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Division II

ASSESSMENT OF IRRADIATION SWELLING EFFECTS ON VVER-1000 CORE SHROUD

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ABSTRACT

VVER-1000 core shroud is significant part of the reactor internals, made of austenitic stainless steel. It was designed to create external shroud of reactor core consisting of fuel assemblies. The irradiation swelling is one of the main degradation mechanisms influencing VVER-1000 core shroud. Swelling refers to the volume change of material under neutron irradiation. It is generally depending on neutron dose, temperature and operational stresses. Degradation occurs only in the special locations of the core shroud.

The assessment of the effect of irradiation swelling of core shroud consists in several steps. First of them is physical analysis to determine the distribution of neutron dose in the core shroud cross-section and the values of internal thermal sources due to gamma radiation. The second step is performing thermal-hydraulic calculations for determination of values of coolant temperature and heat transfer coefficients in the cooling channels of the core shroud and outside of it. The third step is thermal analysis of the core shroud. Radiation swelling is treated as an internal deformation which is calculated according to formula determined empirically based on experiments preformed in research reactor. Finally, elastic-plastic analysis is performed to determine the overall deformation of the core shroud. It was concluded that the changes of core-shroud dimensions during its lifetime remain within acceptable limits.

INTRODUCTION

VVER-1000 is a pressurised water reactor type of Russian design. Its core shroud is significant part of the reactor internals, made of austenitic stainless steel 08Kh18N10T. It was designed to create external shroud of the reactor core consisting of fuel assemblies. Its purpose is to reduce the intensity of neutron flux on the reactor pressure vessel (RPV) wall and to control the flow of coolant in the reactor. Core shroud is a cylindrical shell consisting of five rings, joint together by bolts and thread bars. Rings have cylindrical outer surface and angled inner surface that follows hexagonal shape of fuel assemblies on boundary of reactor core. In the body of the rings there are vertical channels for cooling otherwise fairly robust component. View on the core shroud is in Fig. 1; its position inside the reactor pressure vessel is seen in Fig. 2.

The RPV and the reactor internals are the most important and also most expensive and hardly replaceable components of nuclear power plant (NPP). In view of this, the long term operation of NPP is mainly limited by the lifetime of RPV and its internals. The reassessments of durability and serviceability of NPP critical parts are strictly required for licensing of life extension. All possible degradation mechanisms must be taken into account to perform the assessment adequately. The most important degradation mechanism for core shroud is the irradiation-induced swelling. Swelling refers to the volume change of material under neutron irradiation. It is generally depending on neutron dose, temperature and operational stresses. Degradation occurs only in the special locations of the core shroud, where high temperature (more than 400°C) and high neutron dose occur. For most PWR reactor types, the temperature and neutron dose do not reach sufficiently high values for the radiation swelling to occur, but for VVER-1000 core shroud the conditions for irradiation-induced swelling can be met.

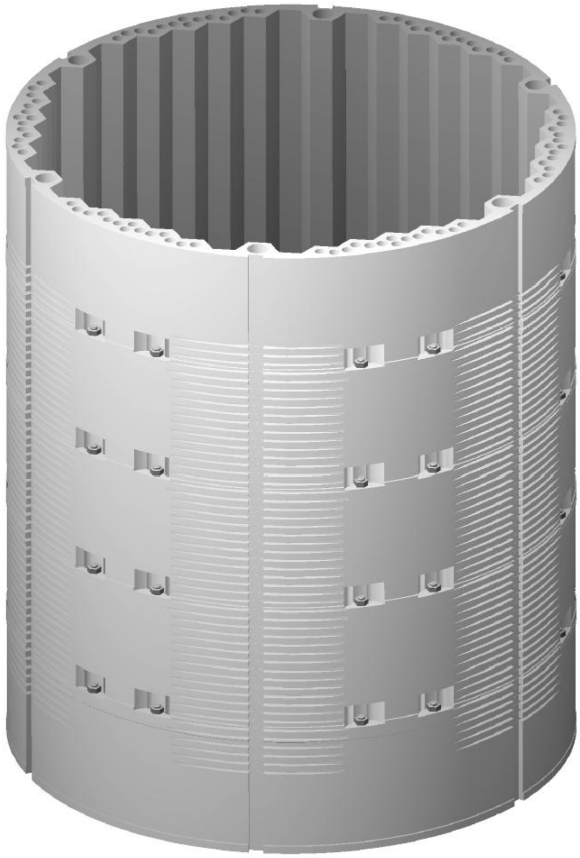

Figure 1. View on the core shroud

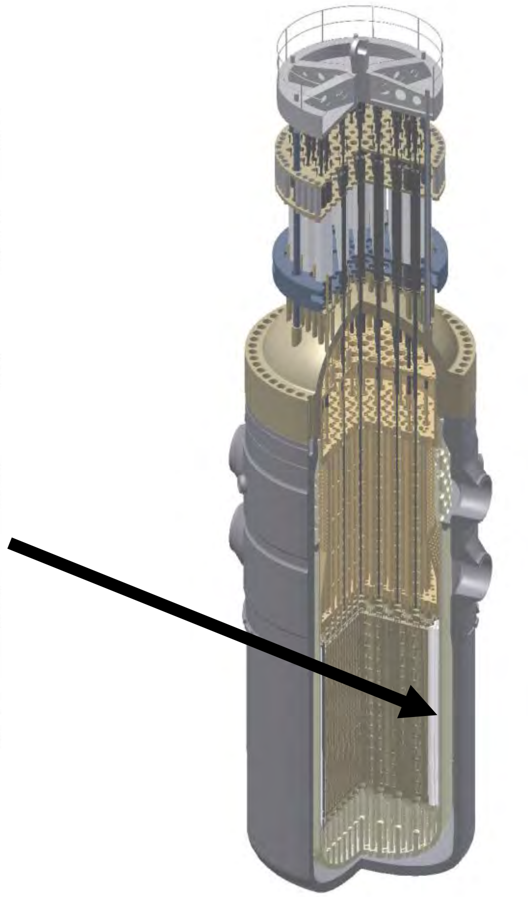

Figure 2. Position of core shroud inside the reactor

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 swelling assessment is given in Fig. 3.

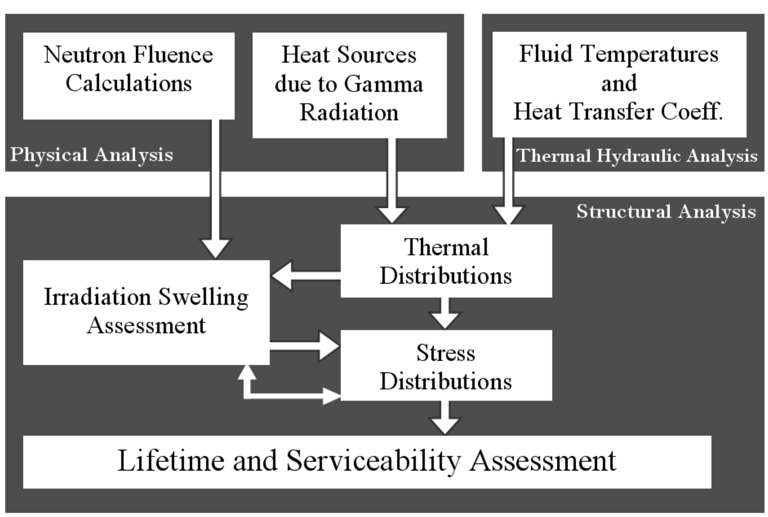

Figure 3. Flow chart of the assessment

Physical Analyses - Neutron Dose and Gamma Heat Calculations

Physical analysis is the first step of the assessment. Neutron transfer code TORT, see Rhoades and Mynatt (1994), was used for this analysis, with application of BUGLE96 cross-sections library, see DLC-185/BUGLE96 (1996). Geometry of the reactor core, core shroud including thread bars, core barrel and reactor pressure vessel, including the volumes filled by water, enters the calculation model. First, analysis on relatively coarse model (121×78×48 cells in R-θ-Z coordinates) was performed to find the most loaded core shroud cross-section (i.e. its position in vertical direction). Due to reactor core symmetry, only the symmetrical segment with angle 60º was modelled. Finally, the refined 2D model of the selected cross-section with 455×300×1 cells in R-θ-Z coordinates was used. Two types of results were determined: (i) distribution of neutron dose in the core shroud cross-section, for the most exposed cross-section and for each campaign during the whole lifetime, (ii) values of internal thermal sources due to gamma radiation. Examples of the results for one selected campaign are presented in Figures 4 and 5. Notice that parts of the model not seen in the figures correspond to the lowest values on the scale. The highest neutron doses and heat sources are close to the inner surface, i.e. close to the reactor core.

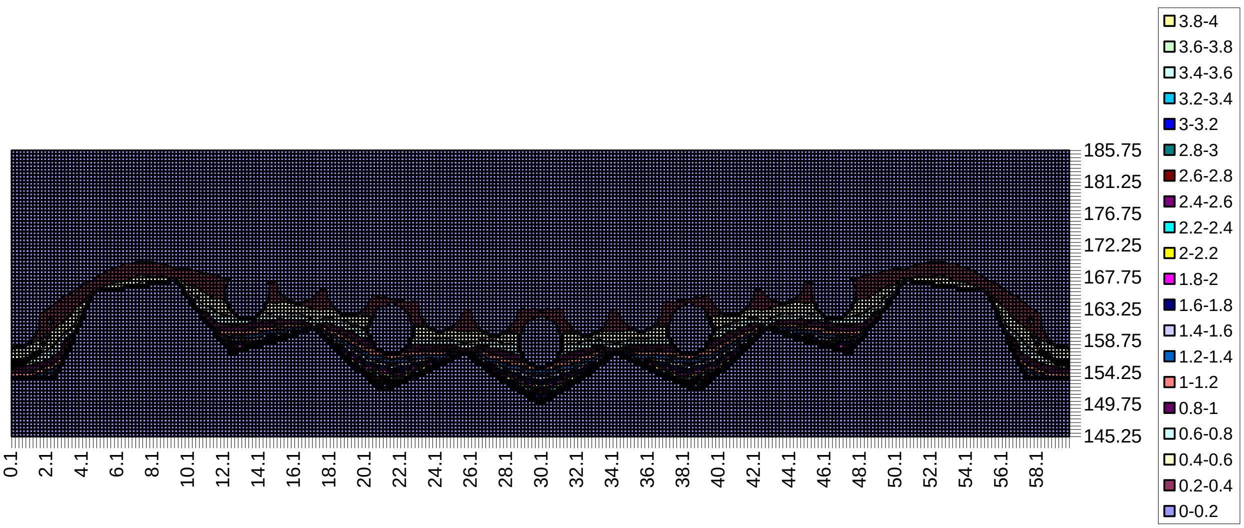

Figure 4. Distribution of neutron dose in the core shroud cross-section (in d.p.a.)

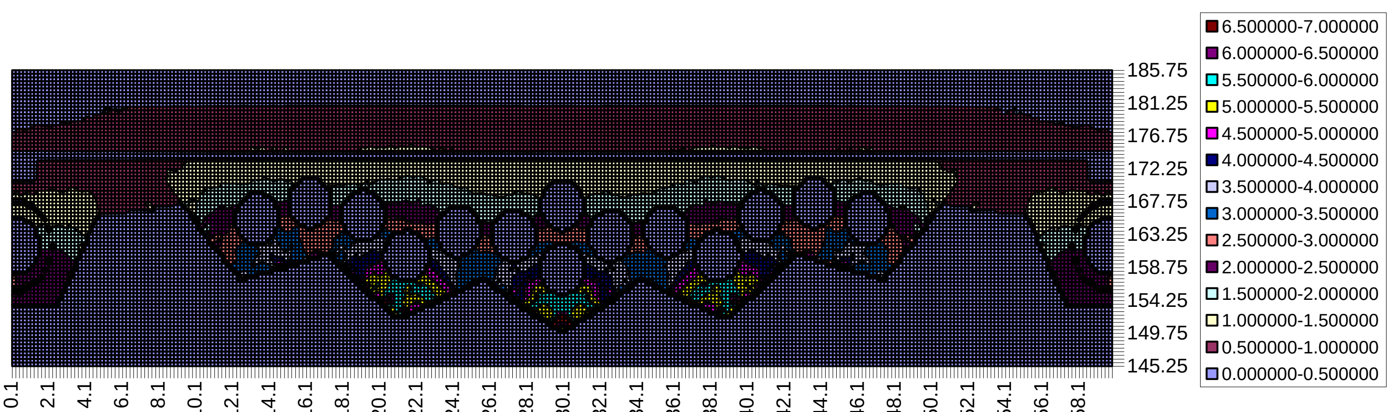

Figure 5. Internal thermal sources due to gamma radiation (in W/cm^3)

It was shown during the analyses that the resulting neutron doses were in accord with other similar analyses performed previously, while the calculated internal thermal sources due to gamma radiation were slightly overestimated. The overestimated internal thermal sources due to gamma radiation were recognised as a bug in the BUGLE96 cross-sections library (notice that the option for calculation of heat sources due to gamma radiation is not often used within the BUGLE96 cross-sections library, as this effect is not significant in most PWRs). Finally, the calculated internal thermal sources were slightly adjusted.

Thermal-hydraulic calculations

Thermal-hydraulic calculations were performed using the RELAP5 code. They were performed on the model of the whole NPP system for steady-state conditions (reactor operation at full power); the results obtained in the core shroud cooling channels and on inner and outer surfaces of the core shroud were considered as the boundary conditions in the subsequent thermal calculations of the core shroud. In particular, thermal-hydraulic calculations provided values of coolant temperature and heat transfer coefficient. The coolant temperature inside the core shroud was taken conservatively as that one of the upper part of reactor core, i.e. 320 ºC, coolant temperature inside the cooling channels and outside the core shroud was taken equal to about 292 ºC.

Thermal analysis

Steady-state thermal analysis of core shroud was performed with using finite element (FE) code SYSTUS. Due to reactor core and core shroud symmetry, two-dimensional FE computational model representing 30 º segment (in circumferential direction) of the core shroud cross section was used. Boundary conditions were taken from the results of thermal-hydraulic analyses. Internal thermal sources due to gamma radiation were taken into account. Example of results for one selected campaign is presented in Figure 6. Maximum temperature of 420 ºC was reached in the middle of ligament between inner surface and coolant channel on the symmetry plane (full grey circle in Fig. 6).

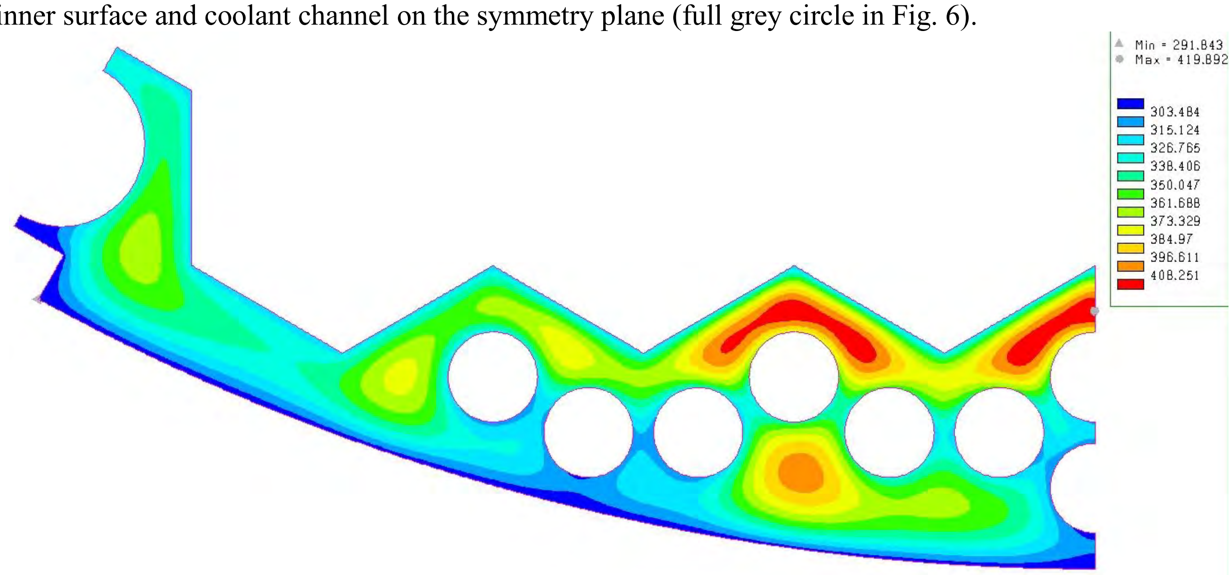

Figure 6. Temperature field in core shroud cross-section (in ºC)

Calculation of swelling deformations

Values of swelling deformations were calculated according to empirical formula based on experiments performed in research reactor; see Votinov et al. (1987) or Sharyi et al. (2004). The input parameters are neutron dose and irradiation temperature calculated for all points of core shroud cross-section in the previous steps of the assessment. The dependency describing volume deformation ε_V due to swelling as a function of temperature is not linear and is expressed by the following equation (1), where Φ is neutron dose [d.p.a] and T is irradiation temperature [°C].

$$\varepsilon_V = \frac{\Delta V}{V} = 0.55(\Phi + 0.1T - 67) \cdot \exp\left[-29.10^{-5}(T-485)^2\right] \tag{1}$$

For $(\Phi + 0.1T - 67) < 0$, i.e. for low dose, $\varepsilon_V = 0$ is taken.

Volume deformation ε_V due to swelling according to formula (1) expressed in %, in dependency on neutron dose Φ [d.p.a] and irradiation temperature T [°C] is presented in Fig. 7.

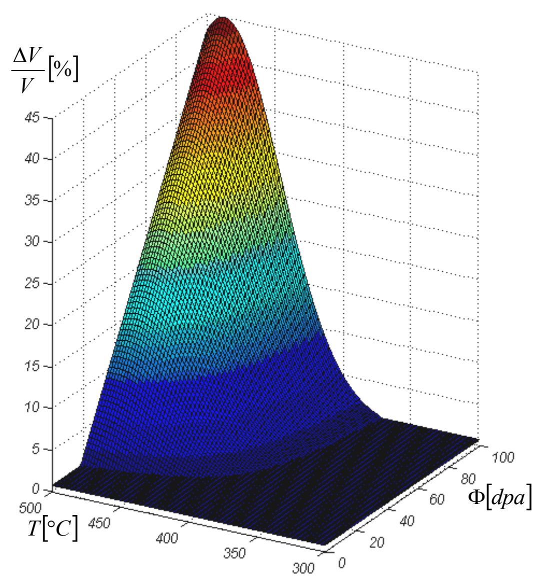

Figure 7.Volume deformation ε_V due to swelling in dependency on neutron dose and irradiation temperature

Elastic-plastic analysis

Core shroud is not significantly mechanically loaded. Therefore, elastic-plastic analyses were conducted only for loading by irradiation swelling occurring in several local areas. The analyses were performed again by SYSTUS code on the same mesh as the thermal analyses. Swelling formations were considered in the FE calculation as initial deformations prescribed in the elements of the FE model (different in different elements). Three components of induced swelling deformation were applied in each element:

$$\varepsilon_x = \varepsilon_y = \varepsilon_z = \sqrt[3]{\varepsilon_V + 1} - 1 \tag{2}$$

Example of induced swelling deformation ε_x for 60 years of reactor operation is given in Figure 8. It is seen that swelling deformation occurs only in limited areas close to inner surface. The maximum value ε_x = 1.5 % is reached in the position of maximum temperature (see Figure 6).

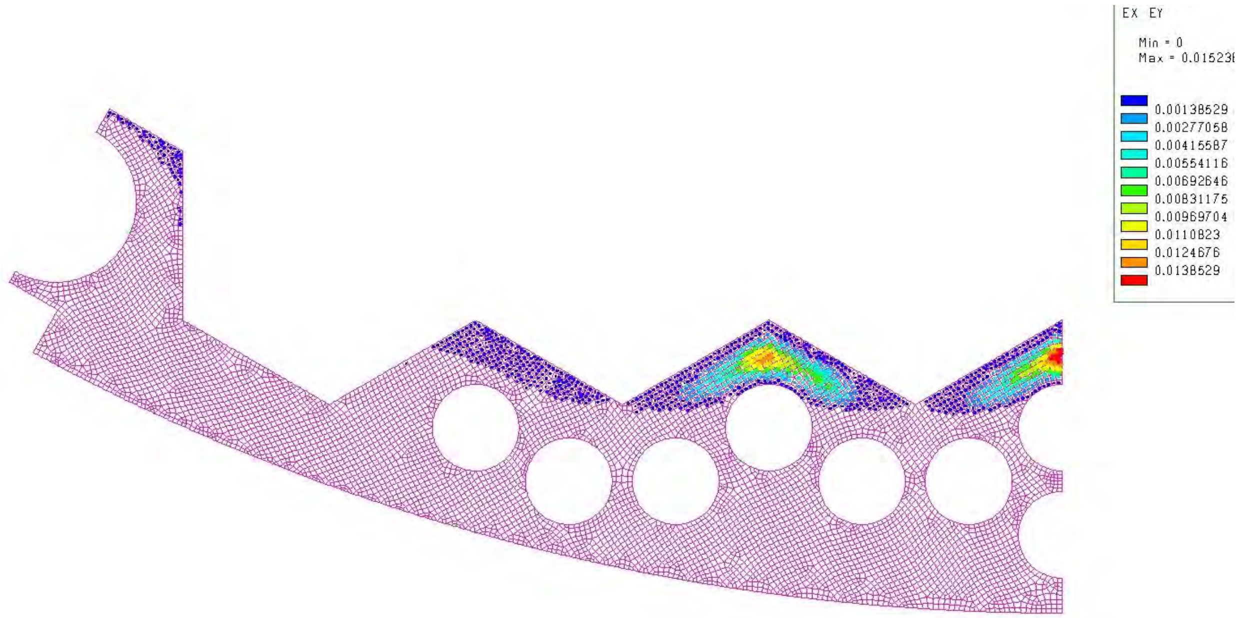

Figure 8. Induced swelling deformation ε_x for 60 years of reactor operation

The difficulty of the whole problem was amplified by the fact that mechanical properties of the austenitic steel are generally depending not only on temperature but also on neutron dose. Radiation damage (strongly varying over the cross section) affects the appropriate stress-strain curve: the values of stress-strain curve are increased, and yield and ultimate strengths get closer together. Due to this fact, model of perfect plasticity was used. The changes of yield strength due to irradiation were taken into account, individually in each element of the FE model.

RESULTS

Main results of the analyses are presented in Table 1. In Figures 9 – 12, there are presented resulting distributions of Tresca stress, axial stress, cumulative plastic strain and radial displacement in the core shroud cross-section, respectively. The results are drawn for 60 years of reactor operation.

It is seen from the results presented in Table 1 that significant onset on swelling starts at the period of long-term operation (beyond design lifetime)

Table 1. Main results of the analyses

Operation	Maximal induced swelling deformation ΔV/V	Maximum stress (Tresca)	Maximum resulting plastic strain	Maximum radial displacement (inwards)	Maximum radial displacement (outwards)
[years]	[%]	[MPa]	[%]	[mm]	[mm]
25	0,4	185	0,0	0,02	0,000
40	2,1	871	0,2	0,5	0,3
60	4,6	925	1,4	1,6	1,2

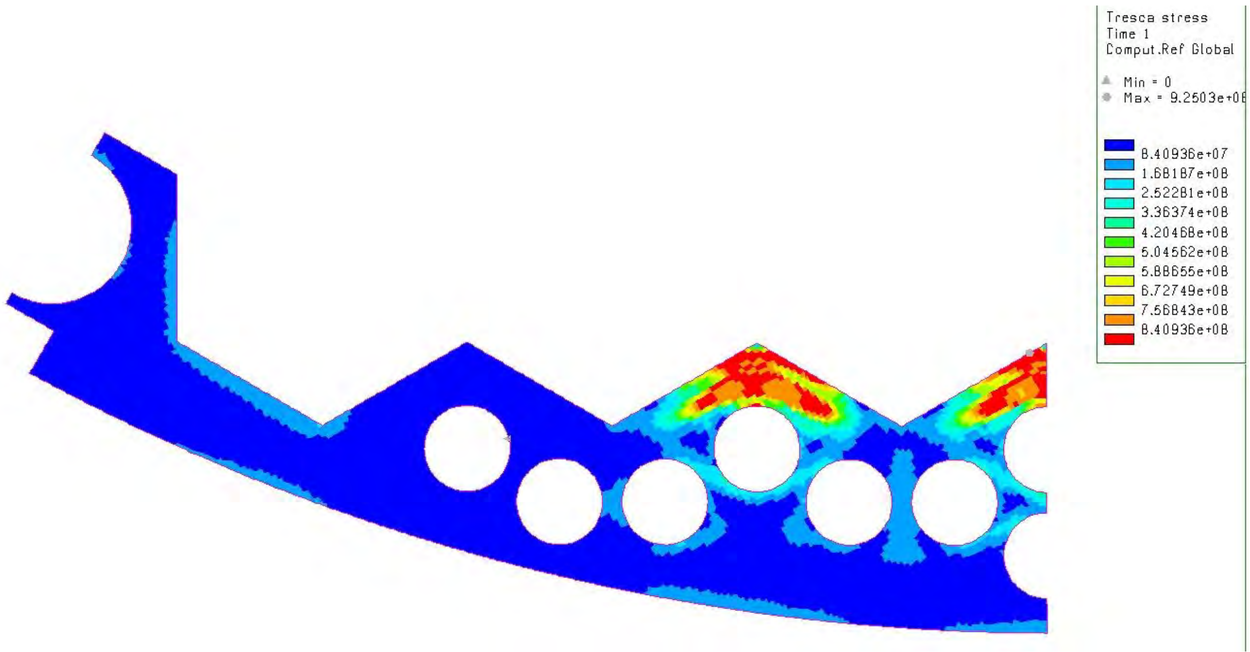

Figure 9. Resulting Tresca stress due to irradiation swelling for 60 years of reactor operation (in MPa)

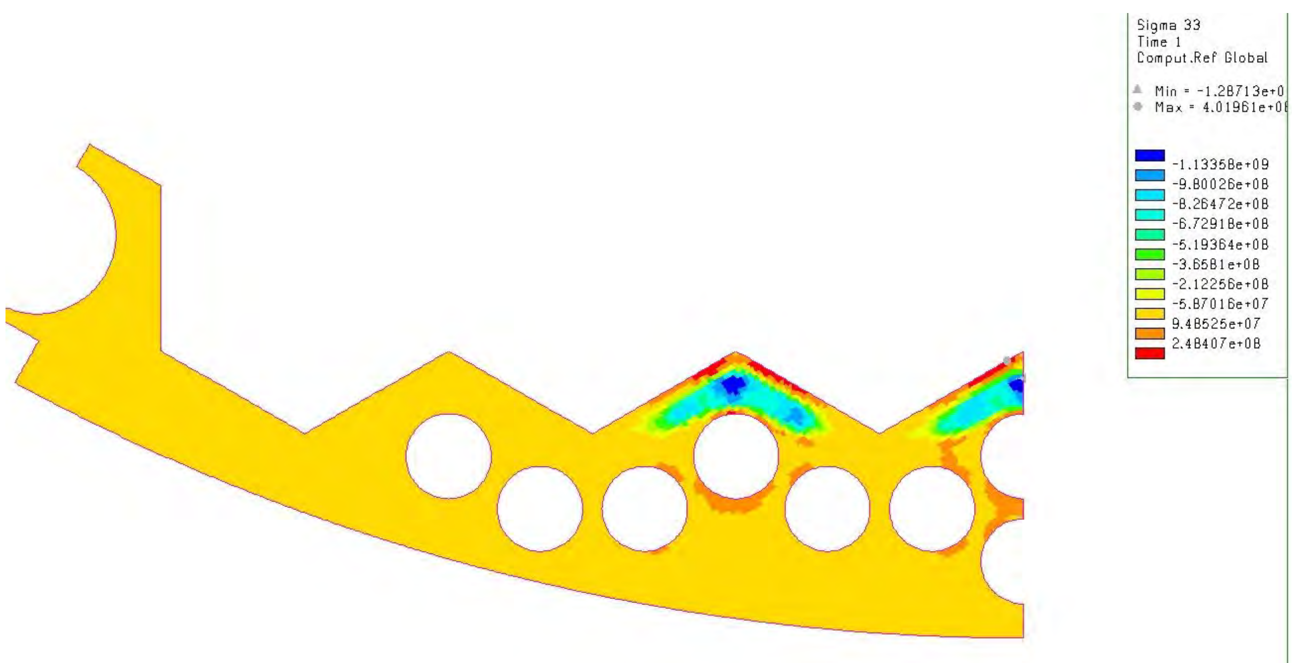

Figure 10. Resulting axial stress due to irradiation swelling for 60 years of reactor operation (in MPa)

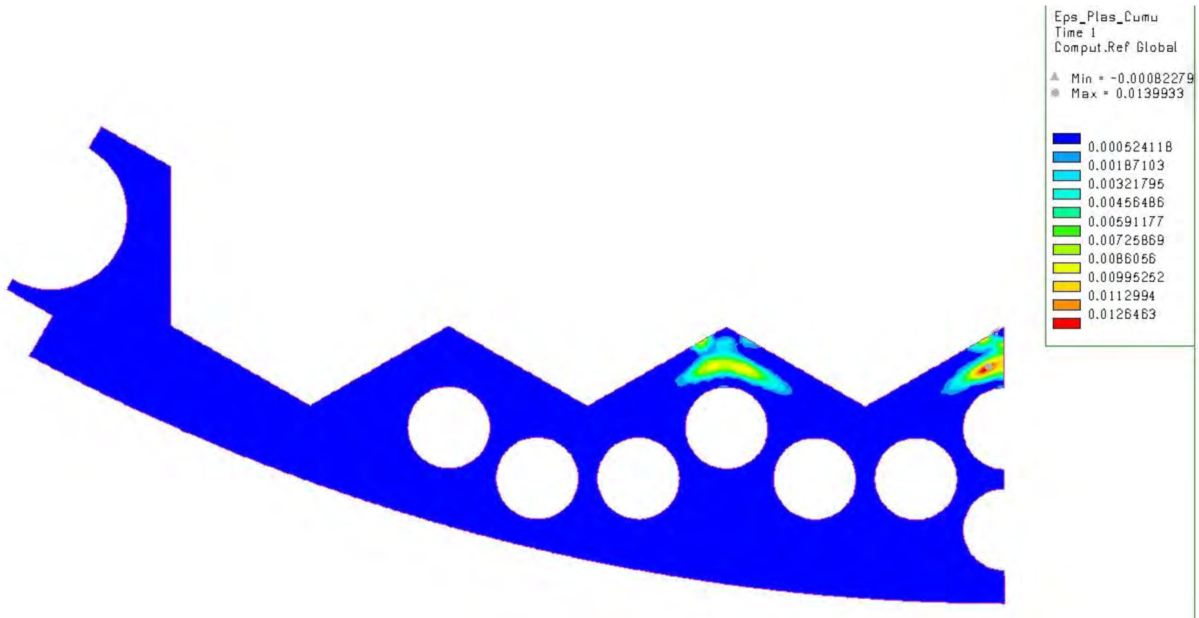

Figure 11. Resulting plastic strain due to irradiation swelling for 60 years of reactor operation

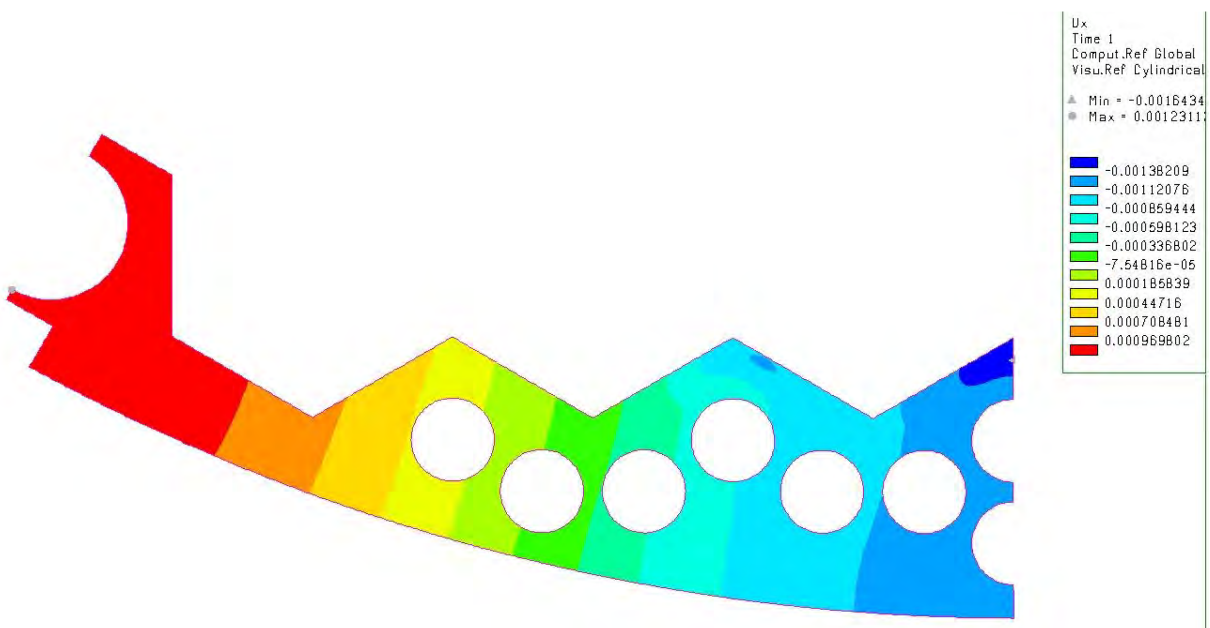

Figure 12. Resulting radial displacement due to irradiation swelling for 60 years of reactor operation (in m)

It is seen from Figure 10 that in the region of maximum swelling (i.e. maximum induced swelling deformation) the maximum compressive stresses occur, while at the "boundary" of this region tensile stresses appear. One side of the symmetrical section of core shroud "moves" inwards (towards the axis of RPV) while the other side "moves" outwards (i.e. in opposite direction).

The resulting change of core shroud shape expressed in terms of radial displacements was finally compared with dimensions of gaps between the core shroud and surrounding structural components (core barrel close to "outer surface" of core shroud and fuel assemblies close to "inner surface" of core shroud). Excessive reduction of these gaps could have a negative impact on the component serviceability and possibility of disassembling of the system. It was concluded that the changes of core-shroud dimensions during its lifetime remain within acceptable limits.

CONCLUSION

Assessment of effect of irradiation swelling for VVER-1000 core shroud was performed, consisting in several steps. The maximum volumetric swelling deformation reaches 4,5% for long-term operation of 60 years. The significant swelling deformation appears when approaching the end of the lifetime. It was shown that the radial displacements of the components (1.6 mm inwards and 1.2 mm outwards) remain in acceptable limits from the point of view of gaps between the core shroud and surrounding structural components during the long-term operation. Even at the end of the assessed period (60 years), significant part of core shroud cross-section thickness remains in elastic state.

In the future, it is planned to perform the core shroud evaluation according to the prepared normative document VERLIFE (2013) in which also the effect of radiation creep will be included. It was also recommended to perform re-analysis of irradiation swelling for the core shroud after issuing new revision of BUGLE96 cross-sections library to establish more precise internal thermal sources due to gamma radiation. Additionally, it was recommended to perform periodical measurements of dimensions of the core shroud to be able to capture the beginning of macroscopic deformations of the component due to irradiation swelling.

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