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# Fracture Mechanics Analysis for Structural Integrity Assessment of Reactor Pressure Vessel under Pressurized Thermal Shock Condition

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## 1. Introduction

A reactor pressure vessel integrity is ensured by a proper margin between its loads bearing capacity given by the vessel design and material properties and the acting loads, which could occur during a plant operation. Thus, it is designed and manufactured according to strict code requirements in order to ensure its structural integrity.

Before the late 1970s, it was postulated that the most severe thermal shock that a pressurized water reactor vessel must withstand is a large break loss-of-coolant accident (LOCA). In this type of overcooling transient, a low-temperature emergency core coolant would rapidly enter the reactor pressure vessel and cool the vessel wall. The resulting temperature gradient in the vessel wall would cause a significant thermal stress, within the inner surface of the wall. However, the stresses due to a system pressure along with the thermal stresses were not considered, since it was expected that during a large break LOCA, the system would depressurized fast and remain at a low pressure. But the occurrence of a pressurized thermal shock (PTS) at one nuclear power plant showed that some overcooling transients could be accompanied by a re-pressurization of the primary system, which would compound the effects of the thermal stresses. When a system pressure remains high or slowly decreases during thermal shock events, an additional stress from the system pressure could increase the possibility of a crack initiation and propagation. To assure the integrity of a reactor pressure vessel under a PTS event, the PTS rule requires that the $RT_{NDT}$ for any material in a beltline should be lower than the PTS screening criterion.

The objective of this study is to evaluate the structural integrity of a reactor pressure vessel under a PTS condition by applying deterministic fracture mechanics.

## 2. Problem Definition

### 2.1 Geometry

The reactor pressure vessel considered in the analysis is a typical 3-loop PWR, which is made of ASTM A 508 Class 3 with an inner radius of 1994mm, a base metal thickness of 200mm, and a cladding thickness of 7.5mm. The postulated defect as a base case is a through-clad surface-breaking semi-elliptical crack of 19.5mm in depth by 117mm in length for $a/c$=1/3 as shown in Fig. 1. Analysis matrix for the sensitivity of the postulated defect is shown in Table I.

### 2.2 Transient condition

One overcooling transient due to an assumed leak is defined as in Fig. 2, for which axi-symmetric loading conditions are assumed.

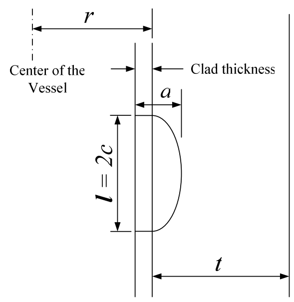


Fig. 1. Schematic illustration of a postulated crack

Table I: Analysis matrix of a postulated defect

| Case | Aspect ratio ( $a/c$ ) | Depth | |
|---|---|---|---|
| | | $a$ | $a/t$ |
| 1 | 1/3 | 19.5 | 0.06 |
| 2 | 1/2 | | |
| 3 | 1/1 | | |

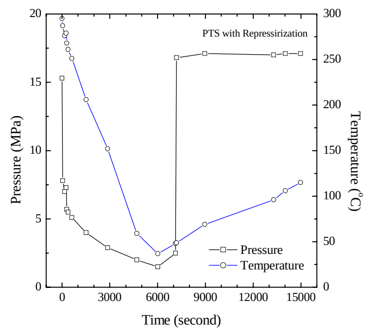


Fig. 2. Transient histories of a PTS with a re-pressurization

## 3. Finite Element Modeling

In this paper, the three-dimensional finite element analyses were performed for the assessment of various cracks in a reactor pressure vessel under a PTS condition. The three-dimensional mesh, generated using the I-DEAS, approximately contains 2,200 elements and 30,000 nodes. Due to the symmetrical boundary condition, only a quarter of the whole reactor pressure vessel was modeled as shown in Fig. 3. The model was designed with 20-node isoparametric quadratic brick elements with reduced Gaussian integration points and 20-node quarter point brick elements for the crack front point.

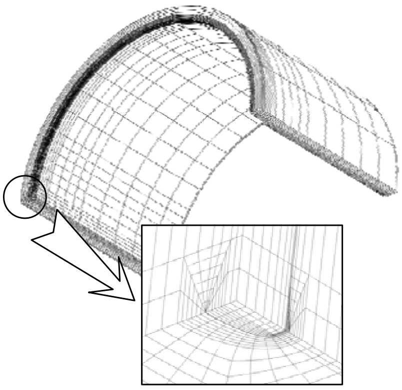

Fig. 3. Three-dimensional finite element mesh for the semi-elliptical surface crack

## 4. Analysis Results

In the PTS event analysis, temperature difference between the inner and outer surfaces of the reactor pressure vessel was considered. Fig. 4 shows the temperature distributions through the vessel wall thickness for Case 1. Due to a difference in the heat conduction of the cladding and base metal, an interface can be clearly found. Fig. 5 shows the stress intensity factor curves for Case 1. The calculated maximum allowable $RT_{NDT}$ is 56℃. The effects of the crack aspect ratio on the stress intensity factor curves are shown in Fig. 6. The stress intensity factor curves decreased with an increasing crack aspect ratio.

## 5. Conclusions

In this paper, the three dimensional finite element analyses were performed to evaluate the integrity of a reactor pressure vessel under a PTS condition. As the crack aspect ratio increases with the same crack depth, the maximum allowable nil ductile transition temperature increases.

## REFERENCES

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[3] The American Society of Mechanical Engineers, Rules for inservice of nuclear power plant components, ASME boiler and Pressure Vessel Code Section XI, Appendix A, 2007.
[4] ABAQUS/Standard Ver.6.7 Software, Dassault systemes.

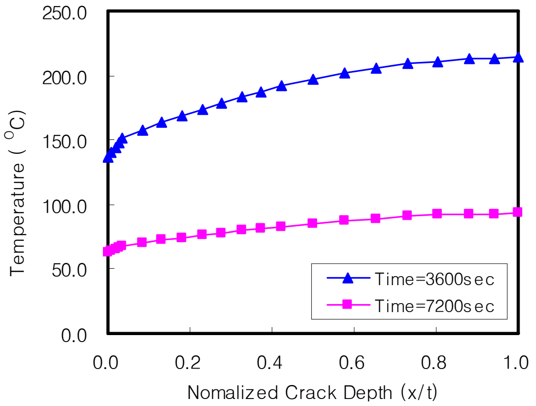


Fig. 4. Variation of the temperature through the reactor pressure vessel wall thickness for Case 1

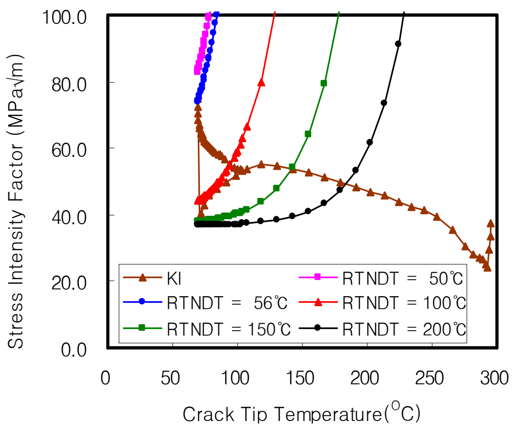


Fig. 5. Determination of the allowable $RT_{NDT}$ by the maximum criteria for Case 1

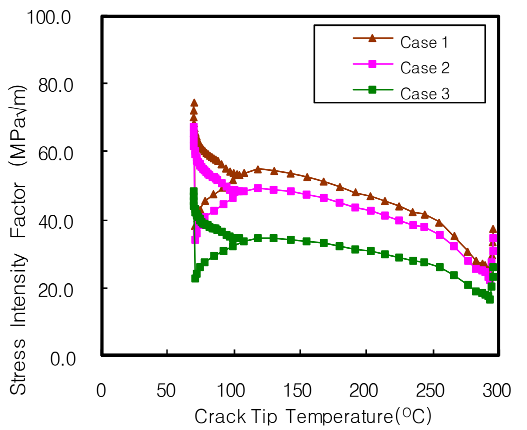


Fig. 6. Comparison of the stress intensity factor with various crack aspect ratios