Structural Integrity Assessment of VVER-1000 RPV under Accidental Cool down Transients

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Abstract. Corrosion, Fatigue and Irradiation embrittlement are the major degradation mechanisms responsible for ageing of RPV (and its internals) of a Pressurized Water Reactor. While corrosion and fatigue can generate cracks, irradiation damage can lead to brittle fracture initiating from these cracks. Ageing in nuclear power plants needs to be managed so as to ensure that design functions remain available throughout the life of the plant. From safety perspective, this implies that ageing degradation of systems, structures and components important to safety remain within acceptable limits. Reactor Pressure Vessel has been identified as the highest priority key component in plant life management for Pressurized Water Reactors. Therefore special attention is required to ensure its structural integrity during its lifetime. In this paper, structural integrity assessment for typical VVER-1000 RPV is carried out under severe accidental cool down transients using the Finite Element Method. Three different accidental scenarios are postulated and safety of the vessel is conservatively assessed under these transients using the Linear Elastic Fracture Mechanics approach. Transient thermo mechanical stress analysis of the core belt region of the RPV is carried out in presence of postulated cracks and stress intensity factors are calculated and compared with the material fracture toughness to assess the structural integrity of the vessel. The paper also include some parametric analyses to justify the methodology.

1. Introduction

Reactor pressure vessel (RPV) of a pressurized water reactor (PWR) system is the key equipment governing the life of the plant. During the service life, RPV is subjected to a variety of steady as well as transient thermo-mechanical loads under normal operation, upset, emergency and faulted conditions. One of the most severe transient loads which may jeopardize the safety of a PWR-PV is the sudden cooling (thermal shock) of the hot reactor vessel under accidental conditions. The RPV of a PWR may be subjected to such condition during loss of coolant accident (LOCA), with the injection of cold water by the emergency core cooling system (ECCS) of the plant. Such a scenario leads to very high tensile thermal stresses near the inner surface of the vessel as shown in Fig. 1. If, under such a condition, a longitudinal crack is present near the inner surface of the RPV, this high tensile thermal stress combined with the tensile stress due to internal pressure can provide high opening stresses to the crack posing a threat to the structural integrity of the vessel.

The material of the RPV must have sufficient ductility in order to safely sustain such loads without the possibility of a brittle fracture. On the other hand, the material of the RPV keeps losing its ductility over years of operation owing to irradiation damage. Assurance of safety demands that the RPV must remain capable of withstanding such severe transient thermal loads without impairing its integrity at all times throughout its service life. Pressure vessel integrity evaluation is therefore an important task in the Plant Life Management (PLIM) of Nuclear Power Plants (NPPs).

2. Material Fracture Toughness and Assessment of Irradiation Damage

Fracture toughness, characterized by the critical Stress Intensity Factor (SIF), K_{IC}, under the framework of Linear Elastic Fracture Mechanics (LEFM), is a strong function of temperature for ferritic materials.

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The critical SIF (or allowable SIF for emergency conditions) for *Cr-Mo-Ni-V* low alloy steel for VVER-1000 RPV as a function of temperature is given by the Russian Design Code [1] as:

$$K_{IC} = 65.2 + 18.12.\exp\{0.0293\,(T - T_K)\}, \qquad (1)$$

where, T is the metal temperature, and T_K is an indexing temperature that represents the ductile to brittle transition temperature (DBTT) of the material. The value of T_K is determined by carrying out charpy impact tests at various temperatures.

As the material ages due to irradiation damage, the loss of ductility is envisaged as an upward shift of DBTT. Based on various experimental and surveillance data on the RPV material, following empirical relationship is given in the Design Code [1] for assessment of irradiation damage in terms of the shift in DBTT:

$$T_K \text{ (after irradiation damage)} = T_{K0} \text{ (initial)} + \delta T_F \text{ (shift due to fluence)} \qquad (2)$$

where,

$\delta T_F = (A_F)\,(\Phi/\Phi_0)^{1/3}$,
A_F = a constant depending on the composition of RPV steel,
Φ = fluence ($E \geq 0.5$ MeV) in n/m^2, and
Φ_0 = reference fluence = 10^{22} n/m^2

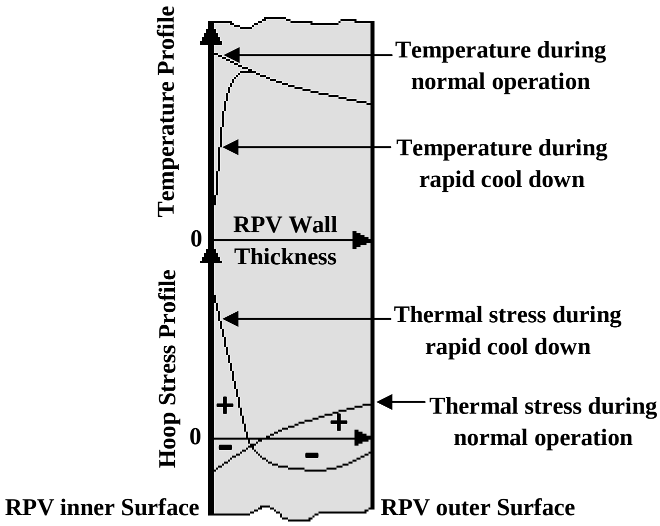


FIG. 1. Temperature and Thermal Stress Distributions across the Thickness of RPV during Normal Operation and during Cool Down Transient.

3. Approach to Structural Integrity Assessment of RPV

Structural integrity evaluation of RPV is a multi-physical problem involving thermal hydraulic, structural and fracture mechanics analyses, in combination with reactor dosimetry calculations. The stages involved in the process are shown in Fig. 2.

The basic philosophy is to postulate severe cracks at critical location of the RPV and determine their fracture mechanical behaviour under the severe most loading conditions. Under the framework of LEFM, severity of a crack is assessed based on SIF. A crack becomes unstable when the applied SIF, K_I becomes greater than or equal to the material critical SIF, K_{IC}. The governing equation for assurance of structural integrity is therefore, $K_I \leq K_{IC}$. This equation must hold good even at the end of service life of the RPV.

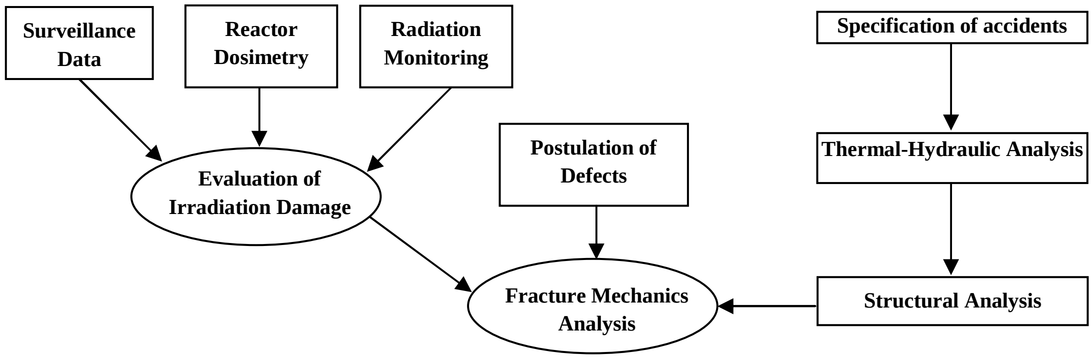


FIG. 2. RPV Structural Integrity Assessment Methodology

4. Structural Integrity Assessment of VVER-1000 RPV under Severe Cool down Transients

India is presently constructing NPP with VVER-1000 type reactors. In view of this, a safety analysis of the VVER-1000 RPV was carried out. The work presented in this paper is a part of the above analysis. Results of RPV integrity evaluation under following three postulated accidental scenarios leading to severe cool down transients are included in the paper:

(1) Inadvertent opening of the pressurizer safety valve followed by its failure to seat.
(2) Leak of diameter 25 mm from the main coolant pipe with four ECCS channels under operation.
(3) Leak of diameter 105 mm from the main coolant pipe with four ECCS channels under operation.

The RPV core belt region, which is the most critical location of RPV for irradiation damage is investigated for the ability to withstand the accidental transients in presence of cracks. The temperature and pressure transients seen by the inner surface of the RPV in the core belt region as a result of the above postulated LOCA conditions are derived from detailed thermal hydraulic calculations. These transients are shown in Fig. 3. Transients-1, 2 & 3 refer to the above three LOCA conditions respectively.

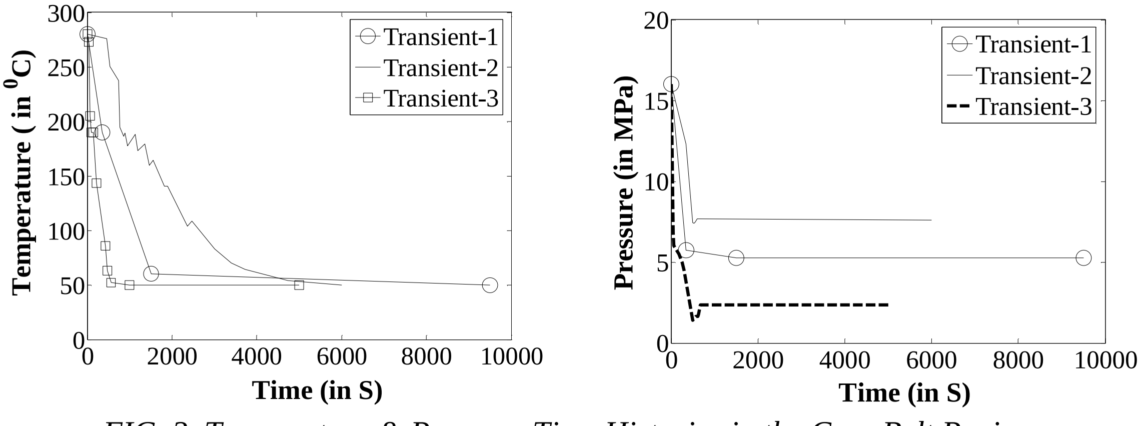


FIG. 3. Temperature & Pressure Time Histories in the Core Belt Region

4.2. Procedure

Transient thermo-mechanical stress analyses of the core belt region are carried out in the presence of postulated cracks using the Finite-Element Method (FEM). The core belt region of VVER-1000 RPV having following dimensions – inner diameter (excluding clad) = 4150 mm, base metal wall thickness = 192.5 mm, and length = 3500 mm, is modeled as a quarter cylinder with symmetrical half-length. Clad is not included in the model. A semi-elliptical longitudinal surface crack with aspect ratio 1:6 (as per ASME Code [2]) and size varying from 10 mm to a maximum of 48 mm (25% of wall thickness as per ASME Code [2]) is postulated to exist at the inner surface. Figure 4 shows the FE mesh pattern of a typical cracked model. Quarter point elements are used in the neighbourhood of the crack to simulate the square root singularity.

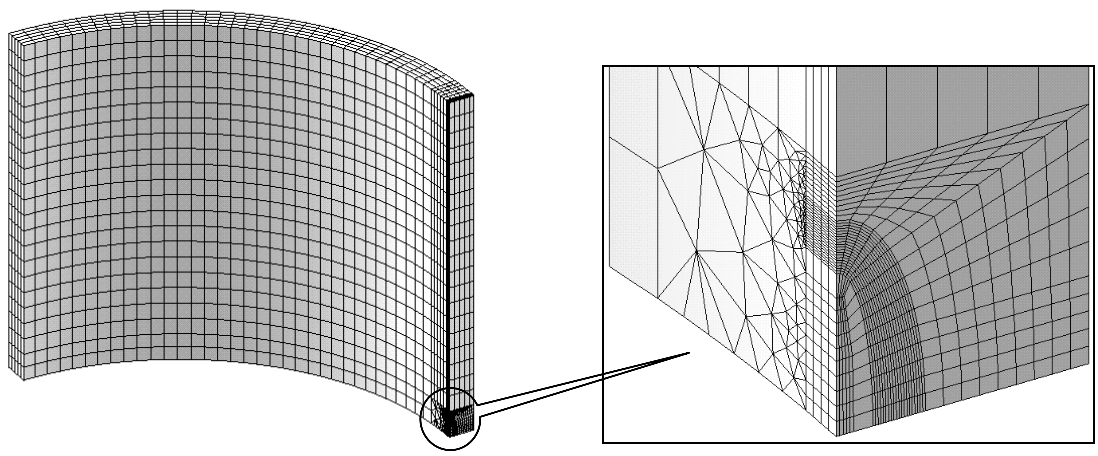


FIG. 4. Finite Element Model of VVER-1000 RPV Core Belt Region with a Postulated Crack

Thermo-mechanical properties of the RPV material required for the analyses are taken from Ref. [3]. For thermal analyses, convective boundary condition is applied at the outer surface of the model, with bulk temperature of 50 ^{0}C and heat transfer coefficient, h, calculated from standard Handbook solution for a vertical cylinder considering laminar flow conditions [4]:

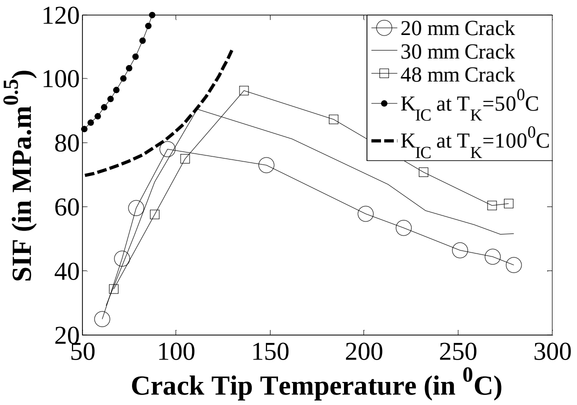
$$h_{Laminar} = \begin{cases} 0.65\ \text{W/m}^2\cdot\text{K – for steady state analysis} \\ 0.325\ \text{W/m}^2\cdot\text{K – for transient analysis} \end{cases}$$

Temporal displacement and stress field solutions are obtained from the thermal-stress analyses spanning the time period of the transient. Following this, fracture calculations are carried out to determine the crack driving force, SIF, at the deepest crack tip at various time points of the transient, from the instantaneous displacement field solution in the vicinity of the crack tip using the formula given in Ref. [5].

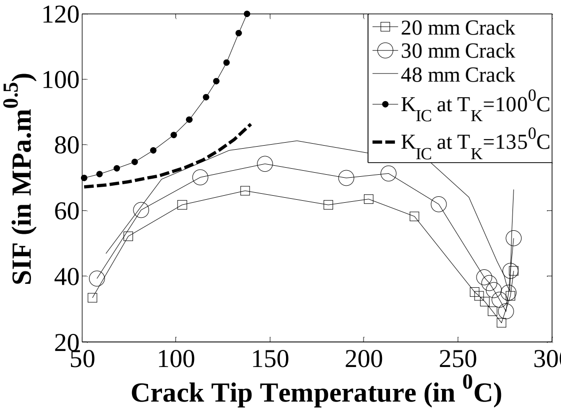
4.3. Results

The temporal values of SIF at the deepest crack tip during the transient are plotted with respect to the crack tip temperature for the spectrum of cracks analyzed, and are compared with the material fracture toughness (also varying with temperature as per Eq. 1) to assess the severity of the cracks leading to brittle fracture.

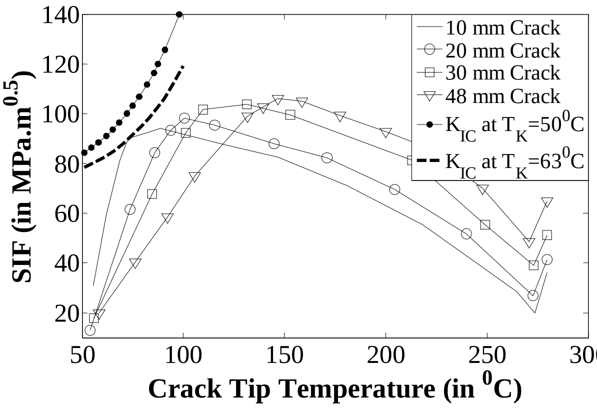
Figures 5(a)-(c) show the above fracture assessment diagrams for transients 1-3 respectively. The minimum value of T_K for which the material SIF curve becomes tangent to the applied SIF curves is considered as the limiting value of the end of life DBTT (corresponding to the particular transient) for assuring structural integrity of the RPV throughout its safe operating life. Therefore, from Fig. 5, it is determined that the end-of-life DBTT of the RPV must be limited to 100 ^{0}C for safety under transient-1, 135 ^{0}C for transient-2 and 63 ^{0}C for transient-3. This means that transient-3 which is a large break LOCA from the main coolant pipe is the most severe of the three events. Considering all the three postulated events, the end-of-life DBTT should be limited to 63 ^{0}C for assuring safe operation of the NPP.



***(a)** For Transient-1*



***(b)** For Transient-2*



***(c)** For Transient-3*

FIG. 5. Fracture Assessment Diagrams for VVER-1000 RPV

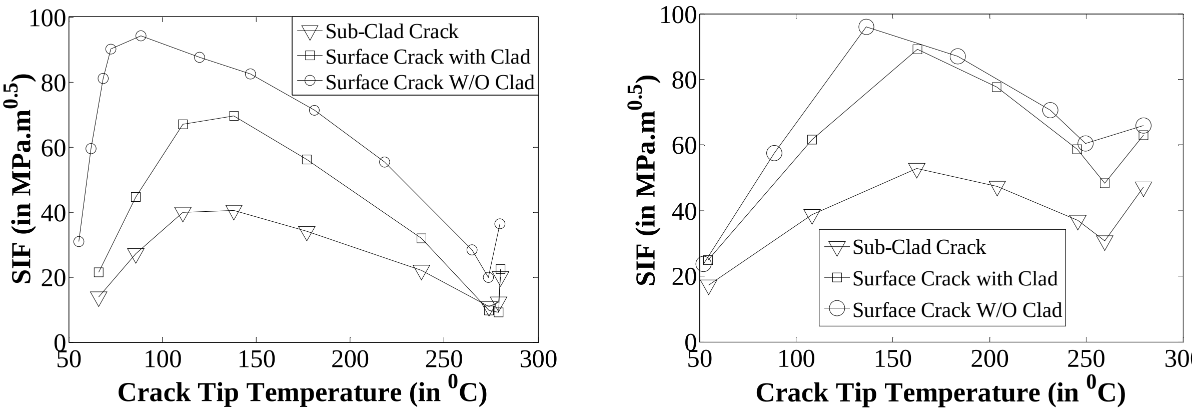
5. Parametric Study

5.1. Effect of Austenitic Stainless Steel Clad

Clad is not considered in the structural integrity assessment of VVER-1000 RPV presented above. A parametric study is carried out to determine the effect of consideration of clad on the integrity evaluation. For this study, austenitic stainless steel clad of thickness 9 mm is included in the RPV cracked model. 10 mm & 48 mm cracks are investigated under the action of transients-3 & 1 respectively. Thermo-mechanical properties for the clad material are taken from Ref. [3]. Three different cases are studied:

(1) Surface crack without consideration of clad

(2) Surface crack with consideration of clad
(3) Sub clad crack



***(a)** For 10 mm Crack under Transient-3*

***(b)** For 48 mm Crack under Transient-1*

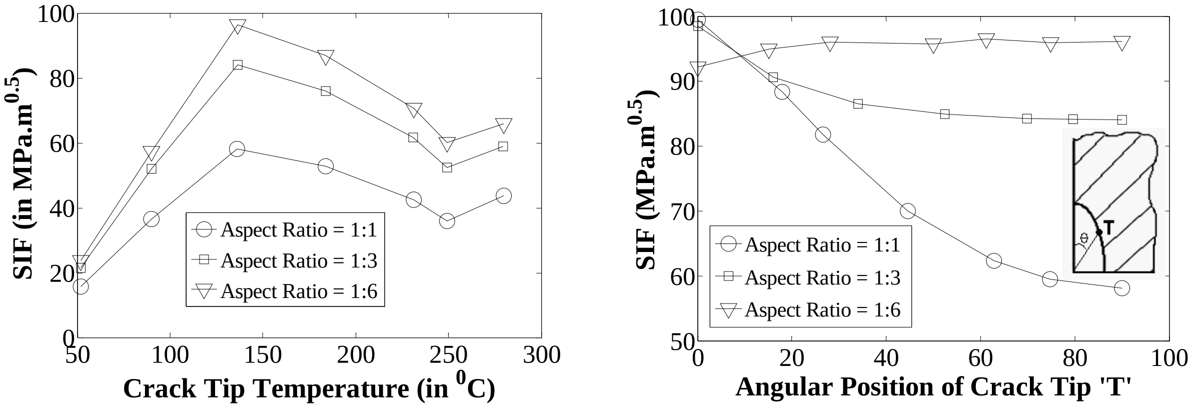
FIG. 6. Effect of Clad on Crack Driving Force at the Deepest Crack Tip

Crack driving forces (SIF) at the deepest crack tip for the above three cases are compared in Fig. 6. It is clearly evident from this comparison that the crack driving force is lowest for the sub clad crack and highest for the surface crack without consideration of the clad. The crack driving force for the surface crack with consideration of the clad is intermediate and substantially less severe than that without consideration of the clad. This shows that clad plays a significant beneficial role in reducing the severity of surface cracks. Therefore, it may be noted from this parametric study that, while ignoring the clad for integrity evaluation is satisfactory for the initial conservative safety assessment at the design/installation stage of NPPs, the clad must be considered in the evaluations done at later stages in the life of NPPs for the purpose of residual life estimation/extension.

5.2. Effect of Crack Aspect Ratio

Shape of the crack has a major influence on the crack driving force. It is known that semi-elliptical cracks produce higher crack driving force than the circular cracks. Russian Code [1] requires postulation of semi-elliptical cracks with aspect ratio of 1:3, whereas the American Code [2] recommends an aspect ratio of 1:6. Therefore, a parametric study is carried out to determine the effect of the crack aspect ratio and identify the one that should be used for the integrity evaluation. For this study, a 48 mm crack is investigated under the action of transient-1 without consideration of the clad. Three different aspect ratios are analyzed, viz., 1:1, 1:3 & 1:6, and the results are shown in Fig. 7. Figure 7(a) shows the variation of SIF at the deepest crack tip with crack tip temperature for the three cracks with different aspect ratios. It can be clearly identified that the crack with aspect ratio of 1:6 produces the most severe crack driving forces. Figure 7(b) shows the variation of SIF along the crack profile at a specific time point of the transient for the three cracks with different aspect ratios. It is found that the crack driving force is fairly uniform along the crack for the aspect ratio of 1:6. Hence, this shape most resembles the natural cracks.

It is therefore recommended that semi-elliptical cracks with aspect ratio of 1:6 should be postulated for integrity evaluations as they most resemble the natural cracks and produce the most severe crack driving forces.



***(a)** SIF at the Deepest Crack Tip* ***(b)** SIF variation along the Crack at 1500 S*

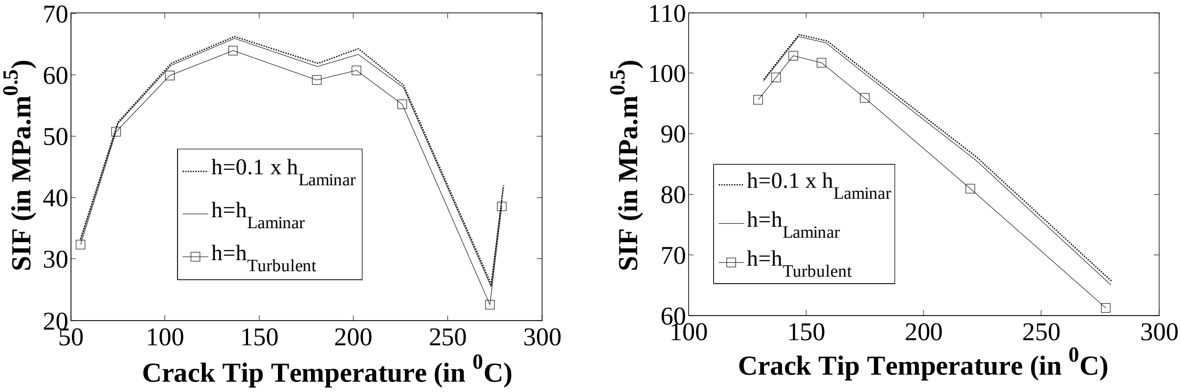
FIG. 7. Effect of Aspect Ratio on Crack Driving Force

5.3. Effect of Uncertainty in Heat Transfer Coefficient

Heat transfer coefficient, h, used in the transient thermal analysis is calculated from empirical correlations that are known to be approximate. Therefore a sensitivity study is carried out to determine the possible magnitude of error in the estimation of crack driving forces due to the uncertainty in the value of h. For this study, a 20 mm crack is investigated under the action of transient-2, and a 48 mm crack is investigated under the action of transient-3. It is assumed that the actual value of h will surely lie in the range ($h_{Laminar}/10$) < h < $h_{Turbulent}$, where, $h_{Turbulent}$ is the value of h calculated considering turbulent flow conditions [4]:

$$h_{Turbulent} = \begin{cases} 8.0\ \mathrm{W/m^2 \cdot K} \text{ – for steady state analysis} \\ 4.0\ \mathrm{W/m^2 \cdot K} \text{ – for transient analysis} \end{cases}$$

Figures 8(a) & (b) show the crack driving forces determined using three different values of h, viz., ($h_{Laminar}/10$), $h_{Laminar}$, and $h_{Turbulent}$. It is found from the Fig. that the maximum possible error in the magnitude of the estimated value of SIF due to the uncertainty in the value of h, is less than 3.5%. Hence, it can be concluded that the accuracy of integrity evaluation is not sensitive to the heat transfer coefficient.



(a) For 20 mm Crack under Transient-2 *(b) For 48 mm Crack under Transient-3*

FIG. 8. Sensitivity of Crack Driving Force to Heat Transfer Coefficient

6.0 Conclusion

Structural integrity evaluations for VVER-1000 RPV core belt region are presented for severe accidental cool down transients occurring due to postulated LOCA conditions. A large break LOCA from the main coolant pipe emerges to be the most severe scenario with the limiting end-of-life DBTT of only 63 ^{0}C and a critical crack length of 10 mm. This is due to the very effective Emergency Core Cooling System of VVER that is capable of compensating large break leaks but at the same time inducing high thermal stresses on the RPV. The *emergency operation procedures*, must therefore, be timely and correctly followed to cope with such accidental transient safely. Based on Eq. 2 and reactor dosimetry calculation results, it is estimated that the limiting value of DBTT of 63 ^{0}C corresponds to a Safe RPV life of 12.5 Full Power Years.

It is however, important to note, that the above results are on the conservative side as the evaluations presented in this paper are carried out without consideration of the austenitic stainless steel clad.

Parametric studies show that the austenitic clad plays a significant beneficial role in reducing the severity of cracks for brittle fracture. Having a lower thermal conductivity, it essentially acts as a thermal buffer delaying the cooling down of the base metal, thereby reducing the thermal stresses. Therefore, while ignoring the clad is satisfactory for initial conservative safety assessment of NPPs, it must be considered in integrity evaluations carried out at later stages for residual life estimation/extension.

It is also shown that the semi-elliptical crack with aspect ratio 1:6 as recommended by the ASME Code most resembles natural cracks and also produces more severe crack driving force, and hence is more justified for integrity evaluations as compared to the crack shapes recommended by other standard Codes.

A sensitivity analysis carried out reveals that the accuracy of the RPV integrity evaluations is practically independent of that of the convective heat transfer coefficient used in the thermal-stress analysis.

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