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Original Article

Round robin analysis of vessel failure probabilities for PTS events in Korea

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ABSTRACT

Round robin analyses for vessel failure probabilities due to PTS events are proposed for plant-specific analyses of all types of reactors developed in Korea. Four organizations, that are responsible for regulation, operation, research and design of the nuclear power plant in Korea, participated in the round robin analysis. The vessel failure probabilities from the probabilistic fracture mechanics analyses are calculated to assure the structural integrity of the reactor pressure vessel during transients that are expected to initiate PTS events. The failure probabilities due to various parameters are compared with each other. All results are obtained based on several assumptions about material properties, flaw distribution data, and transient data such as pressure, temperature, and heat transfer coefficient. The realistic input data can be used to obtain more realistic failure probabilities. The various results presented in this study will be helpful not only for benchmark calculations, result comparisons, and verification of PFM codes developed but also as a contribution to knowledge management for the future generation.

1. Introduction

One significant challenge to structural integrity of the reactor pressure vessel (RPV) in a pressurized water reactor (PWR) is posed by a pressurized thermal shock (PTS) event wherein severe cooling of the core occurs together with, or followed by, pressurization. The temperature differential between the nominally ambient temperature make-up water and the operating temperature of a pressurized water reactor produces significant thermal stresses in the vessel wall. For aged RPVs, these stresses could be high enough to initiate a running cleavage crack that could propagate all the way through the vessel wall [1].

Many studies have been carried out internationally to investigate the structural integrity of reactor pressure vessels under a pressurized thermal shock [2~6]. The International Comparative Assessment Study (ICAS) of Pressurized-Thermal-Shock in Reactor Pressure Vessels was organized in 1996 by OECD/NEA to bring together an international group of experts from research, utility and regulatory organizations in a comparative assessment study of integrity evaluation methods for nuclear RPVs under PTS loading. The analysis results submitted by the participants were compiled in a data-base as a basis for discussions about the predictive capabilities of the analysis methods applied by the participants. As a complementary step to ICAS program on RPV integrity, Probabilistic Structural Integrity of a PWR Reactor Pressure Vessel (PROSIR) benchmarks were prepared and organized by IAGE WG Sub-group of metal components of OECD/NEA in 2003. The objectives were to issue some recommendation of best practices in a probabilistic approach.

IAEA organized a coordinated research project to perform benchmark deterministic calculations of a typical PTS regime with the main comparing effects of individual parameters on the final PTS integrity assessment and then to recommend the best practice for their implementation in PTS procedure. Several benchmark calculations including sensitivity studies were performed and their results were compared to assess the influence of national code requirements and individual parameters. This allowed better technical support for reactor operation safety and life management, and provided a reference for probabilistic evaluations of RPV failure frequency.

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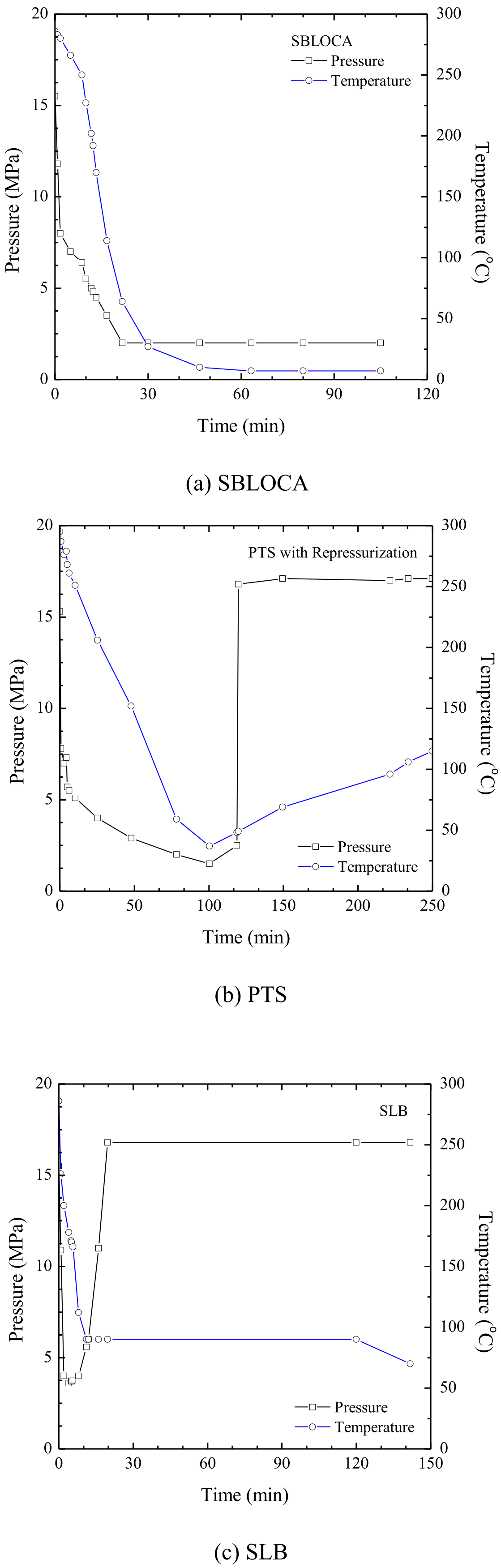

Fig. 1. Transient histories for PTS events.

Table 1
Vessel information for round robin analysis.

lant	C1	C2	A1	A2
Output (MWe)	587	1000	1400	1500
Thickness (inch)	6.5	8.22	9.185	10.095
Clad thickness (inch)	0.125	0.16	0.125	0.125
Inner radius (inch)	66.0	82.015	91.125	97.715
Material	SA508 Cl.2	SA508 Cl.3	SA508 Cl.3	SA508 Cl.3
Cu content (wt%)	0.29	0.03	0.02	0.02
Ni content (wt%)	0.68	0.108	0.03	0.03

Table 2
Analysis matrix for sensitivity.

Parameter	Values
Flaw orientation	Circumferential, Axial
Flaw size (*l/a*)	Infinite, 12, 6
Inspection data for Marshall model	Considering, Not considering
Fluence (10^{19} n/cm^2)	0.5 ~ 9 (0.5, 1, 2, 4, 6, 8, 9)
Copper contents (wt%)	0.02, 0.05, 0.10, 0.30

Table 3
PFM codes used for round robin analysis.

Participant	Code	Reference
P1	R-PIE	[7]
P2	VISA-II	[8]
P3	PROFAS-RV	[9]
P4	FAVOR16.1	[10]

A round robin analysis was organized by Atomic Energy Research Committee of Japan Welding Engineering Society, for international PFM round robin project among Asian countries as a part of ASINCO (Asian Society for Integrity of Nuclear Components) project. Four organizations in Korea participated in the project and their results were compiled. Also, the phase 2 project was launched in 2014 focusing on the assessment of structural integrity of RPV for the events important to safety in the design consideration but relatively low fracture probability. The failure probabilities of the reactor pressure vessel for the low temperature over-pressurization transient and cooldown event were calculated and several sensitivity analyses were performed.

Based on international round robin projects, round robin analyses for vessel failure probabilities due to PTS events are proposed for plant-specific analyses of all types of reactors developed in Korea. Four organizations, that are responsible for regulation, operation, research, and design of the nuclear power plant in Korea, participated in the round robin analysis. The vessel failure probabilities from the probabilistic fracture mechanics analyses are calculated to assure the structural integrity of the reactor pressure vessel during transients which are expected to initiate PTS events. The failure probabilities due to various parameters are compared with each other to assure an understanding of the effects of key parameters.

2. Analysis

2.1. Transients

Among transients used for international round robin analyses, three transients of small break loss of coolant accident (SBLOCA), PTS with repressurization, and steam line break (SLB) are chosen for the analysis, as shown in Fig. 1. The other transients considered in ASINCO round robin analysis are not considered for very low probability of vessel failure.

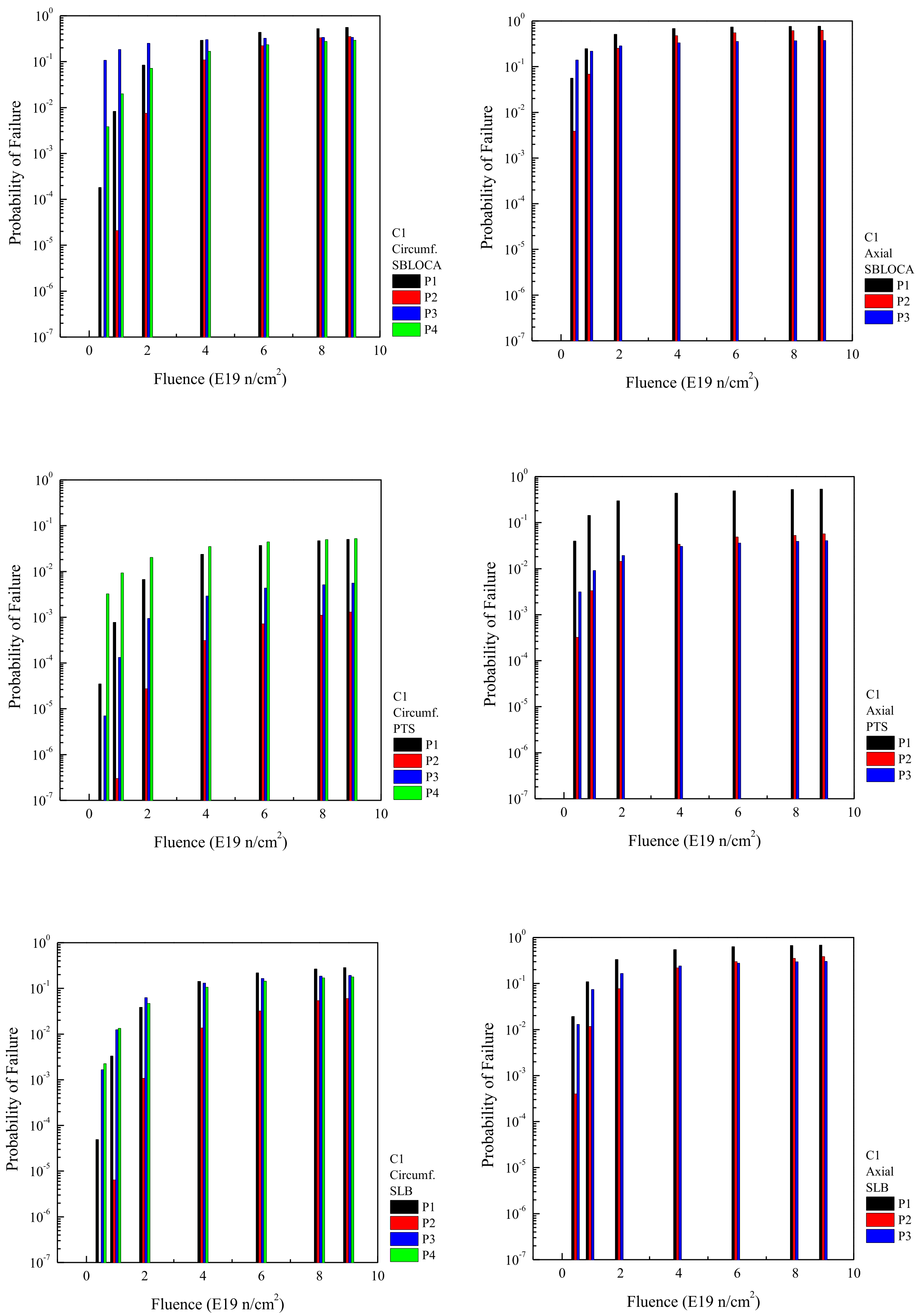

Fig. 2. Comparison of failure probabilities for C1.

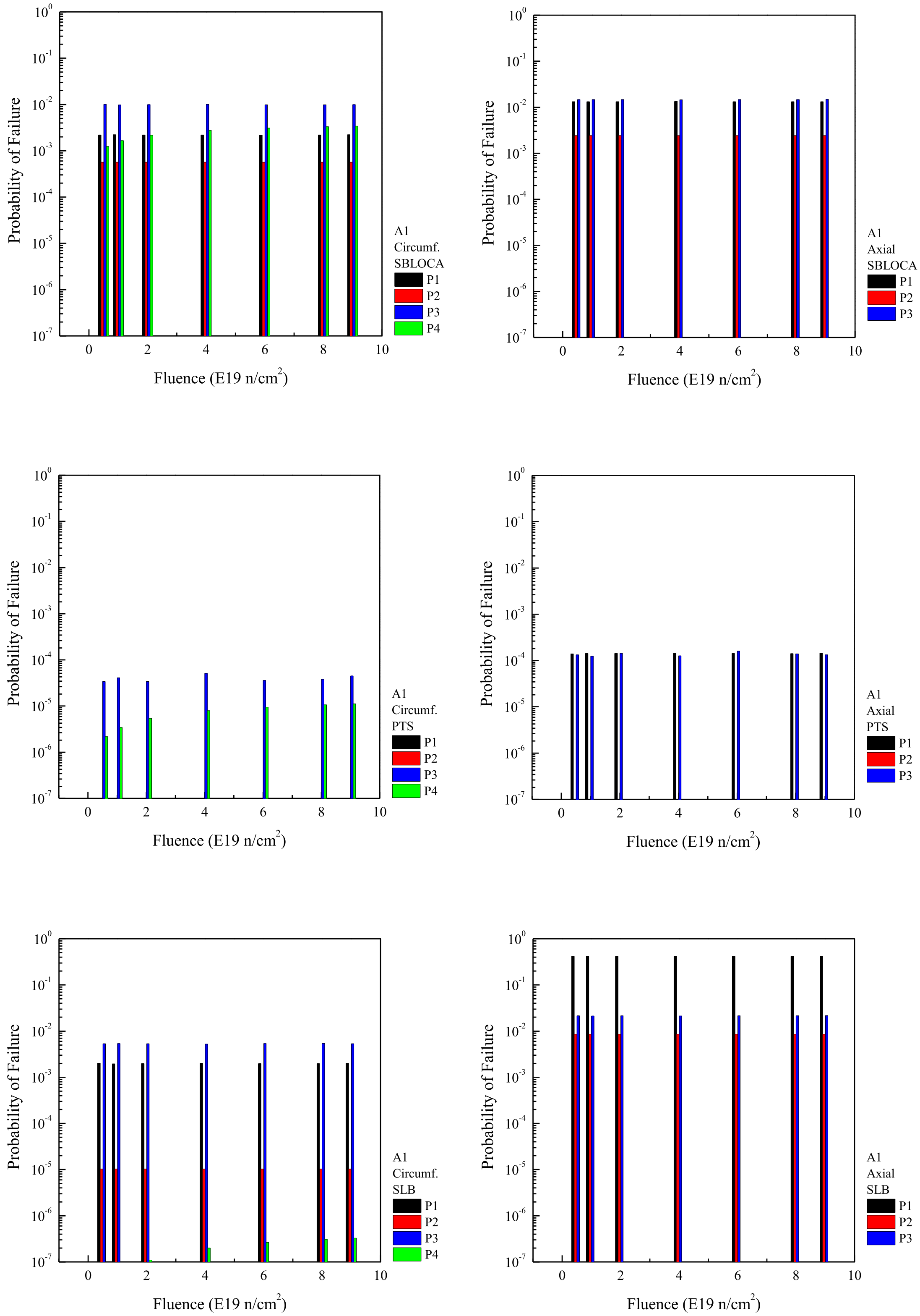

Fig. 3. Comparison of failure probabilities for A1.

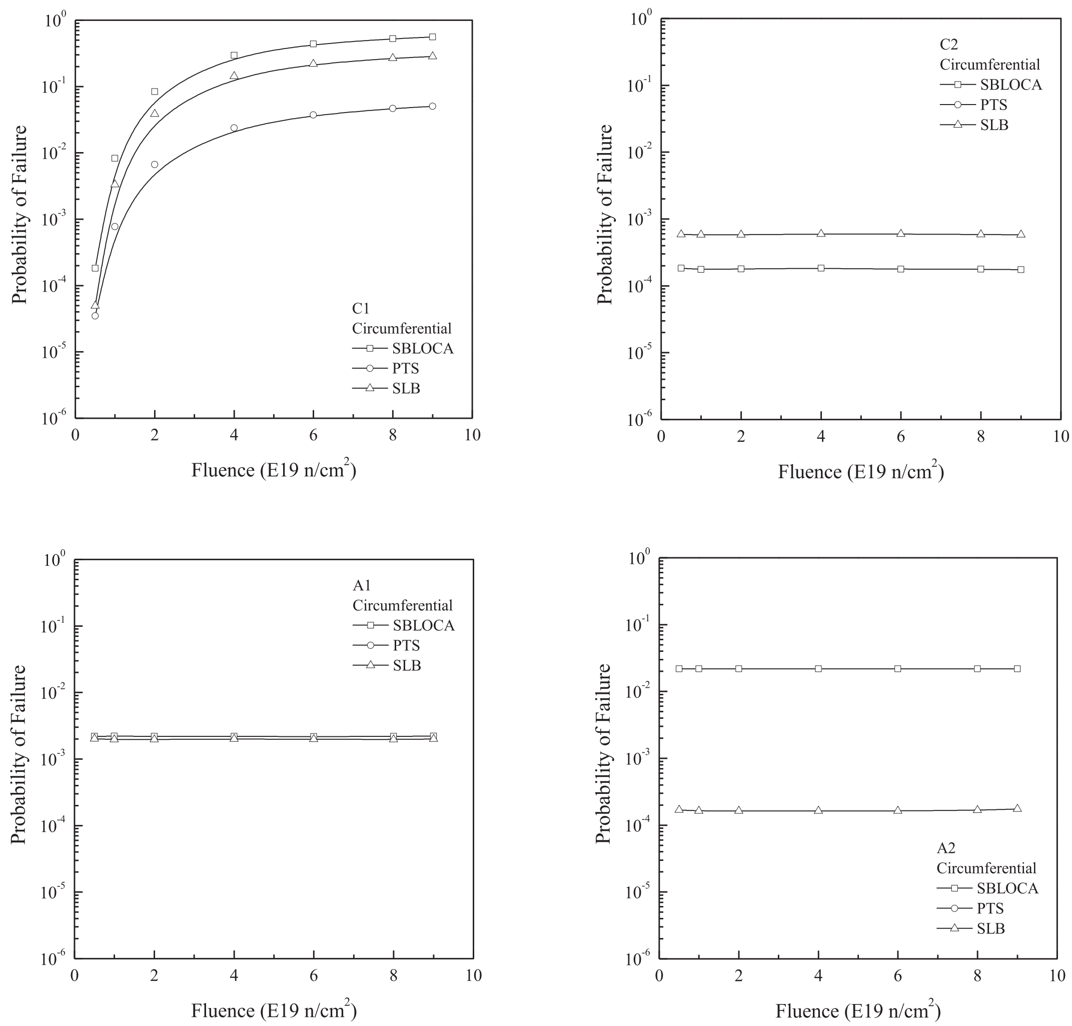

Fig. 4. Probability of vessel failure for circumferential flaw by P1

2.2. Plant specific data

Four types of nuclear power plants are considered in this study, as shown in Table 1. The contents of copper and nickel that augment radiation embrittlement are given and their corresponding uncertainties are arbitrarily chosen to be 20% of the mean values except for C1, the values of which came from the design calculation of the manufacturer. 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 flaw postulated is surface breaking flaw with infinite through-clad in the axial or circumferential orientation.

The upper shelf fracture toughness of C1 plant, $132\text{ksi}\sqrt{\text{in}}$, is used in other plants for comparison purpose because no vessel failures of the advanced reactors were expected for the upper shelf of $200\text{ksi}\sqrt{\text{in}}$ in the preliminary study.

2.3. Sensitivity study

An analysis matrix for sensitivity is shown in Table 2, where the axial infinite flaw with the Marshall model of flaw distribution considering inspection is chosen as a mandatary case. Several sensitivity analyses can be performed as a preference of each participant.

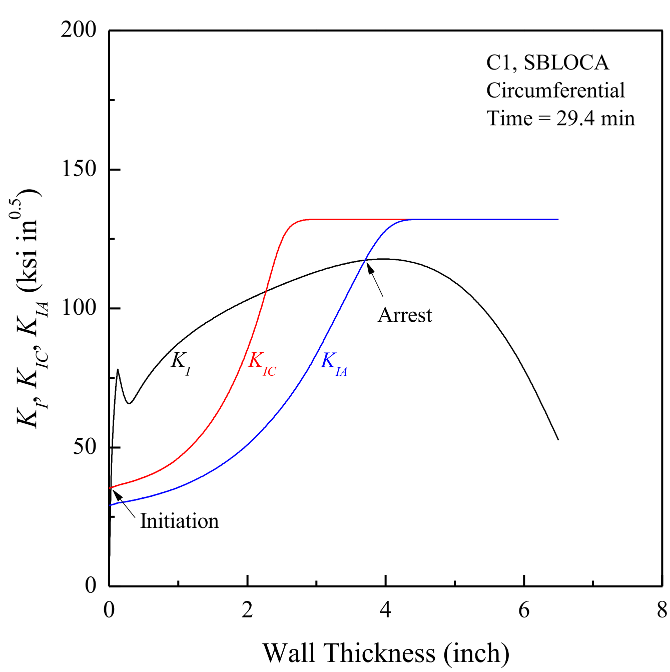

Fig. 5. Determination of initiation and arrest of the flaw.

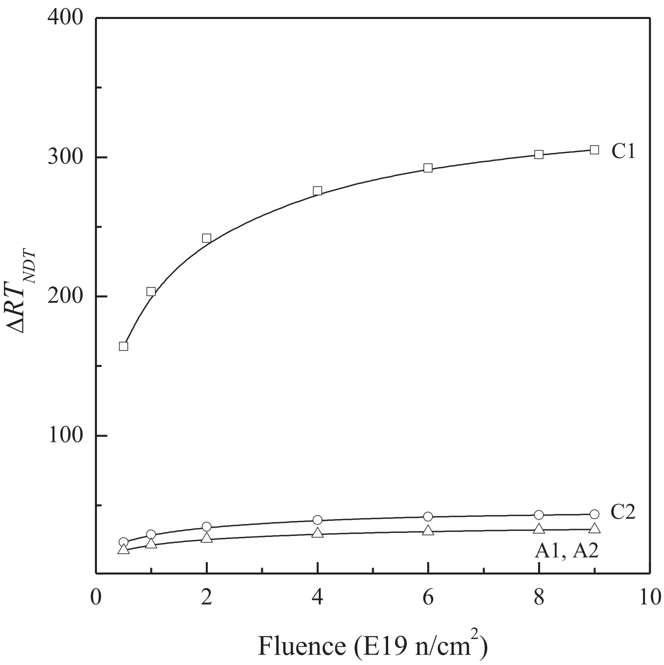

Fig. 6. ΔRT_{NDT} with respect to the fluence.

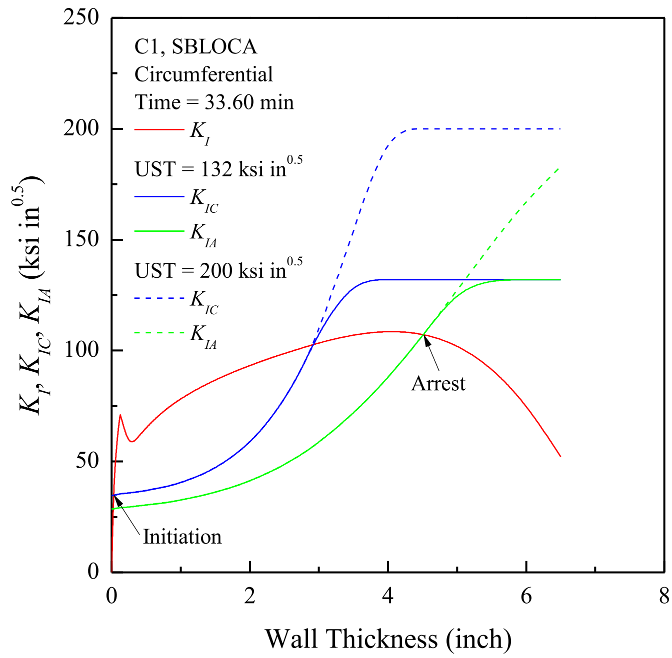

Fig. 8. Comparison of stress intensity factor and fracture toughness profiles for C1.

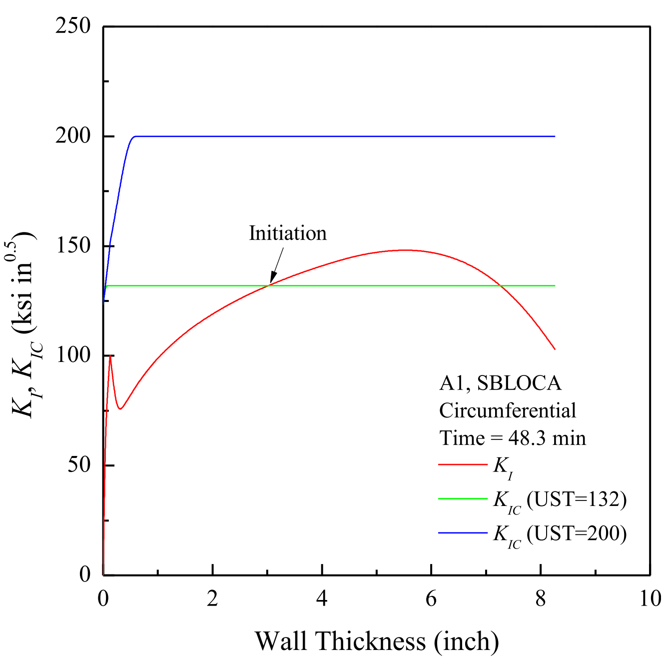

Fig. 7. Comparison of stress intensity factor and fracture toughness profiles for A1.

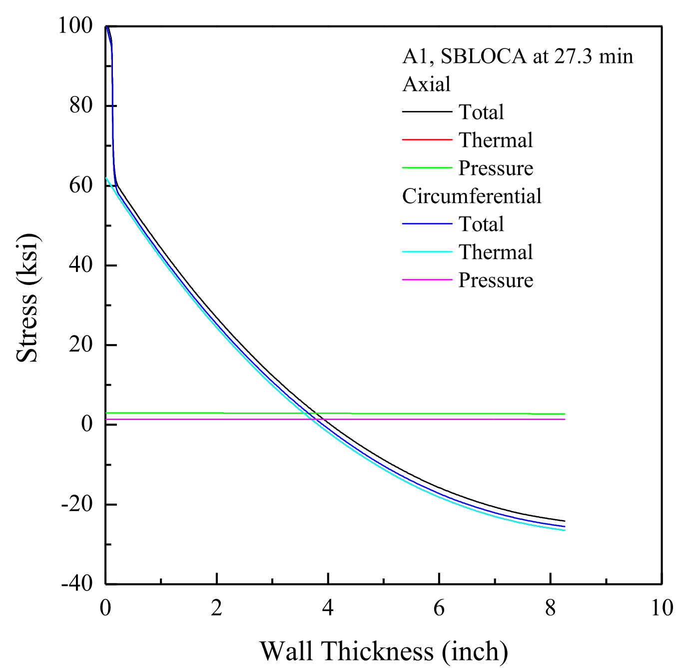

Fig. 9. Stress profiles through the vessel wall at 27.3 min for SBLOCA.

2.4. PFM codes

Four participants used different codes for analysis as shown in Table 3. Regarding flaw orientation, FAVOR assumes all inner surface breaking cracks to be circumferentially oriented. Therefore, P4 performed analyses only for 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 analysis codes. Temperature and axial stress variations along the vessel wall are used to get the stress intensity factors. Also temperature distributions along the vessel

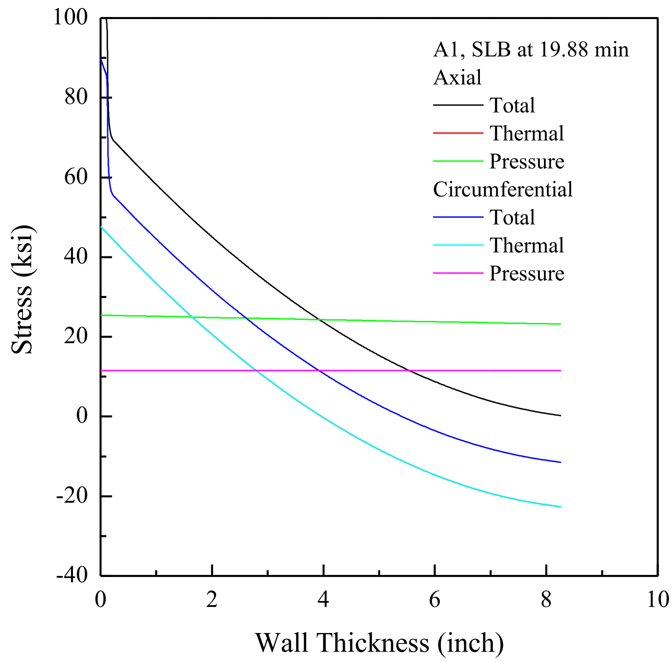

Fig. 10. Stress profiles through the vessel wall at 19.88 min for SLB.

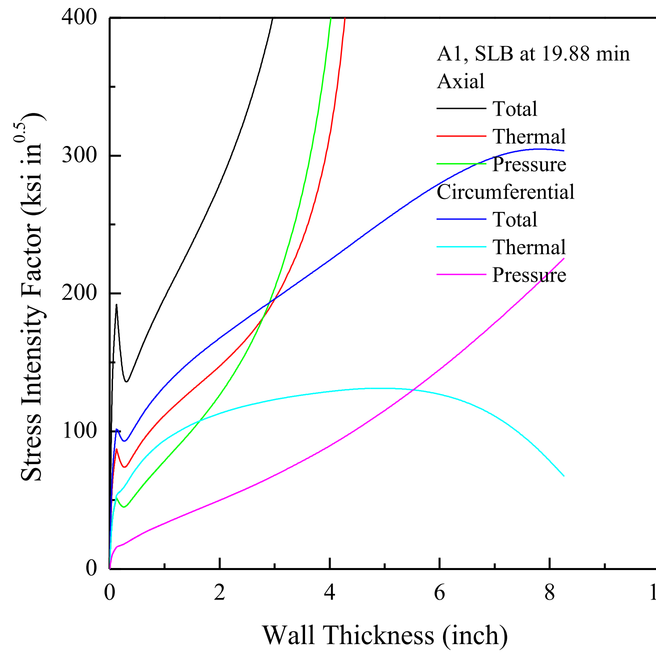

Fig. 12. Stress intensity factor profiles through the vessel wall at 19.88 min for SLB.

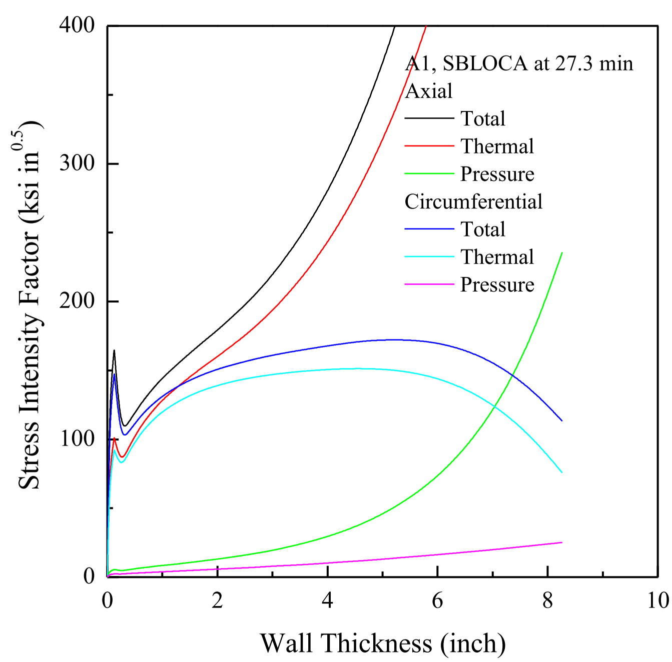

Fig. 11. Stress intensity factor profiles through the vessel wall at 27.3 min for SBLOCA.

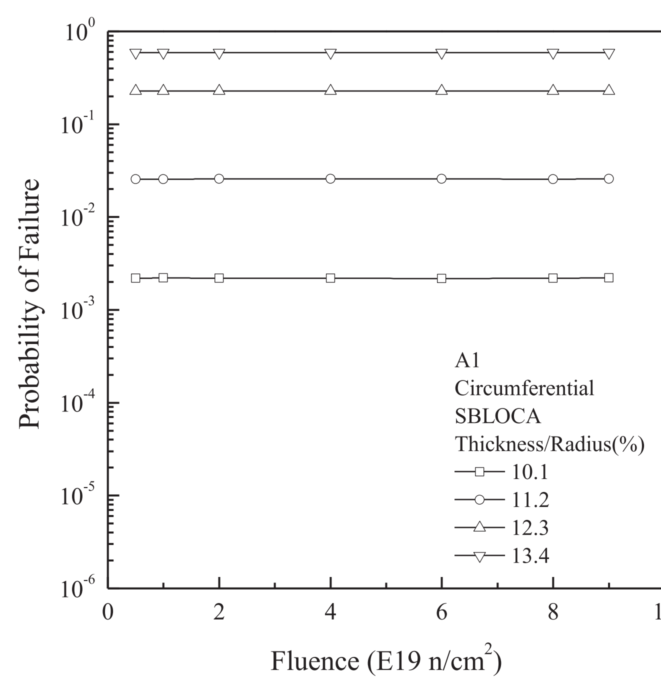

Fig. 13. Failure probabilities with respect to wall thickness.

wall are used to get the fracture toughness. The stress intensity factor and fracture toughness are compared to determine the propagation of the flaw generating the failure of the vessel, which is used to calculate the probability of the vessel failure.

Comparisons of failure probabilities between participants are shown in Figs. 2 and 3 for C1 and A1 plants. The probabilities of vessel failures are shown in Fig. 4 by P1. There are similarities of the trend with respect to the fluence level. But in some cases the difference of probabilities is more than one order of magnitude, which is expected and naturally accepted from the simulation technique to calculate probabilities with different codes. For example, K_I, K_{IC}, and K_{IA} profiles through the vessel wall at a specific time can be shown in Fig. 5. The flaw initiates when K_I is larger than K_{IC} from the thickness of 0.03 inches. It grows a certain distance until K_I is larger than K_{IA}. The flaw is arrested when K_I is smaller than K_{IA}; in this case at 3.70 inches. By repeating millions of simulations of this procedure, probabilities for failure are determined. That's why there are some little uncertainties in the result comparisons between participants.

Another points to be considered are the temperature and stress profiles through the vessel wall. With a small difference of

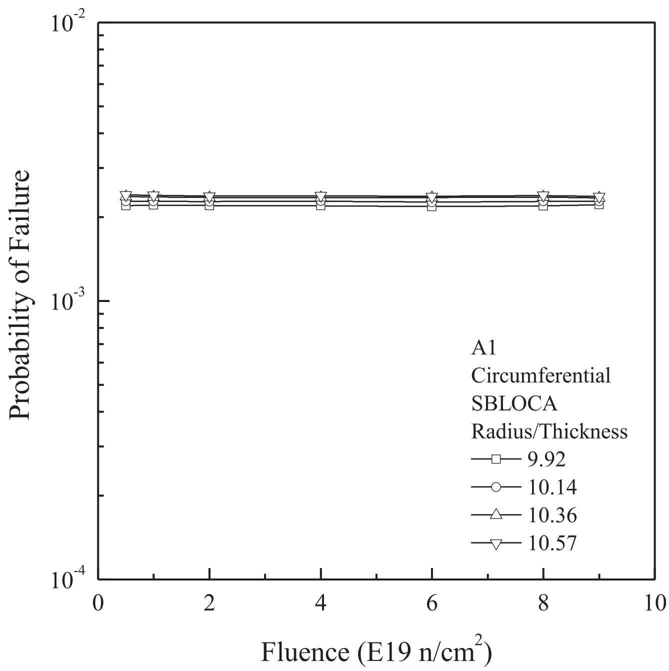

Fig. 14. Failure probabilities with respect to inner radius.

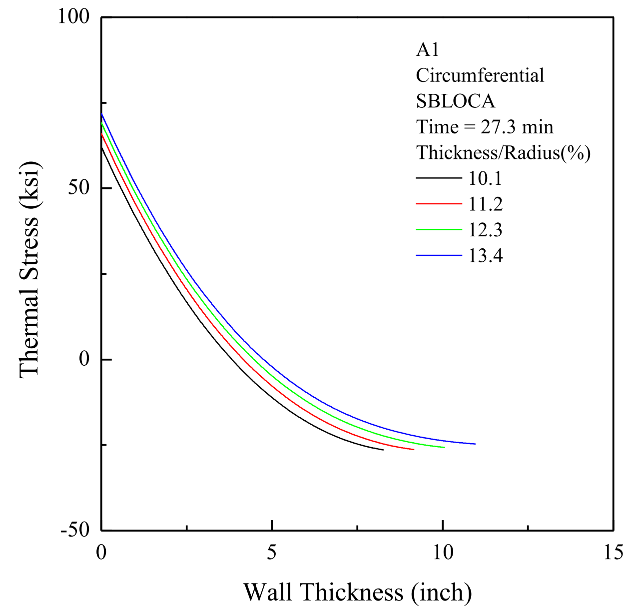

Fig. 16. Thermal stress profiles through the vessel wall.

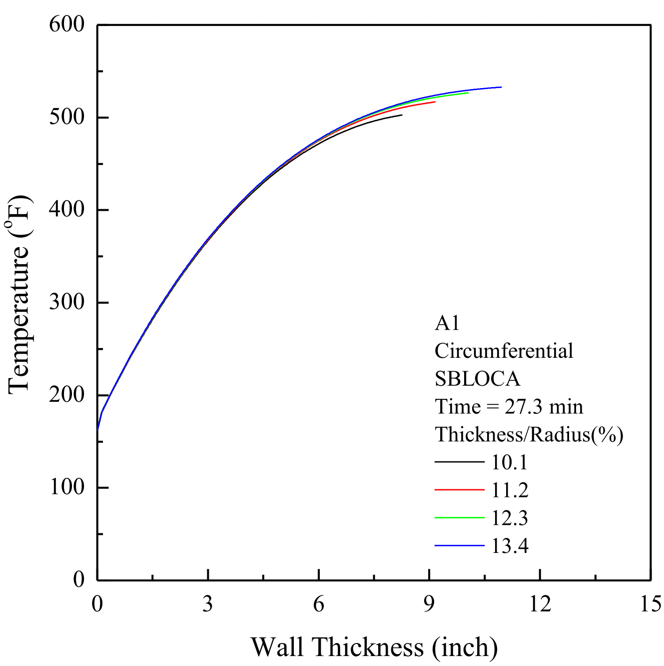

Fig. 15. Temperature profiles through the vessel wall.

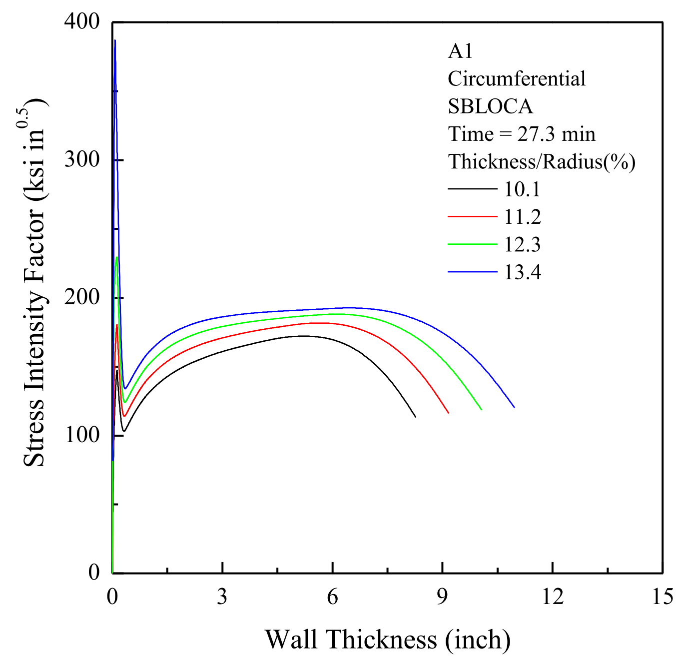

Fig. 17. Stress intensity factor profiles through the vessel wall.

temperature profile, there will be a difference of stress followed by the difference of K_I, K_{IC}, and K_{IA}, which will generate a different probability. For example, the stress intensity factor and fracture toughness are determined for the temperature and stress profiles, as in Fig. 5, and from these profiles the initiation of the flaw specified occurs and propagates until the arrest point where K_I is lower than K_{IA}. These procedures are repeated several times to calculate the failure probability, complicating exact estimation of the difference of results from different codes.

Especially, the underestimation of the failure probabilities obtained by P2 is due to the specific characteristics of the PFM code used for analysis. VISA-II used by P2 has input limitations for transient data. The pressure and temperature histories are input by a polynomial or exponential expression instead of a piecewise data point. In addition, constant heat transfer coefficient is used in the analysis, adding more difference for the estimation of the temperature in the vessel wall. This will generate a difference of temperature and stress followed by the stress intensity factor and fracture toughness. The effect is significant for the transient with a rapid variation of pressure and temperature. The pressure and temperature of PTS and SLB change more rapidly than those of SBLOCA. And therefore, the difference of failure probabilities by P2 is greater

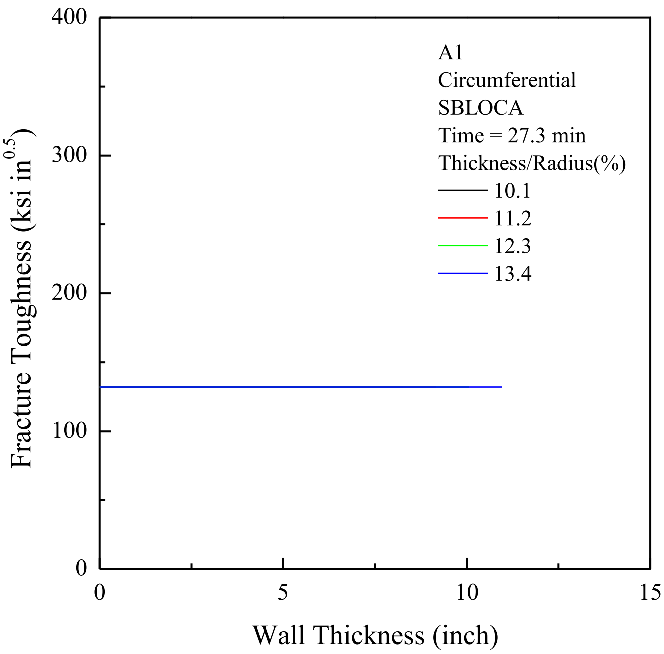

Fig. 18. Fracture toughness profiles through the vessel wall.

in PTS and SLB than in SBLOCA, as shown in Figs. 2 and 3.

Vessel failure probabilities increased with increasing fluence for C1, as expected. For other types of reactors, almost the same probabilities with respect to the fluence level were obtained. This is because the copper contents are so small compared with that of C1, resulting in little effect of fluence on the mean value of the adjustment in reference temperature caused by irradiation, ΔRT_{NDT}, defined as $\Delta RT_{NDT} = (\mathrm{CF})\, f^{(0.28-0.10 \log f)}$, where CF is the chemistry factor, a function of copper and nickel content given in US NRC Regulatory Guide 1.99, Rev. 2 [11], and f is the neutron fluence at any depth in the vessel wall (10^{19} n/cm^2, E > 1 MeV). Fig. 6 shows ΔRT_{NDT} with respect to fluence for four reactors, from which the vessel failure probabilities are expected to be almost the same with respect to the fluence for C2, A1, and A2. The materials for these reactors will not be irradiated too much with increasing fluence due to the very low contents of copper.

All results are obtained based on the most conservative input parameters in the analysis for the licensing purpose and therefore failure probabilities calculated in this study may be excessively overestimated. For example, upper shelf fracture toughness is assigned to be 132ksi$\sqrt{\text{in}}$, which is estimated for the old conventional reactor vessel. NRC indicated that the upper shelf of 200ksi$\sqrt{\text{in}}$exceeds after consideration of irradiation data based on considerably more data and suggested to improve the ASME code [12]. Therefore, if 200ksi$\sqrt{\text{in}}$ is used for the analysis, there are no vessel failures in A1 and A2 plants, as shown in Fig. 7. There is no failure for the case of upper shelf fracture toughness of 200 ksi$\sqrt{\text{in}}$at time = 48.3 min even though the failure is observed for 132ksi$\sqrt{\text{in}}$. This means that plant specific input data should be used to obtain more realistic failure probabilities for the final decision of structural integrity of a reactor vessel due to PTS events.

For the case of C1 plant, which has a smaller wall thickness than advanced reactors, the failure probabilities are almost the same irrespective of the upper shelf fracture toughness. In the profiles of K_I, K_{IC}, and K_{IA} between two values of upper shelf fracture toughness, the flaw initiates and is arrested at the same point, as shown in Fig. 8. With small wall thickness, the temperature at the inner surface of the wall decreases more rapidly than in the case of a thick wall, resulting in small fracture toughness. Therefore, it takes some time to reach the upper shelf value, and during this time most failures occur. That is why almost the same failure probabilities are obtained for C1 plant irrespective of the upper shelf fracture toughness.

By comparing failure probabilities among transients, SBLOCA and SLB were found to be the typical transients in the circumferential and axial flaw orientations, which were characterized by rapid cooling and repressurization, respectively. The time of the maximum number of failures for the advanced reactors are around 28 and 20 min after the event initiates, which exactly correspond to the time of rapid cooling and repressurization in SBLOCA and SLB, respectively.

Figs. 9 and 10 show the stress distributions for axial and circumferential flaws at 27.3 min and 19.88 min into SBLOCA and SLB, respectively. Thermal stresses are the same for both flaw orientations. However, pressure stresses for circumferential flaws were about half of those for axial flaws. Therefore, overall stresses for circumferential flaws were consistently lower than those of axial flaws. Figs. 11 and 12 show the calculated stress intensity factors for each stress component represented by Figs. 9 and 10, which shows strong orientation dependency for all flaw sizes. Considering orientation, the failure probabilities of an axial flaw are always higher than those of the circumferential flaw, as expected.

Even though A1 and A2 have the same material properties with different size of vessel, A2 with bigger size is not tolerable to the PTS event. This can be shown in Figs. 13 and 14, where failure probabilities are calculated with respect to the size of vessel, wall thickness and inner radius. The failure probabilities increase by about one order of magnitude by increasing 1% of the thickness to the radius. The effect of increasing the radius to the thickness is almost negligible even though there is a very small increase in the failure probabilities, as shown in Fig. 14.

Fig. 15 shows temperature profiles of SBLOCA of A1 at time = 27.3 min when the maximum number of failures is found. The temperature difference between the inner and outer surfaces of the vessel wall increases with the increasing thickness, resulting in an increase of the thermal stress, as shown in Fig. 16. But the stress due to the internal pressure decreases with increasing thickness, but it is very small compared with the thermal stress. Therefore, the applied stress intensity factor increases with increasing thickness, as shown in Fig. 17. In addition, the fracture toughness increases with an increase of the temperature, but it is almost negligible, as shown in Fig. 18. That is why the failure probabilities of A2 are larger than those of A1.

The event frequencies are coupled with the results of the fracture mechanics analysis to obtain a frequency of vessel through-wall cracking due to PTS. The sequence frequency and conditional through-wall flaw penetration probability are multiplied to give the frequency of through-wall cracking for each initiator as a function of fluence. These will be finally summed over all initiators to provide an integrated frequency of through-wall cracking, the acceptance criterion per reactor year [13].

4. Conclusions

Round robin analyses for vessel failure probabilities due to PTS events are performed by participants from four organizations in Korea. The vessel failure probabilities from the probabilistic fracture mechanics analyses are calculated and compared with each other, generating the following conclusions:

- Failure probabilities generated are based on the conservative input parameters. More realistic values can be obtained with realistic plant specific input data. For example, using 200ksi$\sqrt{\text{in}}$

of the upper shelf fracture toughness gives no vessel failures for the advanced reactors, A1 and A2.

- Depending on the fluence levels, no increase of failure probabilities was expected except C1 due to the improved material quality especially for the low copper content.
- SBLOCA is generally assumed to be the severest event for all types of plant in circumferential flaws. In the axial orientation, SLB is considered to be the severest event. These two events may be considered to be typical transients for the PTS analysis in the future.
- Advanced reactors with good quality of materials are expected to be tolerable to PTS events, but increasing the size of vessel may be another factor to increase the failure probability.

Declaration of competing interest

No Conflicts of Interest.

Acknowledgments

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Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2020.01.028.

References

[1] M.J. Jhung, S.H. Kim, Y.H. Choi, Y.S. Chang, X. Xu, J.M. Kim, J.W. Kim, C. Jang, Probabilistic fracture mechanics round robin analysis on reactor pressure vessels during pressurized thermal shock, J. Nucl. Sci. Technol. 47 (12) (2010) 1131–1139.

[2] OECD/NEA, Comparison Report of RPV Pressurized Thermal Shock International Comparative Assessment Study (PTS ICAS), NEA/CSNI/R(99)3, NEA Committee on the Safety of Nuclear Installations, Paris, 1999.

[3] OECD/NEA, Proceedings of the CSNI Workshop on Structural Reliability Evaluation and Mechanical Probabilistic Approaches of NPP Components, NEA/CSNI/R(2007), vol. 18, NEA Committee on the Safety of Nuclear Installations, Paris, 2008.

[4] IAEA, Pressurized Thermal Shock in Nuclear Power Plants: Good Practices for Assessment, IAEA-TECDOC-1627, International Atomic Energy Agency, Vienna, 2010.

[5] KINS, PFM Round Robin Analysis on RPV Integrity During PTS by Korean Participants, KINS/RR-686, Korea Institute of Nuclear Safety, Daejeon, 2009.

[6] KINS, 2016, A-Pro2: Phase 2 ASINCO Project for Probabilistic Fracture Mechanics Analysis – Korea Results, KINS/RR-1483, Korea Institute of Nuclear Safety, Daejeon.

[7] KINS, Reactor Probabilistic Integrity Evaluation (R-PIE) Code: User's Guide, KINS/RR-545, Korea Institute of Nuclear Safety, Daejeon, 2008.

[8] F.A. Simonen, K.I. Johnson, A.M. Liebetrau, D.W. Engel, E.P. Simonen, VISA-II: A Computer Code for Predicting the Probability of Reactor Vessel Failure, NUREG/CR-4486(PNL-5775), Pacific Northwest Laboratory, Richland, Washington, 1986.

[9] J.M. Kim, B.S. Lee, T.H. Kim, Y.S. Chang, Development of probabilistic fracture mechanics analysis codes for reactor pressure vessels considering recent embrittlement model and calculation method of SIF – progress of the work, in: PVP2016-63128, 2016 ASME Pressure Vessels and Piping Conference, Vancouver, Canada, 2016.

[10] P.T. Williams, T.L. Dickson, B.R. Bass, H.B. Klasky, Fracture Analysis of Vessels – Oak Ridge FAVOR, v16.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations, ORNL/LTR-2016-309, Oak Ridge National Laboratory, Oak Ridge, 2016.

[11] USNRC, *Radiation Embrittlement of Reactor Vessel Materials*, Regulatory Guide 1.99, Rev. 2, US Nuclear Regulatory Commission, Washington, DC, 1988.

[12] M. Kirk, M. Erickson, W. Server, G. Stevens, R. Cipolla, "Assessment of Fracture Toughness Models for Ferritic Steels Used in Section XI of the ASME Code Relative to Current Data-Based Models," PVP2014-28540, 2014 ASME Pressure Vessels and Piping Conference, Anaheim, California, 2014.

[13] USNRC, Format and content of plant-specific pressurized thermal shock safety analysis reports for pressurized water reactor, in: Regulatory Guide 1.154, US Nuclear Regulatory Commission, Washington, DC, 1987.