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Radiological Characterization of Reactor Pressure Vessel Components for Decommissioning Planning: the Case of the Enrico Fermi Nuclear Power Plant

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Abstract

The decommissioning of nuclear power plants represents one of the most challenging phases of the nuclear industry lifecycle. This process involves the coordinated implementation of engineering, radiological protection and radioactive waste management activities. Within this framework, the radiological characterization of activated and contaminated structures is essential for planning dismantling operations and demonstrating compliance with regulatory requirements.

This work is carried out within the framework of the decommissioning activities of the “Enrico Fermi” nuclear power plant in Trino Vercellese and focuses on the radiological characterization of the Reactor Pressure Vessel (RPV) and its internals. The objective is to analyze the radiological behavior of the main structural components and to provide technical support for the definition of dismantling and radioactive waste management strategies.

The analysis considers the mechanisms governing the radiological inventory, distinguishing between neutron activation of structural materials and surface contamination originating from the primary circuit, in relation to the geometric and operational characteristics of the system. The experimental activity is based on the application of radiological measurement techniques, including high-resolution gamma spectrometry using High-Purity Germanium (HPGe) detectors, complemented by surface contamination measurements and radiological air sampling.

The results highlight a non-uniform spatial distribution of activity and confirm the predominance of activation radionuclides in structural materials, while surface contamination appears to be limited and localized.

The study enables a clear distinction between components affected by volumetric activation and those characterized by surface contamination, providing useful elements for radioactive waste classification, selection of dismantling techniques, and radiation protection planning. The results constitute a technical reference supporting the decommissioning activities carried out by Sogin S.p.A.

The main contribution of this work lies in advancing the understanding of the radiological behaviour of activated components and in the methodological approach adopted for their characterization. The expansion of the sampling program, combined with the integration of experimental data and neutron activation modelling, is expected to enhance predictive capabilities and further strengthen the relationship between theoretical assessments and operational evidence.

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List of Acronyms

ALARA	As Low As Reasonably Achievable
DIS	Direct Ion Storage
EPRI	Electric Power Research Institute
EURATOM	European Atomic Energy Community
FWHM	Full Width at Half Maximum
GSR	General Safety Requirements
HPGe	High-Purity Germanium
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
ISIN	Ispettorato Nazionale per la Sicurezza Nucleare e la Radioprotezione
ISOCS	In Situ Object Counting System
Lc	Critical Level
MCA	Multi-Channel Analyzer
MCNP	Monte Carlo N-Particle
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission Regulation
OECD-NEA	Organization for Economic Co-operation and Development Nuclear Energy Agency
PWR	Pressurized Water Reactor
QA	Quality Assurance
RIA	Radiation in Air
RIC	Radiation in Liquid
RPV	Reactor Pressure Vessel
SCALE	Standardized Computer Analyses for Licensing Evaluation
TDL	Thermoluminescent Dosimeter
TRS	Technical Reports Series
WBC	Whole Body Counter

1.Introduction

The present work is developed within the framework of nuclear power plant decommissioning activities and focuses on the radiological characterization of the Reactor Pressure Vessel and its internals at the Trino Vercellese site. The objective is to analyze the available experimental results and assess their implications for the subsequent dismantling phases.

1.1 General Framework of Decommissioning

The decommissioning of nuclear facilities currently represents a crucial phase in the lifecycle of installations that have used radioactive materials. With the progressive shutdown of first generation reactors and the technological maturation of national programs, dismantling activities have become a structured engineering field, regulated by international standards and supported by extensive operational experience.

The decommissioning process is not limited to the mere demolition of the facility; rather, it encompasses a complex set of technical, radiological protection, and managerial operations aimed at the safe removal of contaminated or activated structures, the management of materials and radioactive waste, and the restoration of the site according to criteria defined by the competent authority.

Such complexity requires an integrated approach involving radiological characterization, activity planning, environmental impact assessment, and the adoption of specific strategies for the most critical components, such as the reactor pressure vessel and internals of Pressurized Water Reactors.

Within this framework, the present work focuses on the radiological characterization activities carried out at the "Enrico Fermi" nuclear power plant in Trino Vercellese and their relevance for the upcoming decommissioning phases.

1.2 State of the Art in PWR Primary Components Decommissioning

The decommissioning of Pressurized Water Reactors (PWR) represents one of the most mature fields in applied nuclear engineering, thanks to the experience accumulated over the past forty years in Europe, the United States, and Japan. This experience has enabled the development of consolidated methodologies for dismantling large components, particularly the Reactor Pressure Vessel (RPV) and internals, which are the parts most affected by neutron activation.

International guidelines, together with operational experience gained from major decommissioning programs, constitute the reference framework within which any characterization and decommissioning activity is currently conducted.

According to the standards of the International Atomic Energy Agency (IAEA), decommissioning planning must be structured as a risk informed process based on radiological hazard assessment and the application of the As Low As Reasonably Achievable (ALARA) principle, both during preliminary characterization and during cutting, handling, and conditioning operations [1]. The IAEA General Safety Requirements (GSR) Part 6 document also emphasizes the importance of integrating radiological, engineering, and logistical aspects, since large components require dedicated strategies to minimize worker exposure and reduce radioactive waste generation.

Organisation for Economic Co-operation and Development Nuclear Energy Agency (OECD NEA) literature confirms these approaches. In the report Decommissioning of Nuclear Facilities [2], the agency identifies the most effective methodologies adopted in industrialized countries, highlighting how the current trend favors immediate dismantling over deferred dismantling. This choice is motivated by the availability of mature underwater segmentation technologies, advancements in radiological instrumentation, and the need to preserve technical expertise within the organizations that operated the plant.

Regarding PWR primary components, studies conducted by the Electric Power Research Institute (EPRI) represent one of the main technical references. The report PWR Decommissioning Experience [3] collects data from numerous United States plants, showing that internals segmentation is primarily performed using mechanical techniques such as diamond wire cutting and robotic blade systems or thermal techniques such as plasma arc cutting, selected based on component geometry and metallurgical condition. Underwater cutting is systematically adopted to ensure natural shielding provided by water and to reduce exposure to gamma radiation from activated radionuclides, particularly 60Co and 54Mn.

Neutron activation analysis remains a key aspect in the radiological assessment of the vessel and internals. IAEA Technical Report Serie (TRS) No. 393 and the more recent TRS No. 399 [4] provide a detailed overview of the most relevant nuclear reactions in steel alloys, including the production of long lived radionuclides such as 59Ni and 94Nb, which are of particular importance for waste classification. These reports highlight the importance of integrating experimental measurements with simulations based on neutron transport codes such as Monte Carlo N-Particle (MCNP) or Standardized Computer Analyses for Licensing Evaluation (SCALE), which are essential for reconstructing historical neutron fluence in areas where direct sampling is difficult or not feasible.

From an operational perspective, significant experience has also been gained from European decommissioning programs. In Germany, the dismantling of PWR plants such as Stade, Obrigheim, and Unterweser. Those plants have provided advanced methodologies for internals management, with particular emphasis on preliminary radiological characterization and the classification of components into waste categories according to national disposal requirements [5]. In France, the EDF (Électricité de France) decommissioning program has developed standardized procedures for reactor pressure vessel removal under optimized radiological protection conditions, with strong integration between neutron fluence modeling and targeted sampling [6].

Another important contribution comes from the United States Nuclear Regulatory Commission (NRC), from its Nuclear Regulatory Commission Regulation(NUREG), particularly through documents such as NUREG 1801 and NUREG CR 6801, which analyze the effects of neutron irradiation on materials and their implications for decommissioning strategies [7]. Although originally intended for plant aging management, the data on metallurgical effects and structural changes in irradiated components are highly relevant for defining cutting, handling, and conditioning strategies.

Overall, international studies converge on several key elements defining the state of the art in PWR decommissioning:

1. The importance of preliminary radiological characterization, including spectrometric analyses, mechanical sampling, in situ measurements, and neutron simulations.

2. The need for robust predictive models capable of estimating residual activity and supporting the selection of sampling locations.

3. The adoption of consolidated cutting and segmentation techniques, with a strong preference for underwater segmentation for radiological protection purposes.
4. The integration of radiological safety, waste conditioning requirements, and site logistics, as emphasized in all major international reports.
5. The standardization of procedures, which is now essential to reduce uncertainties and ensure reproducibility of results.

These contributions define the methodological and operational framework underpinning the present work and guiding the characterization activities carried out at the Trino Vercellese site.

1.3 The Trino Vercellese Site: Historical Overview and Decommissioning Status

The Trino Vercellese site, located in the province of Vercelli, hosts the Enrico Fermi nuclear power plant, the first industrial nuclear power plant built in Italy. Construction began in 1961, and the plant entered commercial operation in 1965. Following the nationalization of the electricity sector, management was transferred to ENEL, the Italian multinational energy company, which operated the plant until 1987. In that year, at the end of the ninth fuel cycle, the plant was permanently shut down as a consequence of the political decision following the national referendum on nuclear power.

The reactor, a Westinghouse designed pressurized water reactor PWR, had a nominal electric output of 270 MWe. During its operational lifetime, the plant generated approximately 26 billion kWh, making it one of the most productive and long lived nuclear facilities in Italy.

From an engineering standpoint, the Trino plant featured a typical PWR configuration, based on the separation of the primary circuit, secondary circuit, and ultimate heat sink system. The primary circuit, housed within the containment building, was responsible for removing the heat generated in the reactor core by means of water maintained in liquid state at high pressure. The main components of the primary circuit included the reactor vessel containing the core and internals, the steam generators, the pressurizer, and the primary pumps, all connected through large diameter piping.

A simplified schematic of the Trino nuclear power plant is shown in Figure 1.1, highlighting the main systems and components of the primary circuit.

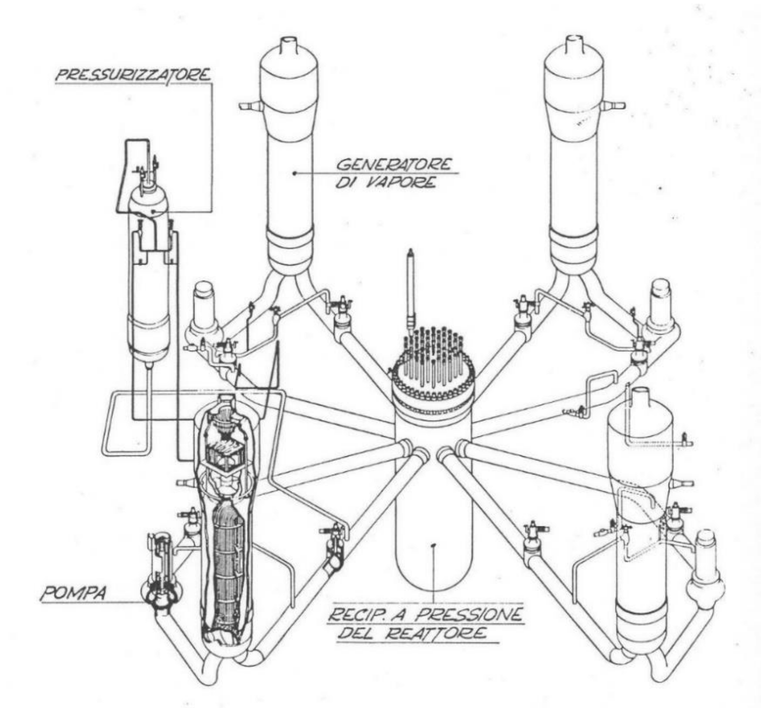
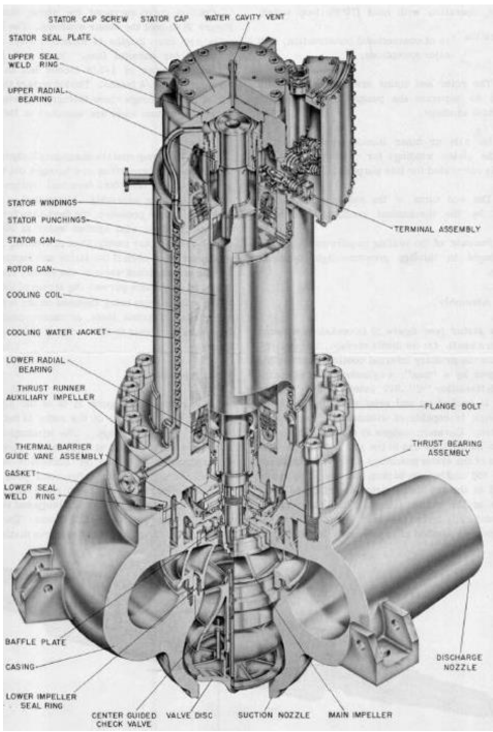


Figure 1.1 Primary circuit schematic of the Trino NPP

The primary pumps, located along the cold legs of the circuit, ensured forced circulation of the coolant through the reactor core and steam generators, thus enabling efficient removal of fission heat.
An example of a PWR primary pump installed at the Trino plant is shown in Figure 1.2.



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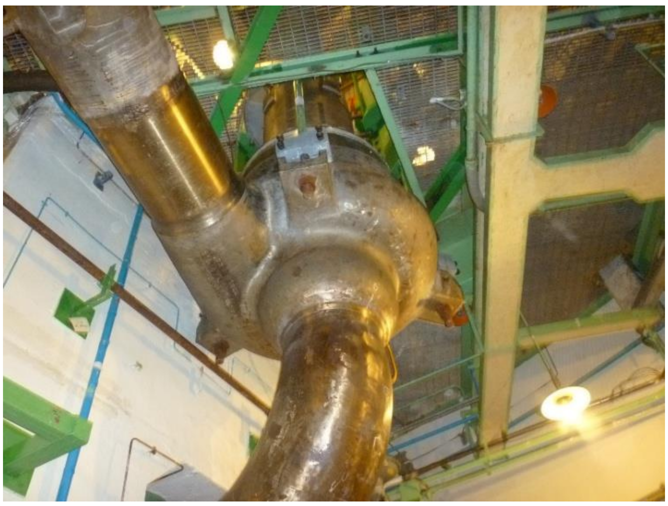
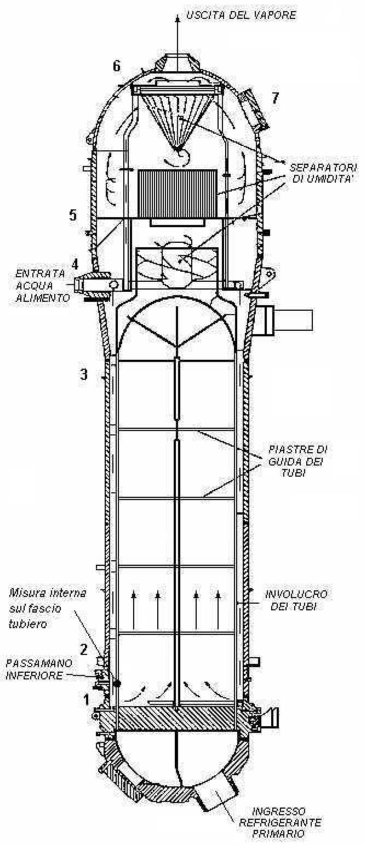


(b)

Figure 2.2 (a)Primary coolant pump schematic, (b) Real primary pump

The steam generators acted as the thermal interface between the primary and secondary circuits. Within them, heat from the primary coolant was transferred to the secondary side water, producing steam to drive the turbine generator system. The steam generators at Trino were vertical shell and tube units, a typical configuration for Westinghouse PWRs of that period. A representative view of a PWR steam generator is shown in Figure 1.3.



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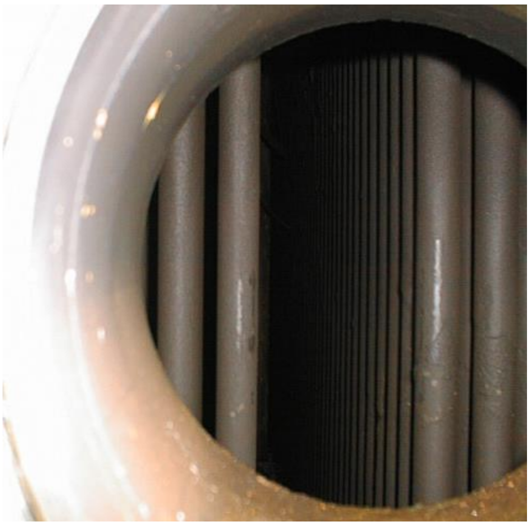
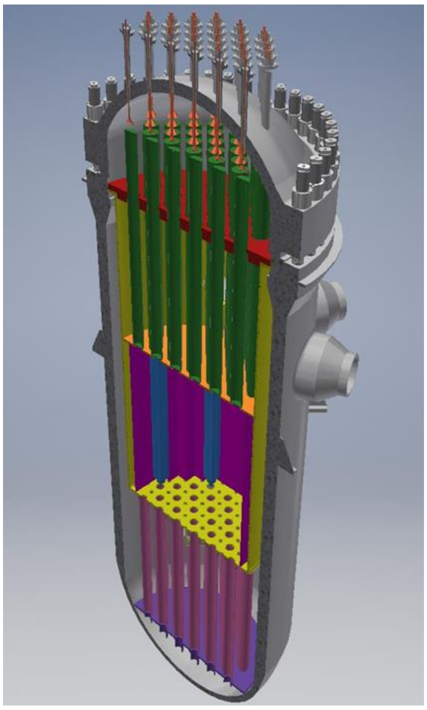


(b)

Figure 1.3 (a) Steam generator schematic, (b) Internal access handrail for in-situ measurements

The reactor pressure vessel represented the main structural component of the nuclear island, housing the core, control rods, and reactor internals. Constructed from thick carbon steel, the vessel ensured containment of the high pressure coolant and maintained the correct core geometry during operation.

Due to the intense neutron flux experienced during operation, the vessel is one of the most highly activated components from a radiological standpoint and represents a key element in decommissioning activities. A view of the reactor vessel of the Trino plant and its main internal components is shown in Figure 1.4.



(a)

(b)

Figure 1.4 (a) Cross-section of the RPV, (b) Position of the RPV within the reactor cavity

Since 1999, the plant has been under the responsibility of SOGIN, the state owned company in charge of decommissioning Italian nuclear facilities. Initial activities focused on the removal of secondary systems. Emergency diesel buildings were demolished, cooling towers dismantled, and the entire secondary cycle including turbines and the turbine hall was decommissioned.

Between 2003 and 2004, chemical decontamination of the primary circuit, particularly the steam generators, was carried out, significantly reducing radioactivity levels. Subsequently, between 2005 and 2008, insulation materials and all primary circuit piping were removed. Between 2013 and 2017, all non-contaminated reactor components were dismantled.

Currently, within the containment building, only the major components remain. These include the reactor vessel with its internals, the steam generators, and the pressurizer. All piping, auxiliary systems, and support systems have been removed and sent for melting.

The Trino plant therefore represents not only a key historical milestone for nuclear energy in Italy, but also a complex and advanced decommissioning project, in which decontamination, dismantling, and remediation activities have already achieved significant results. The final objective is the release of the site for unrestricted use, while temporarily storing the radioactive waste generated, pending its transfer to the National Repository.

1.4 Objectives of the Thesis

The main objective of this work is to critically analyze the radiological characterization activities carried out within the controlled area, with particular attention to the instrumentation employed both in sampling operations and during on site activities. To this end, the study aims to examine the operating principles, performance, and suitability of the radiological instrumentation used on site, in order to assess the correctness of the measurement methods adopted, the reliability of the acquired data, and any technical or operational limitations that may affect the results.

A further objective is the analysis and interpretation of the radiological results obtained from the sampling activities performed on the vessel head. This analysis aims at highlighting the distribution of activity, contamination levels, and the main characteristics of the radiological inventory, relating the results to the plant operational conditions and to the physico-chemical properties of the materials.

In addition to the main objectives, subject to the operational conditions of the site and the availability of data, the work may include a further analysis of samples collected from the upper package and from specific internal regions of the vessel. This activity is considered a secondary and non-mandatory objective, as it depends on the schedule, planned access conditions, and safety requirements during site operations.

Through the analysis of instrumentation and the interpretation of the available experimental data, with possible extension to samples from the internal components of the vessel, the work aims to provide a technical assessment of the quality of radiological measurements and their contribution to the planning of decommissioning activities.

2. Principles of Radioactivity and Radiation Protection

Radioactivity is the physical phenomenon by which certain unstable nuclei, known as radioisotopes, spontaneously transform into more stable nuclei, emitting ionizing radiation. This emission may occur in the form of charged particles or high energy electromagnetic radiation and represents the fundamental mechanism responsible for radiological exposure in nuclear and medical fields. The distinction between stable and unstable isotopes is essential, as only unstable nuclei undergo radioactive decay [8] [9].

2.1 Types of Decay

The main types of radioactive decay are alpha decay (α), beta decay (β^- and β^+), and gamma decay (γ) [8] [9]. Alpha decay involves the emission of particles composed of two protons and two neutrons, characterized by high ionizing capability but low penetration, as they can be stopped by a simple sheet of paper. Beta decay involves the emission of electrons (β^-) or positrons (β^+), with a penetration capacity greater than alpha particles but lower than gamma photons. Gamma decay consists of high energy electromagnetic radiation, which has a high penetrating power through materials, producing relatively low ionization. While charged alpha and beta particles are completely stopped by paper and a 0.5 cm aluminum sheet, uncharged high-energy gamma photons are merely attenuated due to their physical nature. (Figure 2.1)

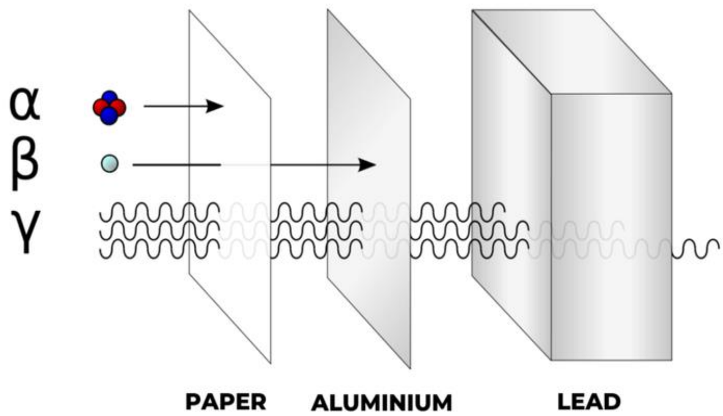


Figure 2.1 Penetration of different types of radiation

2.2 Dosimetric Quantities

Dosimetric quantities are fundamental tools for quantifying exposure to ionizing radiation and assessing the associated biological risk. The absorbed dose (D) represents the energy deposited by radiation per unit mass of irradiated material and is measured in gray (Gy), where 1 Gy = 1 J/kg. However, since different types of radiation produce different biological effects, the absorbed dose alone is not sufficient to evaluate radiological risk [10].

To account for the different ionization capabilities of radiation, the equivalent dose (H) is introduced, measured in sievert (Sv), and calculated by multiplying the absorbed dose (D) by a radiation weighting factor specific to the type of radiation. More strongly ionizing radiation, such as alpha particles, is assigned to a higher weighting factor compared to less ionizing radiation, such as gamma photons.

To evaluate the overall risk to the organism, considering the different sensitivities of tissues, the effective dose (E) is used. This quantity, combines the equivalent dose with tissue weighting factors specific to each organ or tissue. It allows comparison between different exposure conditions and provides an estimate of the risk of stochastic effects, such as radiation induced carcinogenesis. The equivalent dose associated with a single type of radiation is determined by multiplying the radiation weighting factor by the absorbed dose delivered to the tissue. Consequently, the summation of these equivalent doses, each multiplied by their respective tissue weighting factor, yields the effective dose (E), representing the overall radiation risk to the entire organism.

In this framework, three distinct dosimetric quantities can therefore be identified, each corresponding to a different level of evaluation of radiation effects on biological tissues. The *absorbed dose (D)* describes the physical energy deposited by radiation per unit mass of tissue and is directly related to the energy imparted to the material. The *equivalent dose (H)* accounts for the different biological effectiveness of different radiation types by applying radiation weighting factors.

Finally, the *effective dose (E)*, also expressed in sievert (Sv), considers the varying sensitivity of organs and tissues, providing an estimation of the overall risk to the whole body, particularly for stochastic effects such as radiation induced cancer.

$$H_{T,R} = w_R \, D_{T,R}$$

$$E = \sum_T w_T \, H_T = \sum_T w_T \sum_R w_R \, D_{T,R}$$

Where:

- $D_{T,R}$
- $H_{T,R}$
- w_R the radiation weighting factor
- w_T the tissue weighting factor

In Table 2.1 and Table 2.2 the weighting factors for the coefficients w_T and w_R are reported. It should be noted that the sum of all values of w_T is equal to one [10], corresponding to the case in which the whole body is uniformly exposed to radiation.

Table 2.1 Tissue Weighting Factors (w_T) – ICRP 103

Tissue / Organ	Weighting Factor w_T
Bone marrow (red), colon, lung, stomach, breast	0.12 each
Gonads	0.08
Bladder, esophagus, liver, thyroid	0.04 each
Bone surface, brain, salivary glands, skin	0.01 each
Remainder tissues (combined)	0.12

Table 1.2 Radiation Weighting Factors (w_R) – ICRP 103

Radiation Type	**Weighting Factor w_R**
Photons (X-rays, gamma rays)	1
Electrons, muons	1
Neutrons	2.5-20 energy dependent
Protons (other than recoil protons)	2
Alpha particles, heavy ions, fission fragments	20

Finally, other important quantities in radiation protection include activity (A), measured in becquerel (Bq), which indicates the number of decays per second of a radioactive source, and exposure (X), expressed in coulomb per kilogram (C/kg), which quantifies the electric charge produced in air y ionizing radiation (X or gamma rays). The understanding of these quantities is essential for the design of radiation protection systems and for ensuring that operator exposure remains within safe limits.

2.3 Radiation Matter Interactions

When radiation passes through matter, it can interact with atoms and nuclei through different processes, the nature of which depends on the type of particle and its energy. Interactions of photons, charged particles, and neutrons determine beam attenuation and the amount of energy transferred to the material, thereby influencing the physical and biological effects of radiation [8] [9].

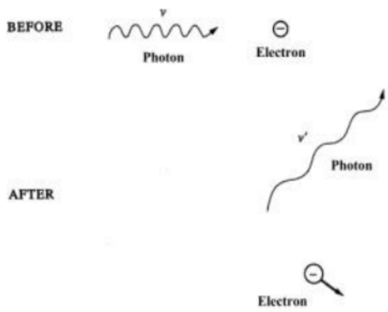
2.3.1 Photons (γ, X)

Photons interact with matter mainly through three fundamental processes: photoelectric effect, Compton effect, and pair production.

In the photoelectric effect, the photon transfers all its energy to an atomic electron, which is ejected. The probability of this process increases in materials with high atomic number and is dominant at low energies. The Compton effect, prevalent at intermediate energies typical of medical diagnostics, consists of an inelastic interaction between the photon and a quasi-free electron, resulting in the production of a Compton electron and a scattered photon with reduced energy. At high energies, above 1.022 MeV, pair production becomes possible, in which the photon interacts with the nuclear field and transforms into an electron positron pair.

The relative probability of each process strongly depends on the photon energy and on the atomic number of the material, thereby determining the attenuation behavior of X rays and gamma rays in different media [8]. In Figure 2.2 the three interactions are schematically shown. Figure 2.3 illustrates the mass attenuation coefficient of lead ($Z = 82$) as a function of energy, highlighting how different photon interaction mechanisms dominate depending on energy. The photoelectric effect prevails at low energies, where the characteristic K, L, and M-edges maximize absorption; Compton scattering becomes dominant at intermediate energies, whereas pair production takes over at high energies. The three main clinical and radiation protection ranges identified are:

- Diagnostic Radiology, dominated by the photoelectric effect in the range between 20 and 150 keV.
- Nuclear Medicine, dominated by the photoelectric and Compton effects in the range between 70 and 511 keV.
- Radiotherapy, dominated by Compton scattering and pair production in the range from over 1.02 MeV up to approximately 20-25 MeV.



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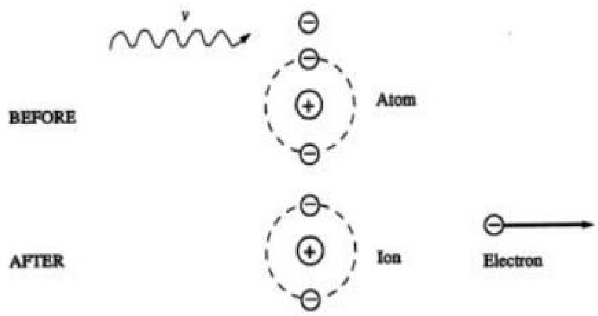

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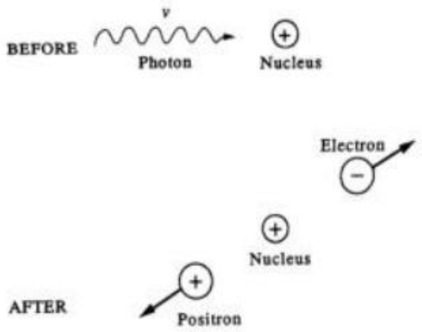


(c)

Figure 2.2 (a) Photoelectric effect, (b) Compton scattering, (c) Pair production

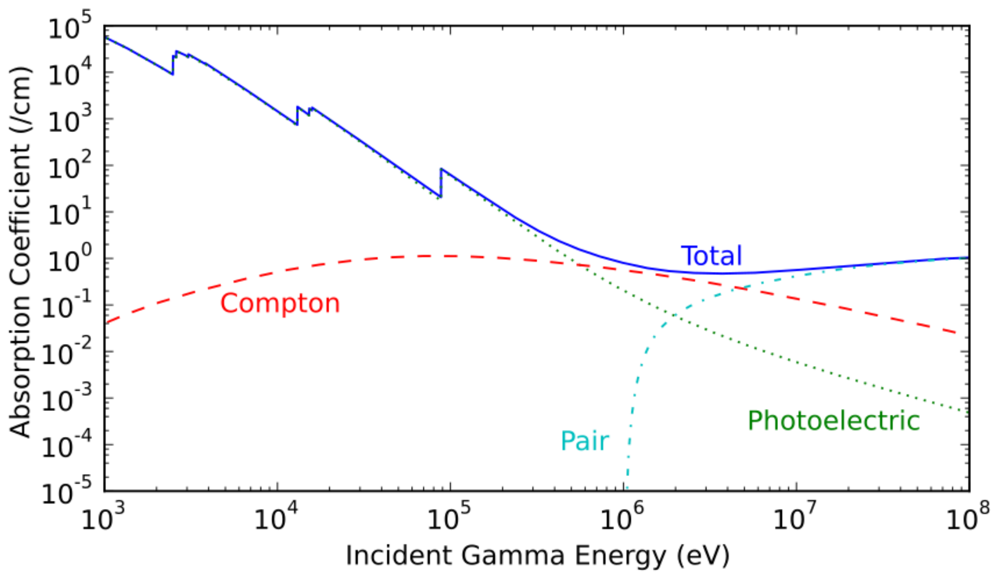


Figure 2.3 *Total attenuation as a function of photon energy for Lead Z=82*

Lead is universally chosen for radiation protection because its high atomic number and density ensure an excellent cross-section for the photoelectric effect, enabling compact and efficient shielding across all these energy ranges.

2.3.2 Charged Particles (α, β, Protons)

Charged particles, such as α, β, and protons, interact with matter primarily through ionization and excitation processes of atoms. Along their path, they transfer energy to orbital electrons, causing either their ejection or their transition to higher energy levels.

Alpha particles, characterized by high mass and double positive charge, produce very intense ionization over an extremely short range. Beta electrons, being lighter, undergo multiple scattering and can generate bremsstrahlung radiation when accelerated or decelerated in the electric fields of nuclei. Protons, with a mass comparable to that of the hydrogen nucleus, lose energy progressively until reaching the characteristic Bragg peak, where the ionization density sharply increases just before the particle comes to rest.

The nature of these interactions determines beam reduction, dose distribution, and the biological effectiveness of different types of charged particles [9].

2.3.3 Neutrons

For completeness, neutrons are also included in the definition of ionizing radiation, although they do not possess electric charge and therefore do not directly ionize matter. Their effects arise from indirect interactions, such as elastic and inelastic scattering with nuclei, neutron capture, and, in the presence of a sufficient flux, nuclear fission, which leads to the formation of daughter nuclei that are typically radioactive.

These fission products can emit additional radiation, such as beta and gamma, contributing to the overall activity of the system. In this context, neutrons will not be treated in detail, as they are not significantly present in the environment under consideration. Since the reactor is currently shut down, there is no neutron production nor radioactivity associated with nuclei generated by ongoing fission processes.

Nevertheless, it is still necessary to manage materials that were activated in the past, such as ^{60}Co formed during reactor operation [9].

The ionization efficiency of different types of particular radiation is schematically presented in Figure 2.4, illustrating the flow and physical nature of the particles emitted during their respective decay processes.

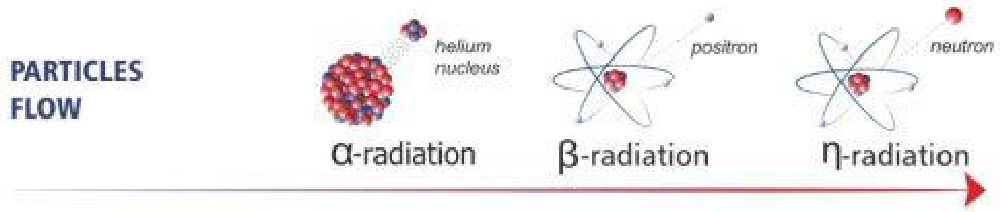


Figure 2.4 Effective ionization efficiency for different types of radiation

2.4 Theoretical and Operational Aspects of Radiation Protection

The three fundamental principles of radiation protection, established by the International Commission on Radiological Protection (ICRP) [10], are:

- Justification: any activity involving exposure to ionizing radiation must be justified, meaning that it must provide a net benefit greater than the associated risks.
- Optimization (ALARA principle): exposure must be kept as low as reasonably achievable, considering economic, social, and technological factors. In this context, operational radiation protection tools are applied.
- Dose limitation: individual doses must not exceed the limits established by current regulations, to prevent deterministic effects and reduce the probability of stochastic effects.

Operational tools of radiation protection:

- Time
- Distance
- Shielding

Reducing exposure time decreases the absorbed dose, as irradiation is proportional to the duration of exposure. Increasing the distance from the source significantly reduces radiation intensity due to spatial dispersion. The use of appropriate shielding, made of suitable materials such as lead, concrete, or water, allows attenuation of the incident radiation level. The attenuation law as a function of the macroscopic cross section of the shielding material can be expressed as follows:

$$I(x) = I_0 e^{-\Sigma x}$$

where I_0 is the initial radiation intensity, $I(x)$ is the intensity after passing through a material thickness x, and Σ represents the macroscopic cross section of the medium, which accounts for the overall probability of interaction between radiation and the shielding material. This exponential law strictly applies to an ideal, narrow beam 1D geometry. In practical 3D scenarios, factors such as radiation scattering (build-up effect) and beam polychromatism cause deviations from a pure exponential trend.

These principles constitute the operational basis for the safe design and management of activities in the nuclear field.

2.5 Regulatory Criteria for Radiation Protection

Italian legislation on radiation protection is primarily based on Legislative Decree of 31 July 2020 n. 101 [11], which transposes Directive 2013 59 EURATOM (European Atomic Energy Community), the European regulatory framework aimed at harmonizing protection criteria against ionizing radiation. The Italian decree is also based on the recommendations of the International Commission on Radiological Protection ICRP, a scientific body that defines principles and exposure limits, and on the guidelines of the International Atomic Energy Agency IAEA, which provide safety standards applied at a global level.

The integration of these three sources, namely European directives, ICRP recommendations [10], and IAEA standards, ensures that the national regulatory framework is consistent with the most up to date scientific criteria and with international best practices in the field of radiation protection.

Considering this international regulatory framework and the alignment of Italy with the standards established by EURATOM, ICRP, and IAEA, Legislative Decree 31 July 2020 n. 101 does not merely adopt general principles but develops a comprehensive and structured system that fully regulates protection against ionizing radiation. The decree defines the foundations of radiation protection and the scope of application of the regulations, establishing criteria for the management of planned, existing, and emergency exposures.

It also establishes the use of reference levels, the classification of workers and working areas, the management obligations required to prevent the exceedance of dose limits, and provisions related to health surveillance. Of particular importance are the dose limits established for exposed workers and for members of the public [10] [11], summarized in Table 2.3, which constitutes the operational reference for exposure assessment and control.

The decree also includes specific provisions concerning the management of radioactive sources and materials, radon risk, emergency planning, and the system of controls and sanctions, designating National Inspectorate for Nuclear Safety and Radiation Protection (ISIN) as the primary regulatory and supervisory authority. The overall framework makes Legislative Decree 101 2020 a comprehensive and coherent instrument, fully aligned with the most advanced criteria of radiological protection at the international level.

Table 2.2 Annual exposure limits - Legislative Decree 2020 n. 101, transposing the European Directive 2013/59/EURATOM.

Per year	Exposed workers		Popolazione
	Category A	Category B	
Effective dose (E)	20 mSv	6 mSv	1 mSv
Equivalent dose (H)			
H, for the lens of the eye	20 mSv	15 mSv	15 mSv
H, for the skin	500 mSv	150 mSv	50 mSv
H, for the extremities	500 mSv	150 mSv	-

3. Operational and Structural characteristics of the Pressure Vessel and Internals for Radiological Characterization

In pressurized water reactors PWR, the components of the primary circuit are characterized by specific radiological issues [3] [7] due to intense exposure to neutrons and gamma radiation generated in the core during operation. The primary circuit includes the reactor vessel, main piping, circulation pumps, pressurizer, and steam generators. These components, operating close to the neutron source and under high temperature and pressure conditions, are subject to activation, contamination, and irradiation phenomena that are of primary importance for radiation protection.

Among all components, the reactor vessel represents the most critical element from the perspective of neutron activation [7]. The vessel is made of low alloy steel and directly surrounds the core, thus being exposed to a high and prolonged neutron flux. The interaction of neutrons with the nuclei of structural materials induces capture and transmutation reactions, leading to the formation of artificial radionuclides with medium to long half-lives.

Impurities and alloying elements such as cobalt, nickel, iron, and manganese can give rise to radioactive isotopes [3] [7] such as ^{60}Co, ^{58}Co, ^{59}Fe e ^{54}Mn, which significantly contribute to the residual radioactivity of the vessel. Vessel activation is a cumulative process, dependent on the irradiation history of the reactor, and results in a progressive increase of emitted gamma dose even after reactor shutdown.

In addition to direct material activation, the primary circuit is affected by the presence of activated corrosion products, generated by erosion and corrosion of metallic surfaces in contact with the primary coolant. These products, transported by the cooling water, are activated in the core and subsequently deposited on the internal surfaces of components, contributing to widespread contamination [3] within the circuit. This phenomenon is particularly relevant during maintenance outages, when the deposition of gamma emitting isotopes on surfaces increases personnel exposure.

The radiological classification of the vessel as highly activated waste represents one of the main challenges in radioactive waste management at the end of plant life, influencing decommissioning strategies [1] [3] and final disposal options.

3.1 General Framework

The analysis of the Reactor Pressure Vessel (RPV) and its internals is carried out from an engineering perspective, with particular reference to the aspects relevant to radiological characterization and the planning of decommissioning activities at the “Enrico Fermi” Nuclear Power Plant in Trino.

The dismantling process of the vessel and its internals is structured into three operational phases, which provide the reference framework for the organization of activities and for the interpretation of characterization results. The main components of the system are therefore analyzed in terms of geometric characteristics, materials, and function, in order to assess their influence on neutron activation and surface contamination phenomena, as well as on the resulting operational implications.

3.2 Dismantling Phases of the Vessel and Internals

Within the decommissioning activities of the “Enrico Fermi” Nuclear Power Plant in Trino, the dismantling of the Reactor Pressure Vessel (RPV) and its internals has been divided into three sequential phases, as reported in the reference technical documentation [12]. The subdivision into three phases is schematically illustrated in Figure 3.1.

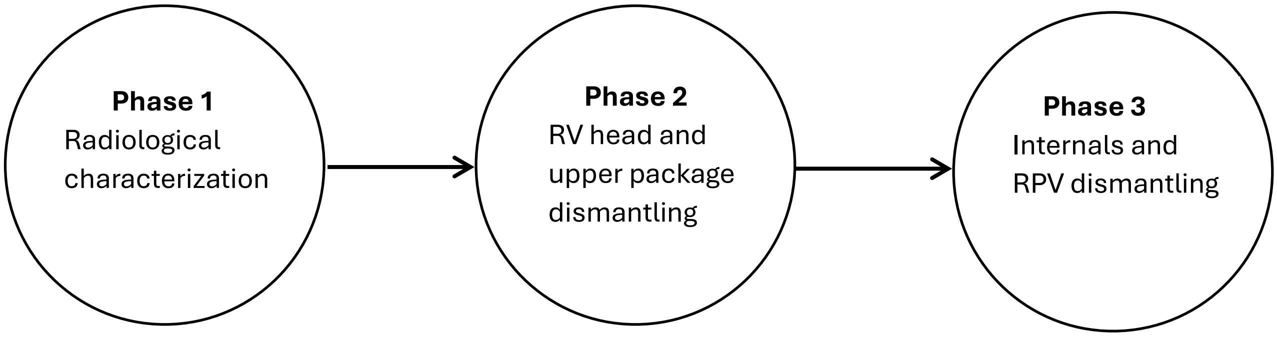


Figure 3.1 Schematic representation of the three phases of vessel and internals decommissioning

The analyzed Operational Plan provides a detailed description exclusively of Phase 1, while the subsequent phases are addressed only in a summary form and referred to future planning documents.

Phase 1 includes the radiological characterization of the vessel and internals, as well as the dismantling of the primary circuit and the auxiliary systems connected to it, excluding large components [12]. The main objective of this phase is to collect the information necessary for the design of subsequent dismantling activities. The operations include the isolation of the vessel through the segregation of the main piping, the removal of primary circuit systems, and subsequently the opening of the vessel and the handling of the internals. In this context, radiological sampling is carried out on the vessel head, upper package, barrel, and the walls of the cylindrical shell, to determine the residual radiological inventory and support material classification.

Phase 2 concerns the dismantling of the vessel head and the upper package [12]. It involves the removal of the main upper internal structures, allowing access to the inner regions of the vessel and preparing the operational conditions for the subsequent phases.

Phase 3 includes the dismantling of activated internals and the cylindrical shell of the vessel and represents the most critical phase from a radiological standpoint [12]. It involves the most highly activated components, located in proximity to the core, and concludes with the removal and segmentation of the vessel itself.

Overall, the subdivision into three phases reflects a gradual approach to dismantling, in which preliminary activities with lower radiological impact are concentrated in Phase 1, while more complex operations characterized by higher activation levels are deferred to the subsequent phases, in accordance with the principles of radiation protection optimization.

3.3 Radiological Characterization Approach of the Vessel and Internals

The Reactor Pressure Vessel (RPV) and the internals of the "Enrico Fermi" Nuclear Power Plant in Trino represent the most relevant structures for the radiological characterization of the facility, as they are subject, during operation, to both neutron activation phenomena and surface contamination from fission and corrosion products transported by the primary coolant. The understanding of the operational, geometrical, and material characteristics of these components is therefore essential for the proper planning of sampling activities, for the estimation of the residual radiological inventory, and for the subsequent management of materials resulting from dismantling.

In the case of the Trino site, the characterization process is part of Phase 1 of the vessel and internals decommissioning project and is aimed at providing reliable experimental data for the design of the subsequent dismantling phases, in compliance with safety and radiation protection requirements.

Phase 1 Radiological Characterization Activities

Within Phase 1, operational activities are aimed at ensuring controlled access to the internal surfaces of the vessel and the internals, a necessary condition for the execution of measurement and sampling campaigns. To this end, preliminary operations are carried out, including vessel isolation and removal of primary circuit systems, as described in the previous section.

Once the internal structures have been made accessible, a systematic radiological characterization program is implemented, based on sampling campaigns representative of the different regions of the vessel and internals. In particular, sampling activities involve components such as the vessel head, the upper package, the barrel, and the walls of the cylindrical shell, selected according to the different neutron irradiation and contamination conditions.

The samples are subjected to radiometric and radiochemical analyses aimed at determining the radionuclide inventory and evaluating the spatial distribution of activity. This information forms the basis for the radiological classification of materials and for the definition of dismantling operational strategies.

3.4 Operational and structural characteristics of the vessel

This section presents the main structural and operational characteristics of the reactor pressure vessel, including its geometrical and functional configuration, the materials and cladding employed, and the main design features of the vessel head.

3.4.1 Geometrical and functional configuration

The reactor vessel is a steel pressure vessel, with an overall height of approximately 11.3 m, consisting of a hemispherical head and a cylindrical shell resting on the Nuclear Shield Tank. The cylindrical shell is divided into upper, middle, and lower sections, as well as a dished bottom, with a constant internal diameter of approximately 3.20 m (Figure 3.2) [12].

During plant operation, the vessel housed the core and ensured the containment of the primary coolant at high pressure and temperature. The four inlet nozzles and four outlet nozzles of the primary coolant, located in the upper shell region, resulted in a non-uniform neutron field and flow regime, leading to activation gradients along both the thickness and the height of the vessel.

Figure 3.2 General view of the RPV

3.4.2 Materials and cladding

The vessel body is made of carbon steel with high mechanical strength, while the internal surface is lined with a stainless steel cladding with a minimum thickness of approximately 4 mm, in direct contact with the primary coolant. This lining has performed a fundamental role in corrosion protection but also represents the main location of surface contamination.

From a radiological perspective, the distinction between base material and internal cladding is crucial. The former is primarily affected by neutron activation phenomena, leading to the production of long lived radionuclides such as ^{60}Co, ^{63}Ni, ^{55}Fe, whereas the latter is mainly subject to contamination from activated products transported within the primary circuit.

3.4.3 Vessel head Configuration and Characteristic

The vessel head, of hemispherical shape and with an internal diameter of approximately 2.95 m, is equipped with 52 penetrations, of which 28 are dedicated to the control rod drive mechanisms [12]. It is connected to the cylindrical shell by means of 42 large diameter studs as shown in Figure 3.3, with a detail in the upper right corner. During operation, the head was exposed to a neutron flux significantly lower than that in the core region. Consequently, its radiological condition is characterized by lower levels of activation and predominantly surface contamination.

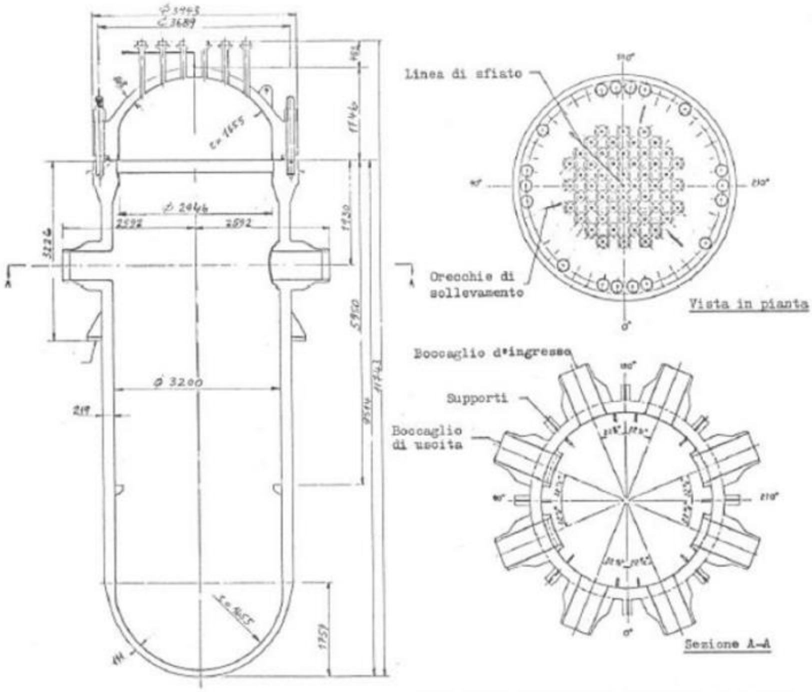


Figure 3.3 Longitudinal section and plan views of the vessel

3.5 Operational characteristics of the Internals

The reactor internals are divided into upper package and lower package, each characterized by specific mechanical and hydraulic functions and by different neutron irradiation conditions [12].

A schematic cross sectional view of the vessel, showing the arrangement of the upper package and lower package, is reported in Figure 3.4.

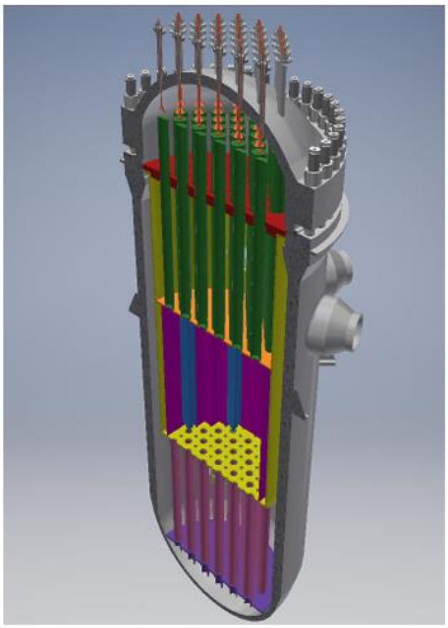


Figure 3.4 Cross sectional view of the vessel with internals

The upper package includes the structures located above the core, such as the upper barrel, the top support plate, the control rod guide tubes, and the associated mechanisms. The lower package includes the core barrel, the baffle assembly, the core support plates, and the lower support structures.

3.5.1 Materials characteristics

The internals are predominantly made of austenitic stainless steels (AISI 304 and ASTM 351 and ASTM 312 alloys) [12], selected for their good mechanical properties, corrosion resistance, and compatibility with the neutron environment. However, these materials are known to be susceptible to neutron activation phenomena, leading to the production of radionuclides such as ^{60}Co, ^{58}Co and ^{54}Mn.

The chemical composition of stainless steels, particularly the presence of cobalt as an impurity, plays a key role in determining the final radiological inventory of the internals.

3.5.2 Irradiation and activation conditions

The internals of the lower package, in particular the baffle assembly and the structures located in the immediate vicinity of the core, have been subjected to the highest neutron fluence during operation. This results in high levels of activation and in a spatial distribution of activity strongly dependent on the position relative to the core.

In contrast, the internals of the upper package are characterized by lower activation levels and a greater contribution of surface contamination, due to contact with the primary coolant and to the deposition of activated corrosion products.

3.6 Implications for Radiological Characterization Strategy

With reference to the status of characterization activities, the vessel head has already been sampled, and the corresponding radiochemical analyses are currently available. The results obtained constitute a consolidated dataset that will be analyzed and discussed in Chapter 6, devoted to the interpretation of the experimental results and their implications for decommissioning activities.

As regards the upper internals, sampling activities on the upper package are currently in progress. The sampling is carried out according to the program defined in *Phase 1* [12] and is aimed at covering, in a representative manner, the main structures of the upper package, considering the different neutron irradiation and surface contamination conditions.

The results derived from the radiochemical analyses of the vessel head and, subsequently, of the upper package will constitute a fundamental element for reconstructing the spatial distribution of activity, for the radiological classification of materials, and for the design of the subsequent dismantling phases, in accordance with the ALARA optimization principle.

Within the context of Phase 1 activities, it is also noted that the author took part, as a technical observer, in several preliminary operations preparatory to characterization. In particular, the reactor cavity flooding operation and the subsequent removal of the upper package were observed. The dedicated sampling system was also examined, including the machinery and the related interchangeable tools and interfaces, designed for sampling both from the upper package structures and from the internal surfaces of the vessel.

This activity allowed the acquisition of direct knowledge of the operational procedures, the configurations of the sampling system, and the installation, assembly, and operating procedures of the equipment, which will be employed in the ongoing operational phases and in subsequent sampling campaigns. Familiarization with machinery and the associated operational sequences represents a relevant element for understanding the technical, radiological, and safety constraints that characterize sampling activities in a controlled environment. The sampling system will be described in Chapter 4.

In Figure 3.5, Figure 3.6 and Figure 3.7, the sampling locations on the vessel head and those planned for subsequent activities on the upper package, barrel, and internal surfaces of the vessel are shown in detail.

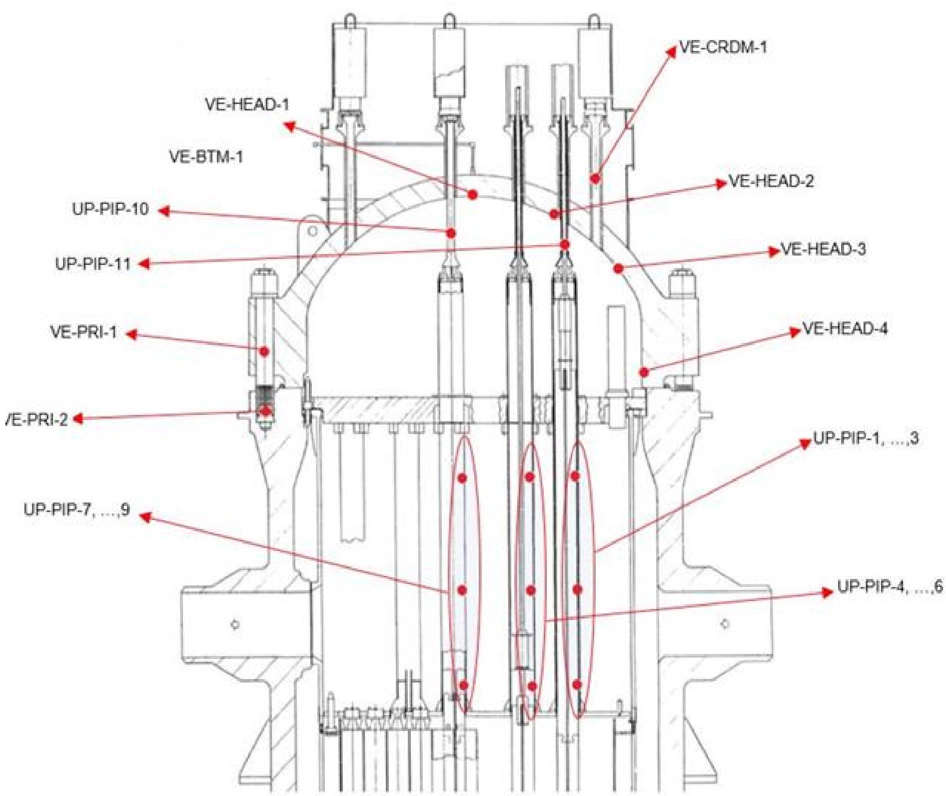


Figure 3.5 Sampling locations on the vessel head and upper package

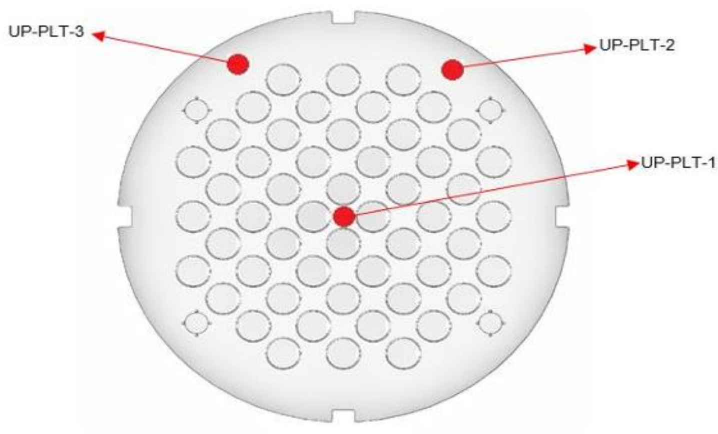


Figure 3.6 planned sampling areas on the upper plate

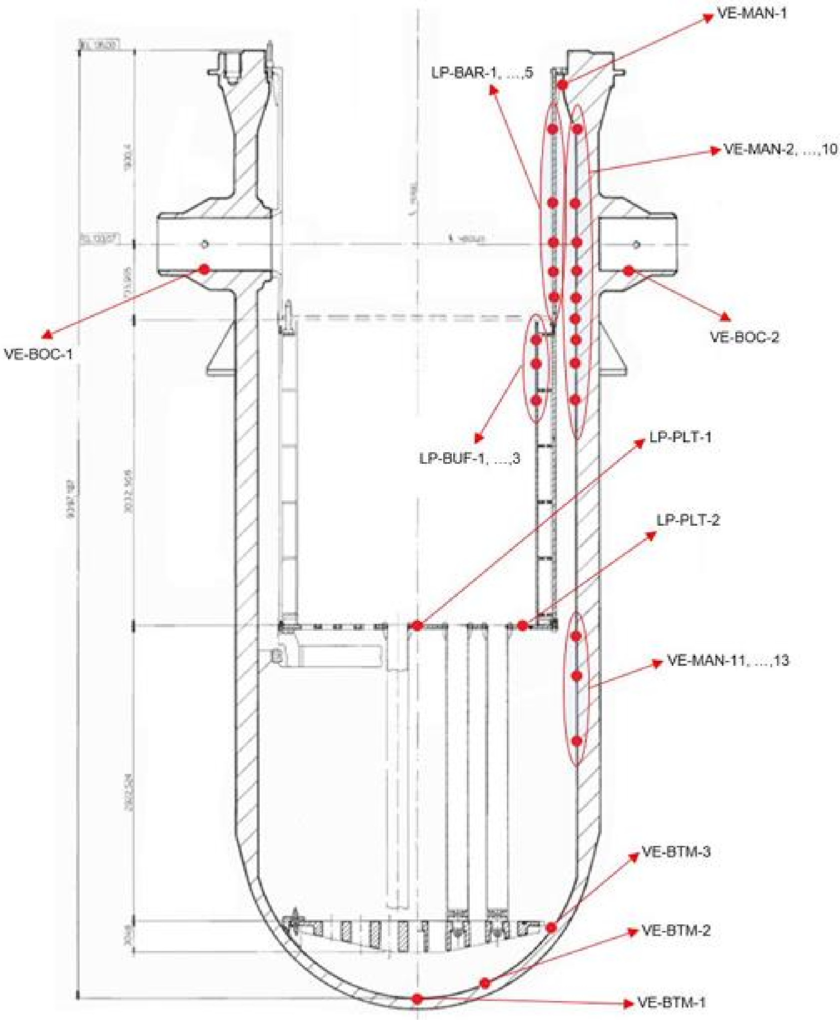


Figure 3.7 Planned sampling areas on the vessel and barrel

3.7 Implications for Subsequent Phases

The main geometrical, functional, and material characteristics of the vessel and internals have been analyzed, highlighting how these aspects significantly influence the planning of sampling activities and the subsequent radiological classification of materials.

In this context, Phase 1 of the decommissioning project plays a central role, as it is aimed at collecting the experimental data necessary for the design of the subsequent phases, in which the dismantling of the most highly activated components will be carried out.

This Chapter thus provides the technical framework required for the analysis of experimental results, which will be discussed in the following section, and for the planning of the following phases of vessel and internals dismantling.

Figure 3.8 shows the cross-sectional view of the Trino NPP reactor building under the post-operam conditions following Phase 1. The configuration of the components, in the primary containment, resulting from the decommissioning activities can be observed.

The RPV head (shown in cyan) has been removed and safely placed at the base floor level (at an elevation of 152.68 m). Through the operation of the polar crane, the upper package (shown in green) have been extracted from the vessel and positioned onto their dedicated storage rack. The remaining components within the reactor cavity consist of the RPV shell housing its lower internals, specifically the core shroud and the lower core plate (shown in violet). In order to ensure adequate biological shielding and optimize radiological protection, the entire cavity is kept completely flooded; the maximum water level is indicated by the solid blue line at an elevation of 144.68 m.

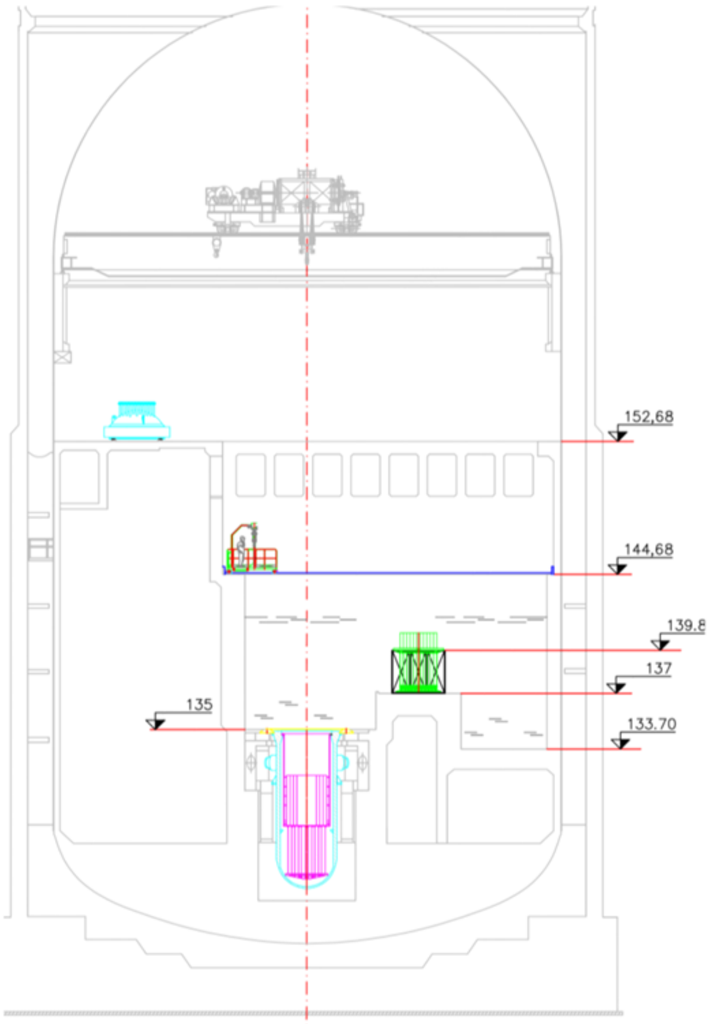


Figure 3.8 Reactor Building Section Post Operam Condition

4. Instrumentation and Radiological Measurement Methodologies

The instrumentation employed at the Trino site for radiological measurements is described in a structured manner, with reference to the operating principles and the corresponding fields of application. The instruments are presented according to a consistent classification, aimed at highlighting their role in the different phases of radiological characterization and control.

4.1 Classification of radiological instrumentation

In this section, radiological instrumentation is organized into five functional categories: instrumentation for direct radiometric measurements, gamma spectrometry systems, instrumentation for radiological air sampling, low-background systems for smear measurements, and personal dosimetry systems. In the following subsections, these categories are described in detail, with reference to their operating principles and fields of application.

4.1.1 Instrumentation for direct radiometric measurements

Radiometric measurement instrumentation includes the set of devices used for the direct detection of ionizing radiation and for the evaluation of dose levels and contamination in the environment. These instruments play a fundamental role in operational radiological monitoring activities, enabling preliminary area characterization, identification of potential contamination sources, and verification of safety conditions during decommissioning operations.

From a physical standpoint, most of these instruments are based on gas filled detectors, scintillation detectors, or solid state detectors, appropriately calibrated in terms of operational radiation protection quantities such as ambient or directional dose equivalent.

Radiometers based on Geiger Muller counters represent one of the most widely used technologies for the detection of ionizing radiation in operational contexts. They belong to the category of *gas filled detectors*, in which incident radiation produces ion pairs that, under the action of an electric field, generate a measurable electrical signal (Figure 4.1). The Geiger Muller counter consists of a sealed tube containing a low pressure gas, enclosed between a cylindrical cathode and a central anode, in which the incident radiation induces ionization processes that give rise to the electrical signal.

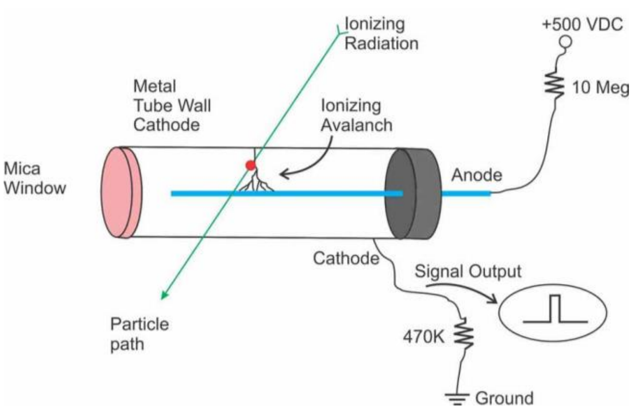


Figure 4.1 Geiger Muller Tube

Figure 4.2 illustrates the different operating regions of a gaseous ionization detector as a function of the applied voltage, showing how the collected charge increases from the initial recombination and ionization chambers up to the proportional and Geiger-Müller plateaus. In particular, it highlights how the distinct curves for highly ionizing α particles, β particles, and γ rays progressively converge in the Geiger region, where high gas amplification renders the output pulse amplitude entirely independent of the primary ionization energy.

The operation of the Geiger Muller counter takes place in the so called Geiger region, as shown in Figure 4.2, characterized by a high signal amplification due to secondary ionization phenomena and discharge propagation throughout the entire sensitive volume of the detector [13]. Under these conditions, each event produces a pulse with substantially constant amplitude, independent of the energy of the incident radiation.

The tube is generally filled with noble gases, such as argon or helium, to which small amounts of quenching gases are added, to prevent continuous discharge phenomena and multiple pulses [13].

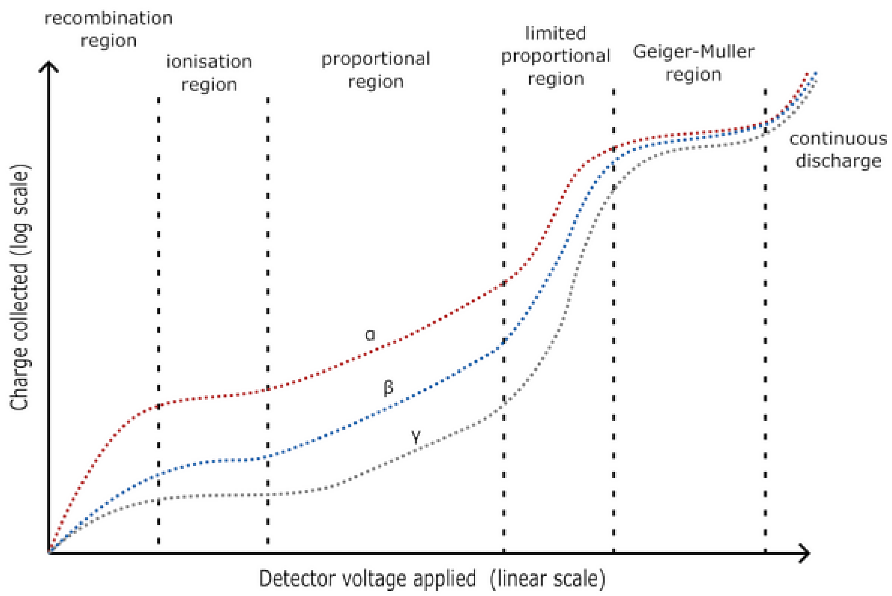


Figure 4.2 Operating regions of gaseous ionization detectors

- **Instruments for dose rate measurement**

Instruments for dose rate measurement are used to determine the intensity of the radiation field in terms of dose per unit time, generally expressed in μSv/h up to maximum values of a few Sieverts. They are essential for the evaluation of exposure conditions and for the classification of areas from a radiation protection perspective.

These instruments can be based on different detection principles. Ionization chambers operate in the ionization region of gas filled detectors, also referred to as the saturation region of charge collection, as illustrated in Figure 4.2, in which all the charge produced by the incident radiation is collected without amplification [13]. The measured current is therefore proportional to the number of ion pairs generated and consequently to the energy deposited in the sensitive volume, making these instruments particularly suitable for accurate dose measurements.

Portable instruments for gamma irradiation measurements are commonly used, such as the Teletector or systems based on scintillators, employed for dose and dose rate measurements directly in the field [15]. These instruments are characterized by high sensitivity, fast response, and robustness, making them particularly suitable for field monitoring applications. However, they present some limitations, including the inability to provide energy information and the presence of relatively high dead times. An example of a portable Geiger Muller counter based radiometer is shown in Figure 4.3.

(a) (b)

Figure 4.3 Radiameter 6150AD6: (a) probe (b) telescopic probe.

- **Instruments for surface contamination measurement**

Instruments for surface contamination measurement are designed to detect the presence of radioactive material deposited on surfaces. They are generally composed of portable contamination monitors equipped with probes sensitive to alpha and beta particles.

Figure 4.4 LB 124 Scint contamination monitor (Berthold)

From a technological perspective, these instruments may employ Geiger Muller counters, scintillation detectors, or gas flow proportional counters, depending on the type of radiation to be detected [15]. The probes are often designed with a large sensitive area to maximize detection efficiency, especially for alpha and beta particles, which have a limited penetration capability.

The response of detectors for alpha and beta particles can be described by characteristic curves with distinct plateaus, as shown in Figure 4.5, which highlight the different operating conditions of the detector as a function of anode voltage [14].

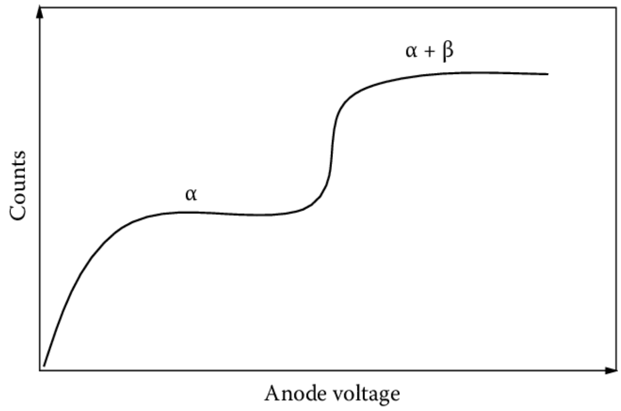


Figure 4.5 Characteristic response curve, α and β contributions.

The measurement is performed near the surface, and the signal is generally expressed in counts per second, with the possibility of conversion into surface activity units through appropriate calibration factors. At the Trino Laboratory, the LB124 SCINT contamination monitor is used (Figure 4.4), which is based on the scintillation principle and employs a thin layer of scintillating material, typically ZnS(Ag), positioned immediately below an entrance window. The incident radiation, in particular alpha and beta particles, interacts with the material producing light pulses, which are subsequently converted into electrical signals for surface contamination measurement.

4.1.2 Gamma spectrometry

Gamma spectrometry represents one of the most important techniques for the qualitative and quantitative characterization of radioactive materials, allowing the identification of radionuclides and the determination of their activity through the analysis of emitted gamma radiation.

This technique is based on the measurement of the energy distribution of photons emitted during radioactive decay processes. Each radionuclide is characterized by a set of specific gamma energies, which constitute a distinctive signature useful for its identification. When gamma radiation interacts with the detector, the energy is deposited through different interaction mechanisms Figure 2.3 [13]. The generated electrical signals are subsequently processed and sorted according to their amplitude, producing a spectrum in which the peaks correspond to specific gamma energies, shown in Figure 4.6.

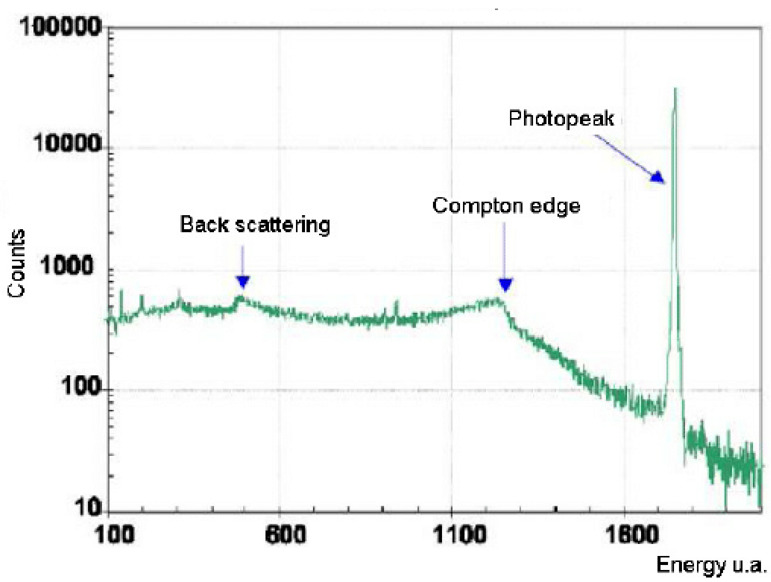


Figure 4.6 Gamma radiation spectrum

Although gamma spectra exhibit common general features such as the photopeak, the Compton continuum, and the backscattering peak, their shape strongly depends on the radionuclide under consideration. In real gamma spectra, the distribution of events is significantly influenced by the specific radionuclide, as each isotope is characterized by well defined emission energies. Radionuclides commonly used in gamma spectrometry may exhibit a single dominant photopeak, as in the case of cesium-137, or multiple distinct peaks, as for cobalt-60. In the case of low energy sources, such as americium-241, the photopeak is instead located in the lower region of the spectrum.

Table 3.1 Photopeak energies (keV) of selected reference radionuclides.

Radionuclide	Photo peak [Kev]
Americium-241	59.5
Cesium-137	662
Cobalt-60	1173 and 1332

Reference radionuclides such as cesium 137, cobalt 60, and americium 241 are widely used in gamma spectrometry because they emit photons at well defined and stable energies, distributed over different energy ranges. The simplicity and recognizability of their spectra, together with their availability as calibrated sources, make them particularly suitable for calibration and performance verification of the detector over the entire energy range of interest.

High purity germanium detectors HPGe are semiconductor devices widely used in high resolution gamma spectrometry. Their operating principle is based on the interaction of gamma radiation with the germanium crystal, within which the deposited energy generates electron hole pairs Figure 4.7. The number of charge carriers produced is proportional to the energy of the incident photon, allowing an extremely accurate energy measurement. Due to the small band gap of germanium, these detectors must be maintained at cryogenic temperatures, generally by means of liquid nitrogen cooling, to reduce electronic noise.

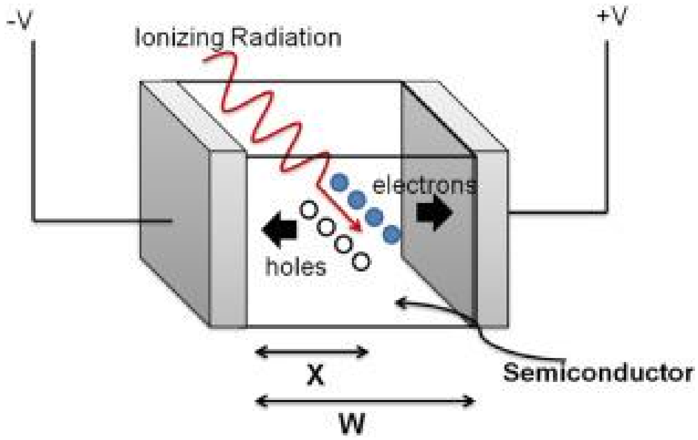


Figure 4.7 Schematic representation of electron-hole pair generation caused by ionization.

The energy resolution of a detector is generally expressed in terms of Full Width at Half Maximum (FWHM), defined as the width of the photopeak measured at half of its maximum height Figure 4.8. A lower FWHM value corresponds to a better capability of the detector to distinguish between closely spaced energy peaks.

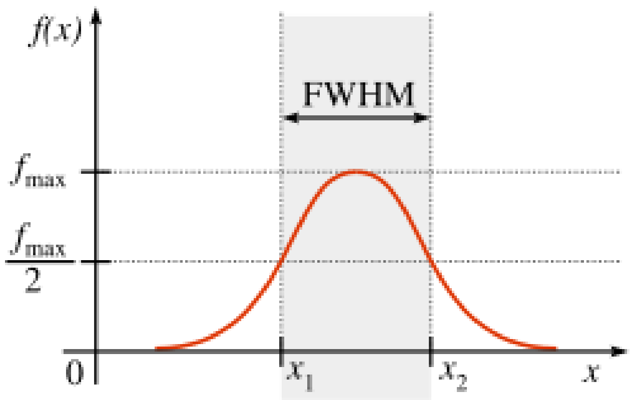


Figure 4.8 Graphical representation of the FWHM of a peak.

The FWHM is generally expressed in keV and can be evaluated at specific reference energies. In the case of HPGe detectors, this parameter is significantly lower than that of scintillation detectors, making them particularly suitable for high resolution gamma spectrometry applications Figure 4.9 [14].

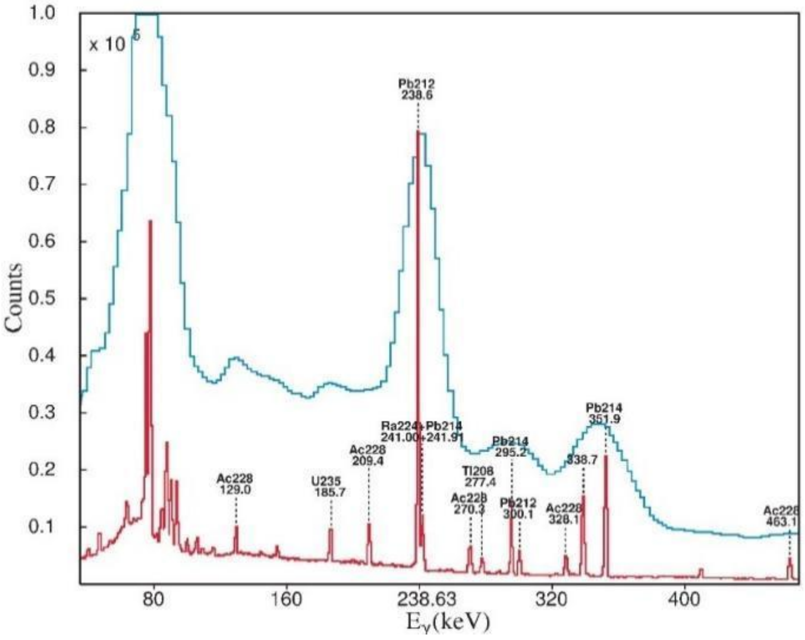


Figure 4.9 Comparison of energy resolution: HPGe (red) vs scintillator (blue)

ACQUISITION CHAIN

A complete gamma spectrometry system consists of a measurement chain composed of several interconnected elements, each with a specific function in the signal detection and analysis process. The detector, typically a High-Purity Germanium (HPGe) crystal, represents the first stage of the chain and is responsible for converting the energy of the incident gamma radiation into an electrical signal. This signal, initially very weak, is collected and preprocessed by the preamplifier, which preserves its informative characteristics.

Subsequently, the signal is sent to the amplifier, which optimizes its amplitude and shape, making it suitable for the analysis stage. The multichannel analyzer (MCA) then performs signal digitization and classification according to amplitude, assigning each pulse to a specific energy channel and thus enabling the construction of the gamma spectrum.

Finally, the acquisition software manages the entire measurement process allowing spectrum visualization, identification of characteristic peaks, and calculation of the activity of radionuclides present in the sample (Figure 4.10) [15].

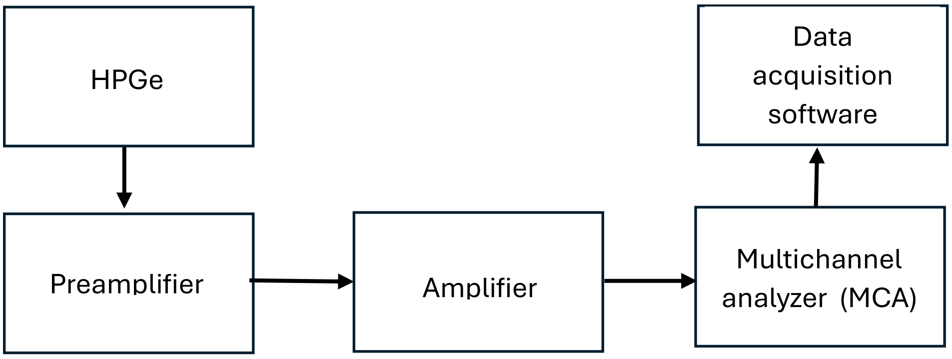


Figure 4.10 Gamma spectroscopy acquisition chain

At the Trino laboratory, integrated systems developed by Canberra are employed, combining both hardware and acquisition and analysis software within a single platform, as in the case of the Genie2000 system. These systems enable automated measurement management, ensuring high reliability in data processing and interpretation of the acquired spectra.

CALIBRATION AND EFFICIENCY

To obtain reliable quantitative results through gamma spectrometry, it is necessary to perform a proper calibration of the measurement system. In particular, the fundamental operations are energy calibration and efficiency calibration, both preliminary to the correct interpretation of the acquired spectrum and to the subsequent determination of the activity of radionuclides present in the sample [13].

Energy calibration aims to establish the relationship between the channel number of the spectrum and the corresponding gamma photon energy (Figure 4.11). It is performed using calibration sources with gamma emissions at known energies, distributed over the entire energy range of interest. The centroids of the photopeaks are identified and a linear calibration relationship is established, allowing each channel to be associated with the corresponding energy. This enables the identification of the radionuclides present by comparing the measured peak energies with reference values.

Quantitative determination instead requires ***Efficiency calibration***, namely the definition of the relationship between detection efficiency and the energy of the emitted photons. Efficiency can be expressed as the ratio between the number of recorded pulses and the number of photons emitted by the source within the same time interval and is influenced both by the characteristics of the detector and by the measurement geometry. The resulting efficiency curve, shown in Figure 4.12, represents the fundamental tool for converting the counts measured in the photopeaks into activity values [14].

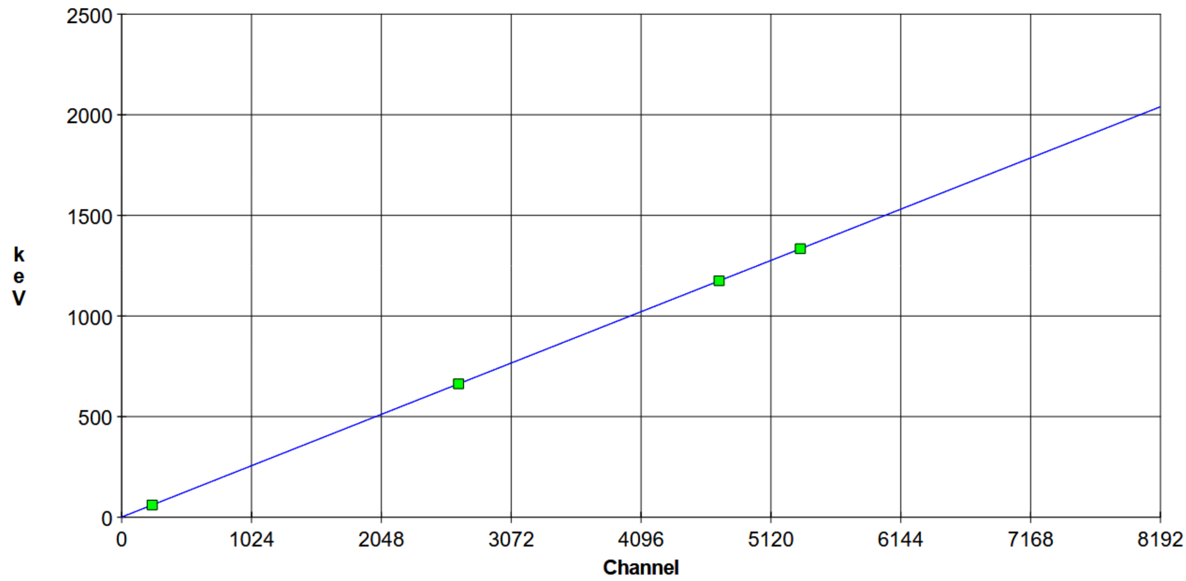


Figure 4.11 Energy calibration curve based on known source gamma emissions energies

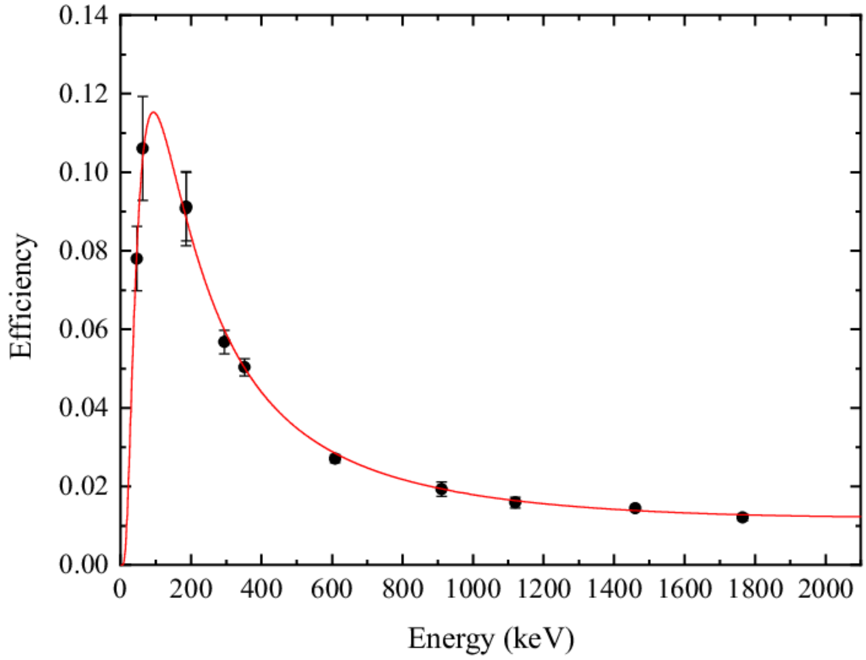


Figure 4.12 Typical detector efficiency as a function of γ-ray energy

In the calculation of activity, it is also necessary to correctly determine the net area of the photopeak, which represents the number of counts associated with the specific gamma energy emitted by the radionuclide. This quantity is obtained by integrating the peak and subtracting the contribution of the underlying background, generally due to the Compton continuum, as illustrated in Figure 4.13.

Figure 4.13 demonstrates the traditional linear background subtraction method, where the photopeak is bound by a lower channel (L) and a higher channel (H), with corresponding boundary counts n_L and n_H. The background continuum beneath the peak is approximated by a straight line connecting these two points over the channel interval ($H - L$), allowing for the mathematical isolation of the true net counts (N) from the gross integrated area.

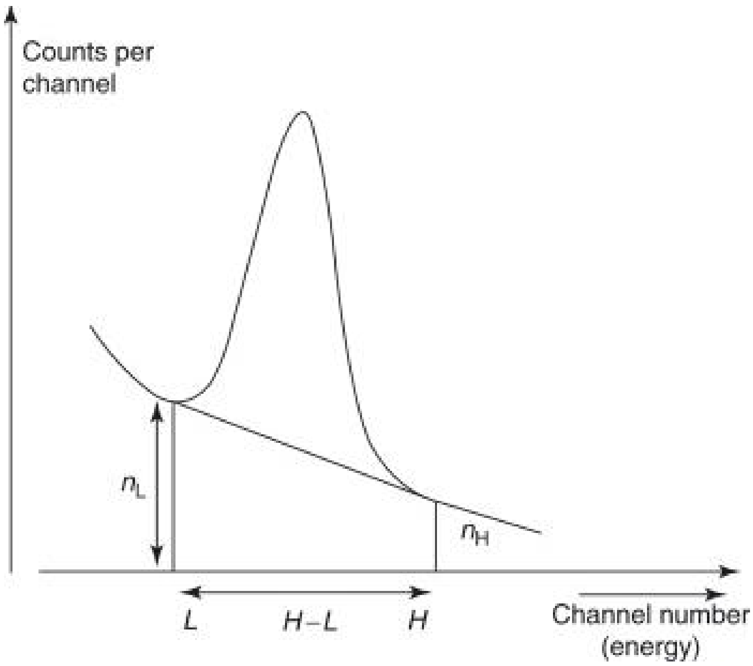


Figure 4.13 Photopeak area with background subtraction

The net peak area represents the fundamental parameter for activity determination, which can be expressed in simplified form as:

$$A\ [\mathrm{Bq}] = \frac{N}{\varepsilon \cdot I_{\gamma} \cdot t}$$

Where:

N = net counts; ε = detector efficiency; I_{γ} = gamma emission probability; t = measurement time

The efficiency curve is not an intrinsic property of the detector alone but depends significantly on the experimental conditions. Several correction factors must be considered, including sample geometry, self-absorption, coincidence summing, and decay during measurement. Self-absorption depends on the density and composition of the sample, while coincidence summing becomes more significant in the presence of high efficiencies and favorable geometries [15].

The quality of the quantitative result in gamma spectrometry depends not only on detector resolution and correct peak identification, but also on the robustness of calibration procedures and the proper application of experimental corrections, which constitute the fundamental basis for reliable activity determination.

Among the instrumentations described, gamma spectrometry represents the technique analyzed in greatest detail, as it constitutes the main tool used for the radiological characterization of the samples considered in this Thesis.

It should be noted that spectrometric analyses on samples taken from the vessel head, as well as those that will be subsequently analyzed, were not performed at the Trino site, but at the laboratories of Nucleco in Rome. In the following chapters, examples of spectrometric analyses carried out in the laboratory will also be presented, to illustrate in an applied manner the methodologies described.

4.1.3 Instrumentation for radiological air sampling

Radiological air sampling represents a fundamental phase in environmental characterization and monitoring activities, as it allows the assessment of the presence of airborne radioactive contamination in the form of particulate matter or aerosol [13]. Such measurements are particularly relevant in decommissioning activities, where dismantling operations may generate dispersion of radionuclides in the working environment.

The RADeCO type air sampler Figure 4.14 is a portable device designed for the collection of known volumes of air, to determine the activity concentration of airborne radioactive particulate matter. The operating principle is based on the forced suction of air through a filter, on which particles containing radionuclides are retained [15].

(a)

(b)

Figure 4.14 (a) RADeCO air sampling systems room air monitoring, (b) operator breathing simulation during operations

The system generally consists of a suction pump, a filter holder, and a flow measurement system, which allows accurate determination of the total sampled air volume. The filters used are typically made of fibrous materials such as glass fiber or cellulose and exhibit high collection efficiency for micrometric particles.

During operation, the sampler draws air at a controlled flow rate for a defined time interval. At the end of the sampling process, the filter is removed and subjected to subsequent radiometric or spectrometric analysis, generally by means of scintillation detectors or high resolution gamma spectrometry [14].

4.1.4 Low Background Instrumentation for Smear Measurements

For the determination of removable contamination collected by smear tests or filter, Berthold instrumentation dedicated to low background alpha and beta counting has been employed [15]. In this configuration, the samples are placed in dedicated holders and inserted into a shielded measurement chamber designed to reduce the contribution of environmental radiation.

Background reduction is achieved through shielding made of materials with high atomic number and anticoincidence systems, which allow the influence of cosmic radiation and natural background to be minimized. This enables high levels of sensitivity, making the instrument particularly suitable for the measurement of very low activities [13].

From the standpoint of the detection principle, the system uses *gas proportional counters operating in the proportional region*, in which the ionization produced by the incident radiation is amplified in a controlled manner while maintaining proportionality between the signal and the deposited energy [13]. This characteristic allows efficient measurement of alpha and beta emissions and their discrimination under appropriate conditions. The instrument available in the laboratory is the Berthold LB790, shown in Figure 4.15.

Figure 4.15 Low-Level detector LB790

4.1.5 Personal dosimetry

Personal dosimetry allows the evaluation of individual exposure of workers to ionizing radiation through the measurement of the personal dose equivalent, $H_p(d)$. The main devices used are classified into passive and active dosimeters, based on different detection mechanisms, including thermoluminescence, ionization, and radiation interaction with sensitive materials [16].

- **Passive dosimeters, Direct Ion Storage (DIS)**

Passive dosimeters, such as the DIS 1 (Figure 4.16), are based on the accumulation of energy in a sensitive material following interaction with ionizing radiation. In Thermoluminescent Dosimeters (TLD), radiation induces the migration of electrons to metastable energy levels traps, as shown in Figure 4.17. Subsequent heating of the material causes the release of energy in the form of light, the intensity of which is proportional to the absorbed dose [16] [17].

Figure 4.16 Passive dosimeter (DIS-1 type)

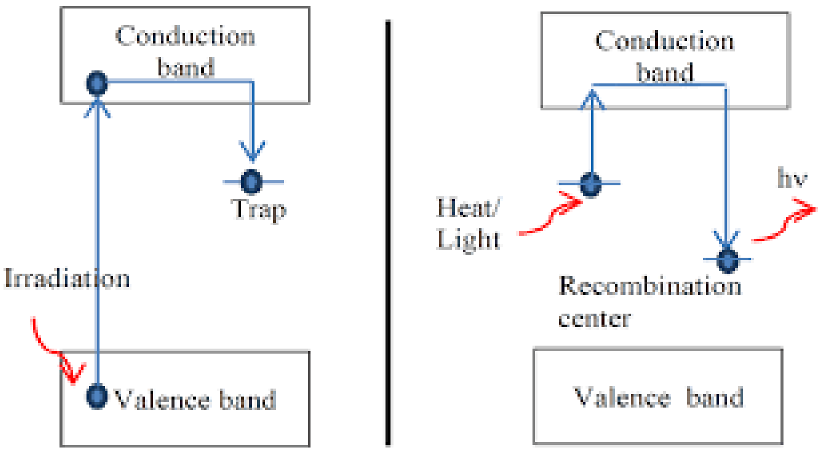


Figure 4.17 Schematic principle of a passive dosimeter

These devices do not provide an immediate readout but allow the determination of the integrated dose over time through laboratory analysis. The measurement is expressed in terms of personal dose equivalent $H_p(10)$, representative of the dose at a tissue depth of 10 mm.

- **Active dosimeters**

Active dosimeters, such as electronic DMC 3000 devices (Figure 4.18), are based on semiconductor detectors, typically silicon p-n junctions. Incident radiation generates electron hole pairs within the material, as previously described in Figure 4.7, producing a charge distribution proportional to the deposited energy and therefore to the absorbed dose [16].

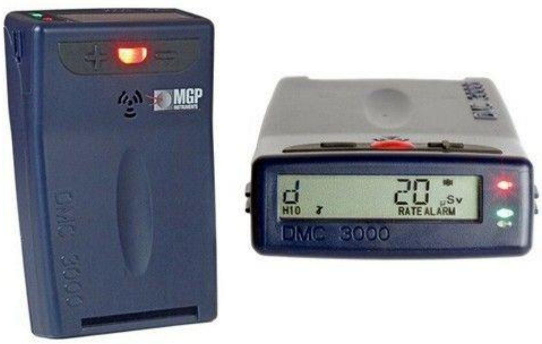


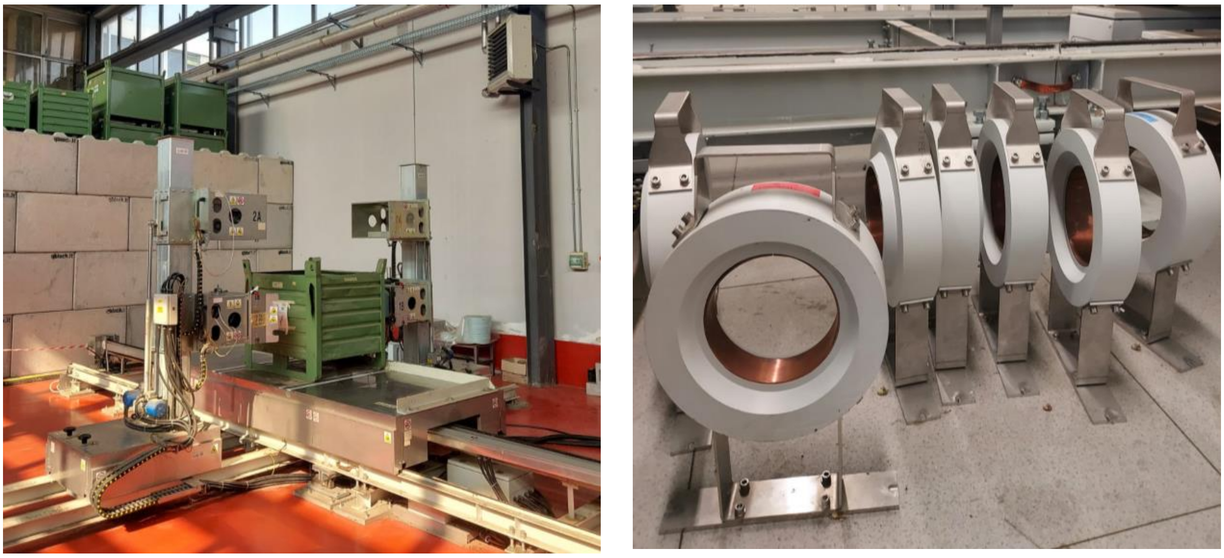
Figure 4.18 DMC 3000 Active electronic personal dosimeter

The electrical signal is processed by an internal electronic system that allows real time display of the dose rate. These devices are also equipped with alarm systems for exceeding preset thresholds, making them particularly useful for operational monitoring in environments with variable radiation fields.

4.2 Instrumentation and Infrastructure of the Trino site

This section describes the main instrumentation and monitoring systems implemented within the Trino site infrastructure, with reference to In Situ Object Counting System (ISOCS), Whole Body Counter (WBC) systems, portal monitors, a sampling device for the radiological characterization of the internals and a dedicated underwater probe for in vessel gamma spectrometry. These systems, although based on measurement principles already described in the previous sections, are integrated into the plant layout and are used in specific operational contexts related to radiological control, waste management, and personnel protection.

•I*SOCS* systems are used for the radiological characterization of materials and radioactive waste directly within the controlled area. These systems are based on high purity germanium detectors and allow non destructive gamma spectrometric analysis of objects without the need for sample preparation. The use of ISOCS enables the determination of radionuclide composition and activity of the waste, supporting its classification and management. Such characterization is essential to verify compliance with clearance levels established by regulations, allowing materials to be released and disposed of in conventional landfills when the activity concentrations fall below the prescribed limits. Additionally, collimators can be employed to restrict and direct the radiation field toward the detector, enabling more localized and geometry controlled measurements Figure 4.19.



(a) *(b)*

Figure 4.19 (a) ISOCS system for radiological characterization of waste containers, (b) Collimators used in ISOCS

•*WBC* systems are employed for the assessment of internal contamination of workers. These systems are typically used at the beginning and at the end of work activities, as well as periodically, to ensure the absence of radionuclide intake. The measurement is performed using high sensitivity detectors, generally based on high purity germanium, capable of detecting gamma emissions originating from radionuclides present within the human body. The results provide an estimate of the internal dose and represent a fundamental tool for the implementation of radiation protection programs Figure 4.20.

(a)

(b)

Figure 4.20 (a) WBC system for internal contamination, (b) WBC internal view

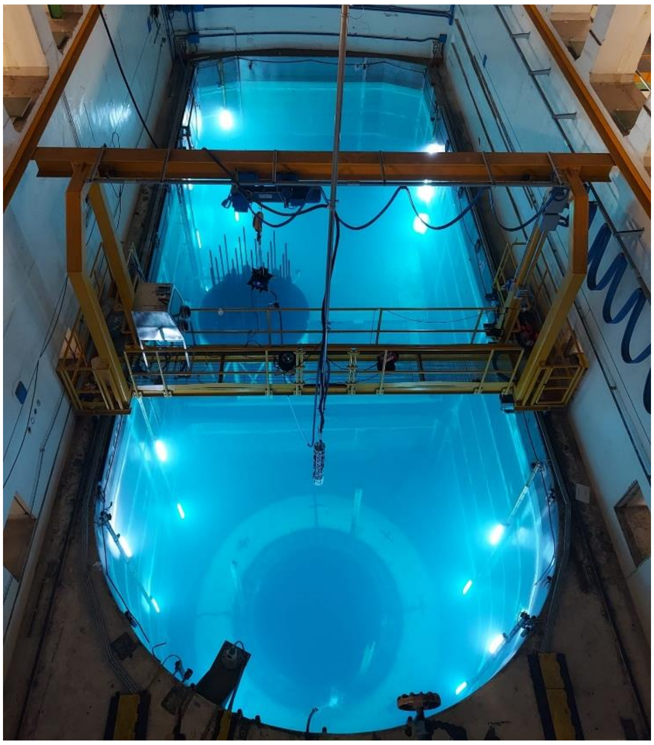
•*Portal monitors* are installed at the exits of controlled areas and are used to verify the absence of radioactive contamination on personnel leaving radiologically controlled zones. These systems operate according to principles like those of surface contamination monitors, employing detectors sensitive to alpha and beta radiation and, in some configurations, also to gamma radiation. The measurement is performed automatically during transit through the portal, allowing rapid identification of potential contamination and preventing its spread outside controlled areas Figure 4.21.

Figure 4.21 Portal monitor for personnel

• A further system is represented by a dedicated *sampling device* is foreseen for the execution of radiological sampling activities on the Reactor Pressure Vessel and its internals. This equipment is designed to operate in submerged conditions and allows the controlled removal of material from selected locations, supporting subsequent laboratory analyses.

The sampling system is intended to be employed during the sampling campaign of the upper package and, through the use of specific inserts and configurations, also for the internal surfaces of the vessel and potentially for components in proximity to the core region. Its design enables flexibility in the selection of sampling points, ensuring compatibility with different geometrical and radiological conditions.

From an operational point of view, the device is positioned and handled by means of the polar crane, allowing precise placement of the equipment on the target surface while maintaining a safe distance between operators and the radiation source. This configuration ensures both accuracy in sampling and compliance with radiological protection requirements.



(a)

(b)

Figure 4.22 (a) Sampling device installed on the polar crane during preparation for in vessel operations,
. (b) Detailed view of the sampling device

As shown in Figure 4.22, the integration of the sampling system with the crane handling infrastructure represents a key aspect for the execution of remote operations within the reactor vessel.

The sampling device therefore represents a fundamental tool within the overall characterization strategy, complementing in situ measurement systems and enabling a detailed assessment of both activation and contamination phenomena in structural materials.

•A dedicated *underwater probe* is used for vessel gamma spectrometry measurements. This probe allows in situ characterization of radiological conditions inside the reactor vessel, enabling measurements without the need for sample removal. These activities are intended as preliminary investigations and do not preclude subsequent sampling operations in later phases of the decommissioning process.

The probe was acquired and made available on site during the period in which the author was present at the Trino facility. It consists of a sealed stainless steel housing with IP68 protection, containing a scintillation detector based on LaBr3 Ce crystal for gamma spectrometry and a Geiger Muller detector for dose rate measurements. This configuration allows simultaneous acquisition of spectral information and dose rate over a wide dynamic range, from environmental levels up to high radiation fields.

The system operates in an energy range of approximately 50 keV to 2 MeV, with high energy resolution, and includes a Geiger Muller detector extending the measurable dose rate up to about 1 Sv per hour. The probe is connected via fiber optic cable to a portable control unit, enabling real time visualization and data acquisition. It is designed for operation in submerged conditions within water filled structures such as the reactor vessel.

The underwater probe represents a complementary tool to the sampling system previously described, enabling localized and rapid radiological assessment within the vessel environment Figure 4.23 [18].

Figure 4.23 Local control unit connected to the stainless steel probe housing

Overall, these systems constitute an integral part of the radiological protection and characterization infrastructure of the Trino site, ensuring control of contamination pathways, safe management of radioactive materials, and continuous monitoring of personnel exposure.

•In addition to the instrumentation and infrastructure previously described, the Trino site is equipped with dedicated facilities for the *temporary storage* of radioactive waste generated during decommissioning activities. At present, three temporary storage facilities are foreseen within the site: two already in operation, used for the interim storage of conditioned radioactive waste, and a third currently under construction, aimed at increasing the overall storage capacity and improving waste management throughout the different phases of decommissioning. The facility currently in operation, is shown in Figure 4.24, which illustrates its configuration and its integration within the site. The test tank is designed to accommodate 380 L overpacks used for the containment and storage of conditioned radioactive waste.

These structures enable the safe storage of radioactive materials prior to their transfer to the National Repository, ensuring temporary confinement and control of radiological conditions in compliance with the applicable regulatory framework [11]. From an operational perspective, they provide increased flexibility in the planning of decommissioning activities, allowing the decoupling of waste generation from final disposal and supporting appropriate procedures for waste classification, segregation, and conditioning. Temporary storage facilities represent a key infrastructural element for the safe and efficient management of radioactive waste throughout the decommissioning process.

Figure 4.24 Temporary storage facility (Test Tank) at the Trino site

4.3 Radiological monitoring systems of the plant

Within the Trino site, several radiological monitoring systems are installed, distributed across the main plant areas and along release pathways, with the aim of ensuring continuous control of radiological conditions and compliance with environmental safety limits. In this context, monitoring systems are commonly identified by specific codes, such as Radiation in Air (RIA) for airborne radioactivity monitoring and Radiation in Liquid (RIC) for liquid effluent monitoring. These systems include perimeter monitors, monitoring systems associated with the spent fuel pool and the temporary storage facility, as well as monitoring of gaseous discharges through the stack and liquid effluents.

Perimeter monitors RIA117 and RIA118 and the systems installed in correspondence with the spent fuel pool and the temporary storage facility are based on ionization chambers. In these devices, ionizing radiation interacts with the gas contained in the detector, producing ion pairs; the charge collected at the electrodes is proportional to the deposited energy and therefore to the dose rate. This configuration allows stable and reliable measurements over time, making these instruments particularly suitable for continuous monitoring of the radiological background and for the detection of any anomalous variations.

Regarding atmospheric particulate monitoring, for example at locations identified as RIA101 and RIA102, air sampling systems equipped with suction pumps are employed. The air is conveyed through the system and first passes through a silicon semiconductor detector, which provides an initial detection of airborne radioactivity. Subsequently, the particulate matter is collected on a filter, which is then analyzed in the laboratory using low background instrumentation, to determine with greater accuracy, the concentration and nature of the radionuclides present.

Monitoring of liquid effluents, as in the case of system RIC 111, is performed using scintillation detectors, typically based on sodium iodide crystals. In this case, the fluid flows through a monitored section, and the emitted radiation interacts with the crystal, producing light pulses proportional to the deposited energy. The system enables continuous control of radioactivity levels in liquids, both under normal operating conditions and during tank discharge operations. During these phases, real time monitoring ensures that dose or activity levels remain below predefined alarm thresholds, thereby guaranteeing compliance with radiological safety requirements.

Monitoring of gaseous discharges through the stack is also carried out using air sampling and airborne particulate detection systems, like those described for environmental air monitoring, with the objective of controlling atmospheric releases and ensuring compliance with regulatory limits. The monitored air includes all flows originating from the primary containment.

Overall, these systems constitute an integrated radiological monitoring network, essential for the control of exposure pathways, the prevention of contamination spread, and the protection of both the environment and personnel during all operational and decommissioning phases of the plant.

4.4 Sampling procedures and sample preparation

Sampling and sample preparation represent fundamental steps in radiological characterization activities, as they directly influence the accuracy and reliability of subsequent measurements. Depending on the type of material and the measurement objective, samples can be analyzed by gamma spectrometry or by low background counting techniques.

With reference to gamma spectrometry, samples may consist of solid or liquid materials collected from different plant areas. Solid samples are typically obtained through mechanical operations such as coring or cutting, to ensure representativeness of the material under investigation. Liquid samples are instead collected in appropriate containers, ensuring homogeneity and avoiding contamination during handling.

Once collected, samples are transferred into suitable measurement geometries. Standard beakers can be used for routine analyses, while Marinelli beakers are commonly adopted when higher detection efficiency is required. The Marinelli geometry allows the sample to surround the detector, significantly increasing counting efficiency, particularly for low activity measurements. The choice of container and geometry depends on the sample matrix, volume, and required sensitivity.

Before measurement, samples may undergo additional preparation steps, such as homogenization, drying, or sealing, to ensure reproducibility and to minimize uncertainties related to geometry and density. A different procedure is applied for samples related to airborne contamination and removable surface contamination. In the case of air sampling systems, particulate matter is collected on filters, which are subsequently analyzed. Similarly, smear tests are performed to assess removable contamination on surfaces, and the collected material is transferred onto dedicated supports.

These filters and smear samples are typically measured using low background counting systems, which provide high sensitivity for alpha and beta emitters. The use of shielded chambers and controlled measurement conditions allows the detection of very low levels of activity, making these techniques particularly suitable for routine monitoring and contamination assessment.

The adopted sampling and preparation procedures are tailored to the type of sample and the measurement technique, ensuring consistency between the collected material and the analytical method, and enabling reliable radiological characterization results.

4.5 Metrological Control and Instrumentation Quality Assurance

The radiological instrumentation available at the Trino site is fully inventoried and managed through a control system that ensures traceability and maintenance of performance over time. All instruments are subject to periodic calibration programs, scheduled at defined intervals and carried out in collaboration with the manufacturer or qualified laboratories, to ensure the reliability and accuracy of measurements.

Calibration activities are documented through calibration certificates, which attest the compliance of the instrument with recognized metrological standards and report its main performance characteristics. These certificates represent a fundamental element for the validation of measurement results and for their traceability.

In operational practice, each laboratory instrument is also equipped with an identification label, reporting the main calibration information, such as the date of the last calibration, the expiry date, and the reference to the calibration certificate. This system allows the operator to immediately verify prior to use, the validity status of the instrument and the suitability of the measurements to be performed, ensuring direct and continuous visual control of metrological compliance Figure 4.25.

In addition to periodic calibrations, the instrumentation is subject to functional checks and intermediate controls, aimed at identifying any deviations in performance. Calibration and verification activities are integrated within the quality assurance system, which includes codified procedures for the management, use, and maintenance of the instruments.

The adoption of structured calibration, verification, and quality control programs ensures the consistency and reliability of radiological measures, representing an essential requirement for characterization and monitoring activities carried out at the site.

Figure 4.25 Example of instrument calibration label

5. Operational Radiological Monitoring Activities at the NPP

The Health Physics Department plays a central role in decommissioning activities, ensuring the control and management of radiological aspects throughout all operational phases. It is responsible for the assessment of radiological conditions in working environments, the protection of personnel, and the monitoring of environmental matrices, contributing to compliance with regulatory requirements and safety criteria. The activities of the department include radiological characterization of materials, support to dismantling operations, management of radioactive waste, and verification of conditions for material release. In addition, the department ensures continuous monitoring of exposures, both external and internal, and the application of radiation protection principles, with reference to dose optimization.

Within the framework of the present work, the activities carried out in the Health Physics Department were aimed at supporting radiological characterization and control operations, providing a direct contribution to the understanding of the operational methodologies adopted at the Trino site.

5.1 Operational Procedures and Methodological Framework

The activities of the Health Physics Department are carried out according to codified operational procedures, aimed at ensuring systematic control of the radiological conditions of the plant and compliance with regulatory requirements. These procedures, defined within technical documentation and internal site procedures [19], are consistent with international principles of radiation protection and radiological monitoring established by the IAEA [20], and cover the entire range of monitoring, sampling, instrumental control, and effluent management activities, including support to plant operations relevant from a radiological perspective.

A first category of activities concerns the radiological monitoring of areas within the controlled zone. This includes the execution of smear tests for the assessment of removable surface contamination, as well as air sampling through suction systems, aimed at determining the presence of airborne radioactive particulate. The collected samples are subsequently analyzed using low background instrumentation or spectrometric techniques, depending on the type of radionuclides to be investigated. A further operational field is represented by the sampling and analysis of specific matrices, such as filters from stack monitoring systems and samples intended for tritium determination. These activities fall within effluent control and allow the assessment of the radiological impact associated with gaseous emissions from the plant.

Relevance is also attributed to metrological control of instrumentation and quality assurance activities. These include calibration operations, functional checks, and quality assurance procedures on monitoring systems, in compliance with the requirements set by ISIN, to ensure proper functioning of the equipment and reliability of the acquired data.

Within waste management activities, the department adopts procedures for radiological characterization of materials, through in situ gamma spectrometry systems ISOCS, which allow the determination of radionuclide inventory and support material classification for disposal purposes, including verification of compliance with clearance levels established by regulations. An additional area of intervention concerns radiological protection support to plant operations, including, for example, procedures for flooding the reactor cavity and subsequent phases of removal and handling of the upper package towards the dedicated support structure. In such contexts, radiological monitoring and releasement of exposure conditions are essential to ensure worker safety and contamination control.

During field activities within the controlled zone, such as in the case of removal of primary circuit loops, routine radiological monitoring activities are also performed, including smear tests and air sampling, carried out before, during, and after operations, to verify the evolution of radiological conditions and the effectiveness of the adopted protection measures.

Overall, the set of operational procedures adopted by the department ensures integrated control of the radiological aspects of the plant, guaranteeing measurement quality and the safety of decommissioning activities.

5.2 Contribution to Radiological Monitoring and Control

During the period spent in the Health Physics Department at the Trino site, the student actively participated in several operational activities, contributing to the execution of the main radiological monitoring and control procedures described in the previous section.

The activities carried out allowed the acquisition of direct knowledge of the operational practices adopted in nuclear decommissioning, with reference to monitoring techniques, control strategies, and radiological risk management, as well as hands on experience in the use of laboratory instrumentation.

5.2.1 Routine Radiological Monitoring in Controlled Areas

Routine radiological monitoring activities were performed in controlled area zones, including the execution of smear tests for the assessment of surface contamination and air sampling through suction systems (Figure 4.14). These activities enabled the development of familiarity with sampling techniques and with the procedures for sample handling and analysis, subsequently carried out using low background instrumentation (Figure 4.15). In support of the described activities, representative images of the monitored areas are reported in Appendix A.

Within this activity, prior to access to the monitored areas and following the measurements, appropriate signs are placed at the entrance of the rooms, providing information on the radiological status of the area through color coded indicators. These indications allow the operator to immediately assess the conditions of the area and the possible presence of contamination, representing an essential operational tool for the safe management of activities Figure 5.1.

Figure 5.1 Example of radiological area classification levels

5.2.2 Airborne Effluent Monitoring at the Stack

The student took part in the monitoring of airborne effluents through stack monitoring systems, in accordance with the procedures described in Section 5.1. In particular, particulate collection filters were analyzed using low background measurements aimed at determining the activity of the radionuclides present. These activities allowed the development of familiarity with operational procedures for sample handling and analysis within the framework of effluent monitoring.

Activities related to the sampling and handling of systems for tritium determination were also carried out, contributing to the control of emissions for radionuclides not detectable by gamma spectrometry. Finally, the student participated in functional checks of stack monitoring systems and detectors associated with discharge lines, to ensure their proper operation and the reliability of measurements over time.

5.2.3 Calculation and Assessment of Liquid Effluent Discharges

The calculation and assessment of liquid discharges were carried out through the application of operational formulations reported in the reference technical documentation [21]. These evaluations were required in relation to the extraordinary management of water volumes stored in storage tanks, following the flooding of the reactor cavity due to the detection of leaks during plant operations.

The main objective of the activity was the verification of the radiological compliance of the discharges, through the determination of releasable activity and comparison with authorized limits. The formulas used for the calculation of concentrations and discharged activities were applied to the available data, derived from monitoring campaigns and measurements performed on the water volumes to be treated. The formulations adopted for data processing, derived from the reference technical documentation [21], are reported in extended form in Appendix B, to ensure completeness and traceability of the calculation procedure. The obtained results did not show a high level of statistical significance. This is mainly due to the limited number of available data, consisting of approximately 20 discharge events in the analyzed period between 2016 and the present.

Furthermore, the reliability of the results was affected by sample heterogeneity. The water conveyed into storage systems originates from flows coming from different areas of the plant, associated with both routine activities and specific operations, such as the flushing of primary circuit loops. This variability in origin conditions and fluid composition resulted in significant dispersion in the obtained results, limiting their statistical representativeness.

Within the data analysis, correlation factors between key radionuclides and difficulty to measure radionuclides were also determined. By way of example, Tables 5.1 and 5.2 report the results related, respectively, to an activation product ^{63}Ni / ^{60}Co and to a contamination product ^{241}Pu / ^{137}Cs.

These specific pairs were selected because ^{60}Co and ^{137}Cs act as easily measurable gamma-emitting "key nuclides" (or indicators), while ^{63}Ni (a pure beta emitter) and ^{241}Pu (an alpha emitter) are typical Hard to Measure (HTM) radionuclides. In nuclear plants, ^{63}Ni structurally correlates with ^{60}Co since both originate from neutron activation of structural steel components, whereas ^{241}Pu co-migrates with ^{137}Cs as they are both fission and transuranic contaminants derived from core fuel degradation, making these correlations physically representative for scaling factor calculations

Table 5.1 Correlation Factor Results for Activation Products (Ni-63 / Co-60)

Parameter	Value
Correlation factor (A_{FC})	5.428
Dispersion factor $(D_{FC})^2$	4.186

Table 5.2 Correlation Factor Results for Contamination Products (Pu-241 / Cs-137)

Parameter	Value
Correlation factor (A_{FC})	8.217 E-7
Dispersion factor $(D_{FC})^2$	8.32

Where (A_{FC}) represents the geometric mean of the correlation factors and $(D_{FC})^2$ indicates the data dispersion; the detailed mathematical formulation and calculations of these parameters refer to Appendix B.

Further correlation analyses were performed for selected difficult to measure radionuclides present in liquid discharges. However, some of the obtained dispersion factors exceed the acceptance criteria ($D_{FC}^2 > 8$), indicating limited statistical reliability [21]. These correlations were not considered suitable for application and are reported in Appendix B for completeness.

Despite these limitations, the analysis provided useful operational support for discharge management, allowing the radiological conditions of the treated volumes to be characterized and supporting decision making related to release operations.

5.2.4 Instrumental Control and Radiological Characterization Activities

Within the framework of instrumental control, activities including calibration procedures, functional checks, and quality assurance operations on various measurement instruments were carried out, together with functional testing of radiological monitoring systems, with the aim of verifying their correct operation and operational reliability.

Figure 5.2 ISOCS system during calibration with reference source positioned in front of the detector

In particular, the student followed the calibration procedures of a spectrometer using a reference source, acquiring direct insight into the operational methods and the verification steps required to ensure the accuracy of the instrumental response Figure 5.2.

The radiological characterization procedure of waste using ISOCS systems was also observed, providing an overall understanding of the methodologies employed for material classification for disposal purposes.

5.2.5 Gamma Spectrometry on a Certified Source

Gamma spectrometry measurements were performed on a certified source to verify the correct operation of the detection system and the validity of both energy and efficiency calibrations. The analyses were carried out using an HPGe detector coupled with a multichannel acquisition system and dedicated spectral analysis software.

The acquired spectrum (Figure 5.3) shows the presence of the main characteristic photopeaks associated with the reference radionuclides used for calibration, in particular ^{137}Cs, ^{60}Co and ^{241}Am, whose emission energies reported in Table 4.1 are well defined and consistent with the expected values [22].

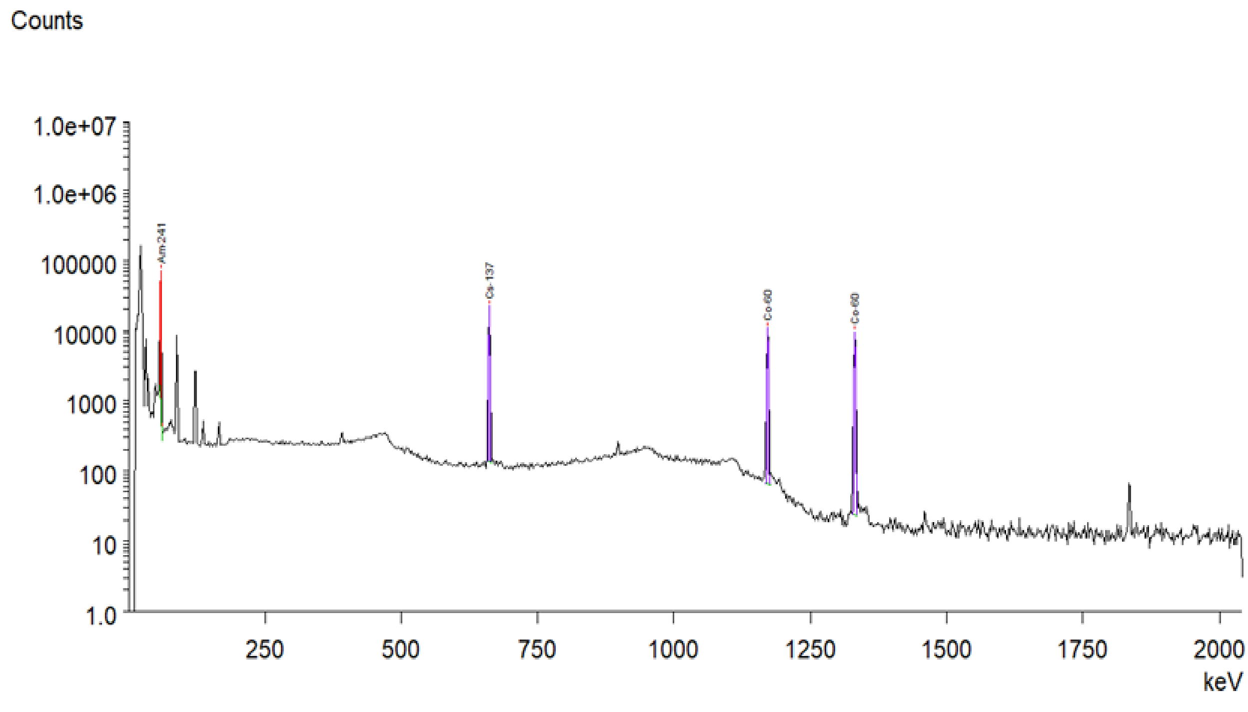


Figure 5.3 Gamma spectrum of the certified reference source showing the main photopeaks

The correct identification of these peaks confirms the validity of the energy calibration and the proper functioning of the measurement chain.

The quantitative analysis allowed the determination of the activities associated with the radionuclides present in the source, in agreement with the expected values for the reference source [22]. In addition, the Minimum Detectable Activity (MDA) values are on the order of a few Bq, indicating the high sensitivity of the system.

The quality of the measurement is supported by the presence of the Compton continuum, while the good definition of the photopeaks highlights the correct behavior of the detector with respect to the main interaction mechanisms of gamma radiation.

The energy and efficiency calibrations of the system are reported in the calibration curve (Figure 5.4).

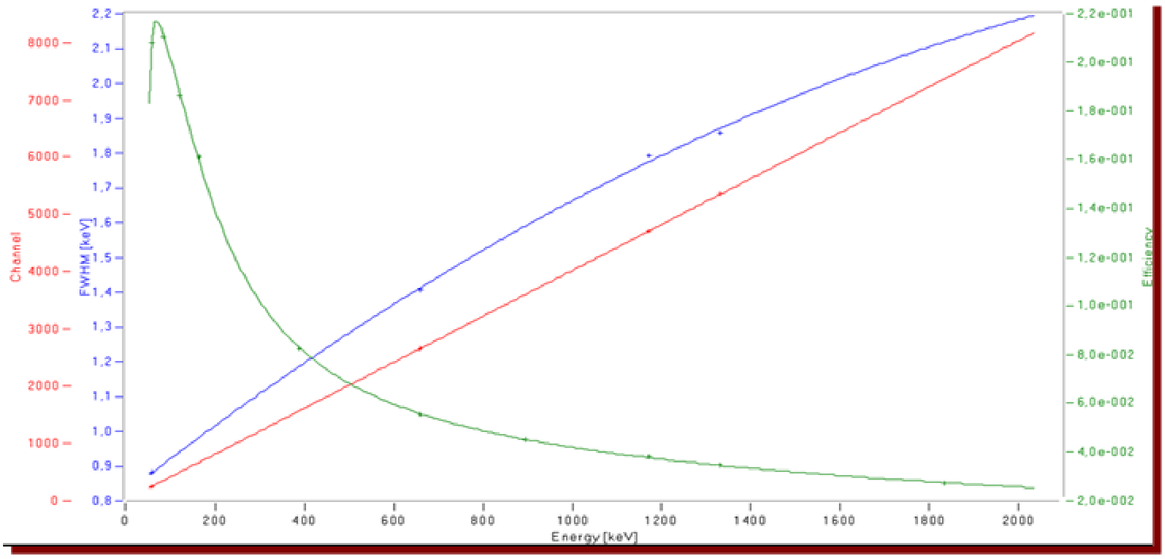


Figure 5.4 Energy and efficiency calibration curves applied to the measurement system

As further confirmation of the system performance, the results of quality assurance (QA) checks, reported in Table 5.3, show that the measured activities for the reference radionuclides are in good agreement with the nominal values of the source, with percentage deviations within a few percent and in any case below the acceptance thresholds [23].

Table 4.3 QA efficiency check results for reference radionuclides

Identified Peaks Summary							
Nuclide	**Reference Activity [Bq]**	**Measured Activity [Bq]**	**Energy [keV]**	**Peak Countrate [cps]**	**Counting Unct. [%]**	**Peak Activity [Bq]**	**Deviation %**
Am-241	**3.370,00**	**3.347,08**					**-0,68**
			59,54	246,95	0,20	3.347,08	-0,68
Co-60	**3.070,00**	**3.179,25**					**3,56**
			1.173,23	70,92	0,36	3.188,92	3,87
			1.332,49	64,16	0,37	3.169,52	3,24
Cs-137	**2.590,00**	**2.621,66**					**1,22**
			661,66	116,81	0,28	2.621,66	1,22
Referencedate of Source:	15/09/2023 12:00:00						

The performed measurements and the associated quality assurance checks validate the performance of the gamma spectrometry system, confirming its suitability for quantitative radiological characterization.

5.2.6 Whole Body Counter Analysis

A measurement was performed using a Whole Body Counter system, aimed at assessing potential internal contamination within the framework of radiological surveillance of personnel. The WBC system (Figure 4.20), based on high resolution detectors, allows the identification and quantification of radionuclides present within the human body through noninvasive techniques.

The measurement was carried out on the student at the end of activities in the controlled area, according to an operational procedure. The results show the absence of internal contamination from artificial radionuclides. In particular, the activities associated with ^{60}Co and ^{137}Cs are below the MDA limit, indicating the absence of significant contributions attributable to artificial exposure. The presence of the natural radionuclide Potassium ^{40}K was instead detected, with a value consistent with the physiological background due to the presence of potassium in the human body [24]. These results, reported in Table 5.4, confirm the proper sensitivity of the system and the validity of the performed measurement.

Table 5.4 WBC activity results and MDA for Co-60, Cs-137 and K-40

Measured activity (Bq)		
Co-60 $\pm 2\sigma$	Cs-137 $\pm 2\sigma$	K40 $\pm 2\sigma$
< 45(MDA)	< 84(MDA)	5204 ± 1536

The spectral analysis did not reveal the presence of peaks attributable to artificial radionuclides, as confirmed by the measurement report [24]. The acquired spectrum, reported in Figure 5.5, shows only contributions attributable to natural background, confirming the good quality of the measurement and the absence of significant interferences.

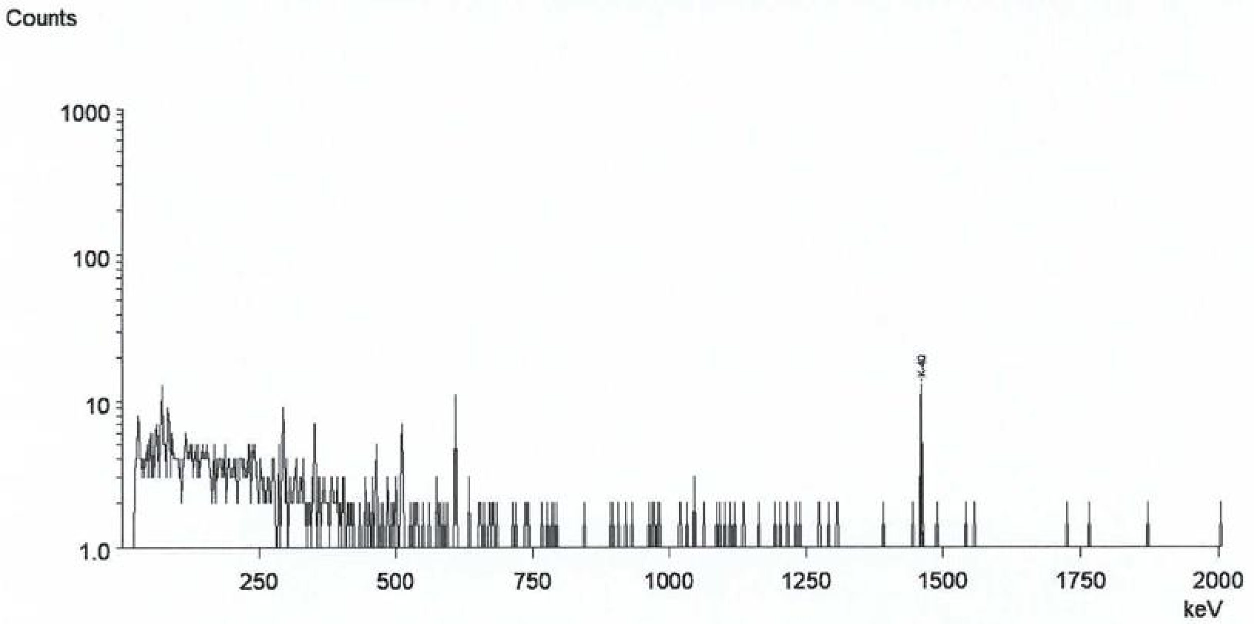


Figure 5.5 WBC gamma spectrum showing natural background and no detectable artificial radionuclides

The Whole Body Counter measurement allowed verification of the absence of internal contamination and provided direct insight into in vivo monitoring techniques, which are fundamental in the field of radiation protection of personnel.

5.2.7 Radiation Protection in Plant Operations

During the performed activities, the student also had the opportunity to observe plant operations related to the flooding of the reactor cavity and the subsequent phases of removal of the upper package, directly witnessing the application of radiation protection measures in a complex operational context. Under these conditions, control of exposure and contamination plays a central role and is ensured through careful planning of activities, the adoption of appropriate collective and individual protection measures, and continuous monitoring of radiological conditions. Attention is devoted to the preliminary assessment of dose levels and to real time verification of the evolution of environmental conditions, to ensure compliance with safety criteria and optimization of exposures according to radiation protection principles.

During the operations, it was also possible to observe inserts and equipment that will be employed in subsequent phases for sampling and radiological characterization of the upper package. These systems play a fundamental role in future activities, as they will allow representative sampling and targeted measurements, contributing to the reconstruction of the spatial distribution of activity and to the definition of dismantling strategies.

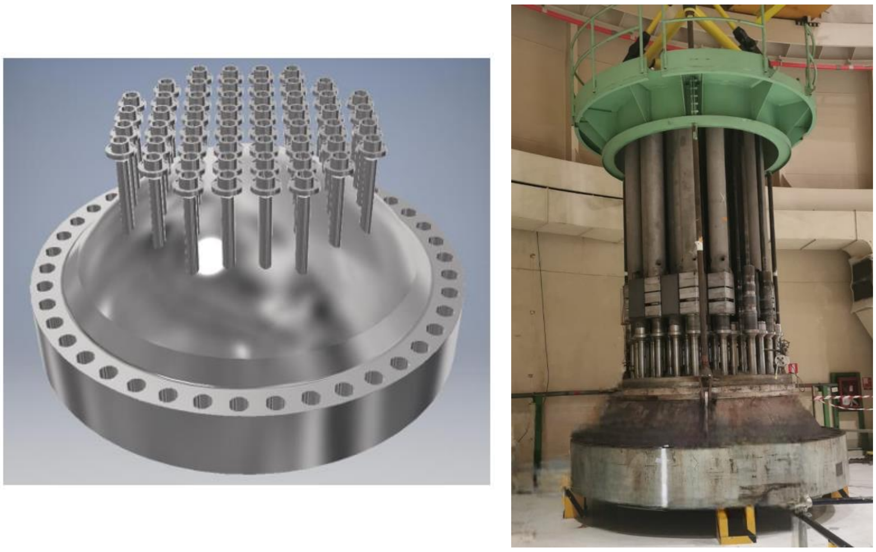
The observations carried out in this context therefore represent a direct link with the analyses developed in the following chapters, respectively dedicated to the radiological characterization of the vessel head and to the discussion of methodologies planned for future sampling on internals.

6. Radiological Characterization of Vessel Head Samples

The analysis and interpretation of the results derived from the radiological characterization of metallic samples collected from the Reactor Pressure Vessel head of the "Enrico Fermi" Nuclear Power Plant in Trino are developed on the basis of the available data. In particular, the results discussed are drawn from the intermediate radiological characterization report prepared by Nucleco S.p.A. [26] and from the associated official dataset [27], which contains the outcomes of the analyses performed on the samples.

The sampling activities are part of Phase 1 of the decommissioning process, aimed at the radiological characterization of the vessel and the internals. In particular, the analyzed samples originate from coring operations carried out on the vessel head, which enabled the collection of representative samples along the thickness of the material, allowing a distinction between surface and volumetric contributions [26].

The geometry of the vessel head is shown in Figure 6.1, which presents a render of the RPV head together with a real photograph of the vessel head, providing a geometric and visual overview of the structure. The geometric configuration and the operational conditions of the vessel head, already described in Chapter 3, directly influence the neutron activation phenomena and the surface contamination observed in the analyzed samples.



(a) *(b)*

Figure 6.1 (a) RPV head render, (b) actual vessel head

The radiological characterization was carried out using an integrated approach that combines high resolution gamma spectrometry and radiochemical analyses, allowing the determination of both gamma emitting radionuclides and hard to measure radionuclides such as pure alpha and beta emitters.

- In particular, the analyses were structured to distinguish two main contributions: surface contamination, expressed in Bq/cm^2, associated with deposition and transport phenomena of radionuclides within the primary circuit;
- neutron activation, expressed in Bq/g, representative of radionuclide production within the metallic matrix because of neutron irradiation during plant operation.

The objective of this chapter is therefore twofold. On one hand, it aims to present the experimental results obtained, highlighting the main radionuclides and the measured activity levels. On the other hand, it provides a critical discussion of the data, analyzing the activity distribution and the implications for the radiological classification of materials and for the planning of subsequent dismantling phases.

6.1 Sampling Campaign and Sample Description

The sampling campaign concerned the materials subject to analysis, with reference to the head of the Reactor Pressure Vessel of the Nuclear Power Plant in Trino.

The sampling operations focused on the reactor head, with the objective of obtaining samples representative of both surface contamination phenomena and neutron activation processes affecting the structural material.

As reported in the reference technical documentation [26], sampling was performed by coring using a magnetic drill equipped with a hole saw, capable of extracting cylindrical samples along the thickness of the vessel head. The sampling locations were selected to ensure spatial representativeness of the analyzed structure, as shown in Figure 6.2, which provides a detailed view of the sampling points on the vessel head, derived from the general representation already introduced in the previous chapters.

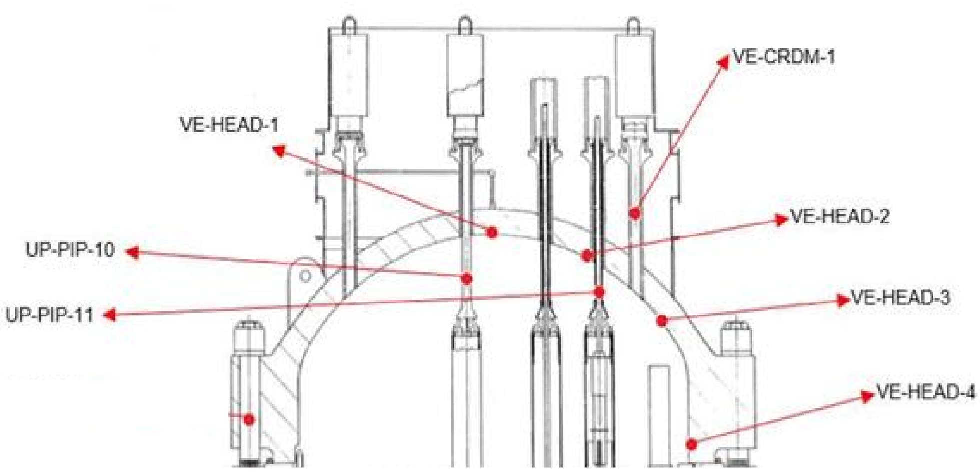


Figure 6.2 Sampling points on the vessel head (detail view)

Overall, four sampling locations were identified on the vessel head, labeled according to the plant operational coding as VE Head 1, VE Head 2, VE Head 3, and VE Head 4. In addition, two further locations related to the thermal sleeves were considered, identified as UP PIP 10 and UP PIP 11. The distribution of these locations on the component surface, shown in Figure 6.2, highlights their position with respect to the vessel head geometry and allows an assessment of their spatial representativeness.

For each sampling location, three cylindrical cores were extracted, for a total of eighteen primary samples related to the vessel head. This configuration was adopted in order to ensure sample traceability and independent validation of the analytical results. In particular:

- one core was allocated to radiological characterization analyses at the Nucleco laboratories;
- a second core was sent to the regulatory authority ISIN for independent analyses;
- a third core was retained on site as a reference sample.

The extracted cores have a length equal to the thickness of the vessel head, approximately 20 cm, and a cross sectional area not lower than 12 cm^2, thus being representative of the entire thickness of the component. The coring process was carried out using dedicated equipment capable of ensuring controlled extraction of representative samples along with the material thickness.

The cylindrical samples obtained in this way allow a direct analysis of the activity distribution, enabling a distinction between surface and volumetric contributions.

From a radiological perspective, coring based sampling clearly distinguishes between the contamination contribution, mainly localized in the surface layers in contact with the primary coolant, and the neutron activation contribution, distributed within the bulk of the material.

Following sampling, the specimens were identified through a double coding system, Sogin and Nucleco, to ensure full traceability throughout all management phases, from sampling to laboratory analyses.

6.2 Radiological Analysis Methodology

The radiological characterization of the samples collected from the vessel head was carried out using an integrated approach that combines high resolution gamma spectrometry and destructive radiochemical methods, ensuring a complete qualitative and quantitative determination of the radionuclides of interest, as described in the reference technical report [26].

Following sampling, the specimens underwent a preliminary pre-characterization phase by gamma spectrometry, performed directly on the as received material. This qualitative measurement allowed the identification of the main gamma emitting radionuclides present and provided useful information for both radioprotection purposes and the planning of subsequent analytical steps.

Subsequently, the samples were subjected to chemical pretreatment through partial dissolution of the metallic matrix, carried out using strong acid attack with HNO3, HCl, and HClO4. The choice of partial rather than total dissolution was driven by the need to avoid excessive concentration of the metallic matrix in solution, which could have compromised the effectiveness of the subsequent radiochemical separations.

For the vessel head samples, the analytical protocol included a distinction between contamination and activation contributions. In particular, the surface portion of the sample, identifiable by its different appearance and directly exposed to the primary coolant, was progressively removed through successive chemical attacks to obtain a solution representative of surface contamination, expressed in Bq per cm^2. The completion of this phase was verified by monitoring the presence of indicator radionuclides, whose disappearance indicated the effective removal of contamination [26].

The remaining portion of the sample, corresponding to the underlying structural material, was instead used for the determination of neutron activation, expressed in Bq per g. This approach allowed an effective separation between surface and volumetric contributions, enabling a correct interpretation of the results in terms of radionuclide origin.

The solutions obtained from the dissolution steps were subsequently analyzed by high resolution gamma spectrometry using HPGe detectors, following energy and efficiency calibration for the different measurement geometries. These measures enabled the determination of the main gamma emitting radionuclides. For such radionuclides, including ^{60}Co, ^{137}Cs, and ^{134}Cs, the analyses were performed directly on aliquots of the solution obtained from sample dissolution.

For radionuclides that are difficult to measure by gamma spectrometry, such as pure beta and alpha emitters, specific radiochemical separation procedures were applied, followed by dedicated measurement techniques. In particular:

- beta emitting radionuclides were determined by liquid scintillation counting;
- transuranic isotopes were analyzed by alpha spectrometry following chemical separation and electrodeposition;
- isotopic discrimination was supported, where necessary, by mass spectrometry techniques.

Measurement accuracy was ensured using analytical procedures compliant with international standards and through the evaluation of the Minimum Detectable Activity, calculated according to the methodologies defined in the reference regulations [26].

Overall, the adopted methodological approach provided a complete and reliable assessment of the radionuclide inventory of the analyzed samples, allowing a consistent distinction between surface contamination phenomena and neutron activation processes within the material.

6.3 Analysis of Experimental Results

The results of the radiological analyses, derived from the Nucleco technical report [26] and the associated dataset [27], are presented as a function of the different sampling locations on the vessel head, in order to highlight their spatial distribution and the differences in activity levels.

A summary of the main results, limited to the most representative radionuclides and to the maximum measured values for each sampling location, is reported in Table 6.1. This table enables a direct comparison among the different areas of the vessel head and highlights the differences between surface contamination and neutron activation contributions.

The complete results of the radiological analyses, including all identified radionuclides and the corresponding analytical parameters, are reported in Appendix C. The appendix also includes the values of MDC Minimum Detectable Concentration and Lc critical level, which are used as criteria for assessing detectability and statistical significance of the measurements, respectively representing the minimum quantifiable limit and the threshold for discrimination from background.

Table 6.1 Maximum Co-60 and Ni-63 activities in vessel head samples (contamination and activation)

Sampling point	Co-60 Contamination (Bq/cm^2)	Ni-63 Contamination (Bq/cm^2)	Co-60 Activation (Bq/g)	Ni-63 Activation (Bq/g)
VE-Head 1	3.76×10^{2}	3.03×10^{2}	1.36×10^{1}	5.44×10^{1}
VE-Head 2	4.40×10^{2}	1.18×10^{3}	2.65×10^{0}	3.16×10^{1}
VE-Head 3	1.89×10^{2}	3.87×10^{2}	4.31×10^{0}	2.12×10^{1}
VE-Head 4	1.98×10^{2}	1.09×10^{3}	1.26×10^{1}	6.50×10^{1}

In the following, the data are first presented in a descriptive form, with reference to the main identified radionuclides and the measured activity levels. Subsequently, an interpretative analysis is provided in order to highlight the physical phenomena underlying the observed results and the implications for radiological characterization and decommissioning activities.

6.3.1 Results for Vessel Head Samples

As reported in Table 6.1, the main radionuclides identified in the vessel head samples are Co 60 and Ni 63, both in terms of surface contamination and neutron activation of the material.

Regarding surface contamination, the highest values are observed at sampling location VE Head 2, which shows activity levels of 4.40×10^{2} Bq per cm2 for Co-60 and 1.18×10^{3} Bq per cm2 for Ni-63. The VE Head 4 location also exhibits elevated Ni 63 levels, comparable to those observed at VE Head 2. The distribution of contamination levels across the different sampling locations is shown in Figure 6.3.

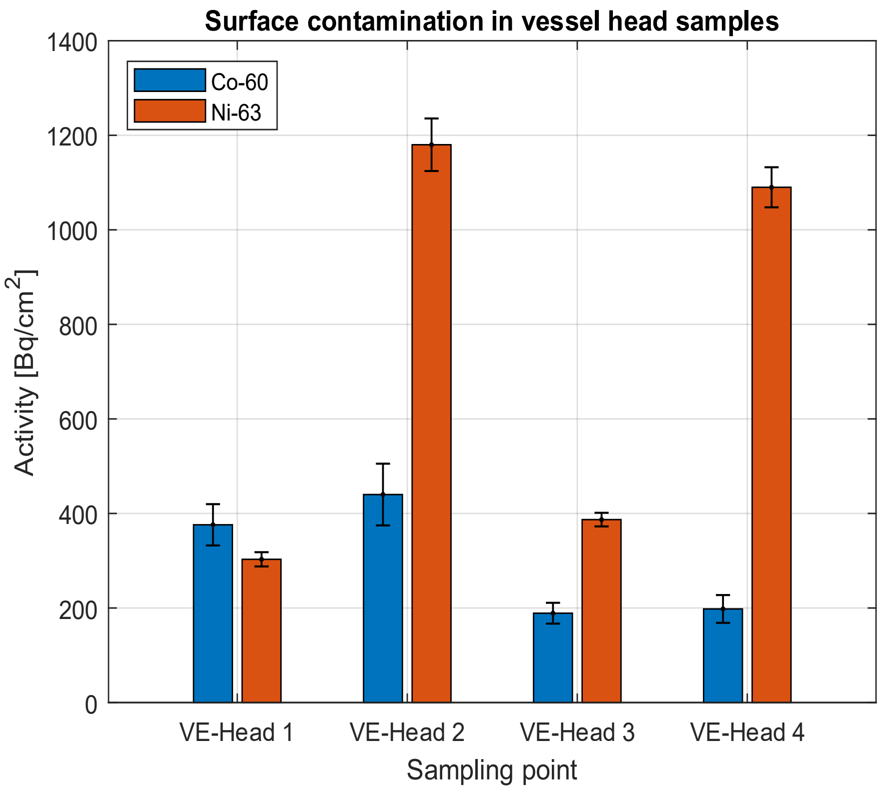


Figure 6.3 Comparison of Co-60 and Ni-63 surface contamination in vessel head sampling points

Regarding neutron activation, the results show a distribution that differs from that of surface contamination. In particular, the maximum Co 60 value is observed at sampling location VE Head 1, while Ni 63 reaches its highest value at VE Head 4, with an activity of 6.50×10^{1} Bq per g.

The VE Head 2 location, although dominant in terms of surface contamination, exhibits lower levels with respect to volumetric activation. The distribution of activation values across the different sampling locations is shown in Figure 6.4.

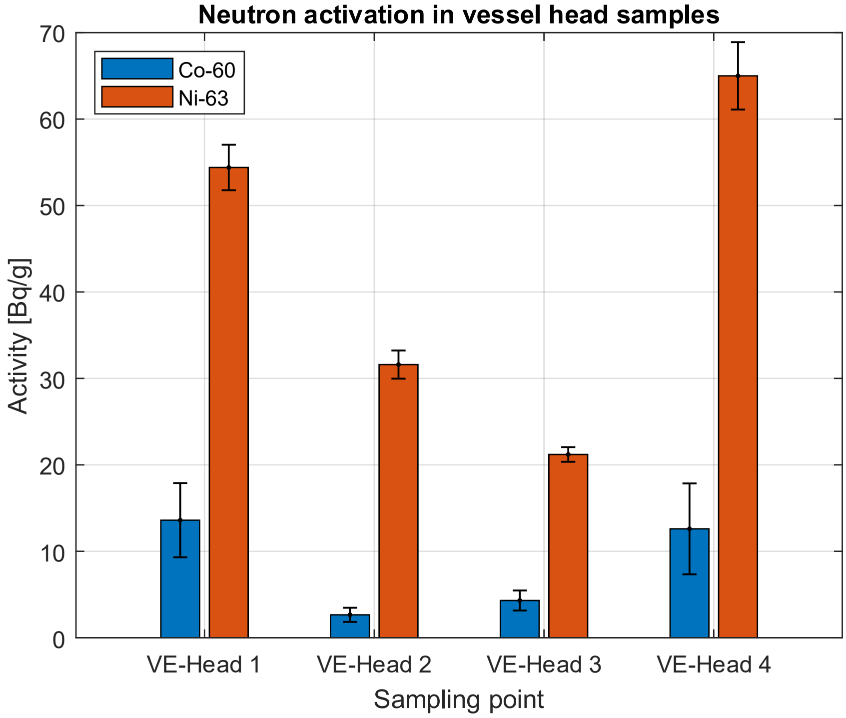


Figure 6.4 Comparison of Co-60 and Ni-63 neutron activation in vessel head sampling points

Overall, the results highlight a non-uniform distribution of activity on the vessel head, with significant differences among the various sampling locations for both analyzed contributions.

6.3.2 Analysis of Thermal Sleeve Samples

In addition to the samples collected from the vessel head, radiological analyses were also performed on two samples from the thermal sleeves, identified as UP PIP 10 and UP PIP 11. Unlike the vessel head samples, the contribution of surface contamination was not considered for these components, as the analyses were focused exclusively on neutron activation of the material.

The results show the presence of the main radionuclides typical of neutron activation in steels, including Co 60 and Ni 63, considered representative radionuclides, with activity levels comparable in order of magnitude to those observed in the vessel head samples. In particular, Ni 63 is the dominant radionuclide also in these samples, confirming the role of nickel in activation processes of structural materials.

A comparison between the two analyzed samples shows differences in activity levels, with sample UP PIP 11 generally exhibiting higher values than UP PIP 10. This difference can be attributed to a different exposure to neutron flux during plant operation, related to the position of the components within the vessel and to the configuration of the neutron field.

The comparison between the two samples is shown in Figure 6.5, which reports the activity levels associated with the main activation radionuclides, including the corresponding measurement uncertainties.

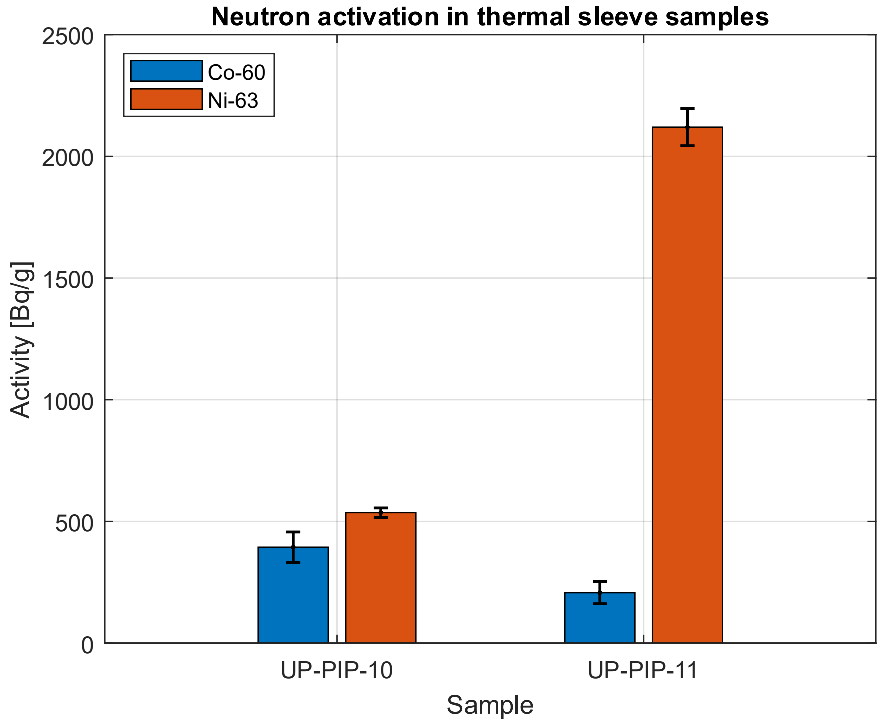


Figure 6.5 Comparison Co-60 and Ni-63 neutron activation in thermal sleeve sampling points

In analogy with what was observed for the vessel head, the activity distribution in the thermal sleeves is also non uniform, confirming the influence of local operating conditions and system geometry. However, the absence of a surface contamination contribution allows a more direct isolation of the neutron activation phenomenon, making these samples particularly significant for the analysis of the radiological behavior of the structural material.

The results obtained for the thermal sleeves are consistent with those observed in the vessel head samples and confirm the predominance of activation radionuclides in determining the radiological inventory of reactor internal components.

6.3.3 Final considerations on Experimental Results

The obtained results highlight the combined presence of surface contamination and neutron activation contributions, which significantly determine the radiological inventory of the analyzed components.

For the vessel head samples, the analysis showed variability in activity levels among the different sampling locations, attributable to the influence of neutron flux distribution, component geometry, and corrosion product transport phenomena within the primary circuit.

The results related to the thermal sleeves confirm the predominance of neutron activation processes in structural materials, showing activity levels comparable in order of magnitude to those observed in the vessel head and differences attributable to varying exposure to the neutron field.

Overall, the set of analyses carried out allows a reliable definition of the radiological inventory of the examined components, providing a consistent and representative dataset for subsequent engineering evaluations and for the planning of decommissioning activities.

6.4 Implications for Radiological Characterization and Decommissioning

The experimental results presented in the previous section allow several relevant considerations to be drawn for the purposes of radiological characterization of materials and the planning of decommissioning activities.

In particular, the analysis of the identified radionuclides, the measured activity levels, and their spatial distribution highlights the most significant aspects related to the operational management of the analyzed components, with reference both to dismantling operations and to the subsequent classification of materials as radioactive waste.

The considerations developed in this section are also integrated with the information reported in the reference technical documentation [28], to frame the obtained results within the operational and regulatory context applicable to decommissioning activities.

6.4.1 Radiological assessment of materials

The analysis of the experimental results allows a reliable definition of the radiological behavior of the materials composing the vessel head and the thermal sleeves, highlighting the main radiological implications of the analyzed materials.

In particular, the predominant radionuclides are Co 60 and Ni 63, which represent respectively the main contribution in terms of external radioprotection and are one of the most relevant radionuclides for activation of metallic materials. The presence of these radionuclides is consistent with neutron activation phenomena in steels and with transport and deposition processes of corrosion products in the primary circuit.

The variability of activity levels observed among the different sampling locations highlights the influence of local conditions such as component geometry, neutron flux distribution, and primary coolant flow characteristics. This variability must be properly considered in the definition of representative characterization strategies, to avoid underestimation or overestimation of the radiological inventory.

The obtained results also confirm the need to distinguish between contamination and activation for a correct radiological assessment of materials. In this context, the adopted methodological approach proves to be adequate to provide a complete and consistent evaluation of the radiological state of the analyzed materials.

6.4.2 Operational aspects of dismantling

From an operational perspective, the results of the radiological characterization provide essential information for planning the dismantling activities of the vessel and its internal components.

The presence of residual surface contamination, particularly associated with radionuclides such as Co 60, implies the need to adopt appropriate radioprotection measures during cutting and handling operations, to limit the dispersion of contaminated particulates and personnel exposure. In this context, containment and ventilation techniques, as well as the use of filtration systems for the control of radioactive aerosols, are of particular importance.

About neutron activation, the presence of radionuclides distributed within the material implies that the entire volume contributes to the radiation field, influencing handling, segmentation, and temporary storage procedures. The variability of activity levels among the analyzed locations suggests the need for accurate planning of operations based on detailed and representative characterization.

The experimental evidence is consistent with what is reported in the reference operational documentation [28], confirming the importance of integrating radiological characterization data into the design and management phases of decommissioning activities, to ensure adequate safety conditions and optimize operational organization.

6.4.3 Management and classification of radioactive waste

The obtained results also play a central role in the management and classification of materials arising from dismantling operations.

The presence of activation radionuclides directly influences waste classification, as these radionuclides contribute to determining activity levels and radiological characteristics of the material. In particular, Ni 63, characterized by beta emission and a relatively long half-life, represents a critical factor in evaluating disposal options, while Co 60 is relevant due to its contribution to dose rate and exposure conditions during operational handling.

The distinction between contamination and activation is fundamental in defining waste management strategies. While contamination can in some cases be removed through decontamination processes, neutron activation is intrinsic to the material and permanently determines its radiological characteristics.

The characterization data therefore effectively supports decision making processes related to waste classification and to the selection of treatment and disposal options, in accordance with the regulatory and operational criteria defined in the reference documentation [28].

Overall, the conducted analysis provides a reliable radiological framework for the examined components, highlighting the most relevant aspects for operational management and material classification. The obtained results represent an essential reference for the planning of decommissioning activities and for the definition of treatment and disposal strategies, which will be further addressed in the following chapter.

7. Theoretical Operational Plan for Vessel and Internals Sampling

The proposed theoretical operational plan for the sampling of the Reactor Pressure Vessel and its internals is developed to support radiological characterization activities in the subsequent phases of decommissioning. The definition of the plan is based on the integration of the experimental evidence discussed in the previous chapters with engineering and radiation protection criteria and is primarily founded on the methodologies and operational approaches reported in the reference technical documentation [28], appropriately adapted to the case under consideration. In this context, the historical operating conditions of the plant, the expected distribution of activation and contamination phenomena, and the technical and radiological constraints associated with operations in a controlled environment are considered.

The proposed plan is structured according to a logical sequence that includes the definition of objectives, the identification of sampling points, the selection of operational methodologies, and the evaluation of the main safety constraints, with the aim of providing a coherent and applicable framework for future dismantling activities.

7.1 Objectives of the Theoretical Plan

The theoretical plan defined in this chapter is aimed at identifying criteria and operational approaches capable of ensuring the representativeness of the collected samples, with respect to the spatial distribution of radionuclides and to the neutron activation and surface contamination mechanisms affecting the different components under investigation.

One of the main objectives is the selection of representative sampling points, enabling a reliable reconstruction of the radiological inventory of materials, taking into account neutron irradiation conditions, the geometrical characteristics of the vessel and its internals, and the different operational exposures experienced during the plant lifetime. In this sense, the plan is intended to support the radiological classification of materials and their subsequent waste management, in accordance with the applicable regulatory criteria.

A further objective is the definition of an operational approach integrating technical and radiological protection aspects, ensuring the feasibility of sampling activities under safe conditions for operators and in compliance with the ALARA principle.

7.2 Criteria for the Selection of Sampling Points

The selection of sampling points within the Reactor Pressure Vessel and its internals is aimed at ensuring the representativeness of samples with respect to the spatial distribution of radiological activity. The sampling locations identified for the upper package and for the internals have already been defined and presented in Chapter 3 (Figures 3.5, Figures 3.6, Figures 3.7), to which reference is made for detailed geometrical description and localization.

In the present context, the focus is therefore placed on the criteria that guided such selection. In particular, the distribution of neutron fluxes within the vessel leads to activation gradients both along the vertical axis and in the radial direction, with higher levels in regions close to the core and progressively decreasing values in upper and more distant areas.

These phenomena are complemented by effects related to the operating conditions of the primary circuit, such as coolant flow distribution, inlet and outlet configurations, and possible stagnation zones, which may significantly influence surface contamination levels. As a result, the sampling plan must include both areas predominantly affected by neutron activation and regions where surface contamination plays a more relevant role.

The selection of sampling points must be compatible with the operational and radiological constraints of sampling activities, in terms of accessibility, dose conditions, and feasibility of execution. Consideration is given to the use of dedicated remotely operated sampling systems, to ensure a balance between data representativeness and operational safety.

7.3 Sampling Methodologies, Equipment, and Radiological Requirements

The definition of sampling methodologies and the selection of the equipment to be employed represent a central aspect in the development of the theoretical operational plan, as they directly influence the feasibility of the activities, the quality of the obtainable data, and the level of radiological protection ensured during operations. In accordance with the reference technical documentation, solutions enabling sampling activities to be carried out under high safety conditions must be prioritized, minimizing personnel exposure and preventing the spread of contamination.

From a methodological standpoint, sampling activities should be oriented toward techniques capable of providing representative samples both for the assessment of surface contamination and for the evaluation of activation within the bulk of the material. To this end, the Operational Plan refers to remotely executable techniques performed under submerged conditions, including boat sampling by electrical discharge machining, surface removal by EDM, drill sampling, and lens sampling using a mechanical milling tool. These techniques allow the extraction of small-mass samples, on the order of grams, thereby reducing shielding, handling, and transport requirements while maintaining adequate radiological representativeness.

The selection among different methodologies must be performed based on the type of component, the required depth of information, and the accessibility conditions of the surface to be sampled. Techniques such as boat sampling and lens sampling are particularly suitable when both an almost intact surface for contamination assessment and a portion of material for activation analysis are required. Drill sampling, on the other hand, enables the acquisition of information on activity profiles along with the material thickness and is therefore particularly useful when reconstructing activation gradients. Surface removal by EDM is more appropriate when the primary interest is limited to the superficial contamination component.

From an equipment perspective, the reference documentation foresees the use of remotely operated systems that can be positioned by operators from the cavity working platform, possibly with the support of the polar crane. In particular, the sampling system can be installed and handled by means of the polar crane, enabling controlled positioning of the equipment in the areas of interest and the execution of sampling operations at a safe distance from the radiation source.

The reaction force systems of the sampling machine may be based on suction or electromagnetic anchoring, thus avoiding the need for dedicated support structures and reducing installation complexity in the field. As shown in Figure 4.22, the use of dedicated equipment ensures high positioning accuracy and operational reliability, contributing to the reduction of operator exposure. This approach is particularly suitable in operational scenarios characterized by limited accessibility, high reliability requirements, and the need to minimize personnel presence in controlled areas.

An additional relevant aspect concerns sample management following collection. According to the Operational Plan, the sample, collected onboard the equipment under submerged conditions, is transferred to the operating platform and prepared for transport, adopting, where possible, shielded configurations between the sample and the operator. This highlights how methodological choices can not be limited to the material removal phase alone, but must also include subsequent steps such as recovery, packaging, temporary storage, and transfer to the analysis laboratory. The possibility that the designated laboratory may be located outside the national territory further emphasizes the need for sampling methods and containment systems compatible with applicable transport regulations.

Alongside sampling techniques, the selection of operational equipment must comply with radiological protection criteria, prioritizing remotely operated solutions and operations conducted under water.

In this perspective, the water layer plays a fundamental operational role, providing effective shielding against gamma radiation emitted by activated components in addition to enabling remote operation. This approach is particularly suitable for handling and sampling components such as the core barrel and the upper package, for which the Operational Plan explicitly provides underwater operations using equipment derived from or adapted from operational procedures. The continuity with equipment and procedures already used within the plant represents an additional advantage, as it reduces the need for new installations and leverages systems already compatible with the reactor configuration.

Regarding radiological requirements, the selection of methodologies and equipment must be consistent with the radiation protection objectives established for both workers and the public. The Operational Plan emphasizes compliance with the optimization principle, the need to limit liquid and airborne releases, the use of containment systems or localized extraction with HEPA filtration where required and the adoption of specific provisions to control both individual and collective dose. Annual dose limits for exposed workers are considered, together with additional daily and monthly constraints under normal operating conditions, making it essential to design activities aimed at reducing exposure time, increasing distance, and employing adequate shielding.

The selection of auxiliary technologies must also be consistent with this approach. In general, the Operational Plan prioritizes mechanical cutting techniques over thermal cutting, to reduce the generation of radioactive particulates, fire risks, and secondary waste. Although this criterion primarily applies to dismantling activities, it also provides a relevant methodological indication for characterization, confirming the preference for controllable, selective processes compatible with capture and filtration systems. Consequently, for vessel and internals sampling, techniques that minimize material dispersion, allow efficient sample recovery, and reduce the need for direct intervention near the radiation source are preferred.

The selection of sampling methodologies, equipment, and radiological requirements should be interpreted as the result of a balance between analytical needs, plant constraints, and radiation protection objectives. The theoretical approach proposed here, consistent with the reference technical documentation, prioritizes remotely operated techniques, small-sized samples, underwater operations, and shielded handling systems, defining an operational framework aimed at maximizing compatibility between characterization effectiveness and operational safety.

7.4 Proposed Operational Sequence

The definition of the operational sequence represents the final stage in the development of the theoretical sampling plan, as it allows the various activities described in the previous sections to be organized in a coherent and functional manner. This sequence is developed based on the indications provided in the reference technical documentation and considers radiological constraints as well as access conditions to the intervention areas.

The proposed operational sequence can be summarized in the following main steps: vessel opening and flooding, removal of upper internals, execution of sampling activities, and subsequent dismantling and segmentation operations.

In the initial phase, the activities involve opening the Reactor Pressure Vessel by removing the vessel head, followed by flooding of the reactor cavity. This operation enables the establishment of a shielded environment for the subsequent phases, exploiting the water layer as an effective barrier against gamma radiation. Performing operations under water represents a key element in reducing operator exposure and ensuring compliance with the ALARA principle.

Subsequently, the removal of the upper internals, particularly the upper package, is carried out by means of remotely operated lifting and handling systems. In the case under consideration, these operations have already been completed, including vessel head removal, cavity flooding, and upper package extraction, while the subsequent phases of the operational sequence are planned within future decommissioning activities. This phase allows access to the internal structures of the vessel and makes the surfaces of interest available for sampling operations.

Once the removal of the upper structures has been completed, sampling activities can be performed on internal components such as the core barrel, vessel surfaces, and other representative structures. These operations are carried out using the techniques described in the previous section.

In a subsequent phase, if required within the scope of dismantling activities, segmentation operations may be performed, including cutting techniques executed under water to reduce dose rates and limit the dispersion of contamination in the working environment.

In the later stages of decommissioning, the activities may extend to the removal of the main components of the primary circuit and large equipment, such as the vessel, steam generators, and pressurizer. These operations involve the controlled handling and removal of components using appropriate lifting and transport systems, depending on their dimensional and radiological characteristics.

Figure 7.1 (a) internals removal, (b) vessel removal, (c) SG cutting, (d) pressurizer cutting

As shown in Figure 7.1, the different operational phases include the removal of internals (a), vessel handling and removal (b), as well as segmentation operations of the steam generators (c) and the pressurizer (d). These activities represent a complex phase of the dismantling process, requiring accurate planning and integrated management of radiological and logistical aspects.

The removal and handling of components are also closely connected to the subsequent phases of material management, classification, and treatment, which follow specific operational pathways depending on their radiological characteristics, as described in the following section.

Overall, the proposed operational sequence is configured as a progressive and structured process, starting from preliminary access and safety preparation activities, continuing with component sampling, and extending to dismantling and removal of the main equipment. This approach enables the effective integration of radiological characterization requirements with safety and operational constraints, providing a useful reference for planning future decommissioning activities.

7.5 Classification and Management of Radioactive Waste

The sampling and radiological assessment activities described in the previous sections are aimed not only at defining the radiological inventory of the components, but also at determining the appropriate management strategies for the materials generated during subsequent dismantling phases. In this context, the planned measurements play a key role in the radiological classification of materials and in the identification of suitable management pathways.

The analyses performed on the collected samples allow the determination of the concentration and distribution of radionuclides within the various vessel and internal components, distinguishing between neutron activation and surface contamination contributions. These data provide the basis for assigning materials to specific radiological categories, supporting decisions related to treatment, temporary storage, or potential clearance.

Materials can be grouped into homogeneous categories based on their radiological characteristics, in accordance with the reference technical documentation. This classification allows differentiation between activated and contaminated materials, decontaminated materials, and potentially releasable materials, facilitating the definition of management strategies according to radiological risk levels and available treatment and disposal options.

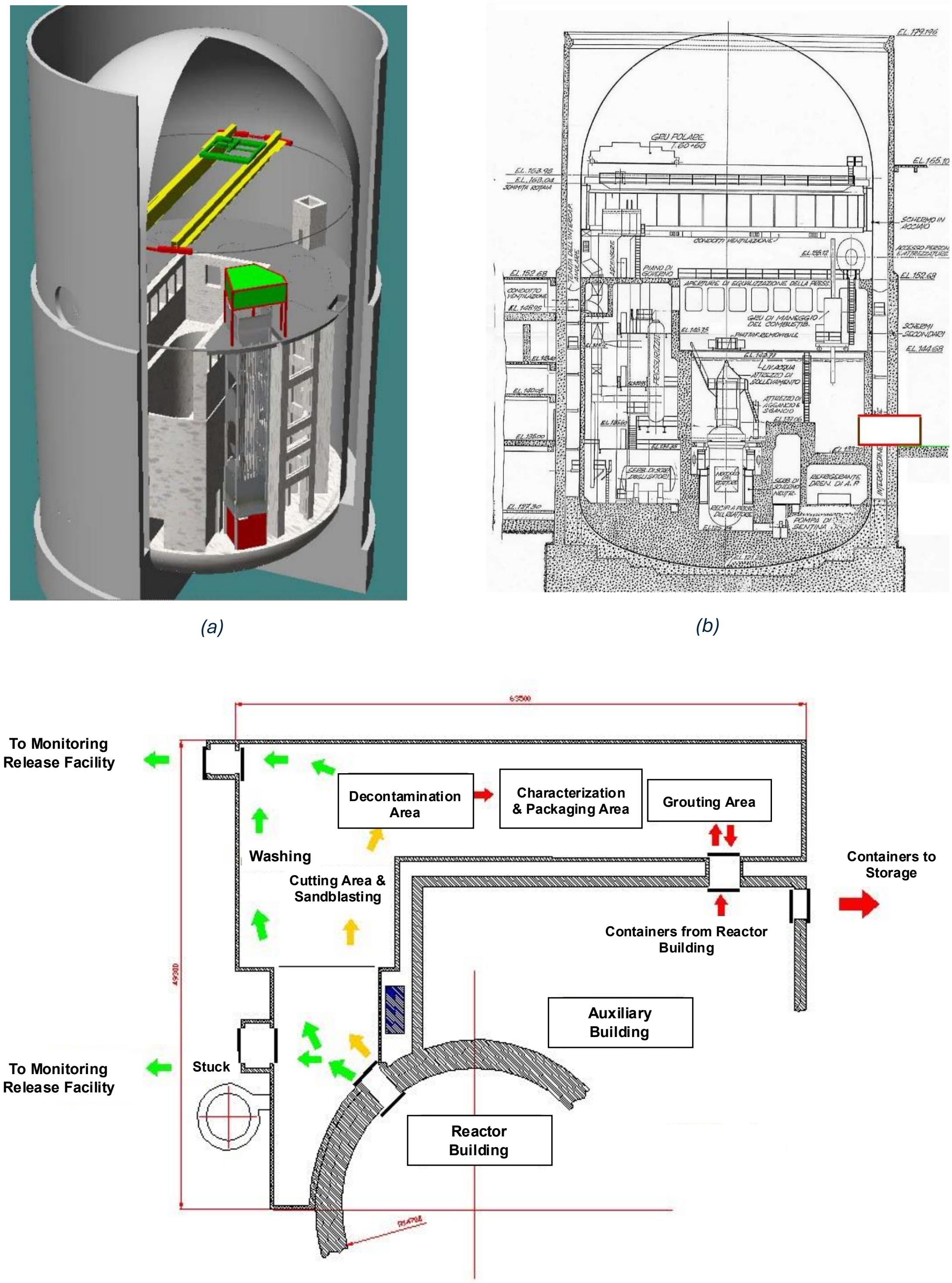
Based on characterization results, materials generated during dismantling activities can be directed toward different management pathways, including treatment and conditioning as radioactive waste, temporary storage pending transfer to the national repository or where compliant with regulatory limits, release as non-radioactive material.

As illustrated in Figure 7.2, materials follow defined management routes starting from their point of generation, through collection, classification, possible treatment, and storage, up to final disposal, depending on their radiological properties.

In this context, the characterization process represents a key step in optimizing waste management, reducing disposal volumes, and promoting material recovery whenever feasible.

From an operational standpoint, material management must be implemented in accordance with traceability and segregation principles, ensuring the separation of material streams based on their radiological classification and preserving information regarding origin, characteristics, and final destination. This approach supports subsequent treatment and disposal phases while ensuring compliance with regulatory requirements.

Finally, the planning of sampling and characterization activities should also aim at minimizing the generation of secondary waste, favoring techniques that limit material dispersion and reduce the quantity of residues produced during operations. This is consistent with an integrated approach to decommissioning, in which radiological characterization plays a central role in optimizing safety, sustainability, and material management.



(a) (b)

(c)

Figure 7.2 Waste route: (a) perspective view, (b) sectional view, (c) plan view.

7.6 Constraints, Limitations and Safety Considerations

The definition of the operational plan must consider a set of radiological, technical, and operational constraints that affect both feasibility and execution of sampling activities.

From a radiological perspective, the presence of activated and contaminated components leads to conditions characterized by potentially high dose rates, requiring the adoption of operational solutions aimed at minimizing worker exposure in accordance with the ALARA principle. In this context, the use of remote handling techniques and the execution of operations under water are essential to ensure adequate safety margins.

From a technical standpoint, accessibility to intervention areas and the geometrical configuration of the vessel and internals represent additional limiting factors that influence equipment positioning and the feasibility of sampling operations. These aspects require careful planning and the use of appropriate handling systems.

Operational constraints include the need to ensure compatibility of sampling activities with the current plant configuration and with decommissioning phases already completed, avoiding interference with ongoing operations and maintaining safe working conditions.

Taking together, these limitations do not represent only constraints but also provide guidance for the development of a coherent, safe, and technically feasible operational plan.

7.7 Comparison with Approaches Adopted in other Nuclear Power Plants

The theoretical operational plan developed in this chapter is consistent with approaches adopted in the decommissioning of similar nuclear power plants, particularly about operations involving the Reactor Pressure Vessel and its internals. The proposed strategies, based on a progressive sequence of access, removal of internal components, sampling, and where necessary segmentation, are in line with documented operational experiences at both national and international levels.

In particular, the removal of internals, the use of underwater cutting techniques, and the subsequent management of activated materials represent well-established practices in the decommissioning of pressurized water reactors. These approaches ensure an adequate level of radiological safety by exploiting the shielding effect of water and reducing operator exposure during the most critical phases, as reported in technical literature [29].

A further element of consistency is represented by the integration of radiological characterization activities with the planning of dismantling phases. In accordance with radiation protection guidelines, the prior definition of the radiological inventory of components constitutes a fundamental prerequisite for material classification and optimization of waste management strategies [28]. This approach reduces operational uncertainties and improves the overall efficiency of the decommissioning process.

The comparison with experiences from other facilities highlights that the proposed theoretical plan is embedded within a consolidated methodological framework, characterized by the use of remote technologies, execution of operations in shielded conditions, and strong integration between radiological and operational aspects. This consistency supports the validity of the adopted approach, confirming its applicability and reliability in the context of future dismantling activities of the vessel and internals at the Trino site.

8. General Discussion

The results obtained from the radiological characterization of the Reactor Pressure Vessel and its internals are critically interpreted, relating them to the geometric, material, and operational characteristics of the system, as well as to the implications for the subsequent decommissioning phases. The analysis is based on the data already presented in the previous chapters, highlighting their engineering and operational significance.

8.1 Interpretation of Radiological Results

The experimental results show an overall consistent pattern among the different components analyzed, despite the presence of local variability attributable to specific operating conditions. Such variability represents an expected feature, as it is directly related to the nature of the radiological phenomena involved.

In particular, a clear distinction emerges between the contribution due to neutron activation and that associated with surface contamination. The predominance of activation radionuclides in structural materials, including Co-60 and Ni-63, indicates that the radiological behavior of the system is primarily governed by the neutron irradiation history of the components, whereas surface deposition phenomena appear more localized and variable.

In cases where surface contamination is negligible, the analysis allows the activation contribution to be isolated, highlighting the relationship between activity levels and the position of the component relative to the neutron field. In this context, Ni-63 assumes particular relevance as a long-lived activation radionuclide, typically classified among Hard-To-Measure radionuclides, whose contribution is representative of the volumetric activation of materials. The distribution of radiological activity is strongly influenced by the geometrical configuration of the vessel and its internals, as well as by the operating conditions of the primary circuit, with neutron flux intensity representing the dominant parameter in determining the level of activation. Regions closer to the core exhibit higher activity levels, whereas more distant areas are less activated and relatively more influenced by surface contamination.

A further element of consistency is provided by the temporal evolution of the radiological inventory. Considering that the final shutdown of the Trino plant dates back to 1987, approximately 39 years have elapsed, corresponding to more than seven half-lives of Co-60. As a result, the residual activity of this radionuclide has decreased to approximately 0.6% of its initial value. Despite this reduction, Co-60 still represents a significant contribution to the radiological inventory due to its high-energy gamma emissions and therefore remains relevant for dose assessment.

Longer-lived radionuclides, such as Ni-63, are less affected by radioactive decay over time and contribute persistently to the overall radiological inventory, playing a particularly important role in the long-term characterization of activated materials.

These findings are fully consistent with the expected behavior of internal components of a PWR reactor, thereby reinforcing the overall reliability of the radiological interpretation.

8.2 Implications for Decommissioning Activities

The interpretation outlined above provides a solid basis for the planning of decommissioning operations. The ability to distinguish between volumetrically activated components and those primarily affected by surface contamination is particularly relevant, as it directly influences the selection of appropriate operational strategies.

Handling of activated materials generally involves more complex procedures and higher dose rates, whereas surface-contaminated components may, in certain cases, be treated through decontamination processes. Accurate interpretation of radiological data therefore supports both reliable waste classification and the optimization of dismantling strategies.

From an operational standpoint, the results contribute to informed decision-making regarding dismantling techniques, material management, and radiation protection measures. Knowledge of the spatial distribution of activity enables more effective planning, with potential benefits in terms of dose reduction and operational efficiency.

Within this framework, the findings are fully consistent with the decommissioning activities carried out by Sogin S.p.A. and provide both technical support and a practical reference for optimizing intervention strategies and radiological management.

8.3 Limitations of the Study and Future Perspectives

Although the results are coherent and robust from an interpretative standpoint, certain limitations should be acknowledged.

The limited number of available samples restricts the ability to fully characterize the spatial variability of radiological activity within the vessel and its internals. In addition, the heterogeneity of operating conditions at different sampling locations introduces variability in the data, reducing the extent to which the observations can be generalized.

Another limitation lies in the absence of neutron activation modelling or numerical simulations, which prevents the integration of a predictive component into the analysis. As a result, the interpretation remains primarily descriptive, albeit consistent with the expected physical behavior.

Future work may address these aspects by extending the sampling programme, thereby improving data representativeness and reducing associated uncertainties. At the same time, coupling experimental data with activation models would enable the development of more advanced predictive tools, enhancing support for decision-making during decommissioning activities.

Validation of the proposed interpretations will also benefit from comparison with data obtained during subsequent dismantling phases. In this perspective, the present study provides a solid technical basis for further developments in radiological characterization methodologies.

9. Conclusions

This study focused on the radiological characterization of the Reactor Pressure Vessel and its internals within the decommissioning framework of the Trino nuclear power plant, with the aim of defining the radiological behavior of the main components and supporting dismantling operations.

The results outline a consistent radiological scenario and allow a reliable assessment of the inventory of the analyzed components. The analysis made it possible to identify the dominant physical mechanisms and to describe the spatial distribution of activity within the system in a coherent manner.

The outcomes provide useful input for operational decision-making, contributing to the definition of material management strategies and radiation protection measures. In this regard, they represent a technical reference for the activities carried out by Sogin S.p.A.

The main contribution lies in the improved understanding of the radiological behavior of the components and in the methodological approach adopted for their characterization. Extending the sampling programme and integrating the results with activation models may further enhance predictive capabilities and strengthen support for future decommissioning operations.

Appendix A - Radiological Characterization Maps

The detailed layout of the monitored area, along with the corresponding radiation exposure levels and surface contamination values, is fully illustrated. Appendix A provides comprehensive visual and quantitative reference through dedicated schematics and localized radiological data maps, focusing on two specific operational zones:

- *Area 1-Health Physics Laboratory* displaying minimal levels of both surface contamination and radiation exposure.
- *Area 22-Boron Tank Room* used for primary circuit loop decontamination flushing. While surface contamination values are low, it presents non-negligible exposure levels that necessitate strict safety precautions.

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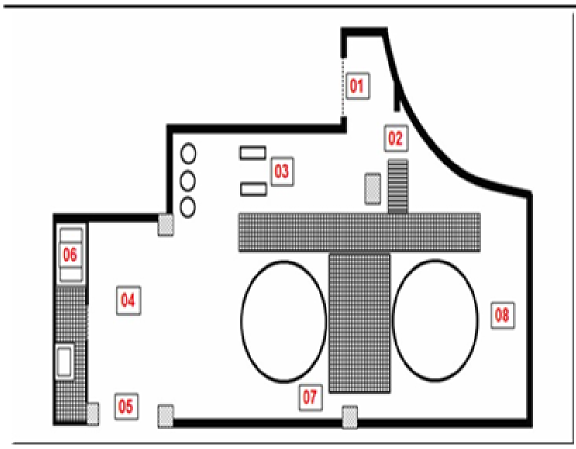
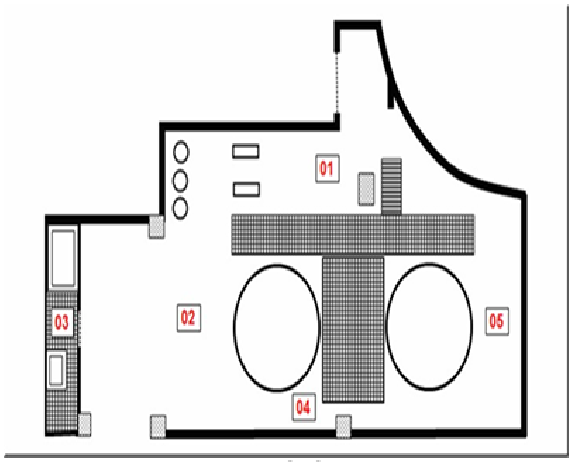
22-Locale serbatoi boro



Contaminazione

Punto di Interesse	Valore (Bq/cm^2)
01 - Pavimento lato sud	1,23E-2
02 - Pavimento antist. scala	3,87E-3
03 - Pavimento antist. pompe	9,84E-3
04 - Pavimento antist. ballatoio	6,74E-3
05 - Pavimento antist. locale casco	1,84E-2
06 - Vent. su ballatoio	6,79E-3
07 - Pavimento lato est	7,01E-3
08 - Pavimento antist. parete nord	4,37E-3
Max:1,84E-2 - Media:8,67E-3	



Esposizione

Punto di Interesse	Valore (µSv/h)
01 - Ambiente ingresso locale	4,00E-1
02 - Ambiente antist. locale casco	2,00E-1
03 - Ambiente grigliato antist. A-B	2,00E-1
04 - Ambiente retro serbatoi	2,00E-1
05 - Ambiente antist. lato nord	4,50E1
Max: 4,50E1 - Media: 9,20E-0	

Particolato	Valore (Bq/cc)
22 - Locale serbatoi boro	5,32E-8

Figure A.1 E.g. of extended routine sheet area layout and measured contamination and dose rate values for Area 22

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Controlli radiometrici giornalieri negli ambienti di Zona Controllata

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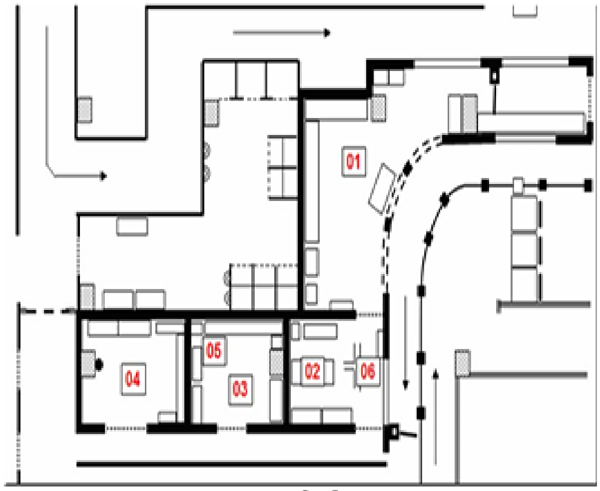
1-Lab. Fisica Sanitaria e Magazzini



Contaminazione

Punto di Interesse	Valore (Bq/cm²)
01 - Pavimento sotto tavolo calcolatori	5,24E-3
02 - Tavolo calcolatori	4,04E-3
03 - Pavimento sotto RIC 308	5,34E-3
04 - Pavimento laboratorio	3,97E-3
05 - Scrivania laboratorio	3,96E-3
06 - Banco del CCA Selo	4,82E-3
07 - Banco del CCA T.EMI	3,76E-3
08 - Pavimento tra laboratorio e ufficio	6,55E-3
09 - Pavimento porta ufficio	7,98E-3
10 - Pavimento ufficio	5,46E-3
11 - Tavolo ufficio	4,28E-3
12 - Pavimento porta magazzino lavanderia	1,09E-2
13 - Pavimento magazzino lavanderia	6,57E-3
14 - Pavimento porta magazzino	5,37E-3
15 - Pavimento magazzino	3,79E-3
Max:1,09E-2 - Media:5,47E-3	

Esposizione

Punto di Interesse	Valore (µSv/h)
01 - Ambiente laboratorio	2,00E-1
02 - Ambiente ufficio	2,00E-1
03 - Ambiente Mag. Lavanderia	2,00E-1
04 - Ambiente magazzino	2,00E-1
05 - Armadietto sorgenti	2,80E-0
06 - Scarico lab. chimico	2,00E-1
Max: 2,80E-0 - Media: 2,00E-1	

Particolato	Valore (Bq/cc)
01 - Lab. Fisica Sanitaria e Magazzini	5,08E-8

Figure A .2 E.g. of extended routine sheet area layout and measured contamination and dose rate values for Area 1

Appendix B - Correlation Factor Calculation Methodology

The determination of discharge correlation factors was carried out according to the formalism proposed in the literature and reported in the document NUREG CR 6567 [25], as well as on the basis of the internal operational procedures adopted at the Trino site [21]. This approach is based on the statistical analysis of the ratios between the activities of difficult to measure radionuclides HTM and those of appropriate reference radionuclides.

For each analyzed sample, the *experimental correlation factor* is defined as the ratio between the activity of a generic HTM radionuclide (x) and that of the selected reference radionuclide (k). Therefore, the i-th ratio within the dataset is calculated as follows:

$$(FC)_i = \frac{a_{x,i}}{a_{k,i}}$$

where $a_{x,i}$ and $a_{k,i}$ represent the activity concentrations of the HTM and reference radionuclides, respectively, measured in the i-th sample.

Starting from these values, the representative correlation factor of the homogeneous group is determined through the calculation of *the geometric mean*:

$$A_{FC} = \left(\prod_{i=1}^{N}(FC)_i\right)^{\frac{1}{N}} = \exp\left(\frac{1}{N}\sum_{i=1}^{N} ln(FC)_i\right)$$

where N represents the number of representative samples.

The *dispersion of the data* is evaluated through the standard deviation of the logarithmic distribution of the correlation factors:

$$D_{FC} = \exp\left(\sqrt{\frac{1}{N-1}\sum_{i=1}^{N}(ln(FC)_i - ln(A_{FC})_i)^2}\right)$$

To verify the statistical consistency of the results, a criterion based on dispersion evaluated at $2\sigma = 2\ln(D_{FC})$ is considered, or equivalently by evaluating the squared dispersion factor $(D_{FC})^2$:

- $(D_{FC})^2 \leq 6$ $\quad$ $Acceptable$
- $6 < (D_{FC})^2 \leq 8$ $\quad$ $Marginal$
- $(D_{FC})^2 > 8$ $\quad$ $Unacceptable$

The validity of the correlation is therefore assessed by imposing criteria on dispersion, according to the reference guidelines [25], and applied operationally in internal procedures [24]. Only in the presence of sufficiently limited dispersion can the correlation factor be considered representative of the analyzed group of materials.

In the present work, the analysis was carried out considering radionuclides representative of different radiological mechanisms. In particular:

- Cesium 137 was used as an indicator of surface contamination
- Cobalt 60 was instead considered representative of neutron activation phenomena in metallic materials

This distinction, consistent with technical literature and operational plant experience [24][25], allows, as a first approximation, the separation of contributions due to contamination from those related to volumetric activation of components.

The results, reported in Table B.1, show that only the correlation associated with ^{63}Ni satisfies the statistical acceptability criteria, while all other radionuclides exhibit dispersion factors exceeding the threshold. It is noted that ^{241}Pu presents a value only slightly above the acceptability limit, whereas the remaining radionuclides show significantly higher $(D_{FC})^2$values and correlation acceptability for each radionuclide values, indicating limited reliability of the corresponding correlation factors.

Table B.1 $(D_{FC})^2$values and correlation acceptability for each radionuclide

Radionuclide	$(D_{FC})^2$	Acceptability Criterion
Ni 63	4.82	Acceptable
Pu 241	8.32	Slightly Unacceptable
Ni 59	9.10	Unacceptable
Am 241	11.05	Unacceptable
H 3	12.75	Unacceptable
Pu 239	12.82	Unacceptable
C 14	92.65	Unacceptable
Fe 55	94.03	Unacceptable
Sr 90	179.64	Unacceptable

Appendix C - Extended Radiological Data from RPV Head

This appendix reports the complete radiological characterization results provided by Nucleco for the samples collected from the reactor pressure vessel (RPV) head and the thermal sleeves.

Following the partial chemical dissolution of each sample, the resulting solution was first analysed by high-resolution gamma spectrometry for the direct determination of gamma-emitting radionuclides, such as Co-60, Cs-137 and Cs-134. These radionuclides can be quantified directly by gamma spectrometry because they emit characteristic gamma rays detectable by the HPGe detector. The corresponding analytical results are reported in Table C.1, which summarizes the activity measured for the gamma-emitting radionuclides.

The remaining radionuclides, including H-3, C-14, Ni-59, Ni-63, Fe-55, Sr-90, Pu isotopes, Am-241 and Cm isotopes, cannot be determined directly by gamma spectrometry because they are pure beta emitters, alpha emitters, or low-energy X-ray emitters. Consequently, aliquots of the same dissolved solution were subjected to dedicated radiochemical separation procedures followed by the appropriate measurement techniques, including liquid scintillation counting, alpha spectrometry or X-ray spectrometry, depending on the radionuclide of interest. The results obtained from these radiochemical analyses are presented in Table C.2.

For each radionuclide, the tables report the measured activity, its associated uncertainty, the Critical Level (Lc) and the Minimum Detectable Concentration (MDC). The reported activity represents the measured radionuclide concentration, expressed either in Bq/g for activation analyses or in Bq/cm^2 for surface contamination analyses. The associated uncertainty corresponds to the combined measurement uncertainty, mainly arising from counting statistics, detector efficiency calibration, sample mass or surface determination. The Lc represents the decision threshold above which the measured signal can be statistically distinguished from the background, whereas the MDC represents the lowest activity concentration that can be reliably detected by the analytical method under the adopted measurement conditions.

The appendix (Table C.1, Table C.2) includes the complete analytical results for the four RPV head sampling locations and the two thermal sleeve samples belonging to the upper internals, providing the full radiological characterization dataset used for the discussion presented in Chapter 6. For clarity and to improve the readability of the tables, the sample identification is reported only in the first row of each group, while the following rows refer to the same sample and list the results for the different analysed radionuclides.

Table C.1 Gamma-emitting radionuclides in RPV head and thermal sleeve samples

ID CAMPIONI SOGIN	ID CAMPIONI NUCLECO	MATRICE	quantità disciolta per la spettrometria gamma	Unità di misura
ITR.001.25-VE-Head 1-contaminazione	TR25C00230	Metallo	13.85 cm^2/60.57 g	Bq/cm^2
ITR.001.25-VE-Head 1-attivazione	TR25C00231	Metallo	42.31 g	Bq/g
ITR.066.24-VE-Head 2-contaminazione	TR25C00232	Metallo	12.56 cm^2/36.82 g	Bq/cm^2
ITR.066.24-VE-Head 2-attivazione	TR25C00233	Metallo	46.95 g	Bq/g
ITR.065.24-VE-Head 3-contaminazione	TR25C00234	Metallo	13.85 cm^2/29.62 g	Bq/cm^2
ITR.065.24-VE-Head 3-attivazione	TR25C00235	Metallo	94.87 g	Bq/g
ITR.069.24-VE-Head 4-contaminazione	TR25C00236	Metallo	13.85 cm^2/41.74 g	Bq/cm^2
ITR.069.24-VE-Head 4-attivazione	TR25C00237	Metallo	41.85 g	Bq/g
ITR.063.24-UP-PIP-10-Thermal sleeve 1	TR25C00238	Metallo	140.69 g	Bq/g
ITR.067.24-UP-PIP-11-Thermal sleeve 2	TR25C00239	Metallo	47.13 g	Bq/g

Co-60				Cs-137				Cs-134			
Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.
3.76E+02	4.37E+01	3.03E-01	1.50E-01	3.12E-01	1.40E-01	2.71E-01	1.34E-01	<Lc		1.50E-01	7.43E-02
1.36E+01	4.29E+00	2.59E-02	1.20E-02	-	-		-	-	-		-
4.40E+02	6.53E+01	3.54E-01	8.67E-02	2.37E-01	1.11E-01	1.76E-01	8.67E-02	<Lc		1.10E-01	5.40E-02
2.65E+00	8.18E-01	1.68E-02	7.79E-03	-	-	-	-	-	-		-
1.89E+02	2.20E+01	8.94E-01	4.43E-01	2.28E-01	1.04E-01	1.78E-01	8.82E-02	<Lc		1.63E-01	8.06E-02
4.31E+00	1.16E+00	1.77E-02	8.37E-03	-	-	-	-	-	-		-
1.98E+02	2.94E+01	1.93E-01	9.50E-02	2.07E-01	7.04E-02	1.05E-01	5.18E-02	<Lc		7.41E-02	3.64E-02
1.26E+01	5.26E+00	2.94E-02	1.29E-02	-	-	-	-	-	-		-
3.94E+02	6.25E+01	3.10E-01	1.52E-01	4.82E-01	1.24E-01	1.96E-01	9.62E-02	<Lc		1.71E-01	8.40E-02
2.07E+02	4.54E+01	4.76E-01	2.30E-01	2.17E+00	4.97E-01	3.08E-01	1.49E-01	<MDC		1.34E-01	6.44E-02

Table C.2 Radionuclides determined by radiochemical analysis in RPV head and thermal sleeve samples.

ID CAMPIONI SOGIN	ID CAMPIONI NUCLECO	MATRICE	Unità di misura	Pu-241			
				Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.
ITR.001.25-VE-Head 1-contaminazione	TR25C00230	Metallo	Bq/cm^2	4.30E+00	1.26E+00	8.50E-01	4.13E-01
ITR.001.25-VE-Head 1-attivazione	TR25C00231	Metallo	Bq/g	-	-	-	-
ITR.066.24-VE-Head 2-contaminazione	TR25C00232	Metallo	Bq/cm^2	1.06E+01	1.47E+00	8.37E-01	4.06E-01
ITR.066.24-VE-Head 2-attivazione	TR25C00233	Metallo	Bq/g	-	-	-	-
ITR.065.24-VE-Head 3-contaminazione	TR25C00234	Metallo	Bq/cm^2	3.42E+00	1.58E+00	1.91E+00	9.29E-01
ITR.065.24-VE-Head 3-attivazione	TR25C00235	Metallo	Bq/g	-	-	-	-
ITR.069.24-VE-Head 4-contaminazione	TR25C00236	Metallo	Bq/cm^2	1.03E+01	1.28E+00	6.98E-01	3.39E-01
ITR.069.24-VE-Head 4-attivazione	TR25C00237	Metallo	Bq/g	-	-	-	-
ITR.063.24-UP-PIP-10-Thermal sleeve 1	TR25C00238	Metallo	Bq/g	7.16E+00	6.07E-01	1.91E-01	9.28E-02
ITR.067.24-UP-PIP-11-Thermal sleeve 2	TR25C00239	Metallo	Bq/g	2.16E+01	4.00E+00	3.74E+00	1.81E+00

Pu-238				Pu-239			Pu-240		
Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.
<Lc		3.32E-02	1.01E-02	2.73E-02	1.18E-02	1.01E-02	5.18E-02	4.27E-02	1.01E-02
-	-	-	-	-	-	-	-	-	-
2.06E-01	1.61E-01	2.93E-02	8.94E-03	4.85E-02	1.43E-02	2.93E-02	1.10E-01	5.50E-02	2.93E-02
-	-	-	-	-	-	-	-	-	-
7.42E-02	5.92E-02	1.24E-02	3.77E-03	2.18E-02	6.16E-03	1.24E-02	5.60E-02	2.31E-02	1.24E-02
-	-	-	-	-	-	-	-	-	-
1.16E-01	5.30E-02	1.37E-02	4.19E-03	3.40E-02	8.23E-03	1.37E-02	6.89E-02	3.24E-02	1.37E-02
-	-	-	-	-	-	-	-	-	-
6.64E-01	5.58E-02	1.87E-03	5.69E-04	6.40E-02	6.23E-03	1.87E-03	2.38E-01	2.27E-02	1.87E-03
1.55E+00	1.81E-01	1.51E-02	4.61E-03	1.29E-01	1.88E-02	1.51E-02	4.06E-01	6.41E-02	1.51E-02

Table C.2 (continued)

Mn-54				Ag-108m				Sb-125			
Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.
<MDC		2.11E-01	1.04E-01	5.45E-01	1.10E-01	2.71E-01	1.34E-01	<Lc		3.45E-01	1.71E-01
-	-		-	2.45E-02	9.64E-03	1.63E-02	7.55E-03	<Lc		2.59E-02	1.20E-02
<Lc		1.66E-01	8.18E-02	5.59E-01	1.07E-01	1.70E-01	8.36E-02	<Lc		2.42E-01	1.19E-01
-	-		-	1.06E-02	5.62E-03	1.08E-02	5.00E-03	<Lc		1.66E-02	7.70E-03
<Lc		1.77E-01	8.75E-02	3.91E-01	7.51E-02	1.79E-01	8.86E-02	<Lc		2.76E-01	1.36E-01
-	-		-	1.78E-02	8.74E-03	1.44E-02	6.84E-03	<Lc		1.74E-02	8.21E-03
<Lc		1.02E-01	5.04E-02	5.14E-01	8.43E-02	9.01E-02	4.43E-02	<Lc		1.63E-01	8.02E-02
-	-		-	5.33E-02	2.30E-02	1.60E-02	7.00E-03	<Lc		2.67E-02	1.17E-02
<Lc		1.73E-01	8.47E-02	8.98E-01	1.60E-01	1.29E-01	6.32E-02	<Lc		2.46E-01	1.21E-01
<Lc		1.67E-01	8.07E-02	1.67E+00	3.82E-01	2.05E-01	9.92E-02	<Lc		4.26E-01	2.06E-01

Eu-152				Eu-154				Nb-94			
Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.
7.31E-01	1.50E-01	4.00E-01	1.97E-01	2.62E-01	8.59E-02	1.43E-01	7.05E-02	<MDC		3.41E-01	1.69E-01
<MDC		2.89E-02	1.35E-02	<Lc		1.48E-02	6.86E-03	<MDC		2.04E-02	9.50E-03
7.81E-01	1.50E-01	2.16E-01	1.06E-01	2.67E-01	7.95E-02	1.32E-01	6.49E-02	2.07E-01	7.33E-02	1.74E-01	8.57E-02
<Lc		1.27E-02	5.90E-03	<Lc		8.84E-03	4.10E-03	<Lc		7.52E-03	3.48E-03
<MDC		3.14E-01	1.55E-01	<MDC		1.35E-01	6.67E-02	<Lc		1.44E-01	7.09E-02
<Lc		9.08E-03	4.29E-03	<Lc		6.34E-03	3.00E-03	<MDC		7.87E-03	3.72E-03
3.41E-01	1.17E-01	2.09E-01	1.03E-01	1.05E-01	4.70E-02	9.29E-02	4.57E-02	<MDC		1.32E-01	6.47E-02
<Lc		1.65E-02	7.25E-03	<Lc		1.15E-02	5.05E-03	<Lc		1.42E-02	6.22E-03
7.61E-01	1.44E-01	2.41E-01	1.18E-01	2.55E-01	6.71E-02	1.08E-01	5.30E-02	2.56E-01	9.52E-02	2.19E-01	1.08E-01
4.95E-01	1.62E-01	3.67E-01	1.77E-01	3.08E-01	9.62E-02	1.28E-01	6.17E-02	3.88E-01	1.74E-01	3.87E-01	1.87E-01

Fe-55				Ni-59+63				Ni-59			
Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.
<MDC		4.59E-01	2.22E-01	3.03E+02	1.51E+01	4.90E-01	2.73E-01	1.34E+00	6.74E-01	1.27E+00	6.23E-01
<MDC		9.31E-02	4.50E-02	5.44E+01	2.63E+00	1.96E-01	9.45E-02	<Lc		3.64E-01	1.80E-01
<MDC		2.36E-01	1.14E-01	1.18E+03	5.56E+01	3.76E-01	1.82E-01	1.09E+00	1.98E-01	3.54E-01	1.75E-01
<MDC		7.74E-02	3.74E-02	3.16E+01	1.63E+00	8.35E-01	4.04E-01	<Lc		4.72E-01	2.33E-01
<MDC		7.29E-02	3.52E-02	3.87E+02	1.44E+01	2.19E-01	1.06E-01	8.65E-01	3.10E-01	5.89E-01	2.92E-01
<MDC		8.35E-02	4.03E-02	2.12E+01	8.49E-01	1.54E-01	7.44E-02	<Lc		2.97E-01	1.47E-01
<MDC		3.46E-01	1.67E-01	1.09E+03	4.24E+01	5.50E-01	2.66E-01	2.17E+00	9.99E-01	1.74E+00	8.53E-01
<MDC		8.16E-02	3.94E-02	6.50E+01	3.90E+00	2.33E+00	1.12E+00	<Lc		6.83E-01	3.39E-01
<MDC		1.56E-01	7.56E-02	5.36E+02	1.93E+01	1.32E-01	6.40E-02	1.45E+00	2.69E-01	4.70E-01	2.31E-01
<MDC		2.62E-01	1.27E-01	2.12E+03	7.64E+01	3.15E-01	1.52E-01	2.13E+00	3.78E-01	6.68E-01	3.30E-01

Sr-90				H-3				C-14			
Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.
<Lc		7.41E-02	3.36E-02	<Lc		4.94E-02	2.35E-02	2.01E+01	6.60E-01	3.43E-02	1.69E-02
-	-		-	-	-		-	-	-	-	-
<Lc		3.09E-01	1.51E-01	<Lc		8.10E-02	3.86E-02	2.17E+01	7.14E-01	3.72E-02	1.83E-02
-	-	-	-	-	-		-	-	-	-	-
<Lc		9.26E-02	4.53E-02	<Lc		4.89E-02	2.33E-02	2.00E+01	6.57E-01	3.42E-02	1.68E-02
-	-		-	-	-		-	-	-	-	-
<MDC		8.95E-02	4.39E-02	<Lc		4.93E-02	2.35E-02	2.02E+01	6.64E-01	3.43E-02	1.69E-02
-	-	-	-	-	-		-	-	-	-	-
<MDC		1.12E-01	5.10E-02	<Lc		1.64E-02	7.79E-02	2.56E+00	8.56E-02	7.15E-03	3.52E-03
<Lc		3.36E-01	1.65E-01	<Lc		1.18E-02	5.63E-03	1.20E-02	2.92E-03	5.60E-03	2.75E-03

Am-241				Cm-244				Cm-242			
Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.	Attività [Bq/cm^2]cont. [Bq/g] att.	Incertezza [Bq/cm^2]cont. [Bq/g] att.	MDC [Bq/cm^2]cont. [Bq/g] att.	Lc [Bq/cm^2]cont. [Bq/g] att.
3.59E-01	6.79E-02	2.10E-02	6.39E-03	<MDC		2.10E-02	6.39E-03	<Lc		2.10E-02	6.39E-03
-	-	-	-	-	-	-	-	-	-	-	-
7.66E-01	1.92E-01	1.22E-01	4.64E-02	8.96E-02	6.64E-02	7.62E-02	2.32E-02	<Lc		7.62E-02	2.32E-02
-	-	-	-	-	-	-	-	-	-	-	-
3.72E-01	6.26E-02	1.72E-02	5.23E-03	3.81E-02	1.97E-02	1.72E-02	5.23E-03	<Lc		1.72E-02	5.23E-03
-	-	-	-	-	-	-	-	-	-	-	-
5.56E-01	1.11E-01	3.49E-02	1.06E-02	5.01E-02	3.31E-02	3.49E-02	1.06E-02	<Lc		3.49E-02	1.06E-02
-	-	-	-	-	-	-	-	-	-	-	-
7.81E-01	6.48E-02	6.04E-03	2.29E-03	9.37E-02	1.50E-02	3.76E-03	1.15E-03	<Lc		3.76E-03	1.15E-03
2.73E+00	2.10E-01	8.78E-03	2.68E-03	5.62E-01	6.26E-02	8.78E-03	2.68E-03	<Lc		8.78E-03	2.68E-03

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