IASMiRT

SMiRT 26
50 Anniversary Berlin/Potsdam

Transactions, **SMiRT-26**
Berlin/Potsdam, Germany, July 10-15, 2022
Division II

NEW APPROACH TO IRRADIATION SWELLING AND CREEP ASSESSMENT OF VVER-1000 CORE SHROUD

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ABSTRACT

The reactor core shroud is an important part of VVER-1000 reactor internals. It is made of austenitic stainless steel and consists of four to six rings joint together by bolts. The inner surface has a hexagonal shape which copies the shape of the fuel assemblies. This component serves to protect other parts of the reactor from radiation damage. Radiation swelling is one of the important degradation mechanisms of the reactor core shroud and is dependent, among other, on irradiation temperature, neutron dose and internal stresses.

The assessment of the effect of irradiation swelling of core shroud consists in several steps. The first step is to calculate the distributions of neutron dose and heat sources due to neutron and gamma radiation. In the next step, the thermal-hydraulic calculations are performed to determine values of both coolant temperature and heat transfer coefficients. The third step is thermal analysis of the core shroud. In the fourth step, the elastic-plastic analysis is performed, taking into account analytical formulae for swelling and creep, implemented in the Czech standard NTD AME (2020). In the paper, only partial results of the assessment are presented, such as distribution of swelling deformation in the core shroud cross-section or change of core shroud shape (in the reactor full power state) for the anticipated end of reactor lifetime.

INTRODUCTION

The core shroud of the VVER-1000 reactor is an important part of the reactor internals. This component is made of austenitic stainless steel. The VVER-1000 core shroud consists of four to six rings (depending on actual NPP unit) joint together by bolts. The inner surface has an angular shape that follows the hexagonal shape of the fuel assemblies on the periphery of the reactor core. The outer surface has a circular shape that follows the shape of the surrounding core barrel. The core shroud rings are equipped with vertical channels that are designed to reduce temperature in the core shroud material.

One of the main degradation mechanisms influencing VVER-1000 core shroud is the irradiation swelling (and irradiation creep). It is caused by fast neutron irradiation at elevated temperatures. Swelling degradation occurs practically only in specific locations of the core shroud, where high temperatures and high neutron dose occur. For most PWR internals, these parameters do not reach values high enough for void swelling to develop, but for VVER-1000 core shroud the conditions for formation of the irradiation-induced swelling can be met. The CAD model of VVER-1000 core shroud is seen in Figure 1.

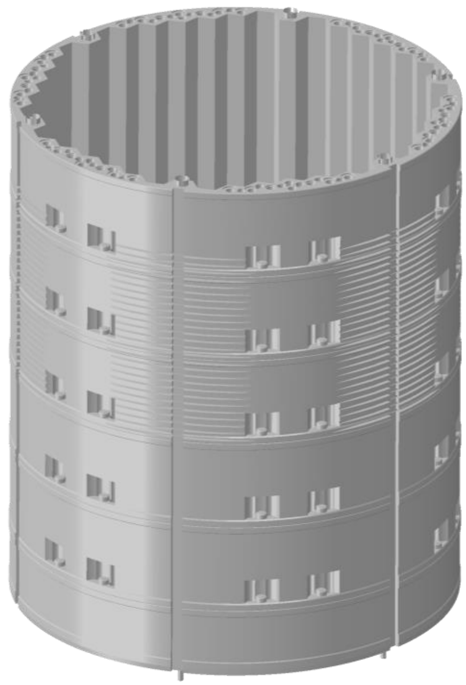

Figure 1. CAD model of the core shroud

STEPS OF THE IRRADIATION SWELLING ASSESSMENT

The assessment of the effect of irradiation swelling of core shroud consists in several steps. The flow chart of the irradiation-induced creep and swelling assessment is given in Fig. 2.

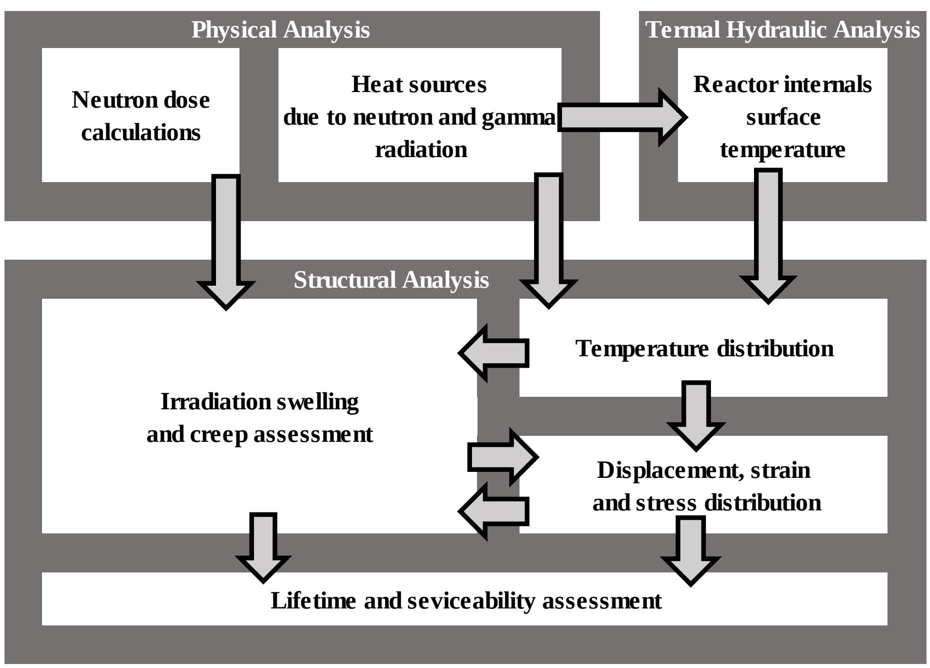

Figure 2. Flow chart of the assessment

PHYSICAL ANALYSES

The first step of the assessment was neutron transport analysis. Using three-dimensional neutron transport code TORT [1], two important results were obtained: (i) distribution of the neutron dose (expressed in d.p.a.) in the core shroud for each fuel campaign during the whole lifetime, (ii) values of internal heat sources due to gamma and neutron radiation. These values were calculated on model consisting of 121 x 78 x 48 cells in R-θ-Z coordinates. Due to reactor core symmetry, only the symmetrical 60° segment was modelled. From these results, the most loaded section was found, on which the load was then calculated on a finer model (455×300×1 cells in R-θ-Z coordinates). The final results were obtained in the form of a 2D horizontal distribution and a 1D axial (Z-axis) distribution.

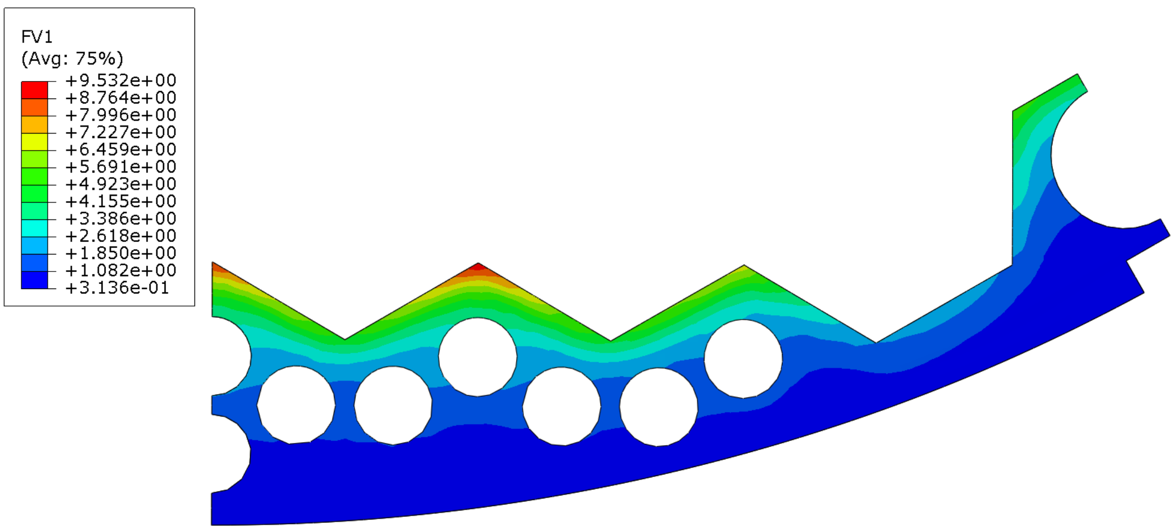

Figure 3. Internal heat sources due to gamma radiation in cross-section with maximum temperature in selected campaign (in W/cm^3)

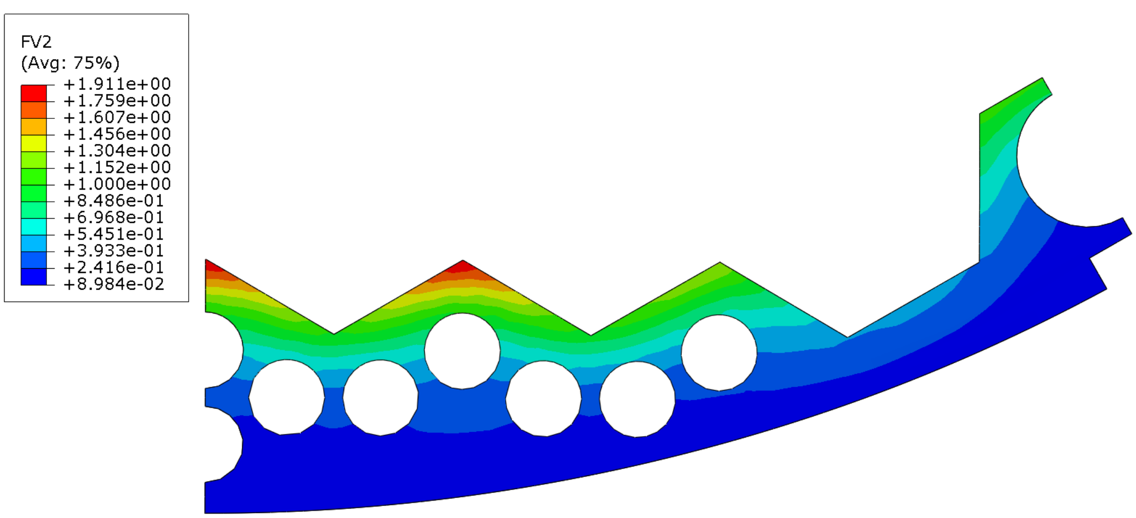

Figure 4. Distribution of neutron dose per selected campaign in cross-section with maximum temperature (in d.p.a.)

THERMAL-HYDRAULIC CALCULATIONS

The system thermal-hydraulic calculations were performed by the RELAP5 program. Resulting fluid temperatures and heat transfer coefficients were used as boundary conditions in the subsequent thermal and structural analysis. Stationary full-power reactor state and cold state was analyzed.

THERMAL AND MECHANICAL ANALYSIS

Thermal and mechanical analysis of the core shroud was performed using finite element (FE) code ABAQUS. Three-dimensional FE computational model representing one symmetrical twelfth (30° segment in the circumferential direction) of the core shroud and core barrel was constructed. The model includes bolts joining the individual rings. Contacts between individual components of the core shroud were modelled, including mutual contacts between core shroud rings and bolts. The structural calculations were performed for the anticipated lifetime (60 campaigns). As a first step of the thermal-mechanical analysis, the inhomogeneous temperature field in the modelled structural components was calculated for each fuel campaign using fluid temperatures and heat transfer coefficients (obtained from thermal-hydraulic calculations) as boundary conditions and internal heat sources due to gamma and neutron irradiation (obtained from neutron transport calculations). In the second step, visco-elastic-plastic analysis was performed successively for each fuel campaign to calculate creep, swelling, and plastic strain and stress fields. The structure was loaded by non-homogeneous thermal field obtained in the first step. The creep and swelling strains were calculated based on the non-homogeneous neutron dose, temperature and stress-strain fields using the formula (1) presented in the text below.

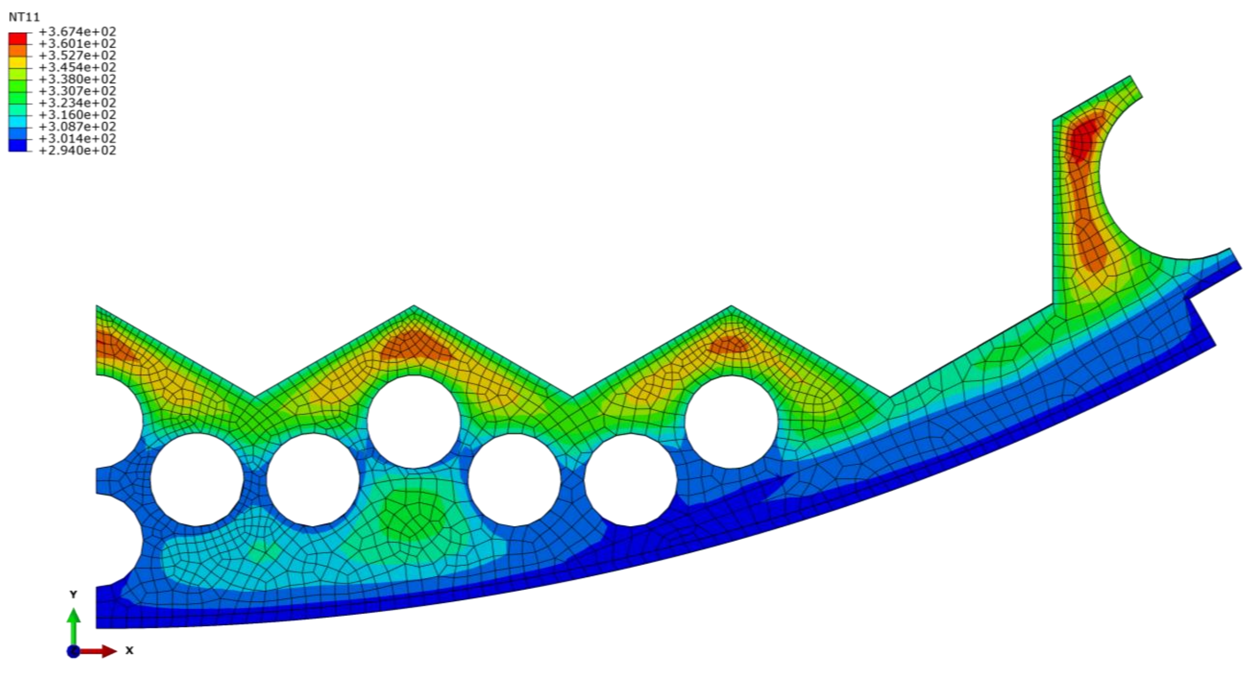

Figure 5. Distribution of temperature in the predicted (selected) campaign for reactor full power state (in °C). Cross-section corresponding to maximum temperature.

The volumetric swelling strains, creep strains and stress and plastic strain fields are coupled according to constitutive formulae given by the standard [2]. These constitutive formulae were implemented using ABAQUS Fortran subroutines. The swelling and creep strains calculated at the end of each fuel campaign were used as initial conditions for the next fuel campaign.

CALCULATION OF SWELLING STRAINS

Values of swelling strains were calculated according to empirical formulae developed on the basis of experiments performed in a research reactor. The input parameters are neutron dose and irradiation temperature calculated for all points of core shroud in the previous steps of the assessment. The dependency describing volumetric strain ε_V due to swelling as a function of irradiation temperature is not linear and is expressed by the following equation (1), where $\dot{S}_o$ is free swelling rate, F is neutron dose in [d.p.a.], Φ is neutron dose rate [d.p.a./s], T_{irr} is irradiation temperature [°C], n_v=1.88; r = $1.825 \cdot 10^{-4}$ °C^{-2}; T_m = 470 °C; c=$1.035 \cdot 10^{-4}$.

$$\dot{\varepsilon_V} = \frac{1}{3}\dot{S_0} = \frac{1}{3} c \cdot n_v \cdot \mathrm{F}^{(\mathrm{n_v}-1)} \cdot \Phi \cdot \exp[-r \cdot (T - T_{\mathrm{irr}})^2] \qquad (1)$$

The equation (1) is valid only for the free swelling, which means that it holds only for a mechanically unconstrained piece of material. The detailed equations for the calculation of the swelling and creep deformations in the presence of stresses and plastic strains are given in the standard NTD AME [2].

Volumetric swelling and creep strains were calculated for the anticipated lifetime of the assessed reactor unit. The magnitude of the swelling strain was compared with its limiting value of 7% which is critical for the formation of limit embrittlement area (LEA).

The change of core shroud shape (core shroud dimensions) due to irradiation-induced creep and swelling was calculated for full power state at the end of the reactor anticipated lifetime.

RESULTS

From Figure 5, it is seen that maximum temperature in the core shroud is 367°C for the predicted (selected) campaign.

The highest values of von Mises stress were found in the ring flanges around the holes for the connecting bolts and were equal to about 300 MPa (not shown in the Figures). Maximum von Mises stress value found in the core shroud cross-section (for full power reactor state, at the end of anticipated reactor lifetime) was 73 MPa, see Fig. 6.

Distribution of axial stress in the core shroud cross-section (for full power reactor state, at the end of anticipated reactor lifetime) is shown in Fig. 7. From this Figure, it is seen that maximum value of axial stress is about 77 MPa.

Distribution of volumetric swelling strain (calculated with consideration of creep strain) in core shroud cross-section (for anticipated end of reactor lifetime) is shown in Fig. 8. From this Figure, it is seen that the highest value of radiation swelling is reached near the large cooling channel, and it takes value of 1.86 %. Since this value is lower than the limit value of 7%, we do not expect the occurrence of the limit embrittlement area (LEA), for details regarding LEA see [2].

Distribution of radial displacements in the core shroud cross-section (for full power reactor state, at the end of anticipated reactor lifetime) is shown in Figure 9. From this Figure, it is seen that the highest values of radial displacement (due to thermal expansion, radiation swelling and creep) are located on the outer surface of the core shroud in the vicinity of the large cooling channel, and they take values of about 9.6 mm. The initial minimum gap in the cold state before the first campaign between the core shroud and fuel assemblies is 4 mm and between the core shroud and core barrel is 2.5 mm. At the end of the reactor anticipated lifetime, in the state of reactor full power, the minimum gap between core shroud and fuel assemblies is 2.77 mm. The minimum gap between core shroud and core barrel in the reactor full power state after the 60th campaign is 0.92 mm. Thus, it may be expected that at the end of anticipated reactor lifetime, there will be no contact between the core shroud and the fuel assemblies and between the core shroud and core barrel (in the state of reactor full power).

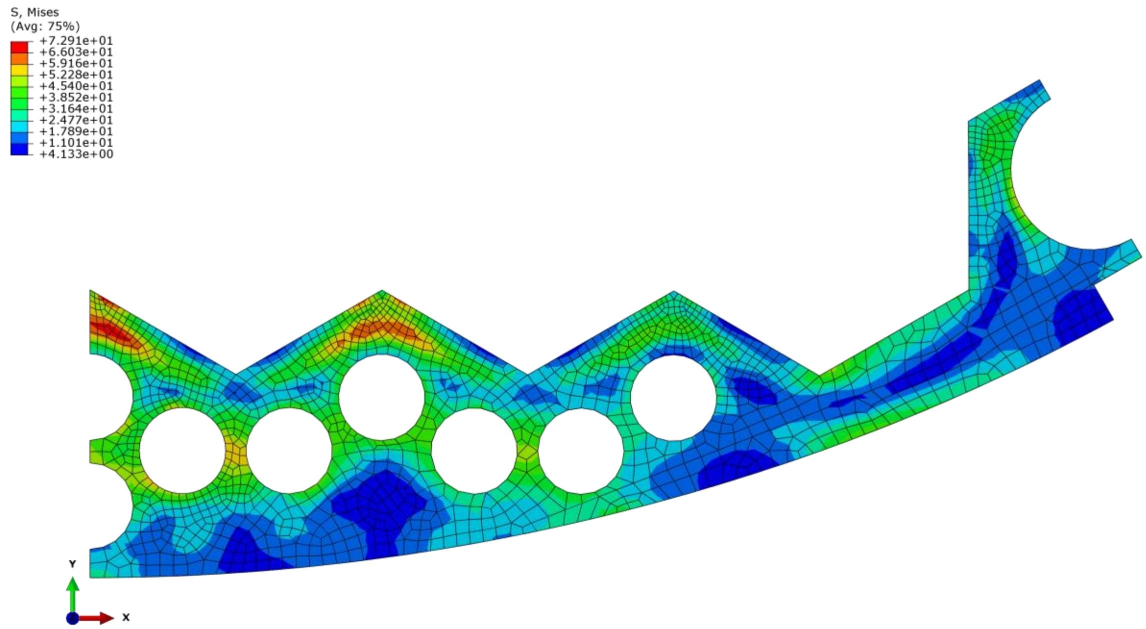

Figure 6. Distribution of von Mises stress for 60 years of reactor operation, for reactor full power state (in MPa)

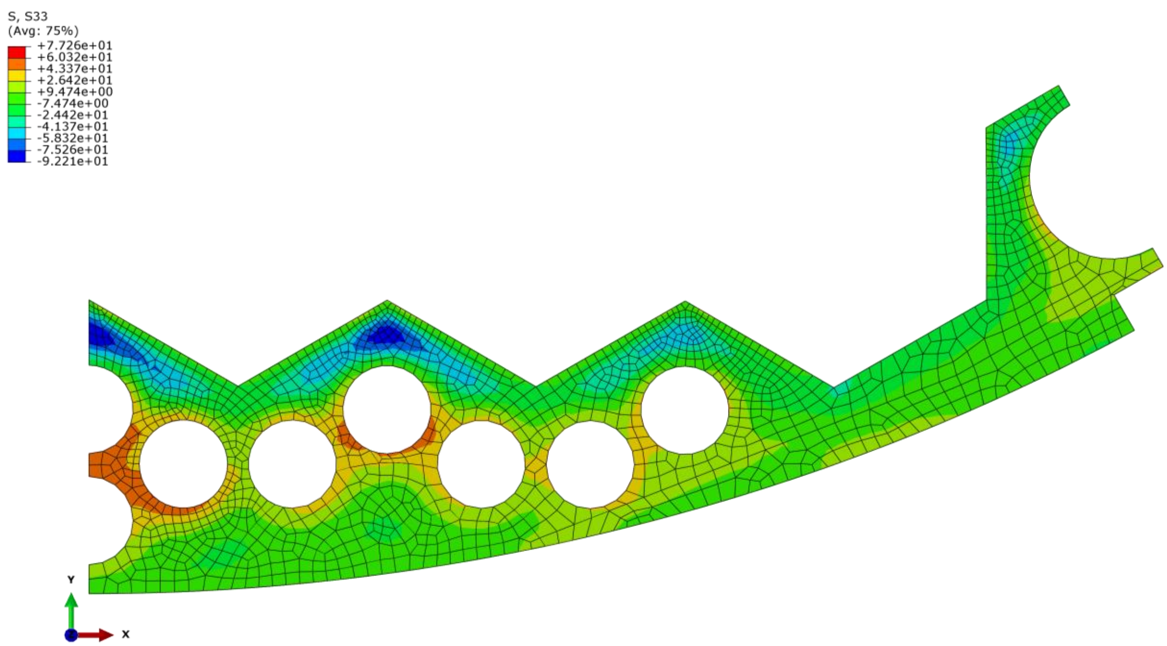

Figure 7. Distribution of axial stress for 60 years of reactor operation, for reactor full power state (in MPa)

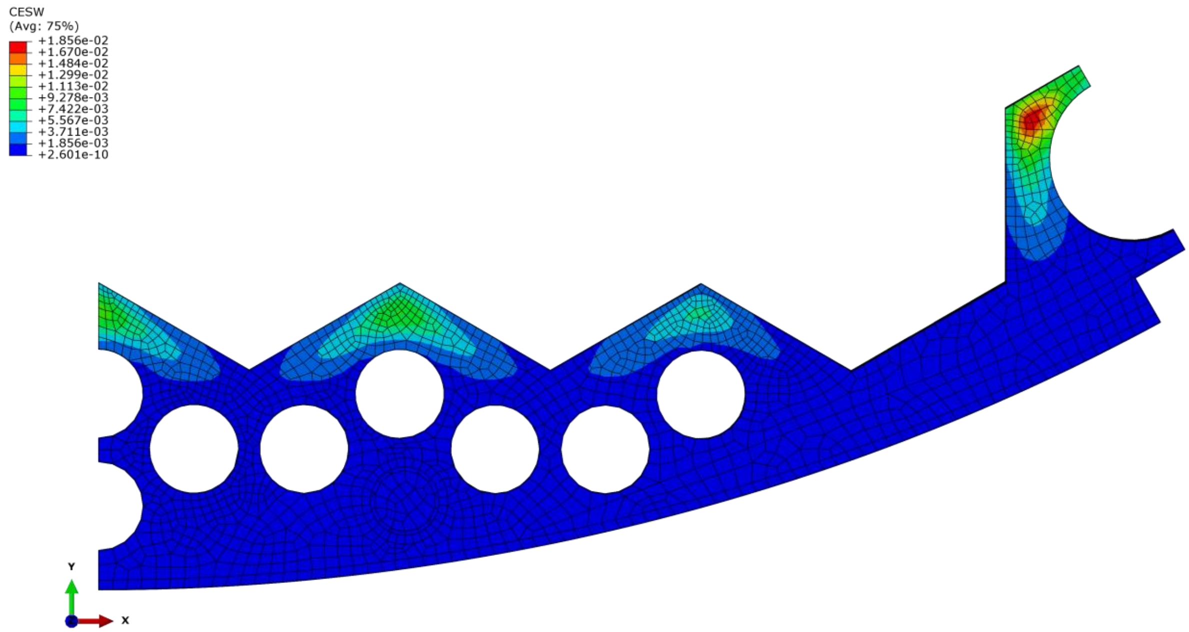

Figure 8. Distribution of volumetric swelling strain (calculated with effect of creep) due to irradiation swelling for 60 years of reactor operation

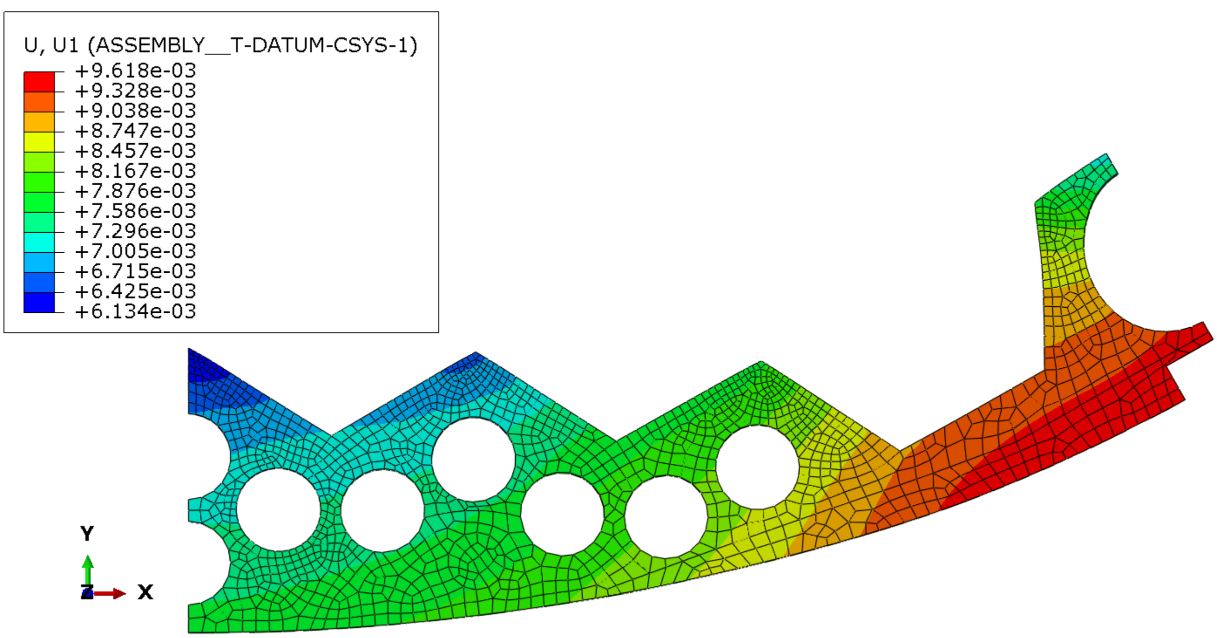

Figure 9. Distribution of radial displacement due to irradiation swelling for 60 years of reactor operation in full power state (in m)

CONCLUSION

In the paper, evaluation of radiation swelling and creep in the VVER-1000 core shroud was performed, for the period up to 60 reactor operation campaigns (reactor anticipated lifetime). The evaluation was performed using analytical formulae for the development of radiation swelling and creep, as presented in the Czech standard NTD AME [2].

The maximum value of swelling volumetric strain (including the effect of creep in the calculation) in the core shroud was found to be 1,86 % after the 60th campaign, which is lower than the critical value for limit embrittlement area development (7%), and therefore the occurrence of the limit embrittlement area (LEA) during the period of 60 campaigns is not expected. Based on the performed evaluations, it was found that at the end of expected reactor lifetime, there will be no contact between the core shroud and the fuel assemblies and between the core shroud and the core barrel (in the state of reactor full power).

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