Estimation of neutron fluence distribution within German PWR components for decommissioning studies

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Abstract. On 15 April 2023, the last German NPPs were shut down. The final shutdown is followed by a post-operational phase in which measures can be carried out to prepare for the NPPs dismantling and decommissioning. One of the tasks in preparation for the dismantling is to acquire accurate knowledge of the activation distribution within the reactor components. The activation levels depend on the neutron fluence within the reactor. Therefore, to achieve this task, a 3D detailed Monte Carlo model of a German PWR has been developed to calculate the neutron fluence characteristics (spectrum and distribution) within the components of such nuclear reactors type. The reliability of the calculations was validated via metal foil-activation measurements carried out in two active PWRs. This paper presents the Monte Carlo model and describes the calculations and measurement procedures.

1 Introduction

The German Atomic Energy Act was amended in 2011 to phase out the use of nuclear energy for the commercial production of electricity. Consequently, the last German Nuclear Power Plants (NPPs) were shut down on 15 April 2023. After their final shutdown, the NPPs should be decommissioned and dismantled [1]. To dismantle the NPPs in a safe, economical, and timely manner, decommissioning studies have to be performed for each of the shutdown NPPs. The present study focuses on the decommissioning of a German Pressurized Water Reactor (PWR), which is the most common NPP type in Germany.

One important task in decommissioning studies is estimating the neutron activation distribution within the NPP components that have emerged during its lifetime operation. Such knowledge can significantly minimize the radioactive waste and contribute to the safety of the operating personnel and the general public. The extent and levels of activation in an NPP depend on the neutron fluence within its components. Hence, neutron fluence is one of the most fundamental parameters on which every decommissioning study is based. This study is, therefore, aimed to estimate precisely the neutron fluence characteristics (spectrum and distribution) within a German PWR via a series of Monte Carlo simulations. In that regard,

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a detailed 3D model of a German PWR was developed, and the neutron fluence was calculated and validated based on metal foil-activation measurements.

The neutron fluence calculations and their validation are the subjects of this paper. Section 2 gives a brief description of the German PWR model and specifies the neutron source used for the Monte Carlo calculations. Section 3 presents the foil-activation monitors, their locations inside the reactors, and describes the activation measurement methodology. Section 4 describes the Monte Carlo simulations setup for the neutron fluence evaluation and the activation calculations in the monitors' locations. In Section 5, the results of the Monte Carlo simulations and the measurements are presented and discussed. Section 6 gives some conclusions and outlook for the next step of the study.

2 Reactor model and neutron source specification

2.1 Reactor model description

The German PWR was simulated using the Monte Carlo code MCNP (version 6.2) [2]. The reactor was modeled in detail based on technical drawings of the studied PWR plants. The material densities and compositions used in the model were based on the available documents from the plants. A schematic view of the model is shown in Fig. 1. The simulated model extends from the reactor core through the Reactor Pressure Vessel (RPV) and its internals into the surrounding components, which include the thermal insulation and the interior and exterior biological shields. The lower internals assembly (i.e., the stool, flow distribution plate, lower core plate) and the upper internals assembly (i.e., grid plate, support column, upper core support, top plate), as well as the foot and head regions of the fuel assemblies composing the reactor core, were modeled in detail to represent the ex-vessel neutron leakage accurately. The fuel assemblies' body, i.e., the active core region, was modeled as a homogeneous cell. Up to the outer wall of the exterior biological shield, the reactor exhibits quarter symmetry in the radial direction. Therefore, the model represents only a quarter of the reactor. The neutron source was defined in the active core region of the model as a fixed source.

2.2 Neutron source evaluation

An accurate evaluation of the neutron source is very important for the reliability of the neutron fluence calculations outside the reactor core. It is generally known that the flux integral over the reactor's operating time (one/several operating cycles or the entire lifetime operation) is required to determine a fixed neutron source for the MCNP calculations. Assuming that the transport properties inside the core change only slightly for the relevant energy range and outside the core they do not change at all, the flux can be determined by the transport equation using a time-dependent fission source [3-4].

The fixed source was determined and specified in the model as a pin-by-pin distribution, where each pin was divided into 32 axial layers. Such a fine pin-wise specification, especially in the outer core region (i.e., the outer fuel assemblies), is essential to obtain accurate neutron fluence results outside the core. The pin-wise distribution and the total source were determined based on: (1) the time-dependent fission neutron yield from each of the fission isotopes for each fuel assembly in 32 axial layers, (2) the pin power distribution (16x16 or 18x18 grid, depending on the studied German PWR type). These data were provided by the owners of the studied PWR plants based on calculations performed using the SIMULATE5 code [5]. The source pin-wise distribution was determined by the fission neutron yield in each pin axial layer weighted by the pin power distribution. The total source was determined by the total summation of all pin sources. The energy distribution needed for the source term was taken as the ^{235}U fission spectrum only. The ^{235}U fission spectrum is considered sufficient because it differs only slightly from the other fission isotopes spectra.

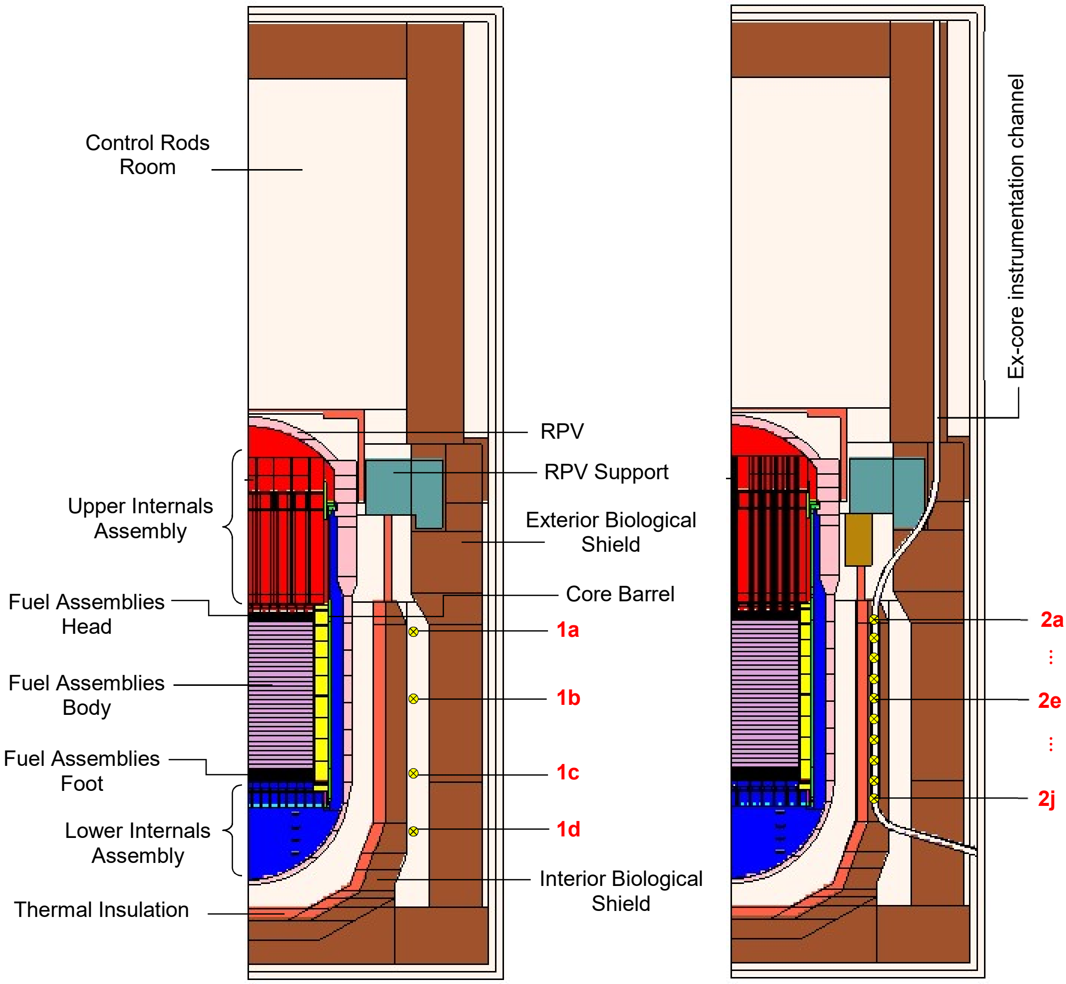


Fig. 1. MCNP model of a German PWR (in red: activation monitors locations).

3 Metal foil-activation measurements

Metal foil-activation measurements have been successfully used in reactor dosimetry for many years [6]. It is an ideal method for collecting information on the neutron fluence in an active reactor. When neutrons activate a metal foil, it emits characteristic energy gamma rays, which can be counted and related to the neutron fluence incident on the foil [7]. Therefore, foil activation measurements can provide sufficient information on the neutron fluence needed for validating, tuning, and optimizing the Monte Carlo calculations.

In this study, Metal foil-activation measurements were performed in two active German PWRs. In one reactor (denoted as reactor No. 1), the activation monitors were placed outside the interior biological shield in different angular positions (monitors 1a-1d). In the second reactor (denoted as reactor No. 2), the activation monitors were placed along one of the empty ex-core instrumentation channels (situated at the angle of 84°), passing through the reactor interior and exterior biological shields (monitors 2a-2j). The locations of the monitors are illustrated in Fig. 1. Each of the activation monitors, pictured in Fig. 2, is made of an aluminum box filled with thin metal foils packed in Kapton tape. The metal foils have a thickness of 0.1 mm and a size of 5 x 5 mm to 10 x 10 mm or 10 mm diameter. Titanium, Iron, Nickel, Copper, Zinc, Indium, and Tin were chosen as the materials for the activation foils. Each of these materials provides information on different neutron energy spectrum ranges (i.e., thermal, epithermal, and fast energy range), which together give a full picture of the energy distribution of the neutron fluence.

Fig. 2. The activation monitor.

The activation monitors were placed in the reactors during the annual refueling outage and irradiated for one fuel cycle (approximately one year). After the activation monitors were removed and recovered, the activation of the metal foils was measured with a coaxial N-type high purity germanium (HPGe) detector (GMX30 P4-76-C-S, 30% efficiency, Ortec-Ametek, Oak Ridge, Tennessee, U.S.A.). The detector was calibrated with a solid $^{152/154}$Eu standard point source. The samples were placed directly on the detector with measurement times between 1 and 60 hours, dependent on the respective activation and measurement sensitivity, that is 1-2 hours for Zinc, and Indium; 2-7 hours for Iron, Nickel, and Tin; and 5-60 hours for Titanium, and Copper. The measurements were started 2 weeks after recovery of the samples and took about 2 months for all samples. The activities were background corrected and recalculated to the date of recovery.

4 Monte Carlo calculations

The developed Monte Carlo model was used to calculate the reactor's neutron fluence characteristics (spectrum and distribution) resulting from its lifetime operation. For these calculations, the neutron source was integrated over the total operating period of the reactor. The neutron fluence spectrum was calculated in 640 neutron energy groups in several segments of the RPV and surrounding components. The neutron fluence distribution was calculated in a fine rectangular mesh of 10x10x10 cm^3 and a more coarse rectangular mesh of 30x30x50 cm^3 covering the entire reactor model. The final fluence distribution map was then obtained by merging the results of both of the meshes based on the values' uncertainties. The knowledge of these neutron fluence characteristics are necessary for the ensuing activation calculations [8].

In addition to the neutron fluence characteristics, the activation of the metal foils was evaluated and compared against the measured values. According to the measurements, the activation were calculated within a small sphere (with a radius of 2.5 cm) around each of the monitor's estimated axial locations in the studied reactors. For these calculations, the neutron source was integrated over the irradiation time of the monitors in each of the reactors. The activation was calculated using the International Reactor Dosimetry and Fusion File (IRDFF-II) data library [9] for all the metal foils reactions of which the data is available in the library. For the reactions of which the data is not available in the IRDFF-II library, the ENDF/B-VII.1 library [10] was used instead.

As is obvious from the scale of the model, the significant challenge in the aforementioned calculations is obtaining reliable statistical results, especially in segments/areas far from the neutron source (i.e., the reactor core). To overcome this challenge and achieve high accuracy results in a reasonable computational time, the weight-windows variance reduction technique was applied in the calculations. The implemented technique is based on the built-in MCNP cylindrical mesh-based weight-windows generator [11]. The cylindrical mesh grid was set to cover the entire model and consists of a fine segmentation that captures the main material differences encountered by the neutrons along their path from the core to the outer boundaries. The weight-window parameters (WWPs) were generated and optimized for each of the aforementioned calculations. The WWPs optimization was performed via iteration sequence, starting with a run with reduced concrete density followed by runs with gradually increased concrete density up to the nominal value. Once the appropriate WWPs were achieved, the neutron fluence characteristics and the monitors' activations were calculated.

5 Results and discussion

The results of the Monte Carlo calculations and measurements are presented in the following figures. It should be noticed that the results are normalized to the one source neutron (denoted as prim-n). Fig. 3 and Fig. 4 present the activation calculations vs. measurements for the ^{113}In (n,ɣ) ^{114m}In reaction. Fig. 3 shows the results obtained for the monitors placed at the angular positions of 26^o, 36^o, 45^o, and 56^o outside the interior biological shield of reactor No. 1. As can be noted, a good agreement was obtained between the calculated and measured activities. The C/E (Calculation to Experiment ratio) values for the Indium foil activation are between 0.75 and 1.46. The same C/E values range was obtained for all the analyzed metal foils. It should be noted that in several monitors, some of the metal foils got destroyed due to the conditions in the reactor. Therefore, getting an accurate measurement value for these foils was impossible (e.g., for the Indium foil in monitor 1a at the 56^o location). Fig. 4 shows the results obtained for the monitors placed along the ex-core instrumentation channel (situated at the angle of 84^o) of reactor No. 2. Since the monitors were manually lowered down from the top of the channel, their exact location inside the channel space has some uncertainty (the channel has a diameter of 15 cm). Therefore, the calculations were performed for three possible locations of the monitors: close to the inner wall of the channel (In), in the middle of the channel (Mid.), and close to the outer wall of the channel (Out). As can be noted, a good agreement was obtained between the calculation and the measurement values. The calculation values of the three possible locations are within the range of the measurement uncertainties. The same observation was noticed for all the analyzed metal foils in the channel.

It should be noted that the agreement between the calculated and measured activities depends on many parameters, such as the geometry approximations and material definitions, the neutron source definitions, the cross-section data, the estimated location of the monitors, and the measurement uncertainties. The calculations and the measurements methodology used in this study were optimized to minimize the impact of these parameters and achieve high accuracy results. The reactor was modeled in great detail, the most updated cross-section data were used, and high-fidelity simulations and measurements were performed.

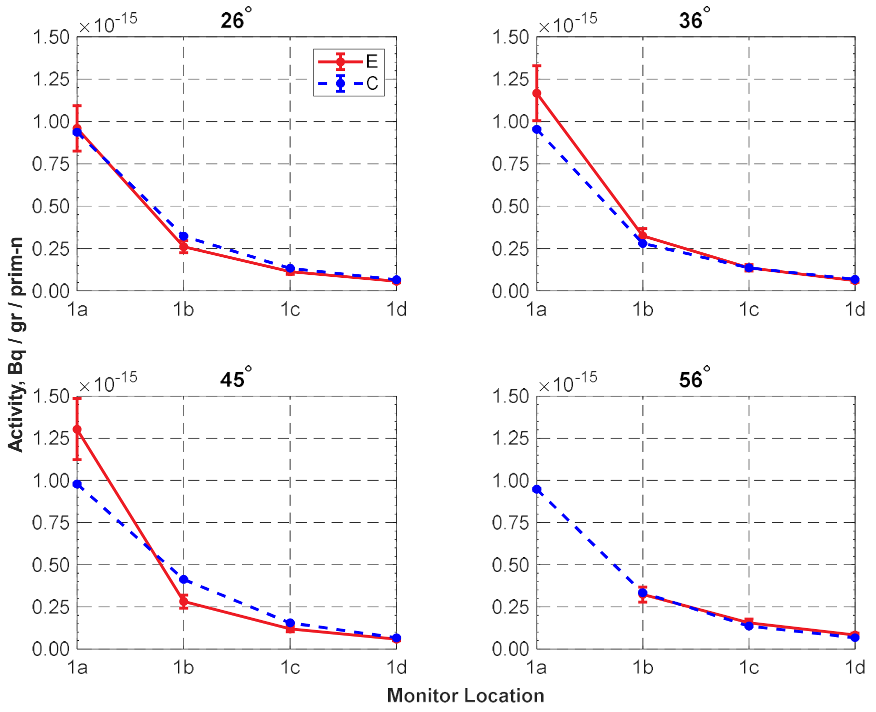


Fig. 3. Activation calculations (C) vs. experimental measurements (E) for ^{113}In (n,γ) ^{114m}In in reactor No. 1.

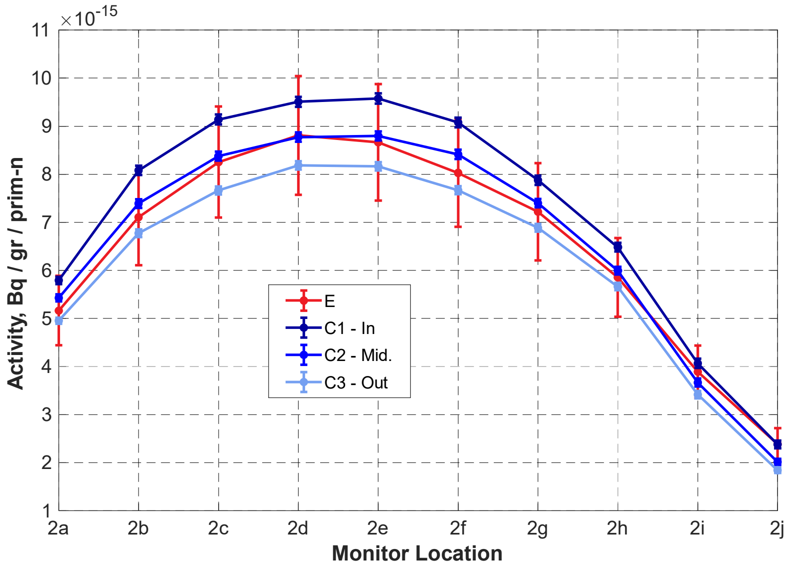
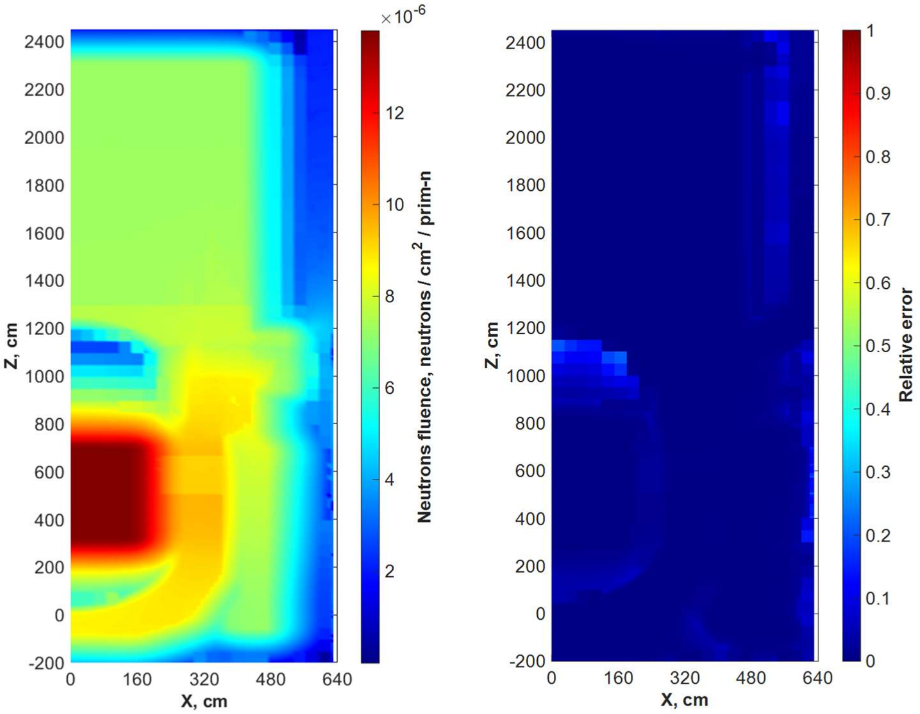


Fig. 4. Activation calculations (C) vs. experimental measurements (E) for ^{113}In (n,γ) ^{114m}In in reactor No. 2.



a. Neutron fluence **b.** Relative error

Fig. 5. Neutron fluence distribution in a German PWR.

The German PWR's fluence distribution map resulting from its lifetime operation is presented in Fig. 5. The figure shows a slice of the fluence distribution map. The map was obtained mainly from the fine mesh calculation of 10x10x10 cm^3. The outer voxels in the reactor geometry and a few voxels inside the RPV (above the top plate and below the flow distribution plate) were replaced by the results from the more coarse mesh calculation of 30x30x50 cm^3 due to the poor statistics in these areas. The combination of both mesh calculations results in a final map with sufficient statistical significance.

6 Summary and Conclusions

In the framework of German NPP decommissioning studies, a 3D detailed Monte Carlo model of a German PWR has been developed based on detailed technical drawings. The model was used to estimate the reactor's neutron fluence characteristics (spectrum and distribution) necessary for the ensuing activation calculations. Measurements were performed in active German PWRs to validate the calculations.

The validation of the calculations was based on metal foil-activation measurements. Several activation monitors, consisting of thin metal foils, were placed in key locations in two active PWRs. The monitors were irradiated in the reactors for approximately one year. After the monitors were removed and recovered, the activation of the metal foils was measured and compared against corresponding activation calculations. The calculated metal foils activation agreed well with the measured values. This fact indicates that the developed Monte Carlo model is reliable and can be used to calculate the neutron fluence in the components of German PWRs.

The neutron fluence spectrum and distribution map resulting from the reactor lifetime operation were calculated. The neutron fluence spectrum was calculated in 640 neutron energy groups in several segments of the reactor components. The neutron fluence distribution was calculated in two rectangular meshes of 10x10x10 cm^3 and 30x30x50 cm^3, covering the entire reactor model. An optimized fluence distribution map was then obtained from a combination of the two mesh calculations. The results of all the calculations were obtained with sufficient statistics due to the implemented variance reduction techniques, i.e., the weight-window parameters and the energy cut.

In the next step of the study, it is planned to extend the reactor model and include the steam generators and other components and far-field reactor building structures. The neutron fluence levels resulting from the neutrons streaming from the reactor core to these rooms will be evaluated. Additional activation monitors will be placed in the active reactors' cavities and generator rooms to obtain additional measurement data for further validation, tuning, and optimization of the developed reactor model.

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