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Materials Challenges of the New Nuclear Power Plant in Hungary

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Abstract

Having the experience accumulated during close to twenty thousand reactor years worldwide one may say that enough knowledge is available on the behavior of nuclear structural materials. In recent years however new questions emerge in close association with the long-term operation of nuclear power plants. The questions are in relation with both the operating and the future plants. Hungary operates four Russian designed VVER-440 nuclear units, and currently, two VVERs are under construction. The new, VVER-1200 type reactors represent the generation 3+ which is the latest and the safest version of the world's reactor fleet in operation. Service life of VVER-1200 reactors is 60 years, and operation beyond this term is foreseen. The structural materials of the main and usually non replaceable pressurized components such as reactor pressure vessel, steam generator and so on have to resist load and environment during the long operation period to ensure the components' structural integrity. It is right to say that the long-term, safe operation of the current and future reactors is ultimately governed by the performance of the structural materials in the mechanical technological systems. After introduction of the evolution process of the VVER structural materials performance the article reviews the major loading and environmental parameters of the operation and the ageing effects induced by the operation. Then the most important materials aspects and challenges are presented and discussed.

Keywords

structural material, long-term operation, reactor safety, corrosion, irradiation embrittlement

1 Introduction

Today, in association with a massive geopolitical restructuring, the world is facing an unprecedented energy shock and other overlapping crises characterized by, among others, these features:

- the European sanctions on imports of fossil fuels from Russia are severing one of the main arteries of the global energy trade,
- basically, all fuels are affected but gas markets are the epicenter (higher energy bills, supply shortages),
- the energy prices have reached levels never seen before,
- the crisis has generated inflation and recession,
- the climate policies and net zero commitments were blamed for high energy prices but the evidence shows that higher shares of renewables correlated with lower electricity prices.

Fig. 1 shows as example an extreme increase of the natural gas and the electricity price in Europe.

The possible solutions for Hungary in line with the country's energy strategy and in response to the energy crisis briefly described above can be summarized as follows. The parallel reliance on nuclear power generation and renewable energy is fundamental. The subsequent service life extension for the four units of Paks Nuclear Power Plant (lately called Paks NPP-1) is already under preparation and the construction of two new units (Paks NPP-2) at the same site is in progress. Addition of two more units into the nuclear fleet in Hungary in a later stage may also be possible. Furthermore, the small modular reactor (SMR) option as a current emerging technology might not be excluded in the future. Among renewables the photovoltaic technology plays the key role. Of course, expansion of energy storage, improvement of energy intensity and increase in energy savings complete the list.

Although nuclear power is heavily influenced by political ideologies, a sober voice has recently been shown

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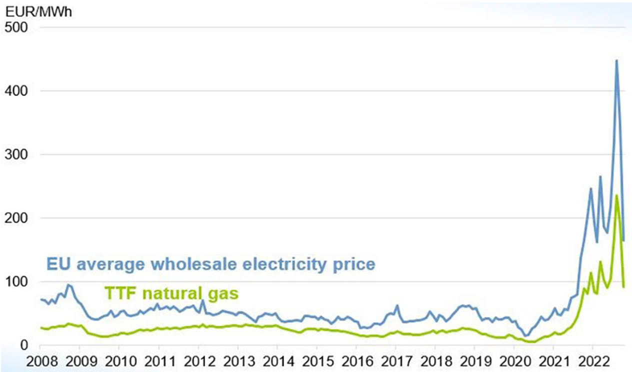

Fig. 1 Change of natural gas and electricity price in Europe [1]

by the recognition of its key role. More than 20 countries from four continents published a declaration in the World Climate Action Summit of the 28th Conference of the Parties to the U.N. Framework Convention on Climate Change in Dubai, UAE, at the end of 2023. The declaration includes working together to advance a goal of tripling nuclear energy generating capacity globally by 2050 and inviting shareholders of international financial institutions to encourage the inclusion of nuclear energy in energy lending policies [2]. Hungary is among the signatories.

Now the biggest task of Hungary's economy is the successful completion of the megaproject, i.e., construction of the two new Russian designed VVER-1200 reactors. Goal of this article is to present the structural materials aspects / challenges of this new nuclear power plant.

2 Role of structural materials

Components of the coolant system in light water reactors are basically made of either low-alloy (ferritic) or high-alloy (austenitic) steels. Cladding of reactor pressure vessel (RPV) and pressurizer inside surface is also austenitic stainless steel. For some parts (e.g., RPV penetrations) and for steam generator (SG) heat-exchanger tubes also nickel-based alloys are used. The combination of operational loading (high pressure) and environmental effects (high temperature, corrosive medium and fast neutron flux) creates an extraordinary harsh environment for the reactor structural materials.

Worldwide operating experience accumulated from 20,000 reactor years may suggest us that a comprehensive knowledge on behavior of the reactor materials exists. Notwithstanding intensive research is going on to better understand the service induced material degradation effects. The overall goal of these research activities is to ensure and improve the safety of operating and that of future nuclear reactors.

Another important driving force behind the research is to meet the major goals of the NPP life management such as power uprate and service life extension. Today, operation of NPPs until safety requirements can economically be ensured became a worldwide tendency, and generally expressed by the name long-term operation (LTO). In the USA, the plants' original 40 years operation license has been extended up to 60 years, then the process from 60 up to 80 years is now under way and, moreover, operation from 80 up to 100 years is under consideration. The units of Paks NPP-1 were originally licensed for 30 years. The commissioning of the units happened between 1982 and 1987. As a result of a systematic technical and a severe legal process this term was extended by 20 more years, and currently, the subsequent (i.e., second) extension process targeting an additional 20 operational years has started.

This means in both USA and Hungary (and, of course, in other nuclear operating countries) that the reactors are supposed to operate more than two times of the lifetimes originally foreseen by the designer. Consequently, the structural materials of the main (usually non replaceable) components have to resist load and environment during this long-term operation to ensure the components' structural integrity. Of course, beside mechanical components, concretes and cables are also critical but these are outside the scope of this article. It can clearly be stated that the long-term, safe operation of the current reactors is ultimately governed by the performance of their structural materials. This statement is, of course, also true for the future reactors.

3 The new reactors at Hungary

Two VVER-1200 model V-527 reactor units are under construction at Paks, Hungary. VVERs are Russian designed pressurized water reactors (PWRs) where the number refers to the nominal electric output of the unit.

3.1 VVER-1200 features

This type of reactor is one of the generation 3+ reactors in the world. The general contractor of VVER-1200 is the Atomstroyexport Engineering Company, part of Rosatom Concern, Russian Federation. The main components of the reactor coolant system are shown in Fig. 2.

The most important technological parameters of the reactors are presented in Table 1.

VVER technology went through on a long evolution process in the former Soviet Union and Russia. The last decades of this process included the world-wide features, i.e., the addition of technological improvements on one hand, and

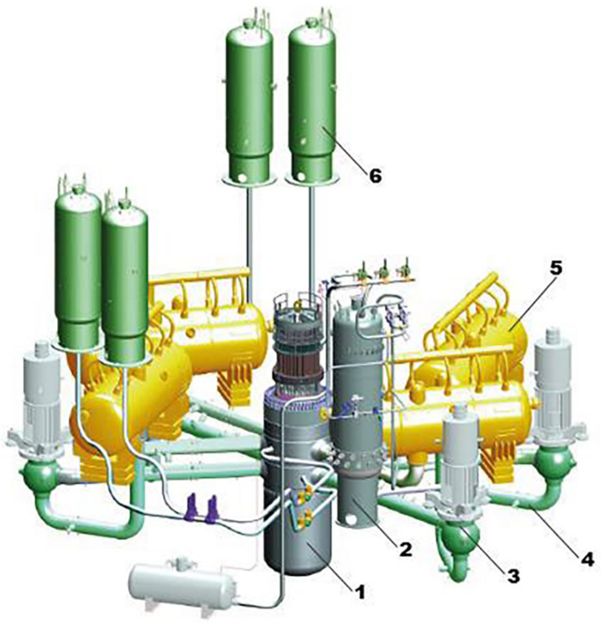

Fig. 2 Reactor coolant system of VVER-1200: 1 – RPV, 2 – pressurizer, 3 – reactor cooling pumps, 4 – reactor cooling loops, 5 – SGs, 6 – hydro-accumulators

Table 1 Main parameters of VVER-1200 reactors

Parameter, unit	Value
Reactor thermal power, MW	3200
Coolant flow rate through reactor, m^3/h	87200
Coolant outlet pressure (absolute), MPa	16.2
Coolant inlet temperature, °C	298.1
Coolant outlet temperature, °C	328.8
Number of loops	4
Max. operation time between refueling, h	14000
Quantity of fuel assemblies in core	163
Service life, year	60
Refueling periodicity, month	18
Availability factor (60 year average), %	92

of new safety technologies such as passive operational principles on the other. To ensure VVER-1200 RPV structural integrity up to 60 years some modifications in the RPV geometry compared to the predecessor VVER-1000 type reactor were implemented by the designer: (1) the vessel diameter was increased by 100 mm and (2) its length was increased by 300 mm. The first modification resulted a lower fast neutron fluence to the vessel wall in general, and the second one a lower fluence (by about 10%) on the welded joint of the supporting shell to nozzle ring and it improved the core cooling conditions in loss-of-coolant accident [3]. Cross section of the VVER-1200 RPV is shown in Fig. 3.

In case of earlier VVERs the corrosion resistant surfaces in the reactor coolant system were ensured either by use of austenitic stainless steel for primary piping,

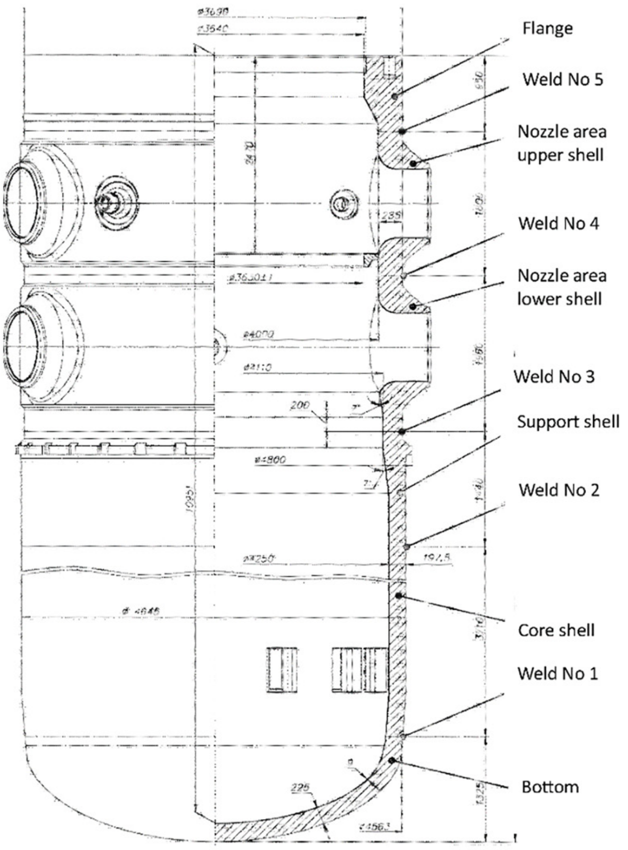

Fig. 3 VVER-1200 reactor pressure vessel

pumps and SG primary collectors, and by cladding of ferritic components such as RPV and pressurizer. In VVER-1200 the primary coolant system is uniformly made of ferritic steels: the RPV material is 15Cr2NiMoVA, a low alloy steel (C = 0.13 to 0.18%), and the pressurizer, the SG housing and collectors, the main pumps as well as the loops are made of 10MnNi2MoVA, also a low alloy steel (C = 0.08 to 0.12%). Consequently, all these ferritic components are cladded.

With this uniform material selection, the number of dissimilar metal welds could be minimized. In addition, this way the formerly used non-alloyed steel 22 K is completely eliminated from the reactor coolant circuit. This relatively cheap steel had shown uncertainty in stability of the long-term mechanical properties. Another basic endeavor is to decrease the overall number of welded joints, the result of which is clearly recognizable on the main circulating pipelines. These DN850 pipes are cladded still in their straight state and are bended only afterwards. Bends with longitudinal welds are not applied.

3.2 Evolution of structural materials performance

The structural materials of the VVER-1200 type reactor are basically the same as those of VVER-1000 type one or, in some cases, their improved versions. To ensure the NPP safety during the entire service life, if we take into account the most essential issue, that the brittle fracture of the RPV has to be excluded, consequently, a low ductile-to-brittle transition temperature (DBTT) should be provided. For this reason, the resistance against radiation damage of RPV material has been improved by development of three variations of 15Cr2NiMoVA steel. These variations mainly differ in their impurity and Ni-content (Table 2).

Out of the materials in Table 2, 15Cr2NiMoVA is used for the RPV flange and the bottom, 15Cr2NiMoVA-A is for both upper and lower shell of the nozzle area and 15Cr2NiMoVA class 1 is for the support shell and the core shell. For welded joints Nr 1 to 5 the wires either Sv-12Cr2Ni2MoAA or Sv-09CrMnNiMoTiAA-VI are used.

Currently, the leading nuclear safety rules and standards, including the world-wide used ASME Boiler and Pressure Vessel Code and the Russian PNAE or its recent successor NP series, say that RV structural integrity is ensured if the following inequation based on the principles of linear elastic fracture mechanics, is met:

$$nK_J < K_{Jc}, \tag{1}$$

where K_J is the stress intensity factor for a postulated flaw, K_{Jc} is the critical value of the stress intensity factor (fracture toughness), and n is a safety factor. One has to concentrate on brittle fracture because ductile fracture would require a larger amount of energy for crack growth.

RPV structural integrity is jeopardized during transient events, e.g., heating up, cooling down or, as most critical, activation of the emergency core cooling system (pressurized thermal shock, PTS [4]). Two safety levels can here be defined [5]:

- level-1 operational safety,
- level-2 operational safety.

In case of level-1 safety the RPV material can fracture only by ductile mode thus the vessel catastrophic failure is ruled out. In case of level-2 safety the RPV catastrophic failure during a PTS loading is impossible for $T_{PTS} \geq 100$ °C and for $T_{PTS} < 100$ °C, although crack initiation and unstable crack growth is not ruled out for $T_{PTS} < 100$ °C.

Considering 100 °C above where a high energy steam-water mixture is present, and taking a ~4 × 1023 m^{-2} fast neutron fluence at the end of the 60 years lifetime and an initial critical temperature of brittleness $T_{cr0} \leq -45$ °C into account, the T_{cr} will reach ~21 °C. This latter one exceeds the temperature necessary to level-1 safety thus the reactor steel 15Cr2NiMoVA class 1 cannot provide level-1 safety but it is suitable for level-2 safety, see explanation in Fig. 4.

Fig. 4 shows two $K_{Jc} - T$ curves (called reference curves); the initial one (left) and the shifted one after 60 years of operation (right); and the fracture toughness limit of ductile failure (K_{Jc} = 200 MPa m$^{1/2}$). The dark grey area shows the temperature window where brittle fracture is impossible, while the lighter grey one shows that in which brittle fracture is unlikely but possible. The material's considerable low initial critical temperature of brittleness was achieved by the metallurgical technology and the embrittlement behavior by very strong restrictions on P- and Cu-contents.

4 Materials challenges

For the discussion of nuclear materials aspects and challenges the well-known chain model of materials science and engineering may help, Fig. 5.

In case of NPP operation – at highest level – materials performance is determined by the most fundamental needs of the society (i.e., the users) such as:

- ensuring reliable energy supply,
- keeping nuclear safety at acceptable level while operating economically and efficiently,
- serving NPPs' long-term operation goals.

Table 2 Chemical composition of VVER-1200 RPV materials, in wt%

Material	C	Si	Mn	Cr	Ni	Mo	V	Cu	S	P	As	Co	Sn	Sb	P + Sn + Sb
								not more than							
15Cr2NiMoVA					1.0–1.5	0.5 0.7	0.10 0.12	0.30	0.005	0.006	0.020	0.03	–	–	–
15Cr2NiMoVA-A	0.13 0.18	0.17 0.37	0.30 0.60	1.8 2.3				0.10	0.005	0.005	0.010		0.005	0.005	–
15Cr2NiMoVA class 1					1.0–1.3			0.06	0.004	0.005	0.010				0.012
Sv-12Cr2Ni2MoAA	0.04 0.12	0.15 0.45	0.45 1.1	1.4 2.1	1.0 1.3	0.45 0.75	–	0.08	0.015	0.012	0.01	0.02	0.005	0.008	–
Sv-09CrMnNiMoTiAA-VI	0.04 0.10			1.2 2.0	0.9 1.3	0.4 0.7	–	0.06	0.012	0.008					–

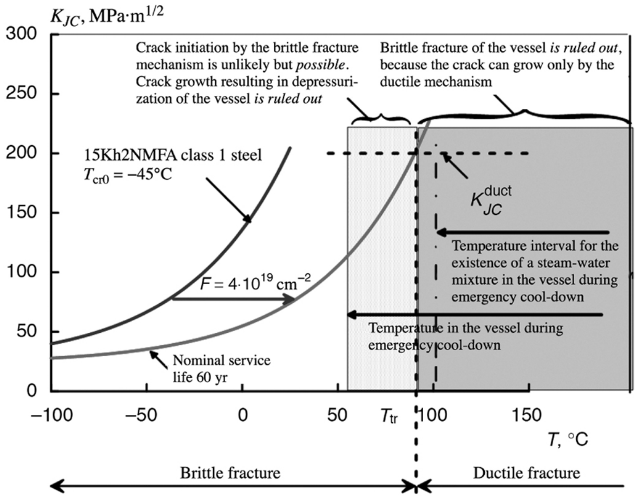

Fig. 4 Securing level-2 operational safety of VVER-1200 RPV, made from steel 15Cr2NiMoVA class 1 [4]

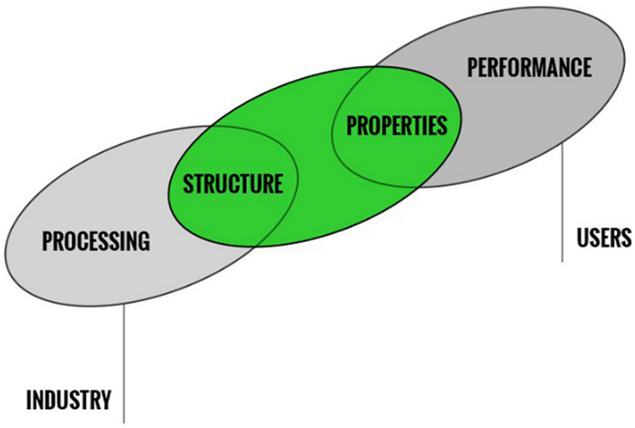

Fig. 5 Chain model of materials science and engineering [6]

Among materials challenges we can identify those issues which have not yet been solved utilizing the knowledge and expertise of 20,000 reactor years. In addition, those can be identified which are in relation to new and/or unanticipated ageing effects and today still not exactly identifiable ones (may be characterized as "unknown unknown"[1] issues) generated by the 60+ operational years of LTO.

Standard review articles, e.g., [7] confirm that the major structural materials challenges of both current and future NPPs are the corrosion in the components of the reactor coolant system and the fast neutron irradiation of the reactor components. Among the numerous corrosion modes, the primary issues are still the stress corrosion cracking (SCC) and – taking into account synergy – the irradiation-assisted stress corrosion cracking (IASCC). When mentioning neutron irradiation of reactor components, it refers to the embrittlement of the ferritic RPV wall. Hereafter these two degradation modes (ageing effects) will briefly be discussed.

1 This phrase was used by Mr. D. Rumsfeld, US secretary of defence in 2002 in relation to the reason of entering Iraq, and adopted by materials scientists later on.

4.1 Stress corrosion cracking

Even though design codes treat SCC as a manageable degradation mode by using materials with known, good behavior, it occurs in NPPs and, of course, in VVER environment [8]. SCC is traditionally presented as a combination of susceptible material, tensile stress and corrosive medium. Material factors are the chemical composition, the microstructure and the surface condition; stresses can be induced by operation or by manufacturing (residual stresses); and the environment can be characterized by the corrosive agents, the flow rate and the electrochemical potential. The crack growth may be intergranular (IGSCC) or transgranular (TGSCC).

Obviously, SCC is a complicated degradation process, having many factors mentioned above that lead to a significant uncertainty in its forecast or even exclusion. This means that current and future plant operators have to face this phenomenon. A good example is the IGSCC discovered recently in the emergency core cooling pipes of some French NPPs [9]. Here, the affected materials were AISI 304 L and 316 L steels which have reached an increased sensibility caused by repair during manufacturing and by stresses as consequence of thermal stratification.

Whereas the materials of the French pipes are non-stabilized austenitic steel, IGSSC has occurred in Ti-stabilized stainless steels in Russian plants too. Fig. 6 shows IGSCC in the down-comer pipes of an LWGR plant (light water cooled, graphite-moderated reactor; Russian acronym is RBMK). The cracks were grown close to the fusion surface in the weld heat affected zone. After this relatively low probability phenomenon was discovered the International Atomic Energy Agency launched a program to determine the root causes. Among others, deformation caused stresses in the pipe inner surface layer, high weld heat input indicated by the weld geometry and justified by the coarse grain structure were identified [10]. Also, sulphate ions due condenser leakages were found on the crack surface.

The stainless steel applied for VVER-1200 pipelines in the primary systems is the Ti-stabilized steel 08Cr18Ni10Ti (equivalent to AISI 321). For RPV cladding the first (i.e., buttering) layer is 07Cr25Ni13A (non-stabilized) and the other layers are made of 08Cr19Ni10Mn2NbA (Nb-stabilized) strip electrode.

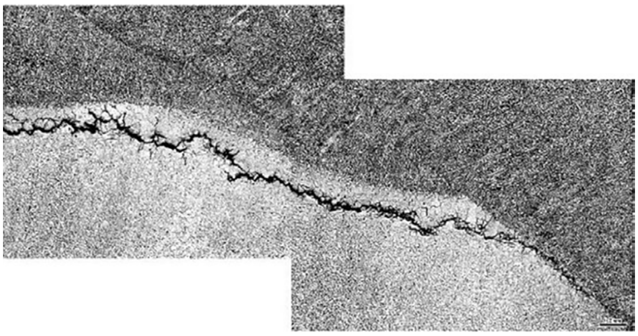

Fig. 6 IGSCC in the weld heat affected zone of a DN 300 pipeline of LWGR plant [10]

Here, the protection against SCC is ensured by alloying either Ti (piping material) or Nb (cladding material) to avoid grain boundary sensitization.

4.2 Irradiation embrittlement

Fast neutrons leaving the reactor core cause complex changes in the microstructure and, consequently, in the mechanical properties of the RPV pressure retaining boundary. The most critical change is the embrittlement which includes the shift of ductile-to-brittle transition temperature (DBTT) and the loss of fracture toughness. As it was stated before RPV integrity is jeopardized during transient operating regimes and when emergency core cooling system is activated (PTS). PTS transient is characterized by thermal stress due to the rapid cooling of the vessel wall and mechanical stress due to the re-pressurization of the vessel at the same time. These result in large tensile stresses in the RPV inside surface. In case a crack exists in the vessel wall near to or on the inside surface and the material has degraded due to neutron irradiation the PTS transient can cause its unstable growth, i.e., the brittle fracture of the RPV.

RPV analysis against brittle fracture is traditionally performed applying the global approach. It means that the material's fracture toughness is expressed by an empirical fracture toughness reference curve which due to its deterministic nature cannot handle the relatively large scatter of the K_{Jc} values in the ductile-to-brittle transition area. To solve this problem probabilistic methods, so called local approach methods, have been developed and applied. The basis of local approach is the weakest link statistical model: cleavage is initiated if the stress reaches its critical value at the weakest point of the material (initiators of cleavage) thus brittle fracture is determined by the probability of coincidence of cleavage initiator and crack front. Cleavage initiators could be non-metallic inclusions, grain boundaries, second phases etc. The most widely adopted probabilistic method is the Master Curve (MC), e.g., [11]. Besides this, the Unified Curve (UC) method has gained increasing attention [12] which was developed by a leading Russian scientific institute. Both MC and UC methods were standardized; the UC is applied for assessing the VVER-1200 RPV resistance against brittle fracture.

Numerous analyses compare these two probabilistic methods. According to the comparisons the most significant difference is visible on the prediction of the material's irradiation embrittlement. The temperature dependence of fracture toughness (reference curve) at fracture probability $P_f = 0.5$ and for specimen thickness $B = 25$ mm for any phase of embrittlement is described by the following equation, according to MC method:

$$K_{Jc(\text{med})}(T) = K_{Jc}^{\text{shelf}} + \beta \times \exp\left[\gamma \times (T - T_0)\right], \quad (2)$$

where T_0 is the reference temperature in °C for 100 MPa√m, T is the temperature in °C, K_{Jc}^{shelf} = 30 MPa√m, β = 70 MPa√m and γ = 0.019; and according to UC method:

$$K_{Jc(\text{med})}(T) = K_{Jc}^{\text{shelf}} + \Omega \times \left[1 + \tanh\left(\frac{T - 130}{105}\right)\right], \quad (3)$$

where K_{Jc}^{shelf} = 26 MPa√m, Ω is a constant and T is the temperature in °C.

According to Eq. (2) when embrittlement increases a lateral shift of the reference curve occurs, but the shape of the curve remains the same. According to Eq. (3) however for the embrittled material a vertical evolvent of the reference curve occurs (Fig. 7).

The basic conclusion of the comparison of MC and UC method, according to [13], is that fracture toughness values practically coincide with each other for initial (non-irradiated) material condition, at least at K_{Jc} < 120 MPa√m, corresponding maximum stress intensity factor under PTS loading. For irradiation material fracture toughness dependence calculated with UC is however more conservative than calculated with MC.

The MC and also UC concepts seem not definitively closed. For example, the standard ASTM 1921 [14] (Standard Test Method for Determination of Reference Temperature, T_0, for Ferritic Steels in the Transition Range) which is based on MC is under permanent improvement (it has had so far more than two dozen of modifications). Therefore, it is not surprising that still new probabilistic models are recommended, see e.g., the one in [15] which

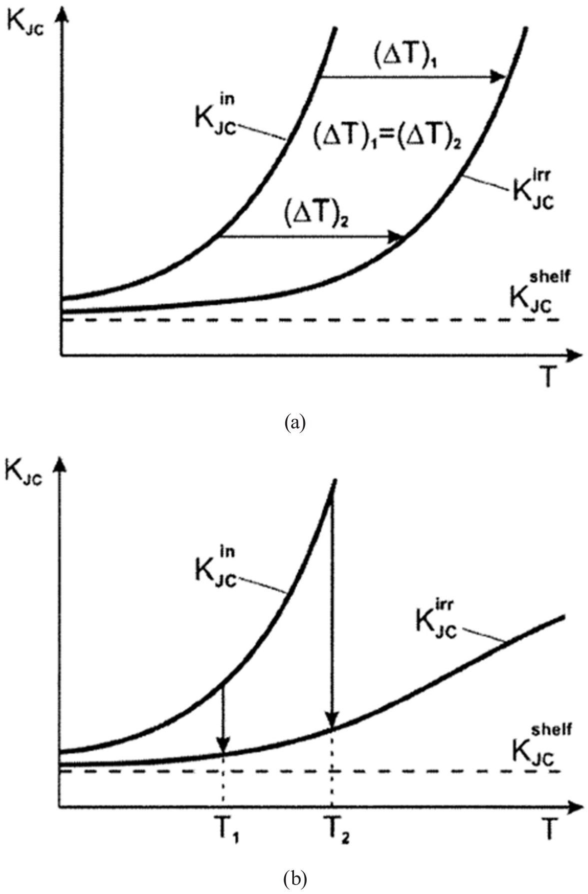

Fig. 7 Transformation of the reference curve for irradiated materials according to MC (a) and UC (b) [13]

intends to eliminate the imperfect theoretical foundations and the lacks in strict correspondence with Weibull statistics of the other models mentioned here.

5 Conclusions, summary

The new reactors including the VVER-1200 units in Hungary utilize the operational experiences accumulated during the decades passed, and its structural materials went through on an evolutionary development. The designer's major focus has been nuclear safety but the optimized service life is also inevitable. The scope of the article was limited to the structural materials whose performance determines the long-term, safe operation of the reactors.

As major service induced ageing effects the corrosion, particularly SCC of austenitic stainless-steel pipelines, and irradiation embrittlement of the RPV ferritic wall were identified. Both degradation effects can be slowed down and keep within the safety limits however their ultimate exclusion is unlikely. Key is the chemical composition especially in relation to maintain resistance against irradiation embrittlement. By decreasing the content of harmful impurity as well as certain alloying elements, e.g., Ni, of the RPV material an extremely low initial transition temperature and, at the same time, a high fracture toughness could be achieved. However, unknown or unforeseen degradation in the late service period cannot be excluded. In addition to materials development the RPV structural integrity assessment method is under serious metamorphosis. The proven in practice however scientifically never justified global, deterministic approach has been replaced by the local, probabilistic approach based on the weakest link statistical method. In case of VVER-1200 the Unified Curve describes the temperature dependence of fracture toughness.

Concerning SCC, the tendency for long-term operation of NPPs clearly justifies that it is still an issue. Due to the quite large number of influencing factors, we can say that it is not a question of a resistant structural material but the question of its occurrence with lower or higher risk. Thus, the best way of decreasing the risk is the properly prepared and performed non-destructive testing and evaluation.

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