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Void Swelling in VVER-1000 Pressure Vessel Internals

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Abstract - *The originally designed lifetime of the commercial VVER-1000 is 30 years. Life extension requires evaluation of pressure vessel and its internals degradation under long-term irradiation. In case of reactor pressure vessel the main problem is embrittlement, which is dangerous considering the high pressure during reactor operation — 15.7 MPa. In case of internals, the main potential limiting factor is a void swelling of the Russian type titanium stabilized stainless 08Cr18Ni10Ti steel used to construct the baffle surrounding the active core. For swelling estimation the precise gamma heating and DPA (displacement per atom) estimation in deeper parts of the baffle ring is necessary. This article presents computational assessment of void swelling estimation in VVER-1000 reactor baffle using MCNP6 code for DPA and gamma heating estimation and further calculation of swelling using ABAQUS code.*

I. INTRODUCTION

Void Swelling estimation for LWR (Light Water Reactor) pressure vessel internals is ambiguous, since the data used for actual analysis are based on the data obtained in fast reactors and extrapolation of swelling rate dependent on fluence [2], others are based on conditions of fast flux or ions irradiation [3, 4].

According to analysis [2], the swelling after 30 years can be 10-20 % and after 60 years of operation in local sites of baffle can be up to 40 %.Other measurements show that bolts of VVER-440 after 15 years of irradiation in identical to operational parameters reach 11-19 DPA. The volume of created nano-voids did not exceed 0.1 % of volume increase.

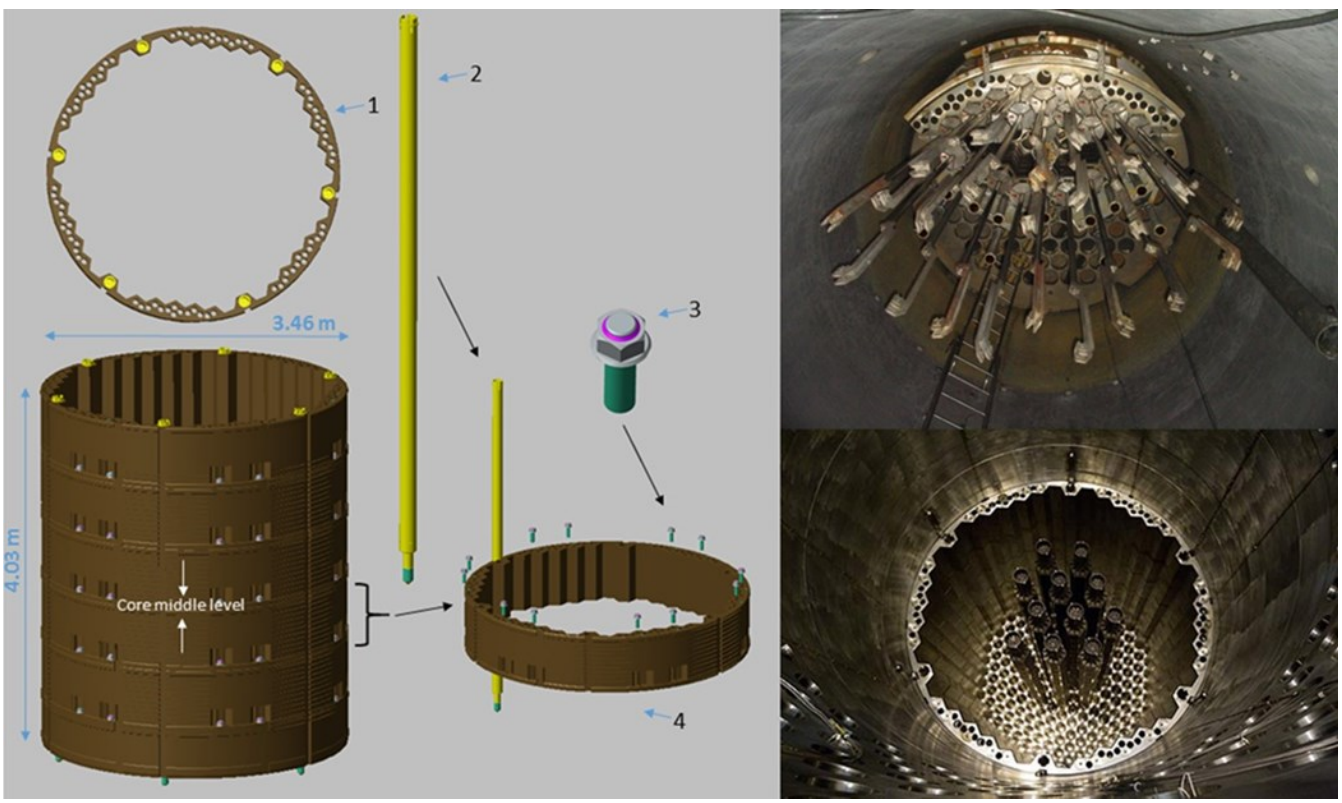

Fig. 1. On the left the general view of VVER-1000 baffle where: 1-reactor baffle (top view), 2-long screws through the baffle height, 3-bolts screwing baffle rings, 4-one of 6 baffle rings.
On the right upper part the VVER-1000 mock-up in LR-0 reactor. On the right down the power VVER-1000 baffle [1]

Swelling rate is a function of DPA, temperature and stress state. For swelling estimation the precise gamma heating and DPA distribution in deeper parts of baffle ring are necessary in LWR operational conditions.

VVER-1000 baffle is described in Figure 1. Temperature map in baffle is calculated based on gamma heating and thermo-hydraulic data for coolant flow in VVER-1000 core. The VVER-1000 mock-up model in MCNP [5] code is used for DPA and gamma heating estimation and further calculation of swelling using ABAQUS [6] code using the model of power VVER-1000.

For MCNP calculation the VVER-1000 mock-up model is used instead of power reactor model, due to the several reasons explained below.

The VVER-1000 mock-up model is V&V (Validation and Verification) and published as a benchmark model in IRPhE-OECD databank [7]. The MCNP calculations using mock-up model are the part of verification process of precise methods of DPA and gamma heating estimation in VVER-1000 internals using Monte Carlo simulations via VVER-1000 mock-up in LR-0 reactor. Due to the fact that the LR-0 reactor is zero power reactor the calculated data, before using for power reactor calculation in ABAQUS code, are scaled to energetic reactor power. Both measurements and calculations with mock-up are normalized to one source neutron.

This makes the scaling process easy and allows the further validation using future planned measurements. This research is mainly focused on method validation for subsequent deployment on power reactor MCNP model.

II. DESCRIPTION OF THE ACTUAL WORK

1. VVER-1000 mock-up in LR-0 reactor

The mock-up core consists of 32 dismountable assemblies with enrichment 2%, 3% and 3.3%. Demineralized water with dissolved boric acid with 4.6 ± 0.1 g/l is used as a moderator. The fuel assemblies are of shortened VVER type, (sintered UO2 pellets, outer diameter 7.53 mm, central inner hole 1.4 mm, active length 125 cm, total length 135.7 cm, Zr alloy cladding tube with outer diameter 9.15 mm, wall thickness 0.72 mm). The mock-up includes radially full scale VVER-1000 barrel, baffle simulators and concrete shielding (see Fig. 2) additional information can be found in [7].

2. VVER-1000 mock-up model

For DPA and gamma heating assessment in VVER-1000 baffle we used MCNP6 code with reliable calculation model of VVER-1000 mock-up. The used model is a fixed source model with directly defined emission. The emission density was validated on experimentally determined power densities [8]. The neutron distribution is also validated in baffle simulator [9]. Applicability of mock-up results to power VVER-1000 reactor has been shown in [10], where the changes in baffle region are negligible in case of neutron flux distribution.

3. MCNP calculation

Gamma heating and DPA calculation in MCNP model of VVER-1000 mock-up were done using mesh tally (grid: 44*88 with the step of 1cm in x or y and 5 cm in z) for the chosen part of baffle see Figure 2. For calculation time saving the fixed source model was used. For primary gamma impact estimation the additional calculation was done. The fixed source calculation correctness was proved with critical mode calculation, where the differences in a neutron/gamma fluxes in each cell in the mesh was less than 1 %, thus in a range of statistical uncertainty of calculation.

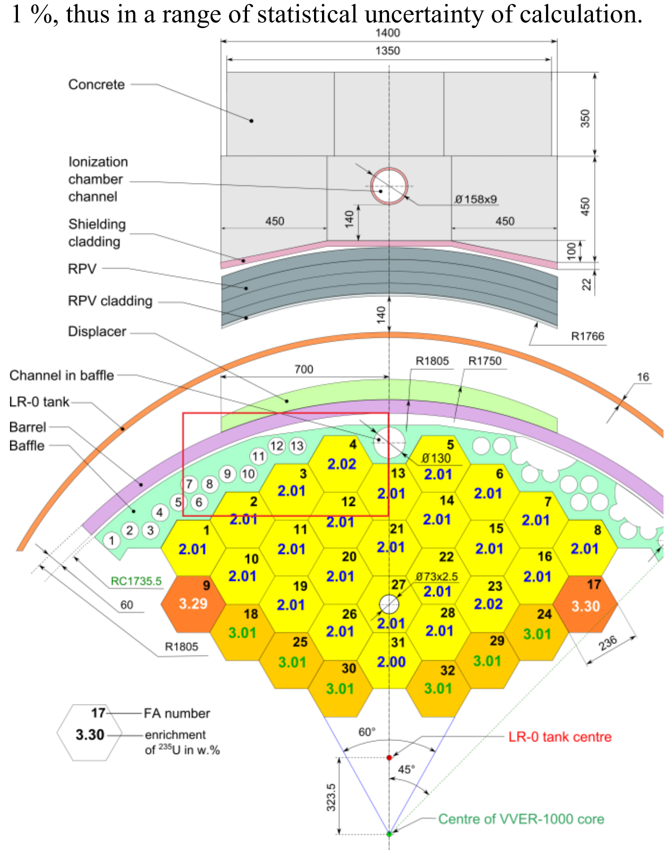

Fig. 2. Section of the VVER-1000 mock-up in LR-0 reactor in the x-y plane (dimensions in mm).

Thermal neutron transport in steel was solved using free gas treatment instead of the recommended S(α, β) for ^{56}Fe in steel because of reported notably better agreement [11].

The DPA was adjusted using Kinchin–Pease formula see equation 1 below [12].

$$DPA = k\frac{E_a}{2E_d} \tag{1}$$

Calculations were done with ENDF/VII.1 [13] library using damage energy production cross section library IRDF-2002 [14] for DPA calculation where E_a stands for available energy, E_d for the Lindhard cut-off energy required for displacing one atom from its normal site and k for displacement efficiency. DPA values are calculated with the convolution of flux and damage cross section library.

The temperature shift in baffle is caused mainly by gamma heating, the rest reactions producing heat (up to 3-5 %) are taken also into account, and the presented results include them. Gamma caused heating got origin by fission, decay of fission products in fuel or neutron interactions. Mainly in thicker parts of the baffle, an important role is played by the gamma originated by thermal capture on ^{56}Fe steel (7.63MeV and 7.65 MeV), or on ^{54}Fe with (9.30 MeV) additional source of high energy gamma in deep parts of baffle is coming from cooling channels on ^{1}H (2.23 MeV). DPA and gamma heating data were scaled to VVER-1000 nominal 3000 MW$_{th}$ power. The scaling is done by multiplying mock-up results with the constant corresponding to number of neutrons in core in case of 3000 MW taking into account core dimension difference (32 fuel assembles Mock-up vs. 163, fuel length difference and axial distribution)

4. ABAQUS

ABAQUS is a multiphysics finite element software which allows to expand its possibilities of standard heat transfer and strain analysis by means of user subroutines written in FORTRAN programming language.

Current calculation model of VVER-1000 core baffle was solved as uncoupled temperature-displacement task with mesh refined to 2×2 mm. Input fields of gamma heating and DPA were prescribed for each finite element of the core baffle and stored in so-called predefined state variables. During calculation those variables are substituted into the mathematical equation for radiation swelling.

The increment of radiation strain tensor $d\boldsymbol{\varepsilon}^{rad}$ at a given time step is presented as the sum of the increments of the radiation creep deformation (deviator component) and the radiation swelling deformation (spherical component):

$$d\boldsymbol{\varepsilon}^{rad} = \frac{1}{3} d\varepsilon_{eq}^{sw}\mathbf{I} + d\varepsilon_{eq}^{cr}\mathbf{n} \qquad (2)$$

where $\mathbf{n} = \partial\sigma_{eq} / \partial\boldsymbol{\sigma}$, σ_{eq} — Mises equivalent stress, $\boldsymbol{\sigma}$ — stress tensor. Radiation strain increments (2) are summarized over all time steps. Under the radiation swelling deformation we understand the first invariant of the total radiation strain tensor, i.e. $\mathbf{I} : \boldsymbol{\varepsilon}^{rad}$.

Calculation of spherical component of radiation swelling $d\varepsilon_{eq}^{sw}$ is done according to the CRISM Prometey model [15, 16] using input data on gamma heating and damaging dose D:

$$\frac{d\varepsilon_{eq}^{sw}}{dt} = C_D \cdot n \cdot D(t)^{n-1} \cdot f_1(T) \cdot f_2(\sigma_m, \sigma_{eq}) \cdot f_3(\varepsilon_{eq}^{pl}) \cdot \frac{dD}{dt} \qquad (3)$$

where

$$\left.\varepsilon_{eq}^{sw}\right|_{t=0} = 0, \;\; d\varepsilon_{eq}^{sw} > 0$$

dt — time period, $C_D = 1.035 \cdot 10^{-4}$, $n = 1.88$,

$f_1(T) = \exp(-1.825 \cdot 10^{-4} \cdot (T - T_{\max})^2)$,

$f_2(\sigma_m, \sigma_{eq}) = 1 + 8 \cdot 10^{-3}(0.85 \cdot \sigma_m + 0.15 \cdot \sigma_{eq})$,

$f_3(\varepsilon_{eq}^{pl}) = \exp(-8.75 \cdot \varepsilon_{eq}^{pl})$,

T — material temperature, $T_{\max} = 470°C$,

$\sigma_m = \mathbf{I} : \boldsymbol{\sigma} / 3$ — average stress, $\mathbf{I}$ — unit tensor, ε_{eq}^{pl} — accumulated plastic strain.

According to [15, 16] we supply expression (3) with relation (4) to account for radiation creep ε_{eq}^{cr}:

$$\frac{d\varepsilon_{eq}^{cr}}{dt} = \left(B_0 \frac{dD}{dt} + \omega \frac{d\varepsilon_{eq}^{sw}}{dt} \right) \sigma_{eq}, \;\; d\varepsilon_{eq}^{cr} > 0, \;\; \left.\varepsilon_{eq}^{cr}\right|_{t=0} = 0 \qquad (4)$$

where $B_0 = 1 \cdot 10^{-6} \, (MPa \cdot dpa)^{-1}$, $\omega = 2.95 \cdot 10^{-3} \; MPa^{-1}$.

Baffle's material yield strength depending on radiation dose and temperature is taken from [16] and is updated along with stresses each time increment during the calculation process.

III. RESULTS

Figure 3 shows the DPA distribution on chosen (see. red frame on the Fig. 2) part of the baffle. The maximum DPA is estimated mainly in vicinity of the large cooling channel.

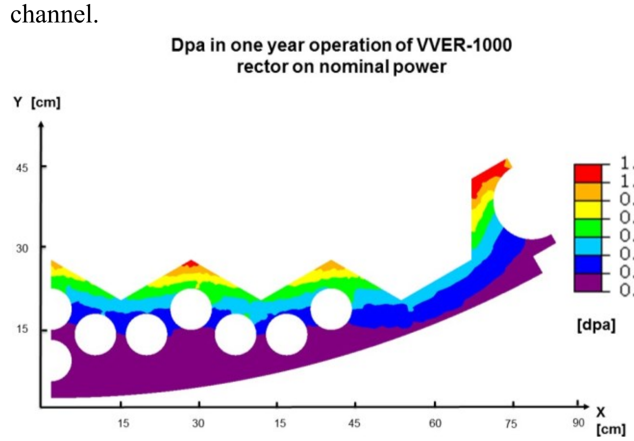

Fig 3. Distribution of DPA in the selected part of baffle.

The DPA distribution strongly depends on pin power distribution on the assemblies closest to the baffle, mainly in their 1-3 rows of pins closest to the baffle. Depending on fuel power (fission source) distribution the max largest area of DPA can be on the other sharp edge of inner baffle part. The maximum DPA during one year of operation in the edges orientated to the core is in the range of 0.8-1.2 DPA per year. Gamma heating is estimated as a total deposited energy due to the fact that the majority (more than 95%) of the deposited energy is coming from gamma. Nevertheless, the presented results include the deposited energy also from neutrons. The typical heating in baffle during nominal power in the height, where the flux is at maximum, varies from 0.5-15 W/cm^3, the maxima's are again on the edges on the inner baffle surface in the range of 6-8.5 W/cm^3 Fig. 4. This additional source of heat is increasing the baffle temperature compared with surrounding water, which is in a range of 280-320 °C. As it is seen from (3), dependence of swelling on the temperature is very significant.

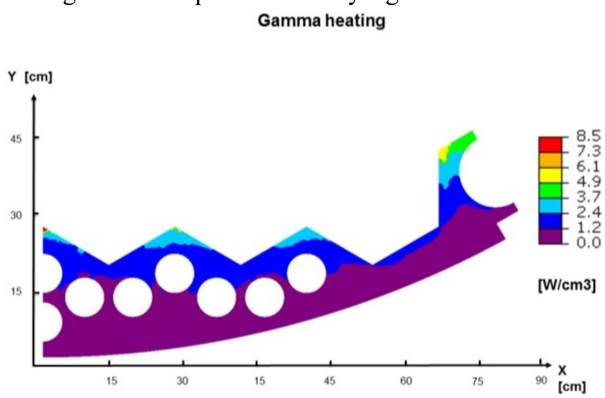

Fig. 4. Gamma heating map in the selected part of baffle.

Heat transfer in the core baffle was calculated with applied gamma heating (Fig. 4) and coolant temperature with heat transfer coefficients from Table 1, which are typical for the VVER-1000 reactor. Resulting temperature field with maximum of 376 °C is shown on Fig. 5.

Table 1. Parameters for the heat transfer task.

	Coolant temperature, °C	Heat transfer coefficient, W·m^{-2}·C^{-1}
Inner surface	320	15000-40000
Outer surface	292	2000-6000
All channels	292	1000-5000

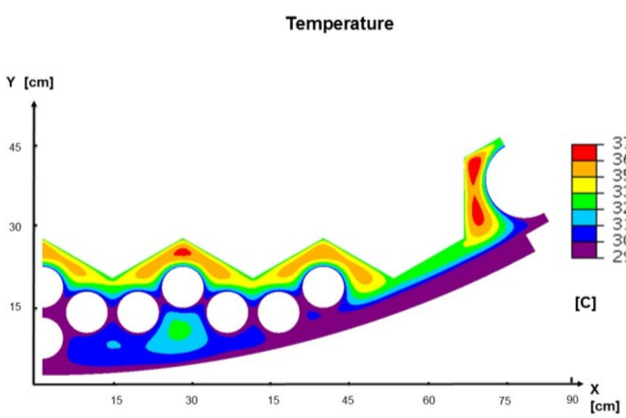

Fig. 5. Temperature map distribution in the selected part of the baffle.

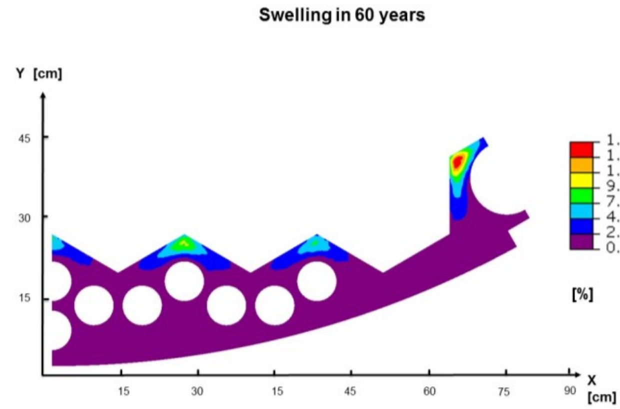

Fig. 6. Swelling map distribution in the selected part of baffle after 60 years of operation.

Calculation uncertainty covers statistical uncertainty of calculation less than 1% in MCNP and convergence tolerance in ABAQUS, equal to 0.003 for the whole strain analysis task. The statistical uncertainties can be neglected, as they are negligible with regard to parameters uncertainties. For parameters uncertainties estimation the sensitive analysis were performed. The main sources of calculation uncertainties are baffle steel density, boric acid concentration, fuel enrichment and gap between fuel pins and the baffle.

For DPA the main contributions to uncertainty are baffle steel density up to 1.4 %, boric acid concentration up to 0.12 %, fuel enrichment up to 0.25 % and gap distance up to 0.8 % giving total uncertainty in DPA 1.6 %.

For gamma heating the corresponding contributions to uncertainty are baffle steel density up to 2.7 %, boric acid concentration up to 0.15 %, fuel enrichment up to 1.4 % and gap distance up to 3.0 % giving total uncertainty in gamma heating values 4.3 %. The total standard uncertainty was determined by summing the values of square standard uncertainty for all of the parameters and taking the square root to convert the variance to a standard uncertainty.

According to [17] strain error in the calculation of swelling is approximately twice higher then uncertainty of input data on both gamma heating and DPA.

IV. CONCLUSIONS

The results presented in this paper estimate the swelling in VVER-1000 reactor baffle after 60 years of operation. The swelling did not exceed 1.6%. The DPA in the region of the big cooling channel and channel number 10 faced to the core reached up to 1.2 DPA per year. Based on the gamma heating distribution the temperature map was calculated. In three areas the maximum temperature reached up to 376 °C. Average gamma heating is 1-4 W/cm^3, but in the corners faced to the active core it is increasing up to 8 W/cm^3.

The distribution of both DPA and gamma heating are very sensitive on the peripheral fuel assembles enrichment and burn up, especially on the closest pin rows to the baffle. This research is the part of V&V process for the methods of precise DPA and gamma heating estimation for further application to power reactors core calculations. The used mock-up core is well defined geometry, there are aims in benchmarking of selected parameters, thus it allows validation of calculation methods, codes and libraries.

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REFERENCES

1. Nuclear power, reactor core, accessed on Feb 17 2017 http://www.nuclear-power.net
2. A.S. Kalchenko, V.V. Bryk, N.P. Lazarev, V.N. Voyevodin, F.A. Garner, Prediction of void swelling in the baffle ring of WWER-1000 reactors for service life of 30–60 years, Journal of Nuclear Materials, June 2013, Pages 415-423, ISSN 0022-3115.
3. V.S. Neustroev, F.A. Garner / Journal of Nuclear Materials 386–388 (2009) 157–160.
4. B.Z. Margolin, I.P. Kursevich, A.A. Sorokin, V.S. Neustrojev: "The Relationship of Radiation Embrittlement and Swelling for Austenitic Steels for WWER Internals", Proceedings of the ASME 2009, Pressure Vessels and Piping conference, Prague, 2009
5. T. Goorley, et al., Initial MCNP6 Release Overview, Nuclear Technology, 180, pp 298-315 (2012).
6. Dassault Systèmes Abaqus Unified FEA 2016. www.3ds.com/products-services/simulia/products/abaqus/.
7. VVER-1000 Physics Experiments Hexagonal Lattices (1.275 cm Pitch) of Low Enriched U(2.0, 3.0, 3.3 wt.% 235U)O2 Fuel Assemblies in Light Water with H3BO3 LR(0)-VVER-RESR-002 CRIT, 2015.
8. M. Koštál, M. Koštál, M. Švadlenková, The pin power distribution in theVVER-1000 mock-up on the LR-0 research reactor, Nuclear Engineering and Design, Volume 242, January 2012, pp 201-214, ISSN 0029-5493.
9. M. Košťál, M. Švadlenková, F. Cvachovec, Calculation and measurement of neutron flux in internal parts of the VVER-1000 mock-up, Ann. of Nucl. En., Vol. 73, (2014), pp. 413-422.
10. D. Harutyunyan, S. Vandlik, M. Schulc, M. Ruscak, E. Novak, Applicability of LR-0 mock-up results to VVER-1000 reactor pressure vessel issues, Annals of Nuclear Energy, Volume 98, December 2016, Pages 157-165, ISSN 0306-4549.
11. M. Košťál, F. Cvachovec, B. Ošmera, Thermal scatter treatment of iron in transport of photons and neutrons, Ann. of Nucl. En., Vol. 37, (2010), pp. 1290-1304.
12. Norgett, M. J., Robinson, M. T. and Torrens, I. M. A proposed method of calculating ν displacement dose rates. Nucl. Eng. Des. 33, 50–54 (1975).
13. M.B. Chadwick, et al. s.l.: ENDF/B-VII.1 Nuclear data for science and technology: cross sections, covariances, fission product yields and decay data Nucl. Data Sheets, 112 (2011), pp. 2887–2996
14. IAEA/NDS IRDF-2002 Damage Cross Section File 2002, https://www.nds.iaea.org/irdf2002/data/irdf2002_damage.dat.
15. A. Sorokin, B. Margolin, I. Kursevich et al. The influence of irradiation on the mechanical properties of materials of WWER-type internals. Voprosy Materialovedenia (Problems of Material Science). — 2011. — №2 (66), pp. 131–151, St. Petersburg (in Russian).
16. Margolin B.Z., Murashova A.I., Neustroev V.S. Analysis of the influence of type of stress state on radiation swelling and radiation creep of austenitic steels. Strength of Materials, (2012) 44: 227.
17. Makhnenko O.V., Mirzov I.V. Two-dimensional numerical analysis of irradiation swelling in WWER-1000 reactor baffle with variation of input data on volumetric heat generation and damaging dose. Strength of Materials, (2014) 46: 689.