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Reactor pressure vessel integrity assessment by probabilistic fracture mechanics – A plant specific analysis

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A B S T R A C T

The probabilistic fracture mechanics method is widely adopted in nuclear power plant industry, especially for the structural integrity assessment of reactor pressure vessels. In this work, the plant specific analyses of Taiwan's boiling water reactors, BWR/4 and BWR/6, under the design transients were performed by the FAVOR (Fracture Analysis of Vessel – Oak Ridge) code. The difference of reactor pressure vessel geometry, alloying elements, neutron fluence, and loading conditions were taken into account in the probabilistic fracture mechanics analyses. The failure probabilities of the axial welds at 64 effective full-power year (EFPY) for the analyzed BWR/4 and BWR/6 are less than 2.8×10^{-10}/year and 1.5×10^{-4}/year, respectively. Furthermore, the circumferential welds present much smaller failure probabilities. The results of this work are useful for the subsequent aging management of the operating reactor pressure vessels.

1. Introduction

The safety of nuclear power plant is always a key issue of nuclear industry. Especially for the plant which has operated more than 20 or 30 years. Because of the material of reactor pressure vessel unavoidably degrades with the neutron irradiation, and the failure probability significantly increases with the operation time.

The phenomenon of material degradation is called radiation embrittlement. Potapovs and Hawthorne [1] first found that the small amount of copper and phosphor can enhance the radiation embrittlement of the reactor pressure vessel materials. In the 1980s, many theoretical studies, micro-structural examinations, and mechanical testing program have been performed to understand the complex process [2]. Several studies on the reactor pressure vessel embrittlement were published by the authors from Unite States [3], Unite Kingdom [4], German [5], Russian [6], and French [7]. The related studies were reviewed by Odette and Lucas [8], which was funded by the Electric Power Research Institute (EPRI). The results show that the radiation embrittlement strongly depends on the neutron flux/fluence, temperature, and alloying elements, such as copper (Cu), nickel (Ni), phosphor (P), and manganese (Mn). Odette and Lucas [9] studied the influence of fast neutron fluence on the radiation embrittlement. They summarized that fast neutron irradiates reactor pressure vessel and degrades the fracture toughness of reactor pressure vessel (RPV) materials, especially the materials in the beltline region of RPV.

To evaluate the structural integrity of reactor pressure vessel, both deterministic and probabilistic assessments were adopted in the past decades. In order to provide adequate safety, the deterministic assessment based on defense-in-depth concept may get over-conservative results. Thus, there is an increasing trend in application of probabilistic fracture mechanics (PFM) for evaluating the structural integrity of reactor pressure vessels in recent years [10–12]. Many computer codes, such as FAVOR, OCA-P, VISA-II and PRAISE were developed to perform the fracture probabilistic analysis based on the Monte Carlo technique. Furthermore, the Fracture Analysis of Vessel – Oak Ridge (FAVOR) [13], which is adopted in this study, can be used to perform risk-informed probabilistic analysis of RPV beltline region under various transients, chemistry concentrations and fast neutron fluences.

In Taiwan, some boiling water reactors started commercial operations in 1970s–1980s. To ensure the structural integrity and the operational stability of nuclear power plants, it is necessary to evaluate the reliability of important plant components, structures, and systems [14]. In this study, the structural integrity of RPV shell welds of Taiwan's BWR/4 and BWR/6 subjected to design transients

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were evaluated by FAVOR code. The plant specific parameters of the RPV shell welds in the beltline region, such as geometries, neutron fluence, chemistry concentration, and design transients were considered and discussed. The results of this work can be used to evaluate the structural integrity of the RPV beltline region, and provide the aging assessment of reactor pressure vessels.

2. Probabilistic fracture mechanics

The deterministic approaches have been mainly employed to design the components of nuclear power plant in the early 1970s. The engineers used safety factor to envelop all of the uncertainties, such as material properties, operation conditions, geometry tolerance, and crack growth mechanisms. The deterministic approaches always result in over-conservative designs. Thus, the practical life of almost of the nuclear power plant is longer than its 40 years design life.

On the contrary, lots of statistic data and Monte Carlo technique were involved in probabilistic approaches to estimate the uncertainties. This risk-inform evaluation can be used to perform sensitivity studies with different inspection policies/arrangements, and to evaluate the structural integrity of nuclear components. Therefore, the probabilistic approach has the ability to handle the issues of aging management. Moreover, by evaluating the over-conservative margin from deterministic approach, probabilistic approach widely adopted in the life extension assessment.

2.1. FAVOR codes

FAVOR, a probabilistic fracture mechanics analysis code, is composed of three main modules: (1) FAVLoad, (2) FAVPFM, and (3) FAVPost. The three modules are briefly described in the following: (1) The temperature, stress, and stress intensity factor through the RPV wall can be determined according to the geometries, thermal-elastic material properties, weld residual stress, and the thermal–hydraulic loadings defined in the final safety analysis report (FSAR). (2) The embrittlement data, corresponding to the RPV alloying elements and the fast neutron fluence distribution in the beltline region, are defined in the second module. Furthermore, the depth, density, and geometry of surface flaws, plate flaws and weld flaws based on the NRC final safety evaluation report were also included in the Monte Carlo probabilistic fracture analysis. Therefore, the conditional probability of initiation (CPI) and conditional probability of failure (CPF) can be obtained by FAVPFM module. (3) In the FAVPost module, the initiation probability or failure probability per year can be calculated by multiplying CPI or CPF and its event frequency, respectively.

2.2. Embrittlement model

The reference nil-ductility transition temperature can be used to determine the level of radiation embrittlement. The initial/unirradiated reference nil-ductility transition temperature ($RT_{\mathrm{NDT(U)}}$) of vessel material can be measured based on the ASME NB-2331 [15]. The fracture toughness of RPV steel, especially in the beltline region, degrades when exposed to the fast neutron. The irradiated reference nil-ductility transition temperature (RT_{NDT}) increases at the same time. The value of RT_{NDT} can be obtained by eq. (1) [16]:

$$RT_{\mathrm{NDT}} = RT_{\mathrm{NDT(U)}} + \Delta RT_{\mathrm{NDT}} + M \tag{1}$$

where M is the margin term. The shift in reference nil-ductility transition temperature is defined as following [17]:

$$\Delta RT_{\mathrm{NDT}} = CF \times f^{(0.28-0.1\times \log f)} \tag{2}$$

where f is the best estimate fast neutron fluence, in the units of 10^{19} n/cm^2 (>1 MeV) at the clad–base-metal interface on the inside surface of the vessel. CF is the chemistry factor, which is a function of copper and nickel content. The chemistry factor for welds and base metal can be found in Tables 1 and 2 of Reg. Guide 1.99, Rev. 2, respectively. In these tables, linear interpolation is permitted.

2.3. Stress intensity factor

Once the through-wall stress distribution at each transient step has being calculated by the 1-D finite element model, the FAVLoad module can generate the stress intensity factor by linear superposition technique. For the 3-D semi-elliptical surface flaw, the stress normal to the crack plane at radial position a' can be roughly defined by a three order polynomial, as shown in the following [13]:

$$\sigma(a') = C_0 + C_1\left(\frac{a'}{a}\right) + C_2\left(\frac{a'}{a}\right)^2 + C_3\left(\frac{a'}{a}\right)^3 \tag{3}$$

where a is the depth of crack and C_0, C_1, C_2 and C_3 are estimated by least square regression analysis for the stress distribution. According to the linear superposition technique, the stress intensity factor can be obtained by eq. (4) [13]:

$$K_{\mathrm{I}}(a) = \sum_{j=0}^{3} K_{\mathrm{I}j}(a) = \sum_{j=0}^{3} C_j\sqrt{\pi a}K_j^*(a) = \sum_{j=0}^{3} C_j\sqrt{\pi a}\,\frac{K_{\mathrm{I}j}^*(a)}{C_j'\sqrt{\pi a}}, \quad C_j' = 1 \tag{4}$$

where the dimensionless quantity $K_j^*(a)$ is the influence coefficient and equals to $K_{\mathrm{I}j}^*(a)/C_j'\sqrt{\pi a}$..

3. Plant specific parameters

In this plant specific study, the structural integrity of two BWR/4 and two BWR/6 units was evaluated by PFM analysis. The key parameters such as geometry, alloying elements, neutron fluence and design loading conditions are described in the following.

3.1. Geometry

Both of the BWR/4 and BWR/6 reactor pressure vessels are made of ASME SA533 Type B carbon steel. The inside surface of RPV is clad in corrosion resistant material, 309 stainless steel. The inner radius of the BWR/4 RPV is 2565.40 mm, and the thickness of RPV and cladding are 130.05 mm and 3.18 mm, respectively. For BWR/6, the inner radius is 2768.60 mm, and the thickness of RPV and cladding are 137.41 mm and 5.08 mm, respectively.

The upper and lower boundaries of the beltline region are defined as the top and bottom of active fuel, respectively. The radiation in this region is significantly larger than other regions, so the beltline region is always the most concern part in the aging and inspection issues.

For BRW/4 reactors, one circumferential weld (W-1102-03) and six axial welds (W-1001-07, W-1001-08, W-1001-09, W-1001-10, W-1002-11, and W-1001-12) are located in the beltline region. For BWR/6 reactors, two circumferential welds (AB and AC) are close to the beltline region, and three axial welds (BE, BG, and BF) are located in the beltline region. The detail locations of axial and circumferential welds are shown in Table 1.