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A fracture mechanics analysis of the PWR nuclear power plant reactor pressure vessel beltline weld

Lih-jier Young

Department of Applied Mathematics, Chung-Hua University, Hsin-Chu, Hsin-Chu 30067, Taiwan, ROC

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Abstract

In this paper, a generic pressured water reactor (PWR) power plant reactor vessel is analyzed. The primary purpose of this work is to assure structural integrity of the irradiated reactor pressure vessel (RPV) beltline weld under normal operating conditions at the end-of-life, as specified by the requirements of 10CFR50, App.G. It is found that suitable margins of safety are maintained at the end-of-life (32 effective full power years or 1.25×10^{23} n/m^2 inner wall fluence).

1. Introduction

The reactor pressure vessel (RPV) is the structural component of the most fundamental importance in a nuclear power plant. The RPV houses the radioactive nuclear fuel which produces heat and the heat thus generated is transmitted to the surrounding media. It is, therefore, one of the major concerns in the nuclear power industry to maintain the structural integrity of the RPV for safety operating the plant.

The RPV undergoes various loadings during normal plant operations such as the heatup, the cooldown, and the hydro-test. Under these operations, the reactor coolant is subjected to temperature and pressure changes of moderate rate and amplitude, resulting in stresses in the reactor vessel wall. A large number of these operations occur during the total life of the plant. In order to establish and demonstrate its integrity under all operation loading, the fracture mechanics analysis of an RPV forms a substantial portion of nuclear safety analysis. A simplified method of performing these analyses for non-brittle fracture will be described in this paper.

2. Description of a PWR RPV

Fig. 1 is a schematic drawing of a typical pressured water reactor (PWR) with steam generator, cold emergency cooling water inlet and internal diameter of about 4.4 m and wall thickness of about 0.225 m in the beltline region (core region). The beltline section of the vessel is fabricated from two forged ring plates, which are welded together by an automatic submerged arc process. Both the plate material, usually ASTM A-533, Grade B, Class 1 plate material or its forging equivalent, ASTM A-508, Class 2 material, and the weld metal used in this region exhibit high toughness in a newly fabricated RPV. The mechanical toughness properties of the weld metal are shown in Table 1. However, after long periods of neutron irradiation the toughness of the material may significantly decrease, resulting in a possible susceptibility of the RPV to fracture.

3. Region of interest

Four regions of the reactor vessel are considered as critical regions in Buchalet and Bamford [1], i.e., the nozzles, the beltline, the closure head-to-flange juncture, and the lower shell-to-bottom head transition. The thickness transition area acts as stress riser giving rise to high stresses. Since the crack driving force is proportional to the stress, crack initiation is more likely to occur in these areas. The stresses developed in the beltline region are not the largest, but this region receives the highest neutron bombardment which decreases the material toughness, thereby increasing the potential of crack instability. Hence, the beltline region is the area of

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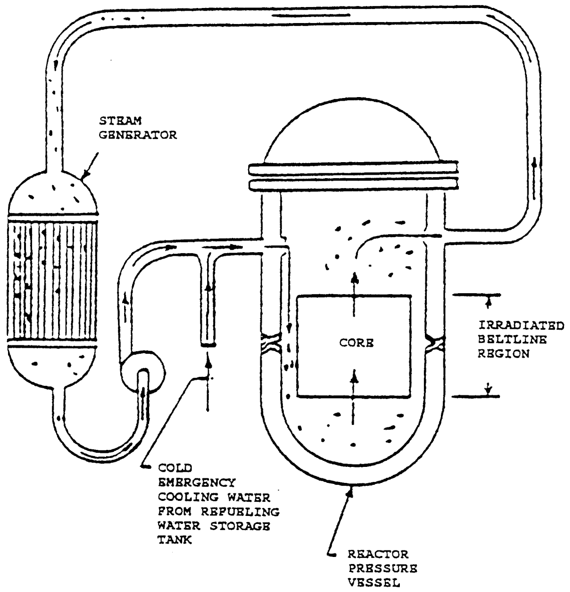

Fig. 1. Illustration of PWR reactor pressure vessel.

Table 1
Mechanical toughness properties of weld metal in beltline region

Impact energy	54.2 N m
Near deductivity transition temperature (T_{NDT})	−17.8°C
Yield strength	475.7 MPa
Tensile strength	579.2 MPa
Elongation rate	28.4%
Cross-section shrink rate	65.3%

interest due to irradiation embrittlement of the vessel material.

4. Method of approach

The fracture mechanics analysis for a PWR reactor beltline region is made following the relevant parameters given in the ASME Boiler and Pressure Vessel Code, Section III (Appendix G). The code requires the maximum postulated flaw to be a surface flaw having a depth of one-quarter of the wall thickness ($t/4$) and a length of 1.5 times wall thickness ($1.5t$). This analysis will examine longitudinal surface flaws. It was found that the combination of the membrane stress at the maximum operating pressure, the thermal stress due to normal heatup and cooldown, and the estimated residual stress at the beltline area remains considerably below the yield strength of the material. This allows the use of linear elastic fracture mechanics (LEFM) methods for determining the applied value of K, K_{app}. This value can be obtained by superposition of K values determined for the individual applied loads with the code safety factor of 2 applied to the K value due to the primary stress components. The K_{app} is then converted into J_{app} simply by

$$J_{app} = \frac{K^2}{E'}, \quad (1)$$

where $E' = E/(1 - \nu^2)$ for plane strain, E is the Young's modulus and ν is the Poisson's ratio of the material.

The LEFM approach is exactly a special case of a more general elastic–plastic fracture mechanics approach based on J fracture parameter as described in NUREG-0744 by Paris, and Johnson [2]. This method will be used to check the crack stability for the postulated flaw at reactor beltline region during normal operations.

5. The determination of J_{app}

5.1. K_{app} due to internal pressure

The applied value of J, J_{app}, in LEFM analysis can be converted from the applied value of K, K_{app}, as described in Eq. (1). The K_{app} is calculated in accordance with ASME code as follows:

$$K_{app} = 2K_p + K_{th} + K_{res}, \quad (2)$$

where $K_p = K$ due to membrane stress by internal pressure, $K_{th} = K$ due to thermal stress for heatup and cooldown, $K_{res} = K$ due to residual stress of welding.

The factor 2 in Eq. (2) is the implicit safety factor specified in the ASME CODE due to primary stresses. The nominal pressure stress to be used in K_p is the hoop stress calculated from the wall without crack for the longitudinal flaw, i.e.,

$$\sigma_p = \frac{pR_t}{t}, \quad (3)$$

where p is the internal pressure, R_t is the inner radius of the vessel, and t is the wall thickness. The stress intensity factor K_{app} for a surface flaw of depth (a), length ($2c$), vessel wall thickness (t) as shown in Fig. 2 is [3]

$$K_{app} = \sigma_p \sqrt{\frac{\pi a}{Q}} F\left(\frac{a}{t}, \frac{a}{2c}, \frac{R_t}{t}\right), \quad (4)$$

where

$$F = 1.12 + 0.053x + 0.0055x^2 + (1 + 0.02x + 0.0191x^2)(20 - R/t)^2/1400,$$
$$x = (a/t)(a/2c),$$
$$Q = 1 + 1.464(a/c)^{1.65}.$$

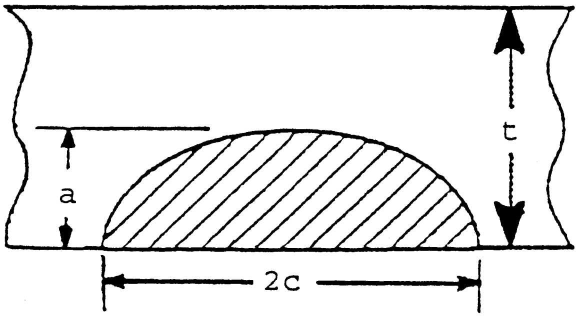

Fig. 2. Stress intensity factor for axial flaw in cylinder subject to internal pressure.

The stress intensity factor, K_{app}, corresponding to the code postulated crack is immediately obtained by replacing a with $t/4$.

5.2. K_{th} due to the thermal stresses

The determination of K_{th} requires the knowledge of the temperature and corresponding stress distribution during heatup and cooldown operations which are determined by actual application and not really part of the fracture mechanics, and therefore, will not be discussed here. However, the temperature and stress profiles through the vessel wall should be fed into fracture mechanics analysis. Once the thermal stress distribution for the maximum heatup and cooldown rate was determined, the K_{th} is calculated through Eq. (5) below

$$K_{\mathrm{app}} = \sigma_{\mathrm{th}} t \sqrt{\frac{\pi a}{Q}} F\left(\frac{a}{t}, \frac{a}{2c}, \frac{R_i}{t}\right), \tag{5}$$

where σ_{th} (t) is the maximum time-dependent thermal stress during the heatup and cooldown operations.

5.3. K_{res} due to weld residual stresses

The weld residual stress distribution is assumed for a double-vee weld in USNRC [4]. The residual stress profile through the weld is parabolic with the maximum tensile stress of 0.12 σ_0 at the mid-thickness of the wall. A conservative of estimated of K_{res} is

$$K_{\mathrm{res}} = 0.12\sigma_0 \sqrt{\frac{\pi a}{Q}} F\left(\frac{a}{t}, \frac{a}{2c}, \frac{R_{\mathrm{t}}}{t}\right), \tag{6}$$

where σ_0 is the flow stress of beltline material at $t/4$ position.

Eqs. (1)–(6) complete the formulation required to establish the applied value of J_{app} throughout normal reactor operations. This J_{app} represents a fracture characterization parameter for a postulated longitudinal surface flaw at its deepest point during heatup and cooldown operations.

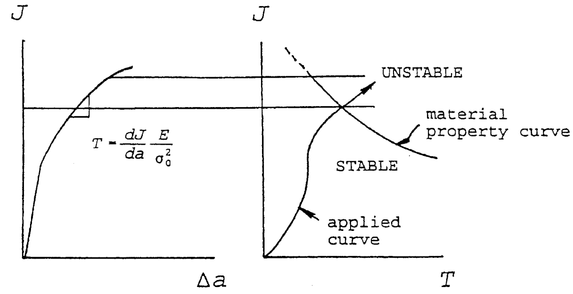

Fig. 3. Basis of tearing modulus for prediction of fracture instability.

6. Determination of the J_{fail} profiles

The material property pertinent to the present fracture mechanics analysis is the J value at instability point, i.e., J_{fail}, in the J–T diagram by Paris et al. [5], where T is the tearing modulus. For safety assessment of nuclear pressure vessel based on the tearing modulus stability concept, it is convenient to present the LEFM and EPFM results in the form of J–T plot. In LEFM the method predicts that crack instability occurs when J_{app} (elastic range) equals J_{fail} which is the failure level in terms of J determined as the ordinate of the intersection point between material J–T curve and applied loading line. In NUREG-0744 it was shown that the J/T loading line is a straight line through the region with a slope of $0.85a$ (σ_0^2/E). Fig. 3 shows the instability occurs when J_{app} is larger than J_{fail}.

7. Effect of irradiation of J–T fracture toughness curves

As indicated in Section 1, the beltline area of the vessel houses the nuclear core and is therefore subjected to irradiation. Thus, the effect of irradiation on the material properties must be taken into account. It has been shown by Chen [6] that a lower bound envelop of J–T plot data can be represented by a hyperbola. Furthermore, the effect of irradiation could be represented by decrease in the hypobola curve constant, C (a product of J and T), as a function of the integrated neutron fluence. It should be pointed out that the maximum accumulated fluence varies through the wall thickness and varies with time during the life of the plant. Fig. 4 shows, for example, maximum calculated fluence profiles through the wall at the end of the plant life (40 years). This fluence profile is based on NRC fluence attenuation model in USNRC [7].

$$\phi(s) = \phi_{\mathrm{surface}}\, \mathrm{e}^{-0.24ts}, \tag{7}$$

where ϕ_{surface} is the maximum fluence at inner surface, s is the fractional distance through the wall x/t and t is the wall thickness.

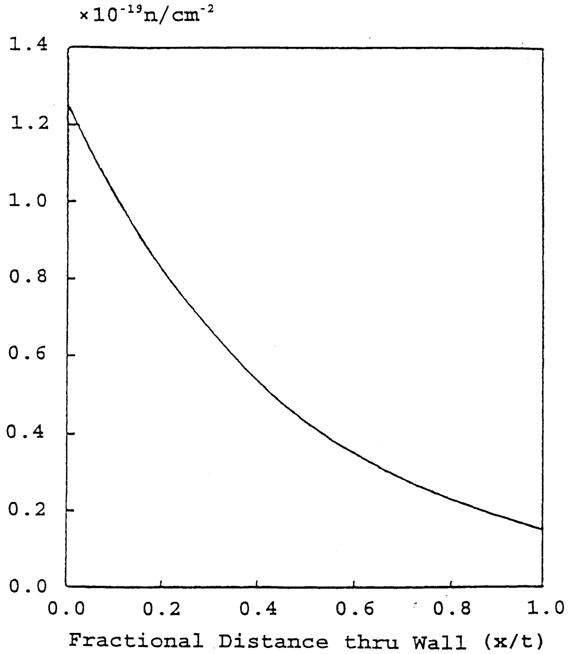

Fig. 4. Fluence profile through wall.

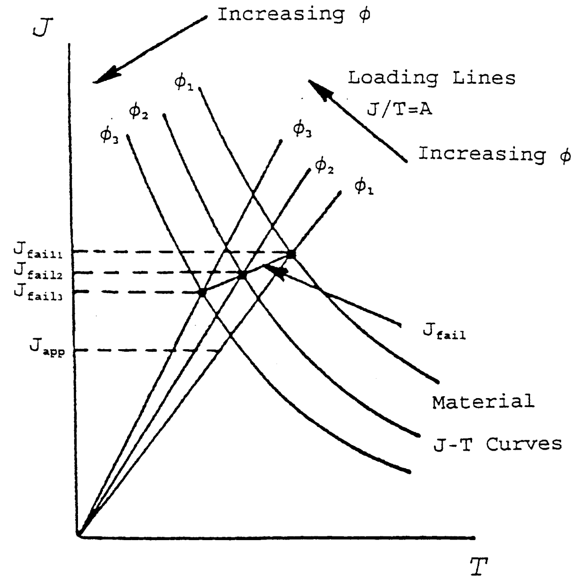

Fig. 5. Schematic of graphical determination of J_{fail}.

Therefore, $C = C(\phi)$ can be determined from the lower bound curves of actual J–T plots obtained at certain ϕ levels in combination with a reasonable interpolation and/or extrapolation scheme by Chen [6]. Once $C = C(\phi)$ is obtained, a family of hyperbolas at different fluence levels can be generated to examine crack instability. The family of these J_{fail} points are plotted onto Fig. 5 as a J-failure line. This line then represents conservative J levels below which no fracture would be expected for $(1/4)t$ postulated flaw.

8. Illustration of numerical results

The maximum heatup and cooldown rate of 37.8°C/h is assumed in the analysis. In addition, a constant reactor coolant pressure, $p = 15.9$ MPa is also assumed to be maintained during the normal operation. The heat conduction temperature solution of vessel wall is obtained analytically and thus the results can be conveniently used in fracture mechanics analysis. That is, the temperature distribution through the wall is obtained at any time during normal heatup and cooldown operations. The corresponding thermal stress distribution is then readily obtained. The material properties and geometric dimensions to be used in the analysis are listed in Table 2. It is noted that the lower bound fracture toughness data from Heavy Section Steel Technology (HSST) program is used for studying crack stability on a generic basis. For the analysis, the weld metal designed as 63w-023 (4T) is to represent the vessel material since it possesses the worst fracture toughness among all of the weld specimens. It is also pointed out that 63w-0.23 specimen was irradiated to 1.25×10^{23} n/m^2 which corresponds to 40 years of reactor operation. Substituting all relevant data in Table 2 into Eqs. (2)–(6), the value of K_{app} for $(1/4)t$ postulated longitudinal flaw in heatup and cooldown operation is obtained, i.e.,

$$K_{\text{app}} = 189\ \text{MPa}\sqrt{\text{m}}. \tag{8}$$

The corresponding values of J_{app} are readily obtained using Eq. (1) giving

$$J_{\text{app}} = 186.5\ \text{KPa m}. \tag{9}$$

To examine the crack stability, the J_{fail} curves similar to those of Fig. 5 were as in Fig. 6. It was observed that at $t/4$ position, the beltline material was capable of sustaining the maximum applied J of 186.5 KPa m (where J_{fail} is 448 KPa m) without crack instability throughout the end of plant life. The corresponding factor of safety

Table 2
Material properties and dimensions of an RPV beltline

Thermal diffusivity (k)	0.04366 m^2/h
Coefficient of thermal expansion (α)	4.1×10^{-6}/°C
Young's modulus (E)	185×103 MPa
Outer radius (R_0)	2.419 m
Inner radius (R_1)	2.194 m
Wall thickness (t)	0.225 m
Flow stress	603 MPa

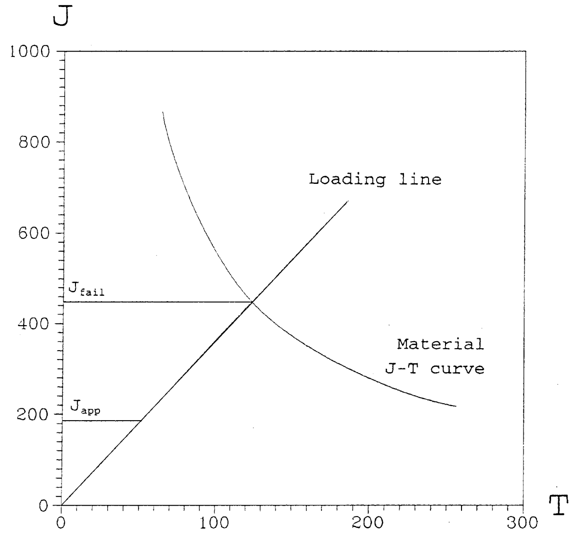

Fig. 6. Crack stability at $t/4$ position of beltline material.

at the end of life was 3.16. Thus, the crack stability was assured.

9. Conclusions

A fracture mechanics analysis was performed to examine the integrity of beltline region of a PWR reactor vessel. The analysis focused on the safety of the beltline material during the normal reactor operations at the end of life. The fracture mechanics loading parameters for the postulated flaws specified in the ASME Code were determined from LEFM approach which was found to be applicable for the situations examined. With the ASME Code factor of safety 2 applied to the primary stresses, J_{app} was compared with J_{fail}. It was found that a suitable margin still existed at the end of the plant life. Thus, the structural integrity of the RPV was maintained and practically, the extended operating life of the vessel might be expected.

Acknowledgements

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