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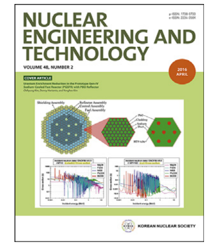

Original Article

Probabilistic Fracture Mechanics Analysis of Boling Water Reactor Vessel for Cool-Down and Low Temperature Over-Pressurization Transients

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ABSTRACT

The failure probabilities of the reactor pressure vessel (RPV) for low temperature over-pressurization (LTOP) and cool-down transients are calculated in this study. For the cool-down transient, a pressure–temperature limit curve is generated in accordance with Section XI, Appendix G of the American Society of Mechanical Engineers (ASME) code, from which safety margin factors are deliberately removed for the probabilistic fracture mechanics analysis. Then, sensitivity analyses are conducted to understand the effects of some input parameters. For the LTOP transient, the failure of the RPV mostly occurs during the period of the abrupt pressure rise. For the cool-down transient, the decrease of the fracture toughness with temperature and time plays a main role in RPV failure at the end of the cool-down process. As expected, the failure probability increases with increasing fluence, Cu and Ni contents, and initial reference temperature-nil ductility transition (RT_{NDT}). The effect of warm prestressing on the vessel failure probability for LTOP is not significant because most of the failures happen before the stress intensity factor reaches the peak value while its effect reduces the failure probability by more than one order of magnitude for the cool-down transient.

1. Introduction

The reactor pressure vessel which encloses fuel assemblies under highly pressurized coolant is the most important component in a nuclear power plant. Therefore, it is important to ensure that brittle fracture of the vessel does not occur during any condition to which the vessel may be subjected over its service lifetime.

In order to evaluate the integrity of the reactor pressure vessel, either a deterministic or probabilistic fracture mechanics (PFM) approach can be used. The deterministic fracture mechanics method, which has been more commonly used, employs the concept of a safety factor that envelops all kinds of uncertainties related to operating loadings, material properties, and damage mechanisms. It seeks a conservative evaluation by assuming the worst and bounding case. By

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Table 1 – Thermal properties.

Coefficient of heat transfer ($W/m^2/K$)	1,817
Poisson's ratio	0.3
Density (kg/m^3)	7,600
Thermal conductivity (Wm/K)	54.60 at 20 °C, 45.80 at 300 °C
Specific heat (J/kg/K)	488.722 at 20 °C, 568.520 at 300 °C
Thermal diffusivity (m^2/s)	14.70E–6 at 20 °C, 10.60E–6 at 300 °C
Thermal expansion coefficient (/K)	1.090E–05 at 20 °C, 1.490E–05 at 300 °C

contrast, the PFM method can directly treat the uncertainties of the main parameters and provide a more realistic result with the use of best-estimate data. Furthermore, the probabilistic method is useful to understand the effect of important parameters on the failure probability by conducting various sensitivity analyses. The probabilistic assessment has become more important recently.

For a pressurized thermal shock (PTS) event, one of the possible major challenges to the integrity of a reactor pressure vessel, many probabilistic assessments have been conducted. In the USA, screening criteria for PTS were determined based on the results of the PFM analyses [1,2]. From 2009 to 2011, PFM round robin analyses were performed amongst Asian countries to establish reliable procedures to evaluate the fracture probability of the reactor pressure vessel during PTS events [3,4]. Qian and Niffenegger [5] reviewed several PFM computer codes and discussed the effects of warm prestressing (WPS) and fracture toughness on the integrity of the reactor pressure vessel subjected to PTS. In addition, Qian et al [6] and Qian and Niffenegger [7] evaluated failure probabilities of the reactor pressure vessel by considering real crack distribution data, two different PTS transients, and different toughness curves.

For operating conditions other than PTS, however, fewer PFM assessments have been done to evaluate the failure probability of the reactor pressure vessel. Huang et al [8] performed PFM analyses for boiling water reactor (BWR) pressure vessels subjected to a low temperature over-pressurization (LTOP) event. Chou and Huang [9] evaluated the failure probabilities of a BWR pressure vessel under normal cool-down transients by considering the revision of the American Society of Mechanical Engineers (ASME) Section XI, Appendix G, which allows the use of the K_{Ic} curve instead of the K_{Ia} curve for generating pressure–temperature limit curves. However, further sensitivity studies are required to understand the effects of different input parameters on the vessel failure probabilities under the LTOP or cool-down transients.

Table 2 – Mechanical material properties.

Average of initial RT_{NDT} (°C)	–30 for weld, 0 for base
Standard deviation of initial RT_{NDT} (°C)	10
Formula of ΔRT_{NDT}	Reg. Guide 1.99
Standard deviation of ΔRT_{NDT} (°C)	0.0
Average of Cu content (wt%)	0.2
Standard deviation of Cu content (wt%)	0.01
Average of Ni content (wt%)	1.0
Standard deviation of Ni content (wt%)	0.02
K_{Ic} (ORNL average curve)	Standard deviation is 15% of average
K_{Ia} (ORNL average curve)	Standard deviation is 10% of average
Flow stress (MPa)	551.6
Young's modulus (MPa)	2.04E5 at 20 °C, 1.85E5 at 300 °C
Yield strength (MPa)	489 at 20 °C, 423 at 300 °C

ORNL, Oak Ridge National Laboratory.

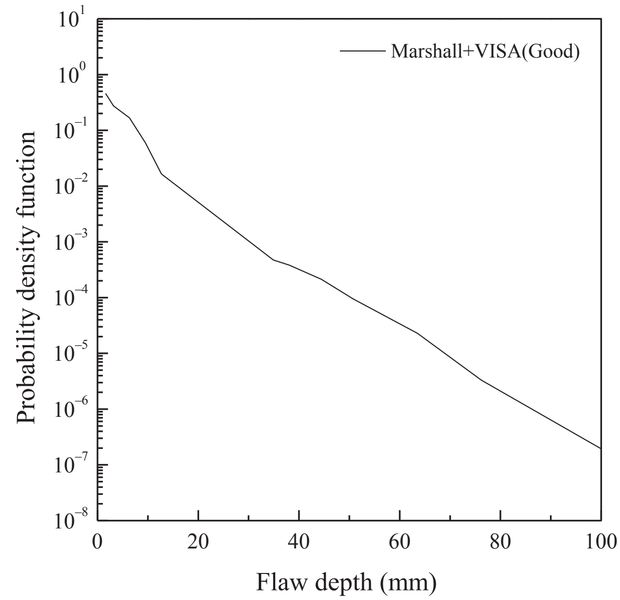

Fig. 1 – Flaw distribution and size for VISA-II model.

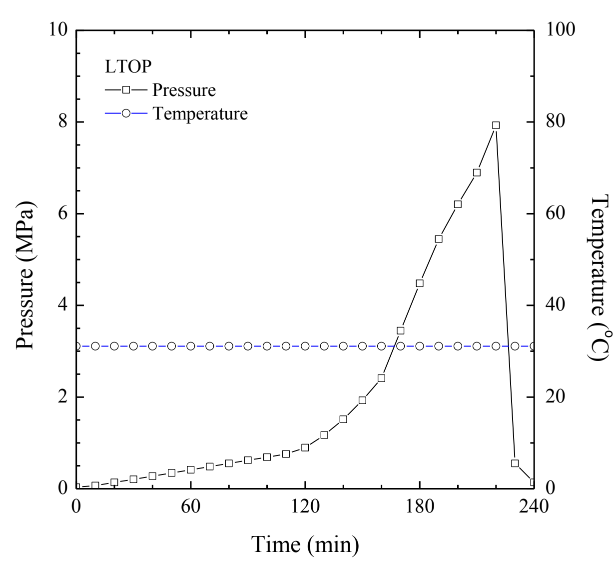

Fig. 2 – Pressure and temperature histories of low temperature over-pressurization.