Evaluation of Failure Probability of BWR Vessel Under Cool-down and LTOP Transient Conditions Using PROFAS-RV PFM Code

Jong-Min Kim [a*], Bong-Sang Lee [a]
[a] *Nuclear Materials Research Division, Korea Atomic Energy Research Institute, Yuseong-gu, Daejeon, Korea*
[*] *Corresponding author: jmkim@kaeri.re.kr*

1. Introduction

The round robin project was proposed by the PFM Research Subcommittee of the Japan Welding Engineering Society to Asian Society for Integrity of Nuclear Components (ASINCO) members, which is designated in Korea as Phase 2 of A-Pro2. The objective of this phase 2 of RR analysis is to compare the scheme and results related to the assessment of structural integrity of RPV for the events important to safety in the design consideration but relatively low fracture probability. Six organizations in Korea, KAERI 1&2, KINS, KNHP-CRI, KEPCO E&C and Kyung Hee University were participated in the round robin analysis. In this paper, analyses results of KAERI 1 for BASE case and sensitivity analyses cases such as different transient, fluence level, copper content, nickel content, initial reference temperature-nil ductile transition and different irradiation embrittlement model are presented.

2. Analysis Method

2.1 PROFAS-RV Code

In this study, PROFAS-RV[1] code which was developed by KAERI for the deterministic and probabilistic fracture mechanics analysis of the reactor vessel was used. New radiation shift correlations in the 10CFR50.61a[2] and stress intensity factor calculation method of RCC-MRx A16[3] were added to the PROFAS-RV code. The parallel programming for multi-core processors with MPI is applied in the code to reduce the computing time of full Monte-Carlo simulation. The PROFAS-RV is being tested with other codes, and it is expected to revise and upgrade by reflecting the latest model and calculation method continuously.

2.2 Problem Definition

The reactor vessel considered in the analysis is a typical BWR with an inner radius of 3200 mm and a base metal thickness of 160 mm without cladding. The material properties and analysis conditions are shown as in Table 1. The effects of initial RT_{NDT}(-30, -15, 0 °C), Cu (0.2, 0.15, 0.1 %) and Ni (1.0, 0.8, 0.6%) content, fluence level (0.02, 0.1, 0.2, 0.3, 0.4, 0.5 (10^{19}n/cm^2)) and RT_{NDT} shift model (R.G. 1.99 rev. 2 and 10CFR50.61a) on the failure probability were evaluated. LTOP (low temperature over pressure) and cool-down conditions were considered as input transient.

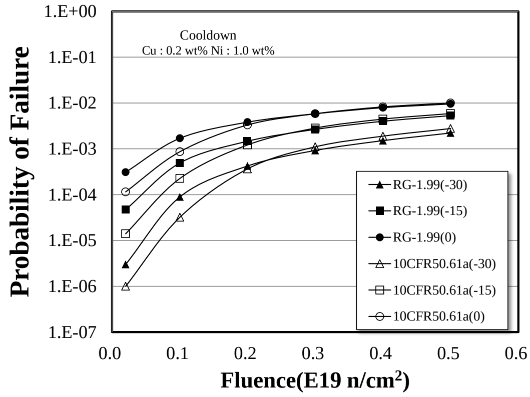

Fig. 1. Comparison of probability of failure between R.G. 1.99 rev. 2 and 10CFR50.61a for cool-down transient.

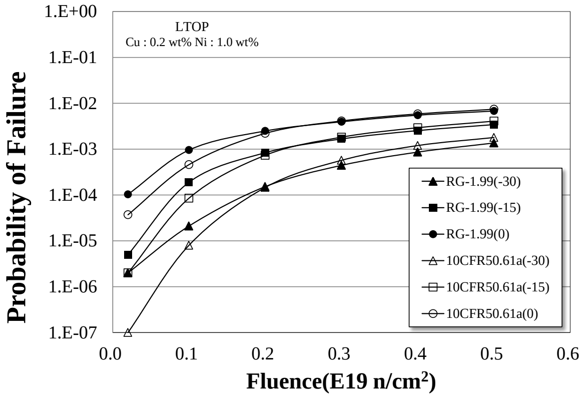

Fig. 2. Comparison of probability of failure between R.G. 1.99 rev. 2 and 10CFR50.61a for LTOP transient.

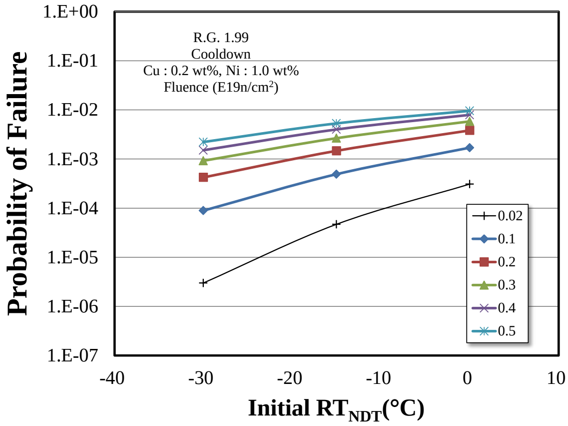

Fig. 3. Effect of initial RT_{NDT} on the probability of failure for cool-down transient.

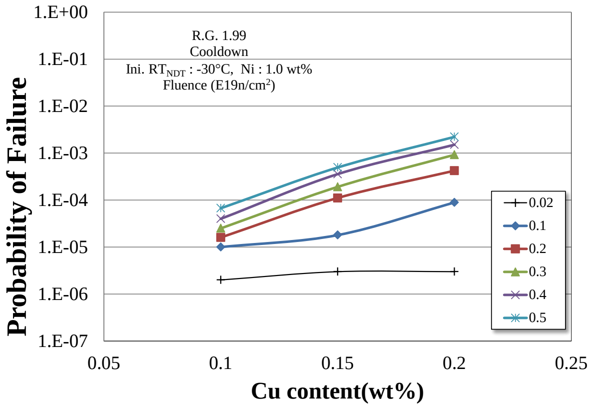

Fig. 4. Effect of Cu content on the probability of failure for cool-down transient.

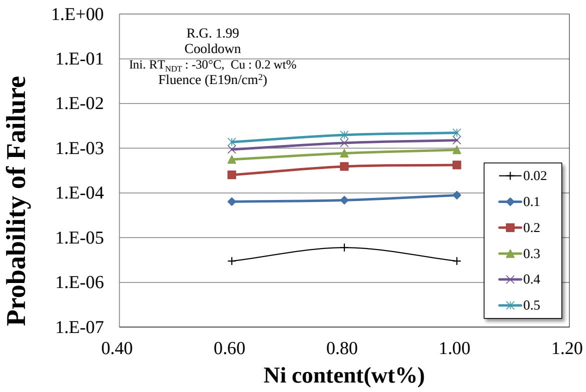

Fig. 5. Effect of Ni content on the probability of failure for cool-down transient.

3. Analysis Results

The results of sensitivity study are shown in Figs. 1-5. The probability of failure of 10CFR50.61a is lower than that of RG-1.99 rev. 2 for fluence lower than 0.2×10^{19} n/cm^2. However, it showed the opposite trend for fluence larger than 0.3×10^{19} n/cm^2. The effect of difference on the failure probability for LTOP transient is more significant for the lower fluence region. Failure probability increases with increasing the content of Cu and Ni and the initial RT_{NDT}. Increasing rates are almost the same for the all fluence ranges except for the fluence level of 0.02×10^{19} n/cm^2, and effects of Ni are lower than that of Cu for increasing failure probability.

4. Conclusions

In this study, probabilistic fracture mechanics analysis was performed for the round robin cases using PROFAS-RV code. The effects of key parameters such as different transient, fluence level, Cu and Ni content, initial RT_{NDT} and RT_{NDT} shift model on the failure probability were systematically compared and reviewed. These efforts can minimize the uncertainty of the integrity evaluation for the reactor pressure vessel.

Table I: PFM Analysis Conditions

Item	Property and analysis condition
Coefficient of heat transfer	1817 (W/m^2/K)
Density	7600 (kg/m^3)
Thermal conductivity	54.60 (W/m^2/K) (20^{o}C) 45.80 (W/m^2/K) (300^{o}C)
Specific heat	488.722 (J/kg/K) (20^{o}C) 568.520 (J/kg/K) (300^{o}C)
Young's modulus	204 GPa (20^{o}C) 185 GPa (300^{o}C)
Poisson's ratio	0.3
Thermal expansion coefficient	1.090×10^{-5} (1/K) (20^{o}C) 1.490×10^{-5} (1/K) (300^{o}C)
Flaw direction	Axial direction
Geometry of flaw	Semi-elliptical
Aspect ratio	a/c = 1/3 (c is the half crack length.)
Mean of Initial RT_{NDT}	0 for base and -30 for weld (^{o}C)^{*2}
Std. dev. of Initial RT_{NDT}	10 (^{o}C)
Prediction Formula of $\triangle RT_{NDT}$	R.G 1.99 or 10CFR50.61a
Std. deviation of $\triangle RT_{NDT}$	0.0
Mean of Cu content	0.2 (wt%)
Std. deviation of Cu content	0.01 (wt%)
Mean of Ni content	1.0 for weld (wt%)
Std. deviation of Ni content	0.02 (wt%)
Mean of P content	0.02 (wt%)
Std. deviation of P content	0.001 (wt%)
Mean of Mn content	1.4 (wt%)
Std. dev. of Mn content	0.02 (wt%)
K_{Ic} (ORNL mean curve)	Stand. Dev. is 15% of mean
K_{Ia} (ORNL mean curve)	Stand. Dev. is 10% of mean
Upper shelf fracture toughness	not considered
Flow stress	551.6 (MPa)
Yield stress	489 (MPa) (20^{o}C) 423 (MPa) (300^{o}C)
Std. deviation of fluence	10% of mean value
Irradiation temperature	276 (^{o}C)
Censoring of mat. prop.	5 times of standard deviation

REFERENCES

[1] Kim, J. M., Lee B. S., Kim T. H. and Chang, Y. S., "Development of Probabilistic Fracture Mechanics Analysis Codes for Reactor Pressure Vessels Considering Recent Embrittlement Model and Calculation Method of SIF – Progress of the Work," Proceedings of ASME 2016 Pressure Vessel & Piping Division Conference, PVP2016-63128, 2016.
[2] Title 10, Section 50.61(a), "Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," The Code of Federal Regulations, Federal Register, Vol. 75, No. 1, 2010.
[3] RCC-MR(MRx) code, "Design and Construction Rules for Mechanical Components of Nuclear Installations Applicable for High Temperature Structures and ITER Vacuum Vessel," AFCEN, Paris, 2010.