Structural Engineering and Mechanics, Vol. 37, No. 3 (2011) 257-265
DOI: http://dx.doi.org/10.12989/sem.2011.37.3.257

Comparison of vessel failure probabilities during PTS for Korean nuclear power plants

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(Received September 25, 2009, Accepted October 12, 2010)

Abstract. Plant-specific analyses of 5 types of domestic reactors in Korea are performed to assure the structural integrity of the reactor pressure vessel (RPV) during transients which are expected to initiate pressurized thermal shock (PTS) events. The failure probability of the RPV due to PTS is obtained by performing probabilistic fracture mechanics analysis. The through-wall cracking frequency is calculated and compared to the acceptance criterion. Considering the fluence at the end of life expected by surveillance test, the sufficient safety margin is expected for the structural integrity of all reactor pressure vessels except for the oldest one during the pressurized thermal shock events. If the flaw with aspect ratio of 1/12 is considered to eliminate the conservatism, the acceptance criteria is not exceeded for all plants until the fluence level of 8×10^{19} n/cm^2, generating sufficient margin beyond the design life.

Keywords: pressurized thermal shock; reactor pressure vessel; structural integrity; stress intensity factor; failure probability; through-wall cracking.

1. Introduction

A pressurized thermal shock involves a transient in which severe overcooling causes a thermal shock to the vessel, while the pressure is either maintained or the system is repressurized during the transient. The thermal stress due to the rapid cooling of the vessel walls, in combination with the pressure stress from either maintaining system pressure or repressurization of the system, results in large tensile stresses on the inside surface of the vessel. At a temperature below the reference nil ductility transition temperature of the material, the decrease in fracture toughness, in combination with the pressure and thermal stresses, could cause a relatively small crack to propagate through the vessel wall. Therefore, it is necessary to evaluate the structural integrity of a reactor pressure vessel under a PTS event (Ryu 2009).

For the quantitative evaluation of the vessel failure risk associated with PTS, the probabilistic fracture mechanics (PFM) analysis technique has been widely used (Baek *et al.* 2009, Dickson 1994). The PFM technique basically checks whether hypothetical flaws on the wall propagate through the vessel wall by comparing the applied stress intensity factor (crack driving force) with the fracture toughness (materials resistance to fracture) during the PTS events. Therefore a

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probabilistic fracture mechanics code called R-PIE (Reactor - Probabilistic Integrity Evaluation) is developed implementing the advanced technologies and new capabilities (Jhung 2008).

In this study, plant-specific analyses of 5 types of domestic nuclear power plants in Korea are performed to assure the structural integrity of the reactor vessel during transients which are expected to initiate PTS events. The temperature distributions are calculated and the stress analyses due to these temperature distributions and internal pressure are performed using the R-PIE code. Stress variations along the vessel wall are used to get the stress intensity factors and temperature distributions along the vessel wall are used to get the fracture toughness. The stress intensity factor and fracture toughness are compared to determine the propagation of the crack causing the failure of the vessel, which is used to calculate the probability of the vessel failure. The through-wall cracking frequency is calculated and compared to the acceptance criterion.

2. Analysis

2.1 RT_{PTS}

USNRC introduced the concept of RT_{PTS} the reference temperature of nil-ductility transition, RT_{NDT}, evaluated for the end-of-life (EOL) fluence for each of the beltline materials, and defined the PTS screening criterion as 270°F for plates, forgings, and axial weld materials, and 300°F for circumferential weld materials in 10CFR50.61 "Fracture toughness requirements for protection against pressurized thermal shock events" (1996). Also, for each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion using the EOL fluence, the licensee is required to implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion. The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated approval of detailed plant-specific analyses, submitted to demonstrate acceptable risk with RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques.

The reference temperature of nil-ductility transition RT_{NDT} is given by the following expression according to the US NRC Regulatory Guide 1.99, Rev.2 (1988)

$$RT_{NDT} = RT_{NDT0} + \Delta RT_{NDT} + M \tag{1}$$

where RT_{NDT0} is the mean value for the initial (unirradiated) value of RT_{NDT} for the RPV region in which the flaw resides. M is the margin which considers the uncertainties of RT_{NDT0} and ΔRT_{NDT}. The irradiation shift formula ΔRT_{NDT} defined as Eq. (2) is the increase in RT_{NDT} due to irradiation-induced embrittlement, which is a function of the copper and nickel content and neutron fluences.

$$\Delta RT_{NDT} = (\mathrm{CF})\, f^{(0.28 - 0.10 \log f)} \tag{2}$$

where CF is the chemistry factor, a function of copper and nickel content and f is the neutron fluence at any depth in the vessel wall (10^{19} n/cm^2, E > 1 MeV) determined as

$$f = f_{surf} \exp^{(-0.24x)} \tag{3}$$

Table 1 RT_{PTS} from surveillance test data

Unit (EFPY)	Method	Cu (wt%)	Ni (wt%)	CF (°F)	Fluence* (surface)	*ff***	RT_{NDT0} (°F)	M (°F)	ΔRT_{NDT} (°F)	RT_{PTS} (°F)
P1 (32)	Weld	0.042	0.647	56.80	3.7548	1.342	−40	56	76.25	92.25
P2 (32)	Weld (S/C)	0.03	0.1225	50.28	4.203	1.367	−20	28	68.72	76.72
P3 (32)	Plate(S/C)-ST	0.0537	0.5575	57.27	3.699	1.339	−20	34	76.69	90.69
P4 (24)	Inter. Lower Shell Weld	0.29	0.68	190.96	3.059	1.295	−10	56	247.36	293.36
P5 (32)	Lower Shell Product	0.04	0.82	26	1.673	1.142	−20	29.69	29.69	39.38

*10^{19} n/cm^2, E > 1.0MeV, ***ff* = Fluence factor = $f^{(0.28-0.10\log f)}$

where f_{surf} is the calculated value of the neutron fluence at the inner wetted surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface.

The RT_{PTS}s are calculated as shown in Table 1 using the surveillance test data. Plant 4 (P4) has a very small margin at the end of design life 24 EFPY but RT_{PTS}s of other plants are very low comparing with the screening criteria of 270°F for plates, forgings, and axial weld materials and 300°F for circumferential weld materials. Therefore the structural integrity of the reactor pressure vessel is maintained for the pressurized thermal shock events and it is not necessary to perform the plant specific analysis. But it was necessary to know how much margins are obtained for the domestic plants for the PTS rule making. Therefore in this study plant specific analysis are performed to generate vessel failure probabilities due to pressurized thermal shock. And the fluence level exceeding the acceptance criteria is estimated to guarantee the safety margin beyond the design life.

2.2 R-PIE code

A probabilistic fracture mechanics code called R-PIE (Reactor - Probabilistic Integrity Evaluation) is developed for the quantitative risk assessment of the RPV at the events of the pressurized thermal shock, which consists of two parts, such as the deterministic analysis and the probabilistic analysis (Jhung 2008). The R-PIE code is similar to previously developed VINTIN (Jang 2007) but contains user-friendly features being written in Visual Basic.

In the deterministic analysis part, the temperature profiles and the resulting thermal stress along the thickness of the reactor pressure vessel are calculated from the given thermal-hydraulic conditions. The distribution of stresses from other sources like pressure and residual stresses are separately calculated. The stress intensity factor, K from each stress components is calculated by the Raju-Newman method (Raju and Newman 1982) using the appropriate influence coefficients for the flaw shapes. Then, the stress intensity factor components calculated for the various stress components such as thermal stress, pressure stress, and residual stress are added to be the total applied stress intensity factor, K_I at the crack tip. This method can be readily applied to calculate the applied stress intensity factors in the base metal of the reactor pressure vessel. However, because of the lower thermal conductivity of stainless steels, the temperature profile in the cladding considerably deviates from that in base metal made of low alloy steels. Also, the thermal stress profiles in the cladding region deviates from that in the base metal which can be fitted as smooth third order polynomials. Special treatment scheme was developed and incorporated into the code to handle such steep deviations in the cladding region (Jang *et al*. 2003).

In the probabilistic analysis part, a variety of statistical parameters such as flaw size, neutron fluence, copper and nickel contents, and the reference temperature-nil ductility transition are simulated for each hypothetical reactor pressure vessel. From the temperature profile and the RT_{NDT}, the mean static fracture toughness K_{IC} and the mean arrest fracture toughness K_{IR} at the tip of the flaws are calculated using the equation derived from the lower-bound fracture toughness (ASME 2004).

Finally, using the mean values and the associated uncertainties, the fracture toughness values are simulated to be compared with the applied stress intensity factors at the tip of the flaws, K_I. If K_I is larger than K_{IC}, the flaw is assumed to initiate and grow a certain distance. Then, at the new flaw size, new values of RT_{NDT}, K_I and K_{IR} are simulated and compared. If K_I is smaller than K_{IR}, the flaw is considered to be arrested. Otherwise, the flaw size is increased again and the arrest check is repeated until the end of the transient. By repeating the above analysis millions of times, a statistically significant conditional probability of the vessel failure for the specific thermal hydraulic boundary condition is determined.

2.3 Transient

From the potential PTS initiating events, hundreds of PTS transient sequences were derived based on success or failure of component actuation and operator actions for P4. The potential PTS transient sequences were quantified and grouped based on similarities in thermal-hydraulic characteristics. In this study, the RETRAN-3D code (EPRI 1996) was used to calculate system pressure, coolant temperature near the vessel wall, and the heat transfer coefficient as a function of time for some of the representative transient sequences. Unlike the system safety analyses in which inputs were calibrated to be conservative in terms of core damage, thermal-hydraulic analyses for PTS transients should be best-estimate analyses (USNRC 1987). Therefore, special care was taken to use appropriate inputs for the thermal-hydraulic analyses.

There are no available transients generated for the other plants except P4 from the thermal hydraulic group. Therefore the same transients developed for P4 are used for other plants. Of all transients, it was found that the through wall cracking is almost due to the small break loss of coolant accident (SBLOCA) case (Jhung *et al.* 2009). Therefore in this study, SBLOCA only is considered to calculate the vessel failure probability.

The I001 transient was one of the sequences derived from SBLOCA at full power with frequency of 2.56×10^{-3}. As shown in Fig. 1, the temperature starts to decrease with cold emergency cooling water injection. System pressure decreases rapidly because the coolant flow rate through the break was greater than the charging and emergency cooling water flow rate. The final coolant temperature was about 90°F.

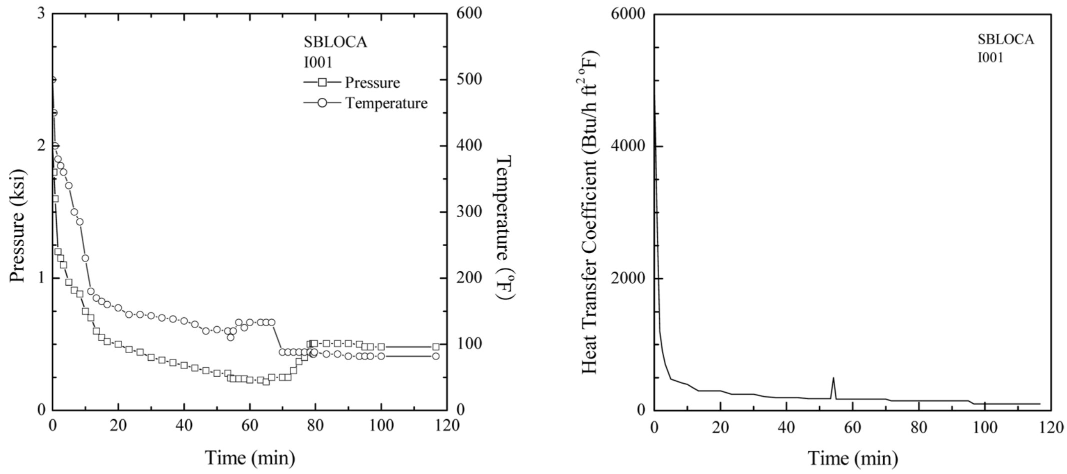

Fig. 1 Transient histories for SBLOCA

2.4 Flaw distribution

Marshall (UKAEA 1982) used the following equation to calculate the probability of crack with depth a.

$$P(a) = 4.06\exp(-4.06a) \tag{4}$$

Then the cumulative flaw density function describing the probability of crack existence larger than a is expressed as following equation.

$$f(a) = \int_0^a 4.06\exp(-4.06a)da \tag{5}$$

If integrated to whole flaw depth range, above equation results in exactly 1, indicating that Eq. (5) is associated with a single flaw.

Also, the probability of non-detection for pre-service inspection is defined as

$$B(a) = \varepsilon + (1 - \varepsilon)\exp(-\mu a) \tag{6}$$

where $\varepsilon = 0.005$ and $\mu = 2.88$ in^{-1}, and is valid for edge cracks and semi-elliptical cracks with a/l = 1/6. Therefore the flaw distribution and size after inspection can be calculated by incorporating Eq. (6) into Eq. (4). As the detected flaws are effectively removed from the population, the net effect is reducing the number of flaws as well as modifying the flaw distribution. After some rearrangement, the cumulative flaw distribution for Marshall with inspection can be expressed by following equation.

$$f(a) = \int_0^a [0.0346\exp(-4.06a) + 6.88\exp(-6.94a)]da \tag{7}$$

As before, if integrated to whole flaw depth range, Eq. (7) will result in exactly 1, but the associated number of flaw is 0.5863 instead of 1 because of the above mentioned reason. The flaw distribution and size considering inspection can be calculated as in Fig. 2.

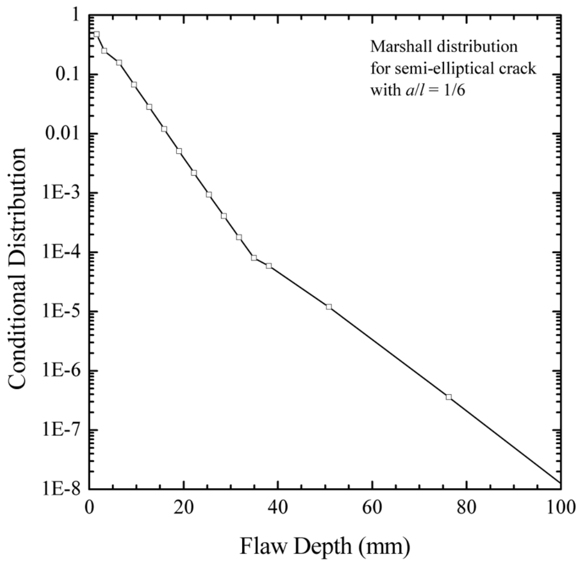

Fig. 2 Flaw distribution and size for Marshall model

Table 2 Plant information

Plant	P1	P2	P3	P4	P5
Output (MWe)	950	650	950	587	1000
Thickness (inch)	7.874	6.63	7.875	6.5	8.22
Clad thickness (inch)	0.197	0.125	0.125	0.125	0.16
Inner radius (inch)	78.4	66	78.5	66	82.015
Material	SA508, Cl.3	SA533B, Cl.1	SA533B, Cl.1	SA508, Cl.2	SA508, Cl.3

2.5 Plant-specific data

Five domestic nuclear power plants are considered in this study and they are shown in Table 2. P4 is the oldest nuclear power plant in Korea. P5 is the first Korean Standard Nuclear Power Plant. P1 is the Framatome type reactor. P3 and P2 are Westinghouse type reactors for 950 and 650 MWe, respectively.

K_{IC}, K_{IA} and ΔRT_{NDT} normal distributions are assumed to be truncated between $+3SD$ and $-3SD$ where SD is the standard deviation. The crack postulated is surface breaking crack with infinite through clad in the circumferential orientation.

3. Results and discussion

The temperature distributions are calculated and the stress analyses due to these temperature distributions and internal pressure are performed using the R-PIE code. Temperature and axial stress variations along the vessel wall are used to get the stress intensity factors. Also temperature distributions along the vessel wall are used to get the fracture toughness. The stress intensity factor

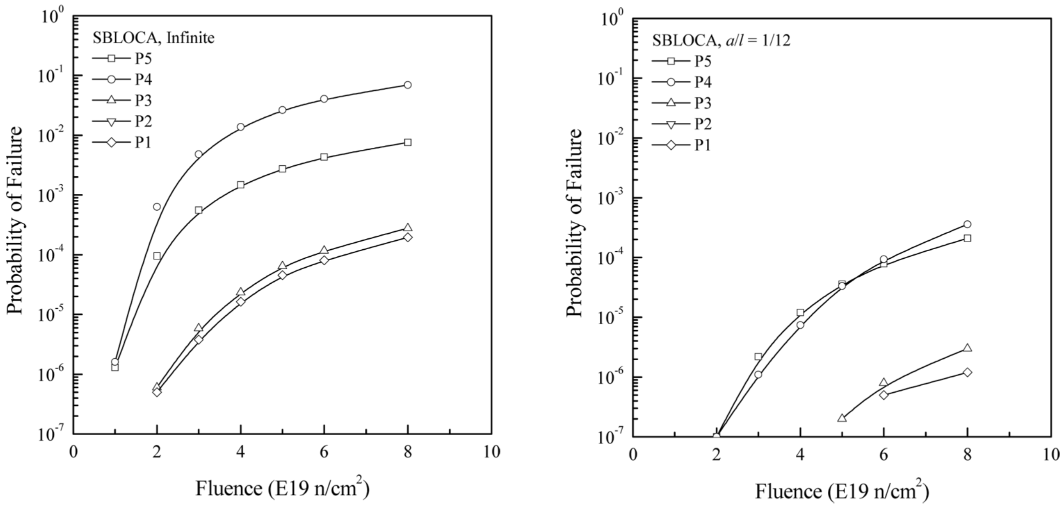

Fig. 3 Probability of vessel failure

and fracture toughness are compared to determine the propagation of the crack generating the failure of the vessel, which is used to calculate the probability of the vessel failure. The probability of vessel failure is shown in Fig. 3.

The event frequencies are coupled with the results of the fracture mechanics analysis to obtain an integrated frequency of vessel through-wall cracking (TWC) due to PTS. The sequence frequency and conditional through-wall crack penetration probability are multiplied to give the frequency of through-wall cracking for each initiator as a function of fluence. These are summed over all initiators to provide an integrated frequency of through-wall cracking, the acceptance criterion of which is 5×10^{-6} per reactor year (USNRC 1987).

By multiplying the sequence frequency and conditional through-wall crack penetration probability, the TWC frequencies are obtained as a function of fluence as shown in Fig. 4, which shows that acceptance criterion is exceeded from the fluence level of 4.48×10^{19} n/cm^2 for P5, 2.60×10^{19} n/cm^2 for P4, more than 8.0×10^{19} n/cm^2 for P3, P2 and P1.

According to the surveillance test, the fluence at the inner surface of the reactor pressure vessel expected at the end of life is shown in Table 3 and the safety margin can be calculated for the structural integrity of the reactor pressure vessel during the pressurized thermal shock events. Except for P4, sufficient safety margins are expected as shown in Table 3.

If the flaw with aspect ratio of 1/12 is used in the analysis for P4, the TWC frequency values are well below 5×10^{-6} at the limiting value of RT_{PTS}, which means that the acceptance criteria is not exceeded until the fluence level of 8×10^{19} n/cm^2. Therefore fluence margin of more than 4.94×10^{19} n/cm^2 is obtained by changing the infinite flaw to the flaw with aspect ratio of 1/12 (Table 3), verifying that sufficient margin beyond the design life is assured for the structural integrity of the reactor pressure vessel during the pressurized thermal shock events. This condition suggests that a higher value of RT_{PTS} may be justified if 5×10^{-6} is maintained as an acceptable limit for TWC frequency.

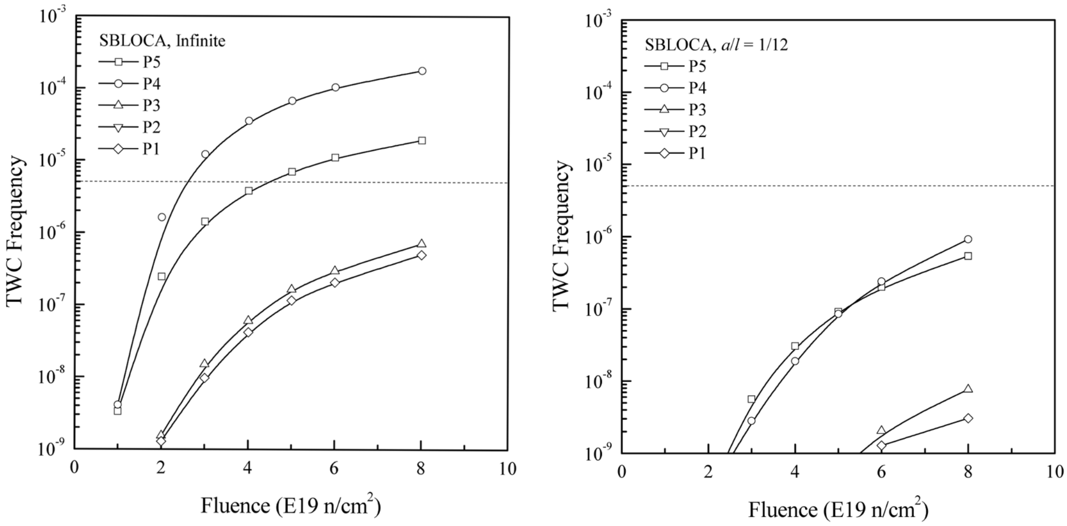

Fig. 4 Through-wall cracking frequency with respect to fluence

Table 3 Fluence level exceeding acceptance criteria of TWC frequency 5×10^{-6} and safety margin

Plant	Fluence (10^{19} n/cm^2)		Fluence at EOL (10^{19} n/cm^2)	Safety margin (%)*	
	Infinite	*a*/*l* = 1/12		Infinite	*a*/*l* = 1/12
P1	> 8.0	> 8.0	3.7548	> 113	> 113
P2	∞**	∞	4.203	∞	∞
P3	> 8.0	> 8.0	3.699	> 116	> 116
P4	2.60	> 8.0	3.059	–15	> 162
P5	4.48	> 8.0	1.673	168	> 378

*Safety margin (%) = (Allowable – Estimated)/Estimated × 100, **No failures.

4. Conclusions

Plant-specific analyses of 5 types of domestic reactors have been performed to measure how much margins it has for the structural integrity of the reactor pressure vessel under PTS of several transients. The vessel failure probabilities from the probabilistic fracture mechanics analyses are combined to the transient frequencies to generate the through-wall cracking frequencies.

The through-wall cracking frequency is compared to the acceptance criterion and it is found that the integrity of the reactor pressure vessel is maintained up to the fluence level of 4.48×10^{19} n/cm^2 for P5, 2.60×10^{19} n/cm^2 for P4, more than 8.0×10^{19} n/cm^2 for P3, P2 and P1. Considering the fluence expected at the end of life by the surveillance test, the sufficient safety margin is expected for the structural integrity of all reactor pressure vessels except for P4. If the flaw with aspect ratio of 1/12 is considered for P4, the acceptance criteria is not exceeded until the fluence level of 8×10^{19} n/cm^2, generating sufficient margin beyond the design life.

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