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Decommissioning of the BR3 biological shield: How a proper data analysis facilitates the D&D process

W. Broeckx [1*], S. Boden[1], B. Rogiers[1], J. Dadoumont[1], Y. Demeulemeester[1], R. Vandyck[1]

[1]SCK CEN, Belgian Nuclear Research Centre, Boeretang 200, B-2400 Mol, Belgium
[*]Corresponding Author, wouter.broeckx@sckcen.be

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Abstract

BR3, a former pilot reactor for Belgian commercial power plants, is currently being decommissioned. Because most technical equipment and installations are already dismantled, the focus of decommissioning is now on the building structure itself. More precisely, on the **activated concrete biological** shield surrounding the former reactor core.

Before defining a dismantling strategy, it is essential to characterize the structure and compare results with the requirements for the various disposal options in the country of the licensee. The initial characterization of the BR3 biological shield was selected as use case within the EU horizon 2020 project INSIDER (INSIDER, 2017). Within this project we applied a **data analysis and sampling design strategy** (INSIDER D. a., 2019), resulting in a **3D** model for the **activity distribution** throughout the entire concrete structure. In a next stage, the model was compared with **conditional** and **unconditional release** limits leading to clear targets on which parts and sections need to be removed to specific disposal sites. These targets served as input for the physical dismantling strategy itself, to decide which techniques are suitable for the job. As part of this project, test sections are defined, mainly for two purposes. On one hand to test and select on **possible dismantling techniques**, like hydraulic hammering and diamond wire cutting. On the other hand, to be used as **verification** of the modelled activity distribution which is initially based on a limited sampling size.

This paper describes the implementation of the data analysis and sampling design strategy as developed in the INSIDER project. Additionally, actual results of dismantling of the test sections demonstrate the soundness of the approach. Throughout the project, the availability of the model and visual representations turned out to be crucial. Not only as a reference and communication aid for dismantling experts and regulatory bodies, but also to optimize the dismantling strategy development.

Introduction

BR3 was a relatively small 10 MWe (about 41 MWth) pressurized water reactor of the SCK CEN (Belgian Nuclear Research Centre). It acted as a pilot reactor for the later nuclear commercial power plants in Belgium, and was brought into operation in October 1962 and shut down in 1987 after 25 years of operation and eleven campaigns. Figure 1 shows a cross section of the reactor building during exploitation. The entire primary circuit (reactor pressure vessel, steam generator, pumps & primary loop, etc.) has been dismantled, as well as the specific pool ventilation system and the anti-missile slabs. The bottom part of the reactor building consists of reinforced ordinary concrete. The remainder of the reactor building and the reactor pool or biological shield consists of **reinforced heavy concrete** (40 wt% Ba). The concrete of the reactor pool (and stainless steel pool liner) close to the reactor pressure vessel is **activated**. Since the reactor pressure vessel was laterally enclosed by a neutron shield tank (NST), the neighboring concrete is not activated, except locally near the hot and cold legs. The presence of the NST created an upwards neutron streaming effect. The activated part of the concrete is marked in red in Figure 1, according to a first, very rough estimation based on a limited amount of exploratory historical measurements.

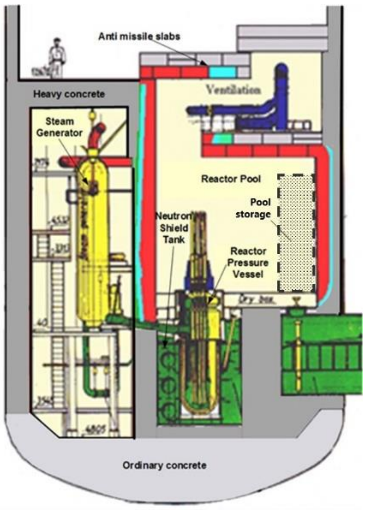

Figure 1: Cross section of the BR3 reactor building during exploitation showing the main components. The building structure is mainly composed of reinforced ordinary concrete (bottom plate) and reinforced heavy concrete (upper structure). The activated concrete is marked in red colour (first rough estimation).

The main challenge faced within decommissioning of such biological shields is the huge amount of potential radiological material it represents. **In Belgium**, this type of radiological material can be divided into three categories, being radioactive waste, material eligible for **conditional release** or **unconditional release**. The latter concerns those materials for which activity levels are below the general clearance levels as specified in the Belgian legislation (Belgium, 20 july 2001). From waste minimization point of view, it is crucial to reliably **localize** those parts of the biological shield which can be unconditionally released in order to economically optimize the corresponding dismantling strategy. Localizing these different material categories throughout the entire structure starts with a proper radiological characterization strategy.

Characterization and decommissioning objectives

The radiological characterization program serves two purposes. On one hand, it is needed to define the proper dismantling strategy, to know which parts need to be removed. On the other hand, since the biological shield represents a total mass of roughly 2200 ton, a certain fraction of this material will eventually be removed from site and characterizing this fraction at the back-end can be a logistic challenge. Therefore, the characterization program also aims to reduce the number of control measurements during the actual dismantling process itself. To tackle both aspects, the initial focus and effort of the project has been on the radiological characterization strategy with the following three objectives:

1. Create a **sufficiently reliable** 3D activity concentration distribution model, that includes uncertainty estimates.
2. Economically optimize volumes in view of **a waste-led approach** taking the following end stage options into account:
    a. unconditional release, aiming at conventional demolishing and recycling of the building materials),
    b. conditional release, for disposing the materials in a category 1 site for classical hazardous materials), and
    c. radioactive waste, for final disposal in a nuclear near surface disposal site.
3. Quantify and localize the different **end-stage volumes**.

This strategy was optimized in line with the **INSIDER** project (INSIDER, 2017) (Improved Nuclear SIte characterization for waste minimization in DD operations under constrained EnviRonment), funded by the European Commission, which focusses on this type of characterization challenges. One of the deliverables is the development of a sampling strategy **statistical guideline** (INSIDER D. a., 2019) and this guideline has been applied and further optimized based on experience gained with this biological shield, which served as one of the use cases of this European project.

Digital twin of the BR3 biological shield

The following sections describe different phases of the radiological characterization and data analysis program conducted at SCK CEN. It is intended to highlight the most important aspects encountered and how it resulted in the development of a **virtual model** (or digital twin) of the reinforced activated BR3 biological shield. It required the use of different statistical methods and modelling techniques. An in-depth description of the applied data analysis techniques and methods can be found on the INSIDER website (INSIDER, 2017), (von Oertzen, to be published).

Pre-existing data analysis

A few limited characterization campaigns were conducted over the years since reactor shutdown in 1987. In a first phase, these **results were combined** with other available information such as 2D plans, operational history, neutron calculations and samples from the reactor pool liner. Based on the analysis of the results from this pre-existing data and characterization objectives, it was concluded that for the different end-stages, the only radionuclides to be taken into account were:

- Co-60, Ba-133, Eu-152 for the activated concrete, and
- Co-60 for the reinforcement bars.

Moreover, the **Co-60 concentration** in the reinforcement bars is **about 5 times lower than the Ba-133** concentration in the concrete **at the same location** on the reference date (1 January 2020). This means that, when the Ba-133 concentration is below the clearance level in the concrete at a specific location, this will also be the case for the Co-60 in the reinforcement bars, at the same location. This moves the focus completely to the concrete activation.

A thorough preliminary analysis of the available data (from concrete samples and pool liner sample-based activity concentration estimates) combined with a 3D geometry (Figure 2) of the biological shield resulted in **a first model** which predicts the **Ba-133 activity distribution** throughout the structure based on the linear distance (through air and concrete) from the reactor core, and depth inside the concrete wall. The spatial concrete structure itself, with a typical **wall thickness of 120 cm**, was discretized into 10 cm x 10 cm x 10 cm cubes) which was the result of a trade-off between post processing time and resolution. In this way a first version of the so-called digital twin of the biological shield was generated, represented by about 622000 cubes (or voxels), over **a volume of about 622 m³**, each with its own spatial coordinates and activity concentrations (including uncertainties) for the three main nuclides.

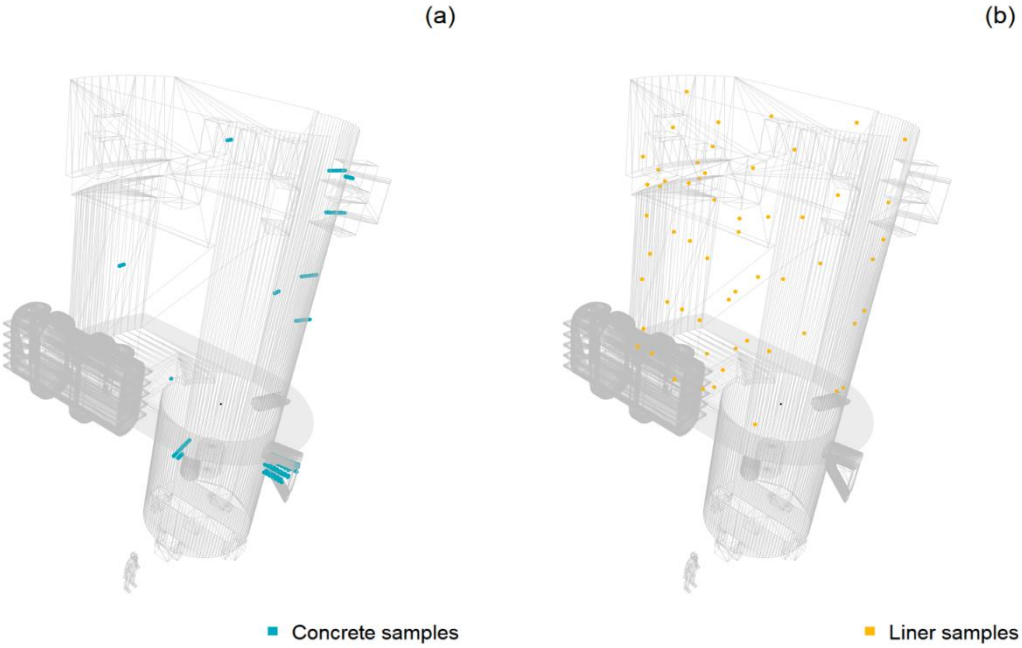

Figure 2: Overview of the boreholes (a) and liner (b) data locations corresponding to the measurement results used in the preliminary data analysis.

Sampling design and analysis

The objectives were not met after this first analysis of the pre-existing data due to uncertainties and too many extrapolations. The most activated region, predicted activation levels exceeded conditional release levels resulting in potential radiological waste. Towards **transition regions**, where activity concentrations enters into unconditional release, **uncertainties where too large**. To reduce the model uncertainties and minimize extrapolation, especially near the most activated regions, additional data gathering and sampling was needed. The **second phase sampling strategy** consisted of two types of data gathering:

- **Primary data**: borehole sampling, sample preparation and in-lab measurements.
- **Secondary data**: quick, straightforward and cheap non-destructive in-situ measurements.

Following the basic principles as described in the INSIDER data analysis & sampling design strategy (INSIDER D. a., 2019), the sampling design for the primary data consisted of **systematic** sampling supplemented with **judgmental sampling** locations (specific structures such as the former reactor pool storage of reactor internals (see Figure 1)). In addition, the expected trend extremes were selected as well (minimum and maximum activation levels at the inner side of the wall), and we relied on the **symmetry** of the activation to maximize the results with a minimum number of samples.

For the **primary data** gathering, 30 samples were taken by **wet core drilling** in agreement with the sampling plan. The cores (diameter 72 mm, length of about 90 cm down at the backside reinforcement bars) were segmented (thickness of 5 to 10 mm) and some of those segments analyzed by high-resolution gamma spectrometry.
Secondary data was gathered via **in-situ total gamma surface mapping**, consisting of about one measurement per square meter or roughly nearly 300 individual measurements (secondary data). The idea was to use these data as secondary information for the activity concentrations within the concrete, potentially in a similar way as how the liner data was used for the preliminary data analysis. Since the measurements were directly performed on the concrete and consisted of a lot more data points compared to the liner data, the liner data was no longer considered relevant for further analysis. The analysis of this dataset showed a few locally unexpected and/or aberrant results and as a consequence the inability to reach the objectives. Just below the bottom of the reactor pool, at the upper part of the Neutron Shield Tank (NST) the activity concentration levels were unexpectedly large, exceeding the limits for conditional release. The inability to in-situ measure or sample this region due to inaccessibility led to a very uncertain extrapolation of the modelled results. Additionally, the presence of a minor contamination locally increased the total gamma measurements (secondary data).
These aspects were tackled in **a third and final measurement and sampling phase** leading to a final dataset (Table 1) consisting of a total of 824 data points: 415 primary and 419 secondary data points, or roughly 1 measurement per m³ of the entire biological shield. This final dataset predicted only conditional and unconditional material categories for the biological shield. The initially predicted waste category near the reactor core, exceeding the criteria for conditional release, was no longer present.

Table 1: Final data overview: Different types and corresponding amounts of data points, in the unfiltered dataset, gathered for constructing the 3D activity concentration distribution.

Data type		Parameter	Unit	Number of data points
Primary	Pre-existing (stage 1)	Ba-133, (Eu-152, Co-60)	Bq/g	184
	New (stage 2)	Ba-133, Eu-152, Co-60	Bq/g	206
	Additional (stage 3)	Ba-133	Bq/g	25
Secondary	New (stage 2)	Total gamma	cps	20 (of 296)
	Repeated (stage 3)	Total gamma	cps	280
	Additional (stage 3)	Total gamma	cps	119

In comparison to the pre-existing data shown in Figure 2, the measurement and sample locations of this **final dataset** are shown Figure 3 and these are used to construct the **reference model**, predicting the 3D activity distribution at about 622000 different locations throughout the biological shield.

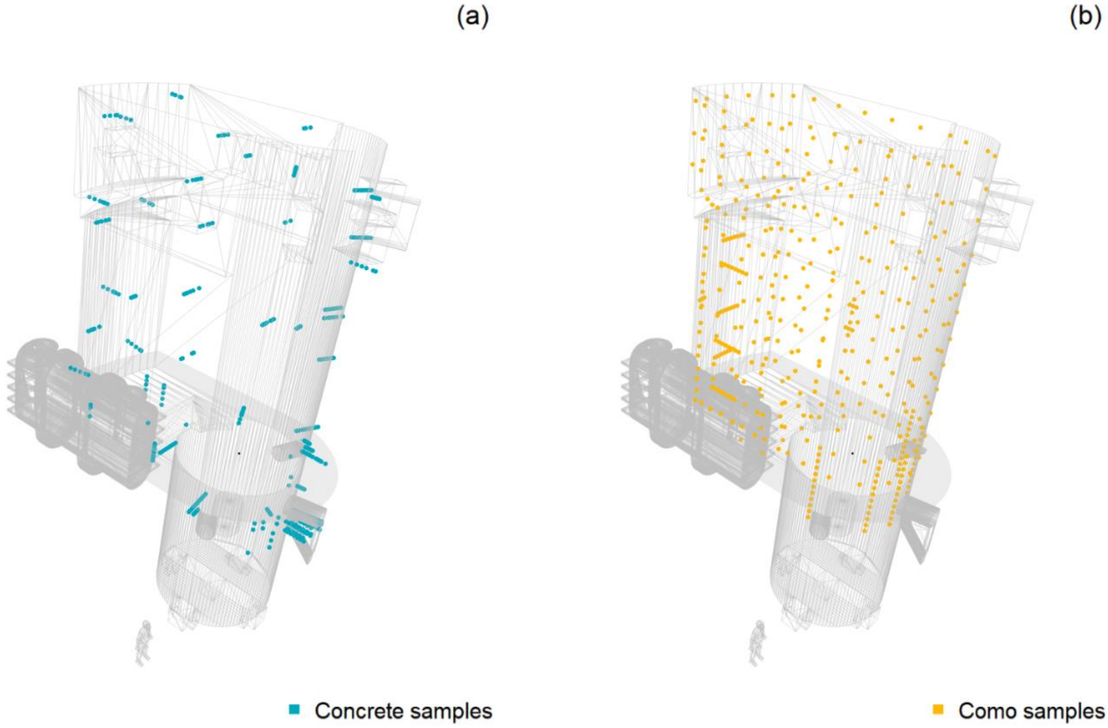

Figure 3: Overview of the borehole (a) and in-situ total gamma (b) data locations used for the final dataset. Como samples refers to the use of the COMO300G (Nuvia) handheld device used for the measurements.

Reliable end-stage volumes

Material clearance legislation in Belgium specifies clearance levels for individual nuclides and the use of the **sum formula** when several nuclides are involved. Besides these clearance levels, it is also specified that concentrations may be averaged out over 1 ton which corresponds to the volume of a 66cm wide cube in the case of reinforced heavy concrete. Having a digital twin of the biological shield with activity concentrations at 10cm resolution enabled us to incorporate both aspects (clearance levels and averaging) through post-processing, resulting in a **prediction of end-stage volumes** eligible for conditional and unconditional release.

The predicted volumes account for uncertainties, such as **measurement** uncertainties of primary and secondary data but also uncertainties of the **different log linear trend and scaling factor models**, as well as **spatial uncertainties**, applied to predict activities in the entire structure. Having all these uncertainties quantified enabled the stochastic simulation of 100 realizations of the biological shield activation, with each model having different end-stage volumes. From these 100 realizations, two statistical summaries were selected to be used as reference to define the dismantling strategy; a **best estimate model (50th percentile)** and a **conservative model (95th percentile)**. The effect of all these uncertainties, based on 100 realizations, on the end-stage volumes is shown in Figure 4. The profile marked in grey is the result of the reference case simulations from which the best estimate and conservative models are extracted.

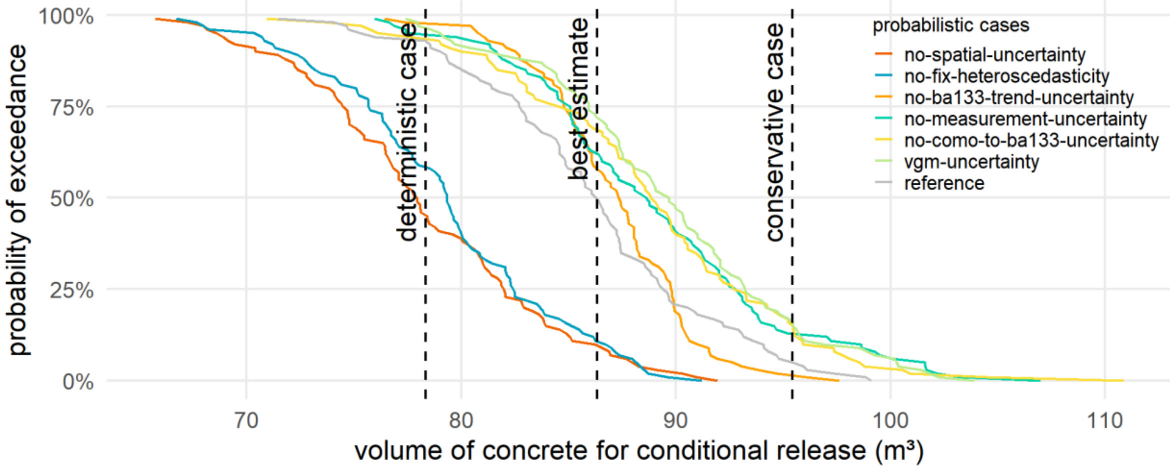

Figure 4: Complementary cumulative distribution function for the volume categorized as conditional release, in function of the different cases with respect to uncertainties

Additionally, a limited **sensitivity analysis** was performed to assess the importance of the different elements in the final dataset. For example, **reducing the primary dataset** down to 15 boreholes (roughly 1 borehole per 40 cubic meter), could result in **extreme under/over estimations** of the volume of concrete for unconditional release. However, the uncertainties can be strongly **reduced** by **combining** the limited higher quality and costly **primary dataset** (in-lab sample measurements) with a large cheap **secondary data** set (in-situ measurements).

From model to dismantling strategy

Once the predicted activity distribution is established the next stage of the decommissioning process, being the actual dismantling strategy, can start. This process involves **several experts and external stakeholders**. On one hand, the proper documentation needs to be prepared to inform the regulatory body (Federal Agency for Nuclear Control, FANC/AFCN) and to apply for a **specific license**. On the other hand, preparing and planning the actual physical dismantling itself requires looking at **practical constraints and logistics**. Furthermore, both aspects run in parallel and depend on the same source data, being the 3D activity model.

To facilitate these processes and the **communication** between different teams and stakeholders, the 3D model (about 622000 rows of data) is translated into a **series of 2D maps**, visualizing the essential information which is needed to meet the initial objectives.

These maps show the different **material categories** in detailed horizontal cross sections for every 0.1 m in height. As mentioned before, two versions of the model exist to include uncertainties; a best estimate model (median activity) and a conservative model (95^{th} percentile). How such a cross section is extracted from the 3D model is shown in Figure 5 while Figure 6 shows a selection of horizontal cross sections of both models. Since it is the intention to **minimize the radiological measurement and control** effort during actual dismantling, the conservative estimated categorization (**95% percentile**) is used as reference for **material segregation**. Meaning that at least all material marked orange in Figure 6 b will be removed from site.

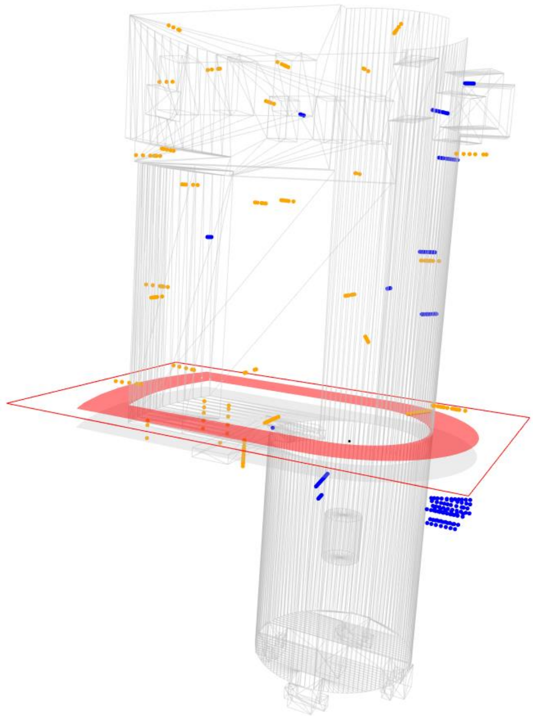

Figure 5: 3D representation of the BR3 biological shield showing the location of one of the horizontal cross sections. Markings in blue and yellow represent different sample locations

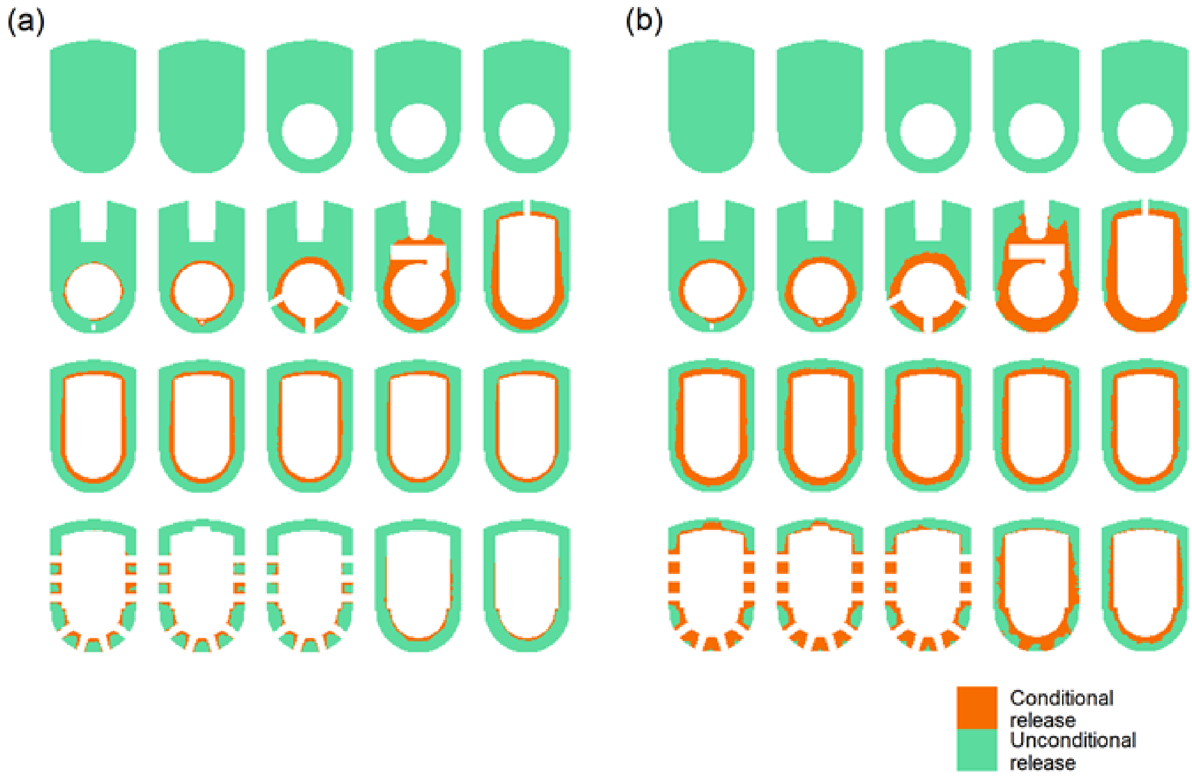

Figure 6: Selection of horizontal sections of the biological shield for different heights. On the left (a) we show the best estimate categorisation (median of the reference case), while the right figure (b) shows the conservative estimate categorisation (95th percentile of the reference case).

According to the defined dismantling strategy, a part of the biological shield will be removed via **pneumatic hammering**, similar to the test case discussed below. For other parts, where structural integrity of the biological shield could not be guaranteed after material removal, **diamond wire cutting** is foreseen to remove parts of the wall as whole. From administrative point of view, the total radiological inventory of the actually removed material will be based on the activities according to the best estimate model.

Test sections and model verification

Before starting the licensing process for the disposal of the fraction for conditional release, the predictions of the **3D model** have been **verified**. The SCK CEN D&D unit performed **tests** on specific regions to in-situ remove the part of the concrete requiring conditional release, from the remaining concrete eligible for unconditional release. For the separation, we applied **remote controlled pneumatic hammering**. The test section consisted of a part of the external wall (Figure 8), from the reactor pool floor up to 2 m height, over the entire wall length.

Figure 7: top view of the BR3 biological shield showing the BROKK pneumatic hammer used to remove materials from the test section. In-frame picture: close up of the curved wall after partial material removal

After removal of a concrete layer to a depth of about 50 cm in the test section, **concrete dust samples** at 12 different locations were collected. Each sample contains concrete dust collected from drilling a 300 mm deep hole (35 mm diameter). Figure 8 shows the location of the samples:

- 8 horizontal samples (087-03-12 up to 19) in the green zone, pointing towards the outside of the reactor building at two different heights (z=0.436 and z=1.436).
- 2 horizontal samples (left: 087-03-08 in the neighborhood of the former reactor pool storage and right: 087-03-11) in the corners of the walls, mainly in the blue (transition) zone (z=0.936).
- 2 vertical samples (left: 087-03-09 in the neighborhood of the former reactor pool storage and right: 87-03-10), mainly in the blue zone (left) and in the orange zone (right).

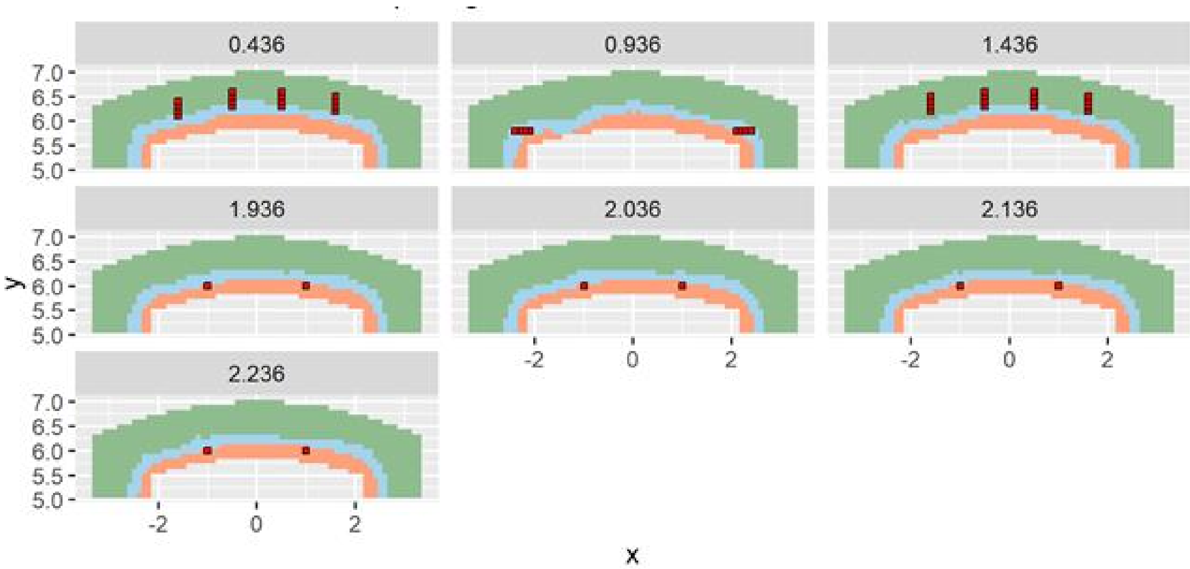

Figure 8: Close up of horizontal cross sections of the test area with indications for the region fulfilling the criteria for unconditional release according to the 95th percentile (green) and conditional release according to the 50th percentile (orange). The difference between both models (blue) represents the safety margin from reliability point of view. Eight dust samples have been mainly taken in the green zone (z=0.436 and 1.436). The four remaining dust samples, two horizontal samples (0.936) and 2 vertical samples (z=1.936 up to 2.236) have been taken in the orange (partly blue) zone

The collected **dust** was analyzed in the lab using high-resolution **gamma spectrometry**. When **comparing** the sum formula calculation for the dust sample measurements with the reference model sum formulas at the **corresponding voxels**, we notice the following:

- For the 8 horizontal samples located in the green zone (z=0.436 and 1.436): Sample measurement results are well below the limit for unconditional release.
- According to the sample measurements of the other 4 samples, which all could exceed the limit for unconditional release according to the model, only one was actually above the release limit but still below the level predicted by the model.

The agreement between the results of the **control measurements and the model is good**, taking the grid size (10 x10x10 cm) into account. Some asymmetry exists between the left and right hand part of the slightly curved wall due to the former pool storage (lead structure) and this is probably more pronounced than the model suggests. The structure itself was already dismantled in 2003 and was not included as such in the model.

Lessons Learned

Very **clear and quantifiable objectives** were defined prior to start the exercise. This allowed the development of an effective sampling plan including the selection of appropriate measurement techniques and the **up-front definition of criteria for the measurements** (e.g. detection limits, uncertainties). Nevertheless, the Belgian regulation changed during the execution, resulting in a modification of the objectives. Fortunately, this change did not impact the sampling strategy nor the decision threshold between unconditional and conditional release.

Detection limits for the primary data were typically more than ten times below the decision criterion and samples were carefully selected. Consequently, there were only **very few results below the detection limit**. The few results below detection limit (7% out of the 415 primary data points for Ba-133) have been discarded with negligible impact on the quality and size of the data set.

Obviously, a problem with **extrapolation** impacted the characterization process (at stage 2) and planning. Extrapolation should be **avoided as much as possible**. This does not only concern the expected activity concentrations (lowest and highest), but as well the physical location. It is obvious that in a certain stage in the dismantling process, it might be difficult or impossible to reach the extremes for performing in-situ measurements or for taking samples. In this case, it is necessary to **foresee the measurements at a later stage** in the project and to **update the existing data analysis** and post processing. Just ignoring the information might result in unacceptable uncertainties. Nevertheless, performing a radiological characterization program in two or three stages can be efficient and effective to tackle areas with higher uncertainties. Historical data, sometimes gathered with different objectives (e.g. analysis liner samples) can optimize the characterization process.

Decommissioning is a **multi-disciplinary operation**. The involvement of **specialized staff** performing the next stages of the decommissioning project has proved to be beneficial. Technical feasibilities/constraints in the next stages might strongly influence the initial characterization program. The use of 3D models and corresponding **detailed 2D maps** of all horizontal cross-sections through the biological shield grid has **significantly improved common understanding** and enhanced **decision taking**.
Measurement and data analysis show that no radioactive waste is expected to be generated during actual decommissioning of this activated biological shield. This is mainly due to the reactor design, with a neutron shield tank (NST) surrounding the reactor vessel, which shielded the concrete near the reactor core from high neutron fluxes during operation. On the contrary, a large volume of very low level activated material, due to the upward neutron streaming effect towards the reactor pool and walls, is inevitable and needs to be disposed to a licensed category 1 disposal site for classical hazardous materials in Belgium.

Acknowledgments

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