Optimized Dismantling Methodology and Importance of Radiological Characterization

Heiko Herbell, Tobias Lopian, Frank Kostroun, Nikolaus Neudert, Jose Pedro Moreira do Amaral
Framatome GmbH
Paul-Gossen-Strasse 100, 91052 Erlangen, Germany
heiko.herbell@framatome.com, tobias.lopian@framatome.com, frank.kostroun@framatome.com, nikolaus.neudert@framatome.com, pedro.moreira@areva.com

ABSTRACT

Recent nuclear dismantling projects are using standardly a decommissioning approach from „hot" to „cold" e.g. from inside to the outside which enables an early removal of the radiological inventory. The overall dismantling sequence mainly consists of:

- Full System decontamination
- Radiological evaluation of the systems using sampling and/or inline radiological measurements
- Segmentation and packing of the Reactor Pressure Vessel (RPV) internals pieces into waste containers
- Segmentation and packing of the RPV pieces into waste containers
- Removal of other large components (for ex. steam generator, pressurizer, …)

The paper focuses on lessons learned and recent experience from radiological evaluation using sampling and/or inline radiological measurements. Preliminary activation calculations are usually carried out on the basis of existing material and neutron fluency data for the radiological characterization of systems and components. The assessment of the existing surface contamination of systems and components is usually carried out by analysing data from wipe tests and material sampling as well as contamination measurements on site.

In the estimation of the surface contamination, there may be significant deviations with respect to the assumed nuclide vector, which describes the activity ratios of the available radionuclides among each other. The so-called tramp uranium, for example, can cause uranium leaks into the primary circuit by fuel rod defects. Said tramp uranium can then significantly increase the activity proportion of the alpha emitters in the surface contamination of primary circuit components as well as secondary components. However, precise knowledge of the activity of the alpha emitters is of particular importance for the radiation protection measures in the dismantling of the affected systems and components and for the final waste declaration.

Experience has shown that these activation calculations can lead to significant deviations in some areas in order to determine the activity in the base material (so-called matrix activity). This is due to conservative assumptions regarding the material composition and deviations in the assumed neutron fluency.

Keywords: *dismantling, decommissioning, radiological characterization, legacy waste*

1 INTRODUCTION

Disposal of large components such as steam generators is one of the major tasks within nuclear industry both for nuclear decommissioning and long term operation (LTO) when large components need to be dismantled or replaced.

The global nuclear fleet is ageing and nuclear dismantling remains one of the major tasks within nuclear industry. Most dismantling projects follow an approach from „hot” to „cold” or from inside to the outside which enables an early removal of the radiological inventory [1].

Framatome is a major international player in the nuclear energy market recognized for its innovative solutions and added value technologies for designing, building, maintaining, and advancing the global nuclear fleet. The company designs, manufactures, and installs components, as well as fuel and instrumentation and control systems for nuclear power plants and offers a full range of reactor services. Framatome in Germany also offers a competitive solutions portfolio for the post-operational-phase and dismantling of nuclear power plants which features: dismantling of large components (RPV, RPV-internals, steam generator etc.), engineering, dismantling scenario studies, system decontamination, sampling, characterization, radiation protection, waste management, waste treatment and backfitting of (mobile) operating systems.

The following paper focuses on lessons learned and recent experience from radiological evaluation using sampling and/or inline radiological measurements. Preliminary activation calculations are usually carried out on the basis of existing material and neutron fluency data for the radiological characterization of systems and components. The assessment of the existing surface contamination of systems and components is usually carried out by analysing data from wipe tests and material sampling as well as contamination measurements on site.

2 REMOVAL OF LARGE COMPONENTS

Steam Generators are large heat-exchangers of varying designs that are used for generating steam in PWR nuclear power plants and at the same side be the barrier between the heavily contaminated primary system and the secondary system which should be almost free from radiological contamination. From a decommissioning perspective, most steam generators share a common basic construction: having an outer, substantial shell covering a large quantity of relatively small-bore tubes that have been exposed to radioactive contamination on the inner surfaces. Approximatively 250 steam generators (SG) of Western PWR type are in operation or are waiting their dismantling. In Europe, around 220 retired steam generators are stored on-site at their respective Nuclear Power Plants (NPPs) premises [2].

Most of these plants have a water access, either by being along the coast or along any large river which indirectly provides water access. This is a key parameter when evaluating the options for the SG decommissioning. The alternative with road transport combined with sea transport may work as well but can complicate the shipment.

One of the main reasons for cost overruns in a decommissioning project is when delays in bottleneck activities will extend the duration of the project. This is well documented in projects performed in the past. One of the potential bottlenecks in PWR decommissioning is the management of the steam generators, especially if they are intended to be disassembled/segmented inside the containment.

Somewhat simplified, there are four options for the site operators to decommission the SGs:

- Option 1 On-site segmentation inside the containment
- Option 2 Treatment in a local on-site waste treatment center

• Option 3 Rip and Ship for disposal of the SG as an entire component
• Option 4 Rip and Ship for processing in an external facility

For options 1 and 2, the site operators can control the full process using their own resources or external contractors. The first option has the lowest investments of the two but at the same time entails the largest risk for delays in the project. The second option requires massive investments and includes a long-term liability but results a lot of opportunities and can provide the employees.

Option 3 may look attractive as it is about disposing the SG as one piece, but experience tells that the disposal of such components is not possible in most countries. In some countries, it may be allowed to dispose SGs as very low level waste (VLLW) components such as France. If so, the main hurdle is then to reduce the activity content and the dose rates down to below the VLLW threshold values. Such operations may be both costly and generating a lot of residual waste from the chemical decontamination. Option 4 is further discussed below.

It may be argued that there is a preference for Option 1 and 2 as both exclude external transports, at least in a short-run, since the waste will have to be shipped for disposal in alternatively interim storage sooner or later.

3 METHODOLOGY FOR RADIOLOGICAL CHARACTERIZATION

Radiological characterization is specifically required when determining waste routes for dismantling of a nuclear power plant but also when it comes to treat radiological waste from operation or disposal of spent large components from LTO projects.

For the purpose of radiological characterization, the following technologies are available:

- Dose rate measurement on site
- Inline measurement
- Sampling of systems and components

This chapter explains each available technology. The radio nuclides which can be determined by the individual technologies are listed [3].

3.1 Dose rate measurement on site

Dose rate measurements directly on site in the power plant can be very helpful for characterizing activation and contamination. With this dose rate mapping, the activation that has led to the formation of high-energy gamma rays (essentially Co-60 and Cs-137) can be characterized in the core areas (such as reactor pressure vessels). Furthermore, the contamination of components from non-nuclear areas (such as the secondary side of steam generators) can be determined qualitatively and semi-quantitatively. This method has proven itself, as it can be carried out without much effort. If unexpected values are detected, the mapping can be refined to obtain a good overview of the actual activity conditions in systems and components. Figure 1 gives an example for location of dose rate mapping of a steam generator.

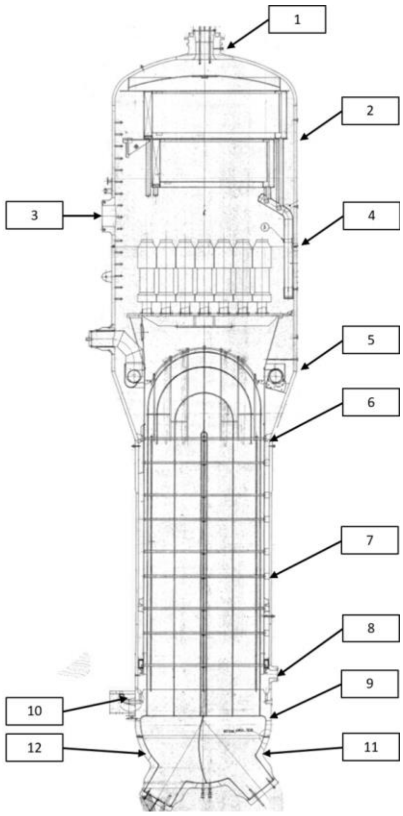
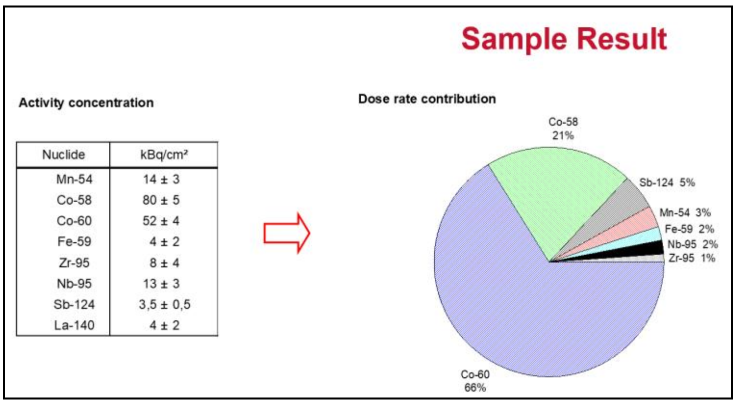


Figure 1: Example for Locations of Dose Rate Measuring Locations of a Steam Generator

3.2 Inline measurement

This method has proven particularly effective in the radiological characterization of surface contamination, for example, of steam generators and primary circuit lines. As part of the inline measurements, gamma spectrometric measurements are carried out directly in the power plant on systems and components. Based on many years of experience, these measurements are carried out at representative locations. A high-purity germanium detector with complete effective 4 pi shielding is used, which has a relatively low sensitivity to gamma radiation from the environment and still shows a good resolution when measuring (see Figure 2). The measurement data obtained in this way can be used to infer the specific gamma activity in Bq/cm^2 and the total gamma activity in Bq of the entire system or the entire component with the help of proven calculation models.



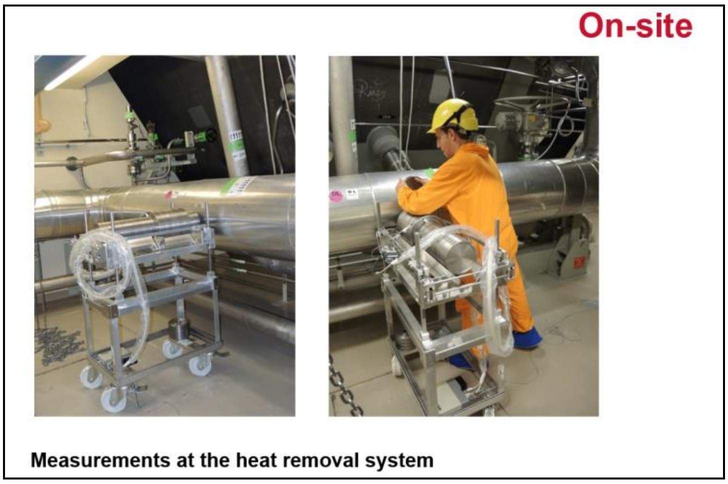


Figure 2: Example of Inline Measurement and Sample Results

3.3 Sampling of Systems and Components

Experience from previous decommissioning projects has shown that material sampling makes a significant contribution to radiological characterization of systems and components. As already mentioned, the aim of the sampling is to determine the activation of the base material and the surface contamination both qualitatively and quantitatively.

Uncertainties in the activation calculation (or calculated activity data) can be significantly reduced by sampling. Important information on deviations can be obtained by comparing the calculated activities in a system or component with the previously performed dose rate mapping.

Sampling in the form of chips or compact specimens has proven its worth. It has been shown that for the analysis of highly active metallic samples, weakly active metallic samples and concrete samples respective masses of 2-5 g, 10-20g and approximately 300 g are sufficient. The following elements and radionuclides are usually required by experts and authorities to carry out a reliable radiological characterization.

Metallic samples:

- Gamma-spectrometric analysis for the determination of Mn-54, Co-60, Nb-94, Sb-125, Cs-134, Cs-137, Ce-144, Eu-152, Eu-154 and others if detected
- Beta emitters: H-3, C-14, Fe-55, Ni-59 and Ni-63
- Determination of total alpha, total beta and total gamma activity
- Element analysis on: Li, C, N, Co and U
- Option: If Cs-137 or a high total alpha content is detected, optional nuclide specific determination of alpha emitters: U-233, U-234, U-235, U-238, Pu-238, Pu-239/240, Am-241, Cm-242, Cm-243/244 and the beta emitter Pu-241

Concrete samples:

- Gamma-spectrometric analysis for the determination of Mn-54, Co-60, Nb-94, Sb-125, Cs-134, Cs-137, Ce-144, Eu-152, Eu-154 and others if detected
- Beta emitters: H-3, C-14, Cl-36, Ca-41 and Fe-55
- Determination of total alpha, total beta and total gamma activity
- Element analysis in: Ca, Al, Si, Fe and U
- Option: If Cs-137 or a high total alpha content is detected, optional nuclide specific

determination of alpha emitters: U-233, U-234, U-235, U-238, Pu-238, Pu-239/240, Am-241, Cm-242, Cm-243/244 and the beta emitter Pu-241

Before analysing active samples, it has proven to be possible to determine whether inactive reference material, restorative or reference samples can be used. Otherwise, preferably weakly activated samples were used for determining the chemical compositions and in particular the impurities. If elements have been identified from the activation calculations that do not make a negligible contribution to activity even at low concentrations, they could - also in retrospect - be included in the analysis program. For example, trace impurities down to the ppb range could be determined using high-resolution Inductively coupled plasma mass spectrometry (ICP-MS) measurement technology.

The determination of the surface contamination can best be done on lens samples or compact sample pieces; alternatively, mechanical surface removal can also take place. The only important thing here is that the surface to be examined is in its original state, i.e., has not been changed. In the analysis laboratory, the contaminated surface of the samples is removed professionally or the surface removal is analysed. For compact samples, a distinction can be made between removable and non-removable contamination if necessary. The radionuclides usually required by experts and authorities are:

- Gamma-spectrometric determination, whereby the following nuclides are explicitly evaluated: Mn-54, Co-60, Zn-65, Nb-94, Ru-106, Ag-108m, Sb-125, Cs-134, Cs-137, Ce-144, Eu-154 and further ones, if detected
- Total alpha activity, total beta activity and total gamma activity
- Beta emitters: Be-10, C-14, Cl-36, Fe-55, Ni-59, Ni-63, Sr-90 and Zr-93
- Option: If Cs-137 or a high total alpha content is detected, optional nuclide specific determination of alpha emitters: U-233, U-234, U-235, U-238, Pu-238, Pu-239/240, Am-241, Cm-242, Cm-243/244 and the beta emitter Pu-241

Experience has shown that the inclusion of existing analysis data, in particular of wipe test samples and a step-by-step analysis, can in some cases significantly reduce the effort involved in determining surface contamination.

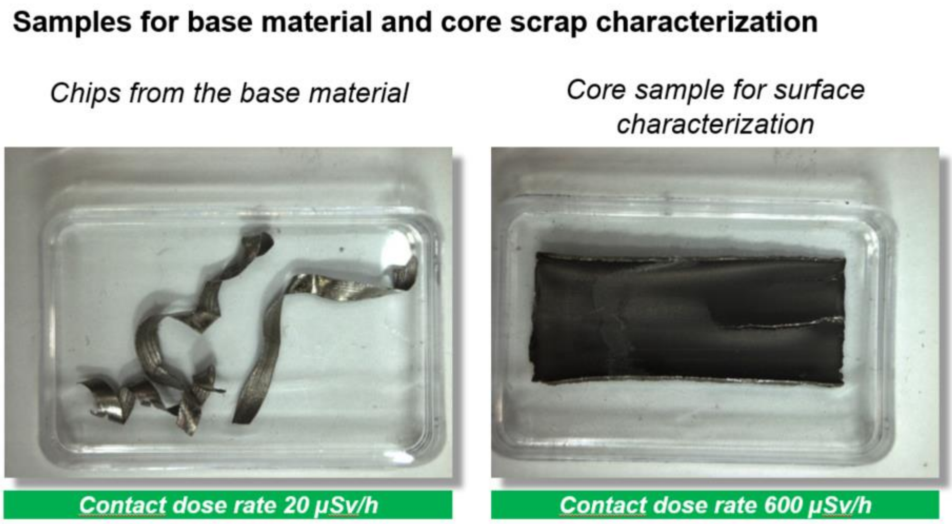


Figure 3: Example for Samples and Respective Contact Dose Rates

3.4 Lessons learned from radiological evaluation of the systems using sampling and/or inline radiological measurements

Preliminary activation calculations are usually carried out on the basis of existing material and neutron fluency data for the radiological characterization of systems and components. The assessment of the existing surface contamination of systems and components is usually carried out by analysing data from wipe tests and material sampling as well as contamination measurements on site.

Recent experience has shown that these activation calculations can lead to significant deviations in some areas in order to determine the activity in the base material so-called matrix activity. This is due to conservative assumptions regarding the material composition and deviations in the assumed neutron fluency.

Also in the estimation of the surface contamination, there may be significant deviations with respect to the assumed nuclide vector, which describes the activity ratios of the radionuclides among each other. The so-called tramp uranium, for example, can cause uranium leaks into the primary circuit by fuel rod defects. Said tramp uranium can then significantly increase the activity proportion of the alpha emitters in the surface contamination of primary circuit components as well as secondary components. However, precise knowledge of the activity of the alpha emitters is of particular importance for the radiation protection measures in the dismantling of the affected systems and components and for the final waste declaration.

Experience has shown that these described deviations have a direct influence on the radiation protection concept during decontamination and dismantling on site, the recycling and packaging concept and the declaration of radioactive waste for the interim storage and final disposal. Experience also shows that uncertainties can only be minimized with a suitable analysis concept in order to obtain realistic data on activation and contamination.

4 IMPORTANCE OF CHEMICAL CLEANING

The primary objective of a system decontamination (SD) is to remove the highest possible amount of radioactive nuclides from systems and components of the primary circuit as well as the connected auxiliary systems. This is accompanied by a reduction of dose rate in the affected areas. Radiation exposure is minimized for staff working on site with regard to preparation and implementation of shutdown and dismantling - following the ALARA principle - as early as possible.

Framatome applies the HP CORD UV procedures among others [4]. It is one of the so-called soft processes and was specifically designed to remove the activity-leading surface areas in a way that is gentle for the base metal. Oxide layers are developed as a multi-cycle process during plant lifetime. A decontamination cycle is applied based on chemical treatment processes necessary for oxide dissolution, consisting of the following different phases:

- Phase 1: Oxidation
- Phase 2: Reduction including removal of nickel / chromium
- Phase 3: Decontamination
- Phase 4: UV decomposition
- Phase 5: Cleaning

The oxidation phase is crucial as a preparatory treatment for the actual 95°C decontamination subsequent oxide dissolution and is usually - like the other phases of the HP CORD UV process – carried out at temperatures up to 95°C. In the course of the oxidation phase, poorly soluble chromium (III) is oxidized with permanganic acid to form more soluble chromium (VI). The effectiveness of the oxidation is strongly diffusion-controlled and therefore a temperature-dependent process.

5 CONCLUSION

Dismantling of large components remain a challenge both for nuclear dismantling and for replacement of large components. Chemical cleaning is widely applied before removal of large component such as steam generators. Chemical cleaning, system decontamination or in case of dismantling the complete NPP a chemical full system decontamination effectively reduces dose rate following ALARA principles.

Removal of large components is widely applied during dismantling of the NPP but also for LTO. Replaced SG are stored on site waiting for disposal. Radiological characterization is required to determine waste routes for old replaced steam generators.

Dose rate measurement on site, inline measurement and sampling of systems and components are used to determine radiological content and distribution of the spent component. Preliminary activation calculations are usually carried out on the basis of existing material and neutron fluency data for the radiological characterization of systems and components. The assessment of the existing surface contamination of systems and components is usually carried out by analysing data from wipe tests and material sampling as well as contamination measurements on site.

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