Brittle fracture strength analysis for reactor pressure vessel of VVER-1000 reactor unit
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Abstract

At the moment 11 nuclear power plants (NPP) are under operation in the Russian Federation. More than half of the reactor units used are VVER-1000 reactors. NPP is a complex engineering structure consisting of plenty of complex equipment. Equipment operates under severe conditions: hard radiation, high temperatures, and high pressure. Nuclear reactor has a finite lifespan of 30 years.

Most nuclear power plants based on VVER-1000 reactors have almost exhausted their lifespan. Therefore, these reactors require a life extension analysis. This paper demonstrates an analysis of brittle fracture strength for VVER-1000 reactor pressure vessel. According to the findings, calculated parameters do not outreach the 10 % limit.

Keywords: VVER-1000, brittle fracture strength, ANSYS;

1. Introduction

Nuclear power plants have 19 % of the total energy output of the Russian Federation and this percentage has been growing each year. It is known that reactor pressure vessels are used under harsh conditions (e.g. high temperatures, high pressures, ionizing radiation) that lead to the constant wearing of structural materials. Some of these materials are usually replaced with new ones during an outage. However, reactor pressure vessel replacement is not possible, hence, it requires studies concerning its strength. In case of improper maintenance or if orders are breached, the equipment could be broken down and it will lead to an accident. Such an accident leads to dose limit overrun for NPP staff and to people around NPP. Moreover, the environment could be contaminated by radiation. Remediation would cost billions of rubles apart from the loss of electric power source, where it could not be replaced.

2. Calculation of thermal fields, stress-strain state, and stress intensity factor.

To calculate temperature fields and stress-strain state a thermo mechanical problem with the dependence of material properties on temperature has to be solved. The following parameters were used as the loads:

- internal pressure [1];
- temperature distribution[2];
- residual stresses [3, 4].

Figure 1 illustrates a temperature distribution and heat-transfer coefficient with boundary conditions at the inner RPV surface during the thermal stress accident.
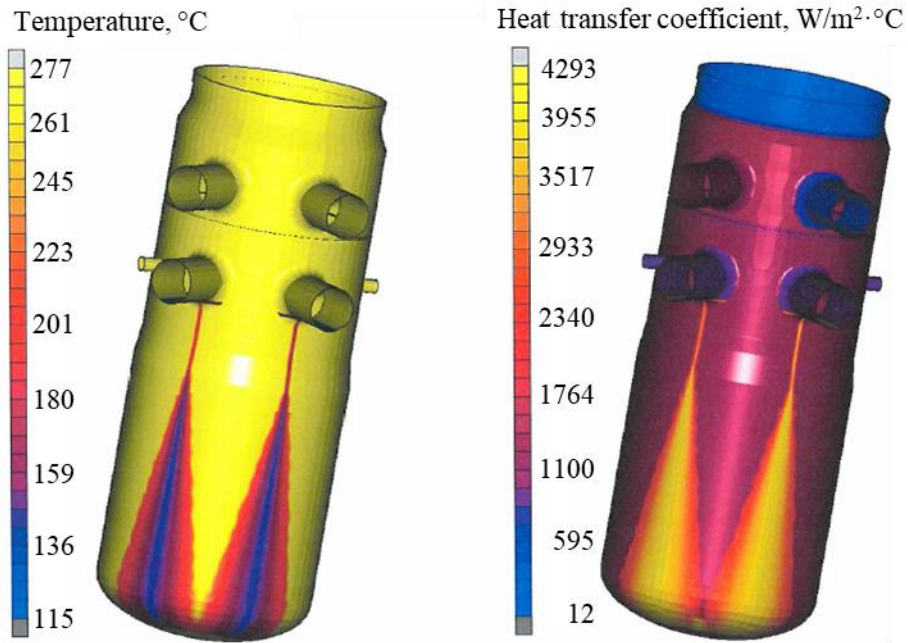

Fig. 1. Temperature distribution and heat transfer coefficient with boundary conditions at the inner RPV surface during the thermal stress accident

Boundary conditions for a discrete finite-difference method (FDM) require use of zeroed transition throughout the RPV axis at the upper end. Moreover, an inner pressure on RPV inner surface should be taken into account. The pressure at the ends of Du 850 and emergency core cooling system (ECCS) nozzles is calculated as follows:

$$p_x = p \cdot R_{in}^2 / \left(R_{out}^2 - R_{in}^2 \right), \tag{1}$$

where: R_{in}, R_{out} signify the inner and outer RPV radii or pipeline radii, respectively, m.

A discrete FDM model shown in figure 2 was used to carry out the calculation of temperature fields and stress-strain state for a longitudinal sub-cladding crack of welded junctions.
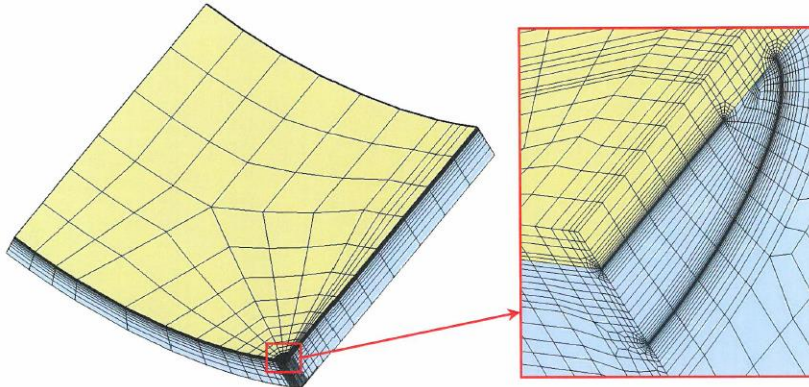

Fig. 2. A longitudinal sub-cladding crack on the metal of welded junctions № 3-5

Figure 3 illustrates an FDM model for the calculation of temperature fields and stress-strain state of the longitudinal sub-cladding crack of Du 850 nozzle.
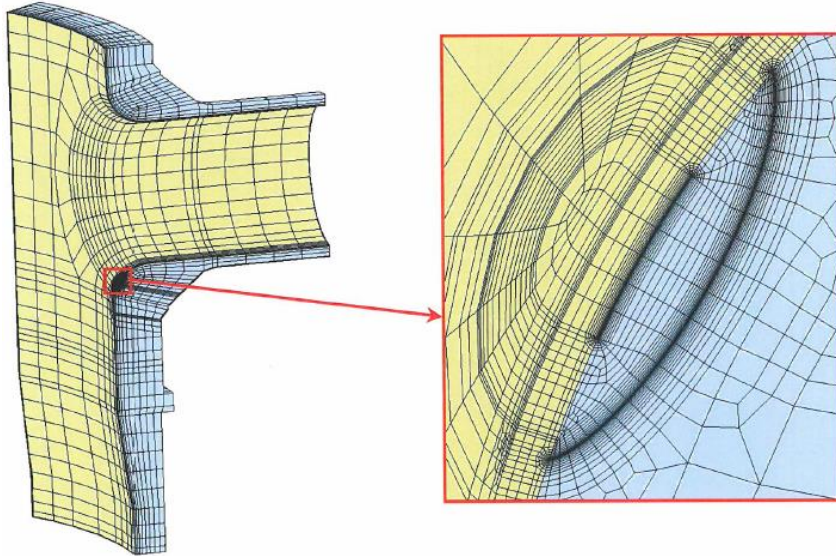

Fig. 3. The longitudinal sub-cladding crack of Du 850 nozzle

Besides, we took into account the residual stresses (welding overlay repair) [3, 4]. Residual stress distribution for at the RPV elements reviewed above is shown in figures 4-6. The origin of the horizontal axis is a surface of overlay welding.
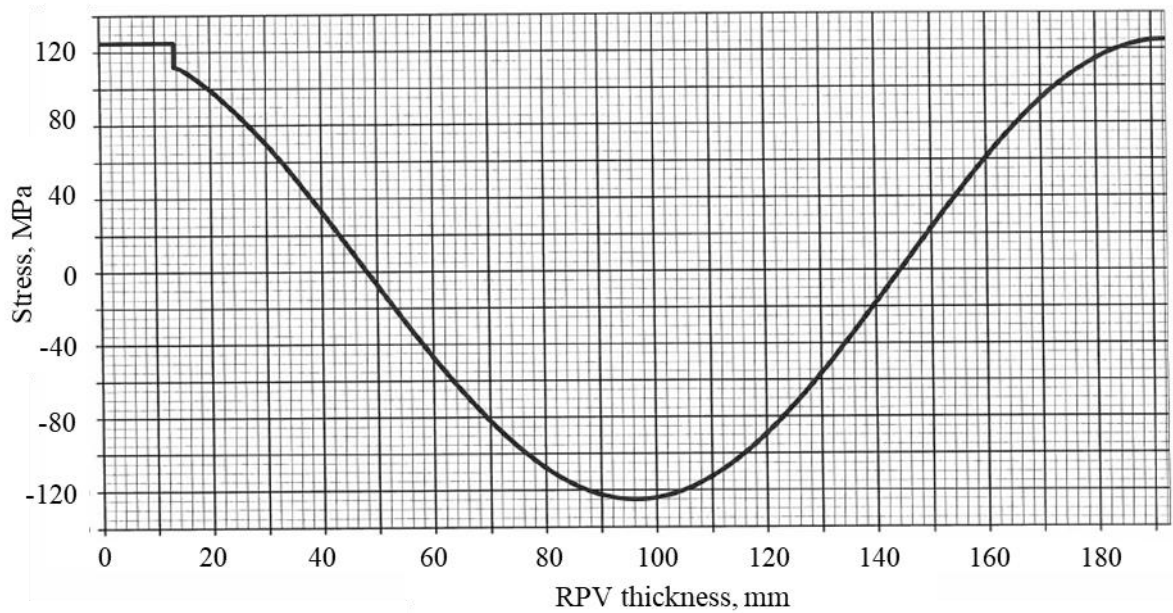

Fig. 4. Longitudinal and circumferential residual stress distribution in the metal of welded junctions №3 and №4
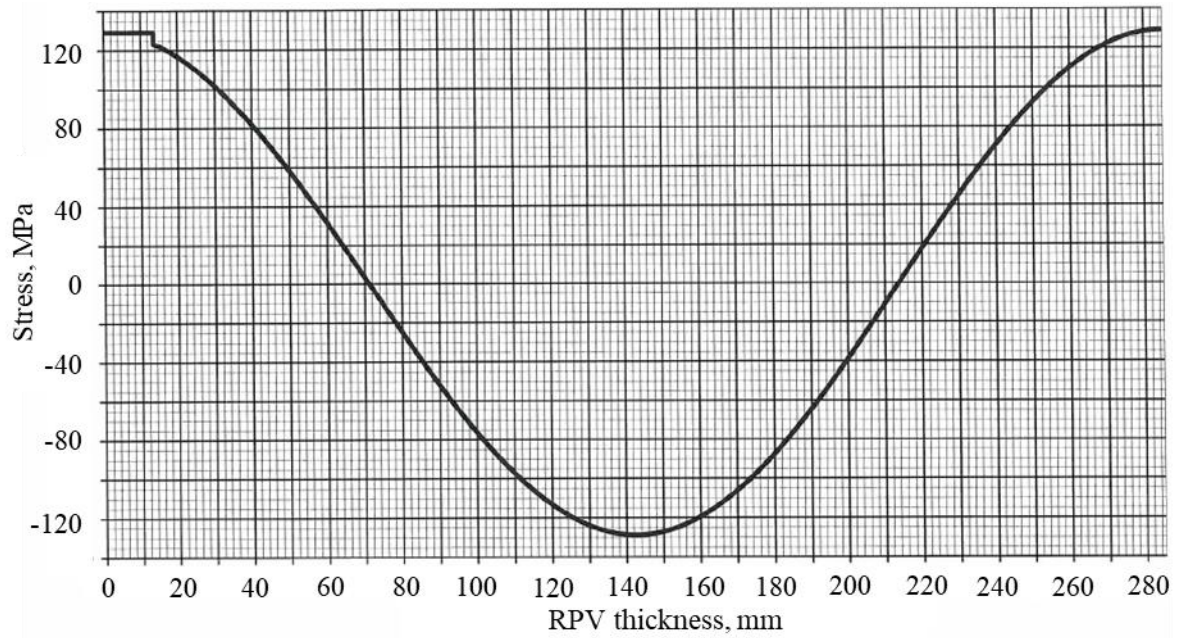
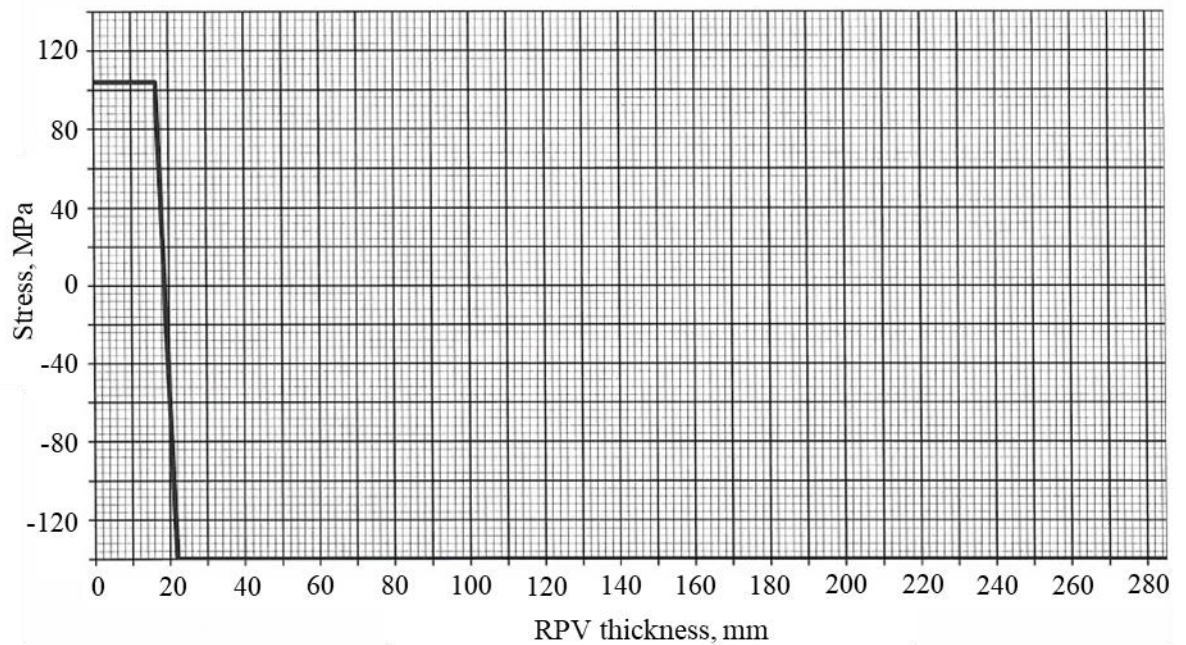Fig. 5. Longitudinal and circumferential residual stress distribution in the metal of welded junction №5

Fig. 6. Longitudinal and circumferential residual stress distribution in the main metal of nozzles

3. Conclusion

Maximal depth of calculated crack for welded junctions №3 and №4 is 14.1 mm. Maximal length of the crack for these junctions is 84.6 mm. Maximal depth of the crack for welded junction №5 is 15.7 mm and its maximal length is 94.2 mm. In the area of main metal, where neutron fluence is maximal, the values of maximal depth and length of crack are 16.3 mm and 97.88, respectively. Calculated parameters do not exceed the limit of 10 %.

As a result of calculations of the pressure permitted for normal conditions, there was found a dependence of fracture viscosity for irradiated part of RPV at the end of extended lifespan of 60 years. Moreover, the conditions of stress intensity coefficient were determined for the overlaying welded metal. The values of these coefficients for irradiated and non-irradiated parts of RPV could be calculated as follows:

$$\left(K_I \right)_1 = n_j \cdot K_I = 1.1 \cdot 54.2 = 60.0 \, \mathrm{MPa} \cdot \mathrm{m}^{1/2} < \left[K_{JC}^H \right],$$

$$\left(K_I \right)_1 = n_j \cdot K_I = 1.1 \cdot 88.5 = 97.4 \, \mathrm{MPa} \cdot \mathrm{m}^{1/2} < \left[K_{JC}^H \right].$$

These values do not exceed the viscosity damage threshold for overlaying weld [5].

According to our findings, it could be stated that equipment parts studied fulfill all the regulatory authorities' conditions for 30 years lifespan extension. The current method is feasible, because, in case of NPP operational period expiry, it will be required to provide a set of decommissioning works. Moreover, the power supply in region will be reduced and hence it will lead to a building of a new NPP or another electric power source. Therefore, current calculations are the optimum option to save energy and financial reserves.

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Materialovedcheskoye obespecheniye raschetnogo obosnovaniya prodleniya sroka ekspluatatsii korpusa i VKU reaktora energobloka № 2 Balakovskoy AES do 60 let. Etap 1.] Opredeleniye iskhodnogo urovnya svoystv osnovnogo metalla i metalla svarnykh shvov KR po pasportnym dannym, ostatochnykh napryazheniy po parametram tekhnologii sborki i svarki dannogo KR, a takzhe otsenka parametrov raschetnogo defekta s uchetom rezul'tatov zavodskogo, vkhodnogo i periodicheskogo nerazrushayushchego kontrolya KR energobloka № 2 Balakovskoy AES, FGUP TSNII KM «Prometey», 2013 (Inv. №320/5198 OKB «GIDROPRESS»).