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Comparative analysis of pressure vessel integrity for various LOCA conditions

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Abstract

In this study, integrity analysis is performed for a classical four loop PWR pressure vessel fabricated from SA533B type ferritic steel. Pressure vessel behavior is analyzed by deterministic and probabilistic methods under transient conditions, which may cause pressurized thermal shock (PTS). In deterministic analysis, the change of material properties and the mechanical state of the vessel are analyzed against changes in coolant pressure and temperature. Probabilistic analysis is performed to obtain pressure vessel beltline region weld failure probabilities in transient conditions. Overall vessel failure probabilities are evaluated based on the results of deterministic analyses. Computer code VISA-II is utilized for the calculation of vessel failure probabilities. Among three cases considered in this study, a medium break loss of coolant accident induced by a 50 cm^2 break in the hot leg yields the highest vessel rupture probability. The maximum nil ductility temperature in all cases is still below the NRC PTS limit. © 2001 Elsevier Science B.V. All rights reserved.

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1. Introduction

The pressure vessel is the most crucial component of light water reactors since it contains fuel assemblies and reactor internals, and holds coolant at high temperature and pressure during operation. Therefore, it is designed and manufactured according to strict regulations. The recent efforts for the life extension of existing nuclear power plants also draw attention to the importance of vessel integrity [1–4].

Pressure vessels contain the primary side of the coolant loop. The integrity of the primary side pressure boundary is of prime importance under all operating conditions since a rupture or leakage results in the loss of coolant which can lead to a consequence of core meltdown. Furthermore, pressure vessels prevent the release of radioactive substances to the environment. Vessel integrity is of particular concern for pressurized water reactors since they are operated at higher pressures and neutron fluxes compared to those for boiling water reactors [5].

Reactor vessels are designed and manufactured according to the accepted standards with wide safety margins against rupture. Pressure vessel steels are chosen among the materials of high fracture toughness, high yield strength, and low irradiation embrittlement. Moreover, change in material properties under different operating conditions is taken into account in pressure vessel steel manufacturing [6].

There are two types of abnormal events which are accounted for in reactor vessel safety analyses; overcooling transients and cold pressurization. Overcooling transients can occur when the emergency core cooling system is activated due to a pipe break in the primary system. Cold pressurization happens when system pressure is increased too rapidly in connection with startup. A pressurized thermal shock (PTS) is such a transient

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that a thermal shock to the vessel is caused by a severe overcooling while either the system pressure is maintained or the system is repressurized during the transient. Large tensile stresses are caused by the thermal stress due to rapid cooling of vessel walls together with the pressure from either maintaining system pressure or repressurization of the system. Maximum stress is observed at the inner surface of the vessel. Relatively small cracks can easily propage through vessel wall under the effect of the pressure and thermal stresses due to the decrease in fracture toughness at temperatures below the nil ductility temperature.

Among American PWRs, 34 events were considered as significant events with regard to thermal shock. Most of the transients were mild and only four were considered to be serious. Two of these events are Rancho Seco transient in 1978 and Crystal River 3 transient in 1980. In both reactors, the malfunction of non-nuclear instrumentation caused erroneous signals, which led to start of the emergency core cooling system and rapid increase of coolant temperature causing an overcooling transient. The reactor vessel was not subjected to any damage in any of these cases.

Since there is no real failure data for nuclear reactor pressure vessels in order to carry out integrity analysis, experience from non-nuclear vessels is employed although major differences exist in manufacturing and design. Experience from nonnuclear vessels shows that the presence of cracks in material can lead to failure. These cracks may occur during manufacturing or reactor operation. Cracks can grow under mechanical, thermal, and corrosion related loads. Especially beltline region welds near the reactor core are of primary concern in fracture analysis. In Reactor Safety Study (WASH-1400), the probability of reactor vessel rupture was estimated to be 10^{-6} per year within an interval of 10^{-5}–10^{-7} [7,8].

In the present study, three overcooling transients were analyzed for a classical four loop PWR. Vessel integrity simulation analysis (VISA-II) code is employed for deterministic and probabilistic calculations in order to determine thermal and mechanical loads and to estimate the failure probability of reactor beltline region welds which can lead to vessel failure [9]. The rules and correlations used in VISA-II are still in accord with the current US and many other international regulatory practice [10].

2. Description of vessel and transient events

2.1. Reactor pressure vessel and materials

A classical four loop PWR vessel whose characteristics are represented in Table 1 is considered and modeled in this study. Two types of ferritic steels have been used to manufacture most of the light water reactor vessels; SA533B and SA508. In this study, SA533B steel which is commonly used in reactor vessel steels is chosen as base metal. This steel contains 0.117% Cu, 0.547% Ni, 0.547% Mo, 0.167% Mn, 0.236% Si, 0.074% Cr, 0.009% P, 0.014% S by weight as alloying elements and impurities [11,12]. Inner surface of the vessel is covered by a cladding made of AISI 308 stainless steel. Properties of base metal and cladding are given in Table 2. The vessel includes three circumferential and six longitudinal welds. Yield strengths of base metal SA533B and weld metal are 313 and 483 MPa, respectively. Ultimate tensile strengths of base and weld metals are 538 and 593 MPa, respectively.

Table 1
Characteristics of a PWR pressure vessel

Overall height	13 660 mm
Inside diameter	4394 mm
Wall thickness opposite core	215 mm
Dry weight of the pressure vessel	434 800 kg
Normal operating pressure	15.98 MPa
Design pressure	17.13 MPa
Normal operating inlet temperature	288°C
Normal operating outlet temperature	327°C

Table 2
Properties of vessel base metal and cladding material

	Base metal	Cladding
Modulus of elasticity, E (GPa)	195.5	186
Poisson's ratio, v	0.3	0.3
Thermal conductivity, k (W/m K)	39.4	16.8
Specific heat capacity, C_p (J/g K)	0.50	0.54
Thermal expansion coefficient, α ($\times 10^{-6}$ K^{-1})	12.4	17.6

2.2. Neutron fluence

Neutron fluence on the inner vessel wall is in the range 1–10 $\times 10^{19}$ neutrons/cm^2. Fluence attenuation through the vessel wall is determined by the relation [13]

$$\phi = \phi_0 \times \exp(-0.0945x), \tag{1}$$

where x is the depth in centimeters from the inner surface. ϕ_0 is taken as 4×10^{19} neutrons/cm^2 in calculations.

2.3. Event descriptions

The first transient is a small break loss of coolant accident (SBLOCA) taking place at full power operation. The break occurs in the hot leg of the primary coolant loop. Transient continues for about 100 min. The temperature starts to decrease with cold emergency core cooling water injection. System pressure decreases

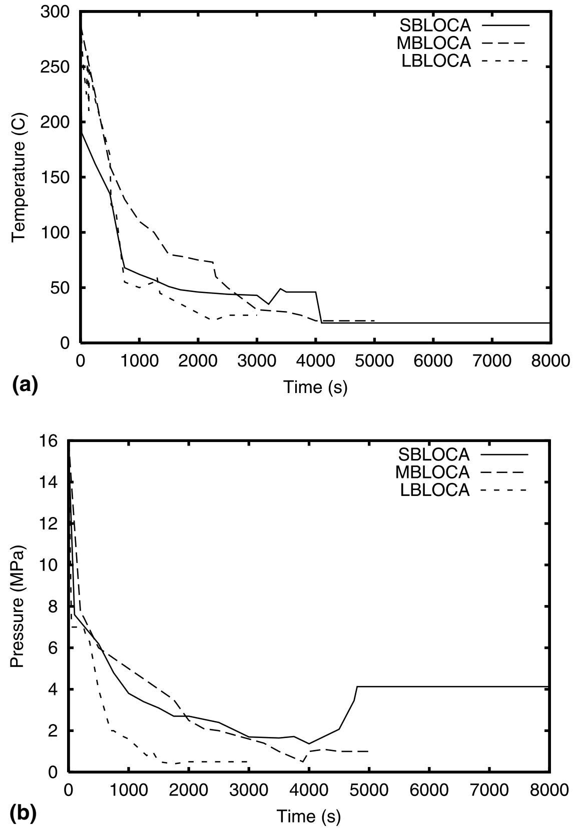

Fig. 1. (a) Temperature and (b) pressure histories of LOCA events.

rapidly because the coolant flow rate through the break is greater than the charging and emergency core cooling flow rate. The final coolant temperature was about 20°C. The second transient corresponds to a break occurred in the hot leg of the primary coolant loop. Cross-sectional area of the break is 50 cm^2. This transient is called a medium break LOCA (MBLOCA) as expressed in WASH-1400. Total transient period is about 5000 s. The third transient is a large break LOCA (LBLOCA) which corresponds to a 200 cm^2 pipe break. Other properties of the event are the same as in MBLOCA. Transient period is 3000 min. Final coolant temperature is about 20°C. Final system pressure is 0.5 MPa. All transient histories are obtained from the thermal hydraulic calculations of RELAP5-MOD3 code [14]. The variation of temperature and pressure during transient events are given in Fig. 1.

3. Failure analysis

Pressure vessel failure analysis include the deterministic and probabilistic methods in order to evaluate the total vessel failure probability and provides an input for probabilistic analysis. Deterministic fracture mechanics analysis include heat transfer analysis as well as the calculation of crack tip stress intensity factors.

3.1. Fracture mechanics analysis

In the analysis, linear elastic material behavior is assumed. Linear elastic fracture mechanics (LEFM) is therefore the base approach for calculations. The fracture toughness values are calculated based on copper and nickel content, fluence, and initial reference nil ductility temperature RT_{NDT0}. The radiation induced shift in RT_{NDT} is calculated using equations in Regulatory Guide 1.99 Revision 2 in the deterministic analysis [13]. The first step in calculations is to evaluate temperature distribution throughout the vessel wall and cladding. This provides an input for thermal and residual stress calculations. Applied stress, thermal stress, and residual stress contributions are evaluated in the assessment of the stress state of the vessel. Stresses induced due to the thermal expansion differences in the base metal and cladding are also accounted. Thus, stress distribution and stress intensity factors are obtained through the vessel. The fluence attenuation and the nil ductility transition temperatures are calculated based on local conditions.

3.2. LOCA analysis

In the transient event analysis, temperature difference between inner and outer surfaces of pressure vessel is considered. Temperature gradient indicates the highest thermal stress at the vessel wall for circumferentially and longitudinally oriented flaws. Total applied stress is the sum of thermal stress, hoop stress and stress induced due to the presence of cladding. During an overcooling event, the stainless steel cladding at the inner surface of a reactor vessel acts to increase the thermal stress in this region. The stainless steel cladding has a greater coefficient of thermal expansion than that for the ferritic steel vessel wall and, therefore, the cladding contracts in greater extent than the ferritic steel having the same thickness. The thermal conductivity of the cladding is lower than that for the base metal, and therefore, it intensifies the temperature difference between the cladding and the base metal. Thus, the cladding exerts pressure on the base metal. Table 3 shows temperature differences between the inner and outer walls for all cases at various stages of all three cases.

Fig. 2 shows the temperature distributions in the vessel wall for five stages of SBLOCA. The maximum temperature difference is observed about 1600 s after the initiation of the transient. Increase in temperature gradient reflects an increase in thermal stress. Fig. 3 also shows the maximum total applied stress for various

Table 3
Temperature difference between inner and outer walls for all transient events

SBLOCA Time (s)	ΔT(°C)	MBLOCA Time (s)	ΔT(°C)	LBLOCA Time (s)	ΔT (°C)
800	148	500	111	300	105
1600	180	1000	160	900	201
2400	149	1500	167	1200	207
3200	104	2000	153	1500	199
4000	66	2500	136	1800	181
4800	42	3000	121	2100	168
5600	41	3500	112	2400	152
6400	41	4000	105	2700	105
7200	29	4500	90	3000	121
8000	2	5000	59		

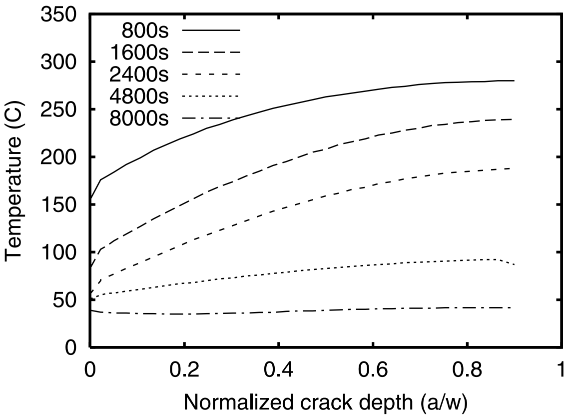

Fig. 2. Temperature distribution in the vessel wall at during SBLOCA.

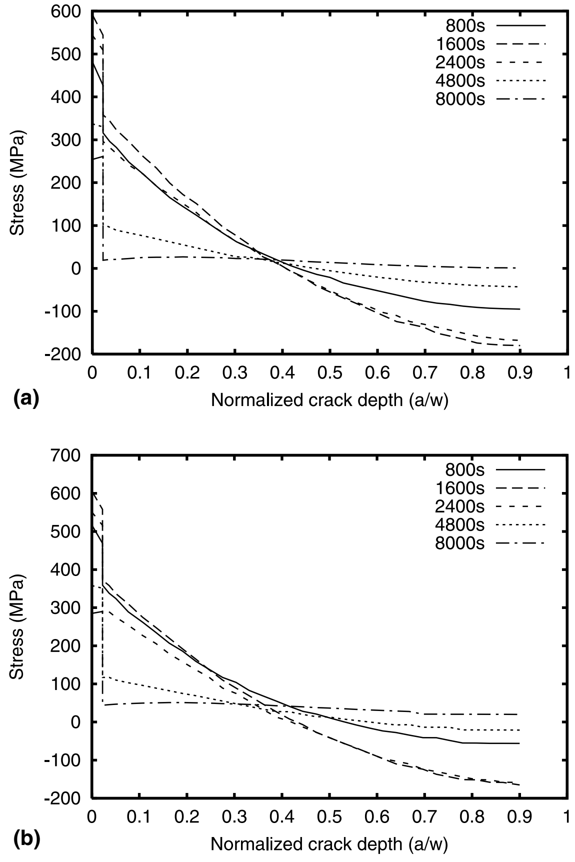

Fig. 3. Stress distribution in (a) circumferential and (b) longitudinal flaws for SBLOCA.

stages of the transient. Actually, temperature gradient increases the contribution of thermal stress which affects the total applied stress for pressure vessel. Another factor for the increase in total stress is the difference in thermal expansion coefficients of base and weld metals. In VISA-II, the mechanical state of the vessel is conservatively treated. It is expected that actual stresses are slightly less than those predicted by the code [9]. Another observation in PTS events is the positive gradients of fracture toughness due to rapid cooling of surface. Fig. 4 shows the stress intensity factors for longitudinally and circumferentially oriented flaws, respectively. Longitudinally oriented flaws are of primary concern in the analysis since they have greater contribution to vessel failure which can be recognized from higher stress intensity factor K_I values of Fig. 4. The maximum stress intensity factor gradients reach the minimum level at the end of the transient.

The maximum temperature difference between the inner and outer surfaces of the vessel is observed between 1000 and 1500 s for MBLOCA as shown in Table 3 and Fig. 5. Maximum temperature difference points out maximum thermal stress in the vessel wall. Fig. 6 shows the total applied stress for longitudinally oriented flaws. Greater values of applied stress is observed between 1000 and 2000 s. At the initiation of transient, temperature difference between inner and

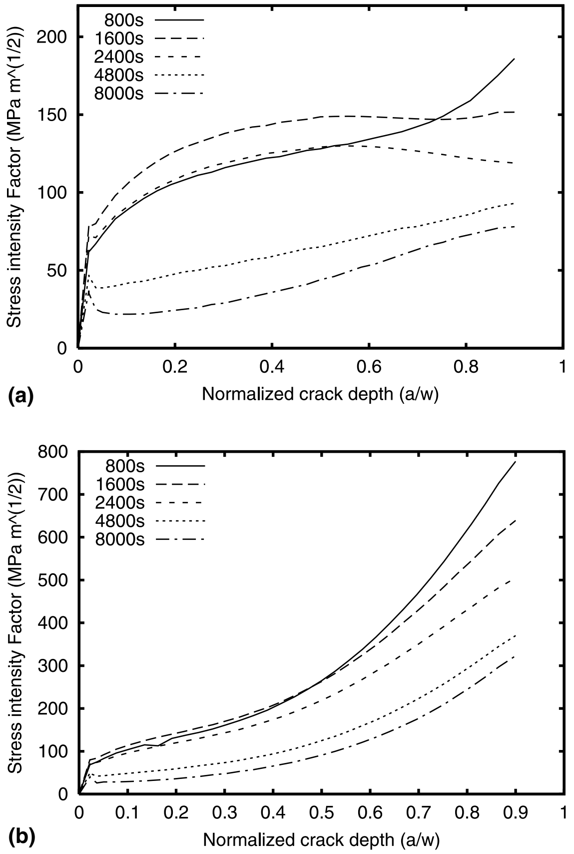

Fig. 4. Stress intensity factors for (a) circumferential and (b) longitudinal flaws for SBLOCA.

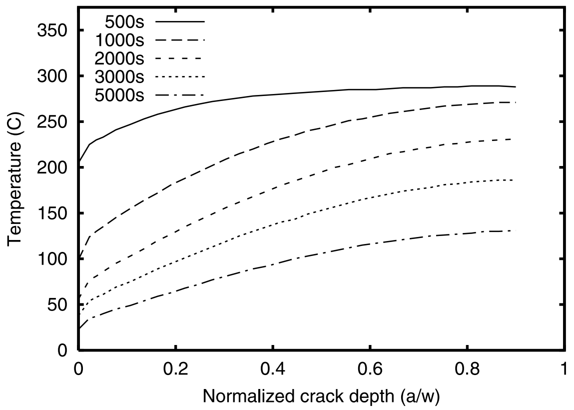

Fig. 5. Temperarure distribution in the vessel wall during MBLOCA.

outer walls is high. Therefore, greater stress intensity factor occurs due to rapid cooling of surface. Fig. 7 shows the variation of stress intensity factors for longitudinally and circumferentially oriented flaws, respectively.

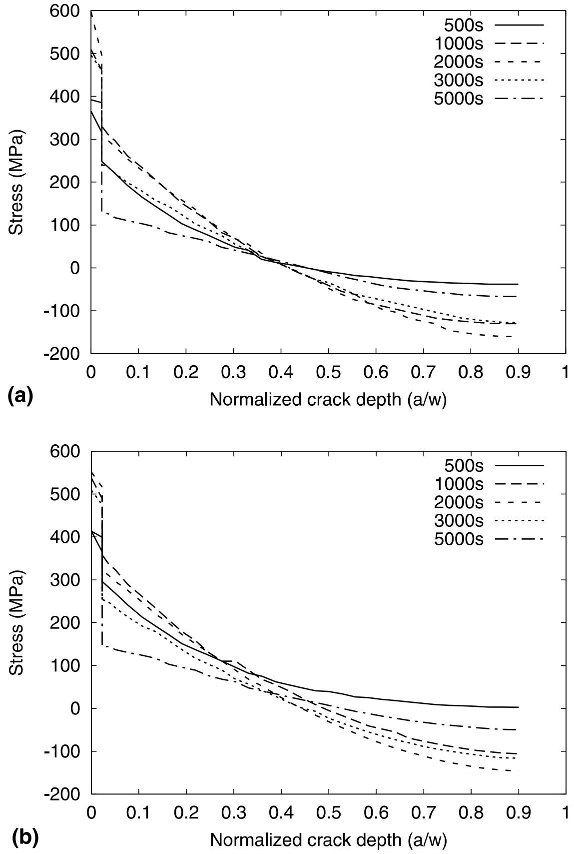

Fig. 6. Stress distribution in (a) circumferential and (b) longitudinal flaws for MBLOCA.

Table 3 shows the temperature differences between the inner and outer surfaces of the vessel wall for LBLOCA. Fig. 8 also shows the variation of temperature at various stages of the event. Maximum temperature difference is observed at 900 and 1500 s when total applied stress is maximum. Total applied stress is higher than that of in the case of MBLOCA since the transient time is short and the coolant discharge rate is high. Fig. 9 show the total applied stress for longitudinally oriented flaws. Since the temperature difference between inner and outer walls is more pronounced at the early stages of the transient, stress intensity factor gradients are more significant for the initial period. Fig. 10 represents the stress intensity factors for longitudinally and circumferentially oriented flaws, respectively.

4. Probabilistic analysis

Probabilistic analysis estimates the probability of vessel failure depending on deterministic calculations.

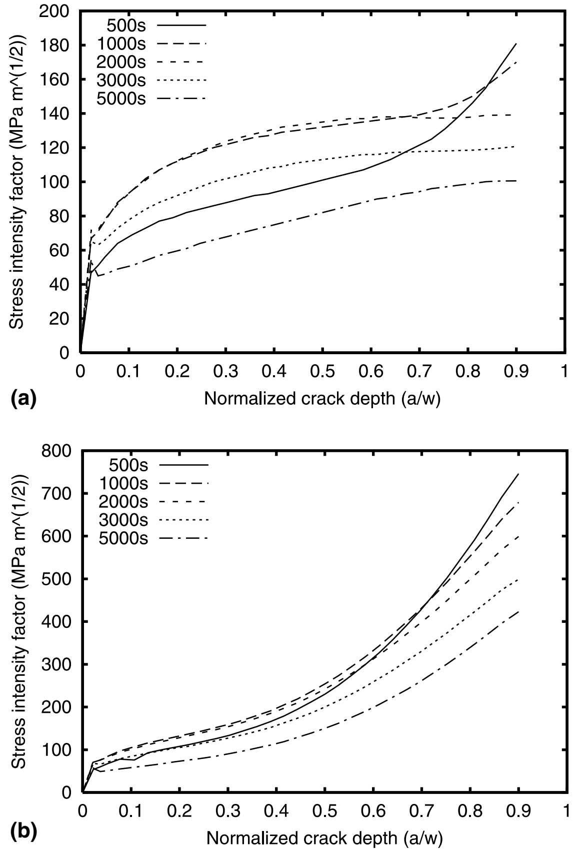

Fig. 7. Stress intensity factors for (a) circumferential and (b) longitudinal flaws for MBLOCA.

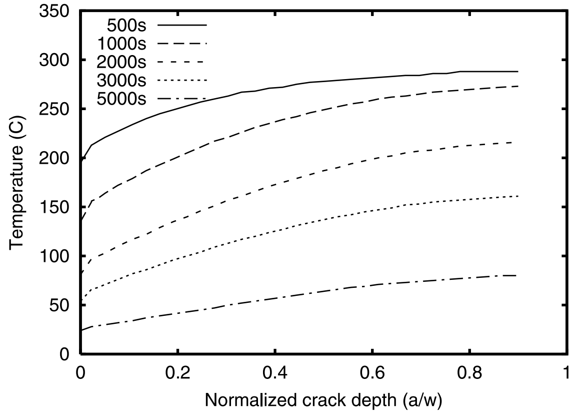

Fig. 8. Temperarure distribution in the vessel wall during LBLOCA.

Only the most critical part of the vessel which are beltline region welds are considered for probabilistic failure analysis.

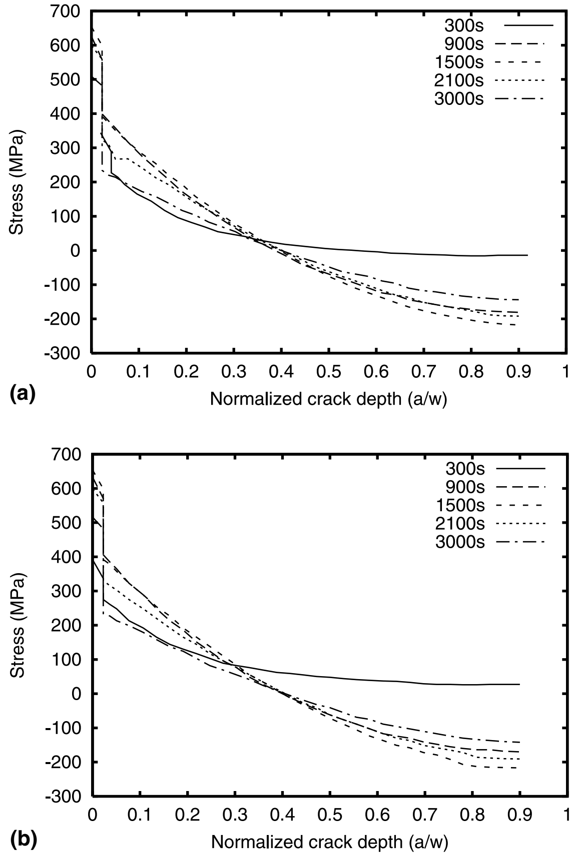

Fig. 9. Stress distribution in (a) circumferential and (b) longitudinal flaws for LBLOCA.

In the probabilistic calculations, first an initial crack is assumed on the inner surface of the vessel. The size of the crack is decided by using a cumulative distribution function such as OCTAVIA which is employed in this study [9,10]. The deterministically calculated stress intensity factor and crack tip temperature are used for this purpose. The material toughness (K_{IC}) is calculated depending upon the crack tip temperature and reference nil ductility temperature (RT_{NDT}). The RT_{NDT} depends on the copper and nickel content, as well as the fluence level. These parameters of the crack tip may be simulated from a Gaussian distribution. The probabilities of crack initiation P_i, failure P_f, and total vessel rupture P_t for three events are presented in Table 4. The probability of initiation for large break LOCA of 200 cm^2 cross-section is the highest due to sharp decrease in temperature and pressure. The probability of vessel failure is the highest for medium break LOCA of 50 cm^2 cross-section since the transient period is shorter compared to the other events. Final temperature and pressure values are nearly identical for the two large break LOCA events, but transient times are different due to discharge rate of

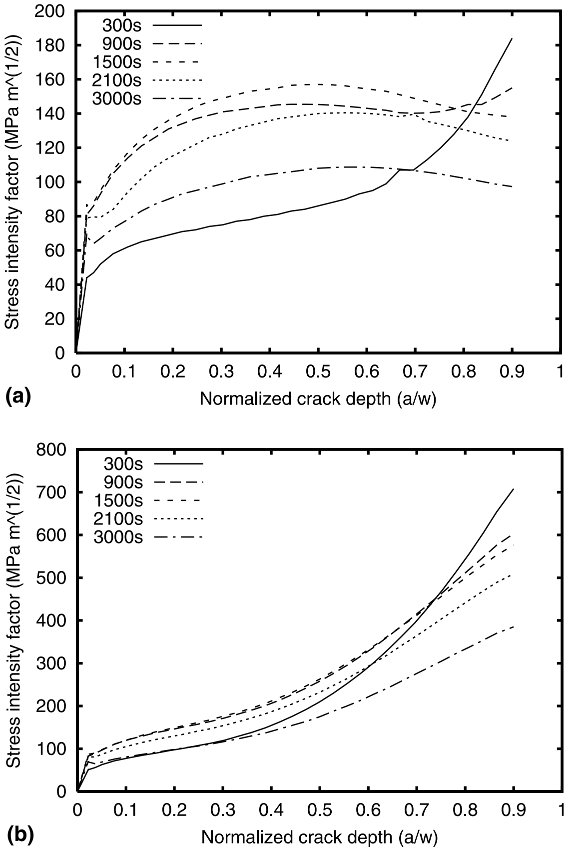

Fig. 10. Stress intensity factors for (a) circumferential and (b) longitudinal flaws for LBLOCA.

coolant and activation of core cooling systems. The probabilities obtained for a transient include flaw initiation and failure in reactor vessel beltline region welds. ΔRT_{NDT}, shift in nil ductility temperature, values are also presented in Table 4 where large break LOCA causes the maximum shift of 77°C.

The NRC PTS screening criterion permits an RT_{NDT} of 149°C for circumferential welds and 132°C for all other beltline materials at the end of life. The initial reference temperature RT_{NDT0} is −6.67°C (20°F) for the material used in the present study. Standard deviations associated with RT_{NDT0} and ΔRT_{NDT} are 9.4°C (17°F) and 15.6°C (28°F), respectively, for weld materials. Then, RT_{PTS} are calculated with the maximum ΔRT_{NDT} according to NRC PTS screening rule [15]

$$RT_{\mathrm{PTS}} = RT_{\mathrm{NDT0}} + M + \Delta RT_{\mathrm{NDT}}, \tag{2}$$

where M is a margin which is determined in terms of the above-mentioned standard deviations. Calculated RT_{PTS} values for three cases are; 104.5°C for SBLOCA, 102.5°C for MBLOCA, and 106.5°C for LBLOCA. Therefore, the maximum RT_{PTS} value remains below the temperature specified in PTS screening criterion.

In order to determine the total failure probability of vessel, the probability of failure of vessel weld and the probability of transient occurrence should also be estimated. Reactor Safety Study (WASH-1400) is employed in the probabilistic assessment of LOCAs [6,7]. According to WASH-1400; the occurrence probabilities for a small break LOCA, a medium break LOCA (50 cm^2), and a large break LOCA (200 cm^2) are 2×10^{-5}, 3×10^{-4}, 1×10^{-4} per reactor year, respectively. Total probability of reactor vessel rupture is the product of probability of transient occurrence and probability of vessel failure. The highest probability for a reactor vessel rupture event is evaluated to be 1.4×10^{-6} for MBLOCA.

5. Conclusions

This study deals with the integrity analysis for a four loop PWR pressure vessel. Analyses are performed for SA533B type ferritic steel and for various LOCA conditions. Transients are generated for a small, a medium (50 cm^2) and a large (200 cm^2) break loss of coolant accidents initiated by hot leg pipe ruptures. Integrity analyses are carried out by means of deterministic and probabilistic methods. Analyses are focused on the reactor vessel beltline region which is the most critical portion of the body due to high neutron fluence. Calculations are performed for longitudinal and circumferential welds. Results of the calculations show that longitudinal welds are more susceptible to failure compared to circumferential welds as expected. The maximum nil ductility temperature shift among all transient events is calculated to be 77°C in LBLOCA of 200 cm^2. RT_{PTS} for this case is 106.5°C and it is still below the NRC PTS limit of 132°C. The highest weld failure and vessel rupture probabilities are calculated to be 4.7×10^{-3} and 1.4×10^{-6}, respectively, for a MBLOCA of 50 cm^2.

Table 4
Probabilities and average and maximum ΔRT_{NDT} for three events

Event	P_{i}	P_{f}	P_{t}	$\overline{(\Delta RT_{\mathrm{NDT}})}$ (°C)	$(\Delta RT_{\mathrm{NDT}})_{\max}$ (°C)
SBLOCA	1.9×10^{-3}	4.6×10^{-5}	9.2×10^{-10}	64	75
MBLOCA	6.4×10^{-2}	4.7×10^{-3}	1.4×10^{-6}	55	73
LBLOCA	6.4×10^{-2}	9.2×10^{-4}	9.2×10^{-8}	55	77

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