Theoretical and Applied Fracture Mechanics 75 (2015) 124–136

Contents lists available at ScienceDirect

Theoretical and Applied Fracture Mechanics

journal homepage: www.elsevier.com/locate/tafmec

The effects of thermal stresses on the elliptical surface cracks in PWR reactor pressure vessel

Usman Tariq Murtaza *, M. Javed Hyder

Department of Mechanical Engineering, Pakistan Institute of Engineering and Applied Sciences, Islamabad 45650, Pakistan

ARTICLE INFO

Article history:
Available online 9 December 2014

Keywords:
Corner crack
Thermal stress
RPV

ABSTRACT

In this study, the effects of thermal stresses on the stress intensity factors (SIFs) of the elliptical corner surface cracks postulated at the nozzle-cylinder intersection of a reactor pressure vessel (RPV) were investigated. A typical RPV of a Westinghouse pressurized water reactor and its set-in nozzle were considered for the analysis. The selected set-in nozzle-cylinder intersection for fracture mechanics analysis is also the highest stress concentration point of the RPV. The numerically computed SIFs for a wide range of corner cracks under pressure and combined (pressure plus thermal) loadings are provided as a reference tool for the fracture mechanics design of the RPV. It was also demonstrated that the operational thermal stresses caused by the provision of the annular chamber for the external reactor vessel cooling, actually reduce the SIF of the corner cracks and they do not endanger the safety of the RPV in normal operating conditions.

1. Introduction

It is widely accepted that material flaws, pre-cracks and fatigue cracks initiated at stress concentration points normally lead to catastrophic failure of engineering structures. The semi/quarter elliptical shape is the general representation for surface cracks in engineering structures. Surface cracks in infinite geometries can be analyzed using analytical approaches whereas in finite geometries numerical or experimental techniques [1–3] are inevitably required for the analysis of surface cracks. Similarly, on stress concentration areas, analytical techniques have not been able to provide stress intensity factors (SIFs) for surface cracks because of variable stress fields and higher stress gradients at the crack area. In practical applications there are many components such as reactor pressure vessels (RPV), which are subjected to thermal loadings in addition to mechanical loadings. The fracture analysis of such components is much more involved due to different behavior of cracks under thermal and mechanical loadings [4]. Under such complex loading conditions, a comprehensive and accurate three dimensional finite element analysis (FEA) is required for evaluation of SIFs along the whole crack front.

Inspired by the work of Irwin [5], many researchers have performed the analysis of surface cracks in different geometries and loading conditions during the last five decades. Some of the early studies used techniques such as alternating method [6–8], boundary elements [9–11], virtual crack extension method [12–14], the line spring model [15–17] and the weight function approach [18]. After the availability of finite element methods (FEM) and computers, relatively higher attention has been given to the analysis of surface cracks using FEM [19–23]. In this context, Newman and Raju [24] beautifully fit the extensive data with double series polynomial to produce an empirical equation for the surface cracks. The analysis of surface cracks in pressure vessels is relatively more important as their catastrophic failure essentially leads to the loss of life and property. Many researchers have provided useful SIF solutions for surface cracks in simple cylindrical pressure vessels [25–29].

The researchers [30–34] have also performed structural integrity analysis of RPVs under different normal and accidental conditions such as pressurized thermal shocks. Thermal stresses caused by the injection of emergency cooling water have been considered for the RPV integrity analysis while most of the studies have ignored thermal stresses caused by the normal operating conditions of the plant. The structural integrity of RPV is such a large-scale research project that it is not adequate to fulfill the fracture evaluation in accordance with ASME code [35] and an additional research work should be taken into account for safe operation of the plant.

To investigate the effects of thermal stresses, under normal operating conditions of the plant, on the corner surface cracks at

* Corresponding author. Tel.: +92 51 2207380/4x3414.
E-mail addresses: maniiut@yahoo.com (U.T. Murtaza), hyder@pieas.edu.pk (M. Javed Hyder).

http://dx.doi.org/10.1016/j.tafmec.2014.12.001
0167-8442/

Nomenclature

PWR	pressurized water reactor
ERVC	external reactor vessel cooling
RPV	reactor pressure vessel
RVI	reactor vessel insulation
HSCP	highest stress concentration point
SESC-FP	semi-elliptical surface crack in finite plate
SESC-TP	semi-elliptical surface crack in thick pipe
SIF	stress intensity factor
a	depth of semi-elliptical surface crack/minor axis of the crack
c	half-length of semi-elliptical surface crack/major axis of the crack
t	thickness of the reactor pressure vessel wall at the set-in nozzle intersection
Q	flaw shape parameter
E	Young modulus of the material
L	length of the SESC-TP specimen
R_i	inner radius of SESC-TP specimen
P_{int}	internal pressure
T_{in}	temperature of the inner surface of the RPV
T_{out}	temperature of the outer surface of the RPV
S_b	remote bending stress on outer fiber
S_t	remote uniform-tension stress
K_I	SIF in mode-I
K_I^p	SIF in mode-I under only pressure loadings
K_I^{ps}	SIF in mode-I under combined pressure plus thermal loadings
K_{Ic}	fracture toughness of the material
ϕ	crack face angle, degrees
ν	Poisson's ratio of the material

the set-in nozzle-cylinder intersection, a typical RPV [36] of the Westinghouse pressurized water reactor (PWR) is selected for the fracture mechanics analysis. The RPV shown in Fig. 1(a) and (b) is a double loop cylindrical pressure vessel with hemispherical bottom and upper head. The reactor coolant, i.e. pressurized light water, enters the reactor vessel through the set-in nozzle (see Fig. 1) and flows through the reactor core where it absorbs heat. The inlet nozzles usually used in RPVs of PWRs are set-in nozzles (see Fig. 2), which have flange set into the vessel wall [36]. After receiving heat, the reactor coolant leaves the RPV through the set-out nozzle. To prevent the reactor coolant heat from being transferred to the surrounding structures a thermal insulation called reactor vessel insulation (RVI) encloses the RPV. Initially in conventional PWRs, there was no liquid flow path between the RPV and RVI as shown in Fig. 3(a). The concept of external reactor vessel cooling (ERVC) depicted in Fig. 3(a) has been of great interest to nuclear system designers since it provides an efficient solution of accident management issues that have been evolved after the Three Mile Island unit 2 incident [37]. The design concepts of retaining the corium (molten core) inside the RPV, during core melting accident, through external cooling of RPV is called ERVC. Core melting accident is one of the severe accidents in PWRs [38,39]. The space between the RPV and the RVI forms an annular chamber (see Fig. 3(a)) which can be flooded with cooling water in accidental conditions through passive valving which directly cool the reactor vessel in the event of core melting.

In normal operating conditions (non-accidental conditions), the space between the RPV and the RVI is filled with air forming an annular chamber of the air (see Fig. 3(b)). Under normal plant operating conditions the reactor cavity cooling system (VRC) is

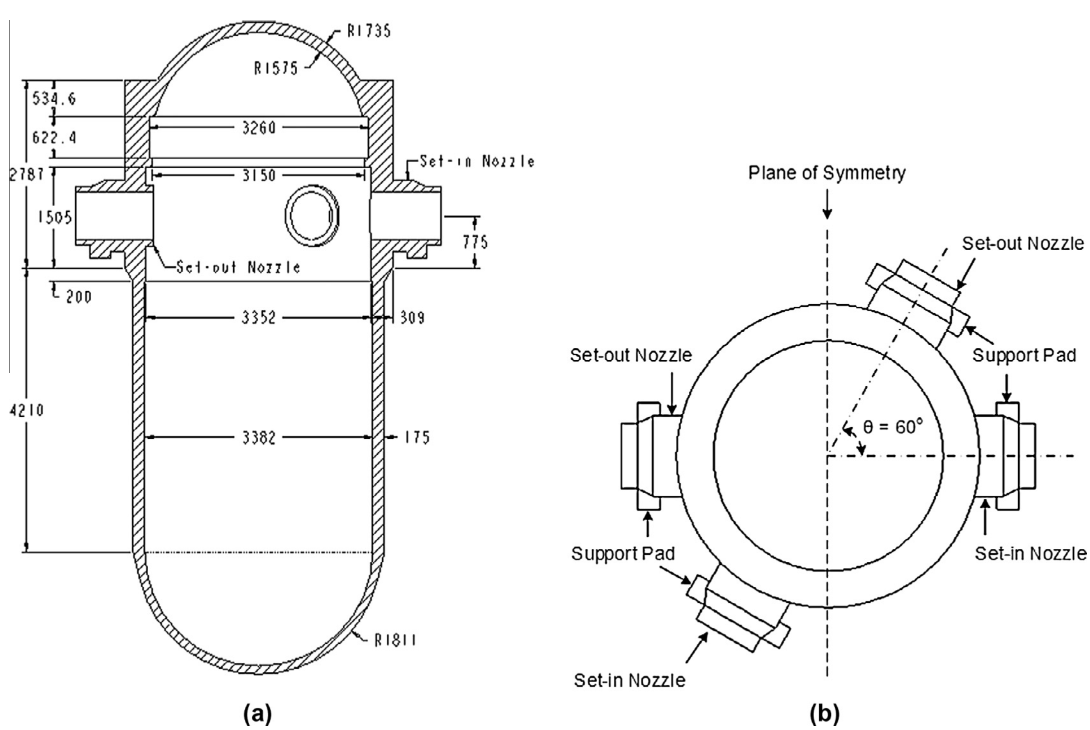

Fig. 1. Engineering drawing of the RPV (a) cut-section view, all dimensions in mm, (b) top view.