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Lifetime Analysis of WWER Reactor Pressure Vessel Internals Concerning Material Degradation

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Расчет долговечности внутренних элементов сосудов давления реакторов типа ВВЭР с учетом деградации материала

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Внутренние элементы реакторов подвергаются воздействию трех основных эксплуатационных факторов: нейтронного и гамма-излучений; статических и динамических механических напряжений и химических веществ, используемых для охлаждения реактора. Исследовано влияние этих факторов на расчетную долговечность внутренних элементов реакторов ВВЭР 440 с целью продления срока их эксплуатации.

Ключевые слова: внутренние элементы реакторов, старение, радиационное охрупчивание, коррозионное растрескивание, расчет долговечности.

Background. The internal structures of water reactors play an important role as they contain the core of the reactor, channel the water flow inside the vessel, support and guide the instrumentation necessary for controlling and monitoring the reactor.

Long-term ageing at WWER operating temperatures results in changes in microstructure and phase composition, and, hence, mechanical properties of structural steels for WWER. Design basis for reactor internals in WWER has been calculated with no account of ageing of these structures due to that time unknown mechanisms of degradation.

This study aims to extend the operational lifetime of WWER 440 type units. Technical margins, the inherent safety and the realized safety improvement measures provide an opportunity for the extension of operational lifetime of nuclear power plant.

Namely, the purpose of this task is to perform evaluative analyses of presumable changes in material properties of WWER 440 reactor vessel internals (RVI). In accordance with these, the study address the issue of availability of degradation mechanisms which limit the lifetime of RVI structural elements.

Nowadays, it is a very important research area, but the available literature contain scarce information on failures of WWER RVI. Within this framework, we consider the following:

(i) systematization of structural plans and basic data required for the analysis, determination of initial data for analyses;

(ii) investigation of the presumable changes in material properties of the RVI;

(iii) perform strength and fatigue analyses of critical structural elements of RVI;

(iv) estimate the technically possible operation time of the structural elements considering the operation load of the 50 and 60 years operation time.

Basically, the strength and fatigue analyses should be performed in accordance with the provisions of the ASME (ASME III (2001)). Calculation methodologies for these structures include strength analysis, fatigue analysis and thermal stratification analysis for systems and components.

Structural Elements. **Boundaries of Analyses**. The subjects of this study are the WWER 440 reactor vessel internals, namely the guide tubes, core barrel, core basket and lower core barrel structures (Fig. 1.). Analyzed boundaries are the edge boundaries of the WWER 440 RVI and its nonwelded joints.

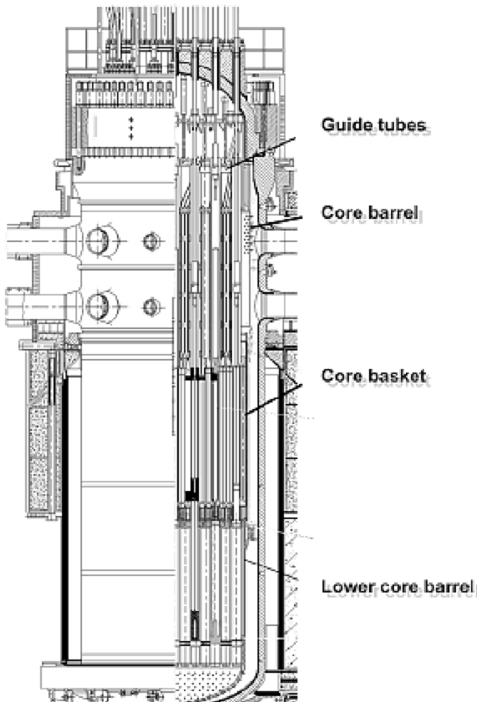

Fig. 1. Scheme of WWER 440 reactor pressure vessels and internals.

Potential Degradation, Material Models and Evaluations. In the following, we summarize the possible ageing mechanisms of WWER 440 RVI (Fig. 2.). Evaluation of these mechanisms is based on service experience, pertinent laboratory data and relevant experience from other industries.

During the evaluation we examine the following: effect of changes of the material characteristics not only on the strength and fatigue analysis, but also on the allowable crack size and crack propagation rates, in case of operation or breakdown.

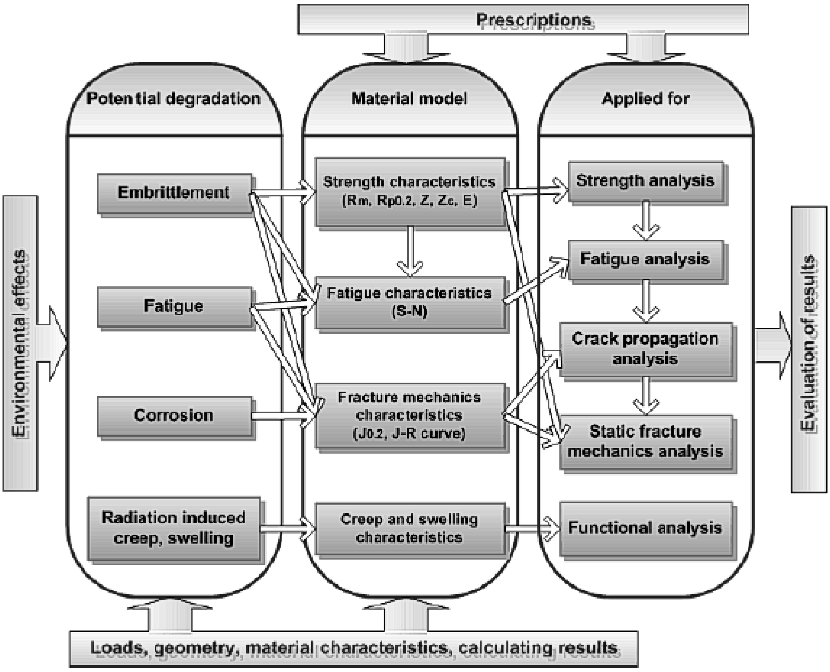

Fig. 2. Evaluation methodology.

The austenitic steels and the base material of the reactor internals are characterized by the ductile fracture mode. At the same time, the effect of irradiation induces the embrittlement mechanisms in the material, which can lead to deterioration of the material toughness. These embrittlement mechanisms can decrease the Charpy test values and the *J–R* curve (and $J_{0.2}$). The *J–R* curve can be applied (in addition to the strength material characteristics) to investigation of crack stability or stable crack propagation. For evaluation of the crack propagation stability two parameters of failure assessment diagram (FAD) are applicable.

Loads and Material Characteristics. Within the framework of the study, the analysis was made by means of the load catalogue (above 100 and 108% power values), which identify the loads, determine load curves and load cycle numbers for cyclic loads expected during the planned extended lifetime.

The main structural material used for WWER 440 for RVI is titanium stabilized austenitic stainless steel 08Ch18N10T (equivalent to A-321 steel); minor parts were made from nickel alloy ChN35VT-VD (this material is used for studs and is tungsten-alloyed) and steel 14Ch17N2. The weld metals Sv08Ch19N10G2B and Sv04Ch19N11M3 were used.

Static Strength Material Characteristics and Crack Growth under Static Load. The RVI are susceptible to several degradation mechanisms in course of reactor operation, the most serious problem being the change of material properties and *J–R* curve due to neutron irradiation. By the requested data to characterize the operational change of material properties are meant the material changes due to irradiation. To perform the calculations the material characteristics and *J–R* curve should be determined concerning 50 and 60 years relevant to the expected maximum fluence.

Material Characteristics Concerning Resistance of the Fatigue Crack Propagation. According to the ASME Code XI:2001 C 3210-1 and other experimental results of the investigation, which were performed in inner atmosphere (in cooling water) of the WWER 440 and on laboratory air, the ratio of the velocity of the crack propagation was around 2.5. Making use of this constant ratio, we determined the curve of the crack propagation concerning the heat-transfer media.

Material Characteristics Concerning Resistance of the Stress Corrosion Crack Propagation. After longer operation, the appearance of stress corrosion cracking cannot be excluded in the reactor vessel internals. This crack propagation is function of the temperature, the media and other outer effects, like the neutron irradiation. To investigate the possibility of the crack propagation, the threshold of the K_{ISCC} stress intensity factor (SIF) is required.

Functional Temperature and Fluence Information. The nominal inlet and outlet temperatures in the reactor vessel are, respectively: 267/296°C for 100% power and 267/299°C for 108% power.

Former research results of neutron physics calculations concern the position of the reactor vessel wall. The fluence of the test sample estimated for 60 years practically equals to the load of the outer surface of the core barrel, and it is ~ 12 dpa. On the basis of study [1, 2] it can be proved that the fluence value of the core boundary can be assessed as that of the core barrel multiplied by ~ 5–10 dpa, namely it can be estimated as 35–60 dpa for 60 years operation time.

According to the NRI study [3], which concerns fracture analysis of the basket of the Greifswald Unit 1, the fluence value of the basket is the following. The calculated maximum dose for the baffle component of Greifswald Unit 1 internals after 15th operation cycle, according to the first calculation method is $1.2 \cdot 10^{26}$ m^{-2} ($E > 0.5$ MeV), i.e., 11.4 dpa [4], and according to the second method is $1.3 \cdot 10^{26}$ m^{-2} ($E > 1.0$), i.e., 19.4 dpa [5].

In the analysis, the first step is to define the maximum neutron fluence values of the critical place of the RVI after 50 and 60 years' operation time. The resulting material characteristics relevant to the maximum fluence will be used in the calculations.

The KFKI Atomic Energy Research Institute has performed calculations concerning the critical environments of the structural elements of RVI in order to determine the expected maximum fluence value. The analyses were made for 30, 50, and 60 years' operation time periods. The irradiation dose was defined in dpa units (displacement per atom), and Table 1 shows the results of the measurements.

Based on our estimations, the results show 20% decrease in the bolt head and 40% decline in the shank of the bolt in radial direction. The estimated irradiation value at the bolt neck is about 85 dpa.

Available Material Data and Material Characteristics Concerning 50 and 60 Years' Operation Time. The base metals (A304, A304L, A316, A316L, A316Ti) of RVI of Western type reactors are nonstabilized and/or nontitanium stabilized austenitic steels, and cannot be considered equivalent to the materials used in WWER type reactors. On the basis of the previous overview of the available analytical and research results [2, 3], the danger cannot be excluded that higher neutron fluence can cause the embrittlement which changes the mechanism

Table 1

The Irradiation Dose in the Site with Maximum Dose

Operation time [year]	Irradiation [dpa]				
	Bolt head, baffle	Shank of the bolt	Core basket	Lower core barrel	Core plate 2–3
30	42	36	34	14	36
50	75	61	58	24	61
60	91	73	70	29	73

of the austenitic steel fracture and invokes of the intergranular type fracture mechanism in the material. To avoid appearance of these particular examination results, material characteristics are required for high fluence values.

Available examination results regarding RVI of NRI Rez and Prometey Institute for 50–60 dpa are the following:

Tensile test → $\sigma - \varepsilon$, YS, UTS, A, RA, and fracture mode.

Instrumented impact → KV and KCV.

Fracture resistance → $J_{0.2}$, $J - R$, and fracture mode.

During calculations we have defined the material characteristics of the core barrel, core basket, shank of the bolt, bolt head and core plate concerning 50 and 60 years operation time relevant to the expected maximum fluence (based on the NRI, Prometey measurement and other data available in the literature).

In accordance with the above-mentioned notions, we have determined:

a) *Strength characteristics* (*YS*, *UTS*, and *RA*) as a function of the size of irradiation. These material characteristics show an increase (except *RA*, which shows a decrease) until these trends turn into saturation (after 60 dpa).

b) *Fatigue curves* based on strength characteristics. Our investigations show that the calculated fatigue curves run above the one constructed according to the PNAÉ [5]. Thus, the earlier fatigue analyses will be valid for the 50–60 years operation time. In the following, we refer only to the curve corresponding to the 60 years' operation time according to $r = 0$ (see Fig. 3).

c) We have also determined the J_c *static crack growth resistance* and the K_{Jc} *stress intensity factor* as a function of the irradiation scale. For example, for 30 dpa $J_c = 15\ \text{kJ/m}^2$ and $K_{Jc} = 51\ \text{MPa}\sqrt{\text{m}}$ (for the base metal), and kinetics of both material characteristics manifests saturation over 30–35 dpa (both in case of the base metal and the weld material).

d) We have defined the *swelling characteristics* of the materials under study after 50–60 years' operation time. The swelling level for 30 dpa in the course of the lower core barrel is 0.3%, in the core basket for 57 dpa is 1.1% , while the swelling level in the bolt head for 75 dpa is 1.8%.

In case of the material characteristics (K_{ISCC}, *da*/*dt*–*K*) concerning crack propagation and crack initiation, these are affected by the environmental effect: determination of these material characteristics are based on particular measurement results. However, there are no measurement data in the literature. Therefore we set out from the earlier conservative approach, where $3 \cdot 10^{-11}$ m/s value for the crack propagation rate is used, irrelevant of the load value.

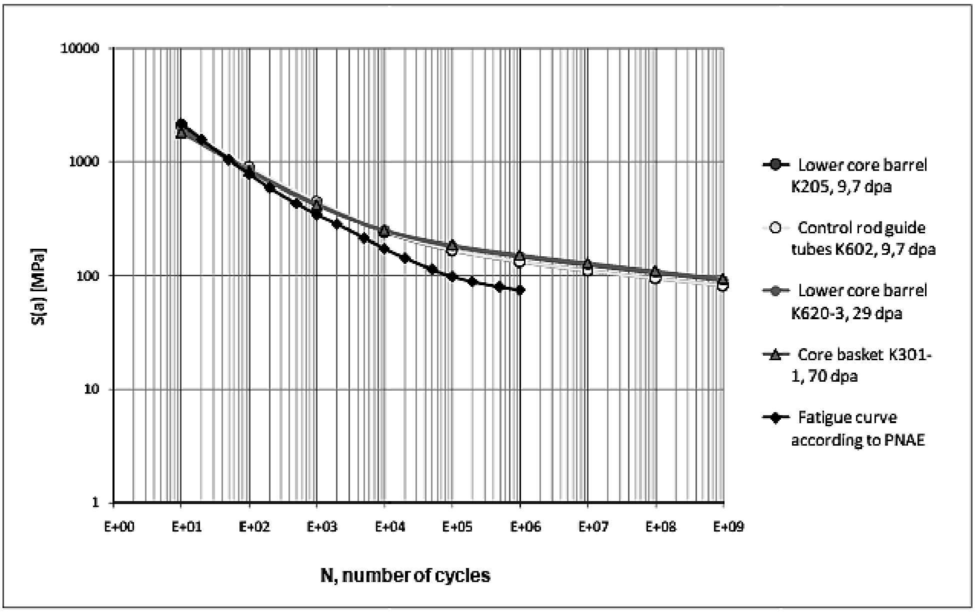

Fig. 3. Fatigue curves corresponding to 60 years' operation time ($r = 0$).

Previous Assessment of the Presumably Alternative Mechanisms of the RVI Material Properties. The degradation of the material characteristics and the degradation of the components are the result of complex environmental effects. The hardening and embrittlement, fatigue and environmental effects (corrosion cracking and propagation) appear as the effect of the irradiation, those are degradation mechanisms, and analysis of these mechanisms is urgently required.

Boundary Analysis. During the analysis we have selected at least one most critical environment, in case of any structural elements, for which we have performed the fracture mechanics calculations. If the cross section did not coincide with height of the zone where the largest irradiation was expected, we would select the second critical cross section of the structural element with the height of the zone.

Necessary Calculations – Analysis of Stable Crack Propagation. As we have already mentioned, the effect of irradiation induces the embrittlement mechanisms in the material, which can lead to deterioration of the material toughness. These mechanisms can decrease the Charpy specimens' test values and affect the *J–R* curve (and $J_{0.2}$). The *J–R* curve can be applied (beside the strength material characteristics) to the investigation of the stableness of the crack or the stable crack propagation. To evaluate the stability of the crack propagation two parameters FAD are applicable.

While the RVI are not pressure elements, it is practical to assume that the material contains a through crack. The ASME standard contains no information on the calculation of through cracks, therefore we have applied the FITNET European Union Standard Collection in our calculations.

The basis of the determination of the allowable crack length was the stress intensity factor K_{I}. We have determined the stress intensity factor supposing a different crack length for every single dangerous environment. Using this information we were able to define the crack size which is tolerable in the structure under 60 years operation. In possession of the FAD, calculation of the allowable crack size can be performed for the critical environment of the structural element. Table 2 contains the allowable crack lengths for some elements.

T a b l e 2

The Allowable Crack Sizes in Some Critical Environments

Environment	T, °C	Irradiation [dpa]	Operation condition	Allowable crack size [mm]
Lower core barrel K205	295	9.7	tv10*	544
Control rod guide tubes K602	262	9.7	tv10*	1280
Lower core barrel K620-3	275	29.0	tv10*	542
Core basket K301-1	332	70.0	NA500 heat circuit fracture*	82

* Final safety report / Load Catalogue Data.

The allowable crack sizes are still large enough to be visually detected. However, our examinations did not reveal such large cracks neither in the marked environments, nor anywhere else in the RVI components.

Necessary Calculations – Fatigue Crack Propagation Analyses. For obtaining data relative to fatigue crack propagation, particular measurement results are needed. Since the operating load cycles include neither significant load, nor large number of cycles, these do not cause fatigue crack propagation. The crack propagation can be resulted only from the vibration of the reactor internals, if the load vibration induces larger stress intensity factor than the threshold SIF of the fatigue crack propagation (ΔK_{th}). The vibration-caused number of cycles is so large that even small crack propagation rate can gradually provide a larger crack size. Therefore the allowable crack size has to be such that the vibration load would not cause crack propagation.

According to the calculations, the fatigue crack propagation preclude so large a crack that could imply stable or instable crack propagation, in case of zone basket and lower core plate. The currently known examinations did not reveal cracks in the dangerous cross section of the control rod guide tubes and lower core barrel.

Stress Corrosion Crack Propagation Analyses. After longer operation the reactor vessel internals cannot be excluded from the appearance of the stress corrosion cracking. We have performed stable crack propagation analysis of a crack appearing in fastener bolts. Since the ASME standard contains no information regarding calculation of the crack appearing in bolt, calculations were performed according to the FITNET standard, based on the BS 7910 procedure. The stress of the shank of a bolt equals to the allowable maximum tensile stress value (according to the PNAÉ standard [5]), which is half of the planned yield stress.

During calculations we have determined the SIF value, from which we have deduced the *J*-integral and fracture toughness of the material as a function of irradiation dose. We have also determined the $J_{0.2}$ value corresponding to fracture.

We have determined the time interval between the initial crack appearance and the moment when the fatigue-corrosion crack propagation reaches the section of the stable crack propagation. The operation time of the bolt significantly depends on the crack shape. When we consider the worst case, namely a circular crack, it takes approximately 3.5 years to reach the stable crack propagation phase in a RVI bolt (Fig. 4.). When we consider a straight crack front, the respective period is approximately 6.1 years (Fig. 4).

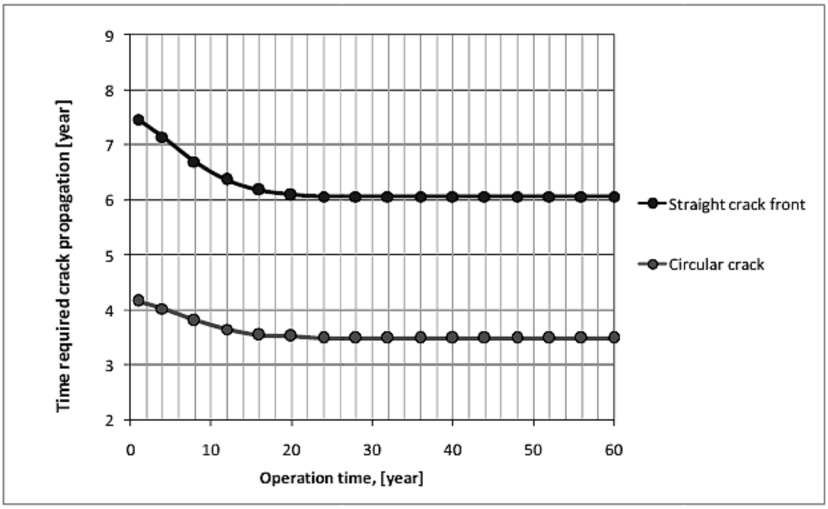

Fig. 4. Time required to reach an allowable stress corrosion crack size in a RVI component.

Swelling Effect. Swelling appears primarily in components exposed to high gamma irradiation. We have determined the level of swelling appearing in the basket wall, in order to demonstrate the distribution of the wall swelling. In the basket wall (at the half of the zone height), swelling along the wall can be described using the swelling relation given by the Prometey Institute [6]), assuming that the value of the irradiation can be decreased by 40% in the wall.

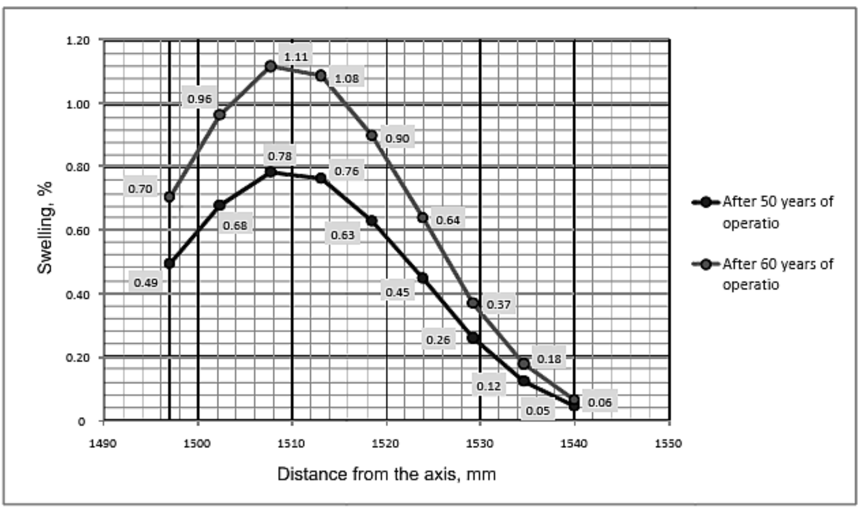

Fig. 5. Swelling effect in the basket wall after 50–60 years' operation.

Figure 5 shows the swelling distribution in the basket wall after 50–60 years' operation time. The swelling effect can induce the second principal stress (and thus, stress biaxiality) in the material.

Conclusions. We have examined the presumable changes in the RVI material properties and estimated the technically possible operation time of the structural elements considering the operation load of the 50 and 60 years' operation time.

The results show that the swelling and the irradiation-induced stress corrosion cracking are the most significant degradation mechanisms in the course of prolongation of the RVI operation time.

From the point of view of the swelling, the critical environments are: basket component at the middle part of the zone, baffle, fastener bolts, and cylinder of the core basket.

Based on measurement results and the presumable irradiation and temperature data, it cannot be excluded that 0.5–1.0% swelling will cause deformation and tightening in the fasteners bolts after 30 years' operation time. The deformation of the baffle can influence the installation, assemblage and joints as well. The appearance of abrasion places on the matt surfaces will be the first sign of this deformation.

From the point of view of the irradiation-induced stress corrosion cracking, the critical sites are the basket fastener bolts which can be characterized by fast irradiated neutron fluence and the stagnant environments. The swelling causes hindered expansion of the bolt, while possible stress corrosion cracking of the bolts can lead to fracture. During the examination, for lack of other information, we supposed the preset load according to the PNAÉ maximum, which is rather a conservative approach with a significant safety factor. For determination of the crack propagation rate for the selected environments further analysis is needed.

Р е з ю м е

Внутрішні елементи реакторів зазнають впливу трьох основних експлуатаційних чинників: нейтронного і гамма-випромінювання; статичних і динамічних напружень та хімічних речовин, що використовуються для охолодження реактора. Досліджено вплив цих чинників на розрахункову довговічність внутрішніх елементів реакторів ВВЕР 440 із метою подовження терміну їх експлуатації.

1. *Strength Analysis, Fatigue Analysis and Thermal Stratification Analysis for Systems, Structures, and Components (SSCs) – Reactor Pressure Vessel, Steam Generator, Pressurizer, Main Circulating Pump, Main Gate Valve, Pipelines, Vessels, Pumps, Heat Exchangers and Valves – Classified into the Safety Classes 1 and 2*, Methodology and Criteria Documentation.
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