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Elastic and elastoplastic fracture analysis of a reactor pressure vessel under pressurized thermal shock loading

Xin Sun, Guozhong Chai*, Yumei Bao

Key Laboratory of E&M, Zhejiang University of Technology, Ministry of Education & Zhejiang Province, Hangzhou 310014, China

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ABSTRACT

Due to the limitation of the manufacturing process and improper operation, there may be defects near the inlet nozzles of reactor pressure vessel (RPV). Under pressurized thermal shock (PTS), the crack region is subjected to a high tensile stress, and the material toughness is gradually decreased in the cooling process. So it is necessary to evaluate the integrity of RPV in the thermo-mechanical coupling field. The 3-D finite element model is established for the beltline region around the inlet nozzles of a RPV. According to the calculated nil-ductility reference temperature, the elastic constitutive relation is firstly established for the fracture analysis. The stress intensity factors along the crack tips are compared with the material toughness. Using the XFEM, the crack propagation path is obtained to verify the predicted results. At relatively low reference temperatures, the ductility characteristics of the nuclear material can be exhibited. In this case, the ultimate bearing capacity of the structure is demonstrated by the elasto-plastic fracture analysis. Finally, the safe range of reference temperature is summed up to prevent the surface crack from running through the whole wall thickness of the vessel during the PTS transient.

1. Introduction

Reactor pressure vessel (RPV) is the main bearing part of nuclear power station, and its integrity assessment has attracted wide attention. In the event of loss of coolant accident (LOCA), the reactor core emergency system is started, and the cooling water is injected into the downcomer of RPV. Under the coupling action of the internal pressure and thermal shock, a high stress is generated on the inner wall of the vessel, especially near the inlet nozzles (Siegele et al., 1999). As the nuclear power plant runs close to the end of life, the fracture toughness of material is reduced by the fast neutron irradiation. As a consequence, the severe pressurized thermal shock (PTS) may cause the defects near the inner surface to penetrate the wall thickness (Qian et al., 2008), and lead to the leakage of nuclear material. Therefore, it is necessary to analyze the structural integrity of RPV under PTS to ensure the safety of the nuclear power plant in the life cycle.

A number of nuclear industry research institutes conducted a lot of researches on the structural integrity assessment of RPV under the condition of PTS. And then the relevant fracture evaluation criterions were established, such as RCCM (2007), ASME (1995), etc. In order to guarantee the accuracy of the evaluation criterions, the international cooperation projects, such as ICAS/RPV-PTS, IAEA CRP9 (IAEA, 2010), were organized to carry out the PTS analysis. By the deterministic fracture mechanics analysis (Jhung and Choi, 2009; Wang et al., 2007), many rules and regulations have been developed for use. The deterministic analysis is mostly based on the linear elastic fracture evaluation criterion. According to the internal pressure and temperature field of RPV under different working conditions, the stress distribution of the vessel wall is obtained by engineering estimation or finite element method. The crack tip stress intensity factor K_I is calculated to be compared with the crack growth fracture toughness K_{IC} and the crack arrest fracture toughness K_{Ia}. If $K_I > K_{IC}$, crack initiation occurs. If $K_I < K_{Ia}$, the crack growth will be interrupted. Based on the above conditions, the structural integrity of RPV can be evaluated. Deterministic fracture mechanics analysis takes crack size, load and strength as certain parameters. This evaluation method is convenient, and extensive application experience has been generated.

In the 1980's, NRC proposed federal regulation 10CFR50.61 (U.S. NRC, 1984). It is based on the nil-ductility transition reference temperature and becomes a screening criterion for PTS evaluation.

* Corresponding author.
E-mail addresses: 1049087232@qq.com (X. Sun), 371228520@qq.com (G. Chai).

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Next, NRC proposed a new project to reevaluate and establish the PTS rules combined with the advanced technology, including probability risk assessment, thermal hydraulic analysis, probabilistic fracture mechanics analysis. In probabilistic fracture mechanics analysis, a large number of uncertainty factors are involved, such as the size and distribution of crack, fracture toughness, neutron fluence, etc. Dickson and Simonen, 2002 discussed the failure probability of crack initiation and propagation obtained by the Marshall distribution and the improved PNNL distribution under the PTS condition. To a certain extent, the improved PNNL distribution can eliminate the unnecessary conservatism in the PTS analysis. Li and Modarres, 2005 proposed a framework which is used to distinguish the aleatory and epistemic uncertainties for PTS analysis. Based on the framework, a simple and reasonably accurate computational approach was developed for reactor vessel failure assessment. In recent years, probabilistic fracture mechanics method has been widely applied to the structural integrity assessment of RPV under the PTS condition. Many research institutions developed the relevant assessment codes, such as FAVOR (Williams et al., 2004), PASCAL (Kanto et al., 2010), etc. By using FAVOR code, Qian et al. (2014) performed a probabilistic analysis on single detected crack. Also, the weld type, size and its manufacturing process were considered. However, it was found that its failure probability was lower than the failure probability obtained by the improved PNNL distribution, which shows that the improved PNNL distribution is still conservative. Therefore, it is necessary to determine more accurate defect models.

So far, the current methods mentioned above have been confined to the prediction of crack growth, which involves the calculation of stress intensity factor and failure probability. However, there are few investigations on the process of damage initiation and evolution in thermo-mechanical coupling field. In order to demonstrate the ultimate bearing capacity of the nuclear material, the fracture behavior on one inlet nozzle of a RPV is revealed by the XFEM (Moës et al., 1999; Moës and Belytschko, 2002). Besides elastic fracture analysis for calculated reference temperature, elastic-plastic fracture analysis is adopted to simulate the ductile crack propagation at relatively low reference temperatures. According to the relationship between the critical internal pressure and reference temperature, the safe range of reference temperature is obtained to ensure the integrity of the RPV structure during the PTS transient. This study expands the research space of the dynamic crack propagation in 3D thermal stress coupled field, and provides a reference for solving the safety and reliability evaluation of the similar components with defects in practical engineering.

2. Analysis model

According to the structural characteristics of RPV, the crack initiation is most likely to occur in the vicinity of the water inlet nozzles. It is assumed that an inner surface crack is located on one of the nozzles. Fig. 1 shows the physical model of a real RPV containing the initial, boundary conditions, and the postulated crack. The inner radius of the RPV is 1994 mm, and the diameter of the two nozzles is 800 mm. The thicknesses of the base and cladding are 200 mm and 8 mm respectively. The shape of the crack in the base material is 1/4 circle, and the radius takes 20 mm, i.e. 1/10 of the thickness of the base wall. Fig. 2 shows the three-dimensional finite element model of the beltline region around the nozzles. The RPV model is established using the finite element program ABAQUS 6.13. The mesh near the nozzle area in the base material is refined, and the mesh in the cladding contains two layers. The current model has 81364 nodes and 49782 elements. In order to effectively deal with the thermo-mechanical coupling problem, the type of elements is chosen as DC3D8 in the thermal analysis, and C3D8R is adopted in the structural analysis.

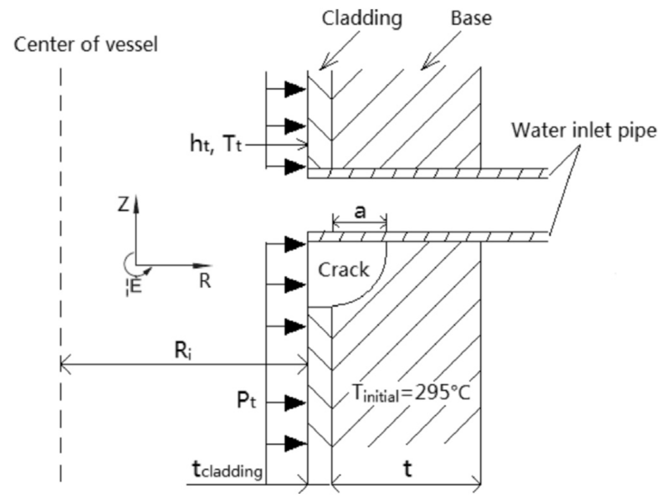

Fig. 1. The physical model of a real RPV with an inner surface crack of a/t = 0.1.

3. Material properties and reference transient

3.1. Material property

The base material is SA508 Cl 3 steel (Tanguy et al., 2005) and the cladding material is austenitic stainless steel (Karlsen et al., 2010). SA508 Cl 3 steel has high strength, good toughness and low sensitivity to irradiation embrittlement. So it is widely used for nuclear power plant. The cladding can protect the ferritic base material against corrosion, and it does not play a major role in the bearing capacity of the RPV structure due to its low strength. Table 1 shows the main physical and mechanical properties of the base metal and cladding. For the yield characteristics of the materials, the stress–strain relations are summarized in Table 2.

3.2. Reference PTS transient

Under the initial working condition, the temperature of the RPV model is assumed to be 295 °C, and the temperature outside the RPV is 20 °C. In case of loss of coolant accident, the cooling water is injected into the downcomer through the inlet nozzles, while the internal pressure may remain at a high level. Then the inner surface of the downcomer is subjected to the pressurized thermal shock. According to the PTS transient of the IAEA CRP9 (IAEA, 2010), Fig. 3 shows the histories of the internal pressure load and the cooling water temperature. Note that the repressure caused by the heating and vaporization of the cooling water suddenly rises to 16.8 MPa at 7185s, and then its fluctuation is very small. So the transient time period of the study is set to 0–7185s. Fig. 4 shows the transient distribution of the heat transfer coefficient between the cooling water and the inner surface of the vessel. In addition, the heat transfer coefficient between the air and the outer surface of the vessel is set to 20 W/(m^2·°C).

4. Methodology

4.1. Calculation principle

In heat conduction calculation, the following equation is introduced as