Review of the official opponent
on the PhD thesis of Eng. Michal Košťál
"Transport of Neutrons and Photons in Construction Parts of VVER-1000 Reactor",
Specialization Nuclear Engineering

It was received for review the thesis in a single volume of 143 pages, contained the following chapters: 1 Preface, 2 Goals of the Thesis, 3 Introduction (5 sections, 7 paragraphs), 4 Methods of Solution (7 sections, 12 paragraphs), 5 Results (5 sections, 21 paragraph, 28 sub-paragraphs), 6 Conclusion, 7 References.

Current status of nuclear power, based on VVER reactors, characterized by the facts that on the one hand pressure vessels of many reactors are approaching the project lifetime limits, but on the other hand works on a significant prolongation of the lifetime of NPPs are in progress, as well as works on increase of the reactors power, and new fuel cycles with improved economic parameters are developed and implemented.

All of this causes an increased interest in the problem of reliable assessment and prognosis of the radiation exposure of pressure vessels and internals.

The most important role in solving this problem play benchmark-experiments which are needed to validate the calculation methods and nuclear data libraries used in the calculation of radiation exposure. In some cases they allow to obtain the integral parameters, which are necessary for radiation exposure determination.

The analysis of the benchmark experiments results can extend understanding of the processes of radiation transport in reactor designs.

A necessary condition for giving to any experiment the "benchmark" status is its so-called "evaluation", or analysis of the experiment including the creation of precise computational model and high-precision calculation of the measured parameters using the most modern methods, codes, and nuclear data libraries, a comparative analysis of calculation and experimental data and their errors, the estimating on this basis the reliability and the absence of contradictions in the data. Only after such an analysis the experiment can obtain the benchmark status and can be included in the international libraries of benchmark experiments. As a rule, it is required several independent high-precision calculations.

The thesis is devoted to the analysis of two types of such experiments: the first - experiments on the full-scale mock-up of VVER-1000 on LR-0 reactor (measurements of the core characteristics and parameters of the neutron and gamma fields in the mock-up in-vessel space and pressure vessel), the second - the measurement in a simple planar geometry of neutron and gamma-radiation transport through the layers of iron and water on the reactor AKR-2. Both of these experiments can play an important role in solving the problems of definition and prognosis of radiation exposure of pressure vessels and internals of VVER reactors.

Thus, it is beyond any doubt that topic of the presented thesis is really urgent.

Researches on the mock-up of VVER-1000, which was created as a result of cooperation of UJV Řež, ZJS Škoda, and Kurchatov Institute, were being conducted for many years by UJV Řež, Kurchatov Institute, University of Defense Brno and other institutions under several multilateral and bilateral (Czech Republic - Russia) international projects. As part of these

researches it was carried out the above-mentioned "evaluation" of experiments using different methods (deterministic and Monte Carlo) and nuclear data libraries.

Author of dissertation took the next step. He analyzed the new experiments on the mock-up, and made the analisys using the modern code MCNPX 2.6.0 and a few modern nuclear data libraries. He performed more detailed analysis of the calculated and experimental pin-by-pin power distribution in the mock-up core. Besides he moved forward in the analysis of uncertainty of the various mock-up calculated characteristics. His analysis extended our knowledges concerning the neutrons and gammas transport in reactor constructions, it allows to create the new useful benchmarks.

Next I will discuss the materials presented in different chapters of the thesis.

In Chapter 4, Section 4.1 ("MCNP Transport Code ") it is presented the methodology of calculation by the Monte Carlo method realized in the MCNPX 2.6.0 code. The choice of the Monte Carlo method and one of the currently available modern codes for the experiments "evaluation" should be definitely supported. I have only one remark to the material of this section: the code MCNPX 2.6.0 is developed in Los Alamos, but not in Oak Ridge as mentioned on page 24.

In section 4.4 ("VVER-1000 Benchmark Description") my attention was attracted by Table 4.4 with the results of the calculation of k_{eff} for different levels of the moderator and different concentrations of boric acid in it. The uncertainty of result and its components, listed in this table, is of particular interest. Obtaining such data is mandatory for the experiment "evaluation" and comparison of experimental and calculation results. Unfortunately, the text does not specify how there were determined the uncertainty components concerned with uncertainties of the characteristics of individual fuel rods and assemblies (fuel enrichment and density, cladding thickness, step between assemblies).

In section 5.1 ("Pin Power Distribution") it was made the detailed analysis of the results of new measurements and calculations of the pins power in the mock-up core. This is a very useful calculation-experimental study, the results of which can be used to justify the engineering methods of calculation of pin-by-pin power distribution in the VVER reactors core, i.e. distribution of the neutron source used in the calculation of the radiation exposure of pressure vessel and internals. The benefit of this study is significant because the source error makes a significant contribution to the error of the calculated values of radiation exposure.

Paragraph 5.2.1 ("Calculational Model") is devoted to effects that are usually ignored in the analysis of mock-up experiments (at the calculation of the neutron source in the mock-up core). These include contribution of delayed neutrons and differences of the spectrum and the number of fission neutrons in various regions of the mock-up due to differences of the spectrum of neutrons causing fission in these regions. The calculations have shown that these effects are indeed small, and can be neglected, but once accurate quantitative evaluation should be done. By the way, in the of power reactors calculations the changes of the spectrum and the number of fission neutrons from region to region can be noticeable and should be taken into account.

Paragraph 5.2.2 ("Resulted Fluxes in VVER-1000 Benchmark") presents the results of detailed calculations of the neutrons and gammas flux density spectra, attenuation coefficients and spectral indices at all points of the mock-up, where the measurements were made. The identified behaviours are analised. This information is very important at solving the radiation exposure problem.

It would be desirable to compare these results with those obtained previously by other investigators; such comparison would allow a deeper analysis. This concerns, for example, attenuation coefficients and spectral indices. In some cases earlier there were found some other correlations between the calculated and experimental data than presented in the thesis.

Unfortunately, in this section it is also found the formal mistakes. For example, the signatures (or legends) to figures 5.30 and 5.31 are reversed. The text after the figure 5.32 contradicts the content of the table 5.23.

Useful new results are presented in the paragraph 5.2.3 ("Particle Flux Behind Core With Diffusion Approach Power Distribution"), where it is shown that the differences in the sources of fission neutrons in the core, calculated in the diffusion approximation (code MOBY DICK) and by Monte Carlo method (MCNPX 2.6.0), do not lead to noticeable differences in the results of calculation of the neutrons and gammas spectra in pressure vessel and internals (except of the point on the barrel surface). This confirms the possibility of using in further analyzes the "diffusion" source, recommended earlier.

The estimation of the uncertainty of the calculation results is a key element of the analysis. Such an evaluation is made in section 5.3 ("Uncertainties in VVER-1000 Benchmark Calculations"), where various sources of uncertainty are discussed. The analysis of those sources of calculation results uncertainty as the uncertainty of the density and composition of the mock-up materials is made for the first time.

Unfortunately, the author of dissertation did not consider another important source of uncertainty, namely, the dimensions tolerances of the mock-up components, the uncertainty in the gaps dimensions between them and the detectors location in the measurements. I would suggest to make it in further researches.

So detailed as the experiment on the VVER-1000 mock-up it is analyzed in the section 5.4 ("AKR-2 Transport Model") the experiment on AKR-2 reactor, in which it was measured neutrons and gammas flux density spectra in simple assemblies consisting of iron and water layers in various combinations (sorry, it should be noted that on the assembly diagram in Figure 5.41 there are not the combinations "D" and "E" for which experimental and calculation results are presented).

Analysis of experiments on the reactor AKR-2, as far as I know, is done the first time.

Author of dissertation solved here the complicated problem of source definition for transport calculations. This problem is specific for "beam" experiments.

For gamma field calculation author applied the effective two-stage method splitting two components – gamma radiation of reactor core and gamma radiation which was born after neutrons capture and scattering in the considered media.

For the first time in the practice of analysis of benchmark experiments for VVER the author of dissertation carried out an analysis using almost all the modern nuclear data libraries. This approach, of course, should be supported. The results of this analysis are presented in Section 5.5 ("Effect of Used Nuclear Data").

Author carried out here for the first time the calculations of neutron spectra in iron taking into account the effect of crystal bonds. The modern versions of ENDF/B library give such an opportunity.

For the VVER-1000 mock-up it is analyzed in detail the influence of the nuclear data library on the calculation core characteristics. For the experiment on the AKR-2 calculations of

neutron and gamma flux density spectra were made as well as integral characteristics – spectral indices and attenuation coefficients – using all nuclear data libaries.

An interesting and useful analysis is made to explain the discrepancy between the results of calculations using the libraries JENDL 3.3 and JENDL 4: in this case the differences of the scattering cross sections (elastic and inelastic) for iron and the corresponding angular distributions of scattered neutrons in that libraries are analyzed. This is an example of an approach that should always be used when analyzing the results obtained using different libraries.

According to the results of AKR-2 experiment analysis, carried out by author of dissertation, it seems to me useful to include this experiment in a number of benchmarks on the problem of radiation exposure of the internals and pressure vessels of VVER.

It is necessary to mention, that experiments on the VVER-1000 mock-up and on the reactor AKR-2, which are analyzed in the thesis, efficiently supplement each other.

Evaluating the work as a whole, I must recognize:

- the theme of the research is urgent and important to solve the problem of increasing the reliability of determination and forecasting of radiation exposure of the VVER reactors internals and pressure vessels,
- the work performed on a modern methodical level: the author of dissertation used the most modern calculation instrument and almost all modern nuclear data libraries,
- the research problems posed for the author of dissertation are fulfilled,
- the results are of doubtless scientific interest, some of them were obtained for the first time, analysis extended our knowledges concerning the neutrons and gammas transport in reactor constructions and allows creating the new useful benchmarks,
- the author carried out a very large amount of calculations and demonstrated a high level of his scientific qualification,
- some shortcomings noted in my review have mainly the formal character, they don't reduce the overall appreciation of the work.

Analysis of the presented thesis allows me to assert with confidence that its author engineer Michal Košťál certainly can be awarded to the degree Doctor of Philosophy in the specialization "Nuclear Engineering."

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