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Ex-Vessel Reactor Dosimetry for Units 5 and 6 of Kozloduy NPP – Current Status and Future Prospects

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Abstract. More than 20 years the Reactor Physics Laboratory of the Institute for Nuclear Research and Nuclear Energy – Bulgarian Academy of Sciences (INRNE-BAS) provides technical support of Kozloduy NPP regarding nondestructive neutron fluence determination of the reactor pressure vessel (RPV). The activities on neutron fluence assessment and related evaluation of RPV metal degradation (embrittlement temperature shift $\Delta \mathrm{T}_F$) as well as the consequent RPV lifetime prediction were started in 1986 and are still ongoing. The Reactor Physics Laboratory together with the Radioanalitical Methods Laboratory carried out reactor dosimetry at Kozloduy NPP Units 1 to 6 for validation/verification of the neutron fluence calculation by activation monitors. For this purpose ex-vessel detectors have been installed and irradiated in the air cavity behind the vessel periodically from 1993. The obtained results have provided important information needed for taking decisions on reactor lifetime management. Further improvements of the RPV neutron fluence determination methodology are foreseen. Large scale international effort is searched to validate these improvements.

Keywords: neutron/gamma fluence, reactor pressure vessel, activation detectors, surveillance.

1 Introduction to Reactor Dosimetry

The reactor pressure vessel of a VVER type unit is referred as the only irreplaceable component. As such, its structural condition appears as a limiting factor of whole unit's operational lifetime. Main factor that causes the structural degradation of the RPV ferrite steel is the radiation embrittlement. The degradation mechanism is expressed in the formation of vacancies, interstitials in the crystalline structure of the steel, as well as in transmutation reactions. This imposes the detailed knowledge about the neutron/gamma irradiation of the RPV as an important part of the lifetime management for the whole unit. The tool that provides data for the neutron fluence accumulated over the RPV during the reactor operation is Reactor Dosimetry (RD).

The evaluation of neutron fluence is performed only by neutron transport calculations since the direct measurements at so called critical places of RPV are not possible. The degradation of RPV practically could be assessed only by indirect evaluation of the critical temperature of embrittlement of vessel metal on the base of neutron fluence with $E > 0.5$ MeV.

The fluence of neutrons with energy above 0.5 MeV is presented by the scalar neutron flux $\Phi(r, \theta, z, E, t)$, $\mathrm{cm}^{-2}\mathrm{s}^{-1}$, as follows:

$$F(r, \theta, z) = \iint \Phi(r, \theta, z, E, t) dE dt\,. \qquad (1)$$

The energy integration is performed over the interval (0.5-15) MeV and the time integration is over the cycle length.

The degradation of metal properties caused by neutron fluence F is measured by the shift of the temperature of neutron embrittlement ΔT_F, $^\circ$C.

The nondestructive method for evaluation of the metal embrittlement under neutron irradiation is based on the relation between the temperature of neutron embrittlement ΔT_F, $^\circ$C, and the neutron fluence F, cm^{-2}. The Russian standard for VVER reactors [1] uses the semi-empirical relation

$$\Delta T_F = A_F \left(\frac{F}{F_0}\right)^{1/3}, \qquad (2)$$

where A_F, $^\circ$C, is a chemical coefficient of neutron embrittlement; F, cm^{-2}; $F_0 = 10^{18}$ cm^{-2}.

These changes in the materials properties are monitored by a surveillance program. Charpy specimens, fracture toughness specimens and tensile specimens, neutron activation detectors, and temperature monitors should be placed in capsules that are located within the reactor vessel. The surveillance capsules are then withdrawn according to a predefined schedule to monitor the reduction in fracture toughness due to neutron irradiation. The withdrawal is performed over large periods during the lifetime of the reactor unit. This schedule, as well as the placement of the capsules away from the most radiation loaded places of the RPV increases the uncertainty of the RPV neutron fluence evaluation. That is why ex-vessel reactor dosimetry [2] is strongly recommended for use simultaneously with the in-vessel capsules to be able to determine more precisely the neutron fluence attenuation through the vessel wall. The ex-vessel dosimetry relies on numerical evaluation of the neutron fluence through the RPV wall, verified by comparison with neutron activation detectors, placed in the air cavity behind the RPV of the reactor unit. It allows more flexibility in reproducing the operation conditions of the unit in comparison to the built-in surveillance capsules monitoring.

The historical preconditions, the current status and the ways for further improvement of the developed in INRNE-BAS Methodology for determination of the neutron fluence on Kozloduy NPP Units 5 and 6 will be reviewed in this paper.

2 Reactor Dosimetry for Kozloduy NPP

The activities on neutron fluence assessment and related RPV metal degradation (embrittlement temperature shift ΔT_F) evaluation as well as the consequent RPV lifetime prediction were started in 1986 and are still ongoing. The Reactor Physics Laboratory together with the Radioanalitical Methods Laboratory carried out reactor dosimetry at Kozloduy NPP Units 1 to 6 for validation/verification of the neutron fluence calculation by activation monitors. Since then both calculations and experimental aspects of the methodology has undergone evolutional improvements to keep it up-to-date with the world standards in the ex-vessel dosimetry.

2.1 Early approach

The capabilities of the calculation tools more than two decades ago were very low, compared to nowadays. This imposes a number of limits on the numerical solution for the neutron and gamma fluence. The synthesis method [3] was used to obtain the three dimensional distribution of the fluence by numerically solving the transport equation in 1D and 2D geometry to obtain the flux Φ for the radial-azimuthal direction $\Phi(r,\theta)$, radial-axial direction $\Phi(r,z)$, and in radial direction $\Phi(r)$. The ASYNT method [4], based on solving the adjoint transport problem, is another method for reconstructing the 3D neutron flux $\Phi(r,\theta,z)$ from 1D and 2D geometry solutions for the flux, that is developed and applied in INRNE-BAS. The 1D and 2D geometry calculations were performed by the discrete ordinates code DORT [5].

The experimental verification of the results for the fluence was performed by placing neutron activation detectors behind the RPV wall of units 1-6 of Kozloduy NPP. The activation detectors chosen (see Table 1) are capable of integrating the neutron flux above certain energy threshold, and in this way to preserve the information of RPV irradiation.

Table 1. Isotopes used as integral neutron monitors for Units 1-6 of Kozloduy NPP

Isotope	Reaction	Daughter nuclide	Threshold, MeV	$T_{1/2}$	Eγ, keV
^{54}Fe	(n,p)	^{54}Mn	1.6	312.5 d	834.8
^{63}Cu	(n,α)	^{60}Co	6.8	5.26 y	1173 1332
^{93}Nb	(n,n')	^{93m}Nb	0.5	16.4 y	16.6 18.7

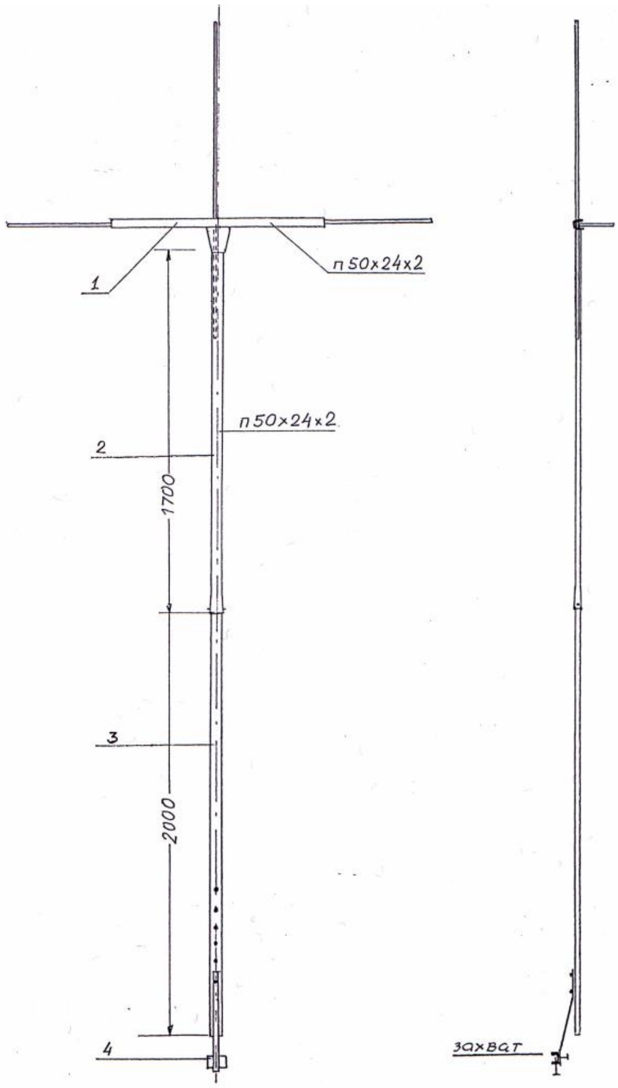

Figure 1. Supporting structure for the neutron activation detectors: 1 – cross-shaped holder for the activation detectors; 2 – upper part of the extender; 3 – lower part of the extender; 4 – bottom support.

The neutron activation detectors were placed on the outer RPV surface near the most irradiated weld using special carrying device (see Figure 1).

The Fe and Cu activation detectors were in the form of long wires wound as spirals, while the Nb activation detectors were hanged on the device in separate containers. In this way the coverage of the fluence in radial and azimuthal direction for more than a sector of symmetry of the reactor core was achieved.

2.2 Current status

The advance of the computer technology as well as the lessons learnt during the decades lead to improvement of both numerical techniques and the detectors placement in order to better evaluate the neutron fluence of the RPV accumulated during one or two subsequent campaigns of the reactor unit. The neutron transport through the reactor system is determined by full 3D deterministic discrete ordinate TORT code [6], that allows more realistic representation (see Figure 2) of the reactor geometry.

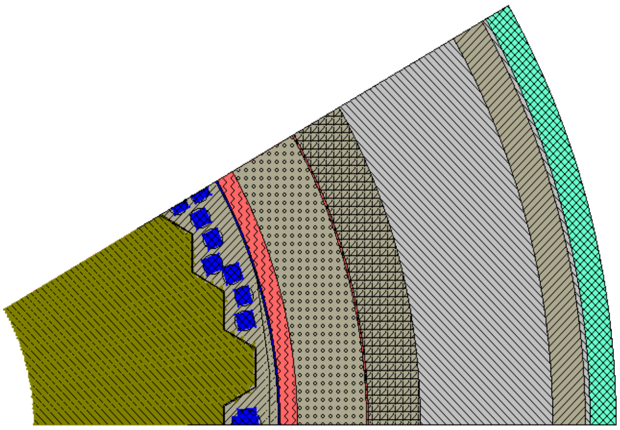

Figure 2. Visualization of TORT 3D calculational model.

The device for placement the activation detectors was redesigned to incorporate the Nb detectors in it, so the placement time in the air cavity behind the RPV and human error rate are significantly reduced.

This improvements lead to obtaining a very good agreement between the measured and calculated activities of the detectors (see Figure 3). The uncertainty of the results for the measured activities of the activation detectors is less than 5% for Fe and Cu. The preparation of measurement samples from the Nb detectors is more complicated than the preparation of Fe and Cu. This leads to a slight increase of the uncertainty for the measured activity of ^{93m}Nb – up to 8%.

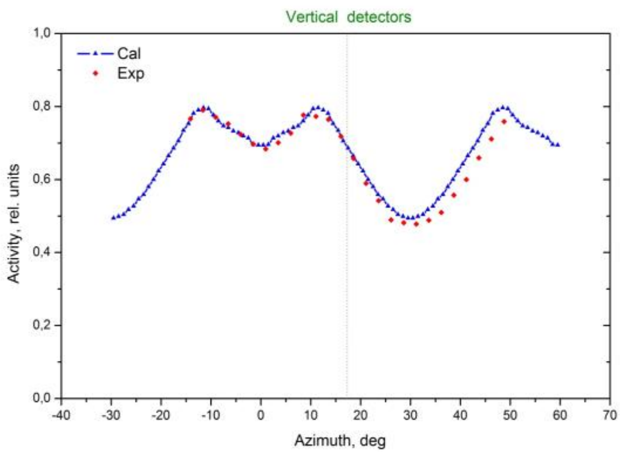

Figure 3. Example of calculated and experimental results for Fe detectors activity (in relative values).

2.3 Research activities and future prospects

Reaching the modern standards in performing ex-vessel reactor dosimetry shifted the focus of scientific research to a number of investigations that will improve the information related to the neutron damage of the RPV steel. Great push brought the implementation in the reactor calculations of the Monte Carlo method using the MCNP code [7] with pointwise cross-sections' description by energy. The influence of the multigroup approximation used in the calculations by the discrete ordinates method was investigated [8], and the sufficient accuracy of the BGL library [9] was proved. Another important research was connected with evaluating the gamma contribution to the total damage of the RPV steel in terms of displacements-per-atom (dpa). It was found negligible [10] in comparison to the neutron damage and generally within the statistical error of the calculations.

Although the nuclear industry is satisfied with the current accuracy of the reactor dosimetry numerical modeling, the possibility for the more precise Monte Carlo calculations opens interesting field for research in the area. The increased precision of these calculations is also due to the close-to-real description of the geometry. This allowed the creation of 360 degree model of a VVER-1000 unit (see Figure 4) that could possibly allow the investigation of local perturbances in the neutron/gamma field by means of Monte Carlo calculations.

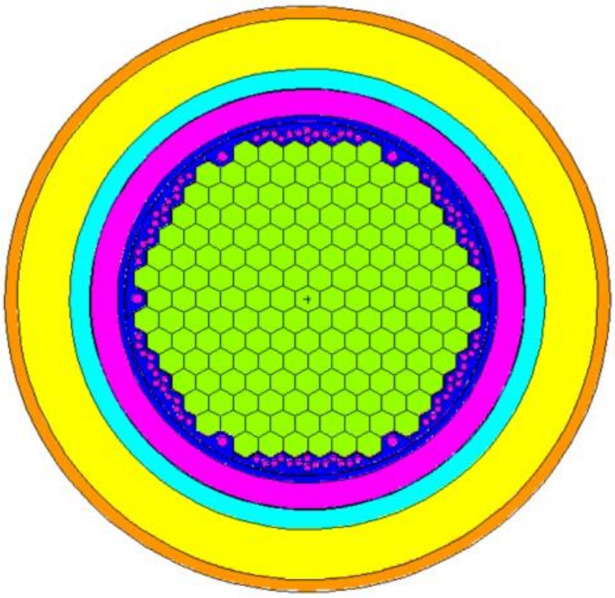

Figure 4. VVER-1000 MCNP model visualization in radial-azimuthal direction.

The calculations performed on such large scale model require substantial calculation effort, as well as knowledge of Monte Carlo statistics controls to ensure the correctness of the obtained results. Based on the experience on other reactor facilities calculations [11] the routine use of this model in the near future has to be expected. Anyway, a considerable effort in validation of the model has to be invested in order to assure the reliability of the obtained results.

3 Conclusion

The Institute for Nuclear Research and Nuclear Energy of the Bulgarian Academy of Sciences provides technical support of Kozloduy NPP regarding nondestructive neutron fluence determination of the reactor pressure vessel for more than 20 years. The reliability of the RPV Neutron Fluence Determination Methodology developed in INRNE-BAS is validated using experimental results and on international benchmarks [12]. The methodology is constantly improved and its evolution is described in the paper. The results of the related investigations increase the trust in the applied approach for determination of neutron/gamma fluence, and set a reliable ground for further investigations in the area.

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