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Probabilistic fracture mechanics analyses of nuclear pressure vessels and piping : a review of japanese activity

Yagawa G., Yoshimura S.
University of Tokyo, Japan

ABSTRACT

Probabilistic Fracture Mechanics (PFM) approaches are regarded as appropriate methods to rationally evaluate plant life since they can consider various uncertainties. In Japan, some research activity on PFM approaches to the integrity studies of nuclear pressure vessels and piping (PV&P) has been performed since 1987. Its purpose is to establish standard procedures for evaluating failure probabilities of Japanese nuclear PV&P, combining the state-of-the art knowledge on structural integrity of nuclear PV&P and modern computer technology. This paper reviews the activity.

INTRODUCTION

Studies on efficient utilization and life extension of operating nuclear power plants have become increasingly important since ages of the first-generation plants are approaching their design lives. In order to predict a remaining life of plant, it is necessary to select those critical components that strongly influence the plant life, and to evaluate their remaining lives by considering aging effects of materials and other factors. In this regard, the Probabilistic Fracture Mechanics (PFM) approaches are regarded as appropriate methods to rationally evaluate plant life since they can consider various uncertainties such as sizes and shapes of cracks, degradation of material strength due to aging effects, accuracy and frequency of PSI and ISI. Several PFM computer programs have been developed and applied in practical situations in the last two decades [1, 2].

In Japan, one research activity on PFM approaches to the integrity studies of nuclear pressure vessels and piping (PV&P) was initiated in 1987 by the LE-PFM subcommittee organized within the Japan Welding Engineering Society under a subcontract of the Japan Atomic Energy Research Institute, and had continued for three years [3]. The activity was followed by the RC111 research committee organized in the Japan Society of Mechanical Engineers in 1991, and finished in March, 1995 [4]. The researchers from 4 universities, 3 research institutes, 3 utilities, 5 venders and 2 software companies participated the activity. The purpose of the activity was to establish standard procedures for evaluating failure probabilities of Japanese nuclear PV&P, combining the state-of-the art knowledge on structural integrity of nuclear PV&P and modern computer technology such as parallel processing. Within the activity, we have set up the following three kinds of PFM round-robin problems on

(a) primary piping under normal operating conditions,
(b) aged reactor pressure vessel (RPV) under normal & upset operating conditions, and
(c) aged RPV under pressurized thermal shock (PTS) events.

The basic part of the last PTS problems is taken from some of US benchmark problems [5, 6]. For these round-robin problems, various sensitivity analyses were performed to quantify effects of uncertainty of data on failure probabilities. Some of the detailed analysis results can be found in elsewhere [7-12]. This paper reviews the activity mentioned above.

OUTLINE OF PFM ANALYSES

Figure 1 illustrates the flow of a typical PFM analysis. Firstly, some random variables are selected according to an analysis model employed. Random variables to be considered include initial crack sizes, accuracy and frequency of nondestructive tests, i.e. PSI and ISI, material properties, cycles and amplitudes of applied loads. Next, crack growth simulations are performed. Fracture mechanics models employed are based on the linear elastic fracture mechanics, i.e. Newman-Raju solutions [13], influence functions [14-17] and others [18], and the nonlinear fracture mechanics, i.e. fully plastic solutions [19] or their combination [20, 21]. Besides, creep crack growth can be simulated based on the nonlinear fracture mechanics [22, 23]. During the crack growth simulation, PSI and ISI are considered, and failure judgements of leakage and break are performed. Cumulative failure probabilities are calculated as functions of operation time in round-robin problems (a) and (b), while failure propabilities for one PTS event is calculated in round-robin problem (c).

When the Monte Carlo algorithm is used to evaluate structural reliability of nuclear components with very low failure probabilities, a large number of samples have to be

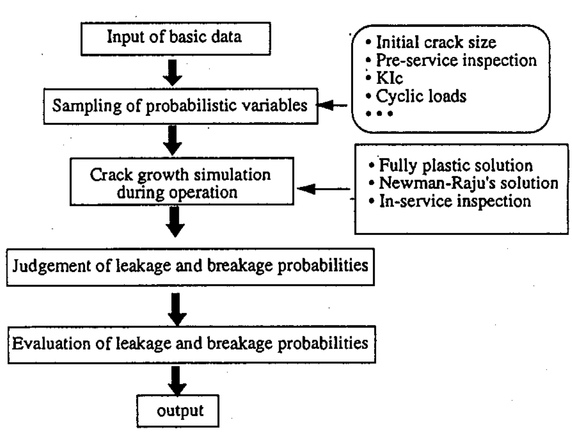


Figure 1 Flow of typical PFM analyses

taken in order to achieve high computational accuracy. This will lead to a very high computational cost. Several improved methods have been proposed so for. The Stratified sampling Monte Carlo (SMC) algorithm is one of such attempts. A parallel processing algorithm combined with the SMC method is very useful [9]. As shown in Figure 2, the computer system includes a parent processor and a number of children processors, which are called network processors. The parent processor controls and renews the calculation data in each cell, while the children processors perform PFM analyses. The parent processor reads the whole input data, and divides a sampling space into many independent small cells. The data of each cell required to the calculation of a fracture mechanics model are sent to the network whenever confirming that any idling processor exists in the network. At the same time, the parent processor receives the analysis results calculated by the children processors. Each child processor continuously monitors the state of the PFM analysis as well as the cell data flowing in the network. If any child processor is idling, it receives the cell data from the network. The present parallel SMC algorithm was successfully implemented on one of massively parallel computers composed of one host processor and 512 network processors, and an excellent parallel performance over 90% was achieved.

ANALYSES OF PIPING UNDER NORMAL OPERATING CONDITIONS

Problem Description

Here we set a round-robin problem for nuclear piping, considering pipe failure protection design based on the Leak-Before-Break (LBB) concept [24].

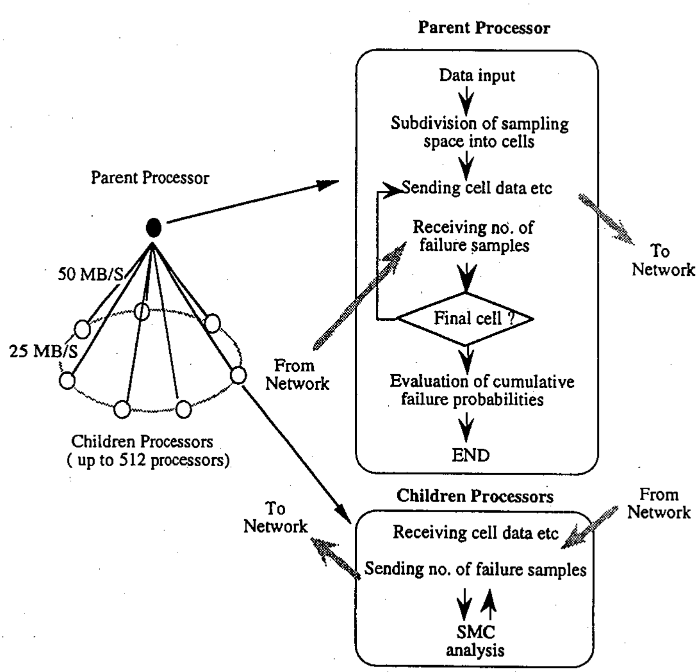


Figure 2 System configuration for parallel SMC algorithm

(1) Analysis model

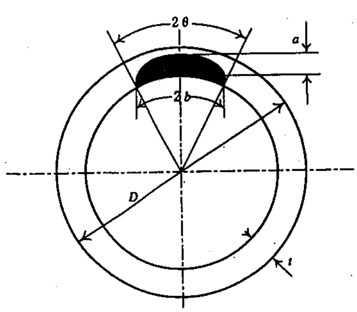
Primary piping of the Light Water Reactors (LWR) with a circumferential inner surface crack is considered as shown in Figure 3(a). When the crack penetrates, the through-wall crack model is used as shown in Figure 3(b). Three typical pipe sizes are assumed as : (a) pipe thickness t = 11.1mm, outer diameter of pipe D_{out} = 114.3mm, i.e. Type 4B, (b) t = 26.2, D_{out} = 406.4, i.e. Type 16B, and (c) t = 37.3, D_{out} = 660.4, i.e. Type 26B. Cumulative failure probabilities of one existing crack, whose unit is 1/crack, are calculated as functions of operation years.

(2) Loading condition

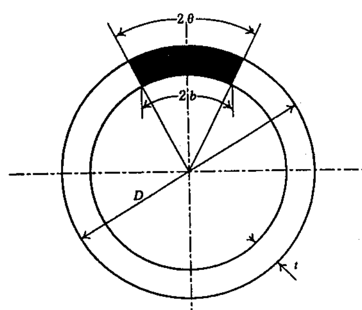
Typical and conservative loading conditions shown in Table 1 are adopted for fatigue crack growth simulation. These were employed in the LBB verification experiments held in Nuclear Power Engineering Corporation in Japan [24].

(3) Initial crack shape

Statistical distributions of initial crack sizes and shape may influence failure probabilities significantly. It is therefore desirable to estimate plant-specific data of the distributions as pointed out in [25]. Unfortunately actual distributions in Japanese power plants are not available yet. On the other hand, semi-elliptical surface cracks are often assumed, and several statistical distributions were obtained and utilized by several researchers. Ref. [7] showed sensitivity studies varying statistical distributions of an initial crack depth.



(a) Part-through crack



(b) Through-wall crack

Figure 3 Geometries of assumed initial crack

Table 1 Loading conditions for analyses of pipe

No	Applied stress ($\times S_m$) Tension max	Tension min	Bending max	Bending min	Cycles /year
1	1.0	0.0	0.0	0.0	7
2	1.5	0.4	0.0	0.0	18
3	1.0	0.75	0.0	0.0	320
4	0.0	0.0	1.0	-1.0	8
5	0.0	0.0	0.5	-0.5	16
6	0.0	0.0	0.1	-0.1	330

Considering these results, the present study employs the Marshall distribution for crack depth (a) [26] which give conservative failure probabilities. As for a crack aspect ratio (c/a), very few researches have been done. We employ here a log-normal distribution [27].

(4) Fatigue crack growth

Only fatigue crack growth based on the Paris' law is assumed, whose coefficients are taken from the fatigue crack growth rate of nuclear pressure vessel steels in water given in the ASME Code Section XI, Appendix A [28]. The curve, which was constructed mainly from the data of pressure vessel steels for nuclear power plants in water, includes the fatigue crack growth data of weldment of STS49 piping.

(5) Failure criteria

Leakage is one of important failure modes of piping. The leakage criterion is simply and conservatively defined as $a/t \geq 0.8$. This value is selected, considering a crack extension from the backside surface.

Pipe break is evaluated by using the two kinds of methods. The one is the Net Section criterion combined with the G factor [24], while the other is the R6 method with a failure assessment curve (FAC) of category 1, option 1 [29]. The G factor is a correction factor to take into account ductile fracture in the case of larger diameter ferritic steel piping. Only when examining break with either the methods, the following loads are applied to the pipe, referring ASME Code Sec. III : internal pressure-induced stress of 0.5Sm and bending moment-induced stress of 1.0Sm, where Sm is an allowable stress.

(6) Material properties

Material assumed here is a carbon steel STS49, which is a typical material for nuclear piping. Its mechanical properties at 302 °C are given as follows : a yield stress σ_y = 232 MPa, an ultimate stress σ_u = 422 MPa, a flow stress σ_f = 327 MPa, Sm = 138 MPa, K_{Ic} = 202.9 MPa$\sqrt{m}$. The temperature of 302 °C is not an actual operation one, but only a reference value.

(7) Nondestructive inspection

PSI and ISI are considered. A probability density function of nondetection of crack of depth (a), P_{ND}, is assumed as in Ref. [27].

(8) LBB Index for PFM Analyses

In PFM analyses, the probability of non LBB, $P_{\neg LBB}$ is regarded as the probability of conditional probability of break after leakage occurs, $P_{L \wedge B|L}$. $P_{L \wedge B|L}$ can be evaluated using leakage probability P_L and the probability of break after leakage $P_{L \wedge B}$ as follows :

$$P_{L \wedge B|L} = P_{L \wedge B} / P_L \tag{1}$$

In the present study, we define the likelihood of LBB, λ_{LBB} with the inverse of Eq. (1) as :

$$\lambda_{LBB} = 1 / P_{L \wedge B|L} = P_L / P_{L \wedge B} \tag{2}$$

We call λ_{LBB} LBB index. Break probability P_B is calculated by :

$$P_B = P_{L \wedge B} + P_{(\neg L) \wedge B} \tag{3}$$

The phenomenon of break without leakage $(\neg L)\wedge B$ corresponds to the situation that a very long surface crack leads to break. Its probability $P_{(\neg L)\wedge B}$ is in general very small. Thus,

$$\lambda_{LBB} \approx P_L / P_B \tag{4}$$

Results and Discussions

Various sensitivity analyses are performed for different pipe diameters, leak detection capabilities, statistical distributions of crack depth, break criteria, crack growth rates, load histories and ISI. Time variations of cumulative failure probability of one existing crack and that of LBB index, λ_{LBB} are calculated. The detail of this analysis can be found in Ref. [12].

(1) Effects of leak detection capability

Figure 4 shows time variations of leakage and break probabilities and LBB index for cases REF, CLM and 1YR for three different pipe diameters. In case REF, leak detection is not applied at all. In case CLM, leak is perfectly detected and a plant operation is stopped just after the crack penetration. In case 1YR, leakage is detected one year after leakage. Case 1YR may be regarded as a practical and conservative case. The figure clearly shows very large failure probability for 4B pipe. This is because the Marshall distribution tends to give conservative results for surface crack problems. To perform more realistic analyses, it is necessary to establish an appropriate method to evaluate embedded cracks. In case REF, leakage and break probabilities tend to be larger, and LBB index gets smaller for smaller pipes. Deterministic analyses for LBB phenomena show that LBB is likely to occur for larger pipes [24]. The present PFM analyses show the same tendency more quantitatively. Let us focus on case REF. After a crack penetrates, break probability increases gradually with time as the crack grows circumferencially. As the results, LBB index decreases with time. During 40-80 years, the decreasing rate of LBB index reduces. In case CLM, break probability means the probability of break without leakage. Figure 5 shows that break probability in case CLM, P_{CLM} is negligible compared with that in case REF, P_B except the first few years. As the result, $P_B = P_{L\wedge B}$. The fact suggests the validity of Eq. (4) as LBB index in an engineering sense. Leakage probability in practical situations may be in between that of case CLM and that of case REF. In Figure 5, break probability of case 1YR is sufficiently small, and is rather close to that of case CLM. LBB indices in 80th year of case 1YR for 4B, 16B and 26B pipes are reduced 50, 500 and 1500 times as much as those of case REF, respectively. These results clearly suggest that leakage detection capability influences siginificantly the likelihood of LBB. In cases CLM and 1YR, LBB indices increase with time. In other words, LBB is more likely to occur with time. This is because crack growth rate is faster in the ligament direction than in the circumferential direction, so that leakage probability increases, while break probability saturates.

(2) Effects of ISI

ISI mainly reduces occurrence rate of leakage or break per year rather than the magnitude of leakage or break probabilities. Figure 5 shows time variations of leakage and break probabilities, occurrence rates of leakage and break per year. In case REF, ISI is not applied at all, while ISI is applied in every 20 years in case IS. Leakage and break probabilities saturate in the 20th year when ISI is first applied. However, ISI does not affect leakage and break

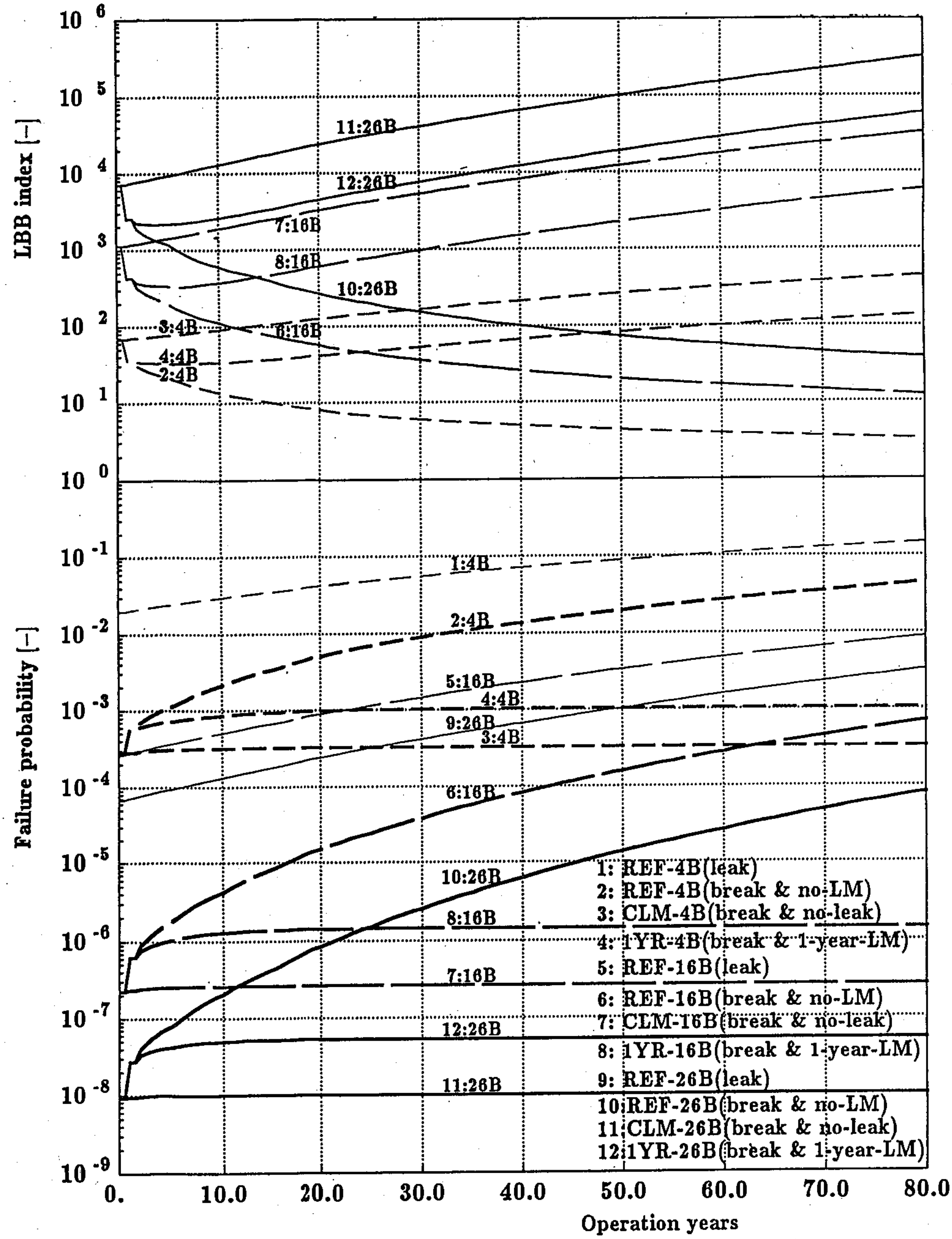


Figure 4 Time variations of failure probabilities and LBB indices : cases REF, CLM & 1YR

probabilities at 40th and 60th years. On the other hand, occurrence rates of leakage and break are significantly reduced. In the present analyses, the nondetection probability of ISI is assumed to be the same as that of PSI. Since ISI is more difficult than PSI, the reduction effect of occurrence rate of leakage and break due to ISI is overestimated to some extent in Figure 5.

ANALYSES OF AGED RPV UNDER NORMAL AND UPSET LOADING CONDITIONS

Problem Description

(1) Analysis model

A plate with a semi-elliptical surface crack is analyzed. Its thickness and width are taken to be t = 0.2 m and 2b = 12.6 m, considering the beltline portion of PWR pressure vessels. The plate is assumed to be subjected to various magnitudes of remote uniform tensile and bending stresses. Cumulative failure probabilities of one existing crack (unit : 1/crack) are calculated as functions of operation years.

(2) Loading conditions

Nineteen kinds of cyclic tensile and bending stresses are chosen from the design loading conditions of Level A (normal operation condition) and Level B (upset condition) which occur in the beltline portion of PWR pressure vessels listed in the Marshall report [26]. Considering practical operating conditions of Japanese nuclear power plants, this loading condition will give us conservative results.

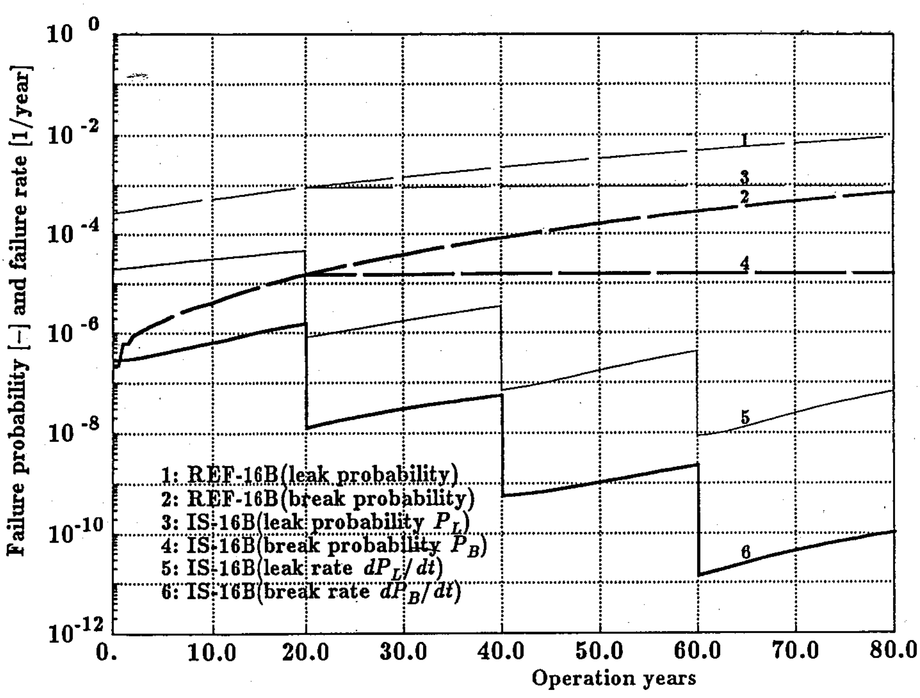


Figure 5 Time variations of failure probabilities and failure rates : cases REF and IS for 16B

(3) Initial crack shape

An initial crack depth (a) and a crack aspect ratio (c/a) of a semi-elliptical surface crack are assumed to be random variables. Crack shapes are assumed to keep the semi-elliptical shape during crack growth.

(4) Fatigue crack growth

As for fatigue crack growth, the Paris' law is used, whose coefficients are taken from the fatigue crack growth rate of nuclear pressure vessel steels in water given in the ASME Code Section XI, Appendix A [28].

(5) Failure criteria

The failure modes considered here are leakage and break, i.e. brittle fracture.

(6) Fracture toughness

Referring to the reduction of the upper shelf Charpy absorbed energy due to neutron irradiation [30] and some experimental data conducted by the LE subcommittee in JWES [3], the time variation of K_{Ic} at 300 °C due to neutron irradiation is formulated. According to this equation, the K_{Ic} value reduces by about 19 % due to neutron irradiation during a 40 year operation.

(7) Nondestructive examination

PSI and ISI are considered.

Results

Various sensitivity studies have been performed varying statistical distributions of initial crack shape, fracture toughness, fatigue crack grwoth rate, accuracy and frequencies of PSI and ISI. The detail of the results can be found in Refs. [7, 8, 11].

ANALYSES OF RPV UNDER PTS CONDITIONS

Problem Description

As for PTS problems, there are some well studied problems, i.e. (a) H. B. Robinson problem [5] and (b) EPRI/NRC PTS benchmark problems [6]. We just employed them.

(1) EPRI/NRC PTS benchmark problem

The original problem of this includes thirteen cases of problems and some additional special problems. Main features of them can be summarized as follows :

- (a) No cladding is assumed.
- (b) Exponential decay of coolant temperature with time is assumed.
- (c) Coolant-wall heat transfer coefficient and internal pressure are assumed to be time-independent.
- (d) Axial (case A1) and circumferential (case B1) infinitely long cracks are assumed.
- (e) Original OCTAVIA crack depth distribution [31] is assumed.
- (f) RT_{NDT} shift and through-wall fluence attenuation in US Regulatory Guide 1.99 Rev.2 [32] are assumed.
- (g) K_{Ic} and K_{Ia} criteria are employed as unstable crack growth and crack arrest criteria, respectively.
- (h) Upper-shelf toughness of 220 MPa$\sqrt{m}$ is employed.
- (i) No warm prestress (WPS) effect is considered.

(2) H. B. Robinson problem

This problem was that of the US H. B. Robinson reactor pressure vessel solved by ORNL in 1985 using OCA-P code [5]. Failure probabilities of vessel were calculated for 28 postulated transients of the reactor. Essence of this PFM problem is as follows :

(a) Cladding is considered.
(b) Coolant-wall heat transfer coefficient and internal pressure are time-dependent, and specific to each of the postulated transients.
(c) Axial infinitely long crack is assumed.
(d) Marshall distribution for crack depth and Marshall crack detection probability [26] are employed.
(e) RT_{NDT} shift of Guthrie equation [33] and through-wall fluence attenuation in US Regulatory Guide 1.99 Rev. 2 [32] are assumed.
(f) K_{Ic} and K_{Ia} criteria are employed as unstable crack growth and crack arrest criteria, respectively.
(g) No WPS effect is considered.

In the present study, one of the 28 transients, called case "9.22B", assuming the main steamline break event, was adopted.

Results

Various sensitivity analyses are performed on the two sets of PTS problems in order to quantitatively evaluate effects of input data on failure probabilities of RPV under PTS events. The input data studied are initial crack shape, the probabilistic distribution of initial crack depth, cladding, RT_{NDT} shift, impurity content, the through-wall distributions of material properties, preservice inspection (PSI) and warm prestressing. The detail of the results can be found in Refs. [7, 10].

CONCLUSIONS

The PFM approaches are expected to play one of key roles in evaluating practical safety margin of aging power plants and in judging whether to operate them over their design lives with a rational basis. Before adopting PFM computer programs to actual problems, it is very important to assure their reliability with well understood standard PFM problems, and to understand quantitative influences of uncertainty of data on analysis results. For this purpose, we have set the following three kinds of PFM round-robin problems ; (a) primary piping under normal operating conditions, (b) aged RPV under normal and upset operating conditions, and (c) aged RPV under PTS conditions. In problem (a), we newly proposed LBB index for PFM analyses. The last problem is chosen from some of the US benchmark problems. For these problems, various sensitivity analyses are performed, varying statistical distributions of initial crack shape, material properties such as flow stress, fracture toughness, fatigue crack growth rate, Cu content, and several operation-related actions such as PSI and ISI. In order to establish standard procedures to evaluate best estimates of failure probabilities of nuclear PV&P components, we will continuously improve the round-robin problems and accumulate sensitivity analysis results.

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