Transactions of the 15th International Conference on
Structural Mechanics in Reactor Technology (SMiRT-15)
Seoul, Korea, August 15-20, 1999 D01/7

The Estimation of Critical RT_{NDT} and Probability of Failure for PWR Vessels

Sung-Gyu Jung[1)], Hyun-Su Kim[1)], Tae-Eun Jin[1)], Chang-Heui Jang[2)] and Il-Seok Jeong[2)]

1) Korea Power Engineering Company, Korea
2) Korea Electric Power Institute, Korea

ABSTRACT: Recognizing the importance of Pressurized Thermal Shock(PTS), U.S. Nuclear Regulatory Commission(NRC) issued the PTS rule, based on SECY 82-465[1], which established PTS screening criteria. In this study, the critical RT_{NDT} and conditional probability of failure of RPVs were evaluated in accordance with the procedure of SECY 82-465 to establish preliminary domestic PTS screening criteria.

1. INTRODUCTION

The PTS events, due to severe overcooling of the reactor vessel followed by repressurization, can occur in PWRs during the PTS transients. Thermal and pressure stresses are both tensile stresses that tend to open existing cracks located near the inner surface of the reactor pressure vessel(RPV). Cumulative neutron radiation exposure makes the RPV material more brittle and, therefore, more susceptible to cleavage fracture. For embrittled RPVs, PTS is the major threat to RPV structural integrity. Response to this concern, the NRC established the limit value of fracture toughness(RT_{NDT}) for 8 PTS events of U.S nuclear power plants and specified screening criteria based on the RT_{NDT} of 270°F for axial weld and 300°F for circumferential weld materials.

10CFR50.61[2] prescribed a RT_{PTS} (the value for the reference temperature for end of life fluence) calculation method for RPV beltline material. The RT_{PTS} could be compared with the NRC screening criteria to determine RPV integrity. However, this screening criteria was developed using conservative assumptions to make it applicable to all plants in the U.S.

This study investigated the applicability of the NRC criteria to 5 PWRs. Major event scenarios were selected from the accident cases analyzed in their Final Safety Analysis Reports(FSAR). The PTS analysis was performed in accordance with SECY 82-465. Critical RT_{NDT} and conditional probability of failure were calculated using deterministic/probabilistic fracture mechanics. Based on these procedure, pilot study was performed to determine whether U.S PTS screening criteria could apply to domestic PWRs.

2. SELECTED PTS SCENARIOS

The U.S. NRC performed PTS evaluation based on actual events, however, a PTS transient has never occurred in Korea. Therefore, this paper considered potential PTS scenarios which were described in FSARs. Main steam line breaks(MSLB), with and without offsite power available, were evaluated for 4 PWRs. Also, inadvertent opening of a steam generator relief or safety valve(IOSGADV) was evaluated for 1 PWR.

3. USED THERMAL HYDRAULIC ANALYSIS DATA

The thermal hydraulic analysis(TH) must be performed to obtain fluid temperature and system pressure for each PTS scenario. Estimates of the downcomer fluid temperatures, which result from the cooldown transients associated with these events, are required to perform the deterministic fracture mechanics evaluation. Both the primary and secondary system have to be modeled for thermal hydraulic analysis. The commercial computer codes typically used for this analysis are RELAP, RETRAN, PHOENICS, etc., which calculate temperature and pressure, heat transfer coefficient, etc., for both systems.

In this paper, however, the objective is to develop a method for evaluating the RPV structural integrity for the specific PTS events described in the FSARs. Therefore, the pressure and temperatures were taken from the FSAR data for each plant which are shown in Table 1, instead of the actual TH analysis results of this study. Because the actual temperature measurements were not available for these plants, the coolant temperature in contact with the inner surface of the RPV was estimated by the stylized exponential formula (1).

$$T(t) = T_f + (T_{init} - T_f)\exp(-\beta t) \qquad (1)$$

where $T(t)$ = Time Depend Coolant Temperature
T_f = Final Coolant Temperature
T_{init} = Initial Coolant Temperature
β = Exponential Decay Constant

Table 1. Final Temperature and Pressure for Assumed PTS Events in Korean PWRs

Plants	Assumed PTS Events	Final Temperature(°F)	Pressure(psig)	β (min^{-1})
A	MSLB(Offsite Power not Available)	270	1,000	0.245
B	MSLB(Offsite Power not Available)	270	1,000	0.334
C	MSLB(Offsite Power Available)	290	Time Variable	0.39
D	IOSGADV	320	Time Variable	0.078
E	MSLB(Offsite Power not Available)	290	1,500	0.14

4. STRESS ANALYSIS AND FRACTURE MECHANICS ANALYSIS

The FAVOR code[3], which was recently developed, was used to perform the stress analysis and deterministic/probabilistic analyses. The critical RT_{NDT} and conditional probability of failure were evaluated for each event scenario. The RPV geometries and thermal elastic properties that were used for PWRs A, B, C, D and E are specified in Table 2 and Table 3.

Table 2. RPV Geometry

Plants	RAD(in.)	Wall Thickness(in.)	Cladding Thickness(in.)	Welding Orientation
A	66	6.5	0.125	Circumferential
B	66	6.63	0.125	Axial/Circumferential
C	78.5	7.875	0.125	Axial/Circumferential
D	78.5	7.875	0.125	Axial/Circumferential
E	82.0	8.50	0.125	Circumferential

Table 3. Thermal Elastic Material Properties for Each PWR

Plant / Material Properties	A		B, C, D		E	
	Base Metal	Cladding	Base Metal	Cladding	Base Metal	Cladding
Material	SA508 CL.2	SB168	SA533B CL.1	SB168	SA508 CL.3	SB168
Thermal conductivity (Btu/hr-ft-°F)	23.3	10.6	23.8	10.6	21.8	10.6
Specific heat (Btu/1b-°F)	0.127	0.126	0.125	0.126	0.13	0.126
Density (1b/ft^3)	489	489	489	489	489	489
Modulus of elasticity (ksi)	28000	28000	28000	28000	28000	28000
Thermal expansion coefficient (/°F)	8.21E-6	8.47E-6	8.36E-6	8.47E-6	8.21E-6	8.47E-6
Poisson's ratio	0.3	0.3	0.3	0.3	0.3	0.3
Vessel initial (stress-free) temperature = 550 °F Coefficient of convective heat transfer = 300 BTU /hr-ft^2-°F						

After establishing temperature distribution across the reactor vessel wall by thermal hydraulic analysis, stress distribution can be determined as a function of time and wall thickness. In addition, The behavior of a postulated crack can be evaluated based on fracture analysis. In this study, both deterministic and probabilistic fracture mechanics were used. The critical RT_{NDT} and conditional probability of RPV failure according to surface RT_{NDT} were evaluated by deterministic fracture analysis and probabilistic fracture analysis, respectively.

4.1 Deterministic fracture mechanics analyses

Once the stress distribution is determined as a function of time and wall thickness, fracture mechanics investigates the behavior of postulated cracks in this stress field. For specific crack geometries, the stress intensity factor, K_I was calculated and compared to a fracture initiation toughness, K_{IC}. If the applied stress intensity factor(K_I) is greater than K_{IC} for a specific crack depth, the crack will initiate and then propagate until it is equal to crack arrest fracture toughness, K_{Ia} which is another material property. Furthermore, if K_I is continuously greater than K_{Ia}, it results in through-wall crack penetration of RPV wall.

Material property, K_{IC} and K_{Ia} can be estimated as a function of metal temperature and the metal reference nil-ductility transition temperature by employing equation (2) which is defined in ASME Section XI for RPV ferritic steel.

$$
\begin{aligned}
K_{IC} &= 33.2 + 2.806\exp[0.02(T - RT_{NDT} + 100^{\circ}\mathrm{F})] \\
K_{Ia} &= 26.8 + 1.233\exp[0.0145(T - RT_{NDT} + 160^{\circ}\mathrm{F})]
\end{aligned} \tag{2}
$$

The units of K_{IC} and K_{Ia} is ksi$\sqrt{\text{in}}$. T is the metal temperature and RT_{NDT} is the adjusted reference temperature (ART) value of the limiting vessel material at the tip of the postulated flaw. For operating nuclear power plants where irradiation embrittlement is a factor, ART can be estimated by using equation (3) which is prescribed in REG. Guide 1.99, Revision 2[4].

$$ART(RT_{NDT}) = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \tag{3}$$

The ΔRT_{NDT} represents an increment of nil-ductility transition temperature resulting from neutron irradiation. Initial RT_{NDT} and Margin are incorporated to consider the uncertainties inherent in those terms.

In order to estimate critical RT_{NDT} for MSLB(with and without offsite power available) and IOSGADV, it was performed assuming a range of RT_{NDT} values; that is, initial RT_{NDT} and Margin(59°F) were fixed and RT_{NDT} was changed as neutron fluence increased according to operation. Thus, it was possible to predict the limiting vessel material condition(critical RT_{NDT}) necessary to prevent vessel failure for each case. In this paper, the critical RT_{NDT} was considered to be the RT_{NDT} where K_{IC} is equal to K_I during the PTS transient at 15% of RPV wall thickness[5].

It was assumed that an axial weld is an infinite flaw while a circumferential weld is a continuous flaw. Because this RT_{NDT} is at the inside-wall, the calculated RT_{NDT} was converted into surface critical RT_{NDT}. The input parameters for fracture analysis are shown in Tables 1 through 3. Also, the evaluation results are shown in Table 4 and Figures 1 through 5, which give the stress intensity factor and each minimum K_{IC} at 15% of wall-thickness.

4.2 Probabilistic fracture mechanics analyses

Probabilistic Fracture Mechanics(PFM) Analyses have been performed to evaluate failure risk of the RPV based on crack propagation frequency and failure frequency caused by PTS events. Probabilistic Fracture Analysis is based on Monte Carlo techniques, i.e., a large number of deterministic analyses are performed on stochastically generated vessels to determine if each vessel, containing a specified number of flaws, will fail when subjected to a specified PTS event at a particular time in the operating life of the vessel.

Table 4. Critical RT_{NDT} vs. Postulated PTS Events for Korean NPPs

Plants	PTS Events	Welding Orientation	$K_I=K_{IC}(ksi\sqrt{in})$	Critical $RT_{NDT}(^{o}F)$
A	MSLB(Without Offsite Power)	Circumferential	60(@20 min)	345.2
B	MSLB(Without Offsite Power)	Axial	65(@25 min)	323
C	MSLB(With Offsite Power)	Axial	80(@25 min)	331
D	IOSGADV	Axial	-	-
E	MSLB(Without Offsite Power)	Circumferential	66(@40 min)	340

The conditional probability of failure for a specified PTS event is estimated by dividing the number of vessels that fail by the total number of vessels simulated. Input data used to compare stress intensity factor and material property(K_{IC}, K_{Ia}) for probabilistic fracture analysis are composed of initial crack size distribution, fluence of surface, Cu and Ni contents, and initial RT_{NDT}. All these input data, as well as K_{IC}, K_{Ia}, were statistically simulated.

Comparing stress intensity factor during the transient with K_{IC}, it was determined whether a crack can be initiated or not. Also, comparing with K_{Ia}, it was determined whether a crack can propagate or not. The conditional probability of failure is then evaluated using these results.

This paper made reference to the SECY 82-465 App. H procedure for PFM analysis. Six axial welds of the RPV plate and two circumferential welds of the RPV forging were evaluated. Though they are somewhat different for each RPV, all axial and circumferential weld volumes were assumed to be 4.41ft^3 and 11ft^3, respectively. Flaw density of weld areas was assumed to be 0.028 $Flaws/ft^3$. Based on these facts, the conditional probability of failure shown in Figure 6 was derived.

In addition, in order to compare with $5X10^{-6}(year^{-1})$, which is the maximum acceptable PTS failure risk prescribed by Reg. Guide 1.154 for determining RPV integrity, the sequence frequency for each scenario should be estimated[6]. Integrated PTS risk is then calculated to combine this frequency with the conditional probability of failure of Figure 6. However, the sequence frequency for each scenario was not estimated in this study, therefore, the integrated PTS risk is yet to be determined.

5. CONCLUSION

Using deterministic methodology, the calculated critical RT_{NDT} values are found to be similar to those of US evaluation results for similar PTS events, as shown in Table 5. Also, evaluation results of the Main Steam Line Break scenario show that the critical RT_{NDT} value is greater than the screening criteria of NRC, at both axial and circumferential welds. Therefore, operational margin is available for this scenario.

Probabilistic fracture mechanics calculations were performed to determine the conditional probability of vessel failure, P(F/E), for 5 postulated domestic PWR transients.

As a consequence of probabilistic fracture mechanics analysis, the conditional probability of vessel failure has a large gap due to a mean input value and standard deviation. Therefore, a higher value of Cu and Ni content, and initial RT_{NDT} for each plant was selected to give a conservative estimation. The conditional probability of vessel failure was then calculated as a function fluence(mean surface RT_{NDT}). These results are shown in Figure 6.

In this figure, the conditional probability of vessel failure of the axial weld area was greater than those of the circumferential weld area for MSLB. Values of the conditional probability of vessel failure of the axial weld for IOSGADV, which is a somewhat less severe transient, were estimated lower than those of the circumferential weld for MSLB.

Evaluation results showed that critical RT_{NDT} based on deterministic/probabilistic fracture analysis were calculated to be greater than the PTS screening criteria, 270°F for axial weld materials and 300°F for circumferential weld materials. Therefore, it is implied that the RPV has a integrity for the assumed scenarios.

This study used pre-established scenarios and the associated thermal hydraulic analyses as PTS events. However, if plant-specific scenarios are selected and thermal hydraulic analysis is performed, then realistic PTS-related event data can be obtained, and definitive screening criteria can be developed for domestic PWRs.

Table 5. Critical $RT_{NDT}(RT_C)$ Value of 8 PWRs in the U.S

Plant Name	PTS Events	Welding Orientation	SECY 82-465 RT_C(°F)
Robinson ('70)	SLB	Axial	321
Robinson ('72)	Stuck SG Valve	Axial	381
Robinson ('75)	RCP Seal SBLOCA	Axial	354
Rancho Seco	Excessive Feed Water Tr.	Axial	295
TMI-2	SBLOCA(PORV Stuck Open)	Axial	209
Ginna	SGTR	Circumferential	378
Crystal River-3	SBLOCA	Axial	250
Prairie Island	SGTR	Axial	400

REFERENCES

1. U.S. NRC, 1982. United States Nuclear Regulatory Commission Policy Issue. SECY 82-465

2. U.S. NRC, 1995. Fracture Toughness Requirements for Light Reactor Pressure Vessels; Final Rule. 10CFR 50.61

3. Dickson, T.L. 1994, FAVOR. A Fracture Analysis Code for Nuclear Reactor Pressure Vessels.
Release 9401, ORNL/NRC/LTR 94-1, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.

4. U.S. NRC, 1988. Radiation Embrittlement of Reactor Vessel Materials.
Regulatory Guide 1.99, Revision 2.

5. Dickson, T.L. 1995. Review of the Pressurized Thermal Shock Screening Criteria for Embrittled Pressurized Water Reactor Pressure Vessels.

6. Selby, D.L. 1985. Pressurized Thermal Shock Evaluation of the H.B. Robinson

Unit 2 Nuclear Power Plant. NUREG/CR-4183.

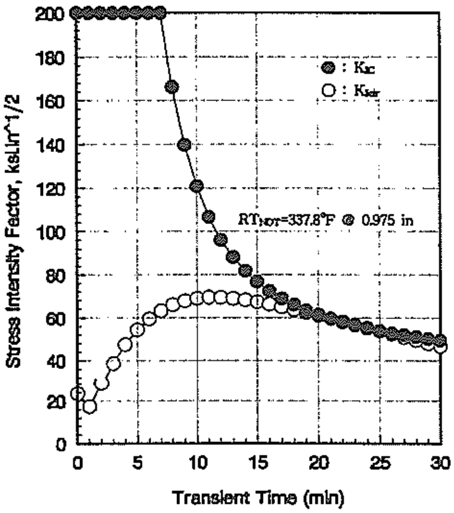

Fig 1. K_I and K_{IC} time histories for 0.975 inch infinite length circumferential flaw assumed for plant A SLB

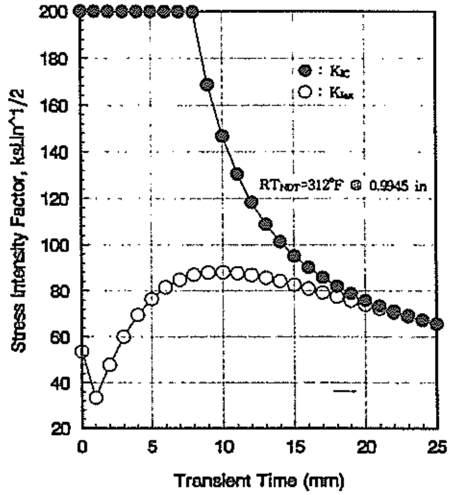

Fig 2. K_I and K_{IC} time histories for 0.9945 inch infinite length axial flaw assumed for plant B SLB

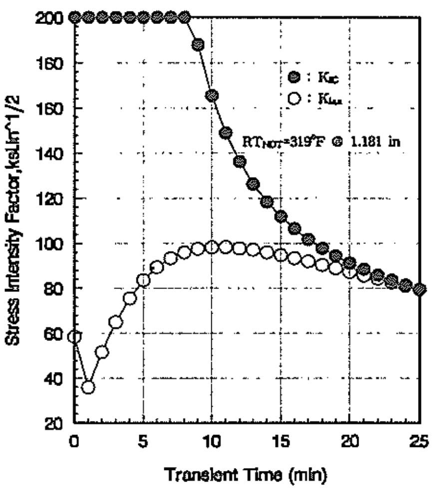

Fig 3. K_I and K_{IC} time histories for 1.181 inch infinite length axial flaw assumed for plant C SLB

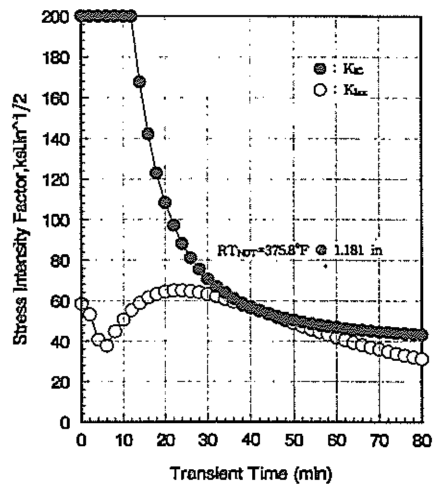

Fig 4. K_I and K_{IC} time histories for 1.181 inch infinite length axial flaw assumed for plant D IOSGADV

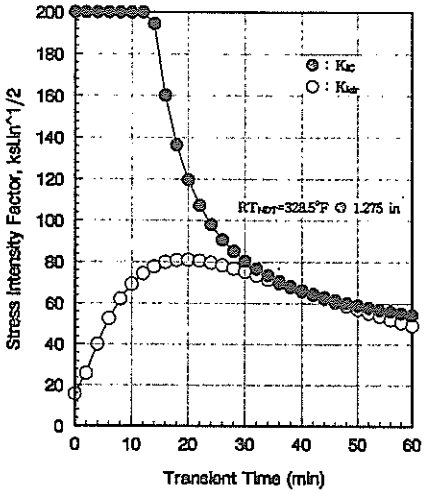

Fig 5. K_I and K_{IC} time histories for 1.275 inch infinite length circumferential flaw assumed for plant E SLB

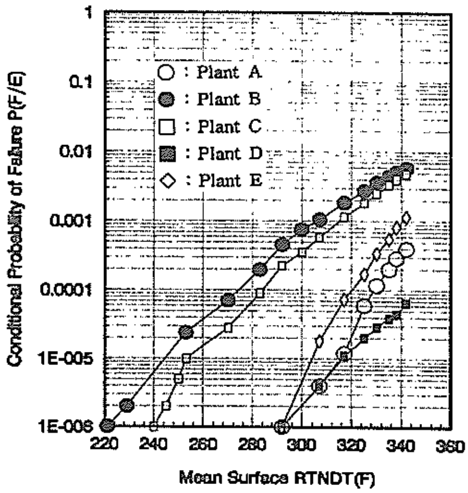

Fig 6. Conditional Probability of Failure for Korean NPPs