not so trivial as far as the North wall is concerned. This is due to the emission of high-energy neutrons by the degrader and the collimator. These high-energy neutrons are dominantly produced in the same direction as the incident proton beam. Therefore, they will impinge on the North wall separating the S2C2 vault from the CGTR vault.

These high-energy neutrons induce the production of ^{22}Na by spallation reactions on some common concrete constituents (Na, Al, Mg, Si) and the neutron capture by Eu, Co or Cs is limited. Therefore, the use of LAC will not prevent the production of ^{22}Na but, the smaller concentrations of Na, Al and Si in LAC EI compared to standard concrete, the production of ^{22}Na is strongly reduced in the North wall and thanks to the rather low lifetime of this isotope (2.6 years), one can use some decay time after the facility shutdown to eliminate most of the produced ^{22}Na. Indeed, the assumption can be made that a Proteus®ONE building will not be immediately decommissioned after the facility shutdown (delayed dismantling). Morevover, the use of LAC EI induce a reduction of final low level waste up to 2 m³ (instead of 88.1 m³ with standard concrete).

Metallic Rods within the area of LAC can be recycled along with the other metallic components of the system as outlined in 4.2.1. Approximately 3 t of metallic rods will be used around the area of LAC – but not all of them will be exposed to a high Neutron flux and activated above clearance level. Only a final measurement at the end of the life-cycle will allow to quantify the amount and level of activated rods.

In conclusion, using low-activation concrete made of Limestone and white cement, the nuclear waste volume obtained after 20 years of operation is limited to 2 m³ instead of 88 m³.

Taking a cooling down period of the facility shutdown and the decommissioning activities into account, we eliminate the important costs related the building deconstruction. In this hypothesis, the total concrete volume could follow the free release procedure and avoid any radioactive waste management for the concrete of the building. Irrespective of above estimations, concrete can only be dismantled after a physical measurement of the activity.

Conclusion

Impact from concrete activation after the working life (§ 4.2.2)	
Public	-
Radiation Worker (in PT Center)	-
Building	88 m³ of radioactive concrete in case of immediate dismantling using standard concrete, 2 m³ of radioactive concrete in case of immediate dismantling using LAC, 0 m³ after cooling down period using LAC
Environment	88 m³ of radioactive concrete to store in specific radioactive treatment center in case of direct dismantling using standard concrete 2 m³ of radioactive concrete to store in specific radioactive treatment center in case of direct dismantling using LAC 0 m³ after cooling down period using LAC

4.2.3 Ground activation - decommissioning

In the case of a PT facility built below the ground level, earth is generally used as natural shielding material and the thickness of the outside concrete walls is reduced. Consequently, some earth will be submitted to intense neutron fluxes.

Previous studies have identified different radionuclides present in earth around accelerator facility such as: ^{7}Be, ^{45}Ca, ^{54}Mn, ^{22}Na,… [1, Table 6.9].

If the concrete walls surrounding the equipment are not thick enough, the neutron flux behind those walls can potentially be large enough to create some activation in the ground self. This is the case for the ground below the Proteus®ONE facility. It can also be true for side walls in case of underground facility.

The study of ground activation below a facility requires a chemical analysis of the ground to be meaningful. This analysis has been performed in case of the Proteus®ONE facility to be installed on the site of UZ Leuven in Belgium. The results of the ground activation study are described in the report [IBA 5] and are summarized here.

The study uses the same MCNPX model as for the shielding design. The model includes a concrete slab of 50 cm thickness below the S2C2 and the treatment vaults. The ground located below the vaults is segmented into small cells of 10 cm thickness to determine the neutrons flux and the production rate of spallation isotopes.

To evaluate the level of activation (up to 1 m depth), we compute the clearance index of the ground considering the list of long-lived isotopes (larger than 1 d) in the table 18. The clearance index is computed using either the clearance levels (CL) defined by the Council Directive 2013/59/EURATOM [14] or the Belgian royal decree of July 20th, 2001 (Local regulation for UZ Leuven PT Center – ARBIS in the table). The clearance index is computed for a constant usage of 20 years using the same workloads as for the shielding design.

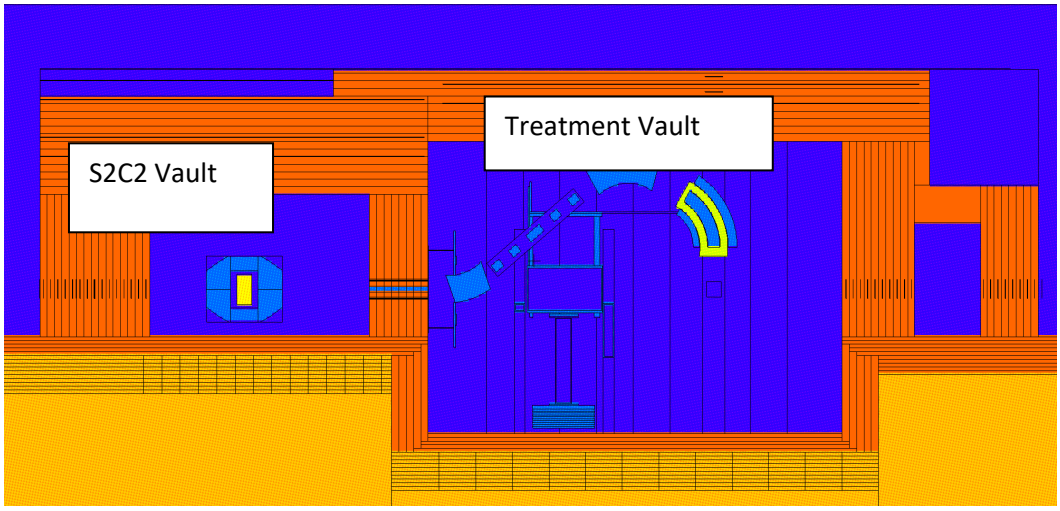


Figure 15. MCNPX modelling of the Proteus®ONE building for ground activation study. The ground below the S2C2 and the treatment vaults is divided into small cells to record the neutron flux.

Isotope	Half-life	CL (EURATOM)	CL (ARBIS)	Isotope	Half-life	CL (EURATOM)	CL (ARBIS)
Be7	53.29 d	10	10	V48	15.97 d	1	0.1
C14	5730 y	1	10	V49	330 d	1*	0.1*
Na22	2.609 y	0.1	0.1	Cr51	27.7 d	100	10
Si32	172 y	1*	1*	Mn52	5.59 d	10	0.1
P32	14.26 d	1000	100	Mn53	$3.7\ 10^5$ y	100	1000
P33	25.34 d	1000	100	Mn54	312.12 d	0.1	0.1
S35	87.51 d	100	100	Fe55	2.73 y	1000	100
Cl36	$3\ 10^5$ y	1	1	Fe59	44.5 d	1	0.1
Ca45	162.6 d	100	100	Co60	5.3 y	0.1	0.1
Ca47	4.54 d	10	1	Cs134	2.06 y	0.1	0.1
Sc46	83.81 d	0.1	0.1	Eu152	13.33 y	0.1	0.1
Sc47	3.35 d	100	10	Eu154	8.8 y	0.1	0.1

Table 18. Long-lived isotopes considered in the ground activation study [IBA 5].

The distribution of clearance index obtained in the first 10 cm of ground below the S2C2 vault is shown in the Figure 19. The peak observed in the middle of the plot corresponds to the neutrons generated by the S2C2 while the peak obtained on the right of the picture is due to neutrons coming from the energy degrader. The maximal value of the clearance index below the S2C2 is 0.14 using the EURATOM CL's and 0.17 with the Local regulation CL's. These values are well below the limit of 1 used to consider the soil as a low-level nuclear waste.

For the treatment room, the modeling was done assuming the nozzle was pointing towards the floor for 100 % of the time, this to enhance the neutron flux traversing the floor slab. Despite that extremely conservative approach, the clearance index in the first 10 cm of ground below the concrete never exceeds a value of 0.2 10^{-3} to be compared with the limit of 1.

These results demonstrate that a thickness of 50 cm for the concrete slab below the Proteus®ONE vault is enough to keep the clearance index below the recommended limit of 1. The thickness of the sidewalls is larger than the 50 cm (2 m), thus, the activation levels will even be below the requested levels.

Those results do not prevent the need of a thicker slab for structural reasons. If so, the impact on the ground will be reduced in consequence.

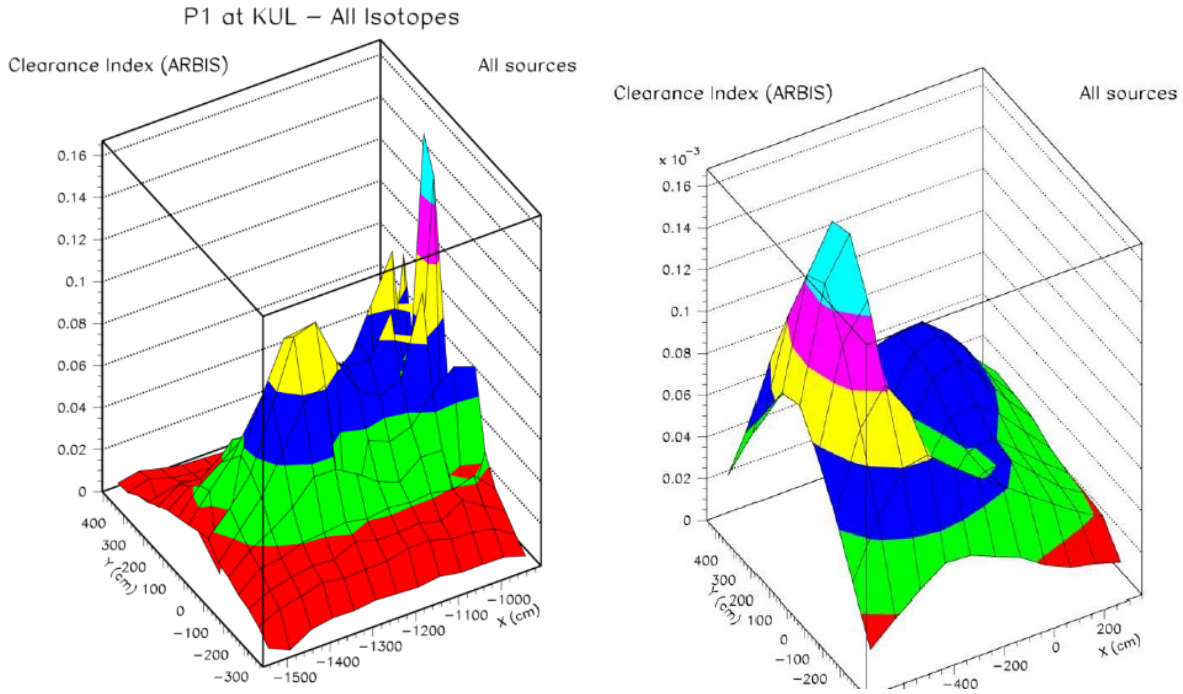


Figure 16. Distribution of clearance index below the concrete slab of the S2C2 vault (left) and Treatment Room (right)

The impact on fauna and flora doesn't need to be considered due to the 'non-impact' on the environment.

Conclusion

Impact from soil activation after the working life (§ 4.2.3)	
Public	-
Radiation Worker (in PT Center)	-
Building	-
Environment	-

5. Conclusion

The total impact on the 'environment' can be summarized as summarized in the following table

Impact		Public	Radiation Worker (PT Center)	Building	Environment
Working life	γ and n doses / dose rates (§ 3)	< 400 µSv/y	< 20 mSv/year averaged over 5 years	-	-
	Activated equipment's (§ 4.1.1)	-	-	-	-
	Activated concrete (§ 4.1.2)	-	-	-	-
	Activated air (§ 4.1.3)	3.45 µSv/y – 80 µSv/y (worst case)	<5 µSv/y	-	-
	Activated water (§ 4.1.4)	-	Application of internal procedure	-	
After the use	Metallic waste (§ 4.2.1)	-	-	To be treated (found) in	To be exported to founder for recycling

				function of activity	
	Concrete waste (§ 4.2.2)	-	-	88m³ if direct dismantling using standard concrete 2m³ if direct dismantling using LAC 0 m³ after cooling down period using LAC	88 m³ of radioactive concrete to store in specific radioactive treatment center in case of direct dismantling using standard concrete 2 m³ of radioactive concrete to store in specific radioactive treatment center in case of direct dismantling using LAC 0 m³ after cooling down period using LAC
	Ground (§ 4.2.3)	-	-	-	-

6. References

IBA Documents – Internal references

[IBA 1] "Concrete Activation in the Proteus®ONE System", F. Stichelbaut, M-ID 39913, September 2013

[IBA 2] "Development of Low-Activation Concrete for the Proteus®ONE System", F. Stichelbaut, M-ID 50897 Rev. 2, February 2016

[IBA 3] "Cooling Water Contamination", A.-K. Nix, M-ID 52970, June 2015

[IBA 4] "Proteus®ONE Shielding Design Using Low-Activation Concrete", F. Stichelbaut, M-ID 52681, February 2016

[IBA 5] "Ground Activation below Proteus®ONE at UZ Leuven", F. Stichelbaut, M-ID 52831, February 2016

[IBA 6] Radiation sources in Proteus One System, F. Stichelbaut, M-ID 36694 Rev. B, September 2014, February 2016

[IBA 7] « Shielding Design for ProteusONE Standard Layout », F. Stichelbaut, M-ID 64063 rev.0, 10 August 2017

External Documents

[1] IAEA Technical Report 283, Radiological Safety Aspects of the Operation of Proton Accelerators, Chapter 6, Environmental Impact of Accelerator, 1988

[2] ICRP (International Commission on Radiological Protection), Publication 74Conversion Coefficients for Use in Radiological Protection against External Radiation, 1996

[3] Neutron H*(10) inside a proton therapy facility: comparison between Monte Carlo simulations and WENDI-2 measurements, V. De Smet et al., Radiation Protection Dosimetry **161** (2014) 417-421

[4] Secondary neutrons inside a proton therapy facility : MCNPX simulations compared to measurements performed with a Bonner Spere Spectrometer and neutron H*(10) monitors, V. De Smet et al., Radiation Measurements **99** (2017) 25-40

[5] Practical Radiotherapy Planning, J. Dobbs et al, 2009

[6] IAEA Safety Standard Series WSG 2-3, Regulatory control of radioactive discharges to the environment, 2000

[7] Evaluation of the Radiological and Economic Consequences of Decommissioning Particle Accelerators, VUB study report EU 19151, 1999

[8] Compendium of Material Composition Data for Radiation Transport Modeling, RJ McConn Jr et al., PNNL-15870 Rev. 1, 2011

[9] Airborne Radioactivity Produced at High-Energy Accelerators", A. Rindi and S. Charalambus, Nuclear Instruments and Methods **47** (1967) 227

[10] Radiation Physics for Health and Environmental Protection, J.D. Cossairt, Fermilab TM-1834, 2007

[11] Radiation Protection for Particle Accelerator Facilities, NCRP 144, 2005

[12] IAEA TEC DOC 1638, setting Authorized Limits for Radioactive Discharges : Practical Issues to Consider, 2010

[13] IAEA Safety Report Series 19, Generic Models for Use in Assessing the Impact of Discharges of Radioactive Substances to the Environment, 2001

[14] Council Directive 2013/59/Euratom laying down basic safety standards for protection against the dangers arising from exposure to ionizing radiation, Volume 57, January 2014

[15] Applications of Concepts of Exclusion, Exemption and Clearance", IAEA Safety Guide RS-G-1.7, Vienna, 2004

[16] Rapport d'essais n° 18-00709-04917 and 04918 from Eichrom Laboratoires, 09.02.2018.

Thank You

/iba

/company/iba

/ibaworldwide

/ibagroup

www.iba-worldwide.com