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The reactor pressure vessel surveillance program of the Belgian nuclear power plants

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Abstract. This paper presents an overview of the Reactor Pressure Vessel surveillance program as it is applied to the seven Belgian Pressurised Water Reactors (PWR). The first part of the paper recalls the original objectives of the surveillance program, detailing the number of capsules that were installed per reactor, the number that have been retrieved so far and discusses the type of dosimeters installed. The second part is a description of the experimental techniques applied to determine the neutron fluences, a work which is performed in the reactor dosimetry laboratory at SCK CEN and is based on qualified activity measurements. Finally, the paper ends by briefly presenting the calculation scheme developed at Tractebel and by providing a statistical analysis of the C/E values over the ~ 400 in-core dosimeters and 800 ex-core dosimeters that have been analysed to date. The method offers satisfactory results with an average C/E on all dosimeters close to 1.0 and a standard deviation comprised between 8 % and 10 % depending on the dosimeter considered.

1 Introduction

In Belgium, the generation of electricity using nuclear power is approaching its end. The first two units have already shut down in October 2022 and February 2023, respectively, and the next three are scheduled, by law, to be permanently shut down in 2025. The final pair have the potential for life extension exception, subject to a government decision expected by the end of the year. The safe operation of these reactors has been and is still guaranteed by the thorough application of the Reactor Pressure Vessel (RPV) surveillance program as designed by Westinghouse in the beginning of the civil nuclear era.

It is in this context of the approaching end of generation that this paper has been written, with the objective of providing a testimony on the dosimetry techniques applied to measure and to evaluate the fast neutron fluence in the dosimeters of the seven Belgian reactors.

After presenting a general overview of the pressure vessel surveillance program, the paper will focus on the experimental techniques applied to measure the fast neutron fluence, which includes:

- Standard neutron dosimetry reactions;

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- A dedicated technique for ^{93}Nb(n,n')^{93m}Nb X-ray measurements, which is of specific importance for long irradiations;
- Determination of the neutron flux gradients in the RPV capsules by measuring the activity of steel samples;
- Evaluation of the thermal neutron flux by re-assessing the Co content in different Co dosimeters;
- ex-vessel neutron dosimetry in Doel 1 and Doel 2.

Then, the paper will present the Belgian industry standard fast fluence calculation route, which was established by Tractebel in the years 2000, and how it was validated on basis of the exceptionally rich experimental database comprising about 400 in-core dosimeters and more than 800 ex-core dosimeters.

2 The Belgian PWR fleet

Belgium has two nuclear power stations operated by Engie Electrabel that are respectively located in Doel (Flanders region) and in Tihange (Walloon region). These sites total seven Pressurized Water Reactor (PWR) units whose characteristics are summarized in Table 1. The commissioning of these reactors was done in the period 1975-1985. From the seven units, two have been recently shut down (Tihange 2 and Doel 3), while Tihange 1 and Doel 1&2 are scheduled to be definitively stopped in 2025, and an extended lifetime until 2035 for Tihange 3 and Doel 4 is being discussed between Engie Electrabel and the Belgian government.

Table 1. Main characteristics of the seven Belgian pressurized water reactors.

Unit	Commissioning year	N° of loops	Initial Thermal Power [MW]	Thermal Power after Power Uprate [MW]	NSSS[†] supplier	Expected shutdown year
Doel 1 Doel 2	1975	2	1192	1311	ACEC Cockerill Westinghouse	2025
Tihange 1	1975	3	2652	2785	ACEC Cockerill Framatome	2025
Doel 3 Tihange 2	1982/1983	3	2775	3054	ACEC Cockerill Framatome	Shutdown in 2022/2023
Doel 4 Tihange 3	1985	3	2988	-	ACEC Cockerill Westinghouse	2035

3 The Reactor Pressure Vessel surveillance program

3.1 Surveillance capsules

The surveillance program for the Belgian reactors was designed by Westinghouse following the recommendations formulated in ASTM E-185 [1]. As is prescribed by these guidelines, the seven Belgian reactors were equipped from the start of operation with

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[†] Nuclear Steam Supply System.

Westinghouse standard surveillance capsules hosting various types of fracture mechanics specimens made from materials representative of the reactor vessel. In addition to these mechanical specimens, the capsules included a series of neutron dosimeters aimed at measuring the neutron fluences. The capsules were fixed at different azimuthal positions on the outer shell of the thermal pad.

Each capsule comprises four neutron dosimeter blocks distributed at different heights. All capsules were equipped with the three standard fast dosimeters Fe, Ni and Cu, while the two newest units additionally included dosimeters made of Nb and Ti. A separate block containing the fissile dosimeters ^{238}U and ^{237}Np was present. In addition to the fast neutron dosimeters, some capsules were equipped with thermal neutron Al-Co dosimeters. A summary of the capsules is presented in Table 2.

Table 2. Summary of the capsules loaded in the seven Belgian reactors.

Unit	Nr of capsules at start-up	Nr of capsules replaced during operation	Azimuthal position (1)	Nr of capsules retrieved	Fast neutron dosimeters
Doel 1	6		13° 23° 33°	6	Fe, Ni, Cu, Np, U
Doel 2	6		13° 23° 33°	6	Fe, Ni, Cu, Np, U
Tihange 1	8		15° 25° 35° 45°	8	Fe, Ni, Cu, Np, U
Doel 3	4	2	17° 20°	6	Fe, Ni, Cu, Np, U
Tihange 2	4	2	17° 20°	6	Fe, Ni, Cu, Np, U
Doel 4	4	2	17° 20°	4	Fe, Ni, Cu, Np, U, Nb, Ti
Tihange 3	4	2	17° 20°	4	Fe, Ni, Cu, Np, U, Nb, Ti

(1) The azimuthal positions have been reduced to the first 45° sector.

The removal calendar of the capsules has been determined by Tractebel Engie with the objective of covering a vessel-equivalent irradiation time representative of at least sixty years of full power operation. This objective could be met for all reactors and was in fact largely exceeded for the last capsules retrieved from the four most recent units for which vessel-equivalent time of more than 75 years could be achieved as is shown in Figure 1.

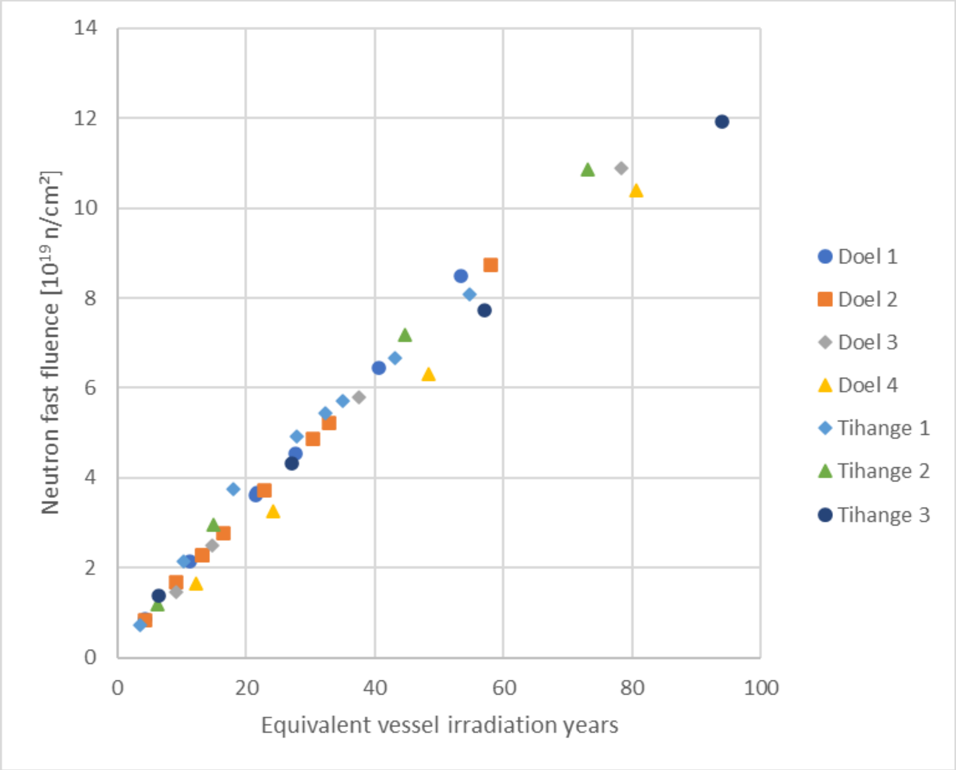

Figure 1. Scatter plot between the fast fluence and the vessel equivalent irradiation time reached by each capsule. Each data point corresponds to a specific capsule.

3.2 Ex-vessel dosimetry

In addition to the surveillance program, ex-vessel dosimeters were installed at both Doel 1 and Doel 2 before the start of cycle 8 (1983). The rationale for installing ex-vessel dosimeters was the following:

- They permit the monitoring of the fluence after the last capsule has been retrieved from the reactor.
- Since they are located outside the vessel, they enable validation of the extrapolation of neutron fluences measured in the capsules to values at the RPV.

At first, the ex-core dosimeters were contained in tubes that were suspended in the reactor cavity at various azimuthal positions. The tubes were fixed to the reactor pit wall close to the reactor mid-plane. In recent years SCK CEN developed a system in which aluminum tubes are fixed to the biological shielding. Tubes containing activation dosimeters can be easily (dis)mounted from those fixed tubes. This system allows faster interventions hence decreases the radiation exposure of the installation team. Moreover it has the advantage that the dosimeters positions remain the same in consecutive dosimetry campaigns.

The ex-core dosimeters are made from Ni, Fe, Cu, Ti, Nb, ^{238}U and ^{237}Np. The dosimetry tubes are installed at 4 azimuthal positions and cover an axial distance of about 2.5 m.

4 Experimental methods

4.1 The reactor dosimetry laboratory at SCK CEN

The γ-spectrometry measurements of the activated dosimeters are performed by SCK CEN. These measurements and the conversion of the measurement results to neutron flux and fluence are based on ASTM standards. The laboratory has been accredited by BELAC (Belgian Ministry for Economic Affairs) according to norms NBN EN ISO/IEC 17025 since 2001. Extensive validation files and work instructions are part of the QA document system of the laboratory. These documents include a full uncertainty propagation calculation and a track of interlaboratory intercomparisons.

The reactor dosimetry laboratory of SCK CEN is currently equipped with five high-purity Ge detectors that are used for the γ-spectrometry measurements and one low-energy Ge detector that is solely used for X-ray measurements. Specific γ-activities of individual samples can be determined with a 1σ uncertainty < 2 %. The Ge detectors that are used for these measurements have different intrinsic efficiencies and moreover each detector has a sample positioning system covering sample-to-detector positions from a few centimetres up to several meters. Two of these detectors are coupled to a fully automated sample changer allowing continuous round the clock operation, which makes them ideally suited for measuring large batches.

A dedicated (destructive) technique is applied for the determination of ^{93m}Nb activities, which are reported with a 1σ uncertainty < 3 %. In brief this technique consists of dissolving the Nb dosimeter in a HF-HNO$_3$ mixture and preparing low-mass samples from this solution (deposits of precipitate on filter paper). The ^{93m}Nb X-rays emitted from such samples are detected with the low-energy Ge-detector that is calibrated with two independent sets of ^{93m}Nb reference deposits. The strong advantage of this technique is that X-ray self-absorption and fluorescence effects are reduced to negligible quantities. [2,3]

A Neutron Activation Analyses (NAA) laboratory is operated by the same group that is running the reactor dosimetry activities. In short this NAA technique allows the determination of the elemental composition of samples (in some cases even down to the ppb level) by irradiation of the sample in the BR1 research reactor (see e.g. [4] for a description of BR1), followed by γ-spectrometry measurements, and applying specific calibrations and analyses software. The availability of this technique offers a strong advantage because it permits, for example, the verification of material certificates, and the characterisation of unknown samples.

4.2 RPV surveillance dosimetry

The fast neutron fluence is determined from the measured specific activities of the dosimeters, using the reactor power history, and the calculated spectrum averaged cross sections (see section 5). Since some dosimeters can be off-centred with respect to the capsule axis as shown in Figure 2, additional measurements are required to determine the radial neutron flux gradient within the surveillance capsule. For this purpose, small samples are cut from the steel dosimetry blocks and from charpy's in the SCK CEN hot cells. From their measured ^{54}Mn activity the radial gradient of the fast neutron flux is determined resulting in a typical value of about 15 % per cm. This is illustrated in Figure 3, that plots the variation of the reaction rate as a function of the radial position for one of the Doel 4 capsules. Each point on the figure is the average reaction rate obtained from three steel samples. The relative uncertainty on these reaction rates is < 1%.

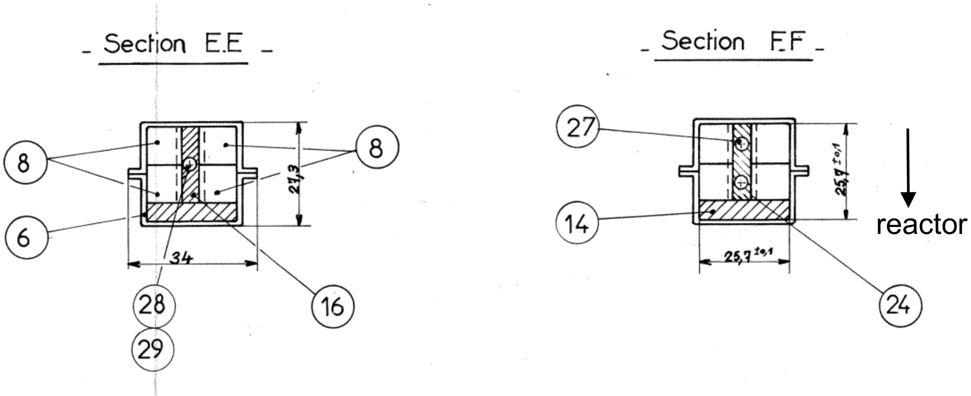

Figure 2. Horizontal crosscuts of a surveillance capsule at different altitudes. Dosimeters are labelled by the numbers 28, 29, 24 and 27. The dosimeters located at section “FF” are off-centered.

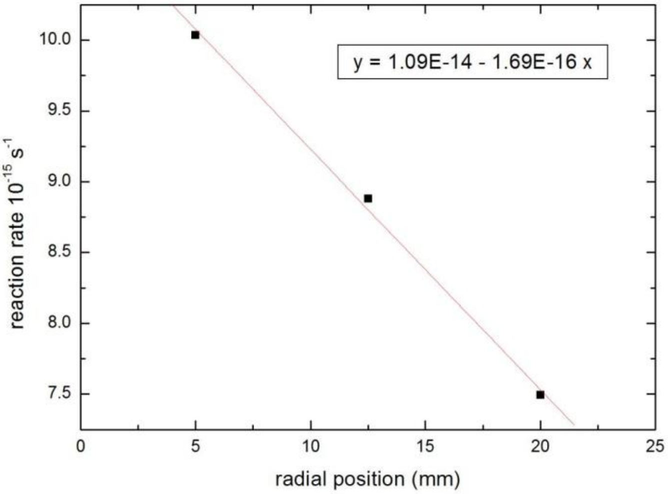

Figure 3. Radial gradient of the reaction rate observed in one of the surveillance capsules.

4.3 Thermal neutron dosimetry

The focus of the RPV surveillance program has historically been on fast neutrons, as these are mainly responsible for the damage in the RPV steel. In view of the future decommissioning of the reactors, it became important to validate the neutronic calculation tools (see next chapter) also for thermal and epithermal neutron flux calculations. Indeed, predicting the thermal (and epithermal) neutron induced activities in structural materials is required for waste management. Therefore it was decided to re-evaluate and compile all Co dosimeters that were present in the Doel 4 and Tihange 3 surveillance capsules. Standard procedures combining the measured activities from the bare and cadmium-covered dosimeters, and using 2200 ms^{-1} cross sections and resonance integrals, were used to determine the thermal and epithermal neutron fluxes.

The outcome of this exercise led to strongly put in question the cobalt concentration reported for these dosimeters in the original material certificates (Al-0.15wt%Co). For this reason, radiochemical analyses (ICP-MS) were performed on a sample and showed a strong deviation from the certificate: Al-0.36wt%Co. The root cause of these deviations is still under investigation. NAA measurements on additional Al-Co dosimeters are considered.

5 Calculation method

5.1 The MCBEND model

The method for calculating the fast fluence at the location of the surveillance capsules or the ex-core dosimeters was established by Tractebel Engie between the years 1995 and 2000 and has remained unchanged since [5].

For each of the cycles in which a capsule or an ex-core dosimeter was present, a time-integrated 2D pin-by-pin neutron source distribution is established by performing a core criticality calculation with the reactor physics code PANTHER version 05.03.R1 [6]. The instant 3D fission source is determined for a dozen of equally spaced burnup steps and is then integrated axially to construct a 2D pin-by-pin source which is in turn time integrated over the cycle length. The axial profile of the fission source is derived from flux map acquisitions, which has the advantage to include the effect of the mixing grids on the neutron source (an effect not accounted for in the PANTHER model). The energy spectrum of the fission source is an average from the ^{235}U and ^{239}Pu fission spectra according to their respective contributions to the fission source.

Once the characteristics of the fission source have been established, it is used as the particle source in a subsequent neutron transport simulation performed with the Monte-Carlo code MCBEND version 11A ru1 [7]. For this purpose, two families of MCBEND models were designed: one for the capsules (see Figure 4) and one for the ex-core dosimetry (see Figure 5). Each of these models is limited to 1/8th of the core. They explicitly include the core, the baffle, the thermal pad with capsules explicitly represented, the vessel, the insulator and the concrete of the reactor pit. All structures (i.e. the fuel pins and the assembly grids) inside the core region are homogenized. A varying density profile is also considered based on thermal-hydraulics calculations done with PANTHER. A constant concentration of 550 ppm of boron is adopted in the model and kept constant during the transport simulations.

As shown on Figure 4, the capsules are modelled as homogenous blocks (stainless steel 304), without representing explicitly the dosimeters. Figure 5 shows the modelling of the tubes hosting the ex-core dosimeters.

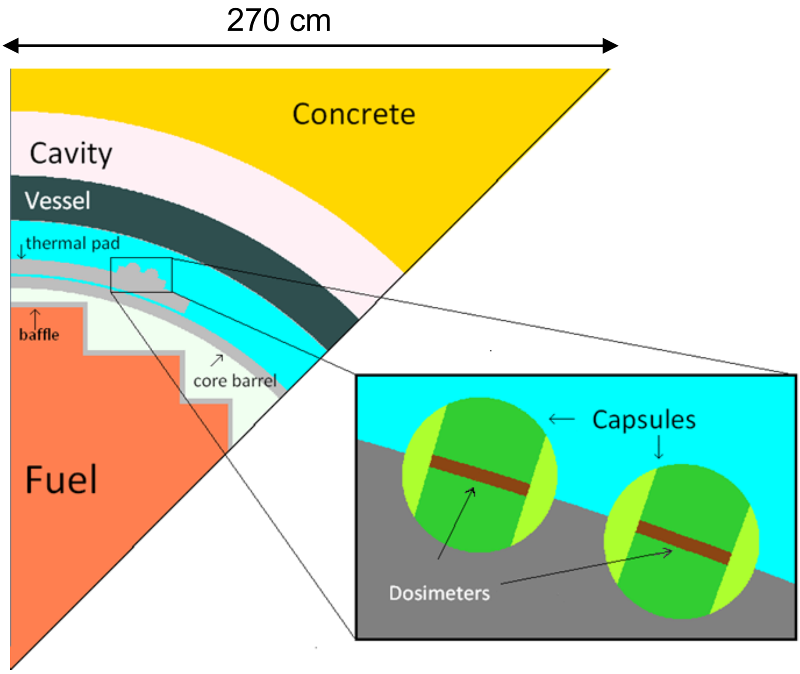

Figure 4. Horizontal cross section of one of the MCBEND models designed to evaluate the fast neutron fluence at the level of the surveillance capsules.

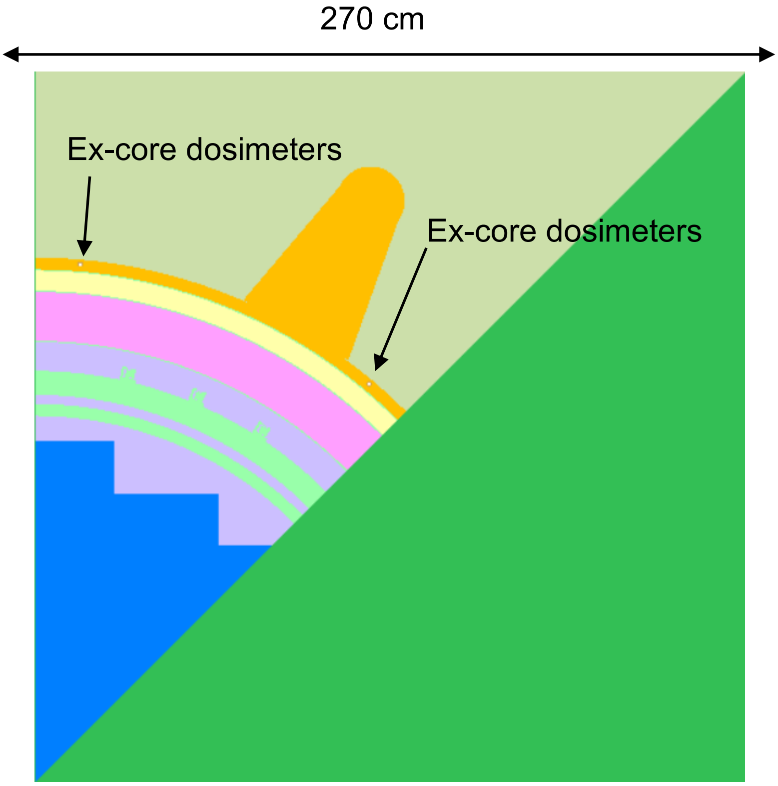

Figure 5. Horizontal cross section of the MCBEND model used to evaluate the fast neutron fluence at the level of the ex-core dosimeters guiding tubes.

Classical biasing techniques such as splitting/Russian roulette and source weighting are applied to boost the convergence of the Monte-Carlo calculations. The biasing parameters have been automatically generated by a specific module of MCBEND named MAGIC which solves the adjoint transport problem in diffusion theory using a simplified geometry.

The neutron transport library used is UKNDL. Even though this library is clearly outdated, it was kept since it provides satisfactory results for this specific application as described in the next section. Testing a more recent library (e.g. JEFF 3.3) could be the object of a future publication.

5.2 Uncertainties on calculated fluences

An uncertainty assessment of the calculated fast fluences at the capsule level has been realized when the methodology was established in 1995 [5]. The outcome of this assessment is summarized in Table 3.

Table 3. Uncertainty budget of the calculated fluence at the capsule level.

Component	Uncertainty (%)
Water density profile	6
Neutron source spatial distribution	8
Shape of the fission spectrum	6
Stochastic uncertainty	1

A total uncertainty of 12 % was obtained by quadratically combining the four uncorrelated components. Note that this uncertainty assessment could be revisited using more modern techniques and quantify some components that had been overlooked in the original exercise such as the uncertainty on cross section data.

No quantification of the uncertainties for the ex-core dosimetry has been realized so far.

6 Comparison between calculations and experiments

6.1 Surveillance capsules

The validity of the theoretical models is demonstrated by comparing the calculated fast neutron fluence (C) with the measured or experimental (E) fast neutron fluence.

Figure 6 shows the C/E ratios obtained for the 397 Fe, Ni and Cu dosimeters that have been retrieved from 35 capsules. Table 4 summarizes the C/E statistics obtained by grouping all fast dosimeters: Fe, Ni, Cu plus Nb and the fissile ones. Note that the fissile dosimeters were not systematically analysed, which explains their relatively low number (6 + 14) compared to the total number of capsules (35). Consistent results are obtained for the three main dosimeter types Fe, Ni and Cu, with an average C/E between 1.02 and 1.03, and standard deviations between 8 % and 10 %, justifying their choice as standard dosimeters in the Belgian context.

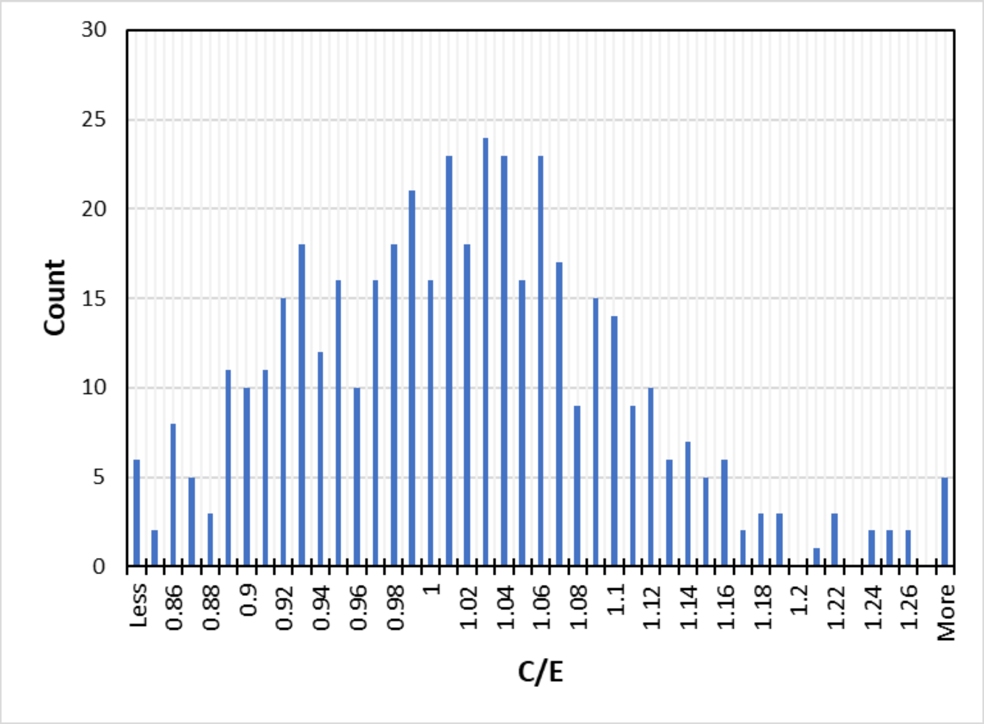

Figure 6. C/E distribution obtained from 397 Fe, Ni and Cu dosimeters retrieved on the surveillance capsules.

Table 4. C/E statistics for the different dosimeters.

C/E	Fe	Ni	Cu	Np	U	Ti	Nb
Average	1.032	1.033	1.021	0.930	0.916	0.913	0.952
σ	0.081	0.100	0.085	0.074	0.082	0.074	0.045
σ [%]	7.9	9.7	8.3	7.9	9.0	8.1	4.7
Count	150	109	138	14	6	19	10
Min	0.857	0.850	0.754	0.829	0.858	0.753	0.876
Max	1.293	1.339	1.412	1.061	1.095	1.027	1.006

The two fissile dosimeters show an average C/E below unity (0.92). The Ti and Nb dosimeters also show C/E below 1. An interesting feature regarding the Nb dosimeters is the low dispersion on the C/E (1σ = 4.7 %) which must however be put in perspective given the low number of such dosimeters analysed. Table 5 presents the statistics obtained by considering a reactor-wise base, which is limited to the three standard dosimeters (Fe, Ni, Cu). An average C/E value close to one is observed for five units (Doel 1, Doel 3, Doel 3, Tihange 2 and Tihange 3) and slightly above unity for Doel 2, Doel 4 and Tihange 1.

Table 5. C/E grouped per reactor obtained from Fe, Ni and Cu. The FLEET column gives the mean value treating each reactor as a single observation.

C/E	Doel 1	Doel 2	Doel 3	Doel 4	Tihange 1	Tihange 2	Tihange 3	FLEET
Average	1.017	1.057	0.987	1.063	1.081	1.001	0.989	1.023
σ	0.083	0.092	0.069	0.070	0.096	0.065	0.073	0.036
σ [%]	8.1%	8.7%	7.0%	6.6%	8.9%	6.5%	7.4%	0.035
Count	52	51	60	46	74	54	60	7
Min	0.754	0.913	0.850	0.918	0.908	0.896	0.859	0.987
Max	1.222	1.277	1.135	1.258	1.412	1.155	1.139	1.081

6.2 Ex-vessel dosimetry

Figure 7 shows a histogram of the C/E ratios obtained from the 886 ex-core dosimeters that have been analysed so far. Just as for the dosimeters retrieved from the surveillance capsules, the statistical distribution of the C/E ratios has a bell-shape curve, with an average of 1.066 and a standard deviation of 0.123. These results are slightly less consistent than the ones obtained from the dosimeters retrieved from the surveillances capsules, which can be explained by a higher source of uncertainty, and notably the positioning of the dosimeters within the guiding tubes.

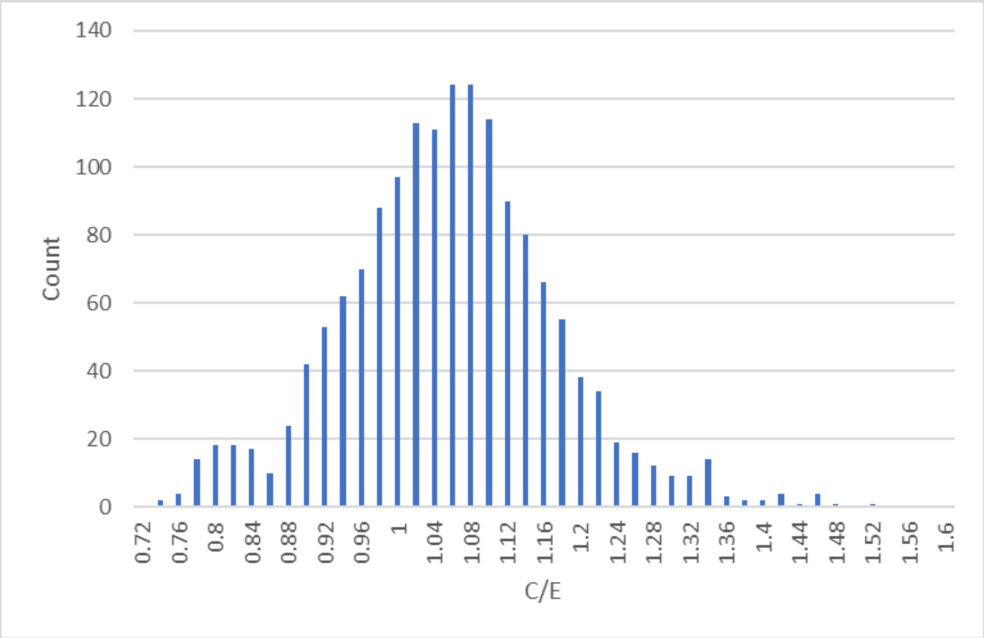

Figure 7. C/E distribution obtained on the 886 ex-core dosimeters.

Table 6 provides the statistics obtained for the different dosimeter types. It is worthwhile noting that among the detectors which are statistically relevant in terms of number, Nb exhibits the lowest dispersion, followed by Fe and Ni.

Table 6. C/E statistics for the different ex-core dosimeters.

C/E	Fe	Ni	Cu	Np	U	Nb
Average	1.070	1.053	1.071	0.843	1.038	1.089
σ	0.119	0.142	0.045	n.a.	0.156	0.080
σ [%]	8.61%	13.52%	4.20%	n.a.	15.07%	7.37%
Count	508	248	4	2	20	104
Min	0.79	0.75	1.00	0.84	0.85	0.90
Max	1.61	1.52	1.11	0.85	1.42	1.25

Finally, Table 7 presents the statistics obtained by grouping the dosimeters on a reactor-wise basis and shows consistent results between both units.

Table 7. C/E statistics grouped per reactor.

C/E	Doel 1	Doel 2
Average	1.066	1.067
σ	0.108	0.137
σ [%]	10.16%	12.89%
Count	452	434
Min	0.78	0.75
Max	1.61	1.52

7 Conclusions

This paper provides a comprehensive overview of the RPV surveillance program as it has been applied in Belgium for almost fifty years. Whilst the experimental approach has remained essentially unchanged since its inception, some notable improvements have been added. These include: a new method to measure the activity of Nb dosimeters; an experimental technique allowing the estimation of the fast neutron fluence gradient within the capsules and, as such, the fast neutron fluence on the capsule centreline; and the introduction of ex-vessel dosimetry.

The analysis of almost 400 dosimeters from the surveillance capsules and more than 850 ex-core dosimeters has enabled the validation of the Tractebel calculation scheme based on PANTHER core physics calculations and MCBEND neutron transport simulations. An average C/E of 1.03 (σ = 0.088) was obtained from the dosimeters extracted from the surveillance capsules and of 1.066 (σ = 0.123) for the ex-core dosimeters.

For the surveillance capsules, the best agreement (in terms of dispersion) between the experiment and the calculation was observed for the Fe dosimeters, confirming their choice as reference dosimeters in the Belgian context. Regarding the ex-vessel dosimetry, the dosimeters providing the best performance are the Fe and Nb dosimeters. As such, these could be selected as the reference dosimeters for future reactors.

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