Procedure for Validating the Life Extension for the WWER-440 Internals at NVNPP Unit 3

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The design lifetime (30 years) of the 1st generation WWER-440 reactor facilities is nearing completion in Russia. One of the major problems is to validate the life extension (LE) of the reactor internals ensuring the core arrangement and free passage of the control and protection system components during different operating modes, emergency modes included. The internals at the 1st generation units are designed so that to enable their replacement. But it requires a lot of funds and time.

The work has been done to demonstrate that the internals may be further safely operated without their components being replaced provided their strength, longevity and serviceability are sufficiently validated.

The complexity of the problem is explained by the fact that the WWER-440 RF internals were essentially designed without the necessary regulatory basis determining the design, manufacturing and operation. Undoubtedly, certain “rules of the game” were observed, largely those arising out of the regulatory technical documents for Kotlonadzor objects (pressure vessels and steam boilers) but the impact of the basic operating effects (neutron irradiation, water environment, residual stresses, vibration) were not properly taken into account.

The regulatory document for the nuclear reactor core support structures being part of the ASME Code [1] was published well after the development period of the 1st generation WWER-440 projects. However, the strength analysis methods for the core supports are presented in this document’s latest update without considering mechanical properties as the result of neutron radiation, water environment impact and vibration. According to the ASME Code the NPP owner (at its design stage) should provide development in all information required for the selection of materials, manufacturing technology and strength analyses for considering the structural materials degradation mechanisms during operation of equipment and pipelines.

The strength analyses for internals at LE largely have analogy to Regulations for Strength Analysis of Nuclear Power Plants [2]. Changes to the properties of the internals material caused by operating factors at LE (08Kh18N10T steel and its welded joints) were considered in terms of their impact on the procedures of respective strength analyses (static, cyclic, brittle strength) and the probability of damage (corrosion cracking) or malfunctions (radiation swelling) during operation.

The increase under irradiation of the yield strength $R_{P0.2}$ by the factor of 3 or 4 and the ultimate strength R^T_m with a fast damping accompanied by the hardening drop beyond the yield strength, the general elongation decrease and the uniform elongation decrease to very low values at not so considerable reduction in area [3], did not require any changes in the static strength analysis and validation of basic dimensions. Non-self-balanced static stresses from mechanical loads in components of the internals are 10-15% of allowable stresses for non-irradiated steel as the analysis of the structure at the minimum irradiation zone does not differ from generally accepted NPP analyses.

In intensive irradiation zones where the said changes of mechanical properties are realized in full, the elongation drop does not require special consideration at such low mechanical stresses in estimating the static strength when the margin factor for $R_{P0.2}$, R_m of the irradiated steel raises to 30…60.

The strength of the core barrel flange and the wall of the core basket the elliptic and flat bottoms was checked from the action of a longitudinal force in all modes with regard for the load from a design-basis accident that was transformed into uniform pressure at the allowable stress equal to the initial yield strength $R^T_{P0,2}$.

The vault's central area is limited for radial displacements by the core basket, is externally strengthened by a shield that rests on threaded pins in the wall of the shield and the radiation hardening of the 08Kh18N10T steel raises the allowable stress at the critical stress to above the initial yield strength of $R^T_{P0,2}$.

The stability analysis for operating loads with regard of dynamic loads at a design-basis accident is performed with stresses limited by allowable values from the seismic resistance analysis according to Regulations [2].

Irregular radiation swelling may cause a growth in the self-balanced stresses and a distortion of the geometry and changes in the component dimensions which may result in a reduction of the coolant flow areas.

The intensity of swelling is growing with the temperature increase. The temperature at the swelling initiation and the incubation period dose at a low radiation damage rate is decreasing.

The overwhelming majority of the available experimental data relate to the area of temperatures exceeding the internals metal temperature in WWER-1000 (270-340 ^{o}C) and WWER-1000 (300-460 ^{o}C) or corresponds to the upper limit of the WWER-440 internals temperatures. Swelling data for irradiation conditions in PWR is practically absent. Swelling estimates for the damage dose of up to 50 dpa (~ $6{\cdot}10^{22}$ cm^{-2}, E>0.1 MeV) reached on the WWER-440 partition for the 30 year of operation, based on different models and experiments, produce the values of 0.01 – 1.03% and it is the estimate based on a theoretical model [4] only that falls beyond this range - 13%.

The absence of any complications during operation with fuel loading and unloading, changes to the dimension and shapes of an internal component does not confirm the pessimism of the outsider estimate (13%) in the range of optimistic estimates for the realized intensity of the radiation swelling and creep effect growing in time in the conditions of WWER-440 and foreign PWRs.

The problem remains urgent, especially for WWER-1000, so the development of the model and its experimental confirmation in the PWR irradiation conditions (dose growth rate, temperature) is one of the priority tasks.

It is evident that it is possible to get reliable values of the internal component stresses and strains during swelling for the 08Kh18N10T steel in the PWR conditions based on the verified dose and temperature swelling model and a 3-D model of the temperature and physical fields for the internals. Low swelling estimates for the WWER-440 internal components make it possible to classify this problem for the WWER-440 internals life extension as a non-critical problem.

High (residual welding and irregular swelling) stresses, changes in the concentration of alloying elements (depletion by Cr) and impurities in the grain boundary region of the 08Kh18N10T steel under irradiation in contact with high-parameter water may cause an irradiation-assisted stress corrosion cracking (IASCC) as the threshold concentration of oxygen exceeds.

The status of research into this problem allows to indicate the estimated threshold damage dose for the PWR environment from which IASCC may obtain (at 340 °C, $2 \cdot 10^{21}$ cm^{-2}, E > 1 MeV [5]).

There are contradictory opinions as to the threshold tensile stresses for the IASCC initiation. They range from the sufficiency of low stresses to the necessity of stresses equal to the irradiated steel yield strength. The compensating measure may consist in improved attention to the structure areas where the threshold damage dose exceeds during inspection.

In case of radiation hardening swelling of the 08Kh18N10T steel leads to an increased asymmetry of the cyclic loading caused by vibrations and pressure and temperature changes; an asymmetry is compensated partially, by radiation creep. The results of vibration changes during commissioning of the 1st generation WWER-440 units, estimates of variable stresses from pressure and temperature effects are an evidence of a low cyclic loading of the internal components on amplitudes of stresses. Stability of vibration stresses may be monitored during operation by diagnosis systems. It enables the use of conservative estimates of the equivalent cyclic loading and the accumulated fatigue damage largely using at that curves of fatigue in the high-cycle region.

The vibration loading is characterized by the maximum root-mean-square amplitude and the maximum frequency of the vibration spectrum as the allowable numbers of cycles are determined according to the design fatigue curve for the 08Kh18N10T steel with the effect of the cycle asymmetry, irradiation and low-oxygen high-parameter PWR water conservatively taken into account. The positive result in estimating the cyclic strength of the WWER-440 internal components permits the assumed conservatism that is justified by the insufficiently studied fatigue in the gigacyclic region with a high asymmetry, corrosion effect and irradiation [6, 7].

The 304 steel fatigue curves according to the curve equation according to Regulations [2] based on the fatigue damage criterion with regard for the initial properties (R_m^T, Z^T) [9] and the properties after irradiation ($1.5 \ldots 1.9 \cdot 10^{22}$ cm^{-2}, E>0.1 MeV, irradiation at 400-450 °C [10]) is more conservative of the experimental data at the symmetric strain cycle (325°C, air, $5 \ldots 10 \cdot 10^{21}$ cm^{-2}, E>0,1 MeV) [7]. The contact with the hydrated PWR water (O_2<5ppb) at 0.001…0.4% s^{-1} reduces the longevity as compared to loading in the air.

The appearance of the fatigue curve for the 08Kh18N10T steel in air accepted for the analysis is determined conservatively just by the dependence of the elastic strain with the steel properties in the initial state and with regard for the maximum impact of the cycle asymmetry [11].

$$\sigma_{aF} = \frac{R_c^T - R_{P0,2}^T}{(4N)^{m_e} - 1} \quad (1)$$

where σ_{aF}, N are the quasi-elastic stress amplitude and the number of cycles respectively,

R_c^T is the true ultimate stress, $R_c^T = R_m^T(1+0.014 Z^T)$,

R_m^T, Z^T is the ultimate strength and reduction in area at tension respectively,

m_e=0.0525+0.132lg(1+0.014Z^T),

R_{-1}^T is the fatigue limit (R_{-1}^T=0.4R_m^T).

It has been assumed that $R_{P0,2}^T$=0.7R_m^T for consideration of the $R_{P0,2}^T / R_m^T$ increase during irradiation.

The type of the fatigue curve at the loading in water is determined by equation (1) and the following equation

$$\sigma_{aF} = \frac{E^T \cdot e_C^T}{(4F_{en}N)^{0.5}} + E^T \cdot e_{ath} \quad (2)$$

where E^T is the elasticity modulus,

F_{en} is the reduction coefficient of the number of cycles N (N_{air}/N_{water}),

e_c is the plasticity characteristic e_c=1.15lg100/(100- Z_C^T),

$Z_C^T = Z^T$ at $Z^T \leq 50\%$, $Z_C^T = 50\%$ at $Z^T > 50\%$ [2].

The threshold strain for austenitic corrosion-resistant steel has been assumed in accordance with the ANL statistical model e_{ath}=0.126% [12].

The F_{en} value depends on temperature, oxygen concentration (at O_2<50ppb) and strain rate [8]

$$F_{en} = 2.5 / exp(T^* \cdot \dot{e}^* \cdot 0^*) \quad (3)$$

There have been assumed the values T*=1 (T>220°C), $\dot{e}^*$=ln10^{-3}, 0^*=0.26 (O_2<50ppb) ensuring conservative estimates at T<220°C, $\dot{e}$>4·10^{-4}% s^{-1}, O_2≥50ppb.

The design fatigue curve equation system looks as follows:

$$[\sigma_{aF}] = \frac{R_m^T(0.3 + 0.014 \cdot Z^T)}{n_\sigma\left[(4[N])^{m_e} - 1\right]}$$

$$[\sigma_{aF}] = \frac{E^T \cdot e_c^T}{(4n_N \cdot F_{en} \cdot [N])^{0.5}} + E^T \cdot e_{ath} / n_\sigma \qquad (4)$$

where $[\sigma_{aF}]$, [N] are the allowable values of σ_{aF}, N,
n_σ, n_N are the margin factors for σ_{aF} and N, 2 and 10 respectively.

The design fatigue curve (T≤350°C, Z^T≤40%, R_m^T≤353 MPa, E^T≤175 GPa, O_2<50ppb, $\dot{e}\leq 4\cdot 10^{-4}\%\ s^{-1}$, n_σ=2, n_N=10) for the 08Kh18N10T steel with regard for the maximum impact of the loading asymmetry, the PWR water environment and radiation is presented in Fig.1. The curve is conservative at $[N]>10^6$ as the value of m_e determined by R_c^T and R_{-1}^T is adopted in this region. The admissibility of the projected cyclic load with regard for vibrations is determined by the accumulated fatigue damage reserve of a_{Nr} determined through the linear summation of 1_{Nr}=1-a_N where a_N is the fatigue damage during the operation time also with regard of vibrations determined by the recorded loading history or an expert estimate of the number of different transients.

At low operating stresses from weight and pressure drops, critical cracks in the internals support components (core basket, core support plate) turn out to be large enough even at a low fracture toughness for the 08Kh18N10T steel and welded joints for the reached exposure doses exceeding the threshold for the loss of plasticity.

Brittle strength analysis is reduced to determining the allowable flaw sizes based on the conservative generally accepted fracture toughness (50 $MPa\cdot m^{1/2}$ [8]) and residual flaw opening as the criterion of the instrumentation resolution during operation.

The absence of pragmatic engineering results in the area of simulating the vibration wear of the internals excludes the problem out of the category of problems solvable by way of analysis. Visual wear recording during periodic inspections of the internals determines the required operative designs for its exclusion.

The above approaches to considering the operating factors causing degradation of the design properties of the internals material in validation of the WWER-440 internals LE have been reflected in methodical instructions worked out at ENES MAE and approved by Gosatomnadzor of Russia [13].

Negative temporal effects of the PWR internals metal degradation mechanisms may be mitigated or excluded if required based on their analysis by rational design procedures and selection of the material as generalized by respective rules and regulations for the internals.

The urgency of their development is evident.

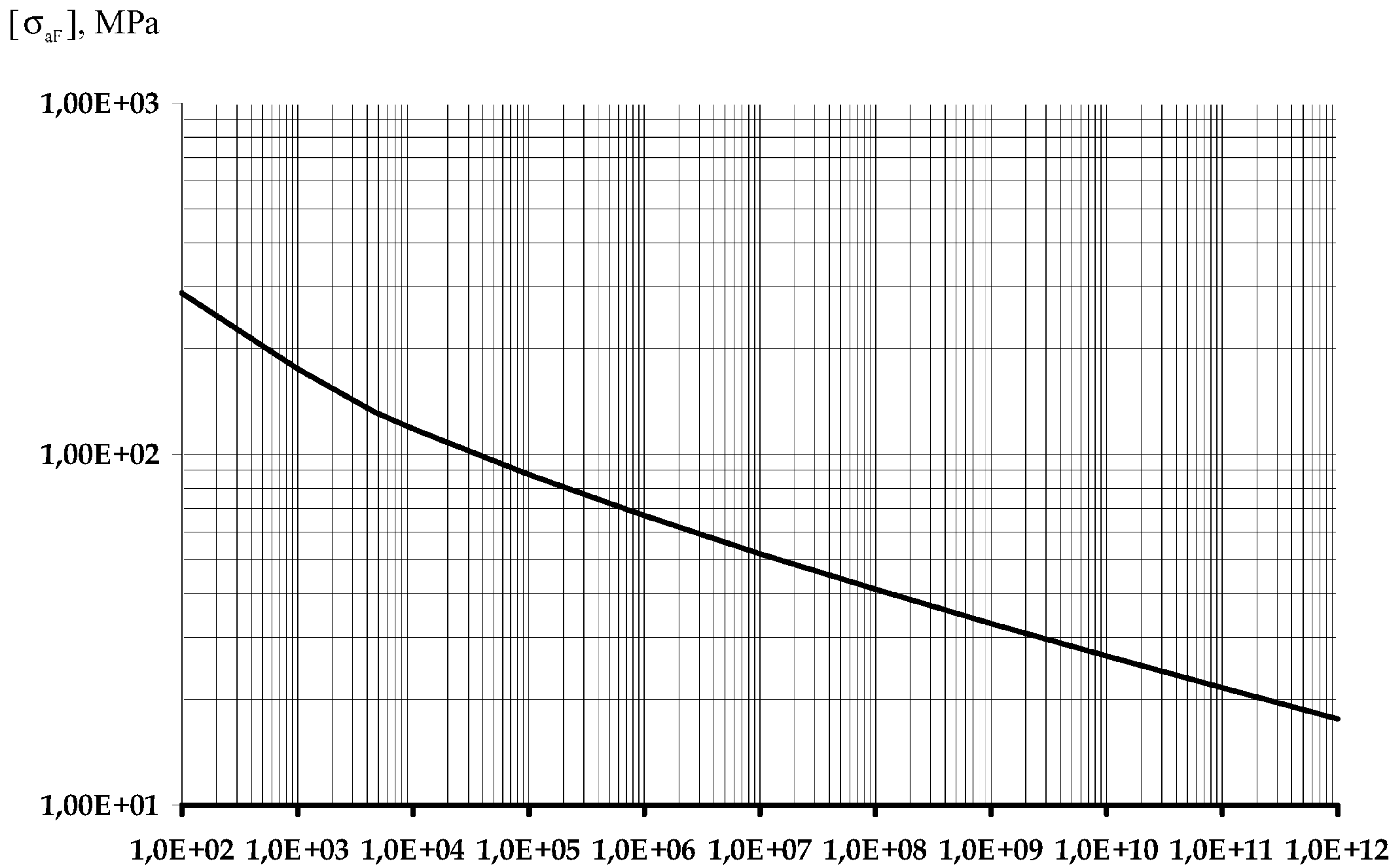

Fig.1. Design fatigue curve for the NVNPP power unit 3 WWER-440 internal components

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