CHAPTER X

SAFETY AND RADIATION PROTECTION

10.1 Introduction

This chapter presents the preliminary safety and radiation protection analysis of the SPES project, in particular the issues related to proton driver and the use of a Direct Target of UCx with a fission rate of 1E+13 fission-per-sec. The target is the most sensitive part of the facility as regard to the radiological aspects. Special care is needed to handle and store the activated material.

10.2 Facility description

The SPES project will produce secondary beams of unstable ions to perform basic research studies in the nuclear physics field with the aim to produce and characterize isotopes far from stability and to study the properties of the nuclear reactions at variation of isospin.

The production method is based on the ISOL technique with direct target: a primary proton beam impinges directly on a UCx target inducing the U fission with a maximum rate of 1E+13 f/s, the fission products are extracted, ionized, selected and reaccelerated to produce the secondary radioactive beam.

It is part of the facility a neutron beam produced by a high current proton RFQ. A proton beam of 30 mA, 5 MeV, impinging on Be or Li target will produce neutrons with a rate in the order of 10^{14} n/s.

10.3 Site and buildings

The facility will be located at LNL and represent an extension of the present accelerator complex. The buildings of the new facility will be constructed in the expansion area of the Laboratory in contiguity with the actual “third experimental hall”.

The species produced by SPES will be reaccelerated by the present ALPI complex.

LNL is located at 2 Km from Legnaro (east) and 1 Km from Ponte San Nicolò (west).

Padova is located 10 Km in the west direction. The highway Bologna-Padova is just on the border of the LNL site area which has a surface of 10000 m^2.

The whole SPES facility will consist of several buildings with design criteria adequate to perform an effective safety environment.

The main buildings are:

10.3.1 Proton driver building

It will host the accelerator for the production of the primary proton beam. The safety in this area is related to the normal operation of a cyclotron accelerator with a proton current of 700 microA and a final energy of 70 MeV. As mentioned above also an RFQ with a proton current of 30-50 mA and 5 MeV energy will be collocated in an adjacent area.

Major radiation hazards are related with the prompt radiation fields (mainly neutrons) and residual radioactivity production due to the interaction of primary beam and/or of the secondaries (neutrons) with the beam line, cyclotron components, target stations, surroundings.
10.3.2 Production Target building

Target
The UCx target is the critical item of the facility from the point of view of the safety due to the large level of radioactivity induced by the fission reactions.
The target follows the design already developed in several laboratories where radioactive beams are operated recently or since long time as EXCYT (LNS) or ISOLDE (CERN) and HRIBF (ORNL).
The total amount of UCx is in the order of 30g and the power released in the target is 8 kW to produce 7E+12 f/s with 200 microA of protons at 40 MeV (or 1.4E+13f/s with 50MeV protons). We consider a fission rate of 1E13 f/s in the following.

Mass selector
The mass selector is the first beam selection element just after the target and +1 source. It is mounted on the same platform of the target itself and will perform the mass selection with a resolution of 1/250.
The expected radiation level is evaluated to be of the same order as the production target in the hypotheses that all produced elements are extracted.

10.3.3 Low Energy building

The Low Energy building is the last part of the new building adjacent to the already existing building and it will host the High Resolution Isotope Separator (HRIS) and Low Energy experimental hall.
At exit of HRIS a Charge Breeder is installed to increase the charge state and to allow a proper acceleration. This stage is performed by an EBIS or an ECR.
The Charge Breeder has efficiency in the order of 3-10% and a large part of the produced beam is expected to be lost inside this element. From the point of view of the radiation protection we consider that the radioactive beam is completely lost inside. A local shielding will be used for this element.
At the entrance of the HRIS the radiation level will be one or two orders of magnitude more than in the Charge Breeder.
The HRIS is expected to have a resolution of $\Delta M/M = 1/15000$, at its output only a beam of selected isotope should be present and the radiation level will be further reduced for the transmitted beam.
At this level the radioactivity is induced by the radioactive beam itself that is expected to be of the order of 1E+11 pps in the optimum case and 1E3 pps for the very rear beams.

10.3.4 Post accelerator building

The post accelerator building hosts the Charge Breeder, a 1/1000 spectrometer and the beam transfer line to drive the beam in the already existing accelerators: the PIAVE RFQ pre-accelerator and the ALPI Linac Accelerator.
Both machines are installed in an existing building. At this level the radioactivity is induced by the beam that is expected to have an intensity of 1E10 pps in the optimum case.
If necessary, additional shielding will be installed to maintain the radiation level at the actual limit.

10.4 Radiation levels

In normal operation the higher radiation level is around the UCx target where the fission process is induced by the proton beam (40 MeV, 200 $\mu$A).
A total activity of 2E+13 Bq is present inside the target with a dose of 4-10 Sv/h at the surface of the target container. The operation temperature of the target is 2000 °C and all elements with melting point lower than
this temperature are extracted ionized in the $1^+$ source and transmitted to the Mass Selector. We consider for this element 1% selection. The HRIS is the following element which delivers a mass and isotope-selected beam to the Charge Breeder. Inside the Charge Breeder the species are further ionized (+n) to allow a proper reacceleration by the PIAVE RFQ and the ALPI Linac. An efficiency of 5% for the Charge Breeder and 40% for the re-acceleration is considered.

After the Charge Breeder a further selection is needed to recover the beam purity before to send the beam by the Low Energy lines to the re-acceleration stage.

A schematic radiation activity for the different stage is reported in table 10.1.

### Table 10.1 Radioactivity at different stage of the facility

<table>
<thead>
<tr>
<th></th>
<th>Activity (Bq)</th>
<th>Gamma dose (Sv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Saturated</td>
<td>after 1 y</td>
</tr>
<tr>
<td>target</td>
<td>2 E+13</td>
<td>1 E+10</td>
</tr>
<tr>
<td>Mass Separator</td>
<td>5 E+10</td>
<td>1 E+9</td>
</tr>
<tr>
<td>Before Isotope Separator</td>
<td>1 E+11</td>
<td>5 E+7</td>
</tr>
<tr>
<td>After Isotope Separator</td>
<td>1 E+10</td>
<td>5 E+6</td>
</tr>
<tr>
<td>After Charge Breeder</td>
<td>5 E+5</td>
<td>1 E-7</td>
</tr>
</tbody>
</table>

#### 10.4.1 Design criteria

The sensitive parameters for the safety and radiation protection are:

- Beam intensity
- Beam energy
- Beam losses
- High voltages
- RFQ Radio frequency

The most relevant issues are:

- Fission fragment production target (7 E+12 f/s)
- High current low energy proton beam (30 mA, 5 MeV)
- Medium current high energy proton beam (350 microA, 70 MeV)

The following parameters are used to the preliminary shielding design of the facility:
Table 10.2 *Design parameter for shielding design*

<table>
<thead>
<tr>
<th></th>
<th>Energy (MeV)</th>
<th>Intensity (microA)</th>
<th>Losses (microA)</th>
<th>HV</th>
<th>particle</th>
</tr>
</thead>
<tbody>
<tr>
<td>Proton source</td>
<td>0.02</td>
<td>30000</td>
<td></td>
<td>+20 KV</td>
<td>proton</td>
</tr>
<tr>
<td>LEBT</td>
<td>0.02</td>
<td>30000</td>
<td></td>
<td></td>
<td>proton</td>
</tr>
<tr>
<td>Trasco-RFQ</td>
<td>0.02 → 5</td>
<td>30000</td>
<td></td>
<td></td>
<td>proton</td>
</tr>
<tr>
<td>BNCT Neutron facility</td>
<td>5</td>
<td>30000</td>
<td>30000</td>
<td>Proton</td>
<td>Neutron</td>
</tr>
<tr>
<td>Cyclotron</td>
<td>0 → 70</td>
<td>350 (design)</td>
<td>7.5</td>
<td>Proton</td>
<td></td>
</tr>
<tr>
<td>Beam dump</td>
<td>70</td>
<td>350 (design)</td>
<td>350 (design)</td>
<td>Proton</td>
<td></td>
</tr>
<tr>
<td>Radioactive target</td>
<td>40 0</td>
<td>200 10^{13} pps</td>
<td>200 10^{13} pps</td>
<td>250+40 KV</td>
<td>Proton Fission Frag</td>
</tr>
<tr>
<td>Mass separator</td>
<td>0.040</td>
<td>10^{13} pps</td>
<td>99%</td>
<td>250KV</td>
<td>Fission Frag</td>
</tr>
<tr>
<td>HRIS</td>
<td>0.250</td>
<td>10^{11} pps</td>
<td>90%</td>
<td>250 K</td>
<td>Fission Frag</td>
</tr>
<tr>
<td>Charge Breeder</td>
<td>0 → 0.250</td>
<td>10^{10}</td>
<td>90%</td>
<td>Fission Frag</td>
<td></td>
</tr>
<tr>
<td>PIAVE</td>
<td>0.250 → 5</td>
<td>10^{7} pps</td>
<td></td>
<td>Fission Frag</td>
<td></td>
</tr>
</tbody>
</table>

A radiation protection programme identifies elements necessary to operate and maintain the SPES accelerator and the source region safely from a radiological point of view. Some radiation protection criteria to be met during the design stage are established.

Location and magnitude of radiation hazards related with have been identified and it is intended to provide the basis to control both the radiation hazards and the exposure of personnel.

The main objectives are:

- Prevent occupational doses over legal limits
- Maintain personnel doses As Low As Reasonably Achievable (ALARA)
- Prevent unplanned exposures
- Minimize spread of contamination

The main topics covered are the following:

- Designation of radiation areas and access control
- Shielding calculations of the accelerator vault
- Induced radioactivity at the accelerator structures and air activity
- Preliminary radiation protection analyses of the SPES target

### 10.4.2 Designation of radiation areas and access control

One of the most important workplace safety issues is to ensure that both individual worker occupational exposure and environmental exposure are maintained below legal limits. Table 10.3 indicate the values of
maximum effective dose admitted in a year exposure of a worker under normal conditions and the
radiation area involved. For 2000 hours of operation of the accelerator, doses constrains, also named
project guidelines, are assumed. The upper limit of effective dose for members of public outside of the
Laboratory is placed a $10 \mu$Sv/y. In the case of a full beam loss the maximum dose received by an
individual in the Laboratory is chosen at $10 \mu$Sv.

Table 10.3 - Limits and project guidelines.

<table>
<thead>
<tr>
<th>Type of area</th>
<th>Effective dose (mSv/y)</th>
<th>Project guidelines (mSv/y)</th>
<th>Max. $H^*(10)$ rate ($\mu$Sv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Not classified</td>
<td>1</td>
<td>0.5</td>
<td>0.25</td>
</tr>
<tr>
<td>Supervised</td>
<td>6</td>
<td>3</td>
<td>1.5</td>
</tr>
<tr>
<td>Controlled</td>
<td>20</td>
<td>10</td>
<td>5</td>
</tr>
</tbody>
</table>

10.4.3 Worker classification

Radiation worker (RW)
These are workers required to be in the radiological areas for operation, maintenance, and field
support. Workers performing special experiments, diagnoses, etc., that regularly access radiological
areas may also be designated as RW.

RW will be classified into two different categories: Radiation workers category A (RWA) and Radiation
workers category B (RWB). RWA are workers that regularly access controlled radiological areas to perform
their job and are likely to be exposed to an annual whole body effective dose higher than 6 mSv. RWB access
occasionally to the controlled radiological areas and are likely to be exposed to an annual whole body
effective dose higher than 1 mSv, but lower than 6 mSv.

Non-Radiation Workers (NRW)
These are administration, scientific, operations, and maintenance personnel whose responsibilities do
not require, normally, access to areas with radiological hazards. The annual effective whole body dose limit
for non-radiation workers is 1 mSv.

10.4.4 Access control
Access control is achieved by defining access zones as a function of expected hazards, and by applying
procedures for movement between these zones. During beam acceleration very high hazards are
anticipated in the accelerator vault and the target zone, and it is needed to assure the restriction of access
to these areas. An Access Restriction System, consisting of physical as well as procedural barriers (locks,
key card access, self-locking doors, monitoring tools), is therefore implemented to control entry.
10.5 Shielding calculations

10.5.1 Cyclotron bunker

Preliminary shielding calculations of the room where the accelerator will be installed and of the target cave were made with an analytical method [1-4], as well as using the BULK-1 tool [5], taken into account the following assumptions:

- maximum energy of the protons 70 MeV
- maximum current of the beam 350 µA
- maximum beam losses due to stripping after interaction with residual gas max. 3%
- constant uniform point losses of 7.5 µA could occur in selected points, bending magnets (more conservative than above)
- thick copper was chosen as material target
- operation time 168 h/week, 2000 hours/y of operation
- two beams of 350 µA at the same time can be accelerated inside the vault
- distance from the magnet to the point of interest equal a 4 m
- in the case of a full beam loss the maximum dose received by an individual in the Laboratory is fixed at 10 µSv.

The lost protons were assumed to strike a thick copper target like a magnet coil o yoke. Copper was chosen because of similar density with iron and stainless steel.

Considering the source at a selected point as a “point source” then the total ambient dose equivalent \( H_{tot,\theta}(E_p, \theta, d) \) due to the neutrons and induced gamma rays at a certain depth of shielding is given by:

\[
H_{tot,\theta}(E_p, \theta, d) = H_0(E_p, \theta)/r^2 \exp[-d/\lambda_\theta]
\]

where \( E_p \) is the energy of incident protons, \( \theta \) is the neutron emission angle with respect to the proton beam, \( d \) is the shielding thickness, \( r \) is the distance between the source point and the point of interest (4 m), \( \lambda_\theta \) is the attenuation length and \( H_0 \) is the source term (Sv m\(^2\) per proton). Given the distance between loss point (bending magnet, worst case), the dose at a given location outside the shield only depends on the facing loss point. For lateral shielding the conservative source term in the forward direction \( H_0 \) was used taking into account that for high-Z target materials (like copper or Ta) the source term remains more or less isotropic.

Figure 10.1 shows the cyclotron vault and the calculated concrete shielding thicknesses for having not classified areas out of the shielding. In the direction of the beam 25 cm of supplementary iron shielding is needed and this is provided by the self shielding of the bending magnet.
Figure 10.2 shows the calculated concrete roof and floor shielding thicknesses leading in an annual ambient dose equivalent much less than 0.5 mSv/y.

**Fig. 10.1 Cyclotron vault shielding**

**Fig. 10.2 A section of the vault reporting the concrete thicknesses of the roof and floor**
Table 10.4 shows the shielding parameters used in the calculations and the resulting thickness required to keep dose rate in the representative points lower than 0.25 µSv/h (not classified area) for losses of 7.5 µA on the bending magnets; concrete of 2.1 g/cm³ density was considered. The direction of the beam is considered in +Y axis, 90° at +X, roof at +Z.

Table 10.4

<table>
<thead>
<tr>
<th>Proton energy (MeV)</th>
<th>Source term (Sv/h) at 1 m</th>
<th>Attenuation length (cm)</th>
<th>Direction considered</th>
<th>Distance from target to estimator (m)</th>
<th>Concrete thickness (cm)</th>
<th>Iron thick. (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>70</td>
<td>227.4</td>
<td>19.6</td>
<td>+Y / -Y</td>
<td>4.4 – 14</td>
<td>350 - 350</td>
<td>25 - 25</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+X / -X</td>
<td>7.2 – 11.4</td>
<td>300 - 300</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+Z / -Z</td>
<td>7 - 5</td>
<td>300 - 350</td>
<td></td>
</tr>
</tbody>
</table>

In the case of a full beam loss the total beam power will be deposited at the point of interest. Dose rates outside the proposed shielding for a full beam loss is reported in table 5.

Table 10.5

<table>
<thead>
<tr>
<th>E (MeV)</th>
<th>Current (µA)</th>
<th>Beam (kW)</th>
<th>Power</th>
<th>Direction considered</th>
<th>Dose rate (µSv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>70</td>
<td>350</td>
<td>24.5</td>
<td></td>
<td>+Y / -Y</td>
<td>10 / 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>+X / -X</td>
<td>46 / 18</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>+Z / -Z</td>
<td>49 / 8</td>
</tr>
</tbody>
</table>

The maximum instantaneous dose rate with the proposed shielding is 49 µSv/h. For 10 µSv accumulated dose by one full beam loss implies that exists enough time to switch off the beam, condition easily met with an interlock system.

10.5.2 Target bunker

The target building will host two targets which can be operated simultaneously or alternatively. To better confine the radioactive areas each target will be hosted in a separate cave. The lay out is presented in fig.10.3
Shielding thicknesses of the target cave were made considering two different scenarios: the one to produce radioactive beams as reported before (40 MeV protons, 200 µA impinging on an UC_x target) but also a proton beam of 70 MeV and 350 µA stopped on a similar target. In figures 10.4 and 10.5 the resulting thicknesses required to keep dose rate lower than 0.25 µSv/h (not classified area) are reported.

10.6 Induced radioactivity in the accelerator structures

The estimation of the induced radioactivity in the accelerator components is important for maintenance interventions and final disposal of radioactive waste. The produced radioactive nuclides depend on the exact chemical composition of the irradiated materials which at the present state are not known. The saturation radioactivity that may be induced by the neutrons generated from the beam interactions, assume that the “target” material surrounds the source point, is given by:

\[ A = N_0 \sigma \varphi_0 \rho \lambda / W \]

where

- \( N_0 \) is Avogadro’s number,
- \( W \) is the atomic weight of the target material in g/mole,
- \( \sigma \) is the production cross section in cm²,
- \( \rho \) is the material density in g/cm³,
- \( \lambda \) is the attenuation length, and
- \( \varphi_0 \) is the neutron source strength in n/sec.

For proton energy of 70 MeV and a point loss of 7.5 µA on a copper target, the neutron source strength is
approximately $2.6\times10^{12}$ n/sec. For concrete as target material (local shielding cave – target containers) with average atomic weight $W = 30$, a $\rho \lambda$ value in the region of interest of $41 \text{ g/cm}^2$, the induced saturation radioactivity is approximately $2.1 \text{ GBq}$ per mb cross section. The total activity estimated in such a way is independent of the detailed geometric distribution of the material surrounding the source point as long as there are no strongly neutron absorbing materials in the path of the neutrons.

For cross sections of the order of a few times $10 \text{ mb}$ the expected total saturation activity will be approximately $20 \text{ GBq}$. With a workload of 2000 hours per year of SPES operation, for long lived species the saturation activity will be much closer to $4.5 \text{ GBq}$. Most of this radioactivity will be fixed in place in the material surrounding the source point and there will be little chance of spread of loose of radioactive material.

The above estimate assumes that the expected beam loss in the accelerator system is concentrated in a given point. In the case of a realistic scenario the losses will occur along the entire length of the accelerator (number of discrete points), the beam losses will be only a small fraction of the considered and induced radioactivity at each point will be only a small fraction of the $4.5 \text{ GBq}$ total saturation activity.

### 10.7 Induced radioactivity in the vault structures

In addition to the neutron activation of the “target” containers, there is also the problem of general activation in the vault containing the cyclotron. Considering a neutron source strength of approximately $2.6\times10^{12}$ n/sec and the empirical relationship for the effective “thermal” neutron flux $\phi_{\text{eff,th}} = Q/A$, where $A$ is the surface area of the equivalent volume sphere and $Q$ is the source strength of neutrons in the concrete vault, the effective flux in the cyclotron room will be $4.3\times10^5 \text{ n/cm}^2\text{sec}$. In the case of concrete shielding with low sodium content of 0.25 per weight, the danger parameter[6] is found to be $7.64\times10^{-11} \text{ Svh}^{-1}$ per $\text{n cm}^{-2}\text{sec}^{-1}$. So the total gamma dose rate at saturation everywhere in the vault will be $0.4 \text{ mSv/h}$; the most of the total gamma dose comes from $^{24}\text{Na}$ due to reaction $^{23}\text{Na}(n,\gamma)^{24}\text{Na}$.
10.8 Air activity of the cyclotron’s vault

The radioactive species which can be produced in certain amounts by neutron activation of the air at an intermediate energy proton accelerator are summarized below in table 6.7.

In practice only the positron emitters $^{11}$C, $^{13}$N, $^{15}$O and also $^{41}$Ar are important as the other species usually does not have time to build up to any significant level for reasonable ventilation rates.

As reported below the saturation activity per unit of neutron flux (energy $\geq 20$ MeV) for all these species is of the order of a few tenths of a Becquerel per cubic meter.

**Table 10.7**

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Neutron energy</th>
<th>Saturation Activity (Bq m$^{-3}$ per cm$^{-2}$ s$^{-1}$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{3}$H</td>
<td></td>
<td>0.80</td>
</tr>
<tr>
<td>$^{3}$He</td>
<td></td>
<td>0.80</td>
</tr>
<tr>
<td>$^{7}$Be</td>
<td>Fast flux (E$_n$ $\geq$ ~20 MeV)</td>
<td>0.27</td>
</tr>
<tr>
<td>$^{11}$C</td>
<td></td>
<td>0.53</td>
</tr>
<tr>
<td>$^{13}$N</td>
<td></td>
<td>0.27</td>
</tr>
<tr>
<td>$^{15}$O</td>
<td></td>
<td>0.21</td>
</tr>
<tr>
<td>$^{41}$Ar</td>
<td>Thermal</td>
<td>0.15</td>
</tr>
</tbody>
</table>

The saturated source strength of $^{41}$Ar produced in the air, for an effective “thermal” neutron flux 4.3E+5 n/cm$^2$sec would be 6.4E+4 Bq/m$^3$ giving a residual radiation field at the centre of the vault of 1.5 $\mu$Sv/h.

As known, for an inert gas (but also true for positron emitters), the contribution to effective dose from external radiation is much greater than that due to inhalation. But even though the radiological hazard due to the air activity is usually small compared to the other sources of the cyclotron the higher sensitivity of the workers associated with the fear of being “contaminated” needs to minimize the leakage of active air into occupied areas.

10.9 Production Target

The production target is designed to sustain a power of 8 KW of proton beam and to produce a fission rate of 7E12 f/s. The active material is 30 gr of UCx in the shape of 7 disks of 4 cm diameter and 1mm thickness each.

The radiation levels in the target area are reported in table 1.

Targets similar to the SPES one are in operation from long time at ISOLDE-CERN and HRIBF-ORNL. The main differences are the total beam power supplied to the target and some details on the disks dimensions and displacement. The SPES target has a diameter 2.7 times the usual one and is designed to cool down by irradiation. The fission rate is 30 times higher than HRIBF but with the same power density released in the target.

The design follows the ISOLDE project and all the mechanical parts, as well as the feed-through for cooling and power supply, are studied and tested at CERN to sustain the high dose rate expected. A pictorial drawing of the Target Ion Source system is reported in fig.10.6.
From the safety point of view the production target is certainly the most dangerous item of the SPES project and special care must be taken to all its details and to all the operations in which it is involved.

In addition to the fission rate during irradiation the target has two more critical points: the operation temperature is around 2000 °C and after irradiation it must be removed and substituted with new one.

The inner container of the target disks is a cylinder with a double entrance window and a beam dump for low energy protons. All these elements are made of graphite. The window is made by two layers of graphite with a thickness of 0.5 mm each. It is the physical separation between the reaction chamber and the proton driver beam pipe. The choice to use a two layers window is suggested by thermal calculation and to improve the safety of the system. The melting point of the graphite is 3500 °C well above the temperature of operation of the target. Special care will be taken to avoid an over heating induced by the proton beam. Even if the target structure can sustain the expected power it is of paramount importance that the proton beam is not concentrated on a small target surface to avoid local over heating and pin holes. The beam shape is monitored by the beam diagnostic system and automatic interlocks on the proton beam will be activated if beam parameters as shape and intensity do not fit the safety limits.

The removal of used plugs and the insertion of new one will be performed without any manual operation by a servomechanism.
10.10 Radioactivity inventory and handling

10.10.1 Target

A correct estimate of radiation and radioactivity levels is essential at the design stage of the facility to incorporate the radiation safety into the infrastructure layout, for predicting the risks of radiation damage and/or material activation as well as for an estimate of the environmental impact of the facility (air, water activation). Remote control handling methods have to be developed especially for the target area. Since the production target consists of fissionable material (mainly ~30 g of depleted $^{238}$U), some trans-uranic elements are produced and have to be controlled at any time throughout the facility. High activation levels have detrimental effects such as large inventories of radiotoxic nuclides, large decay power (gamma heating), as well as radiation damage of components.

The inventory of radionuclides produced in-target was done by ENEA as well as the estimation of the radioactivity produced in the target area.

The actinides and fission fragments production are calculated, as well as the energy spectra of photons emitted by the fission products. In addition, the code gives information on the energy spectra of neutrons emitted in $(\alpha,n)$ reactions and from spontaneous fission. After 30 days of irradiation, inside the ~30g UCx target are collected about 60 mg of fission fragments and a relevant quantity of actinides elements like: $^{236}$U (0.34 mg), Np $^{237}$ (0.52 mg), and Pu$^{239}$ (5 $\mu$g). The corresponding activity is about $10^{13}$ Bq. After 3 months from irradiation the target still shows an activity of $4 \times 10^{10}$ Bq and after 1 year $9 \times 10^9$ Bq. Detailed calculations can be found in Ref. [7].

Because of the high radiation level, the production targets and all the materials surrounding the target itself have to be removed only by a remote handling system. A trail and pneumatic system allows the insertion and disconnection of the target box which has a weight of ~80 Kg (input and output valves included). The activated target is safely sealed in a shielded container made of lead, iron and copper. This container has dimensions less than 1 cubic meter and a total weight of 400 Kg. A remote operated crane allows the deposition of the container inside a special concrete sarcophagus for long term storage.

A site will be dedicated to the storage of the sarcophagi for long term cooling down before the decommissioning.
10.10.2 Preliminary radiation protection analyses of the SPES target

The objective here is a preliminary assessment of some radiation protection quantities. The proton beam interacts with the target disks (28.6 g of UCx) and the structures around the disks (the target configuration is shown in fig. 10.6 and described in detail in [8, 9]). The inelastic collisions produce neutrons and gammas (which can escape the target system) and radioactive residual nuclei (which can emit gamma, beta and alpha particles). The produced neutrons, in turn, can activate other structural components in the surroundings of the target. Here some quantities are estimated [10] by means of computer codes:

- the dose due to the produced neutrons and prompt gammas (see section 2);
- the gamma dose coming from the activated UCx disks (see section 3);
- the gamma dose coming from the activated anticorodal containment (see section 4).

In the target configuration analyzed, the disks are surrounded by a graphite box (2 mm thick) and a layer of tantalum (1.5 mm thick). Then the target is placed inside an anticorodal sphere of radius r=15 cm and thickness 0.5 cm (see fig. 10.6). Even if the MCNP model of the target system is simplified (no model of foreseen blocks of tantalum, copper and other structural materials has been made), this description is considered good enough for assessing the mentioned radiation protection quantities.

![Diagram of target configuration](image)

*Fig. 10.8 Reference target configuration for the calculations.*

Two computer codes have been used for the calculations:

1. MCNPX version 2.5.0 [11], a Monte Carlo particle transport code used with the Bertini-Dresner model [12] for the description of the proton nuclear interactions and with the ORNL model [13] for the description of the fission fragment distribution.
2. SP-FISPACT [14], that is an extension of the activation code FISPACT [15] for neutron energies higher than 20 MeV and for protons. This allows to calculate the activity (Bq), the evolution and the accumulation with time of the residual nuclei provided by MCNPX (about 15% of the atoms provided by MCNPX are expunged because not included in the FISPACT decay library).

The calculations here refer mainly to the ambient dose equivalent $H^*(10)$ (the dose equivalent at a depth of 10 mm in the ICRU sphere), which represents a “reasonable estimate” of the effective dose [16] (at least for
the energy ranges considered here). The flux-to-dose conversion factors are taken from [17] for neutrons and from [18] for gammas.

10.10.3 The neutron dose

The 40 MeV proton beam impinges on the disks and on the structures surrounding them, producing neutrons which can escape the target system. According to MCNPX, the neutrons created in the target for 0.2 mA are $4.9 \times 10^{13}$ neutrons/s. The neutron dose at a distance of 1 m (averaged over the sphere surface) from the centre of the target turns out to be 535 Sv/h. Table 9 summarizes these results for $r=15.5$ cm and $r=100$ cm.

<table>
<thead>
<tr>
<th>Radius of the sphere (cm)</th>
<th>Average neutron flux (cm$^{-2}$ s$^{-1}$)</th>
<th>Average neutron ambient dose equivalent H*(10) (Sv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>15.5</td>
<td>$1.9 \times 10^{10}$</td>
<td>$2.5 \times 10^4$</td>
</tr>
<tr>
<td>100</td>
<td>$3.9 \times 10^{8}$</td>
<td>$5.4 \times 10^2$</td>
</tr>
</tbody>
</table>

Even if there is not spherical symmetry for the neutron dose (because of the angular distribution and the different energy spectra of the neutron flux), the difference taking into account the angular direction is limited: in the forward direction (with respect to the proton beam) the dose reaches values up to 650 Sv/h, laterally up to 490 Sv/h and in the backward direction up to 500 Sv/h.

The dose due to the prompt gammas has been evaluated to be negligible (<2%) with respect to the neutron dose.

The obtained results can be utilized to assess preliminary shielding thicknesses to lower the doses to acceptable limits.

10.10.4 The gamma dose from the UC$_x$ disks

The gamma dose from the activated UC$_x$ disks is evaluated here. The proton inelastic collisions in the UC$_x$ disks are $8.9 \times 10^{12}$ collisions s$^{-1}$. These include fission reactions as well as other nuclear reactions (such as (p,n), (p,pn), etc.). Table 10 shows the reaction rates and the corresponding number of produced residual nuclei.

<table>
<thead>
<tr>
<th>Number of reactions (s$^{-1}$)</th>
<th>Number of produced atoms (s$^{-1}$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fissions</td>
<td>$7.0 \times 10^{12}$</td>
</tr>
<tr>
<td>Other inelastic collisions</td>
<td>$1.9 \times 10^{12}$</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>$8.9 \times 10^{12}$</td>
</tr>
</tbody>
</table>
About 88% of the atoms produced in the disks are fission fragments. If we include also the production of $^1$H ($5.7 \times 10^{12}$ s$^{-1}$), $^2$H ($6.5 \times 10^{11}$ s$^{-1}$), $^3$H ($4.5 \times 10^8$ s$^{-1}$) and $^4$He ($3 \times 10^{12}$ s$^{-1}$), then the total number of produced atoms becomes $2.5 \times 10^{13}$ s$^{-1}$.

The irradiation time is assumed to be seven continuous days (at 40 MeV, 0.2 mA). The activity of the UC$_x$ disks, during irradiation and during the cooling period, is reported in table 11 and shown in fig. 2 and fig. 3. The activity at shutdown is about $2 \times 10^{13}$ Bq and becomes, after one year of cooling, about $10^{10}$ Bq. The last column of table 11 shows the gamma dose for only 1 g of UC$_x$ at 1 m distance without any shielding.

The tritium production in the UC$_x$ is estimated to be $\sim 10^9$ atoms/second. This becomes, after 7 days of irradiation, about $6 \times 10^{14}$ atoms of $^3$H, with an activity of about $10^6$ Bq.

As far as the alpha activity is concerned, while the $^{238}$U has an activity of $3 \times 10^5$ Bq, the maximum alpha activity of the disks is $6 \times 10^8$ Bq. This is mainly (but not only) due to the formation of $^{236}$Pu.

The behaviour of the gamma source is very similar to that of the activity: the ratio between the activity (disintegrations per second) and the gamma source (gammas emitted per second), turns out to vary between 1.3 and 0.3 in the cooling period considered. For the dose calculations also the spectra of the emitted gammas have been taken into account (most of the gammas lie in the energy range between 0.01 MeV and 2 MeV).

### Table 10.11. Activation of the UC$_x$ disks (28.6 g).

<table>
<thead>
<tr>
<th>IRRADIATION STEPS</th>
<th>Activation</th>
<th>Gamma source</th>
<th>Gamma dose* for 1 gram at 1 metre without shielding</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Bq</td>
<td>Bq/kg</td>
<td>cm$^3$ s$^{-1}$</td>
</tr>
<tr>
<td>0 Secs</td>
<td>3.0E+05</td>
<td>1.0E+07</td>
<td>2.3E+03</td>
</tr>
<tr>
<td>1 Secs</td>
<td>4.3E+11</td>
<td>1.5E+13</td>
<td>7.7E+10</td>
</tr>
<tr>
<td>1 Mins</td>
<td>4.1E+12</td>
<td>1.4E+14</td>
<td>5.9E+11</td>
</tr>
<tr>
<td>1 Hours</td>
<td>1.1E+13</td>
<td>3.7E+14</td>
<td>1.4E+12</td>
</tr>
<tr>
<td>1 Days</td>
<td>1.5E+13</td>
<td>5.2E+14</td>
<td>1.9E+12</td>
</tr>
<tr>
<td>4 Days</td>
<td>1.7E+13</td>
<td>6.0E+14</td>
<td>2.0E+12</td>
</tr>
<tr>
<td>7 Days</td>
<td>1.8E+13</td>
<td>6.3E+14</td>
<td>2.1E+12</td>
</tr>
<tr>
<td>COOLING STEPS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1 Secs</td>
<td>1.8E+13</td>
<td>6.1E+14</td>
<td>2.0E+12</td>
</tr>
<tr>
<td>1 Hours</td>
<td>7.4E+12</td>
<td>2.6E+14</td>
<td>6.7E+11</td>
</tr>
<tr>
<td>1 Days</td>
<td>3.0E+12</td>
<td>1.1E+14</td>
<td>2.5E+11</td>
</tr>
<tr>
<td>3 Days</td>
<td>1.6E+12</td>
<td>5.6E+13</td>
<td>1.4E+11</td>
</tr>
<tr>
<td>30 Days</td>
<td>1.6E+11</td>
<td>5.5E+12</td>
<td>1.2E+10</td>
</tr>
<tr>
<td>90 Days</td>
<td>4.5E+10</td>
<td>1.6E+12</td>
<td>2.4E+09</td>
</tr>
<tr>
<td>365 Days</td>
<td>9.0E+09</td>
<td>3.1E+11</td>
<td>3.2E+08</td>
</tr>
<tr>
<td>10 Years</td>
<td>5.6E+08</td>
<td>1.9E+10</td>
<td>2.2E+07</td>
</tr>
<tr>
<td>100 Years</td>
<td>1.1E+08</td>
<td>3.7E+09</td>
<td>2.8E+06</td>
</tr>
</tbody>
</table>

* The method for this gamma dose calculation is reported in [15].
Fig. 10.9 Activity of the UC$_x$ disks during irradiation.

Fig. 10.10 Activity of the UC$_x$ disks after seven irradiation days.

Since the gamma dose is approximately proportional to the gamma flux (if the spectra do not change too much) and the gamma flux is proportional to the gamma source, we can expect that the gamma dose is approximately proportional to the gamma source of table 11. The gamma doses over the anticorodal sphere after 3 cooling days and after 365 cooling days are reported in table 12.

Table 10.12 Calculated γ dose. The radius of the anticorodal sphere is r=15.5cm.

<table>
<thead>
<tr>
<th>Source</th>
<th>Average γ dose on the anticorodal sphere</th>
<th>Max γ dose on the anticorodal sphere</th>
</tr>
</thead>
<tbody>
<tr>
<td>3 cooling days</td>
<td>1.6·10$^{12}$ s$^{-1}$</td>
<td>−4 Sv/h</td>
</tr>
<tr>
<td>365 cooling days</td>
<td>3.7·10$^9$ s$^{-1}$</td>
<td>−8 mSv/h</td>
</tr>
</tbody>
</table>
The doses are still high even after 365 cooling days. Monoenergetic gamma rays of 0.6 MeV (that is the average energy of the gammas) can be attenuated by a factor of 10 by a lead shielding thickness of about 1.6 cm [18].

10.10.5 The gamma dose from the anticorodal

The activation of the anticorodal sphere around the target is evaluated here. The neutron flux averaged over the anticorodal turns out to be $2.8 \times 10^{10} \text{ cm}^{-2} \text{ s}^{-1}$ with a flux-averaged energy of 1.8 MeV. The activation results, obtained from this neutron flux, are shown in Table 13: after one irradiation week, the activation turns out to be $2.3 \times 10^{10}$ Bq. The gamma source (s⁻¹) of the anticorodal (due to neutron activation) is negligible (<0.2%) with respect to the gamma source of the UC₆ disks (due to proton activation).

### Table 10.13 Activation of the anticorodal sphere

(inner radius=15 cm, outer radius=15.5 cm, $\rho$=2.7 g/cm³, mass=3.9 kg).

<table>
<thead>
<tr>
<th>IRRADIATION STEPS</th>
<th>Activation</th>
<th>Gamma source</th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Bq</td>
<td>Bq/kg</td>
<td>cm⁻³ s⁻¹</td>
<td>s⁻¹</td>
<td>Average energy (MeV)</td>
<td>Sv/h</td>
</tr>
<tr>
<td>1 Secs</td>
<td>2.1E+09</td>
<td>5.2E+08</td>
<td>1.4E+06</td>
<td>2.0E+09</td>
<td>0.5</td>
<td>3.4E-08</td>
</tr>
<tr>
<td>1 Mins</td>
<td>3.6E+09</td>
<td>9.2E+08</td>
<td>2.5E+06</td>
<td>3.7E+09</td>
<td>0.8</td>
<td>9.3E-08</td>
</tr>
<tr>
<td>1 Hours</td>
<td>1.6E+10</td>
<td>4.1E+09</td>
<td>1.1E+07</td>
<td>1.6E+10</td>
<td>1.0</td>
<td>5.1E-07</td>
</tr>
<tr>
<td>1 Days</td>
<td>2.0E+10</td>
<td>5.1E+09</td>
<td>1.7E+07</td>
<td>2.4E+10</td>
<td>1.4</td>
<td>9.6E-07</td>
</tr>
<tr>
<td>4 Days</td>
<td>2.2E+10</td>
<td>5.7E+09</td>
<td>1.9E+07</td>
<td>2.8E+10</td>
<td>1.5</td>
<td>1.2E-06</td>
</tr>
<tr>
<td>7 Days</td>
<td>2.3E+10</td>
<td>5.7E+09</td>
<td>2.0E+07</td>
<td>2.9E+10</td>
<td>1.5</td>
<td>1.2E-06</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>COOLING STEPS</th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Bq</td>
<td>Bq/kg</td>
<td>cm⁻³ s⁻¹</td>
<td>s⁻¹</td>
<td>Average energy (MeV)</td>
<td>Sv/h</td>
</tr>
<tr>
<td>1 Secs</td>
<td>2.0E+10</td>
<td>5.2E+09</td>
<td>1.8E+07</td>
<td>2.7E+10</td>
<td>1.5</td>
<td>1.2E-06</td>
</tr>
<tr>
<td>1 Hours</td>
<td>6.5E+09</td>
<td>1.6E+09</td>
<td>8.8E+06</td>
<td>1.3E+10</td>
<td>2.0</td>
<td>6.9E-07</td>
</tr>
<tr>
<td>1 Days</td>
<td>2.1E+09</td>
<td>5.2E+08</td>
<td>2.8E+06</td>
<td>4.2E+09</td>
<td>2.0</td>
<td>2.2E-07</td>
</tr>
<tr>
<td>3 Days</td>
<td>2.3E+08</td>
<td>5.8E+07</td>
<td>3.1E+05</td>
<td>4.5E+08</td>
<td>2.0</td>
<td>2.4E-08</td>
</tr>
<tr>
<td>30 Days</td>
<td>4.9E+06</td>
<td>1.2E+06</td>
<td>2.8E+03</td>
<td>4.1E+06</td>
<td>0.6</td>
<td>8.3E-11</td>
</tr>
<tr>
<td>90 Days</td>
<td>3.3E+06</td>
<td>8.4E+05</td>
<td>2.3E+03</td>
<td>3.3E+06</td>
<td>0.6</td>
<td>7.1E-11</td>
</tr>
<tr>
<td>365 Days</td>
<td>1.7E+06</td>
<td>4.3E+05</td>
<td>1.2E+03</td>
<td>1.7E+06</td>
<td>0.7</td>
<td>3.8E-11</td>
</tr>
<tr>
<td>10 Years</td>
<td>9.5E+04</td>
<td>2.4E+04</td>
<td>5.7E+00</td>
<td>8.3E+03</td>
<td>0.3</td>
<td>8.1E-14</td>
</tr>
<tr>
<td>100 Years</td>
<td>1.3E+03</td>
<td>3.3E+02</td>
<td>8.6E-02</td>
<td>1.3E+02</td>
<td>1.0</td>
<td>3.8E-15</td>
</tr>
</tbody>
</table>

* The method for this gamma dose calculation is reported in [15].
The dose rate due to beta emission has not been evaluated here. Since the anticorodal is thin (5 mm) this contribution will not be negligible compared with its gamma emission.

The activation of the “front-end” (that is of the part of the remaining materials after the removal of the sphere containing the target) from the beam losses and from the neutron flux has not been considered here. Nevertheless, if there are some anticorodal materials at a distance of r=30 cm from the centre of the target, the neutron activation (of the anticorodal) can be estimated. Since the neutron flux varies approximately with the square of the distance, at r=30 cm the neutron flux can be estimated using that at r=15.5 cm by multiplying by the factor (15.5^2/30^2)=0.267. Then the activations (in Bq/kg) and the gamma doses (in Sv/h of 1 g at 1 m) can be obtained by multiplying the values of table 13 by the same factor 0.267.

Table 14 report the total radioactivity of the target with shielding as can be used to evaluate the total dose during the operation of target handling.

The target radioactivity will reach a saturation level at 2E13 Bq after 14 days of continuum operation. The radioactive dose during the cooling time was evaluated and reported in Tab.13.

After 14 days of cooling down, the target can be removed with some care but in quite safe conditions. A minimum shield of 2 cm of lead and an operation distance of 2 m, allows an eventual manual operation with a total dose of 1mSv/h. An operation of 5min gives a dose of 83 µSv, quite low respect to the 20 mSv max dose/year of classified personnel.

### 10.10.6 Radioactive gases

Several elements produced in the target have melting point less than 50°C and are considered as gases. Although the objective is to ionize as large a fraction of radioisotopes in the 1+ source, a significant fraction is expected to leak as natural atoms and is handled as residual gas in the pumping system. To mitigate the risk to activate the vacuum system and to avoid the release of radioactive gasses in the environment, two actions are actuated:

1) a cryogenic trap will be used after the HRIS to prevent the eventual contamination of the beam line and the vacuum system of the post accelerator. A design of such a trap is currently performed at SPIRAL2, it...
is based on cryogenic layers at different temperatures and active carbon, to stop with high efficiency the different radionuclide gasses according to their properties.

2) the exhaust gas of the vacuum system will be filtered and stored in silos for a given time and the release will be submitted to the measure of radioactivity and further filtering.

The evaluated activity due to the gas is 4E12 Bq after 3 months of operation and it reduces to 2.7E9 Bq and 7E8 Bq after 3 months and 1 year of storage respectively.

10.10.7 Induced radioactivity in air and refrigerating fluids

The presence of a neutron flux may be responsible for the activation of air and refrigerating fluids in the target area. An evaluation will be performed for SPES, but as an indication one can consider the study performed for ISAC at TRIUMF [19] which quote the following annual release rates produced by a proton beam of 500MeV, 100µA on a UCx target of ~100 gr.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Annual release rate (Bq)</th>
<th>Bq gr⁻¹</th>
</tr>
</thead>
<tbody>
<tr>
<td>3H (12y)</td>
<td>1.5E5</td>
<td>1.7E-5</td>
</tr>
<tr>
<td>7Be (53d)</td>
<td>1.2E6</td>
<td>1.4E-4</td>
</tr>
<tr>
<td>11C (20m)</td>
<td>8.6E9</td>
<td>9.5E-1</td>
</tr>
<tr>
<td>13N (10m)</td>
<td>6.9E10</td>
<td>7.7</td>
</tr>
<tr>
<td>15O (2m)</td>
<td>1.0E11</td>
<td>11</td>
</tr>
<tr>
<td>41Ar (109m)</td>
<td>4.1E9</td>
<td>4.6E-1</td>
</tr>
</tbody>
</table>

1 year=3.1E7 s, 
air flux=0.23 m³/s, 
Air density=1250 gr/m³

To overcome the risk of dispersion of radioactivity a careful design of the ventilation system will be performed.

The refrigerating fluids of the target station will have a separate circuit and will be monitored for radiation activity as well as the refrigeration fluid of the mass separator.

No problems are expected for the other systems as the activation is induced by neutron reaction and the larger neutron field is confined around the production Target.

10.10.8 Ventilation system

The ventilation system will be designed to maintain the target area at lower pressure to avoid the escape of airborne radioactivity. Three separate systems with controlled differential pressures will be used in the Target Building: a small one for the target cave, a larger one for the surrounding area (mass separator) and finally the main ventilation system of the Target Building. The zoning is such that air flows from areas of lower to areas of higher potential contamination. The Target cave is the most critical area respect to the presence of contaminated air and the ventilation system will be designed to maintain as low as possible the air change in this zone. Special care will be devoted to seal it at the best.

The system will be designed according to the request of a nuclear ventilation system with HEPA and charcoal filters monitored for pressure loss and radiation activation.

10.10.9 Concrete study for radiation protection applications

The radiation tolerance of concrete is an important item for the safety of high radiation structures. Special care is devoted to the study of cement based materials to be used for bunker construction and for the sarcophagus of long term target containment.
The radiation damage induced on concrete affects the fire resistance and the long term stability. For the SPES project special materials will be used in the critical points.

A specific study on this subject is undergoing at Padova University, Dipartimento Costruzioni e trasporti. In the following the basic approach is outlined.

The study of the long-term behaviour of cement-based materials in relationship with nuclear management requires the assessment of a long-term stability of concrete vessels with emphasis on material-mechanical properties. Different environmental conditions - atmospheric and water saturated environments - could affect the material and structural response. Corrosion and air carbonation of reinforced concrete can be identified as the two main phenomena which could jeopardize the long-term performances of cemented containers.

Additionally, concrete vessels can undergo radiation-induced degradation by alkali-silica reaction of aggregates: the effect of Ar ion irradiation on the reactivity of crystalline and amorphous quartz to alkali has been examined by Ichikawa and Koizumi [20] for clarifying whether radiation from nuclear reactors accelerates the degradation of concrete by inducing alkali-silica reaction of aggregates. Distorted amorphous quartz generated on the surface of quartz by irradiation of a 200 keV Ar ion beam is at least 700 times and 2.5 times more reactive to alkali than crystalline and regular amorphous quartz, respectively. The high reactivity of the distorted amorphous quartz indicates that the degradation of concrete by alkali-silica reaction is possible to be induced by nuclear radiation even if the aggregates are inert to alkali before the irradiation.

Rutherford back-scattering spectrometry, combined with ion channelling’s data, suggest the possible formation of a radiation-induced metastable phase in the damaged region, which may be analogous to pressure- or temperature-induced phase transformations in other ilmenite-group oxides. In particular, these materials transform to either the lithium niobate or the orthorhombic perovskite structure at high pressures and temperatures. These results and similar investigations on the olivine system suggest that ionicity, composition and melting temperature may play important roles in the radiation response of ceramics, and particularly in predicting the relative radiation tolerance of materials within a solid-solution series.

On the basis of such first experimental evidences, a fully coupled chemo-thermo-hydro-mechanical model able to describe concrete behaviour under medium and high temperatures is proposed to assess the radiation damage induced by nuclear radiation within concrete vessels and radioactive leaching within waste storage concrete systems. The model incorporates coupled elastoplastic and damage behaviour plus creep effects. The stress–strain numerical model is derived according to thermodynamic consistency and is based on real experimental findings [21-26].

The objectives of the modelling are:

- characterization of the safety performances of ordinary, high performance and ultra-high performance concrete structures devoted to radiation protection and nuclear waste storage;
- definition of long-term scenarios for risk assessment and concrete durability to radiation phenomena and radioactive leaching;
- evaluation of minimum cemented thicknesses for environmental safety assessment in standard conditions;
- evaluation of the structural response under extreme accidents (fire, earthquake, blast) and proposal of remediation measures to possible spalling, fracturing, collapse.

10.11 Safety analysis

10.11.1 Normal situation

The radiation protection program reports on the zone classification of the facility and on the access controls.
The respect of the radiological zone will be monitored by specific barriers and real-time dose measurement. The access control system and the dose monitoring will be integrated in the operation control system of the facility to prevent dangerous situations, as loss of primary proton beam over the allowed limits and over-contamination of the facility components.

The radioactivity transported in the facility will be monitored and at least two barriers are installed between radioactive zone and the environment. Interlock valves ensure the containment of the target and a second barrier is the vacuum system. The target area is inside a two layer bunker with controlled and separate ventilation.

Additional monitoring and bunkers will be added in the existing rooms if necessary, according to the radiation level of elements as High Resolution Isotope Spectrometer or Charge Breeder.

An adequate fire prevention system will be installed in the new buildings and integrated with the existing one. The water used for fire extinguish will be collected in separate sump to allows the radioactivity controlled before the intake in the exhaust system.

10.11.2 Incidental situations

The incidental situations will be evaluated and managed in such a way that all significant impacts will be confined inside the site and no hazards to the environment will be introduced.

The major incidental situation is related to the sublimation of the target. This can be induced by a malfunctioning of the heating control system or by an error in the primary-beam handling.

The impact on the installation is the contamination of the beam lines downstream from the target and the over-contamination of the vacuum system.

The incident is mitigated by the valves along the beam line that will be closed by the control system.

The alarm for this incident can be deduced by several parameters: the heating of the target (controlled by a pyrometer and thermocouples), a sudden decrease of the vacuum in the target box (controlled by vacuum measurement).

The parameters (current, shape and energy) of the primary beam are controlled by the beam monitor system and fast valves are introduced on the beam line to interlock the beam in case of beam anomalies.

The activity injected into the facility is two times the expected one owing a target efficiency of 50%

No radiological impact on the environment is expected as the vacuum system exhaust is submitted to a storage tank.

In the scale of hazard the next serious incident is represented by the target window breaking. It has a similar impact on the installation but with fewer consequences from the contamination point of view.

The main alert comes from the vacuum monitor in the target and +1 source vessel.

The same safety systems as in target sublimation will be activated.

10.11.3 Accidental situations

The accidental situations that may cause a release of radioactive material to outside of the site will be carefully considered and countermeasures will be adopted to minimize the impact on the environment with the goal to reduce the dispersion to 1Bq/gr.

The worst situation is fire in the target bunker.

The fire prevention system will act to extinguish the fire without the use of water and the bunker will be permanently closed. All ventilation ducts and the pipelines of the refrigerating fluids will be interlocked.

The bunker will be constructed as small as possible and will be closed inside a second bunker.

The injured target station will be put off line and the access will be allowed only after a careful evaluation of the contamination eventually present inside the bunker.

The main risk of radioactivity dispersion is related to the ventilation system. The very worst case is that the whole target material is put into suspension and is processed by the ventilation system.
In this case the larger part of the particulate is trapped in the HEPA filters, if all gaseous elements are released outside the facility the radiological impact is evaluated to be of 2 mSv without the charcoal filtering and 0.4 mSv if charcoal filters are effective [19].

No dispersion of radioactivity is expected if the target vessel does not break.

10.12 Risk evaluation

A detailed analysis of the risk will be performed by a specialized company with special regard to the radiation aspects. Generic and specific hazards will be analyzed. Among them earthquake, fire, overflow, explosion will be considered.

[7] ENEA fpn-p815-007