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REACTOR COOLANT LOOP PIPING

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FAILURE PROBABILITY OF PWR REACTOR COOLANT LOOP PIPING

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ABSTRACT

In nuclear power plant design, postulation of a double-ended guillotine break (DEGB) in the primary coolant piping introduces many design problems. These include asymmetric blowdown, pipe whip restraints, and the combination of loads to DEGB and a safe shutdown earthquake (SSE). The U.S. Nuclear Regulatory Commission (NRC) contracted the Lawrence Livermore National Laboratory (LLNL) to conduct a probabilistic assessment of the reactor coolant piping of all existing nuclear power plants to investigate if the probability of occurrence of DEGB would be small enough to safely eliminate the postulation of DEGB in the design requirements. The assessment was divided into two parts: DEGB due to direct crack growth of flaws in the piping and DEGB indirectly induced by sources other than crack growth, such as the failure of component supports.

This paper describes the results of assessments performed on the pressurized water reactor (PWR) coolant loop piping of Westinghouse and Combustion Engineering plants. For direct DEGB, consideration was given to crack existence probability, initial crack size distribution, hydrostatic proof test, preservice inspection, leak detection probability, crack growth characteristics, and failure criteria based on the net section stress failure and tearing modulus stability concept. For indirect DEGB, fragilities of major component supports were estimated. The system level fragility was then calculated based on the Boolean expression involving these fragilities. Indirect DEGB due to seismic effects was calculated by convolving the system level fragility and the seismic hazard curve. The results indicate that the probability of occurrence of both direct and indirect DEGB is extremely small; thus, postulation of DEGB in design should be eliminated and replaced by more realistic criteria.

INTRODUCTION

In nuclear power plant design, postulation of a double-ended guillotine break (DEGB) in the primary

coolant piping introduces severe design loading conditions, which include asymmetric blowdown, pipe whip, and the combination of loads due to DEGB and a safe shutdown earthquake (SSE). These conditions result in difficulties and excessive costs in areas of design, construction, and maintenance (e.g., unnecessary radiation exposure of maintenance personnel). It is believed by many that DEGB is an extremely unlikely event and that this design assumption might cause more harm than good. Since some of the older operating plants were not designed for these loading conditions, extensive plant retrofitting may be necessary to meet current requirements.

The U.S. Nuclear Regulatory Commission (NRC) contracted the Lawrence Livermore National Laboratory (LLNL), Livermore, CA, to conduct a probabilistic assessment of the primary coolant piping of all existing nuclear power plants in the U.S., including both PWR and boiling water reactors (BWR) to investigate if the probability of occurrence of DEGB is small enough to safely eliminate the postulation of DEGB in the design requirement.

Since the system design and the piping arrangement of nuclear steam supply systems vary significantly from one vendor to another, we did a vendor-to-vendor assessment. This paper summarizes the results of the assessment performed on the reactor coolant loop piping of two PWR vendors: Westinghouse (W) and Combustion Engineering (CE). We further divided the assessment into two parts for each vendor: DEGB due to direct crack growth of flaws in the piping welds and DEGB indirectly induced by sources other than crack growth, such as the failure of component supports. They are referred to hereafter as direct and indirect DEGB.

In both the direct and indirect DEGB analyses, two types of variability, or uncertainty, in many important parameters were considered. One, which is called random uncertainty in this study, represents the physical inherent randomness; and the other, which is called modeling uncertainty, is associated with the lack of knowledge or detailed information about the parameters to model them precisely. First, we evaluated the inherent randomness for each parameter.

if the randomness was small, we used a deterministic number; otherwise, a distribution was estimated. If the modeling uncertainty was significant for that parameter, another distribution characterizing the modeling uncertainty was also assigned.

In the direct DEGB analysis, we used a Monte Carlo simulation to calculate the leak or DEGB probability at a weld joint considering the randomness of the parameters, and the Latin Hypercube sampling technique of experimental design to generate a set of runs to describe modeling uncertainty. In the indirect DEGB analysis, both types of uncertainty were represented in the fragility descriptions of component failure modes. The uncertainties were carried along throughout the calculation with the fragilities.

Since the details of the indirect DEGB analysis are the subject of another paper (Ref. 1), only a brief summary of indirect DEGB is presented here.

DIRECT DEGB

GENERAL METHODOLOGY

Direct DEGB is defined as DEGB caused by crack growth and instability in the piping weld joints. To assess the crack growth, we used a probabilistic fracture mechanics approach. The basic idea was to simulate the crack state as the plant changes throughout its life-time. Figure 1 is a simplified

Figure 1. Flow chart for the probabilistic assessment of piping integrity.

flow chart of this approach. The left column represents the analytical procedure, and the right column indicates the input information and various simulation models used for each step of the analytical process.

For each weld joint of the piping system, we calculated the leak or DEGB probability using a Monte Carlo simulation technique. We considered the variability due to inherent physical randomness in many of the parameters studied. The simulation starts with the random selection of sample crack sizes and the calculation of conditional probabilities associated with these crack sizes. The fracture mechanics theory is then applied to calculate the growth of these cracks and to determine if pipe fracture, either leak or DEGB, would occur as the cracks grow during the plant lifetime. Various crack detection capabilities or procedures, such as the preservice inspection, hydrostatic proof test, and leak detection, are simulated.

In this study, we considered only fatigue crack growth. Stress corrosion cracking is believed to be insignificant in PWR reactor coolant loop piping. One reason is that the oxygen content of the reactor coolant in the PWR systems is low. Also, the reactor coolant loop piping is stress relieved, and the residual stress is believed to be low. The fatigue crack growth takes into account the cyclic stress history due to various thermal transients and postulated seismic events. The failure criteria applied is either the critical net section stress approach or the tearing modulus instability approach depending upon their applicability to the material characteristics and the geometric conditions of the pipe.

The above calculational procedure yields the conditional leak or DEGB probabilities (conditioned on the existence of a crack and on the occurrence of an earthquake of given peak ground acceleration) as a function of time for all weld joints. The failure probability of the entire piping system, which includes many weld joints in the reactor coolant loops, can be obtained if crack existence probabilities for the weld joints and the probabilistic seismic hazard information regarding the peak ground acceleration and the occurrence rate at the site are given. Table 1 summarizes the input information and various simulation models for W and CE plants.

ASSESSMENT OF W AND CE PLANTS

There are about 60 W reactor units and 15 CE reactor units in the U.S., of which we studied 35 of the 46 W units located east of the Rocky Mountains and 13 CE units. Since the sample sizes are fairly large, we believe that the results give a good, overall picture of PWR primary coolant loop piping.

Input Information and Simulation Models

Geometry and Material Properties. The reactor coolant loop pipes are connected to loop components at both ends, and there are no intermediate supports. Figure 2 shows the general arrangements of typical primary coolant loops of W and CE plants. In the case of W plants, we show a four-loop model; however, there are earlier W plants with two or three loops. The reactor coolant loop system of CE plants consists of two loops, each of which has two branches; each branch is equivalent to one loop of a W system. There is only one three-loop CE plant in operation, which we did not include in this study.

The major geometric distinction between CE and W primary coolant loops is that two branches of a CE loop share one common hot leg and one steam generator; therefore, the hot leg of CE plants is substantially larger than either the suction leg (or crossover leg in W plants) or the discharge leg (or cold leg in W plants). All three legs of the PWR primary coolant piping are rather short relative to their diameters.

The major factor contributing to the difference in the analysis between W and CE primary coolant loop piping is the material used. Austenitic stainless steel type 316 is used in W systems, while ferritic or carbon steel SA516 Grade 70 is used in CE loop piping. As a result, differences exist in the simulation models for the W and CE plants in areas such as the ultrasonic flaw detection, crack growth characteristics, and failure criteria, which we describe later.

Initial Crack Size Distribution. In this study, we considered only circumferential cracks at the weld joints; since DEGB is our major interest, we did not include longitudinal cracks. Two-dimensional cracks of semielliptical shape on the interior pipe surface are assumed (Fig. 3). Several distributions on crack depth a have been proposed in various studies as shown in Fig. 4 (Vol. 5, Ref. 2). We used the Marshall distribution (Ref. 3), which is considered very conservative; however, we made a slight modification to eliminate the physical impossibility of having a crack depth greater than the pipe thickness b . The modified Marshall distribution has the following marginal density function:

$$p(a) = \frac{e^{-a/\mu}}{\mu(1 - e^{-b/\mu})} \quad 0 \leq a \leq b \quad (1)$$

$\mu = 0.246 \text{ in.}$

The modified Marshall distribution is considered the best-estimate model of crack depth distribution due to randomness. To account for the modeling uncertainty, we adopted a triangular distribution for the parameter $1/\mu$ considering the Marshall distribution ($\mu = 0.246 \text{ in.}$) as the median, or the 50th percentile. Figure 4 also shows the curves for the upper and the lower bounds.

A truncated lognormal probability density function (Fig. 5a) is used to describe the distribution due to randomness for the other shape parameter, the aspect ratio β , which is the ratio of crack half-length to depth (or b/a). Figure 5b shows the complementary cumulative marginal distributions for various values of ρ , which is defined as the percentage of cracks with β greater than 5.0 (Vol. 5, Ref. 2). The distribution with ρ equal to 10⁻² is considered the best-estimate model. Another lognormal distribution on the parameter β_m (defined in Fig. 5a) is used to describe modeling uncertainty for the distribution of the aspect ratio. The 5th percentile and 99.9th percentile curves due to modeling uncertainty are also shown in Fig. 5b.

Inspection Detection Probability. Ultrasonic testing is the most often used method of nondestructive inspection in nuclear power plant pipes. The probability of nondetection P_{ND} has been quantified in many studies (Refs. 2 & 4). Based on the available data, we found P_{ND} to be adequately characterized by the following relationship:

$$P_{ND} = 1/2 (1 - c) \text{ERFC}(\sqrt{\ln [A/A']}) + c, \quad (2)$$

Table 1. Input information and simulation models for calculation of direct DEGB.

Input	Plants	
	W	CE
	A sample of 35 units (17 Plants) out of 46 total units east of the Rocky Mountains	13 units (8 plants) out of 15 total units in the U.S.
Piping material	Type 316 Stainless Steel	SA516 Grade 70 Ferritic Steel
Initial crack size distribution	Depth ^{*, a} Aspect ratio ^{*, b}	Random--modified Marshall dist. Modeling--triangular distribution (Fig. 4) Random--truncated lognormal (Fig. 5) Modeling--lognormal (Fig. 5)
Probability of nondetection	$F_{ND} = 1/2 (1-\epsilon) \text{ERFC} (v \ln A/A') + \epsilon$ $v = 1.60$ $A' = 0.98 \text{ in.}^2$ $\epsilon = 0$	$v = 1.33$ $A' = 0.20 \text{ in.}^2$ $\epsilon = 0.005$
Stress history	Provided by W	Provided by CE
Dead weight, pressure, thermal expansion ⁺ , seismic ⁺	Random--lognormal Modeling--lognormal	Random--lognormal Modeling--lognormal
Crack growth model*	Figure 6	Figure 7 and Table 2
Failure criteria	Critical net section stress	Tearing modulus inatability
Detectable leak rate	3 gpm	3 gpm
Crack existence probability*	Poisson distribution with the rate parameter equal to 10^{-4} per cubic inch of weld volume	Poisson distribution with the rate parameter equal to 10^{-4} per cubic inch of weld volume
Seismic hazard curves ⁺	Generic	Generic and site specific

* Distribution was used in representing random uncertainty.

⁺ Modeling uncertainty was considered.

where

$$A = \pi/4 a D_B, \text{ if } (2b < D_B) ,$$

$$\pi/2 ab, \text{ if } (2b > D_B) ,$$

 D_B = beam diameter, and

$$A' = \pi/4 a' D_B.$$

ERFC is the complementary error function. Here, we used a value of 0.005 for ϵ to represent the lower bound value of F_{ND} and a beam diameter of 1 in. For the cast austenitic stainless steel used in W plants, $v = 1.60$ and $a' = 1.25 \text{ in.}$ are used. In the case of ferritic steel used in CE plants, $v = 1.33$ and $a' = 0.25 \text{ in.}$ are appropriate. These values reflect

the fact that the large columnar grains resulting from the casting process make the ultrasonic inspection less effective in the thick stainless steel pipes than in the ferritic steel pipes.

Crack Growth Characteristics. The subcritical fatigue crack growth is an important phenomenon that leads to pipe failure under low-level cyclic stress conditions. There are sufficient data available to characterize the fatigue crack growth rates (da/dN) for both A316 stainless steel and SA516 grade 70 ferritic steel. They can be represented by a Paris model in terms of two important parameters: the cyclic stress intensity factor ΔK and the load ratio R .

Figure 2a. General arrangement of W reactor coolant loops.

Figure 6 shows the data and the least square fit curve using a Paris model for stainless steel in a reactor coolant (or water) environment (Vol. 5 of Ref. 2). The equation that represents the model is also shown in Figure 6. It has a coefficient C and an exponent m . The scatter, or randomness, of the data can be characterized by considering the coefficient C to be lognormally distributed if the exponent m is taken to be fixed. Figure 6 also shows the 10th and 90th percentiles of C . The threshold value of $4.6\sqrt{\text{ksi in.}}$ for $(\Delta K/(1-R))^{1/2}$, below which the crack growth rate is considered to be negligible, provides a reasonable and somewhat conservative number.

The fatigue crack growth model for carbon and low-alloy ferritic steel used in this study is a modified version of the reference fatigue crack growth

Figure 2b. General arrangement of CE reactor coolant loops.

Figure 3. Geometry of a semielliptical inner surface crack.

Figure 4. Various complementary cumulative marginal crack depth distributions.

model contained in Appendix A of Section XI of the ASME code. It has the following form (Ref. 5):

$$\frac{da}{dn} = Q [C \Delta K^m] \quad (3)$$

where the bracketed expression represents the ASME code reference fatigue crack growth model (Fig. 7). The multiplication factor Q is a lognormally distributed random variable with values for the logarithmic mean and standard deviation listed in Table 2 for different values of R and ΔK . This fatigue crack growth model was developed from the same data base as the ASME reference model. However, the ASME model was based on a 95% global confidence limit on the mean of the data and was intended to be conservative for design purposes (Ref. 6). The growth

model used here and represented by "Eq. (3)" was intended to properly describe the data with randomness characterized by Q . This model is therefore considered to be our best estimate model. The ASME model is conservative by an approximate factor of 1.5 as compared to the median of "Eq. (3)".

Failure Criteria. Two criteria are available to assess the stability of cracks in structures that are stressed beyond the region of applicability of the linear elastic fracture mechanics. One is the critical net section stress approach and the other is the tearing modulus stability approach. In principle, the tearing modulus approach using the elastic-plastic fracture mechanics theory is the more logical approach. Crack instability occurs as the J-integral J and tearing modulus T exceed the corresponding

Figure 5a. Modified lognormal distribution for the crack aspect ratio.

Figure 5b. Various complementary cumulative marginal distributions for the crack aspect ratio.

values J_{IC} and T_{JR} of the materials' J-resistance curve. Unfortunately, calculating J and T is a complicated task and usually requires a finite element procedure. The net section stress approach, on the other hand, circumvents this difficulty (Ref. 7). It is based on the experimental results of a center-cracked stainless steel panel in tension. The crack is considered unstable when the ligament net section stress exceeds a critical value. The net section stress approach is applicable only in very ductile materials.

Figure 6. Fatigue crack growth rate data for 304- and 316-type stainless steels as a function of effective stress identity shown along with the least square curve fit.

In the assessment of W primary coolant piping, we used the net section stress approach. The critical net section stress is the flow stress, which is the average of ultimate and yield stresses. We used this approach because the stainless steel in W reactor coolant loop piping is very ductile, and because of the results of a sensitivity study using both criteria discussed above (Vol. 6 of Ref. 2). The results indicated that the critical stress was reached before J exceeded J_{IC} . In other words, the net section stress approach governs the tearing modulus approach.

In the case of CE reactor coolant loop piping, a tearing modulus approach is used, because the ferritic material used in CE piping is less ductile than stainless steel. It is further observed that the critical net section stress is dependent on the pipe diameter (Ref. 7); the critical net section stress decreases as the pipe diameter increases. The use of the net section stress approach might not be appropriate for the hot leg of CE reactor coolant loop piping since its inside diameter, which is approximately 42 in., might be too large to fall within the range for which the critical net section stress has been determined.

We assessed the stability of cracks in both the radial and circumferential directions. Instability in the radial direction results in leak; however, instability in both the radial and circumferential directions are required to induce a DEGB. The stability assessment of CE plants in the radial

Figure 7. Reference fatigue crack growth curves for carbon and low-alloy ferritic steels. (Appendix A to ASME Code, Section XI.)

direction was based on tabulated results for part-through complete circumferential cracks (Refs. 8 and 9). In the circumferential direction, we used an approximate approach (Ref. 10) that incorporates known linear elastic solutions for through-wall part circumferential cracks (Ref. 11), and linear elastic and elastic-plastic solutions for center-cracked panels (Refs. 8, 9, and 12).

Seismic Hazard Curves. To assess the seismic effect, we used seismic hazard curves to represent the frequency of occurrence of earthquakes of different intensities. We used plots of the annual frequency of

exceedance vs peak ground acceleration to represent the randomness of earthquake events. The modeling uncertainty is represented by curves of various percentile levels of a distribution. The curve corresponding to the median of the distribution due to the modeling uncertainty is considered to be the best estimate.

There are two kinds of seismic hazard curves used in this study, generic and site-specific. Generic seismic hazard curves were developed based on existing site specific hazard curves for six plants dispersed over the eastern and midwestern states (Ref. 13). Figure 8 shows the generic seismic hazard curves.

Table 2. Constants associated with the random variable Q in the fatigue crack growth model for carbon and low-alloy steels: ΔK in ksi $\sqrt{\text{in.}}$; $W = (R - 0.25)/0.4$; $W' = 1 - W$.

Ranges for R and ΔK	Mean of $\ln Q$	Standard Deviation of $\ln Q$
$R < 0.25$		
$\Delta K < 19$	-0.408	0.542
$\Delta K \geq 19$	-0.408	0.542
$R \geq 0.65$		
$\Delta K < 12$	-0.367	0.817
$\Delta K \geq 12$	-0.367	0.817
$0.65 > R > 0.25$		
$\Delta K < 12 + 7W$	$-0.367W - 0.408W'$	$0.817W + 0.542W'$
$\Delta K \geq 12 + 7W$	$-0.367W - 0.408W'$	$0.817W + 0.542W'$

Figure 8. Generic seismic hazard curves for sites east of the Rocky Mountains.

They were used for the analysis of W plants located east of the Rockies since W plants are numerous and most do not have site specific seismic hazard curves. In the case of CE plants, we used the same generic seismic hazard curves for the many sites. However, we used site specific curves for San Onofre and WPPSS plants since they are located on the West Coast where the generic curves are not applicable. We also used site-specific seismic hazard curves for Palo Verde, because it is not close to any of the six sites whose site specific curves form the data base of the generic curves. These site specific seismic hazard curves are documented in Ref. 14.

Other Input. Westinghouse and Combustion Engineering provided all plant specific information including the pipe geometry and the stresses.

Generally, conservatism and uncertainties exist in the stresses calculated by design engineers. After reviewing the analytical methods used in the stress calculation and after reducing the conservatism, we estimated the distributions of these stresses to account for both random and modeling uncertainties (Refs. 14 & 15).

In this study, leak is considered to be detected if a crack results in 3 gpm or more of coolant leakage. The crack is then considered to be repaired once it is detected.

We neglected the inservice inspection, because its quality varies greatly from plant to plant, and it cannot be modeled with reasonable confidence. We get conservative results by not taking into account the effect of inservice inspection.

We calculated the crack existence probability assuming that cracks in weld joints occur as events of a stationary Poisson process. It appears that a value of $10^{-4}/\text{in.}^3$ is a reasonable estimate for the crack rate per unit weld volume (Ref. 2).

The earthquakes are also assumed to be Poisson distributed. The earthquake occurrence rate of any specific intensity can be estimated based on the seismic hazard curves (Ref. 14). Furthermore, the distribution associated with the earthquake occurrence time can be obtained from the assumed Poisson distribution (Ref. 14).

Analyses and Results

We performed two types of analyses: best-estimate analysis and the uncertainty analysis. The former considers only the best-estimate models and the latter takes into account the full range of distribution due to modeling uncertainty.

We considered random and modeling uncertainties for many important parameters. Although modeling uncertainties exist in almost all of the parameters, we studied the modeling uncertainty for only those parameters where the effect is deemed more significant: crack depth, crack aspect ratio, thermal expansion stresses, seismic stresses, and seismic hazard curves.

The models for those parameters whose modeling uncertainties were neglected are naturally the best-estimate models. For parameters for which modeling uncertainties were considered, the best-estimate models are those that correspond to the median, or 50th percentile, of their modeling uncertainty distributions.

Figures 9 and 10 show the leak and DEGB probabilities for W plants (Ref. 15), and Figures 11

Figure 9. Leak probabilities for W reactor coolant loop piping.

Figure 10. Direct DEGB probability for W reactor coolant loop piping.

and 12 show those for CE plants (Ref. 14). Figures 9-12 show both the best-estimate and the uncertainty analysis.

Best-Estimate Analysis. For each of the plants considered, we calculated the best-estimate probabilities of leak and DEGB using the best-estimate models and the procedure outlined in Fig. 1.

The best-estimate analysis yields a single point estimate of the failure probability for a reactor coolant loop piping. In the case of W plants, the results are plotted as empirical cumulative distributions based on the results of the 17 plants (or 35 units) analyzed and are intended to make characterize the total population of W plants east of the Rockies. Since our analysis covers most of the CE

Figure 11. Leak probability for CE reactor coolant loop piping.

Figure 12. Direct DEGB probability for CE reactor coolant loop piping.

plants, the results are presented as plant-specific cases. Plants 1 through 4, a total of 7 units, are late model plants. We also analyzed a composite plant including the worst combination of stresses and geometries of four (6 units) earlier CE models.

The best-estimate results indicate that the leak and DEGB probabilities for W and CE plants are very small. The best-estimate leak probabilities for W plants range from 1.2×10^{-8} to 1.5×10^{-7} per plant year, and the best-estimate DEGB probabilities range from 1×10^{-12} to 7×10^{-12} . The

best-estimate leak probabilities for CE plants range from 1×10^{-8} to 3×10^{-8} , and the best-estimate DEGB probabilities range from 5×10^{-14} to 5×10^{-13} . These ranges are rather small; they are less than one order of magnitude except for the leak probabilities for W plants, where the range is slightly greater than one order of magnitude. This indicates that plant vintage does not have a significant effect on the direct DEGB results since both older and newer plants are included in this analysis.

Another interesting observation is that the median best-estimate probabilities of leak and DEGB for CE plants are slightly lower than those of the W plants. The difference is less than one order of magnitude for leak and is somewhat greater than one order of magnitude for DEGB. It is surprising to find the closeness of the corresponding ranges of best-estimate leak and DEGB probabilities between W and CE plants since there are substantial differences in the general arrangement of reactor coolant loop systems, in the geometry of the pipes, and in the materials used.

Uncertainty Analysis. The values of the parameters and the simulation models used to calculate leak or DEGB probability are not known with certainty. Therefore, a range of values or a distribution for leak and DEGB probability, considering the whole range of the distributions due to modeling uncertainty, is necessary in addition to the point estimate produced in the best-estimate analysis. This would provide confidence bounds on the leak and DEGB probabilities in addition to the results calculated from the best-estimate analysis. To account for the modeling uncertainty, we used the Latin Hypercube sampling technique of experimental design. Readers are referred to relevant literature for a detailed description of this technique (Ref. 16).

The basic procedure for the uncertainty analysis of a plant is as follows:

1. For each parameter, divide the distribution due to modeling uncertainty into n equiprobable intervals.
2. Select a random value within each interval. Therefore, this selected random value corresponds to a specific percentile of the distribution. A total of n values spread over the distribution for each parameter is obtained. Repeat this process for all m parameter for which modeling uncertainty is to be considered.
3. Randomly combine a value without replacement from each distribution to form a set. A total of n sets of m values are obtained.
4. Calculate the leak and DEGB probabilities according to the procedure presented in Fig. 1 for each of the combinations. Each result constitutes a sample. A total of n samples are obtained.

Based on these samples, a distribution of leak or DEGB probability can be constructed for each plant.

Since there are many plants and it is impractical to analyze each one, we performed the uncertainty analysis for two W plants one of which had the highest best-estimate leak probability and the other, which had the highest DEGB probability. The results of these uncertainty analyses provide valuable information about the modeling uncertainty in estimating the failure probabilities of these two plants. They are also expected to provide insight on the effects of modeling uncertainty on other W plants. The sample leak and DEGB probabilities of these two plants are plotted on horizontal lines passing through their corresponding best-estimate data points in Figs. 9 and 10. The calculated upper and lower bounds as well as the 10th and 90th percentile of the sample distribution are also marked.

The uncertainty analysis of the plant with the highest calculated best-estimate leak probability make us 90% confident that the leak probability of that plant is less than 2.5×10^{-7} per plant year. In the same sense, we are 90% confident that the DEGB probability of the plant with the highest calculated best-estimate DEGB probability is less than

7.5×10^{-11} per plant year. To make inferences to other W plants, we can say, based on Figs. 9 and 10, that there is a 90% confidence that almost all W plants east of the Rockies have leak and DEGB probabilities less than 2.5×10^{-7} and 7.5×10^{-11} per plant year, respectively.

In the case of CE plants, we performed the uncertainty analysis for every plant under consideration since there were relatively few plants involved. In Figs. 11 and 12, the upper and lower bounds as well as the 10th and 90th percentile are marked on horizontal lines as was done for the two W plants. Since we performed an uncertainty analysis for each CE plant under investigation, we conclude that the upper bound probability values for leak and DEGB are 2×10^{-7} and 1×10^{-10} per plant year, respectively, for CE plants.

INDIRECT DEGB

METHODOLOGY

Indirect DEGB is that caused by sources other than flaws in the piping, such as missiles and the failure of structures and components connected or adjacent to the primary coolant piping. Studies conducted in this research program (Refs. 1 and 2) indicate that seismically induced failure of primary coolant loop component supports is the major source of indirect DEGB. The pipe break scenario starts with seismic excitation, which causes a component support to be stressed beyond its capacity. The primary coolant loop component collapses when the supports fail. This leads to a pipe break since the pipe can be severed by the falling component to which it is connected. Following this scenario, the assessment of indirect DEGB includes the following tasks:

1. Estimate the conservatism and the uncertainty in the calculated structural responses for various loading conditions, such as dead weight, thermal expansion, pressure, and seismic loads.
2. Identify failure modes and develop for each a fragility description. Each fragility description represents the probability of structural failure conditioned on the occurrence of an earthquake of given peak ground acceleration. Figure 13 shows a typical fragility description.
3. Calculate the overall system level fragility with the Boolean expression, which accounts for all of the significant failure modes and the associated fragility descriptions. It has the same mathematical form as the individual fragility descriptions.
4. Calculate the probability of indirect DEGB, P_{DEGB} , by convolving the system level fragility and seismic hazard.

ASSESSMENT OF W AND CE PLANTS

We assessed 46 W reactor units located east of the Rockies in this study (Ref. 13), but we did not analyze all of them extensively, because this would have been impractical. Furthermore, the required input information was incomplete. We therefore decided therefore, to perform a detailed analysis on the two lowest seismic capacity units, which include one earlier model that was designed without considering DEGB and a later model that was designed with current criterion of combining SSE and DEGB. The results from these two units provide an upper bound on the P_{DEGB} for all 46 units.

To get an overall assessment of all 46 W units, we evaluated the other 44 units using these two units

Figure 13. A typical fragility description.

as a guide. A single probability distribution on all 46 units was obtained by statistically merging the individual probability distributions (See Table 3). The median annual probability of DEGB has an upper bound value of 3.3×10^{-6} . The same median value for all 46 units is 10^{-7} and is more than one order of magnitude less. The 90th percentile value for the lowest capacity unit is 2.3×10^{-5} per plant year, which can be viewed as the absolute upper bound probability for a DEGB to occur.

Eight CE plants, or a total of 13 units, were analyzed in detail (Ref. 17). Site-specific seismic hazard curves were used whenever they were available; see Table 4 for the results. It is clear that the probability of DEGB is also small. Plant 8 has the lowest capacity, or the highest DEGB probability. The 90th percentile value of P_{DEGB} for Plant 8 is 5.2×10^{-5} , and the median is 6.4×10^{-6} per plant year. These leak and DEGB probabilities values support the contention that DEGB is an extremely unlikely event.

SUMMARY AND CONCLUSION

The objective of this study is to estimate the probability of DEGB, the postulation of which introduces many design problems for the reactor coolant loop piping. If the probability of occurrence of DEGB is indeed extremely small, it will then be safe to eliminate DEGB in the design of reactor coolant loop piping. Our study, which included a large sample of W plants east of the Rocky Mountains and most CE plants in the U.S. was divided into two parts: DEGB due to direct crack growth instability of existing flaws in the piping and DEGB indirectly induced by sources other than flaws, such as the failure of component supports.

In the direct DEGB assessment, the effort was to study the growth of postulated existing cracks in the pipe welds and to determine if these cracks would

Table 3. Annual probabilities of indirect DEGB, P_{DEGB} , for W plants.

Case	Seismic hazard curves	Percentile of the distribution for P_{DEGB}		
		10th	50th	90th
Lowest seismic capacity units				
Designed with SSE + DEGB	Generic	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}
Designed without DEGB	Generic	1.0×10^{-7}	2.4×10^{-6}	2.0×10^{-5}
All 46 units	Generic	2.0×10^{-9}	1.0×10^{-7}	7.0×10^{-6}

Table 4. Annual probability of indirect DEGB, P_{DEGB} , for CE plants.

Plant	Seismic hazard curves	Percentile of the distribution for P_{DEGB}		
		10th	50th	90th
1	Site specific	4.0×10^{-9}	3.8×10^{-16}	1.0×10^{-13}
2	Site specific	3.5×10^{-18}	4.6×10^{-17}	3.2×10^{-14}
3	Site specific	8.0×10^{-11}	2.9×10^{-9}	1.5×10^{-7}
4	Generic	1.1×10^{-10}	1.3×10^{-8}	3.0×10^{-7}
5	Generic	2.3×10^{-8}	6.1×10^{-7}	6.1×10^{-6}
6	Generic	9.0×10^{-10}	6.6×10^{-8}	1.2×10^{-6}
7	Generic	6.6×10^{-8}	1.4×10^{-6}	1.1×10^{-5}
8	Generic	5.0×10^{-7}	6.4×10^{-6}	5.2×10^{-5}

become unstable and result in leak or break during the plant lifetime. Plant conditions, such as the stress history due to normal operation, various postulated transients, and seismic events were simulated as they affect the growth of the cracks. We also modeled crack detection capabilities, such as preservice inspection, hydrostatic proof test, and leak detection.

In the indirect DEGB area, we considered DEGB due to the failure of reactor coolant loop component supports in a seismic event. We estimated fragilities associated with credible failure modes and calculated probabilities of DEGB using fragility descriptions and seismic hazard curves.

In both the direct and indirect DEGB analyses, we considered variabilities due to both inherent randomness and modeling uncertainty in many important parameters. In the direct DEGB analysis, randomness in many parameters was considered in a Monte Carlo simulation to calculate the leak or DEGB probability at a weld joint; the Latin Hypercube sampling technique was used to handle modeling uncertainty. In the indirect DEGB analysis, both variabilities are attached to the fragility descriptions throughout the calculation.

The results indicate that probability due to direct DEGB is extremely small. The best-estimate probabilities of direct DEGB using the medians of the distribution of the modeling uncertainty ranges from 5×10^{-14} to 7×10^{-12} per plant year for both W and CE plants. From the uncertainty analysis, considering the whole range of modeling uncertainty, we conclude that a direct DEGB probability of 3×10^{-10} per plant year can be considered as the absolute upper bound for W and CE plants.

Indirect DEGB is a more likely event compared to direct DEGB; however, the probability of indirect DEGB is, also very small. For the lowest seismic capacity W plant, the median probability of DEGB is 3.3×10^{-6} per plant year. The corresponding indirect DEGB probability at the 90th percentile is 2.3×10^{-5} . Even for this lowest capacity plant, these probability values are still very small. For all 46 W units east of the Rockies as a whole, the median probability is more than one order of magnitude lower. The probability values for CE plants are also very low. The upper bound probability values are compatible with those of the W plants.

It is clear from the results obtained for W and CE plants that the probability of a DEGB in PWR primary coolant loop piping is extremely small; therefore, it is safe to eliminate the postulation of DEGB in the design requirements.

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