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Probabilistic fracture analysis for boiling water reactor pressure vessels subjected to low temperature over-pressure event

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ABSTRACT

This paper performs the probabilistic fracture analyses for boiling water reactor pressure vessels subjected to the low temperature over-pressure event. Using the FAVOR code, which was developed by the Oak Ridge National Laboratory in America, a conservative model with the plant specific data of domestic BWRs in Taiwan to the USNRC requirements is built. A hypothetical transient of low temperature over-pressure event which could severely challenge the integrity of reactor pressure vessel of BWR is assumed as the loading condition for the probabilistic fracture mechanics analyses. The effects of performing inservice inspection on the probability of failure are also simulated. The computed low probability of failure indicates that the analyzed reactor pressure vessel can maintain sufficient reliability even without performing any inservice inspections for the circumferential welds. It also indicates that extensive inservice inspections on shell welds cannot promote the compensating level of safety significantly. Present results can be regarded as the risk incremental factors compared with the safety regulation requirements on reactor pressure vessel degradation. The developed probabilistic fracture mechanics model is helpful for the subsequent BWR regulation in Taiwan.

1. Introduction

To ensure the nuclear safety and structural integrity of components of nuclear power plant, the inservice inspections (ISIs) should be performed periodically on the welds of pressure retaining components such as reactor pressure vessels (RPVs), piping, nozzles and some other important components which are specified definitely in the ASME Boiler and Pressure Vessel Code, Section XI. For RPV, the most critical pressure boundary component, the ISI requirement of the ASME Code since 1989 has been amended to essentially 100% of volumetric examination on all shell welds due to the radiation embrittlement, stress corrosion cracking (SCC) and fatigue. Subsequently it was also employed for the regulation of reactor pressure vessels by the United States Nuclear Regulatory Commission (USNRC) into the Code of Federal Regulation 10CFR 50.55a.

However, the environmental and operational impact on structural integrity of boiling water reactors (BWRs) is not as severe as the pressurized water reactors (PWRs). Thus the failure probability of shell welds within the beltline region of BWR RPV can be expected to be lower than that of PWR, especially for the circumferential shell welds which undergo the axial stress around half of the hoop stress imposed on the axial welds. The results of probabilistic fracture mechanics (PFMs) analyses conducted by the BWR Vessel and Internal Project (BWRVIP, 1995), USNRC (1998) and Wichman et al. (1999) have demonstrated that the fracture occurrence of beltline circumferential welds is indeed orders of magnitude less than that of beltline vertical welds, and accordingly concluded that it is not necessary to perform the ISI for the circumferential welds to maintain the required safety margins. Accordingly, the USNRC further issued Generic Letter 98-05 (1998) which permitted the BWR licensees to request permanent relief from ISI requirements of circumferential welds. Over the past decade, nearly half of BWRs in the United States have already been approved for the relief request thus significantly reduced the associated substantial ISI cost and person-rem exposure. By contrast, the ISI for all RPV shell welds of BWRs in Taiwan so far are still required to be performed every 10 yr to the ASME Code with "essentially 100%" examination. In other words, at least 90% of volumetric examination on the circumferential welds still has to be performed because of the merely allowable 10% reduction from obstructions by interior components defined in 10CFR 50.55a.

Based upon the PFM analysis results, the USNRC concluded that the failure probability of RPV circumferential welds of limiting BWRs was sufficiently low (8.2×10^{-8}/yr at most) and finally accepted ISI relief request from utilities, but the axial welds still have to be inspected due to the relatively higher failure probability (4.4×10^{-4}/yr) (USNRC, 1998). The results were computed by using the Fracture Analysis of Vessels-Oak Ridge (FAVOR) code developed by the Oak Ridge National Laboratory (ORNL). In the

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paper, the PFM analyses for BWR RPV of Chinshan nuclear power plant in Taiwan are performed by using the same version of FAVOR code from ORNL. Very conservative assumptions consistent with the early work done by the USNRC were first taken into account in this work, including the numbers and size distributions of inner surface breaking flaws, failure criterion and the transient of low temperature over-pressure (LTOP) event. By introducing the plant specific data regarding the RPVs of Chinshan nuclear power plant, the PFM models can be built separately for the axial and circumferential shell welds in the beltline region. Further, the effects of ISI on the structural integrity are also studied here. Present work could be of importance to the basis for the safety evaluation and regulation on RPV of our domestic BWRs.

2. Analysis method

Based on the risk-informed concept, the PFM approaches for safety evaluation of RPV have been in development for over 20 yr such as VISA-II, VIPER and FAVOR (Dickson et al., 2004). The results are all computed by the Monte Carlo simulations considering every random variable which affects the toughness of RPV welds. Probability of failure (POF) is simply the total number of failures divided by the total number of simulations.

In this study, the failure of RPV is conservatively defined as a crack with the stress intensity factor greater than the fracture toughness ($K_I > K_{IC}$), and ignore the potential crack arrest due to mitigation of radiation embrittlement among deeper wall region. For consistency with original analyses by the USNRC, the manufacturing flaws were assumed to be all surface breaking flaws without the existence of any embedded flaws. A very conservative surface breaking flaw distribution, the PVRUF-Exponential best estimate distribution was concerned. Its complementary cumulative distribution function of the crack depth, s, can be represented by the following equation (USNRC, 1998):

$$\Pr\{S > s\} = 1 - 0.0608e^{-\frac{s-4}{3.8298}} \quad s \geqslant 4\ \text{mm} \tag{1}$$

The above equation describes the flaw distribution of the crack depth greater than 4 mm and its entire probability is 0.0608. Flaws with crack depth less than 4 mm were all assumed to be 4 mm uniformly with the probability of 0.9392. The flaw density of PVRUF-Exponential best estimate distribution is 108 flaws per vessel (USNRC, 1998). Fig. 1 shows the numbers of flaws against the percentage of the vessel wall thickness with the geometry for the

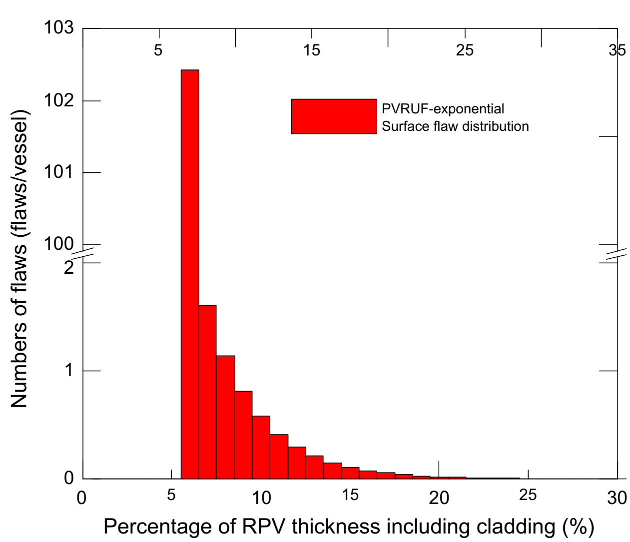

Fig. 1. Numbers of surface breaking flaws against the RPV wall thickness.

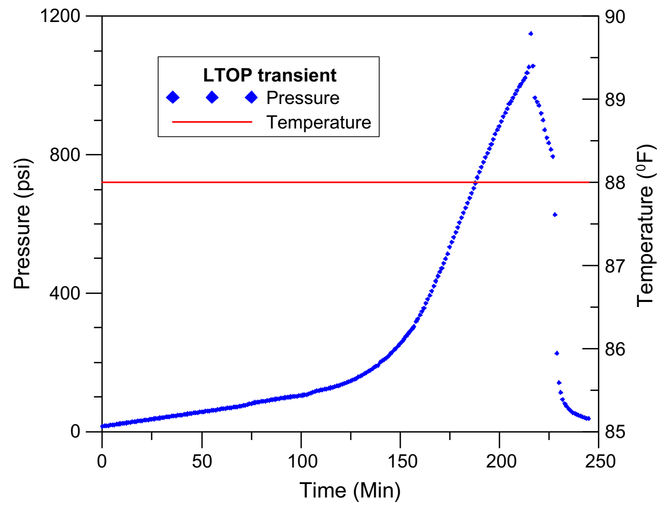

Fig. 2. The hypothetical transient of LTOP event.

RPV of Chinshan nuclear power plant. All flaws were conservatively assigned with the aspect ratio of 10. Although the Marshall flaw distribution used by BWRVIP additionally considered the effects of SCC crack growth on welds during operation, it has been shown that the PVRUF-Exponential distribution is still more conservative to calculate the POF of RPV shell welds (USNRC, 1998).

The input of loading condition is a hypothetical transient of LTOP event for RPV, which is a beyond design basis event with a constant temperature of 88 °F and pressure of 1150 psi (USNRC, 1998). The USNRC concluded that the LTOP event severely challenges the BWR RPV integrity and should be prevented strictly, even though it has never been happened in the United States. The precursors which may induce the LTOP event are inadvertent injection, condensate injection, control rod drive (CRD) injection and loss of reactor water cleanup (RWCU). It can be resulted from a series of operator errors during cold shutdown and non-recovery failure with an occurrence frequency of 1×10^{-3}/yr (USNRC, 1998). Compare to the analyses of USNRC, The pressure and temperature during LTOP in the study are real distribution and shown in Fig. 2, which are input in the load file then read by the pre-processer "FAVLOAD" of FAVOR code. Therefore the distributions and histories of temperature, stress and stress intensity factor through the wall thickness for each time step can be computed and then output to the processer "FAVPFM". On the other hand, the plant specific data about neutron fluence variation, material properties and chemical compositions also have to be input for the analyses. The consequent results represent the conditional probability of failure of RPV under LTOP, and should be multiplied by the occurrence frequency of 1×10^{-3}/yr to estimate the POF.

3. Plant specific data

Two GE BWR/4 reactors are currently operating in the Chinshan nuclear power plant in Taiwan. They began commercial operations in 1978 and 1979 respectively. Generally speaking, the factors affect the irradiative deterioration on fracture toughness of metals including neutron fluence intensity, initial reference temperature of nil-ductility transition (RT_{NDT}), and chemical composition. The decisive elements of chemical composition are Cooper (Cu) and Nickel (Ni) contents which correspond to the chemistry factor (CF) to assess the radiation embrittlement. The Regulatory Guide 1.99 (1988) addresses the radiation embrittlement as the following equation: