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Probabilistic Fracture Mechanics Based Assessments for Aged Nuclear Structural Components

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ABSTRACT

Probabilistic Fracture Mechanics (PFM) approaches are regarded as appropriate methods to rationally evaluate life of aged plant since they can consider various uncertainties. In Japan, a series of research activities on PFM approaches to the integrity studies of nuclear structural components have been performed since 1987. This paper reviews some key results on PFM-based assessments for aged nuclear structural components, i.e. 1) probabilistic interpretation of the Leak-Before-Break (LBB) concept, 2) a PFM model of aged pipe considering variation of seismic loading, and 3) PFM-based maintenance for steam generator (SG) tubes.

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 pre-service inspection (PSI) and in-service inspection (ISI).

In Japan, a research activity on PFM approaches to the integrity studies of nuclear structural components was initiated in 1987 by the LE-PFM subcommittee organized within the Japan Welding Engineering Society (JWES) under a subcontract of the Japan Atomic Energy Research Institute (JAERI), and had continued for three years [1]. The activity was followed by the RC111 research committee organized in the Japan Society of Mechanical Engineers (JSME) in 1991, and finished in 1995 [2]. Succeeding it, a PFM subcommittee organized in JWES started again in 1996 and have continued for 5 years [3]. These activities are summarized in [4]. Main results obtained from the activities have been published in a number of papers [5-15]. This paper reviews some key results on PFM-based assessments for aged nuclear structural components, i.e.

1) probabilistic interpretation of the Leak-Before-Break (LBB) concept,
2) a PFM model of aged pipe considering variations of seismic loading, and
3) PFM-based maintenance for steam generator (SG) tubes.

OUTLINE OF PFM ANALYSES WITH AGING EFFECTS

The flow of a typical PFM analysis is as follows. 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 of aged structural components, cycles and amplitudes of applied loads. Next, crack growth simulations are performed. Fracture mechanics models employed are based on the linear elastic fracture mechanics and the nonlinear fracture mechanics or their combination. During the crack growth simulation, PSI and ISI and time-dependent change of material properties are considered, and failure judgements of leakage and break are performed. Failure probabilities, i.e. break or leakage are calculated as functions of operation.

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 taken in order to achieve high computational accuracy. This leads to a very high computational cost. Several improved methods have been proposed so far. The Stratified sampling Monte Carlo (SMC) algorithm is one of such attempts. A parallel processing algorithm combined with the SMC method is very useful [7].

PROBABILISTIC INTERPRETATION OF LBB CONCEPT

Problem Description

We set a round-robin PFM problem for aged nuclear piping, considering pipe failure protection design based on the Leak-Before-Break (LBB) concept [16]. Then we attempt probabilistic interpretation of the LBB concept, using a newly proposed LBB index. The detail of the analyses were described in Refs. [10, 11].

Primary piping of the Light Water Reactors (LWR) with a circumferential inner surface crack is considered. When the crack penetrates, the through-wall crack model is used. 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.

Typical and conservative loading conditions are adopted for fatigue crack growth simulation. Other failure mode of pipe such as stress corrosion cracking (SCC) can be also considered [3, 17].

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. Unfortunately actual distributions in Japanese power plants are not available yet. Semi-elliptical surface cracks are often assumed, and several statistical distributions were obtained and utilized by several researchers. Ref. [5] showed sensitivity studies varying statistical distributions of an initial crack depth. Considering these results, the present study employs the Marshall distribution for crack depth (a) 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. Comparative study between an initial surface crack model and an initial embedded crack model can be found in [12].

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.

Leakage is one of important failure modes of piping. The leakage criterion is simply and conservatively defined as $a/t \geq 0.8$. Sophisticated criteria for leakage can be employed, considering capability of leak monitor.

Pipe break is evaluated by using the two kinds of methods. The one is the Net Section criterion combined with the G factor [16], while the other is the R6 method with a failure assessment curve (FAC) of category 1, option 1. 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 $0.5S_m$ and bending moment-induced stress of $1.0S_m$, where S_m is an allowable stress.

Material assumed here is a carbon steel STS49, which is a typical material for nuclear piping. PSI and ISI are considered.

LBB Index

In the present analyses, we define the likelihood of LBB called LBB index, λ_{LBB} as follows :

$$\lambda_{LBB} \equiv \frac{1}{P_{B|L}} = \frac{P_L}{P_{L \wedge B}} \tag{1}$$

where $P_{B|L}$, P_L, $P_{L \wedge B}$ are the conditional probability of break after leak, the probability of leak and the probability of both leak and break, respectively. The bigger λ_{LBB} is, the more likely LBB occurs. The probability of break is defined as follows :

$$P_B = P_{L \wedge B} + P_{(\neg L) \wedge B} \tag{2}$$

Since $P_{(\neg L) \wedge B}$ is negligibly small compared with $P_{L \wedge B}$ under the present analysis condition, Eq. (1) is simply reduced as :

$$\lambda_{LBB} \cong \frac{P_L}{P_B} \tag{3}$$

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.

Figure 1 shows time variations of leakage and break probabilities and LBB index for the following three cases for three different pipe diameters.

Case REF :	Leak detection is not applied at all.
Case CLM :	Leakage is perfectly detected and plant operation is stopped just after the leak detection.
Case 1YR :	Leakage is detected one year after leakage.

The figure show that for smaller pipes, leakage and break probabilities tend to be larger, while the LBB index gets smaller. The 26B pipe is more than one order safer than the 4B pipe through the life in terms of P_L, P_B and λ_{LBB}. Deterministic analyses for LBB phenomena show that LBB is likely to occur for larger pipes [28]. 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 circumferentially. As the results, LBB index decreases with time. During 40-80 operation years, the decreasing rate of the LBB index reduces.

In Case CLM, break probability means the probability of break without leakage, since leakage is completely detected. Break probability in Case CLM, P_{CLM} is negligible compared with that in Case REF, P_B except the first few years.

Leakage probability in practical situations may be in between that of Case CLM and that of Case REF. As shown in the figure, break probability of Case 1YR is sufficiently small, and is rather close to that of Case CLM. The LBB indices in the 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 suggest that leakage detection capability influences the likelihood of LBB significantly. In Cases CLM and 1YR, the LBB indices increase with time. In other words, LBB becomes more likely to occur with time. This is because crack growth rate is faster in the ligament direction than in the circumferential direction under the present analysis condition, so that leakage probability increases, while break probability saturates because of leak detection effects.

PFM MODEL OF AGED PIPE CONSIDERING VARIATION OF SEISMIC LOADING

Seismic load is one of the dominant loads in the failure assessment of pipes. Its probabilistic variation was not taken into account in the past PFM studies. We have developed a new PFM model of aged pipe considering probabilistic variations of seismic load. The detail of the model and analysis results are described in Ref. [3, 15].

Analysis Model

Probabilistic variation of seismic load for pipe is caused due to the following two reasons. The one is the variation of amplitude and frequency of the earthquake acting to nuclear plant building. This is expressed with seismic hazard curve. The other is the variation of seismic stress acting to the pipe, which is propagating within the building. This should be precisely expressed with a seismic analysis model of the building including piping. Reliable data for this variation is very few. In the present study, the variation of the seismic stress is expressed using a log-normal distribution, referring response factors of components used for Seismic PSA [18]. These two kinds of variation in seismic loading are considered in the PFM model of pipe.

Basic analysis conditions are almost the same as the above LBB analyses, except loads to judge break criterion. The amplitude and frequency of earthquake are determined using seismic hazard curve, while the variation of seismic stress is expressed with the log-normal distribution.

Results and Discussions

Three different hazard curves are compared. Generally, the hazard with higher probability of occurrence gives higher break probability. Within the whole hazard curve, the regions of extremely low and high in ground acceleration are not effective for failure probability. This is because extremely low seismic load does not cause failure of pipe and because extremely high acceleration has very low probability of occurrence. Thus the break probability of pipe is sensitive to the medium region of the hazard curve, i.e. probability of occurrence is around 10^{-2} - 10^{-3} per year.

The variation of seismic stress and the amplitude and frequency of earthquake significantly affect the break probability, but not the leak probability. We also compared with the long-normal distribution and the normal distribution in expressing the variation of seismic load. Using the log-normal distribution results in much higher break probability than the normal distribution because of longer tail of the log-normal distribution. We may need to introduce a certain cut-off value when using the log-normal distribution.

PFM-BASED MAINTENANCE OF SG TUBE

SCC is a main degradation mode for SG tubes made of Inconel 600, which has been significantly affecting PWR's availability. Some old SGs have been replaced because of severe SCC damage in SG tubes and high maintenance costs. Thus, it is strongly demanded to establish a rational method of optimizing maintenance strategy for SG tubes, considering both safe operation and efficient maintenance of plant.

In PSA of first kind, core damage frequency (CDF) is taken as the end-point of risk. Ranking of components to be inspected next is also performed based on such risk [19, 20]. Although core damage frequency (CDF) value is very low, unscheduled stop of plant often results in loss of huge amount of money, and further accumulate social damage in public. Thus it is also desired to develop the rational method to manage such unscheduled stop of plant. To deal with these issues, we focus on leakage and break probabilities of SG tubes with their related costs.

Analysis Model

A model is constructed, by modifying pc-PRAISE code so as to evaluate primary-secondary leakage and SG tube rupture during 60-year operation. Failure mechanism considered is stress corrosion cracking (SCC) located in roll expansion zones at the top of tube-sheet. The model assumes creation of an initial semi-elliptical circumferential surface crack with a fixed crack depth and log-normally distributed crack length at the inner surface of SG tube after some crack initiation period. Net-section criterion is adopted as break criteria. A whole SG tubes in a 4-loop unit is analyzed. Sensitivity studies are performed on inspection accuracy, inspection interval, sampling inspection and crack propagation law. Considering longer-cycle operation in the future, three kinds of inspection intervals of 12, 18 and 25 months are calculated. Three kinds of inspection accuracy, i.e. non-detection probability are considered. "Normal" inspection is defined in the level that 40% through wall (TW) defects are detected with the probability of 0.5. This corresponds to a conventional eddy current testing (ECT) using bobbin coils. "Better" inspection is for the level that 20% TW defects are detected with the probability of 0.5. "Best" inspection is for the level that 10% TW defects are detected with the probability of 0.5. The inspection accuracy of "Better" and "Best" expects future development in the SG tube inspection.

Cost Analyses

To quantitatively evaluate maintenance strategies, failure probabilities are converted into cost. Maintenance strategy examined here is the amount of investment to improve inspection accuracy. The cost of leakage and rupture in *t*-th year per unit are calculated as follows :

$$CostOfLeakage = C_{leak} \times p_{leak}(t) \tag{4}$$

$$CostOfBreak = C_{break} \times p_{break}(t) \tag{5}$$

where C_{leak} and C_{break} denote the anticipating costs of leakage / break, and $p_{leak}(t)$, p_{break} (t) represents the probabilities of leakage / break for *t*-th year per unit. The cost of repairing SG tubes in the *t*-th year per unit is also defined as follows :

$$\text{Cost of Repairing SG Tubes of a 4-Loop Unit} = N_{tubes} \times C_{repair} \times p_{repair}(t) \tag{6}$$

where N_{tubes}, C_{repair}, and $p_{repair}(t)$ denote the number of SG tubes in a 4-loop unit, the cost of repair, and the probability of repairing one SG tube in *t*-th year, respectively.

On decision-making for long range investment, it is required to consider time value of money. In such a case. the discounted cash flow method (DCF) is often used to evaluated the long-range investment. Here net present value (NPV) is calculated as an index of investment. The NPV is one of the most fundamental financial indexes for decision-making based on DCF. At the time of T, if NPV(T) > 0, it is justified to be worth while investing, namely, keeping operation of the unit.

Results and Discussions

Sensitivity analyses

ISI interval of 12, 18 or 25 months affect leakage probabilities very little. Leakage probability in the case of "Better" inspection is two orders lower than that of "Normal" one. The "Best" inspection significantly reduces the leakage probability by five orders, compared with the "Normal" inspection over the 60-th operation year. Thus leakage probability is highly sensitive to inspection accuracy.

Cost analyses

Figure 2 shows the effect of inspection accuracy on NPV. Until 10 year operation, three types of inspection accuracy do not affect NPV. However, in the case of "Normal" inspection, operation over 10 years is no longer profitable. In the case of "Better" inspection, operation over 30 years is no longer profitable.

CONCLUSIONS

Aging of nuclear structural components are key issues in structural integrity and safety assessment of nuclear power plant. To take into account such aging issues in probabilistic assessments, we have been precisely studying on round-robin PFM models for aged nuclear structural components and some application methodology in Japan. This paper reviewed some key results on PFM-based assessments for aged nuclear structural components, i.e. 1) probabilistic interpretation of the Leak-Before-Break (LBB) concept, 2) a PFM model of aged pipe considering variation of seismic loading, and 3) PFM-based maintenance for steam generator (SG) tube. The first example showed that the LBB index provided us quantitative information on the likelihood of LBB concept. The second example showed how probabilistic variations of seismic loading influenced failure probability of aged piping. The third example showed that the PFM-based cost analysis could be a powerful tool for determining maintenance strategy of aged components.

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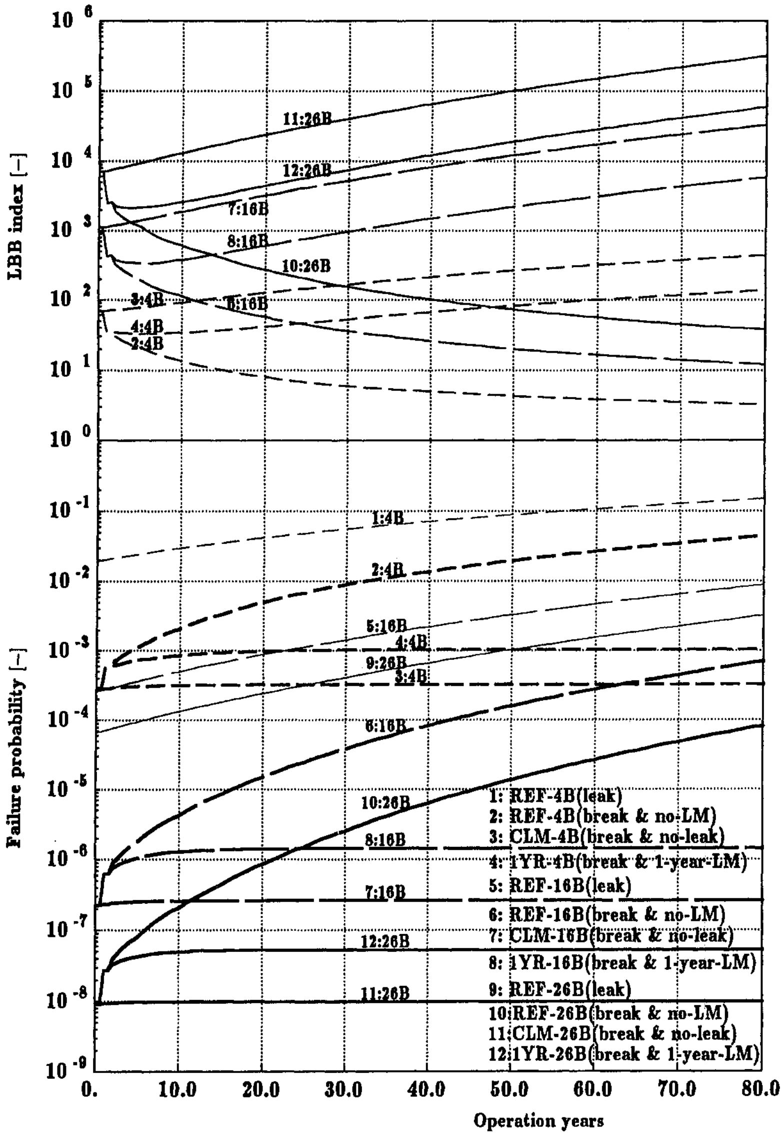

Fig.1 Time Histories of Failure Probabilities and LBB Indexes

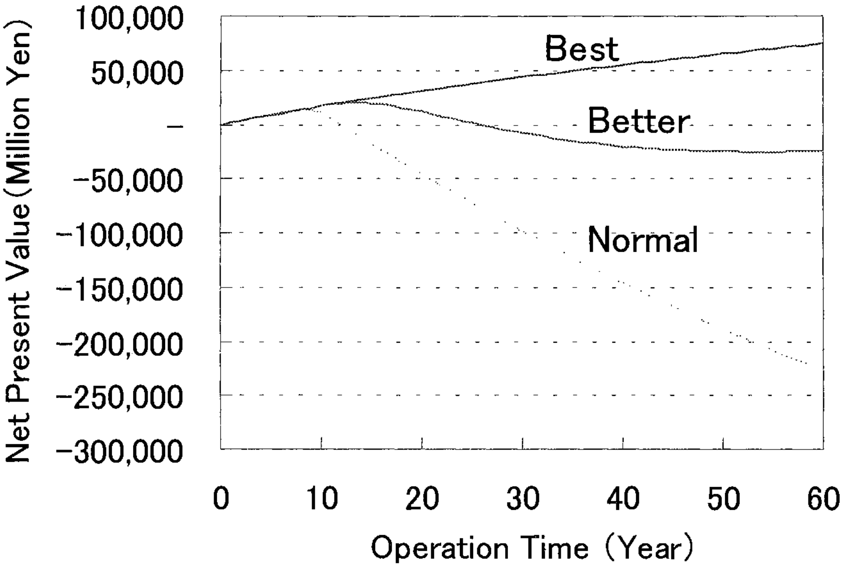

Fig. 2 Net Present Values for Various Inspection Accuracy