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ACTIVITY CHARACTERIZATION AND WASTE MANAGEMENT IN THE FIR1 TRIGA DECOMMISSIONING PROJECT

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ABSTRACT

FiR1 TRIGA research reactor was in operation from 1962 until 2015 and its dismantling was finalized in March 2024. All LILW nuclear waste generated from the dismantling project will be sent for final disposal to Loviisa NPP site, operated by power company Fortum. This requires careful waste activity characterization to fulfill Loviisa's waste acceptance criteria.
Activity characterization of the FiR1 project is based on both calculations and measurements. Preliminary results were calculated with MCNP and ORIGEN-S codes. These have been further validated and refined during the project by sample measurements to form material-wise nuclide vectors. Finally, the packed waste is characterized with the ISOCS gamma spectrometric measurement setup.
We describe the characterization methods, how they we utilized throughout the process and how the characterization results were reviewed and assessed. We also introduce specific challenges related to characterizing TRIGA research reactors and the connections with other aspects in decommissioning planning.

1. Introduction

FiR1 is a 250 kW TRIGA Mark II type research reactor located in Espoo, Finland. The reactor commissioned in 1962. It has been used for training, medical and industrial isotope production and various research purposes, including a structural modification for BNCT cancer treatments in the mid 1990's. The reactor was shutdown for financial reasons in 2015. Spent nuclear fuel was removed in 2020. Other active structures were dismantled in 2023-2024. According to a contract between VTT and Fortum power company, the radioactive dismantling waste is sent for final disposal in Loviisa NPP site operated by Fortum.

Characterizing the activated reactor structures is mandatory for planning the dismantling and waste management methods (identifying structures and activities to be dismantled) and radiation safety (dose rates and contamination control).
Activity characterisation in the FiR1 reactor decommissioning project is an iterative combination of calculations and measurements.
Preliminary estimates of activated areas and calculated nuclide vectors were formed and validated with sample measurements mainly before the shutdown. Some results were also refined during dismantling, since e.g. the reactor core structures could not be drilled before.
After dismantling, gamma active key nuclides in each waste package is measured with ISOCS gammaspectrometer (Figure 1).

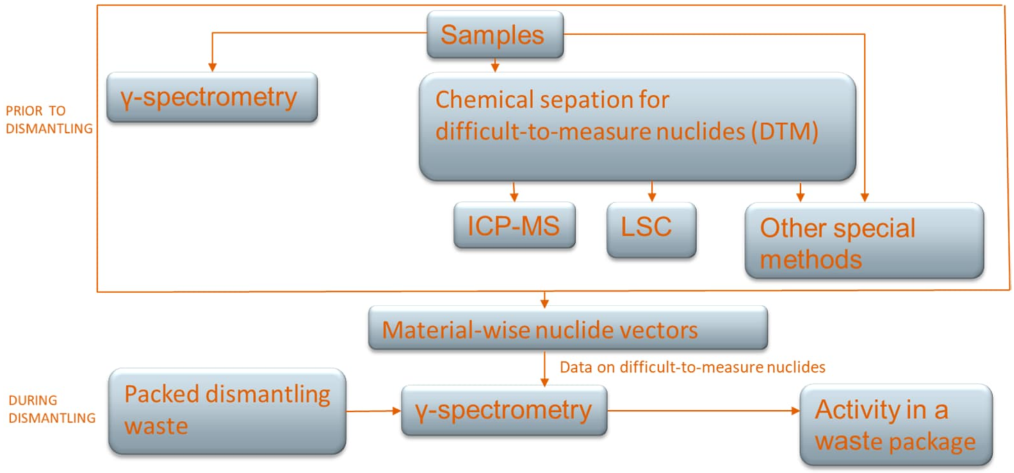

Figure 1: Characterisation approach.

Main challenges in research reactor activity characterizations are related to variety or materials and lacking input data. E.g. the long-term safety of waste final disposal mainly depends on long-lived difficult-to-measure nuclides (Figure 2), which are less relevant for the safety of mechanical dismantling (direct dose rates). Even if there is any original documentation on material compositions, they are typically compiled for estimating material mechanical properties during constructions and thus not directly reliable for activation calculations.
Certain materials can also cause chemical effects during final disposal, e.g. aluminium corrosion, even if their activity is relatively small.

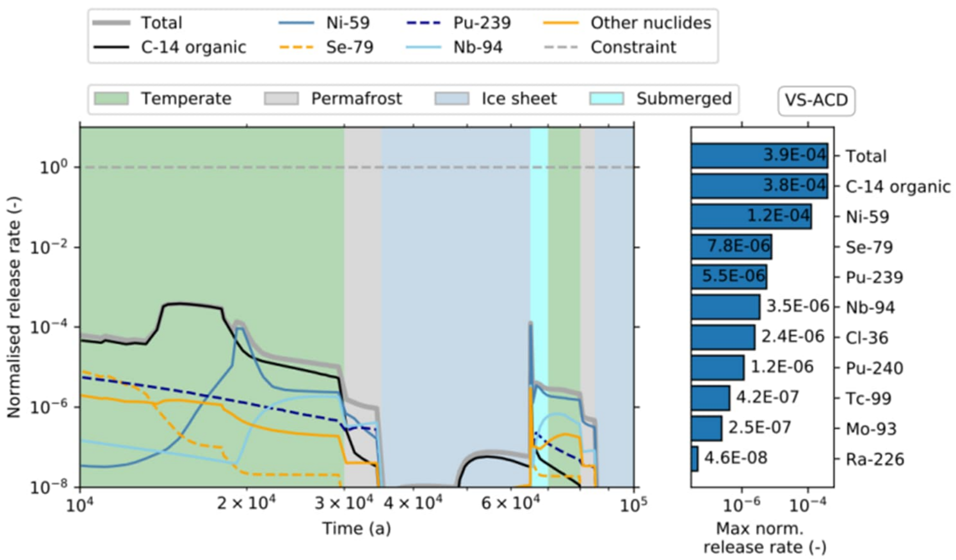

Figure 2: Normalized release rates from FiR1 decommissioning waste to the environment through the time period of 10 000 – 100 000 years. [1]

This paper summarizes the applied methods for active characterizations and waste management. Some specific challenges and lessons learned for characterizing research reactors and other older facilities are also discussed.

2. Calculations

In the preliminary phase, VTT conducted activity calculations using a model that combined several MCNP neutron flux models representing different reactor operation phases to

ORIGEN-S point-kinetic calculations to take into account the operation hours in each configuration (Figure 3). Since this model assumes that the target is mathematically homogeneous, the ORIGEN-S calculations were repeated for all the reactor main components and structures separately. [2] These results were used in the preliminary waste estimates and dismantling planning.

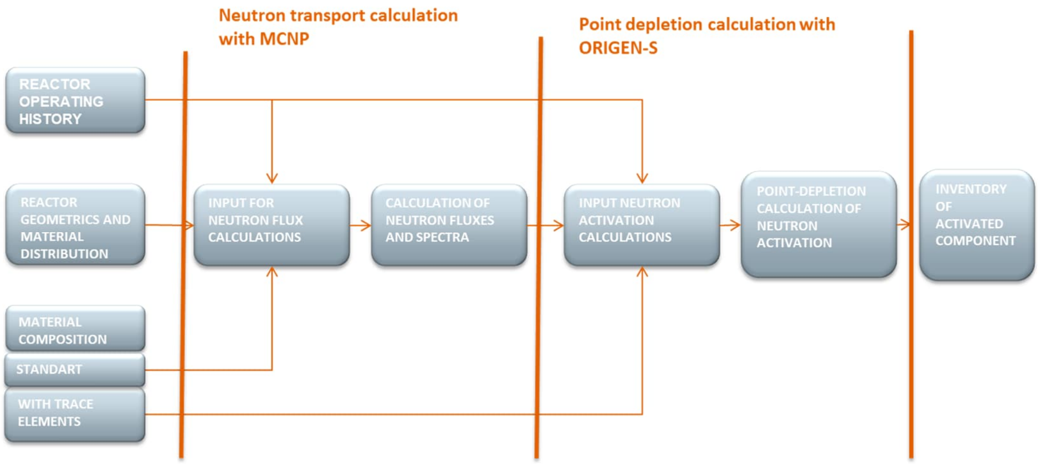

Figure 3: Calculations scheme

Two significant challenges in activity calculations were structural modifications during the operating history and missing data on material compositions. The reactor power increased in 1967 from 100 kW to 250 kW, the horizontal neutron beam tubes were plugged in late 1980's and the thermal column was replaced with an epithermal BNCT station in the mid 1990's. Moreover, there had been dozens of different core configuration, some even non-symmetrical. Altogether the calculations were formed by dividing the operating history into four phases

1) 1962-1967 with thermal power of 100 kW
2) 1967-1990 with open neutron beam tubes
3) 1990-1995 with plugged beam tubes and graphite thermal column
4) 1995-2015 with the thermal column replaced by BNCT station

Even this was a simplified model, since some components had been used inside the tank for shorter while. Moreover, e.g. material compositions of several components were unknown (or the provided data was unreliable), so several assumptions had to be made and these required re-iteration during the dismantling to validate the measured activities.

However, collecting and analyzing samples of e.g. concrete prior to dismantling enabled estimating e.g. the amount of activated concrete around the horizontal beam tubes sufficiently (Figure 4). This had a large impact on final amounts of waste [3].

Therefore, main lessons learned from the calculations would be to collect data operating history data and reference materials as early as possible.

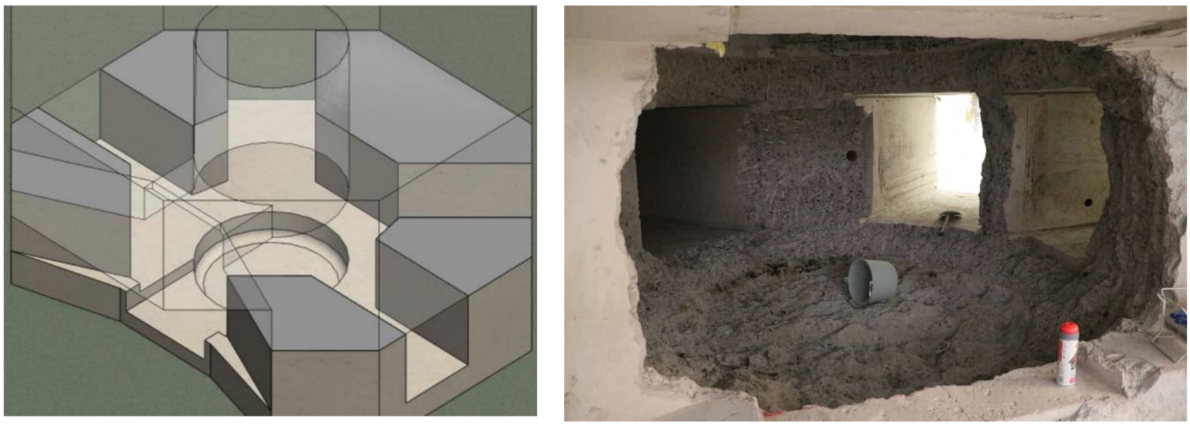

Figure 4: CAD model of the reactor biological after removing the activated concrete (left). Photo of concrete structure around two weeks before finishing the work (right).

3. Sampling and measurements

Prerequisite for representative sampling is in understanding of the material homogeneity/heterogeneity and radionuclide distribution in the material and their behaviour during sampling and storage. Practical challenges in the FiR1 project are typically related to the variety of material, relatively low specific activities and handling volatile nuclides. Some examples can be listed [4]:

- If the activity of DTM nuclides is below MDA, the vectors can only be validated using measured impurities from inactive reference samples.
- In case of small quantities of several steel types, the best approach has been to form only one vector using conservative assumptions and checking with the waste acceptor that the list contains all the relevant nuclides.
- A relevant gamma active key nuclide cannot be identified for some of the special materials (lithiated shielding plastics, Fluental neutron moderator, bismuth).
- How to measure the nuclides as a result of contamination from reactor operation (isotope production and activation analysis)? This required identifying gamma active key nuclides from historical records of reactor operation.
- Since some of the materials are inaccessible before dismantling commences, These vectors need to be further refined and validated during dismantling
- Release of volatile nuclides during sampling required developing special tools and methods to collect released nuclides [5]

Altogether around 230 material samples were collected and analyzed during the decommissioning project (does not include smear samples, air filters or water samples) Especially, the analysis of the DTM nuclides has required a significant effort, since reference samples from real activated materials were not available. Many of the developed methods have been validated with various international intercomparison exercises. [6-8]

4. Nuclide vectors

Characterisation of FiR1 decommissioning waste prior to dismantling has focused on formation of material specific scaling factors (or nuclide vectors). Formation of the scaling factors is based on ISO 21238:2007 standard [9].

The variety of materials especially in research reactors poses a significant challenge in formation of nuclide vectors. VTT decided to form vectors for all activated assuming that e.g. different aluminium types are handled using the same vector. Even with this simplification, altogether this still means that 15 scaling factors were formed [10].
Moreover, since at the beginning there was very little data available on material compositions and operating history, practically characterization was an iterative process meaning updating the original calculated results using conservative assumption based on data from the sample measurements. The process in illustrated in Figure 3.

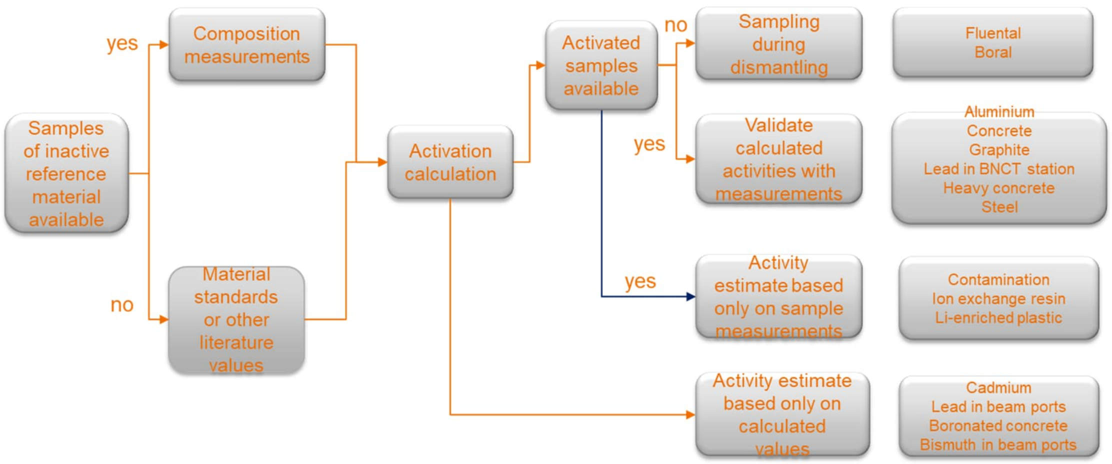

Figure 5: Approach for forming nuclide vectors.

5. Gamma spectrometric waste package measurements

Most of the waste packages generated from the decommissioning project were characterized with HPGe gamma spectrometry using ISOCS efficiency calibration software [11]. The method was chosen due to its cost efficiency, relative simplicity and ability to measure large waste streams, which are generated in the decommissioning projects.

Most of the waste was packed into 200 litre barrels, which were then scanned with HPGe detector while being continuously rotated with a rotating table. A suitability assessment [12] required by the national regulation [13] was made for measuring 200 litre waste barrels, which was approved by the waste acceptor Loviisa NPP. In the suitability assessment the standard methods for measurements were determined and the uncertainties of using ISOCS efficiency calibration were assessed. The uncertainties assessed were the effect of mismatches between the modelled and measured geometries for the most typical waste types generated during the decommissioning, the detectors ability to characterize waste with activity contents below clearance level, the detectors ability to characterize waste packed in heavily shielded packages and the determination of maximum allowed dead time in measurements for measuring active waste packages. As a part of the suitability assessment a quality control program for the measurements was also established which was used to ensure the condition of the used measurement equipment and the qualified performance of the measurements.

Some of the waste was dismantled and packed to other packages than barrels, mostly due to demolition technical reasons or due to Loviisa waste acceptance criteria limits for package external dose rate. For example, the reactor internal parts (i.a. irradiation ring, reflector) were put into heavily shielded steel packages, which were then measured and modelled using the ISOCS system. The waste acceptor required separate characterization reports for all the waste that was not characterized in 200 litre barrels. These waste packages could only be transported to the waste acceptor after they approved these characterization reports. The compilation of these separate characterization reports proved to be very time consuming, since they had to include the used ad hoc measurement methods and geometry models, the assessment uncertainties of the used characterization method and the assessment of the reliability of the results via validating them with e.g. comparing the them to computationally calculated activities. Later in the decommission project many larger waste units that were originally planned to be characterized in their entirety, were dismantled further to fit into 200 litre barrels, due to cost efficiency of characterizing waste with standardized method approved by the waste acceptor.

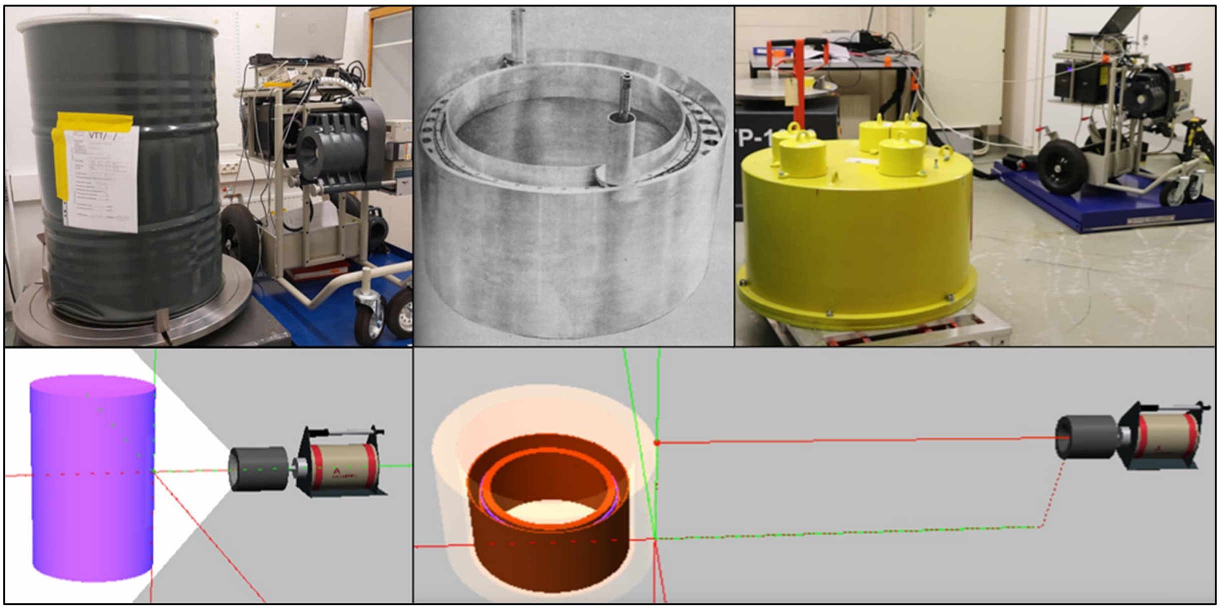

Figure 6. On the left is the ISOCS measurement setup and efficiency calibration model of a 200 litre waste barrel. On the right is the ISOCS measurement setup and ISOCS efficiency calibration model of the shielded irradiation ring of the FiR 1 reactor (shown in the grayscale figure).

For the ISOCS characterization of waste packages the activity distribution in the package needs to be as homogeneous as possible to achieve reliable results. This was achieved very well with screening the waste to different activity categories with hand-held measurement instruments. The concrete waste packages generated from the biological shield were very homogeneous due to the used demolition method, hammer drilling the activated concrete in small layers and collecting the generated waste into barrels in these small batches. For most waste categories, the correspondence of the activity concentrations measured from small samples and the waste packages were excellent. Also, the nuclide proportions between key nuclides and other measured nuclides from small samples and the waste packages were very close to each other, which suggests that the determined nuclide vectors can reliably be applied to the inventory of the waste packages.

The challenges in the ISOCS characterization of the waste included performing the measurements in the reactor hall in very tight spaces, which required good communication with the demolition personnel. In the reactor hall also some low background levels of Co-60 were measured in the background monitoring program measurements. This required careful selection of the waste measured waste in the different stages of the project since the activity of the measure package had to be high enough to make the low background level insignificant for the results. The packages with activity contents too low to measure them at the reactor hall were sent to a low background facility for the ISOCS measurements with another HPGe detector.

6. Summary

Early and comprehensive characterization of radionuclide inventories is highly valuable for all later planning. It provides necessary input for both dismantling planning, radiation safety and waste management.

We strongly encourage decommissioning operators to invest sufficiently to that important phase early on. Especially in old facilities, uncertainties e.g. in material compositions and lacking data on reactor operating history can cause challenges and results in very laboursome iterative process.

Reflecting to new build projects, collecting inactive reference materials of the reactor structures is very essential so that activation calculations can be performed with reliable input data. Validating the calculated estimates with measurements also require systematic development of the measurement methods. International intercalibration exercises are an example of valuable method development. This also provides an opportunity to systematically document the best methods.

Successful waste package measurements during dismantling requires active communication and agreeing on waste acceptance criteria with the waste acceptor as early as possible. Nuclide vector approach with ISOCS gamma spectrometric measurements provides a systematic setup, but contains various technical details related to e.g. measurement validation, package geometry, dose rates, sufficiently low background and reporting the results.

Nevertheless, FiR1 reactor dismantling was successfully conducted in 2023-2024 with relative low radiation doses to the personnel. Although FiR1 was the only research reactor in Finland, experiences from its dismantling can be utilized to prepare for decommissioning of large power reactor later in the future.

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