Activation calculation for the dismantling and decommissioning of a light water reactor using MCNP™ with ADVANTG and ORIGEN-S

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Abstract. The decommissioning of a light water reactor (LWR), which is licensed under § 7 of the German Atomic Energy Act, following the post-operational phase requires a comprehensive licensing procedure including in particular radiation protection aspects and possible impacts to the environment. Decommissioning includes essential changes in requirements for the systems and components and will mainly lead to the direct dismantling. In this context, neutron induced activation calculations for the structural components have to be carried out to predict activities in structures and to estimate future costs for conditioning and packaging. To avoid an overestimation of the radioactive inventory and to calculate the expenses for decommissioning as accurate as possible, modern state-of-the-art Monte-Carlo-Techniques (MCNP™) are applied and coupled with present-day activation and decay codes (ORIGEN-S). In this context ADVANTG is used as weight window generator for MCNP™ i. e. as variance reduction tool to speed up the calculation in deep penetration problems. In this paper the calculation procedure is described and the obtained results are presented with a validation along with measured activities and photon dose rates measured in the post-operational phase. The validation shows that the applied calculation procedure is suitable for the determination of the radioactive inventory of a nuclear power plant. Even the measured gamma dose rates in the post-operational phase at different positions in the reactor building agree within a factor of 2 to 3 with the calculation results. The obtained results are accurate and suitable to support effectively the decommissioning planning process.

1 Introduction

The boiling water reactor (BWR) Philippsburg 1 (KKP1) and the pressurized water reactor (PWR) Neckarwestheim 1 (GKN I) were shut down and entered directly the post-operational phase which is covered by the operating licence held by the nuclear power plant owner. The decommissioning of a nuclear power plant (NPP) which is licensed under § 7 of the German Atomic Energy Act, following the post-operational phase requires a comprehensive licensing procedure including in particular radiation protection aspects and possible impacts to the environment. Decommissioning requires a decommissioning licence. This will include essential changes in requirements for the systems and components and will mainly lead to the immediate dismantling of the nuclear power plant which is one of the two fundamental decommissioning strategies in Germany.

In this context, neutron activation calculations for the structural components of KKP1 and GKN I have to be carried out to predict occurring activities and to estimate future costs for conditioning and packaging. To avoid an overestimation of the radioactive inventory and to calculate the expenses for decommissioning as accurate as possible, modern state-of-the-art Monte-Carlo-Techniques (MCNP6 [1]) combined with automated variance reduction tools (ADAVANTG [2]) are used and are coupled with the activation and decay module ORIGEN-S [3, 4]. This paper presents the calculation procedure and the obtained results with a validation by measured activations and gamma dose rates.

2 Calculation procedure

As pointed out in the introduction, the calculation of the activation inventory is done by a combination of two computational methods. First of all, a MCNP™ model has to be generated (cp. chapter 3) based on technical drawings of the reactor building. Additionally, the material composition and densities of all included components have to be identified. An essential part of the Monte-Carlo-simulation is the representation of the core (chapter 3.2) to map the appropriate neutron emission. Especially for the BWR model the choice of the correct water densities plays an important role because of occurring high gradients in axial direction inside the reactor pressure vessel (RPV).

According to the expected neutron flux distribution, the model is then divided into different segments, where

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activation can be detected in relation to the local material composition.

For the first MCNP™ calculations, splitting and Russian roulette techniques were used as variance reduction tools and showed good results. Since ADVANTG was released in the second half of 2015, it is used by WTI as variance reduction tool for MCNP™ generating energy and space bias cards for the source and an adequate weight window mesh.

The results of the final MCNP™ calculation are reaction rates, neutron spectra and total neutron fluxes for the different segments. These results are used to execute an activation calculation with ORIGEN-S. The basis of this calculation are the burnup history of the reactor during its operation period and detailed material compositions including relevant trace elements to determine nuclide activities for several reference times. In a post-processing tool all information are merged to determine the specific activities of the components.

3 Calculation procedure

Exemplarily, Figure 1 shows an overview drawing and the derived MCNP™ model of KKP1. Not only neutron absorbing and moderating structures but also gaps which support neutron streaming are included. The structure is divided in some hundred segments for which activities are determined.

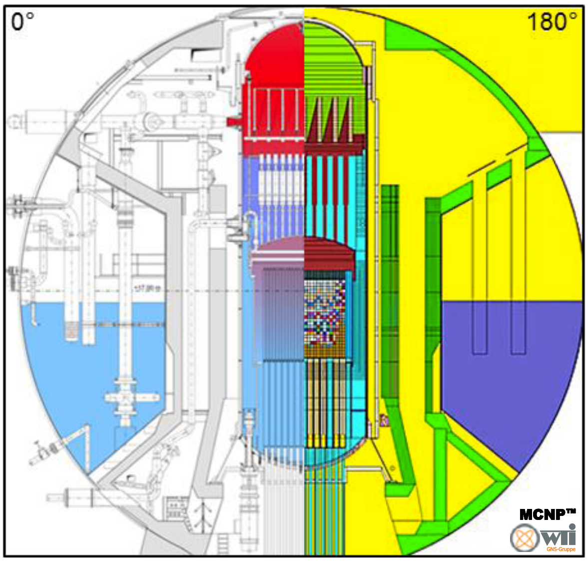

Figure 1. Overview drawing and the derived MCNP™ model

The calculation of neutron fluxes is based on the source distribution in MCNP™. An accurate core model must represent the neutron leakage and spectrum at the core boundary as far as they are relevant for activation reactions outside.

The local fission neutron source is derived from the axial burnup distribution of each fuel assembly in each cycle. Beside the burnup distribution, the water density influences the neutron leakage especially concerning the BWR-core model. Regarding the target to model an accurate representation of the core, transitions in fuel assembly shuffling influencing the neutron leakage have to be taken into account for both reactor types. Considering these aspects, steam-qualities with a local resolution are prepared for four operational phases of the BWR during full power reactor operation, where fuel shuffling schemes determine the neutron leakage from the core: (1) Start-up phase with radially flat power distribution, (2) introduction of higher enriched fuel and reduced neutron leakage, (3) introduction of spectral shift operation and transition to low-leakage fuel patterns and (4) introduction of additional orifices in second fuel assembly row with optimized flow rate in the outer fuel assembly rows. The relevant characteristics of the entire operational history of KKP1 are covered by these phases.

Regarding the PWR-core model of GKN I, the temperature distribution inside the vessel has been taken into account by varying the water density in axial direction. The boron concentration could be assumed as constant because no significant differences between the different cycles integrally appeared. Two different operational phases could be identified.

The local emission probability of neutrons is based on the local burnup-behaviour. The analysis of the provided distributions shows that fuel assemblies can be merged to different fuel regions in the case of the BWR. This simplification saves computing time because effectively only the outer fuel assembly regions influence the leaking neutron flux and thus the activation in the ex-core regions. In the case of the PWR each fuel assembly was modelled because of the asymmetric distributions during the operational phase.

The material composition of the active zone in both reactor types is based on a homogenized mixture of fuel, structural materials and water. The fissile material of the fuel consists of ^{235}U and ^{239}Pu. It is adjusted so that a k_{eff} of one in the MCNP™ model is achieved.

Because of the four operational different phases in KKP1, four separate MCNP™ runs with different core-models have to be set up. For GKN I two models resulted. These runs are reassembled in ORIGEN-S to calculate the activation considering the influences of all particular cycles.

4 Results

This chapter summarizes the results of the MCNP™ and coupled activation calculations and shows a validation of the calculated specific activations and the derived gamma dose rates.

4.1 General results of the calculation procedure

Exemplary the obtained neutron flux density distribution in the containment is shown in Figure 2 for the PWR GKN I. It can be seen that even in far distances from the neutron source, i. e. the modelled core, the total neutron flux density distribution is smooth indicating small relative errors. Additionally, the fulfilment of the ten statistical checks of the volume detector tallies in this region show that the MCNP™ calculation is converged.

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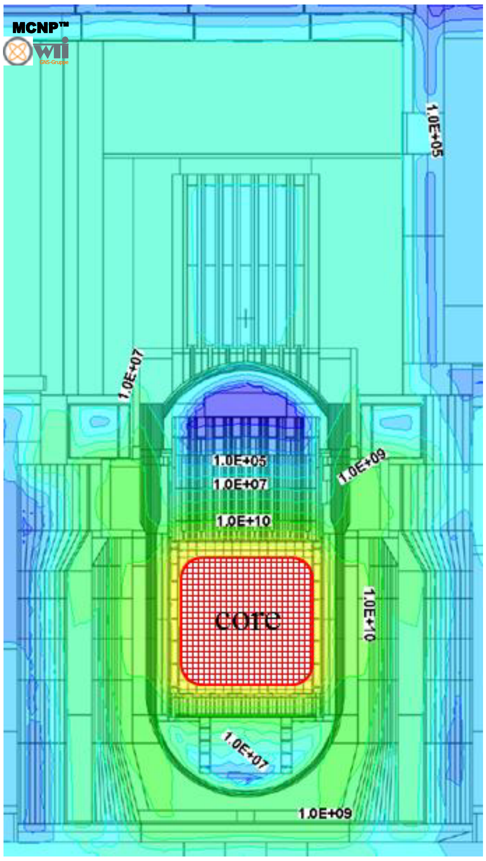

Figure 2. Neutron flux density distribution (1/(cm² s)) near the reactor pressure vessel of GKN I

To validate the calculated activities with ORIGEN-S, samples were taken inside the RPV, from the low activated regions inside the containment, and bore hole samples were taken from the biological shield. In this paper, only the results of the bore hole samples taken at the middle of the active zone and 4 m above the active zone are presented. The results are similar for PWR and BWR. The comparison of measured (M) and calculated (C) activities is presented as fraction (C/M) for the BWR KKP1.

Table 1 shows the obtained results using measured element concentrations as basis for the activation calculation. The bore hole samples contained not only the concrete and the armed concrete structure of the biological shield, but also a small probe of the RPV. Thus, the activities of both components could be measured and compared to the calculations. With regard to the publication of M. Pantelias and B. Volmert [5], where the validation of the neutron transport calculations with an in situ full-cycle foil activation for a PWR under exact-known boundary conditions is presented, about the same C/M-factors resulted as shown in Table 1. The activities of H-3 are overestimated by the calculation which can be explained by the following characteristic of H-3: After H-3 is formed it partially escapes from the components due to its high mobility. Generally, it can be seen that the activities calculated by the developed computational chain (cp. chapter 3) represent the measured activities very well.

Table 1. Comparison of measured (M) and calculated (C) activities for different bore hole samples (biological shield and RPV)

bore hole sample position	nuclide	concrete of biological shield			steel sample of RPV
		towards RPV	in the middle	towards annulus	
mid level of the active zone	H-3	7.1	6.7	-	*
	C-14	2.2	-	-	*
	Mn-54	*	*	*	2.3
	Co-60	0.9	5.6	-	1.2
	Cs-134	1.4	3.8	-	*
	Eu-152	4.1	3.0	-	*
	Eu-154	4.1	2.6	-	*
4 m above the active zone	H-3	19	2.9	2.8	*
	C-14	2.2	0.1	-	*
	Mn-54	*	*	*	6.8
	Co-60	0.9	-	4.6	1.8
	Cs-134	2.0	-	-	*
	Eu-152	5.7	-	2.0	*
	Eu-154	5.1	-	-	*

*: Not measured, -: Measured activity below detection limit

4.2 Results of the gamma dose rate calculations

In KKP1, the calculated activities are used to estimate the occurring dose rates in the post-operational phase. To validate this approach, the calculated nuclide activities are converted to photon spectra by ORIGEN-S and transferred to MCNP™ as source term. The occurring dose rate inside the containment can then be determined at different given receptor points. In the case of the validation, the gamma dose rate according to $\dot{H}^*(10)$ was measured at eleven receptor points (cp. Figure 3) after the decontamination of the primary circuit (Remark: The RPV was filled with water). Table 2 summarizes the results as C/M deviation. In the second row of Table 2 the results are shown without modification, i. e. a rotationally symmetric activation of the shroud. In this first calculation the deviations estimated for the receptor points at the level of the active zone (M4 to M8) are slightly different from the others. Including rotationally heterogeneous activation of the shroud, the results shown in the third row arise. This is due to the major contribution of the shroud to the total dose rate. Thus azimuthal varying heterogeneous activation has to be considered in dose rate calculations especially in the near region of high activated components.

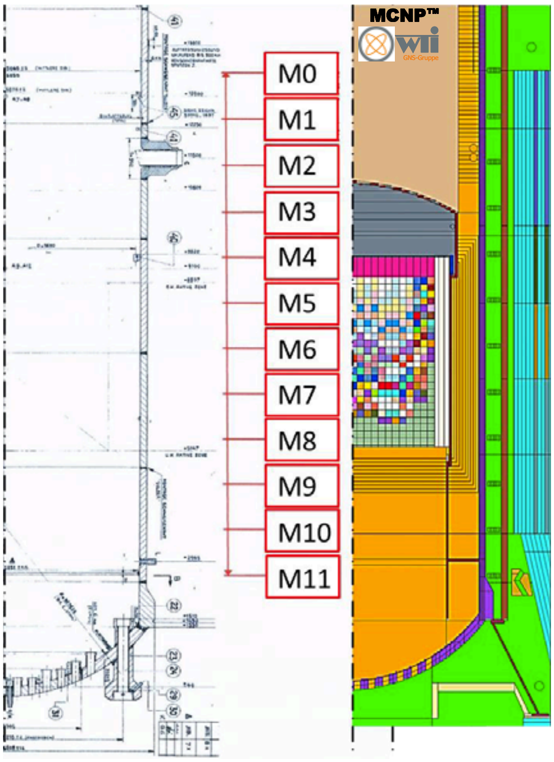

Figure 3. Receptor points M0 to M11 between reactor pressure vessel and biological shield for gamma dose rate measurements in KKP1 (BWR)

Table 2. Comparison of measured (M) and calculated (C) gamma dose rates at different receptor points in KKP1.

receptor point	C/M	C/M (corrected)
M0	1.2	1.2
M1	2.0	2.0
M2	3.0	2.7
M3	1.9	1.7
M4	3.1	2.3
M5	3.1	2.2
M6	3.2	2.1
M7	2.6	1.8
M8	4.0	3.0
M9	2.9	2.6
M10	2.9	2.9
M11	2.0	2.0

5 Conclusions and Outlook

The results presented in this paper clearly show that the developed procedure to determine activities can be applied for the calculation of the activation inventory of any type of nuclear power plants. The obtained results and even the estimated gamma dose rate predictions are very accurate and still conservative. In future, the calculated local activation inventories for a PWR will be used to calculate gamma dose rates occurring in this nuclear power plant type after shutdown. These results will be compared to measured dose rates. Based on the WTI experience, the developed calculation procedure can be used for the calculation of radioactive inventories of any type of nuclear facilities. The results are valuable contributions for the decommissioning planning and especially required for conditioning planning and the determination of package types and quantities.

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