Further development, validation and verification of the Monte-Carlo code TRAMO on ex-vessel experiments of the Greifswald NPP

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Abstract. TRAMO is a Monte Carlo code which is specially designed for geometries of nuclear power plants. In recent years, TRAMO has undergone a number of technical developments. In addition to the conversion to modern Fortran, standardized models for VVER-440 and VVER-1000 for the evaluation of monitor experiments on the surface of the reactor pressure vessel were developed, and the corresponding pre- and post-processing was automated. The new TRAMO version is continuously validated and verified. Here we present comparisons with experimental data from unit 1 of the Greifswald nuclear power plant and with MCNP calculations. Earlier calculations with TRAMO showed good agreements in the middle region of the reactor core, but significant deviations outside. A possible explanation is the wrong documentation of the experimental data – especially the heights of the monitors. Assuming a downward shift of the monitor positions by about 24cm, experimental and calculated data show very good agreement.

1 Introduction

VVER reactors have been in operation in many countries - especially in Eastern Europe and China - often for several decades. In contrast to reactors of western design, the distance from the reactor core to the reactor pressure vessel is smaller here, so that the neutron irradiation in the reactor pressure vessel is higher which could lead to its embrittlement. Therefore, obtaining accurate predictions of neutron fluence distributions in the reactor pressure vessel is an important component for safe operation, especially for the purpose of achieving lifetime extensions. This is typically also specified in regulations within the scope of licensing procedures.

To monitor the radiation exposure, the operation of the reactors was and is accompanied by monitoring measurement campaigns. In particular, a large number of measurement series are available from the Greifswald nuclear power plant. In addition, it was possible to obtain material samples from the reactor pressure vessels in the course of decommission.

The measurement results are used to validate computer programs in this area. Monte Carlo codes now play a prominent role here, since the modeling of the complex geometries

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can in principle be carried out exactly and the computing time required to obtain statistically acceptable results is becoming shorter and shorter due to very fast computing machines and various acceleration methods.

In this paper, the results of an extensive monitoring measurement campaign in the 12th cycle at unit 1 of the Greifswald nuclear power plant are subjected to renewed consideration and compared with calculations of the Monte Carlo codes TRAMO and MCNP. After a short presentation of the latest developments concerning the program TRAMO, there follows a description of the experimental setup as well as comparisons of measurement results and calculated values.

2 TRAMO & MCNP

TRAMO [1] is a Monte Carlo code developed at Helmholtz-Zentrum Dresden-Rossendorf for the calculation of gamma and neutron fluences of nuclear reactors and their internals. It can handle reactor-typical axis-parallel geometries, which means that in principle all structures of a reactor out to the biological shielding can be modeled in great detail. For the input of an external source, special routines have been implemented that facilitate the input of pin distributions within each fuel elements. Starting from a given neutron source distribution, TRAMO calculates integrated fluences and dose values for all volumes and their outer surfaces as well as spectral fluences within a multi-group approximation for volumes and surfaces determined in advance by the user.

In recent years, TRAMO has undergone extensive technical development, which is still ongoing [2,3]. Among other things, the code has been converted to a modern Fortran and the code readability has been noticeably improved. Standard input data sets have been created for both VVER-440 and VVER-1000, based on Russian standard models used for DORT calculations of the nuclear power plants. Information on geometries and material compositions was available from previous projects. The corresponding material data such as microscopic cross sections and scattering matrices were generated with the program NJOY21 [4] and based on the nuclear data library ENDF-BVII.1 [5]. MODAJ [6] was used to create the macroscopic data. There are a total of 33 compositions based on 63 nuclides. In the core, heating has been accounted for by twelve different water densities. In the cases where (α,β)-factors for thermal scattering were available, they were also used.

In order to obtain statistically reliable energy spectra of fluences, the application of variance-reducing methods is necessary. The program TRAWEI is available in the program package TRAMO for the generation of energy- and location-dependent weights within the framework of the weight-window method. Recursive Monte Carlo techniques were used to determine weight data sets for all heights, which considerably accelerated the Monte Carlo calculations. A detailed description of the technique and the program TRAWEI can be found at [7,8]. In addition, a source biasing can be used for optimizing the source with specific regard to the different result areas.

For standard calculations, a shell script can be used to automate the calculation process. The user selects the reactor type, the data library, the energy group structure and the orientation and positions of the monitors, after which appropriate input files are selected and calculations are performed.

TRAMO also has the option of a neutron energy and a gamma energy cutoff. Neutrons whose energy falls below a certain predefined value are not followed up in the calculation. This has the advantage that statistically reliable results can be obtained even in the upper energy groups. For the calculation of the neutron fluences at the monitor positions, the calculations with different cutoffs are combined.

The code package with TRAMO, TRAWEI, MODAJ and additional support programs is validated against many activation calculations on Russian VVER reactors and within international benchmarks [9-14].

MCNP is generally used as a reference code for the calculation of neutron and gamma fluences, since it is used worldwide and thus has excellent validation. A new very detailed geometry model was developed for the calculations with MCNP. In addition to all internals up to the reactor pressure vessel, this also includes the water tank located further outside.

3 The Experiment

Four VVER-440/230 units were operated at the Greifswald nuclear power plant until 1990. It was recognized early on that high neutron fluences could occur in the reactor pressure vessels, which could have led to its embrittlement. For this reason, the operation was accompanied by a whole series of monitoring processes. Among other things, there were extensive monitoring programs on the reactor pressure vessels. One of these measurement campaigns is used here to validate the Monte Carlo programs. Specifically, this is the monitoring measurement campaign in cycle 12 of unit 1 [15]. This consisted of ten equidistant height positions from -0.1cm to 250.1cm with respect to the lower edge of the active core, five approximately equidistant angles at 26.6°, 40.0°, 53.6°, 66.8°, and 80.2° and eight activation reactions. The activation reactions are ^{46}Ti(n,p), ^{54}Fe(n,p), ^{58}Ni(n,p), ^{59}Co(n,γ), ^{63}Cu(n,α), ^{109}Ag(n,γ), ^{93}Nb(n,n'), and ^{93}Nb(n,γ).

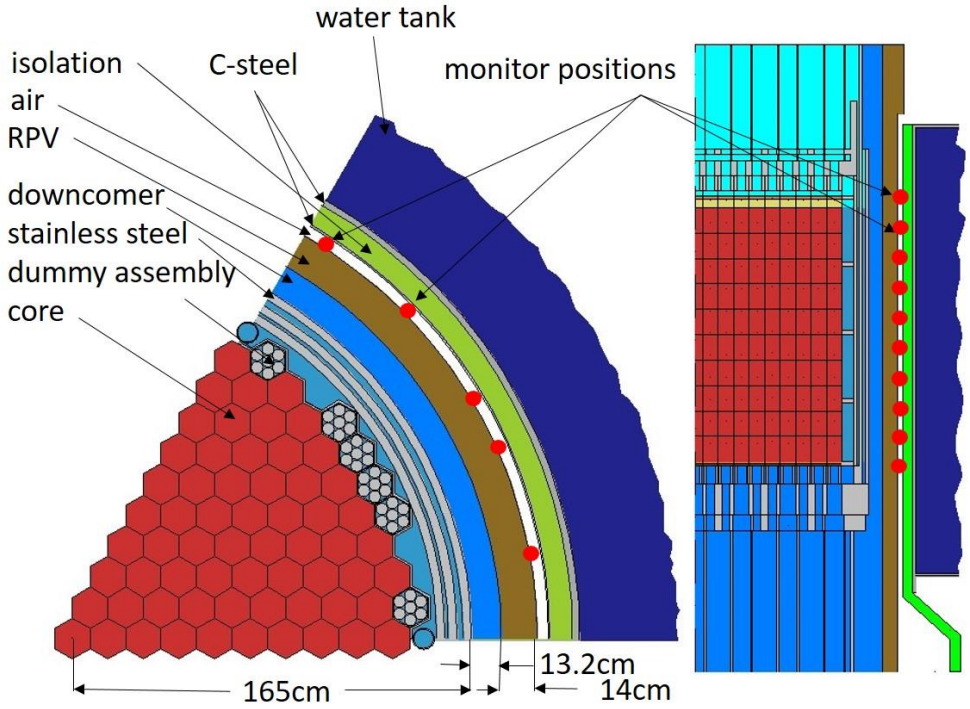

Fig. 1. Horizontal (left panel) and vertical cut (right panel) of a 60° sector of unit 1 of the Greifswald NPP. Inserted are the positions of the monitors.

Measured values are available for 393 out of 400 measurement locations. This large data set is very attractive for validating codes on VVER-440 reactors because on the one hand it covers almost the entire reactor pressure vessel within the reactor core and on the other hand the activation reactions of the monitors cover different regions of the neutron spectrum.
Fig. 1 shows schematically the geometry model used and the positions of the monitors. Due to the symmetry, the modeling of a 60° sector of the reactor is sufficient. Correspondingly, the monitor positions for 66.8° and 80.2° are shown at the angles of 6.8° and 20.2°, respectively. For geometrical details see [13]. To create the geometric models, drawings from the manufacturer and documents from the former operators of the nuclear power plant were used. The six specially marked fuel assemblies in the left panel are shielding assemblies that were used in operation cycles 12 to 15.

Calculations had already been performed for these data with an earlier version of TRAMO [16]. Although these showed reasonable agreement between experimental and calculated data in the central region of the reactor core, there were significant discrepancies above and below it. Unfortunately, this observation was not adequately discussed in the paper and it is not made clear whether all height locations were calculated at all. It was speculated that the discrepances have been caused either by inaccuracies in the neutron source data on fuel burnup or from inexact positioning of the monitors.

4 Results

All data points were recalculated using both MCNP and TRAMO. As an example, results for the reaction ^{54}Fe(n,p) are shown in Fig. 2, left panel. The reaction rates are shown in comparison with experiment, calculations with TRAMO and calculations with MCNP for all height points. Each point plotted in Fig. 2 represents the average value over the five azimuthal angles.

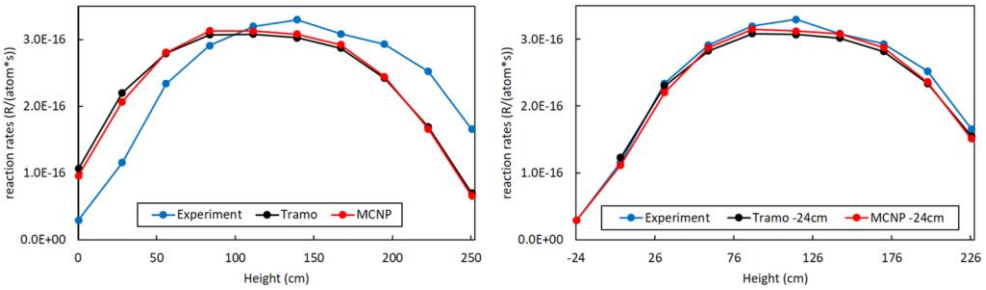

Fig. 2. Comparison of experimental and calculated reaction rates of ^{54}Fe(n,p) monitors. Left panel: Calculations with documented positions of the monitors. Right panel: Calculations with monitors shifted by 24cm downwards.

The typical shape of the curve can be seen, which approximately reflects the power distribution over the height. The agreements of the MCNP and TRAMO calculations are excellent for all heights. On the other hand, calculated and experimental values agree well only in the region of the center of the reactor core. In contrast, there are discrepancies away from the center of the reactor core The larger the distance, the bigger the deviations. It can also be seen that the calculated values below the center of the reactor core are above the experimental values, while above they are below the experimental values. This behavior is reproduced for each individual angle and each measured reaction (subject to some variability). A closer look at the shape of the curves suggests the idea that a vertical shift of

the measurement points could provide for better agreement. The mean value of the shift was determined to be 24cm downwards out of detailed analysis with MCNP. All results were then recalculated for the hypothetical geometry in which all monitors were 24cm lower. The righthand panel of Fig. 2 shows the corresponding comparison of the reaction rates. Since the lowest monitor position in the TRAMO model falls in an area not represented in detail, it is missing in the figure. Regardless, there is excellent agreement between the experimental data and both those calculated by TRAMO and now MCNP.

In Fig. 3, the C/E ratios averaged over the five angles are plotted with the shift of -24cm for three selected reactions. Here, ^{54}Fe(n,p) represents a reaction in the fast energy range. ^{93}Nb(n,n') is particularly interesting for the reactor dosimetry, as it covers the range about 1MeV well. ^{59}Co(n,γ) has the highest cross-section in the thermal range. Fig. 4 shows the C/E ratios at a height near the center of the active core as a function of the azimuthal angle. In addition, C/E ratios for all reactions are presented in Table 1 for comparison with TRAMO and in Table 2 for comparison with MCNP.

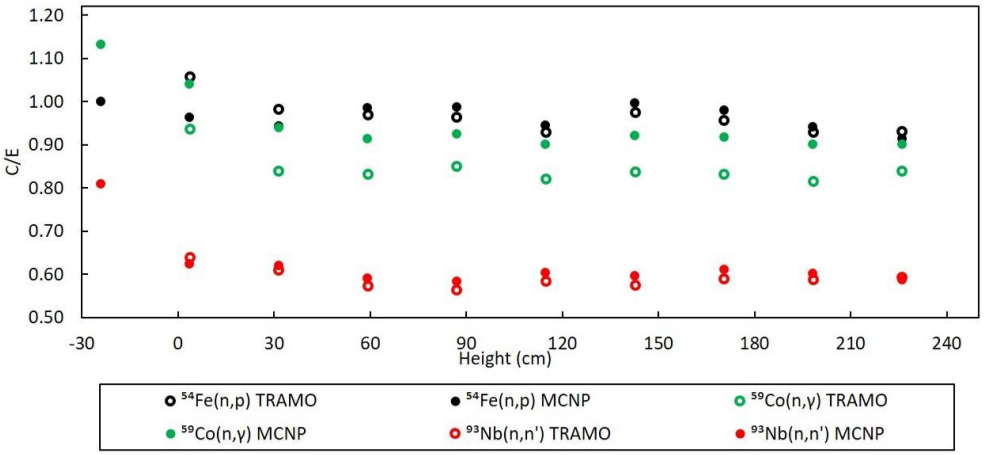

Fig. 3. Angle-averaged C/E ratios of reaction rates of three selected monitors. The monitors are shifted by -24cm in the calculations.

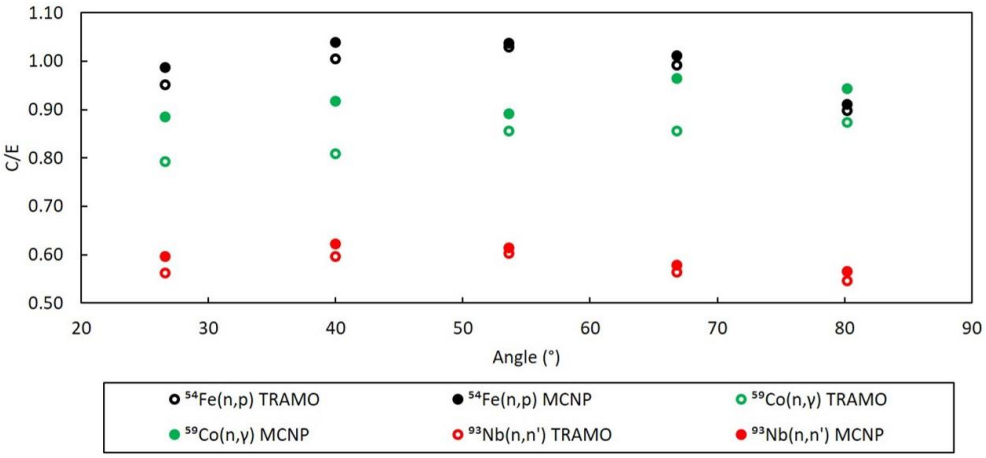

Fig. 4. C/E ratios of reaction rates of the same monitors as in Fig. 3; adjusted height = 142.7cm.

Table 1. Angle-averaged C/E ratios for TRAMO. Monitors are shifted by -24cm in the calculations.

Height (cm)	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{93}Nb(n,n')	^{93}Nb(n,γ)	^{59}Co(n,γ)	^{109}Ag(n,γ)
226.1	0.95	1.01	0.93	0.81	0.59	0.89	0.84	0.86
198.3	0.98	0.99	0.93	0.87	0.59	0.84	0.82	0.88
170.5	0.98	0.99	0.96	0.87	0.59	0.9	0.83	0.85
142.7	0.98	1.08	0.98	0.97	0.57	0.87	0.84	0.88
114.9	0.97	0.98	0.93	0.8	0.58	0.86	0.82	0.85
87.1	0.96	1.08	0.96	0.96	0.56	0.86	0.85	0.91
59.3	0.98	1.09	0.97	0.96	0.57	0.91	0.83	0.89
31.5	1.04	1.13	0.98	0.98	0.61	0.96	0.84	0.89
3.7	1.06	1.20	1.06	1.02	0.64	1.03	0.94	0.94
mean	0.99	1.06	0.97	0.92	0.59	0.84	0.84	0.88

Table 2. Angle-averaged C/E ratios for MCNP. Monitors are shifted by -24cm in the calculations.

Height (cm)	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{93}Nb(n,n')	^{93}Nb(n,γ)	^{59}Co(n,γ)	^{109}Ag(n,γ)
226.1	0.97	1.02	0.91	0.80	0.59	0.95	0.90	0.95
198.3	1.02	1.01	0.94	0.89	0.60	0.91	0.90	0.98
170.5	1.02	1.01	0.98	0.89	0.61	0.98	0.92	0.95
142.7	1.02	1.10	1.00	1.00	0.60	0.94	0.92	0.97
114.9	0.99	1.00	0.95	0.82	0.60	0.94	0.90	0.95
87.1	1.02	1.12	0.99	0.98	0.58	0.94	0.92	1.00
59.3	1.01	1.12	0.99	0.97	0.59	1.00	0.91	1.02
31.5	0.99	1.07	0.94	0.94	0.62	1.08	0.94	1.04
3.7	0.95	1.07	0.96	0.94	0.62	1.15	1.04	1.08
-24.1	0.91	1.12	1.00	0.78	0.81	1.43	1.13	1.20
mean	0.99	1.06	0.97	0.88	0.62	0.79	0.95	1.01

It is difficult to give a reasonably reliable value for the errors. The statistical errors of the calculations are generally less than 1% for the individual group fluences. Only in the thermal energy groups do errors of up to about 10% occur. The error in the calculated reaction rates is then also less than 1%. Other error quantities relate to the modeling of the geometry, the material composition or the homogenization of the materials, the reaction

cross sections of the monitors and, of course, the experimental data. A general systematic error of 3% – probably related to the detector limit – and individual statistical uncertainties between 1% and 10% are given for the latter.

First of all, it should be noted that the C/E ratios in Fig. 4 have only a slight angular dependence within 10%. This holds true for other heights as well. Thus, averaging for the discussion of the height dependence (Fig. 3) appears justified for all reactions.

It can be seen that for the different reactions in Fig. 3 the C/E values are comparable over the height, and that they are relatively close to each other for TRAMO and MCNP calculations. Unfortunately, the data points for the lowest monitor are missing for TRAMO as mentioned above. The C/E ratios in Fig. 3 vary only slightly with height.

^{54}Fe(n,p) is a reaction in the fast energy range. The neutrons scatter only rarely, so that the source distribution is essentially reflected. TRAMO and MCNP data show very good agreement with each other and the C/E ratios are close to 1. This is a good indication that the source distribution has been modeled correctly.

^{59}Co(n,γ), on the other hand, is a reaction in the thermal energy range. Moreover there is a strong resonance at about 130eV. Here, the TRAMO results are slightly below those of MCNP and both are almost consistently below 1. Neutrons can be thermalized in the RPV or scattered back at the insulation or water tank. In order to obtain reasonable results, it is therefore necessary to model the structural elements as accurately as possible. In addition, the resonances in the group cross-sections must be resolved as precisely as possible. The results show that this was achieved quite well. One possible reason for the lower values in the TRAMO calculations could be that a constant water temperature was used in these calculations. On the one hand, the degree of validation of TRAMO in the thermal and resonance energy range is not yet as high as in the energy range above 1 MeV. This requires further investigations.

In case of ^{93}Nb(n,n') it is striking that the C/E ratios for are very low compared to the other reactions, namely only about 0.6. This behavior is known and has been documented in previous work [13]. The measurements were performed by gamma spectrometry, which are very difficult because of the very low gamma energy of the emission lines from the ^{93m}Nb. The activity of the reference source can be called into question.

Overall, TRAMO and MCNP show good agreement in terms of reaction rates and in the corresponding C/E ratios. Errors in the calculation methods of the programs can be largely ruled out. Other comparisons of experimental data with calculated results determined with this model show good agreement, so that it is also unlikely that a model (geometric) error is the cause of the shift (Fig. 2). The most plausible therefore seems to be incorrect recording of the heights in the experiment, especially since a constant shifting of all monitors leads to excellent results.

5 Summary and Conclusion

The code package TRAMO has been extensively modernized and some functionalities have been added. In addition to the transfer to modern Fortran, the inclusion of standardized models for VVER-440 and VVER-1000 reactors is noteworthy. On the basis of these, it is possible for external users to carry out fluence calculations independently. This concerns in particular calculations for monitors located at the surface of the reactor pressure vessel. The revised pre- and post-processing enables a fast preparation of the data. The new TRAMO version is continuously validated and verified.

For this paper, results of monitor experiments in the 12th cycle of the Greifswald nuclear power plant, unit 1, have been calculated with both TRAMO and MCNP. Previous observations showed a good agreement of experimental and calculated data in the center region of the reactor core. However, there were increasing deviations as the distance from

the middle of the reactor core became larger. A possible solution to the problem could be found. If one assumes an (unintentional) shift of the monitors by 24cm downwards, excellent agreements of experimental and calculated reaction rates are obtained. Comparison calculations with MCNP support the assumption that there may be a mistake of the experiment.

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