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Role of in-service stress and strain fields on the hydrogen embrittlement of the pressure vessel constituent materials in a pressurized water reactor

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ABSTRACT

The in-service brittle fracture of a structural component of a nuclear power plant (NPP) is a problem of major concern in engineering. During nuclear energy generation, the wall of the nuclear reactor pressure vessel (RPV) is exposed to a hydrogenating environment leading to a fracture phenomenon known as hydrogen embrittlement (HE). This in-service failure is ruled by hydrogen diffusion from the hydrogenating source (the inner side of the RPV) towards certain places inside the vessel where hydrogen is accumulated and microstructural damage is located. The diffusion process is highly influenced by the stress and plastic strain distributions. For achieving a realistic estimation of the hydrogen accumulation by diffusion, both the in-service thermal stress and the manufacturing induced residual stress and strain (due to tempering heat treatment) must be taken into account. In this paper, a numerical analysis of the hydrogen diffusion assisted by stress and strain is carried out to find out the hydrogen accumulation within the wall of a real *pressurized water reactor* (PWR) for diverse heat treatment conditions. Results reveal the key role of the in-service thermal stress which enhances the hydrogen diffusion through the constituents' materials of a PWR pressure vessel.

Abbreviations: BWR, boiling water reactor; EAC, environmentally assisted cracking; HE, hydrogen embrittlement; LWR, light water reactor; NPP, nuclear power plant; NRPV, nuclear reactor pressure vessel; PWR, pressurized water reactor; RPV, reactor pressure vessel; VVER, Vodo-Vodyanoi Energetichesky Reaktor; WWER, water–water energetic reactor; 1D, one-dimensional; A, zone of wall corresponding to the cladding layer; B^+, zone of the base material layer with tensile residual stress; B^-, zone of the base material layer with compressive residual stress; C, hydrogen concentration; C_0, hydrogen concentration in a material free of stress and strain; D, hydrogen diffusion coefficient in the metal; D_A, hydrogen diffusion coefficient in the metal of stainless steel (layer A); D_B, hydrogen diffusion coefficient in the metal of low carbon steel (layer B); J, hydrogen flux; K_S, hydrogen solubility; $K_{S\varepsilon}$, hydrogen solubility component depending on plastic strain; r_0, inner radius; r, radial coordinate; R, molar gas constant; t_{temp}, tempering time; T_{temp}, tempering temperature; t_{serv}, in-service time; T, absolute temperature; V_H, partial molar volume of hydrogen; w, width of nuclear reactor pressure vessel wall; w_A, width of the 1st layer (cladding) made of stainless steel; w_B, width of the 2nd layer (base material) made of low carbon steel; z, axial coordinate; ε^P, cumulative plastic strain; ε^P_{ij}, components of the cumulative plastic strain tensor; ε^P_A, cumulative plastic strain at the cladding layer; ε^P_{B+}, cumulative plastic strain at the *tensile residual stress* zone of the base material layer; ε^P_θ, circumferential cumulative plastic strain; $\varepsilon^P_{\theta,A}$, circumferential cumulative plastic strain at the cladding layer; $\varepsilon^P_{\theta,B}$, circumferential cumulative plastic strain at the base material layer; $\varepsilon^P{}_z$, axial cumulative plastic strain; $\varepsilon^P{}_{z,A}$, axial cumulative plastic strain at the cladding layer; $\varepsilon^P{}_{z,B}$, axial cumulative plastic strain at the base material layer; σ, hydrostatic stress; σ_A, residual stress at the cladding layer; σ_{B+}, tensile residual stress at the base material layer; σ_{B-}, compressive residual stress at the base material layer; σ_θ, circumferential stress; $\sigma_{\theta,A}$, circumferential stress at the cladding layer; $\sigma_{\theta,B}$, circumferential stress at the base material layer; σ_r, radial stress; σ_z, axial stress; $\sigma_{z,A}$, axial stress at the cladding layer; $\sigma_{z,B}$, axial stress at the base material layer

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1. Introduction

Nowadays nuclear energy is generated in large amounts in nuclear power plants (NPP) all around the world [1]. Although this source of energy exhibits interesting advantages in relation to other energy sources [2], safety is a key issue since it generates mistrust in society in the matter of nuclear energy [3]. Consequently, the structural integrity assessment of the main components of a NPP is a key task in nuclear engineering. In a common *light water reactor* (LWR), energy is generated as a result of a fission chain reaction that is developed in nuclear fuel cans within a nuclear reactor pressure vessel (NRPV), commonly described as reactor pressure vessel (RPV) since such an acronym directly implies the relation with nuclear energy. Among LWRs, there are two types of fission nuclear reactors: (i) *boiling water reactor* (BWR) and (ii) *pressurized water reactor* (PWR). As a result of the nuclear fission reaction, a high amount of energy is liberated and a temperature increment is produced [4]. This energy is used for heating water inside the RPV that is pumped towards a steam turbine for transforming such an energy in electricity in the so-called primary circuit of the NPP.

The hot water of the primary circuit and the ionization resulting by the nuclear fission reaction act as hydrogenating sources, which can promote a harmful fracture phenomenon known as hydrogen embrittlement (HE) leading to catastrophic failure [5,6]. Thus, the hydrogen accumulated at the inner surface of the RPV is initially adsorbed and later the atomic hydrogen is absorbed and diffused within the material lattice towards certain places where hydrogen is accumulated up to reaching a critical concentration, leading to microstructural damage linked to HE. The hydrogen diffusion is highly influenced by stress and plastic strain fields, as shown in papers [7,8]. Thus, three factors help the HE phenomenon to occur: (i) a cathodic environment generating hydrogen, (ii) an adequate stress-strain state and (iii) a material susceptible to HE, cf. [7,8].

In a PWR all such factors converge. Firstly, the *cathodic environment* is produced as a consequence of the hydrogen generation due to both the hydrogen dissociation of the coolant water and the atomic hydrogen from irradiation due to chain nuclear fission reaction. Secondly, an in-service *stress state* exists caused by the *combined* action of thermal stress (as a result of the thermal gradient between the hot inner surface and the outer surface of the RPV) and the residual stress generated during material manufacturing, including a heat treatment of tempering whose aim is to remove residual stresses. Finally, the third factor (*material*) also appears since two layers of different steels (bi-material) with different susceptibilities to HE are used in a pressure vessel of a fission nuclear reactor. In a PWR, the geometry of a common vessel is a cylindrical body of large sizes (inner radius, r_0, about 1.77 m, and height about 14 m) ending by two half-spheres. Thus, in the outer side of the PWR pressure vessel, a common low carbon steel (e.g. A533B or A508) is used as *base material* (structural element) whereas a fine layer of stainless steel (316 L) is used as cladding material in the inner side exposed to the harsh environment. The main function of such a layer is to protect the base material, highly susceptible to environmentally assisted cracking (EAC) and HE [9–13], from the harsh environment. However, the effect of hydrogen is not fully removed since this layer is also affected by EAC and HE [14–19].

The aim of this paper is to analyse the effect of in-service stress and plastic strain states on the HE susceptibility of a pressure vessel widely used in one of the most important PWR designs, namely WWER-440, WWER (*water-water energetic reactor*) being the English name of the Russian model VVER (*Vodo-Vodyanoi Energetichesky Reaktor*). Since the vessel of such a model WWER-440 represents a general PWR pressure vessel, the conclusions of this study can be considered for the whole models of PWR. To achieve this goal, the model for hydrogen diffusion assisted by stress and strain previously developed [7,8] is applied taking into account the *global* (total) stress and plastic strain fields resulting from the *combined* action of (i) in-service thermal stress and (ii) residual stress-strain fields caused by manufacturing (considering diverse tempering treatment conditions). In this way, the distribution through the vessel wall width of the hydrogen concentration is revealed for different heat treatment conditions: *tempering time* (t_{temp}) and *tempering temperature* (T_{temp}). From these results, the best conditions to be considered in the tempering process of the constituent materials of a RPV wall can be estimated, so that the HE damage can be avoided.

2. Stress-strain state in a reactor pressure vessel

The in-service stress and strain fields undergone by a RPV wall (*global* stress and strains fields) are here estimated as the addition of the residual stresses generated during the manufacturing process (after tempering) [20] and the thermal stress state generated by the gradient of temperature that exists between the hot inner vessel face (in contact with the hydrogenating environment) and the cold outer vessel face (in contact with a conventional environment).

The analysis is carried out for a real cylindrical vessel of a PWR type WWER-440. The geometry of the RPV is obtained from the data given in [20]: inner radius, r_0 = 1.77 m; width of the stainless steel first layer, w_A = 8 mm (zone A in Fig. 1); width of low carbon steel second layer, w_B = 142 mm (zone B in Fig. 1).

In [20] the residual stress and plastic strain distributions in the radial direction of the RPV obtained after diverse tempering conditions are idealized in the way shown in Fig. 2a and Fig. 2b, respectively. According to this, the residual stress and plastic strain are uniformly distributed along the cladding width (w_A, zone A of stainless steel). With regard to the base material layer (second layer of low carbon steel of width, w_B), two intervals are considered. The first one of tensile nature residual stress (denoted as B^+ hereafter) is extended over a zone of width two times the cladding layer thickness ($2w_A$). The second one of compressive nature residual stress (denoted as B^- hereafter) is extended over the remaining RPV wall (i.e., w-$3w_A$, w being the whole RPV width considering both layers, i.e., $w = w_A + w_B$).

According to the idealized distributions stated in [20], only three parameters are required for defining the residual stress and plastic strain states at the cladding layer: width of cladding layer (w_A), residual stress at cladding layer (σ_A) and plastic strain at cladding layer (ε_A^P). On the other hand, four parameters are needed for the base material layer, namely: width of base material layer (w_B), tensile residual stress at second layer ($\sigma_B{}^+$), compressive residual stress at second layer ($\sigma_B -$) and plastic strain at the tensile