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Applicability of LR-0 mock-up results to VVER-1000 reactor pressure vessel issues

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ABSTRACT

Evaluation of neutron fluence in a reactor pressure vessel (RPV) together with surveillance specimen programs for RPV materials are one of the most important parts of in-service inspection programs that are necessary for realistic and reliable assessment of RPV residual lifetime. This paper covers transport of neutrons through the RPV of a VVER-1000 nuclear reactor. This problem is of increased importance as it concerns issues around VVER NPP life extension. With regards to the construction (reduced thickness of the lateral reflector), this issue plays greater role in VVER reactors than in Western types of PWR reactors.

RPV material degradation depends mainly on neutron flux and spectra. Both quantities can be calculated or measured. This paper compares MCNP calculations and measurements on zero-power experimental reactor LR-0 with TORT calculations for VVER-1000. The goal is to find a reasonable method for precise estimation of neutron fluence and attenuation factor through the RPV. The calculations were performed with MCNP stochastic code and TORT deterministic transport code. The measurements were performed in a VVER-1000 mock-up placed in reactor LR-0 (Research Center Řež).

1. Introduction

VVER-1000 reactors can operate without difficulties for at least 40 years due to their basic design (Ballesteros et al., 2004). The prolongation of their lifetime is primary related to RPV radiation damage which is caused by various reactions. RPV stainless steel damage is mostly caused by neutrons which displace atoms from crystal lattices directly by scattering or indirectly due to momentum conservation, when an atom moves in the opposite direction as a photon after collision. Stainless steel degradation could also be caused by formation of gases (^{1}H, ^{4}He) inside material by (n, α) or (n, p) reaction. 1H can contribute to corrosion while ^{4}He forms bubbles which may cause swelling. The degradation effect from gamma rays in the case of VVER-1000 does not exceed 4% of total Displacements Per Atom (DPA) Ilieva et al., 2009.

Thus mainly fast neutrons cause embrittlement of RPV material which is characterized by the increase in a ductile-to-brittle transition temperature (DBTT) that marks the transition between low toughness brittle and high toughness ductile fracture regimes. The Eq. (1) is given for a temperature shift in the case of VVER-1000 (Nikolaeva et al., 2001). Radiation damage caused by neutrons depends on the energy of the neutrons, i.e. on the energy spectrum. With higher neutron energy the probability of penetration through RPV is increasing. In this article, the results of stochastic and transport calculations of neutron flux will be compared with fast neutron spectra experimental results.

$$\Delta T_F = T_F - T_{Cr} = A_F \Phi^{1/3} \tag{1}$$

- A_F - radiation embrittlement coefficient ($A_F = 20$ for welds; $A_F = 23$ for the main metal 15KH2NMFAA steel).
- T_{Cr} - the critical temperature of brittleness of a material in its non-irradiated state.
- T_F - the critical temperature of brittleness after irradiation.
- Φ - fast-neutron fluency in units of 10^{22} m^2 (E > 0.5 MeV).

For the correct applicability of LR-0 mock-up results to VVER-1000 reactor it is important to study uncertainties caused by geometry and material discrepancies compared to VVER-1000. For stochastic calculations, a proved MCNP benchmark model was used (Košt'ál et al., 2012a). For transport calculations, TORT 3D code with a proven model of VVER-1000 power reactor (Marek et al., 2014) was used.

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2. Experimental and calculation methods

2.1. Experimental methods

The experiments were carried out at room temperature in a VVER-1000 transport mock-up placed in zero-power light water research reactor LR-0. The VVER-1000 mock-up is a radial full scale transport benchmark including baffle, barrel, RPV simulator, and concrete shielding model (see Fig. 1). The mock-up core consists of 32 shortened (the length of the fissile column is 125 cm) VVER-1000 type fuel assemblies with different enrichments (see Fig. 1). Criticality for this loading is achieved with moderator (demineralized water with diluted boric acid with concentration 4.6 ± 0.1 g/kg) which fills a core to level H_{cr} = 150 cm, when two control clusters (three absorbing B_4C rods only, positions 19 and 23) are inserted to 60.4 cm above the bottom of the fissile column. More details can be found in Košťál et al. (2015).

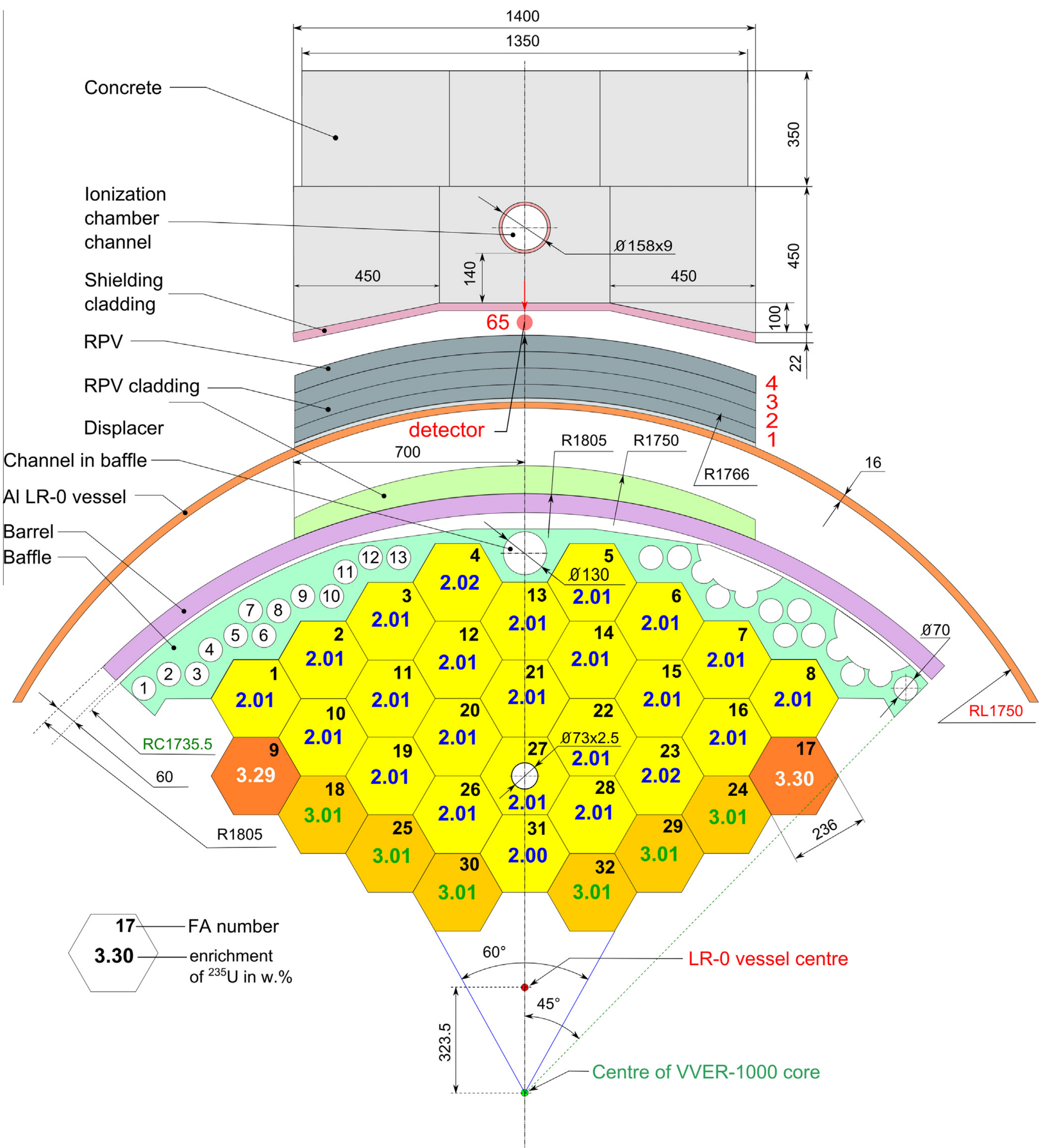

Fig. 1. Section of the mock-up in the x-y plane (Dimensions in mm). Basic configuration.