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Radiological characterization of the reactor pressure vessel of Trino NPP for dismantling purposes

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ABSTRACT

The decommissioning of a Nuclear Power Plant (NPP[1]) is based on the knowledge of radioactivity content and its distribution. The reactor pressure vessel and its internal components have the highest radioactive levels in the plant, considering the contributions of the residual radioactivity due to neutron activation and the surface contamination. This paper presents the steps of the radiological characterization process carried out on the pressure vessel and internals of "Enrico Fermi" NPP. The activation data, computed by analytical procedures in a previous study, are analysed and semi-empirical models have been developed to determine contamination levels where activation is not expected. Non-invasive external dose rate measurements have been performed for a preliminary validation of the activation profiles and to be used in the contamination models. Afterwards, an optimal sampling plan has been adopted to quantitatively validate the current radiological framework 30 years after the end of the lifecycle, in order to get the radiological classification for the optimal choice of the cutting techniques and of the number of radioactive waste containers.

1. Introduction

At the end of the life of a Nuclear Power Plant, the decommissioning phase starts. It is a complex, multidisciplinary, expensive project, which is generally longer than decommissioning of other conventional power plants. Nuclear decommissioning has not only techno-economic implications, but also social and environmental impacts (Invernizzi et al., 2016). The objective is to dismantle the components of the plant, to manage the radioactive wastes and convert the site to new use. The process must be optimized in accordance with the ALARA principle, finding the right trade-off among the different techniques, in order to minimize extra costs, radiological doses, project delay and environmental impacts (IAEA Technical report, 1998; IAEA Safety Standards, 2006). The most critical components are the Reactor Vessel Internals (RVI) and the Reactor Pressure Vessel (RPV), since they have the highest content of radioactivity in the plant. The first step in a dismantling process of a RPV is the radiological characterization; its aim is to determine the current radioactivity inventory, i.e. quantities, types and distribution of the radionuclides, in term of both residual activation levels (specific mass activity: Bq/g), which are the major contributions to the total plant radioactivity, and contamination levels (specific superficial activity: Bq/cm^2) (IAEA Technical report, 1998). Residual activation is due to the interaction of the neutron flux with structural materials during the reactor operation (Glascock, 2003); contamination is due to the deposition of some radioactive materials transported by the coolant. In particular, contamination radionuclides are corrosion and erosion products coming from the pipes walls, which were activated when they passed through the core (crud), and fission and irradiated fuel products released from the micro-crackings in the fuel claddings (IAEA Technical report, 1998; IAEA Safety guide, 1999; Bartlett, 1968). Therefore, activation generates gamma, X and beta radioactive decays, while contamination, which is partially fixed and partially removable (IAEA Technical report, 2005), is also responsible of alpha radioactive decays. There are technical, economic and environmental advantages associated with an accurate radiological characterization, in addition to the improvement of radiation protection of the workers involved in dismantling operations. Indeed, the radiological characterization allows the best dismantling strategy to be established, that is the technologies and kind of environment for the RPV and RVI segmentation and cutting, particularly whether to adopt remote or hand working, and the convenience of a superficial decontamination of some components. Moreover, it's possible to establish the

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[1] Abbreviations: DA: Destructive Analysis, HLW: High Level Waste, HTM: Hard To Measure, ILW: Intermediate Level Waste, MDA: Minimum Detectable Activity, NDA: Non Destructive Analyses, NPP: Nuclear Power Plant, NST: Neutron Shield Tank, RPV: Reactor Pressure Vessel, RVI: Reactor Vessel Internals, VLLW: Very Low Level Waste.

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waste inventory with the associated radiological classification, therefore the management and the prevision of decommissioning schedule before starting the effective dismantling, together with the choice of the waste containers types and number, the way of filling them, according to the national regulations and the features of the waste repository. Finally, the radiological characterization also allows a preliminary estimation of the decommissioning cost (NEA Report, 2013; Park et al., 2016). To properly carry out the radiological characterization, it is necessary to know the history of the plant, in terms not only of energy production, but also of changes occurred in the plant, besides geometrical dimensions, materials and their locations, the masses of the components, and any available historical radiological data. Indeed, the choice of the characterization method to define the radioactive inventory depends on the quantity and reliability of previous information, and on the plant configuration too. Implementation of the radiological characterization is done by both analytical and experimental procedures. The analytical procedures involve the use of mathematical models and the application of computer codes (Yanagihara et al., 2001; Croff, 1983). The experimental procedures are Non Destructive Analyses (NDA) and Destructive Analyses (DA) techniques. NDA consist of non-invasive “in situ” measurements and laboratory analyses that do not modify the physico-chemical composition of examined materials, like the gamma-ray spectrometry (Anthoni et al., 2004), whereas an example of DA is the radiochemical analysis of samples (Xiaolin, 2007). Analytical procedures suffer from uncertainties in input data, discretization of the domain and application of some models, whereas the problems involved by experimental procedures are the choice of the measurement points, their accessibility and the difficulty to detect low-energy radiation (NEA Report, 2013). Therefore, in most cases, the radiological characterization requires the joint use of experimental procedures and analytical ones (IAEA Technical report, 1998). The best approach would be to carry out the complete radiological characterization with simple and inexpensive techniques involving a low dose commitment. However, these techniques usually provide only partial results that need to be integrated with the results coming from the application of more complex, long and expensive techniques (IAEA Technical report, 2007). For example, the characterization with simple NDA measurements of dose rate and gamma-ray spectrometry provides the knowledge of the activity of gamma emitting radionuclides only. Therefore, radiochemical destructive analyses on some significant samples must be done in laboratory in order to determine the activity of “Hard To Measure” (HTM) radionuclides (also said “Hard To Detect”), which decay beta and alpha, and the values of the scaling factors, i.e. the relationship between the activity of a HTM radionuclide and the activity of a gamma emitter (Taddei et al., 2015; IAEA Technical report, 2009). As regards the international experiences of radiological characterization of RPV for decommissioning, different strategies have been applied, as reported in the literature. The preliminary characterization of RPV and RVI of Chooz A (305 MWe), the first French PWR in operation and now in dismantling, was obtained on the basis of a 3D activation model that was calibrated through some measurements of activity performed on a few samples. The results, to be validated, were compared with the radiological characterization of other USA reactors already dismantled, whose features and operational life were similar to Chooz A (Ehrhard, 2016; Grenouillet, 2009). On the contrary, in the case of V1-1 and V1-2 VVER reactors at the Jaslovske Bohunice site in Slovakia, the activation inventory was determined by an intensified sampling campaign and dose rate measurements at sampling spots. This activity was performed in 2011, respectively five and three years after the definitive plant shutdown of unit 1 and 2. Totally 125 samples were taken and a gamma-ray spectrometry was performed on all samples, but the complete radiochemical analysis was carried out on only 13 of them. A remote machine was used to take samples from safety and control rods assemblies and neutron flux measurement instruments, placed in a High Level Waste (HLW) storage, and from RVI and RPV. Manual drillings were performed only for basic RPV materials. The total radioactivity of the two reactors of V1 NPP was 2.61×10^{17} Bq few years after the shutdown and Fe-55 was the main isotope: 72.78% (Kristofova et al., 2012). In the case of Rancho Seco PWR of 913 MWe at Herald site in California, the results of the initial analysis on RPV and RVI activation were obtained by neutron and activation codes (IAEA Technical report, 1998). Then, the data were combined with several empirical measurements of dose rate, which were carried out easily after the removal of the RPV closure head and before the segmentation of RVI. It was determined that the stainless steel liner inside the RPV was more activated than the RPV carbon steel shell (EPRI Technical Report, 2008). Although activity levels may vary among different reactors, according to their irradiation histories and physical characteristics, a common feature is that the internals are more activated than the RPV and the internals closer to the reactor core are definitively the ones most activated. Furthermore, studies on the research PWR reactor BR3 (Belgian Reactor No. 3) have shown that the activity of RVI remains at high levels even 30 years after the end of its operation and a dose rate that did not allow direct operations without shielding was detected in the core zone (Massaut, 1998; Demeulemeester et al., 2001). The purpose of this paper is to show the process and the first results of the radiological characterization of the RPV of Trino NPP (Italy), whose lifecycle ended about 30 years ago (shut down in 1987) and it is under decommissioning, with the aim of the green field. After a brief operational and physical description of the reactor with its current configuration (paragraph 2), the available radiological data are analysed in paragraph 3. They are included in a database of the activated components (Genova et al., 1995) whose reliability has to be verified, since it was determined analytically and validated through the comparison with a limited number of samples. Therefore, some measurements of dose rate have been carried out near the external wall of the RPV in order to provide a preliminary qualitative validation of the activation data. Furthermore, the radiological contamination on the inner surfaces of the peripheral components of the RPV, which are not considered activated according to the database on residual activation, has been computed through the application of dedicated semi-empirical models, that are specifically developed to use the contamination data known for other plant components (paragraph 4). Afterwards, a sampling plan has been adopted for the quantitative validation of both activation and contamination levels. In particular, the process for an optimal subdivision of the components of the RPV and RVI into homogeneous groups is described, minimizing the number of samples to be taken and identifying sampling locations (paragraph 5). Some samples have already been taken and analysed (paragraph 6). Eventually, the results of the radiological characterization affect the decommissioning process, with the radiological classification of activated and contaminated components, the choice of better cutting techniques and the prevision about the number of containers for radiological waste (paragraph 7).

2. Operational data of Trino NPP and physical characteristics of RPV & RVI

Trino NPP was a PWR of 870 MWt and 280 MWe. It operated from 1965 to 1987, on 9 operating cycles, for 10.6 years of full power operation (870 MWt), totally producing more than 25 TWh without relevant incidents, Table 1.

The main dimensions and materials of the RPV are shown in Fig. 1 and Fig. 2, whereas Fig. 3 represents both RPV and its RVI.

The RPV was designed by Westinghouse and built by Combustion Engineering. RPV walls are made of carbon steel 302 grade B, a low-grade Mn-Mo steel, with an austenitic stainless steel 304 inner liner of 4 mm minimum thickness. The coolant four inlet and four outlet nozzles are connected to the RPV at the same axial height. The thickness of the RPV is reduced at the hemispherical head and bottom, while increasing at the flanges. Externally, the cylindrical shell of the RPV is coated with insulating material (fiberglass), which has already been removed from