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Original Article

Vessel failure sensitivities of an advanced reactor for SBLOCA

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ABSTRACT

Plant-specific analyses of an advanced reactor have been performed to assure the structural integrity of the reactor pressure vessel during transient conditions, which are expected to initiate pressurized thermal shock (PTS) events. The vessel failure probabilities from the probabilistic fracture mechanics analyses are combined with the transient frequencies to generate the through-wall cracking frequencies, which are compared to the acceptance criterion. Several sensitivity analyses are performed, focusing on the orientations and sizes of cracks, the copper content, and a flaw distribution model. The results show that the integrity of the reactor vessel is expected to be maintained for long-term operation beyond the design lifetime from the PTS perspective using the design data of the advanced reactor. Moreover, a fluence level exceeding 9×10^{19} n/cm^2 is found to be acceptable, generating a sufficient margin beyond the design lifetime.

1. Introduction

The advanced power reactor developed in Korea is an evolutionary pressurized water reactor based on the well-proven Korean Standard Nuclear Power Plant design. It incorporates a number of design modifications and improvements to meet the utility need for enhanced safety. It was developed to meet basic design requirements of, for example, 4000 MWth of rated thermal power, a 60 year-life time, and a tenfold lower probability of core damage and accidental radiation release compared to current nuclear power plants [1].

There are many transients considered during the design stage of an advanced reactor. One of them is a pressurized thermal shock event, which represents a significant challenge to the structural integrity of the reactor pressure vessel (RPV) of a pressurized water reactor. Severe cooling of the core occurring together with, or followed by, pressurization presents the possibility of reactor vessel failure, especially for an aged reactor or one operating beyond its designed lifetime.

To ensure the structural integrity of the reactor pressure vessel during a PTS, deterministic approaches have mainly been employed. However, the results obtained are too conservative to allow a rational evaluation of plant safety due to the accumulation of conservatism in related factors. Therefore, the probabilistic fracture mechanics analysis has been widely used for quantitative evaluations of the vessel failure risk associated with a pressurized thermal shock [2–4].

In this study, plant-specific analyses were performed for an advanced reactor to ensure the structural integrity of the reactor pressure vessel during a transient event, which is expected to initiate a PTS event. The vessel failure probabilities from the probabilistic fracture mechanics analyses are calculated and combined with the transient frequencies to generate the through-wall cracking frequencies, which were compared to the acceptance criterion. By performing several sensitivity analyses, the integrity of the reactor vessel was investigated to determine the fluence level allowed during its operation.

2. Problem definition

2.1. Description of reactor

The reactor vessel of the advanced reactor is composed of a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head. The major design improvements incorporated in the reactor vessel design include an enhancement of its core monitoring capability, larger operating margins, a higher power level, and lower failure rates of fuel elements for greater plant availability and reliability. The life-time of the reactor pressure vessel is improved to 60 years dueto the use of low carbon steel, which has lower

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contents of Cu, Ni, P, and S compared to the previous designs [1].

2.2. Transient for analysis

A small break loss of coolant accident (SBLOCA), one of transients considered in the OECD/NEA PROSIR project [2], is considered in this study. The temperature and pressure decrease very rapidly during this event, as shown in Fig. 1, reaching final values of 7 °C and 2 MPa, respectively. This transient has no repressurization stage characterized by a typical PTS transient. Therefore, it is expected that the temperature is the major factor affecting the results of an analysis.

2.3. Flaw distribution

Marshall [5] used the following equation to calculate the probability of a crack with depth a.

$$P(a) = 4.06\exp\left(-4.06\ a\right) \tag{1}$$

The cumulative flaw density function describing the probability of the existence of a crack larger than a is expressed by the as following equation:

$$f(a) = \int_0^a 4.06\ \exp(-4.06a)da \tag{2}$$

If integrated to the entire flaw depth range, the above equation results in a value of precisely 1, indicating that Eq. (2) is associated with a single flaw. The flaw distribution and size not considering an inspection can be calculated. Moreover, the probability of non-detection during a pre-service inspection is defined as;

$$B(a) = \varepsilon + (1-\varepsilon)\ \exp\left(-\mu\ a\right) \tag{3}$$

where $\varepsilon = 0.005$ and $\mu = 0.113386\ \text{mm}^{-1}$, and is valid for edge cracks and semi-elliptical cracks with depth over length $(a/l) = 1/6$. Therefore, the flaw distribution and size after inspection can be calculated by incorporating Eq. (3) into Eq. (1). As the detected flaws are effectively removed from the population, the net effect is to reduce the number of flaws while also to modify the flaw distribution. After some rearrangement, the cumulative flaw distribution for Marshall with an inspection can be expressed by the following equation.

$$f(a) = \int_0^a [0.0346\ \exp(-4.06a) + 6.88\ \exp(-6.94a)]da \tag{4}$$

As before, if integrated to the entire flaw depth range, Eq. (4) will return a result of exactly 1, but the associated number of flaws is 0.5863 instead of 1 owing to the aforementioned reason. The flaw distribution and size considering an inspection can be calculated as shown in Fig. 2.

2.4. Vessel information

The pressure vessel considered in this study has a thickness of 233 mm including the cladding and an inner radius of 2315 mm. Its material is SA508 Cl.3. The contents of copper and nickel which augment radiation embrittlement are 0.02 and 0.03 wt%, respectively. Their corresponding uncertainties are arbitrarily chosen to be 20% of the mean values. K_{IC}, K_{IR} and ΔRT_{NDT} normal distributions are assumed to be truncated between $+3SD$ and $-3SD$, where SD is the standard deviation. The crack postulated is a surface breaking crack with infinite through-clad in the axial or circumferential orientation.

3. Analysis

3.1. RT_{PTS}

The US NRC introduced the concepts of RT_{PTS} the reference temperature of nil-ductility transition, RT_{NDT}, evaluated for the end-of-life (EOL) fluence for beltline materials, and defined the PTS screening criteria as 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials in 10CFR50.61 [6]. In addition, for each pressurized water 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

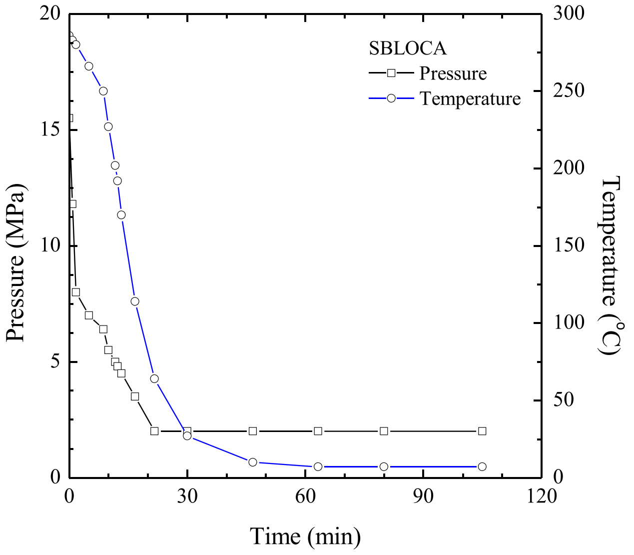

Fig. 1. Transient histories of a SBLOCA.

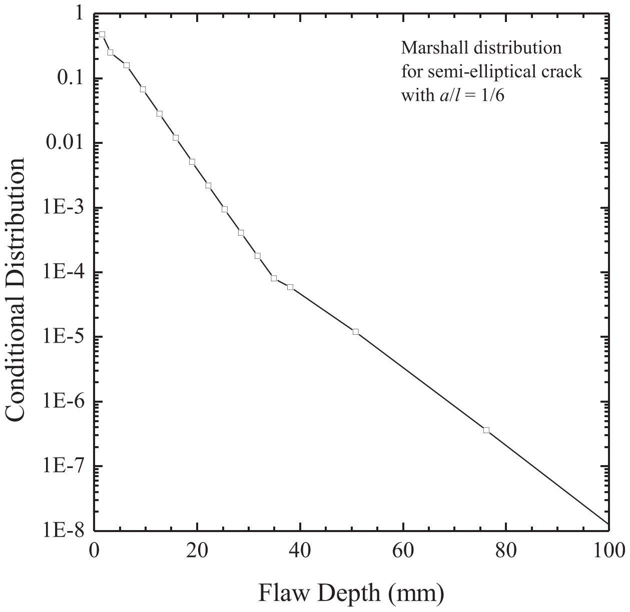

Fig. 2. Flaw distribution and size for the Marshall model.

that are reasonably practicable to avoid exceeding the PTS screening criterion.

The reference temperature of the nil-ductility transition RT_{NDT} is given by the following expression according to the US NRC Regulatory Guide 1.99, Rev.2 [7]:

$$RT_{NDT} = RT_{NDT0} + \Delta RT_{NDT} + M \tag{5}$$

Here, RT_{NDT0} is the mean for the initial (unirradiated) value of RT_{NDT} for the RPV region where 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. (6) is the increase in RT_{NDT} due to irradiation-induced embrittlement, which is a function of the copper and nickel contents and the neutron fluences,

$$\Delta RT_{NDT} = (\mathrm{CF})\, f^{\,(0.28-0.10\ \log f)} \tag{6}$$

where CF is the chemistry factor, a function of the copper and nickel content given in US NRC Regulatory Guide 1.99, Rev.2 [7], and f is the neutron fluence at any depth in the vessel wall (10^{19} n/cm^2, E > 1 MeV).

The fluence factors are shown in Fig. 3, where the fluence factor does not change too much for fluence exceeding 8×10^{19} n/cm^2. Therefore, it is expected to be sufficient to investigate the effect of fluence on the vessel failure probability up to a fluence level of 9×10^{19} n/cm^2.

3.2. R-PIE code

A probabilistic fracture mechanics code known as R-PIE (Reactor - Probabilistic Integrity Evaluation) is developed for quantitative risk assessments of RPVs during the pressurized thermal shocks [8]. The R-PIE code is similar to the previously developed VINTIN [9] but contains user-friendly features as it is written in Visual Basic.

A variety of statistical parameters, in this case the 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 an equation derived from the lower-bound fracture toughness [10].

Using the mean values and associated uncertainties, the fracture toughness values are simulated for comparison 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 propagate 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 vessel failure for a specific thermal hydraulic boundary condition is determined.

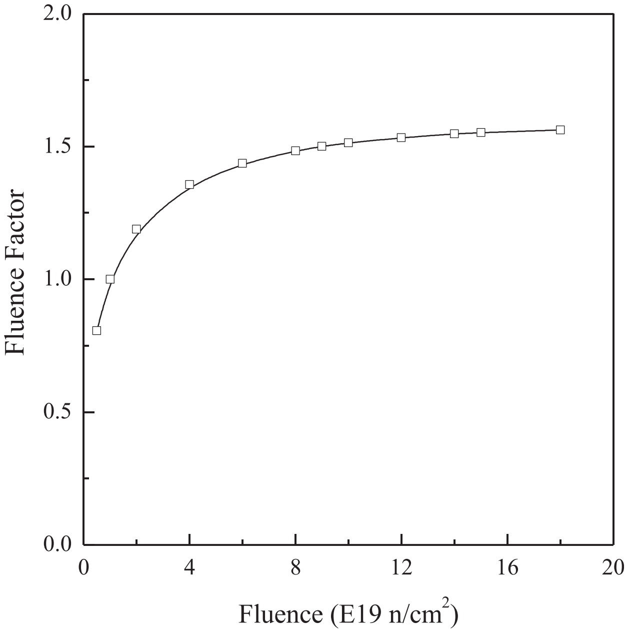

Fig. 3. Fluence factor with respect to the fluence.

3.3. Analysis matrix

The circumferential infinite crack with the Marshall model of flaw distribution considering inspection data is chosen as a basic case. Several analyses are done for a sensitivity study, as shown in Table 1.

4. Results and discussion

The temperature distributions are calculated and the stress analyses due to these temperature distributions and internal pressure are conducted using the R-PIE code. Temperature and axial stress variations along the vessel wall are used to determine the stress intensity factors. Moreover, the temperature distributions along the vessel wall are used to calculate the fracture toughness. The stress intensity factor and fracture toughness are compared to determine the propagation of the crack generating the failure of the vessel, which is then used to calculate the probability of the failure of the vessel.

The probabilities of vessel failure are shown in Fig. 4. Nearly identical vessel failure probabilities were obtained with an increase in the fluence. This arises because the copper content is very low, resulting in little effect of the fluence on the mean value of the adjustment of the reference temperature caused by irradiation, ΔRT_{NDT}, as defined by Eq. (6).

Fig. 5 shows ΔRT_{NDT} with respect to the fluence, from which the vessel failure probabilities are expected to be nearly identical with respect to the fluence for a copper content of less than 0.05 wt%. The material will not be irradiated too much due to the very low content of copper. The vessel failure probabilities are shown in Fig. 6 with respect to the copper contents. This confirms again that the failure probabilities are nearly identical at copper content of less than 0.05 wt%.

The effect of crack orientation on the failure probability is shown in Fig. 4. As expected, the failure probabilities for an axial crack are higher than those for a circumferential crack by an order of 1 due to the increase in the axial stress, resulting in a stress intensity factor.

Crack sizes defined by the length over depth are investigated,

Table 1
Analysis matrix for sensitivity.

Subject	Values
Crack orientation	Circumferential*, Axial
Crack size (length/depth)	Infinite*, 12, 6
Inspection data for the Marshall model	Considering*, Not considering
Fluence (10^{19} n/cm^2)	0.5–9 (0.5, 1, 2, 4, 6, 8, 9)
Copper content (wt%)	0.02, 0.05, 0.10, 0.20, 0.30

[a] Base case.

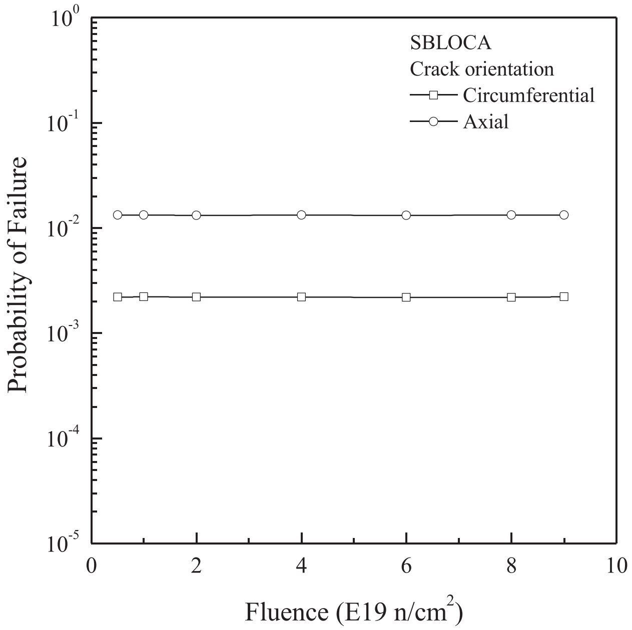

Fig. 4. Probability of vessel failure with respect to the crack orientation.

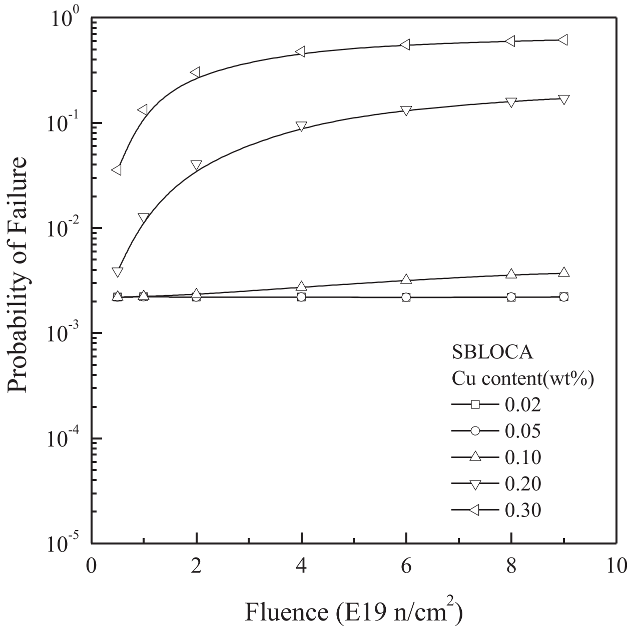

Fig. 6. Probability of vessel failure with respect to the copper content.

and their failure probabilities are shown in Fig. 7. No failure occurred for a length of less than 12 over the depth of the crack. Therefore, there will be no failure due to the SBLOCA if a more realistic crack size is considered in the analysis by eliminating conservatism. An infinite crack size, which is used for regulatory purposes, is considered to generate an overly conservative failure probability.

The flaw distribution of the Marshall model can be defined considering the inspection data obtained from the pre-service inspection. If these data are more realistically used to determine the flaw distribution, the failure probabilities decrease by an order of 1 as shown in Fig. 8. Therefore, the inspection data of the flaw distribution is considered to be an important factor to use when calculating the vessel failure probability.

Histograms of the number of initiations, arrests, and failures for a SBLOCA are shown in Figs. 9 through 11, respectively. As can be expected from the transient histories, the highest number of initiations is obtained between 20 and 30 min, when the temperature decreases very rapidly. Most of the flaws initiated are arrested before 30 min, but some of them are not arrested, as shown in Fig. 10. When a flaw initiates, propagation and arrest occur repeatedly to arrive at through-wall cracking [3]. This expalains why the number of arrests exceeds the number of initiations. If the initiated flaw propagates and arrives at or beyond 75% of the vessel wall, it is considered to have failed. The highest number of failures is observed between 20 and 25 min when the temperature

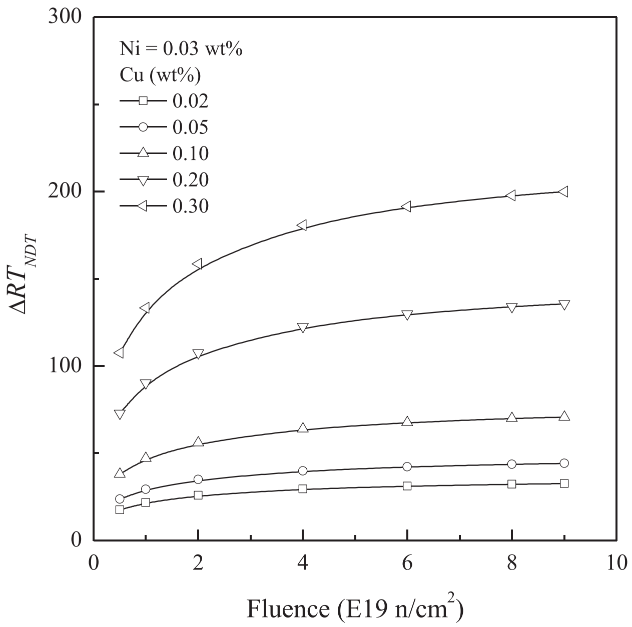

Fig. 5. ΔRT_{NDT} with respect to the fluence.

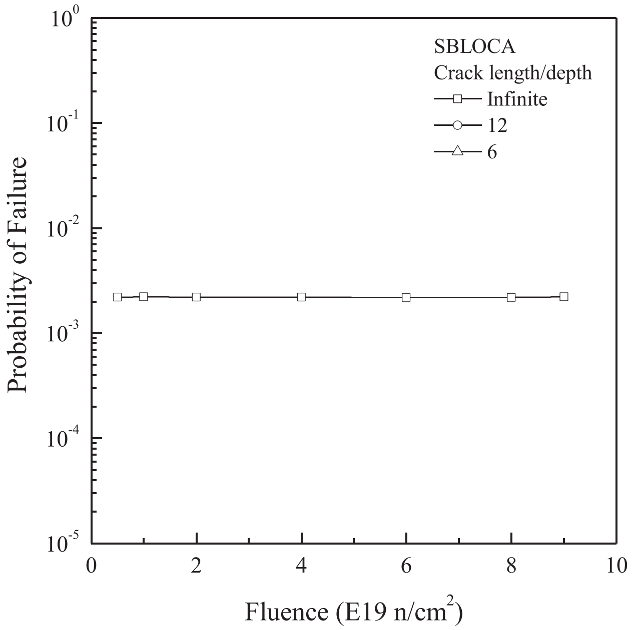

Fig. 7. Probability of vessel failure with respect to the crack size.

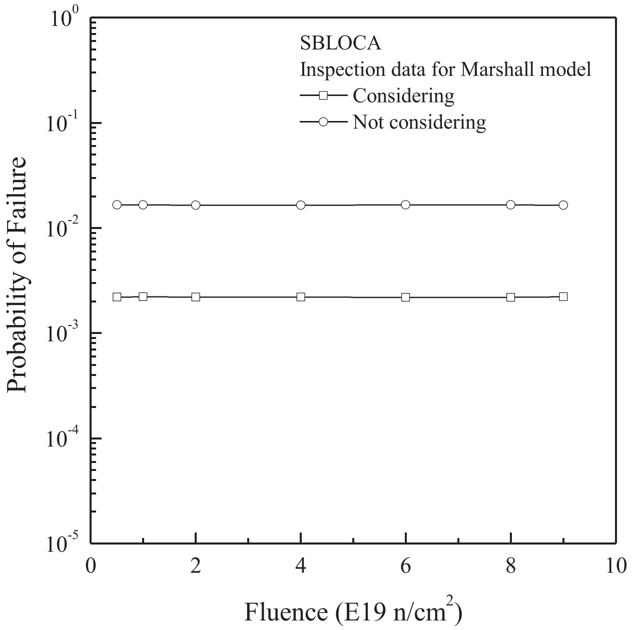

Fig. 8. Probability of vessel failure with respect to the inspection data for the Marshall model.

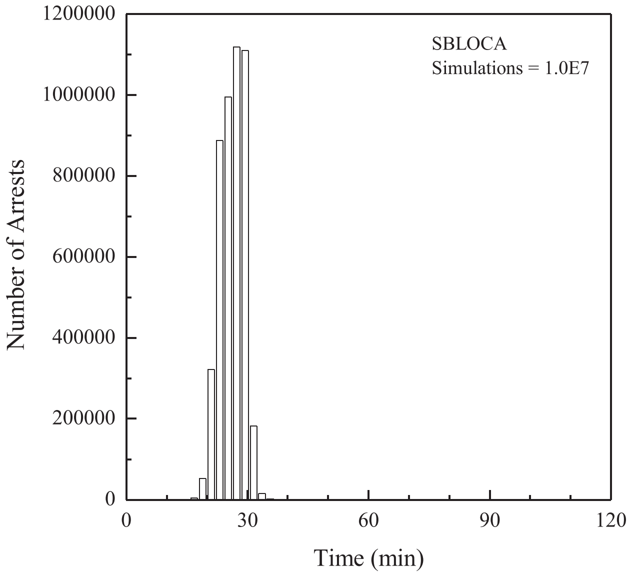

Fig. 10. Number of arrests for a SBLOCA.

decreases with the generation of low fracture toughness. Beyond 30 min, relatively few failures are observed due to decreasing stress intensity factor resulting from the low pressure without repressurization.

A histogram of the number of failure observations is shown in Fig. 12. It is expected that most failures are observed at low flaw depths of less than 40 mm, which is about 17% of the wall thickness. No failures are observed for a flaw depth larger than 45 mm.

The event frequencies are coupled with the results of the fracture mechanics analysis to obtain the frequency of vessel through-wall cracking (TWC) due to PTS. The sequence frequency and conditional through-wall crack penetration probability are multiplied to determine the frequency of through-wall cracking for each initiator as a function of the fluence. These values are summed over all initiators to provide the integrated frequency of through-wall cracking, and is compared to the acceptance criterion of 5×10^{-6} per reactor year [11].

By multiplying the sequence frequency of 5.0E-4 [12] and the conditional through-wall crack penetration probability, the TWC frequencies of a SBLOCA are obtained as a function of the fluence, as shown in Figs. 13 through 16 for various sensitivity factors, in this caes the crack orientation, crack size, copper content, and inspection data, respectively. The fluence levels exceeding the acceptance criteria are determined from Figs. 13 through 16.

The integrity of the RPV is maintained up to a fluence level of more than 9×10^{19} n/cm^2 for the case of an infinite circumferential crack with a copper content of 0.02 wt%, which is typical data for the type of advanced reactor studied here. This value was obtained from the most conservative assumption of an infinite crack, allowing the conclusion that the advanced reactor is resistant to a

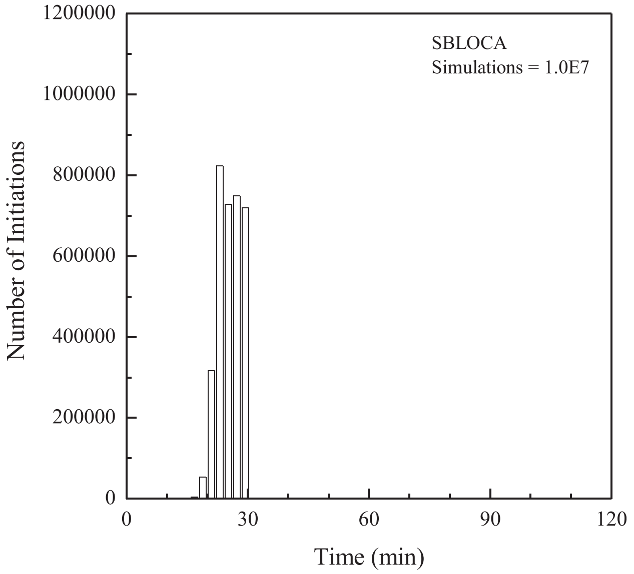

Fig. 9. Number of initiations for a SBLOCA.

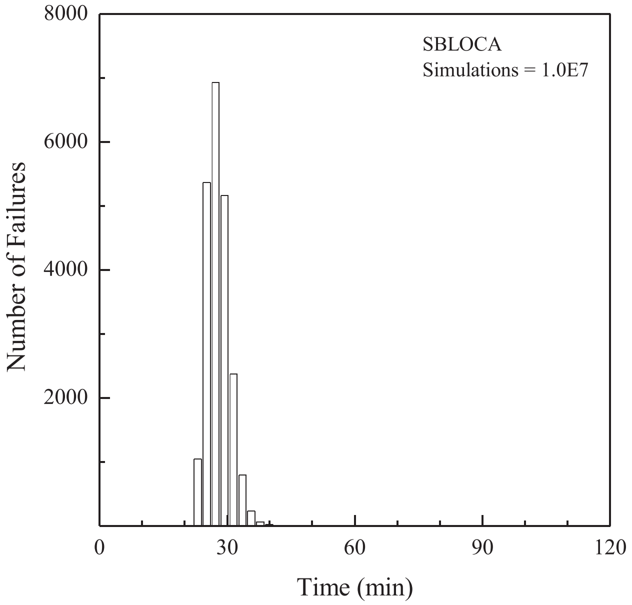

Fig. 11. Number of failures for a SBLOCA.

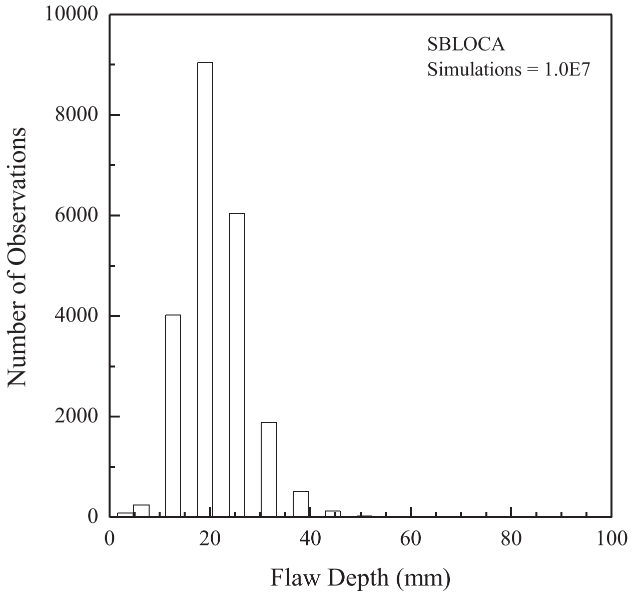

Fig. 12. Histogram of failure observations for a SBLOCA.

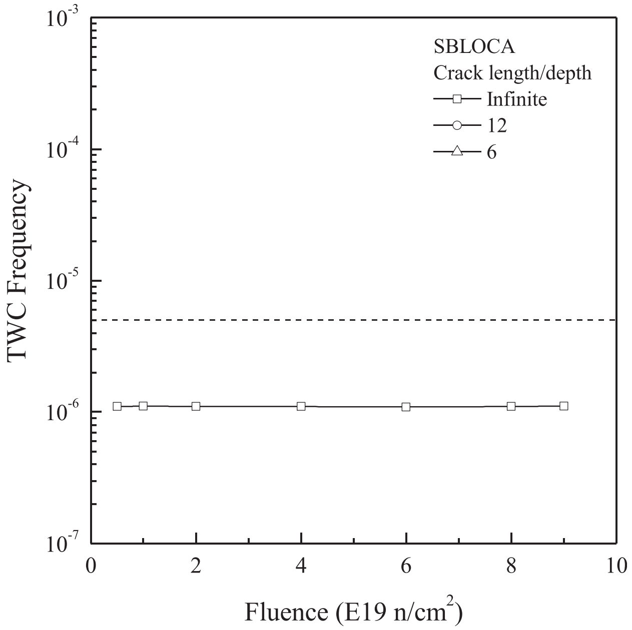

Fig. 14. Through-wall cracking frequencies with respect to the crack size.

SBLOCA during long-term operation in terms of PTS.

If there is an axial crack, or if the reactor vessel has more than 0.20 wt% of copper content, or if inspection data is not available from a pre-service inspection, a more detailed plant-specific analysis needs to be performed. At present, the advanced reactor has a circumferential crack only, a copper content of 0.02 wt%, and good PSI data availability to determine the flaw distribution. Therefore, a long-term operation of the reactor vessel during a PTS transient is expected to present no problems with regard to integrity.

5. Conclusions

Plant-specific analyses of an advanced reactor were conducted to ensure the structural integrity of the reactor pressure vessel during a transient event, which is expected to initiate a PTS event. The vessel failure probabilities from probabilistic fracture mechanics analyses are combined with the transient frequencies to generate the through-wall cracking frequencies, which were then compared to the acceptance criterion. By performing several sensitivity analyses, the conclusions below could be obtained.

- The integrity of the advanced reactor during a SBLOCA is maintained up to a fluence level of more than 9×10^{19} n/cm^2 for a circumferential flaw, with a copper content of less than 0.2 wt%, for a flaw distribution model developed considering inspection data.

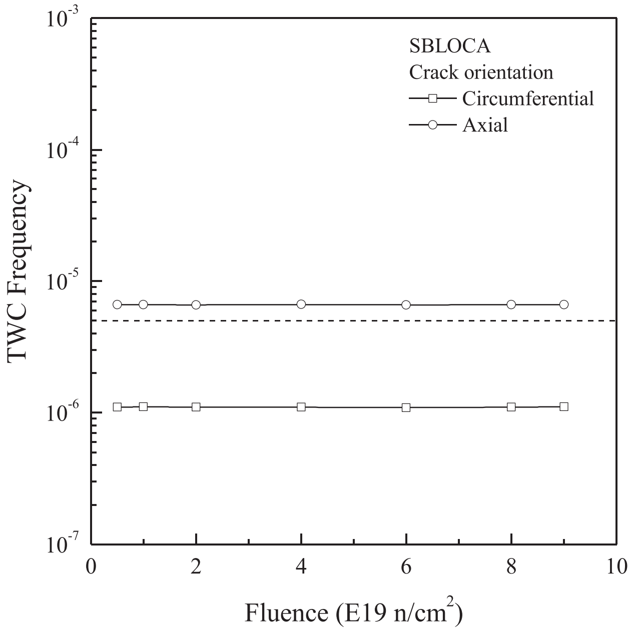

Fig. 13. Through-wall cracking frequencies with respect to the crack orientation.

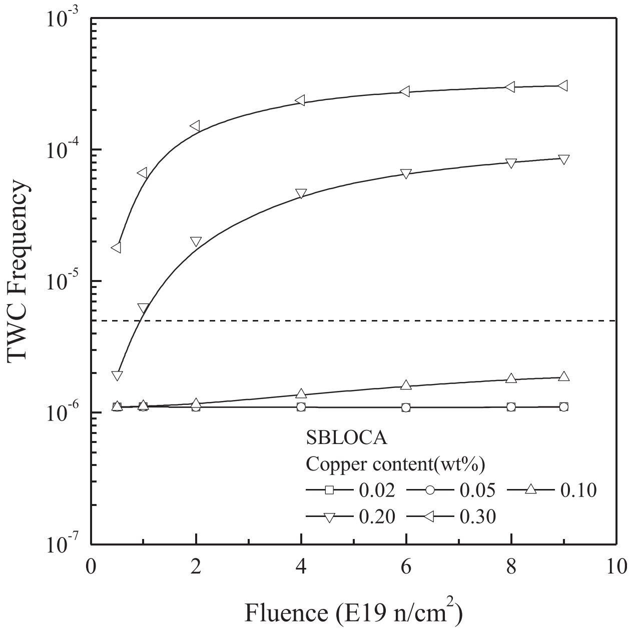

Fig. 15. Through-wall cracking frequencies with respect to the copper content.

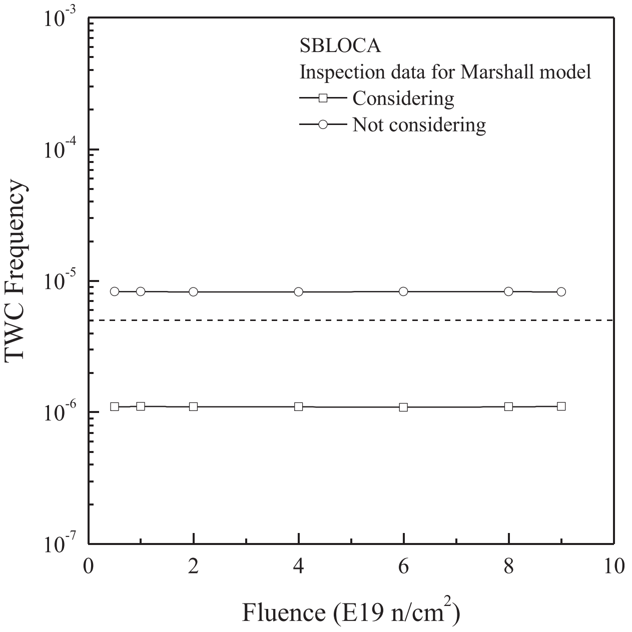

Fig. 16. Through-wall cracking frequencies with respect to the inspection data of the Marshall model.

- Especially for the infinite flaw, which is the most conservative assumption, a fluence level exceeding 9×10^{19} n/cm^2 is acceptable, generating a sufficient margin beyond the design lifetime.

All of these results were obtained based on several assumptions pertaining to the material properties, flaw distribution data, fluence at the end of the design life and the transient data of the pressure, temperature and heat transfer coefficient. If needed, more realistic failure probabilities can be obtained by using more realistic data.

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.2019.05.008.

References

[1] H.G. Kim, K.S. Choi, Design characteristics of the advanced power reactor 1400, in: International Conference on Opportunities and Challenges for Water Cooled Reactors in the 21 Century, IAEA-CN-164, International Atomic Energy Agency, Vienna, 2009, pp. 96–98.
[2] OECD/NEA, in: Proceedings of the CSNI Workshop on Structural Reliability Evaluation and Mechanical Probabilistic Approaches of NPP Components, NEA/CSNI/R(2007)18, NEA Committee on the Safety of Nuclear Installations, Paris, 2008.
[3] IAEA, Pressurized Thermal Shock in Nuclear Power Plants: Good Practices for Assessment, IAEA-TECDOC-1627, International Atomic Energy Agency, Vienna, 2010.
[4] KINS, in: A-Pro2: Phase 2 ASINCO Project for Probabilistic Fracture Mechanics Analysis – Korea Results, KINS/RR-1483, Korea Institute of Nuclear Safety, Daejeon, 2016.
[5] UKAEA, An Assessment of the Integrity of the PWR Pressure Vessels, Second Report by a Study Group under the Chairmanship of Dr. W. Marshall, United Kingdom Atomic Energy Authority, 1982.
[6] N.R.C. US, *Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events*, 10 CFR Part 50.61, U.S. Nuclear Regulatory Commission, Washington, DC, 1996.
[7] N.R.C. US, Radiation Embrittlement of Reactor Vessel Materials, Regulatory Guide 1.99, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, 1988.
[8] KINS, *Reactor Probabilistic Integrity Evaluation (R-PIE) Code: User's Guide*, KINS/RR-545, Korea Institute of Nuclear Safety, Daejeon, 2008.
[9] C. Jang, "Treatment of the thermal–hydraulic uncertainties in the pressurized thermal shock analysis, Nucl. Eng. Des. 237 (2007) 143–152.
[10] ASME, Boiler and Pressure Vessel Code, Section XI, The American Society of Mechanical Engineers, New York, 2004.
[11] N.R.C. US, *Format And Content of Plant-specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactor*, Regulatory Guide 1.154, US Nuclear Regulatory Commission, Washington, DC, 1987.
[12] N.R.C. US, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995, NUREG/CR-5750, U.S. Nuclear Regulatory Commission, Washington, DC, 1999.