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Radiological characterization for future fusion reactor decommissioning: balancing accuracy, time, and radiation protection

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Abstract Future fusion reactors will produce radioactive waste, primarily neutron activation products and tritium, necessitating robust decommissioning strategies. This study compares radiological characterization techniques for fusion waste management, focusing on destructive versus non-destructive methods, and evaluates the advantages and disadvantages of both by applying them to two case studies of nuclear fusion plant decommissioning, namely the ENEA Frascati Tokamak Upgrade (FTU) and the planned ITER, the International Thermonuclear Experimental Reactor. Gamma emitters can be readily quantified using solid-state detectors, while almost pure beta emitters like Fe-55 and Ni-59 traditionally require destructive methods. However, for ITER-scale inventories, Cadmium Telluride (CdTe)-based X-ray spectrometry offers a viable non-destructive alternative, combining good energy resolution, compactness, and potential reduced secondary waste. In contrast, the FTU decommissioning, which is a peculiar case due to its stringent clearance levels for material reuse and recycling (below 1E+03 Bq/kg), necessitates destructive methods for nuclides such as Fe-55 and Ni-59 because non-destructive X-ray spectrometry would require impractically long measurement times. This comparative analysis highlights the importance of tailored characterization approaches, balancing accuracy, efficiency, and radiation protection, and provides insights applicable to decommissioning in different nuclear facilities.

1 Introduction

The operation and eventual decommissioning of future nuclear fusion reactors will generate radioactive waste. However, this waste stream will significantly differ from that of fission reactors. Specifically, it is expected to be characterized by a predominance of activation products and tritium, while lacking transuranic elements and fission products. This compositional difference effectively eliminates proliferation risks associated with transuranic isotopes and simplifies long-term waste management compared to fission waste [1]. The absence of spent fuel production in fusion reactors during operation and decommissioning represents a significant advantage over fission technology. Fusion waste, therefore, will not present the challenges associated with high heat generation and the accumulation of long-lived alpha-emitting nuclides, which are associated with spent fission fuels [2]. With the implementation of effective refurbishment strategies for liquid tritium breeding materials (such as LiPb, Li, and FLiBe) and the application of robust detritiation techniques for contaminated components, the dominant source of radioactive inventory in fusion reactors will be activation products and residual tritium. The presence of activation products is primarily due to neutron-induced activation of solid metallic materials in the main machine components and of the concrete in the biological shield, resulting from exposure to the intense neutron flux emitted by the reactor plasma. Importantly, the extent and nature of activation in both metals and concrete are strongly influenced by the presence of impurities, even at trace concentrations. These impurities can give rise to long-lived radionuclides with significant radiological impact, thus playing a critical role in the design of decommissioning strategies and waste classification [3–5]. The intense neutron flux experienced by plasma-facing components results in significant specific activities at the end of irradiation [2]. Due to activation, structural components like the blanket and the divertor will require replacement during the operational lifetime of future fusion power plants [3–5]. The decommissioning process will subsequently produce radioactive waste necessitating treatment and conditioning, which will, in turn, generate secondary waste products [3]. Recognizing the complexities of radioactive waste management as evidenced by fission reactor experiences, a well-defined and effective approach to handling radioactive materials and waste from fusion facilities will be a decisive factor in their siting, licensing, and decommissioning. The global pursuit of underground or near-surface repositories for fission waste has been hindered by significant public resistance, technical hurdles related to safety and environmental protection, and concerns regarding the long-term burden placed on future generations. In response, the fusion community is prioritizing waste minimization and recycling initiatives to minimize the necessity for deep geological disposal, thereby reducing the economic and logistical demands of fusion energy deployment [6]. A comprehensive strategy for minimizing

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radioactive waste in fusion reactors requires the integration of several key elements. These include: the selection of an optimized fuel cycle, the precise choice of structural materials, the implementation of well-defined maintenance and dismantling procedures, the establishment of effective industrial-scale recycling processes for activated materials, the facilitation of regulatory clearance for materials with very low radioactivity, and the development of robust techniques for treating contaminated materials, with a particular emphasis on detritiation [6, 7]. Minimizing the reliance on geological repositories and optimizing the sustainability of fusion waste necessitates a lifecycle integration approach, encompassing both operational and decommissioning phases. Central to this strategy is the acquisition of detailed qualitative and quantitative radionuclide data across all materials and components. This mandates the implementation of advanced radiological characterization techniques, encompassing measurement protocols, methodologies, and predictive modeling. These analytical tools must be carefully engineered to achieve a balance between high-resolution radionuclide analysis for specific isotopes and the rapid assessment of large waste streams. All analytical processes must adhere to stringent radiological safety standards, reflect cost considerations for plant maintenance and decommissioning, and comply with the ALARA (i.e., As Low As Reasonably Achievable) principle.

Within the context of fusion waste management, radionuclides such as Mn-54, Fe-55, Ni-59, and Co-60 are of critical importance due to their substantial influence on long-term waste disposal strategies. Consequently, these nuclides have been extensively investigated through activation calculations [8]. Unlike readily quantifiable gamma emitters like Mn-54 and Co-60, which can be analyzed using non-destructive techniques that avoid material sampling and physical or chemical processing, such as conventional gamma spectrometry, Hard-to-Measure (HTM) radionuclides like Fe-55 and Ni-59 present significant challenges due to their limited gamma emissions. These HTM nuclides, specifically Fe-55 and Ni-59, which decay via electron capture, typically require destructive analysis using Liquid Scintillation Counting and radiochemical separation followed by low-energy X-ray spectrometry, respectively [9]. Ensuring representative sampling of the original material is pivotal in destructive characterization, as it directly impacts the accuracy of subsequent analyses. However, the large volume of fusion waste expected over a reactor lifetime would benefit from the adoption of non-destructive characterization techniques, since they offer generally faster analysis times, reduced secondary waste generation, and the elimination of potential biases introduced by inadequate sampling. While beta-induced X-ray spectrometry has been successfully applied to tritium detection in fusion materials [10], such an approach was not designed to provide information about the activated materials occurrence. To address the challenges of characterizing activation products in fusion waste, non-destructive methodologies developed for waste streams from fission reactors, accelerators, and industrial and medical applications can be leveraged [11–13].

This paper presents a comparative study of decommissioning strategies for fusion facilities, using the ENEA Frascati Tokamak Upgrade (FTU) as a real case study and comparing it with the planned decommissioning of the future ITER facility. FTU is a compact, high-field tokamak located at the ENEA research center in Frascati, Italy, which operated for over two decades and has undergone decommissioning with a focus on material reuse and recycling due to its relatively lower activation levels [14]. ITER is a large-scale international experimental tokamak currently under construction in France, designed to demonstrate the long-pulse operation of a fusion power plant producing significant net power and expected to generate substantial levels of neutron-induced activation in its in-vessel components and surrounding structures [15]. The most relevant activation products identified for both FTU and ITER scenarios are summarized in Table 1, highlighting key radionuclides, their half-lives, emission characteristics, and activation reactions.

Table 1 Most occurring photon-emitting radionuclides from neutron activation in fusion reactors (adapted from [16])

Radionuclide	X-ray energies (keV) [probabilities]	Gamma energies (keV) [probabilities]	Half-life	Neutron activation reaction
^{51}Cr	4.95 [0.20], 5.95 [0.03]	320.08 [0.10]	27.7 days	^{50}Cr(n,γ)^{51}Cr, ^{54}Fe(n,α)^{51}Cr
^{54}Mn	5.41 [0.23], 5.97 [0.03]	834.85 [0.99]	312 days	^{54}Fe(n,p)^{54}Mn
^{55}Fe	5.89 [0.25], 6.51 [0.03]	–	2.7 years	^{54}Fe(n,γ)^{55}Fe, ^{56}Fe(n,2n)^{55}Fe, ^{58}Ni(n,α)^{55}Fe
^{59}Ni	6.91 [0.30], 7.68 [0.04]	–	76,000 years	^{58}Ni(n,γ)^{59}Ni, ^{60}Ni(n,2n)^{59}Ni
^{60}Co	–	1173.29 [0.99], 1332.49 [0.99]	5.27 years	^{59}Co(n,γ)^{60}Co
^{93}Mo	16.55 [0.61], 18.67 [0.10]	–	4,000 years	^{92}Mo(n,γ)^{93}Mo, ^{94}Mo(n,2n)^{93}Mo
^{94}Nb	–	702.62 [0.98], 871.10 [0.99]	20,300 years	^{93}Nb(n,γ)^{94}Nb
^{181}W	6.24 [0.01], 9.74 [0.22]	56.90 [0.51], 66.18 [0.14]	121.2 days	^{180}W(n,γ)^{181}W, ^{182}W(n,2n)^{181}W

This study is relevant as it investigates the limits of non-destructive characterization techniques (NDT), such as semiconductor-based methods including High-Purity Germanium (HPGe) detectors for gamma spectrometry and Cadmium Telluride (CdTe) detectors for X-ray spectrometry to analyze X-rays associated with Auger electron emission, for potentially replacing destructive methods in radioactive waste management within fusion facilities. A central focus is the optimization of accuracy, efficiency, and radiation protection in fusion waste management. Significantly, the results and methodologies discussed are also relevant to decommissioning challenges in diverse contexts, including accelerator and fission facilities.

2 Materials and methods

A robust characterization protocol for activated solid fusion waste must incorporate techniques for efficient activity distribution mapping across large surfaces. HTM radionuclides, characterized by low or absent gamma emissions, including beta emitters and electron capture nuclides, present significant analytical challenges. While their charged particle emissions (electrons or positrons) offer a potential detection pathway, the short mean free path of these particles necessitates close coupling between the detector and the radiation source. Attenuation in surrounding media, including air and detector components, can prevent particle detection if this proximity is not maintained. The primary analytical techniques employed for the identification and quantification of these nuclides are Liquid Scintillation Counting and mass spectrometry [9 for a comprehensive description of such techniques]. Liquid Scintillation Counting is a radiological technique that indirectly detects charged particles emitted during radioactive decay, whereas mass spectrometry identifies atomic masses of nuclides within a sample, extending its applicability beyond radiological characterization. These analytical techniques, Liquid Scintillation Counting and mass spectrometry, are inherently destructive, necessitating the complete dissolution of the original sample through chemical processes. The resulting solution is then incorporated into a liquid matrix for measurement. A key advantage of these destructive methods is their ability to potentially achieve very low detection limits, commonly below 1 Bq/kg for radiological activity in practical laboratory settings. Achieving these detection levels, however, is time-intensive, typically requiring multiple days, with a significant portion of that time devoted to sample preparation. Moreover, destructive characterization techniques (DT) analyze only a small portion of the material. Given that radiological conditions within a nuclear facility are often highly variable, a well-planned sampling strategy is therefore essential. This strategy must ensure that samples selected for measurement accurately reflect the heterogeneity of radiological conditions within each plant component or device, to provide a reliable overall picture. Effective sampling strategies demand a thorough characterization of the fusion plant, encompassing material composition, spatial distribution, neutron irradiation history, and the development of activation distribution models. Additionally, the quantification and propagation of uncertainties associated with sample inhomogeneity are essential components of the overall measurement uncertainty analysis.

The detection of gamma-emitting radionuclides is efficiently accomplished using solid-state semiconductor detectors. Gamma photons, emitted from the activated material, exhibit low interaction probabilities with air, enabling them to reach the detector sensitive volume with minimal attenuation. The operational principle of a semiconductor detector under reverse bias relies on the formation of an electric field within the depletion region. Photon interactions within this region lead to the creation of charge carriers, which are then efficiently transported to the electrodes by the strong electric field, enabling signal detection. The quantity of charge produced, which is directly proportional to the energy deposited by the incident photon within the detector, is first processed by a charge-sensitive preamplifier—typically located close to the detector to minimize electronic noise—which converts the charge signal into a voltage pulse. This pulse is subsequently shaped by a shaping amplifier to enhance the signal-to-noise ratio and define the temporal characteristics of the pulse. The shaped signal is then digitized by an analog-to-digital converter (ADC) and analyzed by a multichannel analyzer (MCA) or digital signal processor (DSP), which builds the energy spectrum by sorting the events according to their pulse amplitude. Each component of this electronic chain plays a critical role in preserving energy resolution and ensuring accurate photon detection. To mitigate thermal generation of charge carriers and maintain acceptable performance, these detectors require cooling. Elevated temperatures result in increased leakage current noise, which negatively affects the energy resolution of the detector [17 and references therein]. The higher atomic number of Germanium ($Z = 32$) compared to other semiconductor materials, in particular Silicon ($Z = 14$), results in increased gamma photon interaction probability. This translates to greater detection efficiency, provided the detector possesses a depletion volume of sufficient size to facilitate photon interactions within the crystal. The fabrication of Germanium crystals with minimal impurity concentrations ($< 10^{10}$ atoms/cm^3) is essential for achieving large depletion volumes. These high-purity crystals, referred to as HPGe detectors, are produced through controlled crystal growth processes, enabling the formation of detectors with varied geometries and dimensions. The depletion volume typically ranges from a few cubic centimeters in small planar detectors to several hundred cubic centimeters (up to ~ 500 cm^3 or more) in large coaxial configurations [17].

Semiconductor detectors utilizing CdTe leverage the combined high atomic numbers of Cadmium ($Z = 48$) and Tellurium ($Z = 52$), coupled with a sufficiently large band gap energy, to facilitate operation at near-ambient temperatures. For X-ray energies, CdTe demonstrates a substantially increased photon interaction probability per unit path length relative to Germanium detectors [17]. Due to this enhanced sensitivity, CdTe detectors have been proposed for the non-destructive detection of low-energy characteristic X-rays emitted during the electron capture decay of radionuclides such as Fe-55 and Ni-59—nuclides traditionally quantified using destructive techniques [16]. For Fe-55, the dominant X-ray emissions arise from the K-shell transitions of the daughter Mn-55,

with K-α lines at approximately 5.90 keV and a combined intensity of ~ 25%, as well as a K-β emission at ~ 6.49 keV with an intensity of ~ 3.4% per 100 decays. In the case of Ni-59, electron capture decay leads to the formation of stable Co-59, during which characteristic K-α X-rays from Cobalt are emitted at approximately 6.915 keV and 6.930 keV, with an intensity of about 10% and 20% per 100 decays, respectively. These emissions fall well within the detection range of CdTe detectors, offering a viable non-destructive alternative to radiochemical separation followed by Liquid Scintillation Counting.

Although low-energy or ultra-low-energy HPGe detectors are also capable of detecting X-rays in this energy range, CdTe detectors present several practical advantages. These include superior photon absorption efficiency at low energies due to higher atomic numbers, compact design with reduced detector thickness, and the ability to operate without cryogenic cooling. Even if HPGe detectors offer better energy resolution overall, CdTe detectors still provide adequate resolution in the 5–7 keV range for many applications, with the added benefit of simplified system integration. As such, CdTe represents a practical and effective solution for in situ or field-based measurements of hard-to-measure radionuclides in fusion waste.

For the following considerations, this work utilizes two distinct and commercially accessible semiconductor detectors with specific characteristics suited for the energy ranges of interest. The first detector is a p-type coaxial HPGe detector with a radius of 25.2 mm. This configuration is often deployed for in situ measurements, utilizing portable systems that incorporate shielding collimators to limit the detector angular acceptance. The specified HPGe detector enables gamma spectrometry within the approximate photon energy interval of 50 keV–6 MeV. The second detector is a CdTe detector with a radius of 2.82 mm, protected by a 100 μm thick beryllium window. Due to its Peltier cell cooling mechanism, the CdTe detector is considerably lighter and more compact than the liquid nitrogen-cooled HPGe detector. This compactness renders the CdTe detector particularly advantageous for in situ sampling of various reactor component locations. The described CdTe detector is optimized for spectrometry within the approximate photon energy range of 5 keV–60 keV.

A model developed by Marzo et al. [16] is then considered for evaluating the measurement time necessary to attain statistically significant quantification limits [18] for specific radionuclides, as a function of their activity concentration, when employing HPGe or CdTe detectors for gamma or X-ray photon detection, respectively. The model accounts for various parameters, including detector response function, sample self-absorption effects, and measurement geometry.

Currently, the most efficient method to simulate radiation transport in media through either complex (many volumes or "cells," several radioactive sources, and different materials) and large (few centimeters to hundreds of meters) geometries is the Monte Carlo simulation. This approach involves running a lot of 'histories' of the particle of interest in order to reproduce all the possible paths traveled by the radiation in a virtual world. The process is realized by means of random samplings from all the probability density functions ruling physical phenomena. For this work, the MCNPX (Monte Carlo N-Particle eXtended) code has been used, in its version MCNPX 2.5.0 [19]. Dose rates are evaluated in the simulation using point detectors positioned at locations of interest (namely the MCNP F5 tally type detector). The photon flux spectra obtained are then multiplied by the Effective Dose Conversion Coefficients from ICRP-116 [20], yielding the energy-integrated Effective Dose rates.

3 Radiation protection considerations

When considering the adoption of DT versus NDT characterization methods, radiation protection evaluations, in accordance with the ALARA principle, are essential, also with respect to the collective dose [21] that can still pose significant challenges for decommissioning of nuclear fusion plants.

Destructive techniques may involve extended periods of manual manipulation, leading to potentially significant radiation doses to the operators, depending on factors such as the number of samples, the quantity of radioactive material involved, and the chemical processing time. On the other hand, non-destructive techniques enable remote measurements of large material volumes, potentially reducing occupational exposure. Compared to DT, they require shorter time spans in proximity to the radiation source and inherently avoid the handling of radioactive materials and associated contamination risks. However, some proximity may still be necessary, for example, to correctly position the detector relative to the source, thereby requiring that the duration of close exposure must be minimized. In such a sense, performing remote, NDT using HPGe or CdTe detectors, with careful consideration of their positioning and orientation, might be a viable solution to improve radiation protection in high radiation scenarios.

The radiation protection challenges inherent in DT and NDT methods can be further illustrated through a semi-quantitative analysis for the following case studies referred to ITER.

3.1 Case 1—destructive radiological characterization of a 50 mg ITER activated steel sample

In the case of a destructive radiological characterization of a 50 mg sample of activated steel from ITER, which contains both Fe-55 and Co-60, an effective dose to the operator of approximately 3 mSv from external exposure alone over an 8 h work shift is estimated. This occurs when considering a typical procedure for chromatographic isolation of chemical elements [22], a method often employed to perform separated Liquid Scintillation Counting of the extracted fraction of Fe-55. This significant dose is primarily attributed to the presence of Co-60, which dominates the external dose in this sample, despite having a lower activity concentration ($1.06 \cdot 10^{10}$ Bq/kg according to [15]) compared to Fe-55 ($3.54 \cdot 10^{12}$ Bq/kg [16]). This estimation considers an exposure scenario in which the

operator spends 40% of the time near a point source at a distance of 30 cm, 40% near a 10 mL vial at 100 cm, 10% handling a 50 mL beaker in direct contact with the source, and 10% in contact with a 5 mL syringe, using the external exposure coefficients from [20]. Considering that the characterization of a single point may require several samples (at least 3–5) to ensure representativeness, an operator working an 8 h shift and processing 10 samples at the same time could receive an estimated dose of 30 mSv in a single day. It is important to highlight again that the previous dose estimate accounts only for external exposure and does not include the potential internal contamination that may occur during the execution of such procedures.

Furthermore, performing DT characterization often requires samples to be collected directly from the field. For instance, in the previous example, metal chips are typically obtained from large radioactive components using low-speed drilling techniques, with the operator working in close proximity to the source. For a 200 cm × 200 cm × 1 cm activated steel component (approximately 320 kg) from ITER, with previously reported activity concentrations [16], the effective dose from external exposure to an operator working for 15 min at a distance of 1 cm to perform drilling and collect metal chips is estimated to be approximately 750 mSv by Monte Carlo calculations. Since Co-60 is the major contributor to the external irradiation of the operator, only this radionuclide has been considered in the simulation, and it was assumed to be homogeneously dispersed in the metal matrix with the stated activity concentration. A volumetric gamma source was modeled, with the simulation run of 10^{10} nps (number of histories run in the code), and statistical errors maintained below 0.04% for MNCP F5 tally type detectors. Also in this case, the dose estimate accounts only for external exposure and does not include the potential internal contamination—particularly from potential gaseous species such as tritium—that may occur during the drilling procedures.

3.2 Case 2—non-destructive radiological characterization of a massive ITER activated steel component

For the same massive component as before, 200 cm × 200 cm × 1 cm activated steel component (approximately 320 kg) from ITER, with previously reported activity concentrations [16], the effective dose from external exposure to an operator working for 15 min at a distance of 100 cm to position the HPGe detector for characterization is estimated at approximately 160 mSv. This evaluation utilized the external exposure coefficients from [23], modified with a point kernel calculation to account for the extended planar radioactive source. Given that Co-60 provides the major contribution to the dose, the self-attenuation of its 1.17 and 1.33 MeV photons within the 1 cm thickness of the slab was not considered. Internal contamination in this case can be neglected due to the preserved integrity of the component.

When comparing NDT versus DT characterization methodologies in the examples provided, the massive exposure of the operators to carry out the latter techniques is quite evident, especially regarding the sampling stage to produce metal chips for the chemical and radiological analyses. Reducing or removing such contribution to the exposure would require devices for remote sampling. However, the inherently closer interactions often required, such as drilling operations on metal components, pose substantial engineering hurdles for remote platforms in terms of design, weight, robustness, force application, and even monitoring and controlling the execution of the operations via remote media. Moreover, DT carry the significant additional risk of potential internal contamination for working personnel encompassing the entire process of sampling, chemical processing, and subsequent analyses, leading to higher exposures with respect to NDT.

In the case of NDT radiological characterization of a massive ITER activated steel component, the effective dose to the operator for 15 min work during the placement of the HPGe detector at a distance of 100 cm is even quite high (about 160 mSv in 15 min), but there is no contribution other than such external exposure, as the integrity of the component is preserved. Such contribution can be effectively minimized through the introduction of remote handling, allowing the operator to control the remote platform from a safer distance, and inherently avoiding internal contamination. Building a remote platform for an HPGe in-field positioning and acquisition is quite easier than building a remote platform to perform in-field drilling and sample collection: attention has to be paid to the correct positioning of the device that is generally at a distance of 100 cm or higher from the component as to keep the entire object in the field of view of the collimation selected for the detector.

Given the occupational exposure limits of 20 mSv per year (averaged over 5 years) and a maximum of 50 mSv in any single year [24], the advantages of non-destructive methodologies for minimizing personnel exposure in radioactive waste management are apparent. While the reported evaluation highlights a specific scenario with a potential external dose for non-destructive characterization, the inherent ability to perform remote measurements allows for significant dose rate reductions by maintaining or increasing distance from the radioactive source. Crucially, these methods also eliminate the risk of internal contamination associated with destructive techniques, where external doses around 30 mSv are estimated alongside the potential for internal hazards. This aligns with the ALARA principle by effectively minimizing both direct and secondary radiation hazards.

However, DT still remain necessary for some specific cases, e.g., the characterization of tritium or other pure beta emitters radionuclides. The key point is that NDT methodologies should be used more and more extensively, fully leveraging their true capabilities in radiological characterization. These techniques align with the ALARA principle and, when appropriate scaling factors are available for the waste stream under analysis, may also allow for the determination of HTM radionuclides.

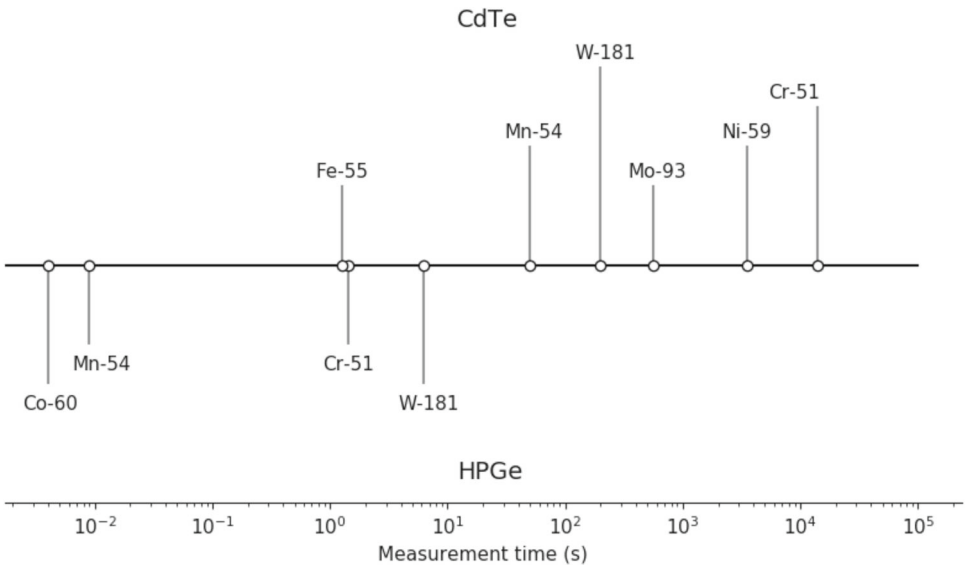

Fig. 1 Timeline summarizing the measurement time necessary to collect a statistically significant signal corresponding to the quantification limit for the significant radionuclides expected in the ITER radioactive inventory of the first wall, 10 years after neutron irradiation. Results for CdTe and HPGe are reported above and below the timeline, respectively, (adapted from [16]). These specific calculations are based on the following hypotheses: background of 50 counts per day in the spectral region of interest of the corresponding radionuclide, 50 cm between the detector and the source with the latter completely filling the detector field of view. Photons are emitted by radionuclides dispersed in a matrix having physical properties of a AISI 316L steel with 2 cm thickness, i.e., density 8 g/cm^3, linear attenuation coefficient from Akkurt et al. [26]

4 Case studies of non-destructive radiological characterization

Building upon the principles of gamma and X-ray spectrometry using HPGe and CdTe detectors, as outlined above, the model developed by Marzo et al. [16] provides a framework for evaluating the measurement time necessary to achieve statistically significant quantification limits for specific radionuclides, given the photon energies specified in Table 1. Applying this model to the decommissioning scenario of the future ITER fusion reactor indicates that most radionuclides within the anticipated radioactive inventory (up to a few GBq/kg, even 10 years after neutron irradiation ceases [16]) can be quantified within a few minutes. Additionally, the model predicts that Fe-55 and Ni-59 can be quantified within a few hours using the X-ray detector, demonstrating its viability as an alternative to more complex and time-intensive destructive analytical methods (Fig. 1).

During the decommissioning of the ENEA Frascati Tokamak Upgrade [14], the effectiveness of non-destructive techniques was evaluated. Since many of the FTU components were intended for reuse or recycling, the characterization protocols are designed to verify that radioactivity levels were within regulatory limits, ensuring they were not radiologically significant [21]. Consequently, longer measurement times were necessary to achieve the required detection limits, which were below 1 kBq/kg for the relevant radionuclides [25]. Furthermore, the planned reuse of plant structures prevented destructive sampling methods like cutting or slicing. This limitation meant that radiation from internal parts had to travel through significant material to reach the detector, reducing the efficiency of detection, particularly for lower energy photons.

Figure 2 presents a model-based analysis of measurement time as a function of material thickness traversed by gamma photons before detection. In this first analysis, an HPGe detector is simulated at a 50 cm distance from the plant structure, replicating typical in situ measurement conditions. Key modeling assumptions include complete filling of the detector field of view by the source material and the use of AISI 316L steel activated material properties, with its linear attenuation coefficient derived from Akkurt et al. [26]. Many structures within the FTU exhibit thicknesses spanning from a few to tens of millimeters, which translates to measurement times ranging from several hours to multiple days for most radionuclides under consideration. These measurement durations are considered manageable within routine decommissioning characterization practices. Figure 2 illustrates specific examples of material thicknesses associated with various components and structures encountered during the ENEA FTU decommissioning campaign. The figure correlates these thicknesses with the measurement times necessary to achieve the quantification limits [18] mandated by regulatory provisions. Specific examples of measurement times are presented in Fig. 2, including in-port diagnostic supporting structures (0.2 cm thickness), support structures for the toroidal limiter tiles (1.8 cm thickness), and port flanges (3 cm thickness). Additionally, external accessories of the main plant, such as the FTU deuterium pellet injector tank (0.6 cm cylindrical steel structure), are also depicted, as they require characterization for regulatory clearance regarding reuse or recycling. While these components can be effectively characterized within a few hours, structures with steel thicknesses exceeding several centimeters may require measurement times extending to days, thus presenting a significant challenge for the accurate quantification of radionuclide occurrence.

Regarding CdTe detectors, while exhibiting a relatively low intrinsic detection efficiency due to their inherent crystal lattice properties, they compensate with a higher overall detection efficiency compared to HPGe detectors. This improvement is achieved

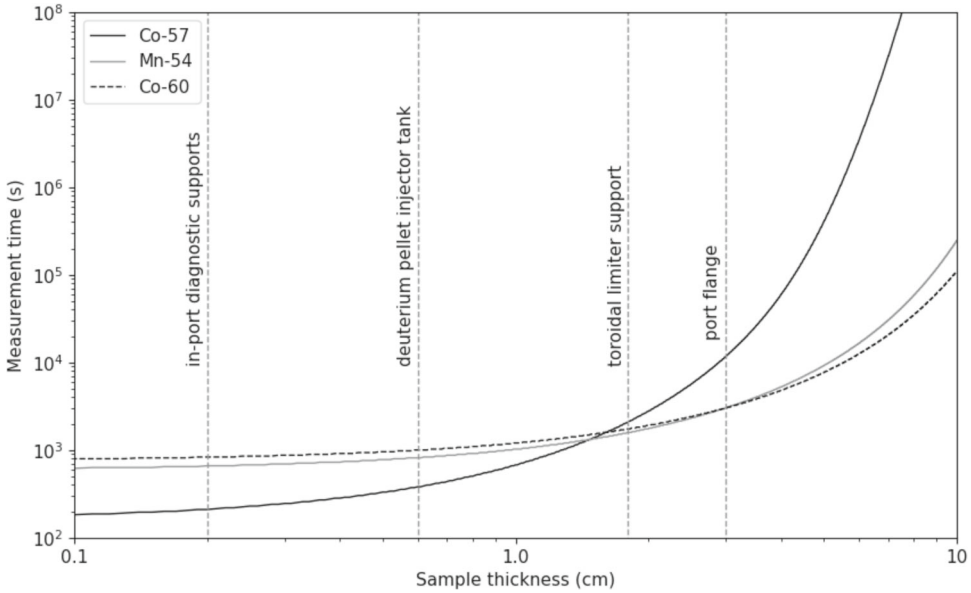

Fig. 2 Measurement time needed to ensure the detection limits required by law as a function of the sample thickness of activated AISI 316L steel. These specific calculations are based on the following hypotheses: HPGe detector, background of 100 counts per day in the spectral region of interest of the corresponding radionuclide, 50 cm between the detector and the source, and source material of 0.2 cm placed inside the plant structure under consideration

through enhanced geometrical efficiency, which requires the CdTe detector to be positioned in close proximity to the radiation source, making CdTe detectors particularly well-suited for detailed, localized mapping of radiation fields. As an illustrative example, a conventional HPGe detector is capable of characterizing a substantial steel surface area in a single measurement. It should be emphasized, however, that certain HPGe detectors—when equipped with appropriate collimation systems such as the In Situ Object Counting System (ISOCS) or the BSI Mobile Spectrometer—can also perform detailed surface characterizations, enabling the detection of potential activation inhomogeneities. Nevertheless, for high-resolution or small-spot surface measurements, these systems often require bulky and heavy collimation setups, which can limit their maneuverability and ease of use in confined or complex geometries. In such cases, compact detectors like CdTe offer a more practical solution for localized mapping of radiation fields in situ assessments. Moreover, a CdTe detector, when integrated with a precise positioning system, allows for detailed surface characterization, facilitating the study of activity inhomogeneities. Consequently, a CdTe detector possesses the potential to quantify the radionuclides expected in the ITER radiological inventory within measurement durations of a few hours (Fig. 1). However, this capability does not extend to the clearance levels required for FTU materials. In this scenario, typical measurement collection times would be on the order of years for radionuclides such as Fe-55 and Ni-59.

Figure 3 illustrates the measurement times necessary to achieve the detection and quantification limits, as defined by Currie [18], as a function of the activity concentration of Fe-55 and Ni-59, demonstrating the potential of CdTe detectors for detecting these Hard-to-Measure (HTM) radionuclides when their activity concentration is significant. Specifically, at the expected activity concentration in ITER, Fe-55 is readily quantifiable, while Ni-59 requires hundreds of seconds for detection and approximately 5000 s for statistically significant quantification. In contrast, the limitations of CdTe become evident when considering the lower activity concentrations of these radionuclides in the case of FTU. Under typical laboratory timescales, CdTe detectors are unlikely to enable their detection in this scenario. Another limitation of X-ray detection becomes apparent in the presence of high-energy gamma emitters, which can necessitate substantial corrections to X-ray photon count rates or even lead to signal loss due to Compton scattering [27], potentially requiring destructive analysis.

5 Conclusions

The operation and decommissioning of future fusion reactors will generate radioactive waste, primarily composed of neutron activation products and tritium. Given the absence of specific fusion plant decommissioning guidelines, establishing comprehensive waste management processes and protocols at each stage of the plant's lifecycle is crucial. This necessitates thorough qualitative and quantitative characterization of the radionuclides occurring in materials and components. Radiological characterization commonly employs both destructive and non-destructive methodologies. Gamma-emitting radionuclides are typically quantified using gamma spectrometry with solid-state detectors. Beta-emitting radionuclides, such as Fe-55 and Ni-59, are generally quantified via destructive techniques, which, while reliable and potentially more sensitive, involve time-consuming chemical sample preparation.

This work evaluates the potential and investigates the applicability limits of NDT by applying them to two distinct scenarios in the context of nuclear fusion plant decommissioning. For facilities with substantial radioactive inventories, such as the future ITER reactor, CdTe detector-based X-ray spectrometry [16] offers a viable non-destructive alternative. This technique combines favorable

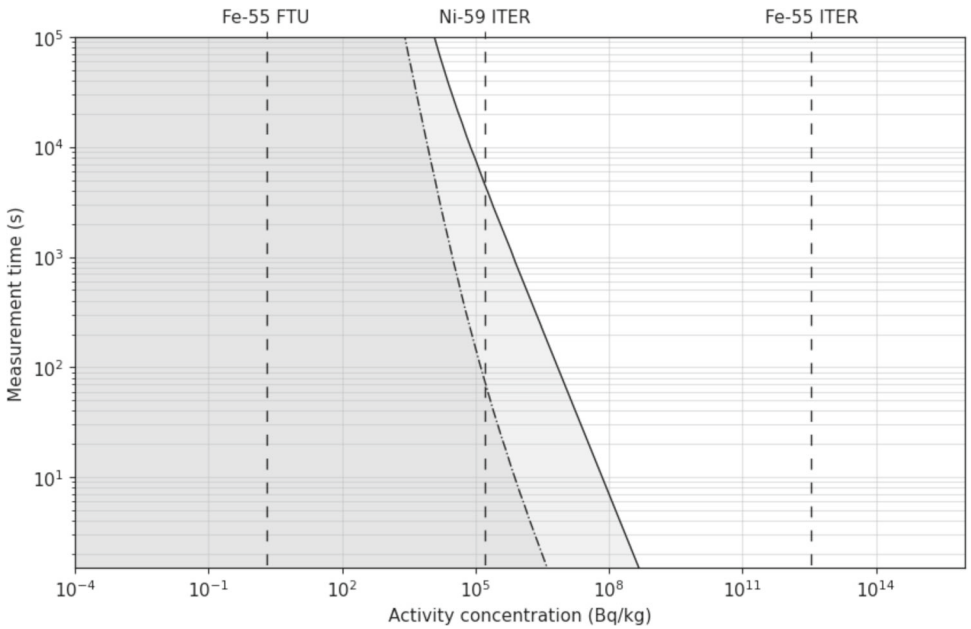

Fig. 3 Measurement times required to reach the detection and quantification limits [18] for Fe-55 and Ni-59 as a function of their activity concentration, calculated using the model developed by Marzo et al. [16] for a CdTe detector. Ni-59 in the case of FTU is not shown because it falls outside the abscissa range. These specific calculations are based on the following hypotheses: background of 50 counts per day in the spectral region of interest of the corresponding radionuclide, 50 cm between the detector and the source, and source material with the physical properties of AISI 316L steel

Table 2 Critical issues in the adoption of DT and NDT characterization approaches, given a specific detection limit provided by Law or by Authority provisions

Characterization approach	Overall time effort	Radiation protection	Uncertainty
Destructive techniques	Number of samples to ensure representativeness. Necessary amount of material to be sampled. Multi-step samples preparation.	Secondary radioactive waste production. Hands-on activities. Radioactive dust production during drilling or milling.	Inhomogeneity. Activation distribution and geometric configuration models. Cross-contamination.
Non-destructive techniques	Large (compared to destructive techniques) measurement time for ensuring the detection limit.	Set-up of equipment in close proximity to object to be monitored.	Simulation model to evaluate efficiency (e.g., Monte Carlo codes). Low detection efficiency (compared to destructive techniques).

energy resolution and compact detector dimensions [28] with reduced costs, while also preventing the generation of secondary waste in analytical procedures, thereby aligning with the ALARA principle.

Table 2 Summarizes critical issues associated with DT and NDT characterization approaches, addressing aspects such as overall time effort, radiation protection, and uncertainty sources, considering a specific detection limit provided by law or regulatory provisions.

In the specific case of the ENEA Frascati Tokamak Upgrade (FTU) decommissioning, a unique scenario arises due to the planned reuse or recycling of the majority of plant components. Italian legislation necessitates ensuring clearance levels below 1 kBq/kg for each radionuclide of interest [25]. While gamma emitters can be quantified within practical measurement durations, ranging from hours to days, using solid-state detectors, this is often not feasible for nuclides decaying via electron capture and emitting very low-energy X-rays. For these nuclides, achieving detection limits comparable to clearance levels would necessitate impractical collection times extending to years. Therefore, destructive characterization techniques remain indispensable in these cases, where, however, they do not typically pose a significant radiation protection issue for handling, given the low activity of the samples at clearance levels.

On the other hand, in the case of the future ITER reactor and its significant radiological activity concentration inventory, CdTe detector-based X-ray spectrometry [16] can be elected as a meaningful solution for accurate and extensive mapping of components to be characterized. Compactness, light weight, and maneuverability allow for easier remote operations with a robotic platform with respect to heavier and larger HPGe detectors. Moreover, CdTe can afford detection of some HTM nuclides such as Fe-55 and Ni-59, avoiding the execution of destructive characterization techniques for those radionuclides, preventing secondary waste, and optimizing the overall radiation exposure.

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Data availability The datasets generated or analyzed during the current study are not publicly archived, as all relevant data supporting the findings are available within the cited references or can be obtained from the authors upon request.

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