UCRL-91731
PREPRINT
CONF-841042--39

DOUBLE-ENDED BREAKS IN REACTOR PRIMARY PIPING

Garry S. Holman

MASTER

This paper was prepared for submittal to
The 12th NRC Light Water Reactor Safety
Research Information Meeting
Washington, D.C.
October 22-26, 1984

October, 1984

Lawrence Livermore National Laboratory

DOUBLE-ENDED BREAKS IN REACTOR PRIMARY PIPING*

Garry S. Holman

Lawrence Livermore National Laboratory
University of California
P.O. Box 808, Livermore, California 94550

UCRL--91731

DE85 003145

1. INTRODUCTION

The Lawrence Livermore National Laboratory (LLNL), through its Nuclear Systems Safety Program, is performing probabilistic reliability analyses of PWR and BWR reactor coolant piping for the NRC Office of Nuclear Regulatory Research. Specifically, LLNL is estimating the probability of a double-ended guillotine break (DEGB) in the reactor coolant loop piping in PWR plants, and in the main steam, feedwater, and recirculation piping of BWR plants. For these piping systems, the results of the LLNL investigations will provide NRC with one technical basis on which to:

(1) reevaluate the current general design requirement that DEGB be assumed in the design of nuclear power plant structures, systems, and components against the effects of a postulated pipe break.

(2) determine if an earthquake could induce a DEGB, and thus reevaluate the current design requirement that pipe break loads be combined with loads resulting from a safe shutdown earthquake (SSE).

(3) make licensing decisions concerning the replacement, upgrading, or redesign of piping systems, or addressing such issues as the need for pipe whip restraints on reactor coolant piping.

In estimating the probability of DEGB, LLNL considers two causes of pipe break: pipe fracture due to the growth of cracks at welded joints ("direct" DEGB), and pipe rupture indirectly caused by the seismically-induced failure of critical supports or equipment ("indirect" DEGB).

Although these investigations are limited to the reactor coolant piping noted above, the techniques used to assess reliability are sufficiently general that they could be conveniently applied to other piping systems not included in the present LLNL investigations.

*This work was supported by the United States Nuclear Regulatory Commission under a Memorandum of Understanding with the United States Department of Energy.

EAB

2. APPROACH

To arrive at a general conclusion about the probability of DEGB in the reactor coolant loop piping of PWR plants, LLNL is taking a vendor-by-vendor approach. For each of the three PWR vendors (Westinghouse, Babcock & Wilcox, and Combustion Engineering) the principal tasks are to:

(1) estimate the probability of direct DEGB taking into account such contributing factors as initial crack size, pipe stresses due to normal operation and sudden extreme loads (such as earthquakes), the crack growth characteristics of pipe materials, and the capability to non-destructively detect cracks, or to detect a leak if a crack penetrates the pipe wall. To do this LLNL developed a Monte Carlo simulation methodology, implemented in the PRAISE computer code.

(2) estimate the probability of indirect DEGB by identifying critical component supports or equipment whose failure could result in pipe break, determining the seismic "fragility" (relationship between seismic response and probability of failure) of each, and combining this result with the probability that an earthquake occurs producing a certain level of excitation ("seismic hazard").

(3) for both causes of DEGB, perform sensitivity studies to identify key parameters contributing to the probability of pipe break.

(4) for both causes of DEGB, perform uncertainty studies to determine how uncertainties in input data affect the uncertainty in the final estimated probability of pipe break.

LLNL has completed generic evaluations of DEGB probability for plants with nuclear steam supply systems manufactured by Westinghouse (Fig. 1) and by Combustion Engineering (Fig. 2).[1,2,3] The results of these evaluations indicate that the probability of DEGB from either cause is very low. Therefore, this result suggests that the DEGB design requirement -- and with it related design issues such as coupling of DEGB and SSE loads, asymmetric blowdown, and the need to install pipe whip restraints -- warrants a reevaluation for PWR reactor coolant loop piping.

In the Westinghouse and Combustion Engineering evaluations, LLNL designated a single reference, or "pilot" plant, as a basis for methodology development as well as for extensive sensitivity studies to identify the influence that individual parameters have on DEGB probabilities. Thus, each pilot plant was used to develop and "shake down" the assessment methodology that was later applied in the corresponding generic study for each vendor.

In the generic study of reactor coolant piping manufactured by each of these vendors, LLNL evaluated individual plants, or groups of plants sharing certain common or similar characteristics, to arrive at an estimated DEGB probability (including uncertainty bounds) characteristic of all plants. Thus, the generic evaluation represented a "production" application of the assessment methodology.

For Babcock & Wilcox PWR plants, LLNL has estimated the probability of indirect DEGB for each of two representative plants: one plant with the raised loop nuclear steam supply system, and one plant with the lowered loop configuration. LLNL has also obtained and reviewed information required for an evaluation of direct DEGB for the representative raised loop plant.

The objectives and approach of the BWR study are essentially the same. LLNL is currently limiting its investigation to Mark I plants, which have recirculation piping particularly susceptible to the effects of intergranular stress corrosion cracking (IGSCC), and is beginning with a pilot study based on the Brunswick plant operated by Carolina Power & Light. As part of the BWR investigation, LLNL has developed a probabilistic IGSCC model which considers crack initiation as well as the effect of stress corrosion on pre-existing cracks; a prototype has been completed and implemented in the PRAISE code. LLNL is also developing a PRAISE model to consider stress redistribution among weld joints due to the failure of intermediate pipe supports; this was unnecessary in the PWR evaluations because reactor coolant loop piping is supported solely by the loop components; preliminary results indicate that intermediate support failure is important only for earthquakes of twice the SSE or greater. The BWR pilot study is scheduled for completion in December 1984.

3. PROBABILISTIC FRACTURE MECHANICS MODELS

Over the past several years, probabilistic analysis techniques have gained increased acceptance as a method for evaluating the safety of nuclear power plants. One application has been through probabilistic risk assessment (PRA) of event sequences potentially leading to radioactive releases. A different application, which will be discussed here, probabilistically evaluates the adequacy of individual systems, structures, or components to resist failure when subjected to postulated loads.

In essence, a typical component evaluation compares some measure of its strength -- material yield stress, for example -- against the stress resulting from anticipated loads applied to it. If strength exceeds stress, the component is considered adequate for the postulated loads. Should stress exceed strength, however, the component is presumed to fail.

As illustrated schematically by Fig. 3, a deterministic calculation compares point estimates of stress and strength to evaluate component adequacy. Generally, these are nominal values established according to conservative load limits and material strength parameters such as those defined by the ASME Code.[4] In component design, the application of "safety margins" provides an added measure of conservatism. The safety margin compensates for uncertainty associated with many factors, including:

- o variability in nominal material strength, that is, actual strength may be lower than that specified in the analysis.
- o degradation in material strength during plant operation, such as radiation embrittlement.
- o variations in postulated loading conditions such as pressure and temperature transients.
- o load conditions generally regarded as having secondary significance and which are therefore neglected in the evaluation.
- o unanticipated load conditions.
- o simplifications made in modeling a physical system.
- o approximation methods used to calculate stresses and resultant component response.

Stress and strength limits are generally set according to specific design considerations. It is not unusual that an evaluation based on "worst case" stress and strength values outside of the design scope will predict a negative safety margin, in other words, failure.

The deterministic approach embodies a significant degree of inherent conservatism, stemming from many sources:

- o the margin between code allowable limits and actual failure.
- o the margin between design conditions and code limits.
- o the particular analytic techniques used to predict component response to applied loads.
- o input conditions used in predicting component response.

These conservatisms generally add together; thus, the more parameters involved, the more conservative a deterministic evaluation tends to be.

The probabilistic approach replaces the fixed values with random variables, each of which has a statistical distribution. Thus, variations in strength and stress about their nominal (or "best-estimate") values are explicitly considered. When plotted together (see Fig. 4), the area where these distributions overlap represents the probability that stress exceeds strength, in other words, that the component will fail. Instead of setting out to determine if a design is adequate and by what safety margin, a probabilistic evaluation estimates the failure probability ("reliability") of the design. The design is considered adequate ("safe") if the failure probability is acceptably low. What constitutes "acceptably low" is subject to judgement, usually taking into account the potential consequences of failure; the more serious the consequences, the lower the tolerable failure probability.

By distributing each parameter statistically, a probabilistic evaluation yields results that more closely reflect reality. Moreover, probabilistic techniques can take event occurrence rate into account, and therefore more realistically weight the relative effects of frequent vs infrequent load events on overall reliability. Statistical uncertainties attached to each distribution can be carried through the analysis to estimate the uncertainty in the predicted reliability.

Because the simultaneous interaction of many individual -- and often deterministically unrelated -- factors is reflected in a single result (i.e., failure probability), probabilistic techniques provide a convenient, yet powerful basis for sensitivity studies. For example, the effect of material property selection (strength, crack growth behavior) on piping reliability can be weighed against that non-destructive examination (inspection interval, crack non-detection probability).

The LLNL evaluations of DEGB in reactor coolant piping represent one application of probabilistic fracture mechanics to the subject of pipe failure. In these evaluations, the probability of pipe break or leak resulting from crack growth at welded joints ("direct" DEGB) is estimated using the procedure schematically illustrated in Fig. 4. The left column represents the analytic procedure, the right column the input information and analytical models used at each step of the simulation. The procedure, implemented in the PRAISE (Piping Reliability Assessment Including Seismic Events) computer code detailed in References 5 and 6, is summarized in the following discussion.

For each weld joint of a piping system, the leak or break probability is estimated using a Monte Carlo simulation technique. Each replication of the simulation -- and a typical simulation includes several thousand -- begins with a pre-existing flaw having initial length and depth randomly selected from appropriate distributions. These distributions in turn relate the conditional probability of crack existence. Fatigue crack growth is then calculated using a Paris growth law model, to which are applied stresses associated with normal operating conditions and postulated seismic events. The influence of such factors as non-destructive examination (NDE) and leak detection is also considered through the inclusion of appropriate statistical distributions (e.g., probability of crack non-detection as a function of crack depth). Leak occurs when a crack grows through the pipe wall, break when failure criteria based on net section collapse or tearing instability are exceeded.

Completing all replications for a single weld joint and tabulating those cracks that cause failure yields the failure probability as a function of time at that weld, conditioned on a crack existing at the joint and on an earthquake of given ground acceleration occurring. By combining the results for all welds in a particular piping system, and then performing a systems analysis incorporating crack existence probability (a function of the total volume of weld material) and seismic hazard (which relates the occurence rates of earthquakes as a function of peak ground acceleration), the non-conditional probabilities of leak and DEGB are obtained.

It is important to emphasize that this procedure is not a PRA utilizing event tree and fault tree analysis. Instead, the procedure incorporates deterministic (either empirical or analytic) models into a probabilistic "framework" that allows the results of deterministic growth calculations for literally thousands of individual cracks to be consolidated, along with the effects of other factors such as NDE intervals and earthquake occurrence rates, into a single convenient result, namely leak or break probability of a particular piping system. This result could, in turn, provide input for that part of a PRA event tree using the probability of pipe system failure.

4. DOUBLE-ENDED GUILLOTINE BREAK INDIRECTLY INDUCED BY EARTHQUAKES

4.1 General Approach

If earthquakes and large LOCAs are considered as purely random events, the probability of their simultaneous occurence is negligibly low. However, if an earthquake could cause DEGB, then the probability of simultaneous occurence would be significantly higher. Our study of direct DEGB in reactor coolant piping concluded that earthquakes were not a significant contributor to this failure mode. However, another way in which DEGB could occur would be for an earthquake to cause the failure of component supports or other equipment whose failure would in turn would cause a reactor coolant pipe to break. We refer to this scenario as "indirect" DEGB. Evaluating the probability of indirect DEGB involves the following steps:

- o 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.

- o identify critical components whose failure could induce a DEGB. For each component, identify failure modes and their corresponding fragility descriptions. Each fragility description represents the probability of structural failure conditioned on the occurrence of an earthquake of given peak ground acceleration.

- o calculate the overall "plant level" fragility to account for all significant failure modes and the associated fragility descriptions.

- calculate the non-conditional probability of indirect DEGB by convolving the plant level fragility with an appropriate description of seismic hazard. Seismic hazard relates the probability of occurrence of an earthquake exceeding a given level of peak ground acceleration.

Typical descriptions of seismic hazard and fragility are shown in Fig. 5.

4.2 Design and Construction Errors

The LLNL analyses of indirect DEGB probability assumed systems and components that were free from design and construction errors. Because in practice such errors are a real possibility, it is important to assess their potential effect on the probability of pipe break. In principle, design and construction errors could be treated probabilistically in the same way that any other parameter is treated, if a distribution of errors could be established. However, because actual NSSS heavy component support failures are exceedingly rare, developing a meaningful distribution may not be possible. Therefore, a limited sensitivity study was performed to determine what degree of error would be required to significantly change the probability of indirect DEGB.

In this study, plausible construction errors were first identified and the corresponding reduction in the capacity of critical equipment estimated. The indirect DEGB probability for Zion was then recomputed to determine the resultant effect on the probability of indirect DEGB. This study indicated that only very gross construction errors -- errors that would presumably be detected by the stringent quality control measures applied to reactor coolant piping -- could significantly increase the probability of indirect DEGB.

5. SUMMARY OF RESULTS

5.1 Probability of Direct DEGB in Reactor Coolant Loop Piping

We completed probabilistic analyses indicating that the probability of direct DEGB in reactor coolant piping is very small for Westinghouse PWR plants located east of the Rocky Mountains. These analyses calculated the growth of as-fabricated surface flaws at welded joints, taking into account loads on the piping due to normal operating conditions and seismic events. Other factors, such as the capability to detect cracks by non-destructive examination and the capability to detect pipe leaks, were also considered. In this study, we performed "best estimate" calculations for each of 17 sample plants (33 plant units), obtaining 17 point estimates of DEGB probability as well as 17 point estimates of leak probability. These point estimates described "best estimate" distributions of DEGB probability and leak probability. The median values (50% confidence limit) of these distributions provide generic point estimates of DEGB and leak probabilities characteristic of all plants east of the Rocky Mountains.

The results of our evaluations indicate for Westinghouse plants east of the Rocky Mountains that:

- the "best estimate" probability of direct DEGB ranges from 1.1×10^{-12} to 6.3×10^{-12} events per plant-year, with a median value (50% confidence limit) of 4.4×10^{-12} events per plant-year.

- the "best estimate" probability of leak (through-wall crack) ranges from 1.3×10^{-8} to 1.5×10^{-7} events per plant-year, with a median value of 1.1×10^{-7} events per plant-year. The significantly greater probability of break compared to DEGB supports the concept of "leak before break" in PWR reactor coolant loop piping.
- uncertainty analyses indicated that the 90th percentile values of DEGB and leak probabilities for the sample plant with the highest probability of direct DEGB are 7.5×10^{-10} and 2.4×10^{-7} events per plant year, respectively.

Through sensitivity studies, we found that normal operating loads, such as stresses due to pressure and thermal expansion, were the dominant contributors to pipe failure; earthquakes had a negligibly small effect on the probability of failure.

Plant-specific evaluations were performed for reactor coolant loop piping at two west coast plants: Trojan and Diablo Canyon. For Trojan, the median probability of direct DEGB was 2.2×10^{-13} events per plant year, with 10th and 90th percentile values of 2.6×10^{-17} and 1.0×10^{-9} events per plant-year, respectively. The estimated median probability of leak was 5.9×10^{-8} events per plant year, with 10th and 90th percentile values of 2.0×10^{-8} and 1.5×10^{-7}, respectively. These values are comparable to corresponding generic DEGB and leak probabilities for plants east of the Rocky Mountains. As in our generic evaluations, we found that normal operating loads, such as stresses due to pressure and thermal expansion, were the dominant contributors to pipe failure; earthquakes had a negligibly small effect.

For Diablo Canyon, earthquakes contributed more significantly to the probability of direct DEGB. Using seismic hazard curves that we derived from three independent seismic hazard evaluations of the plant site, we estimated the median probability of direct DEGB to be 2.5×10^{-11} events per plant-year, about one order of magnitude higher than that for plants east of the Rocky Mountains. Although earthquakes less than about two times the SSE had only a negligible effect on DEGB probability, we found that the simultaneous occurrence of earthquake and DEGB dominated failure for earthquakes above this level. Furthermore, conditional probabilities of leak and DEGB (i.e., given that an earthquake of a given intensity occurs) were equal for earthquakes in this range, suggesting that pipe rupture, and not pipe fracture, became the mode of failure. This contrasted with our results for other plants, which showed that DEGB was typically several orders of magnitude less likely than leak.

Recognizing the increased importance of seismic effects, we performed an extensive series of sensitivity calculations in lieu of a detailed uncertainty analysis and investigated the effect that earthquakes had on the estimated probability of direct DEGB in the reactor coolant loop piping at Diablo Canyon. In particular, we repeated our best-estimate analyses for various values of maximum ground acceleration level as a check on our extrapolation of seismic hazard to five times the SSE. We also estimated the probability of direct DEGB using each of the three independent seismic hazard evaluations, both in extraplolated and unextrapolated form. The results of this sensitivity study indicated that the median probability of DEGB is relatively insensitive to the particular seismic hazard curve selected from among those used in our evaluation.

Our results for Westinghouse plants are presented in Table 2. The results of our generic study of Combustion Engineering PWR plants indicated that the probability of a direct DEGB in reactor coolant loop piping is similarly low (see Table 3). An interesting result here was that the probability of direct DEGB for the carbon steel piping used in these plants was typically higher than that for the more ductile stainless steel piping used in the Westinghouse plants, if the effects of non-destructive examination were neglected. However, the greater certainty of crack detection in carbon steel roughly equalizes the direct DEGB probabilities for the two types of reactor coolant loop systems, a clear illustration of the ability of probabilistic techniques to consider how the interaction of seemingly unrelated parameters can affect overall pipe reliability.

The results of this study also indicated that the probability of an earthquake causing a direct DEGB is as negligible for Combustion Engineering reactor coolant loop piping as it is for the eastern Westinghouse plants.

5.2 Probability of Indirect DEGB in Reactor Coolant Loop Piping

We completed probabilistic analyses for 46 Westinghouse plants located east of the Rocky Mountains indicating that the probability of indirect DEGB in reactor coolant loop piping is very small for these plants. In evaluating the probability of indirect DEGB for each plant, we first identified critical components and determined the seismic "fragility" of each. We then determined for each component the probability that its failure could lead to DEGB. Finally, we estimated the non-conditional probability of indirect DEGB by statistically combining generic seismic hazard curves for the eastern U.S. with a "plant level" fragility derived from the individual component fragilities.

The results of our analyses (see Table 4) indicated for Westinghouse plants east of the Rocky Mountains that:

- the critical components whose failure would result in DEGB were the reactor pressure vessel supports, the reactor coolant pump supports, and the steam generator supports. For the Zion Unit 1 plant used in our pilot study, the overhead crane in the containment building was also a critical component due to its atypical design. More typical crane designs, supported on rails mounted to the containment structure near the dome, did not contribute significantly to the probability of indirect DEGB.
- the best-estimate probability of indirect DEGB (50% confidence limit) is about 10^{-7} events per plant year, with an upper bound (90% confidence limit) of 7×10^{-6} events per plant year.
- the best-estimate probability of indirect DEGB for one "lower bound" plant designed for the combination of safe shutdown earthquake (SSE) and DEGB loads was 3.3×10^{-6} events per plant year, with an upper bound (90% confidence limit) of 2.3×10^{-5} events per plant year.

- the best-estimate probability of indirect DEGB for another lower bound plant designed for SSE alone (no DEGB loads) was 2.4×10^{-6} events per plant year, with an upper bound of 2×10^{-5} events per plant year.

We also estimated the probabilities of DEGB for two west coast plants, San Onofre Unit 1 and Diablo Canyon Units 1 and 2, using site-specific seismic hazard curves derived from the results of several independent seismic hazard evaluations. As in our evaluations of plants east of the Rocky Mountains, we assumed that the RPV supports, reactor coolant pump supports, and steam generator supports were the critical components whose failure would lead to DEGB. The results of these analyses indicated that:

- the median probability of DEGB in the Diablo Canyon reactor coolant loop piping is 1.7×10^{-6} events per plant-year, with a 90% confidence limit of 2.2×10^{-5} events per plant year. These values are about the same as those for the lowest seismic capacity plants east of the Rocky Mountains.

- the median probability of DEGB in the San Onofre Unit 1 reactor coolant loop piping is 5.4×10^{-8} events per plant-year, with a 90% confidence limit of 9.5×10^{-7} events per plant year. These values, estimated using seismic hazard curves that asymptotically approached 1.05g maximum PGA (denoted as SONGS Set 1 in Table 4), are over one order of magnitude lower than those for the lowest seismic capacity plants east of the Rocky Mountains.

- the probability of indirect DEGB is a strong function of seismic hazard. A sensitivity study performed for San Onofre Unit 1, for which we used a second set of seismic hazard curves extrapolated out to five times the SSE (denoted as SONGS Set 2 in Table 4), showed a two order of magnitude increase in indirect DEGB probability. This contrasts sharply with the results of our evaluations of direct DEGB probability, which was shown in general to be only weakly affected by earthquakes. Nevertheless, even when very large earthquakes are considered, the San Onofre results are still on the same order as those for the lowest seismic capacity plants east of the Rocky Mountains.

The probability of DEGB due to crack growth at welded joints is typically four to five orders of magnitude lower than that of DEGB indirectly caused by the seismic failure of heavy component supports. Thus, our analyses clearly point to indirect causes as the dominant mechanism leading to DEGB in reactor coolant loop piping.

We also performed a limited sensitivity study to determine what <u>degree</u> of design or construction error would be required to significantly change the probability of indirect DEGB. From this study, we concluded that only gross design and construction errors of implausible magnitude could substantially increase the probability of indirect DEGB beyond the values predicted.

An evaluation of Combustion Engineering plants indicated the same general results, with the probabilities of indirect DEGB in reactor coolant loop piping typically lower than for the Westinghouse plants (see Table 5).

6. CONCLUSIONS

In general, the results of our evaluation indicate that the probability of DEGB in the reactor coolant loop piping of Westinghouse and Combustion Engineering plants is extremely low. Our results further indicate that:

- o indirect causes are clearly the dominant mechanism leading to DEGB in reactor coolant loop piping.

- o earthquakes have a negligible effect on the probability of direct DEGB. On the other hand, the probability of indirect DEGB is a strong function of seismic hazard, but is nevertheless low even when earthquakes significantly greater than the safe shutdown earthquake are considered.

- o only very large design and construction errors of implausible magnitude could significantly affect the probability of indirect DEGB in reactor coolant loop piping.

On the basis of these results, we recommend that the NRC seriously consider eliminating DEGB as a design basis event for reactor coolant loop piping in Westinghouse plants. Elimination of the DEGB requirement would accordingly allow pipe whip restraints on reactor coolant loop piping to be excluded or removed, and would eliminate the requirement to design supports to withstand asymmetric blowdown loads.

We also recommend that the current requirement to couple SSE and DEGB be eliminated. Recognizing however that seismically induced support failure is the weak link in the DEGB evaluation, we further recommend that the strength of component supports, currently designed for the combination of SSE plus DEGB, not be reduced. The support strength could be maintained in spite of a decoupling of DEGB and SSE by replacing the present combined load requirement with a factor applied to SSE load alone. This factor would be defined in such a way that the support strength would remain unchanged.

Our study indicates that the probability of DEGB in reactor coolant loop piping is sufficiently low under <u>all</u> plant conditions, including seismic events, to justify eliminating it entirely as a basis for plant design. This represents a fundamental change in design philosophy that has potential impact far beyond the single issue of SSE and DEGB coupling. Elimination of reactor coolant loop DEGB would require that replacement criteria be developed as a basis for various aspects of plant design, including, but not necessarily limited to:

- o blowdown loads on the reactor vessel and RPV internals
- o primary coolant discharge rate
- o containment pressurization
- o jet impingement loads
- o environmental effects
- o support loads
- o pipe whip

Any NRC rulemaking action defining general replacement criteria will have to be based on a comprehensive approach taking into account causes of pipe failure, break size and potential effects on plant design, acceptable levels of safety requirements, and criteria for regulating the postulation of pipe break. In the near term, however, the results of the evaluation reported here now provide NRC with one technical basis for making case-by-case licensing decisions applicable to reactor coolant loop pipng.

References

1. "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant", Lawrence Livermore National Laboratory, Livermore, California, Report UCID-18967, NUREG/CR-2189, Vols. 1-9 (September 1981).

   Vol. 1: Summary
   Vol. 2: Primary Coolant Loop Model
   Vol. 3: Non-Seismic Stress Analysis
   Vol. 4: Seismic Response Analysis
   Vol. 5: Probabilistic Fracture Mechanics Analysis
   Vol. 6: Failure Mode Analysis
   Vol. 7: System Failure Probability Analysis
   Vol. 8: Pipe Fracture Indirectly Induced by an Earthquake
   Vol. 9: PRAISE Computer Code User's Manual

2. NUREG/CR-3660, "Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants", Lawrence Livermore National Laboratory, Report UCID-19988, NUREG/CR-3660 (1984)

   Vol. 1: Summary Report
   Vol. 2: Direct DEGB, Eastern Plants
   Vol. 3: Indirect DEGB
   Vol. 4: Direct DEGB, West Coast Plants

3. "Probability of Pipe Failure in the Reactor Coolant Loops of Combustion Engineering PWR Plants", Lawrence Livermore National Laboratory, Report UCRL-53500, NUREG/CR-3663 (1984)

   Vol. 1: Summary Report
   Vol. 2: Direct DEGB
   Vol. 3: Indirect DEGB

4. Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, New York, New York.

5. D.O. Harris and H.H. Woo, et al, "Fracture Mechanics Models Developed for Piping Reliability Assessment in Light Water Reactors", Lawrence Livermore National Laboratory, Report UCRL-15490, NUREG/CR-2310 (April 1982).

6. T.Y. Lo, et al, "Failure Probability of PWR Reactor Coolant Loop Piping", Lawrence Livermore National Laboratory, Report UCRL-86249 (February 1984). Presented at the ASME Pressure Vessel and Piping Conference, San Antonio, Texas, June 17-21, 1984.

7. R.D. Streit, "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant: Pipe Fracture Indirectly Induced by an Earthquake," Lawrence Livermore National Laboratory, Livermore, California, Report UCID-18967, NUREG/CR-2189, Vol. 7 (September 1981).

Table 1

Parameters Considered in Developing Component Fragilities

Structural Response
o Ground spectrum used for design
o Structural damping
o Site characteristics (rock or soil, shear wave velocity, thicknesses of different strata)
o Fundamental frequency of internal structure if uncoupled analysis was performed
o Interface spectra for NSSS points of connection to structure if uncoupled analysis was conducted
o Input ground spectra resulting from synthetic time history applied to structural model
NSSS Response
o Method of analysis (time history or response spectrum, etc.)
o Modeling of NSSS and structure (coupled or uncoupled)
o NSSS system damping
o NSSS fundamental frequency or frequency range
o If uncoupled analysis was performed, whether envelope or multi-support spectra were used.

TABLE 2

Probabilities of Direct DEGB and Leak for Reactor Coolant Loop Piping in Westinghouse PWR Plants

(events per plant year)

	Confidence Limit[1]		
	10%	50%	90%
<u>Plants East of the Rocky Mountains</u>[2]			
DEGB	5.0×10^{-17}	4.4×10^{-12}	7.5×10^{-10}
Leak	5.6×10^{-10}	1.1×10^{-7}	2.4×10^{-7}
<u>West Coast Plants</u>[3]			
Trojan (DEGB)	2.6×10^{-17}	2.2×10^{-13}	1.0×10^{-9}
Trojan (Leak)	2.0×10^{-8}	5.5×10^{-8}	1.5×10^{-7}
Diablo Canyon (DEGB)	see text	2.5×10^{-11}	see text
Diablo Canyon (Leak)	see text	3.8×10^{-7}	see text

(1) A confidence limit of 90% implies that there is a 90% subjective probability (confidence) that the probability of leak or direct DEGB is less than the value indicated.

(2) Generic seismic hazard curves used.

(3) Plant-specific seismic hazard curves used.

TABLE 3

Best-Estimate Probabilities of Direct DEGB and Leak for Reactor Coolant Loop Piping in Combustion Engineering PWR Plants

(events per plant year)

	DEGB	Leak
Palo Verde 1,2,3	4.5×10^{-13}	1.5×10^{-8}
San Onofre 2,3	1.0×10^{-13}	2.2×10^{-8}
WPPSS 3	6.1×10^{-14}	1.8×10^{-8}
Waterford 3	9.0×10^{-14}	1.8×10^{-8}
Group A (1) Composite	5.5×10^{-14}	2.3×10^{-8}
Westinghouse(2)	6.3×10^{-12}	1.2×10^{-7}

(1) Composite plant enveloping data for Calvert Cliffs 1 & 2, Millstone 2, Palisades, and St. Lucie 1 & 2.

(2) Results for Westinghouse sample plant with highest probability of DEGB.

TABLE 4

Annual Probabilities of Indirect DEGB
for Westinghouse PWR Plants

(events per plant-year)

	Confidence Limit(1)		
	10%	50%	90%
Lowest Seismic Capacity Eastern Plants(2)			
Designed for SSE + DEGB	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}
Designed for SSE alone	1.0×10^{-7}	2.4×10^{-6}	2.0×10^{-5}
All 46 Eastern Plants(2)	2.0×10^{-9}	1.0×10^{-7}	7.0×10^{-6}
West Coast Plants(3)			
San Onofre Unit 1			
SONGS Set 1	3.1×10^{-10}	5.4×10^{-8}	9.5×10^{-7}
SONGS Set 2	1.3×10^{-7}	4.7×10^{-6}	4.9×10^{-5}
Diablo Canyon Units 1,2	4.0×10^{-7}	1.7×10^{-6}	2.2×10^{-5}
Median for West Coast Plants	2×10^{-7}	3×10^{-6}	5×10^{-5}

(1) A confidence limit of 90% implies that there is a 90% subjective probability (confidence) that the probability of indirect DEGB is less than the value indicated.

(2) Generic seismic hazard curves used in evaluation.

(3) Site-specific seismic hazard curves used in evaluation

TABLE 5

Annual Probabilities of Indirect DEGB for Combustion Engineering PWR Plants

	Confidence Limit(1)		
	10%	50%	90%
Group A Plants			
Calvert Cliffs	2.3×10^{-8}	6.1×10^{-7}	6.1×10^{-6}
Millstone 2	9.0×10^{-10}	6.6×10^{-8}	1.2×10^{-6}
Palisades	5.0×10^{-7}	6.4×10^{-6}	5.2×10^{-5}
St. Lucie 1	1.2×10^{-8}	3.8×10^{-7}	4.1×10^{-6}
St. Lucie 2	6.6×10^{-8}	1.4×10^{-6}	1.1×10^{-5}
Westinghouse Lowest Capacity Plant	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}

(1) All probabilities are given as events per plant year. A confidence limit of 90% implies that there is a 90% subjective probability (confidence) that the probability of indirect DEGB is less than the value indicated.

(2) Generic seismic hazard curves used in evaluation.

TABLE 5 (cont.)

Annual Probabilities of Indirect DEGB for Combustion Engineering PWR Plants

	Confidence Limit (1)		
	10%	50%	90%
Group C Plants			
Palo Verde 1,2,3 (2),(3)			
Site-Specific	4.0×10^{-19}	3.8×10^{-16}	1.0×10^{-13}
Generic	2.4×10^{-12}	5.4×10^{-10}	1.1×10^{-7}
San Onofre 2,3 (3)			
Site-Specific Set 1	3.5×10^{-18}	4.6×10^{-17}	3.2×10^{-14}
Site-Specific Set 2	5.0×10^{-17}	1.1×10^{-11}	2.1×10^{-9}
WPPSS 3 (2)	8.0×10^{-11}	2.9×10^{-9}	1.5×10^{-7}
Waterford 3 (2)	1.1×10^{-10}	1.3×10^{-8}	3.0×10^{-7}
Westinghouse Lowest Capacity Plant	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}

(1) All probabilities are given as events per plant year. A confidence limit of 90% implies that there is a 90% subjective probability (confidence) that the probability of indirect DEGB is less than the value indicated.

(2) Generic seismic hazard curves used in evaluation.

(3) Site-specific seismic hazard curves

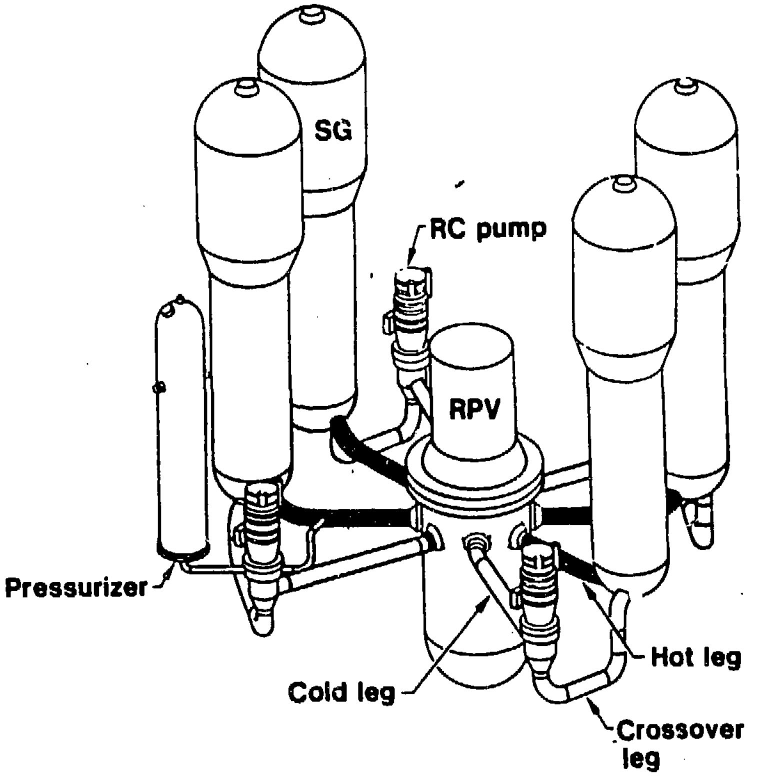

Fig. 1. Typical reactor coolant loop piping arrangement in a Westinghouse pressurized water reactor nuclear steam supply system.

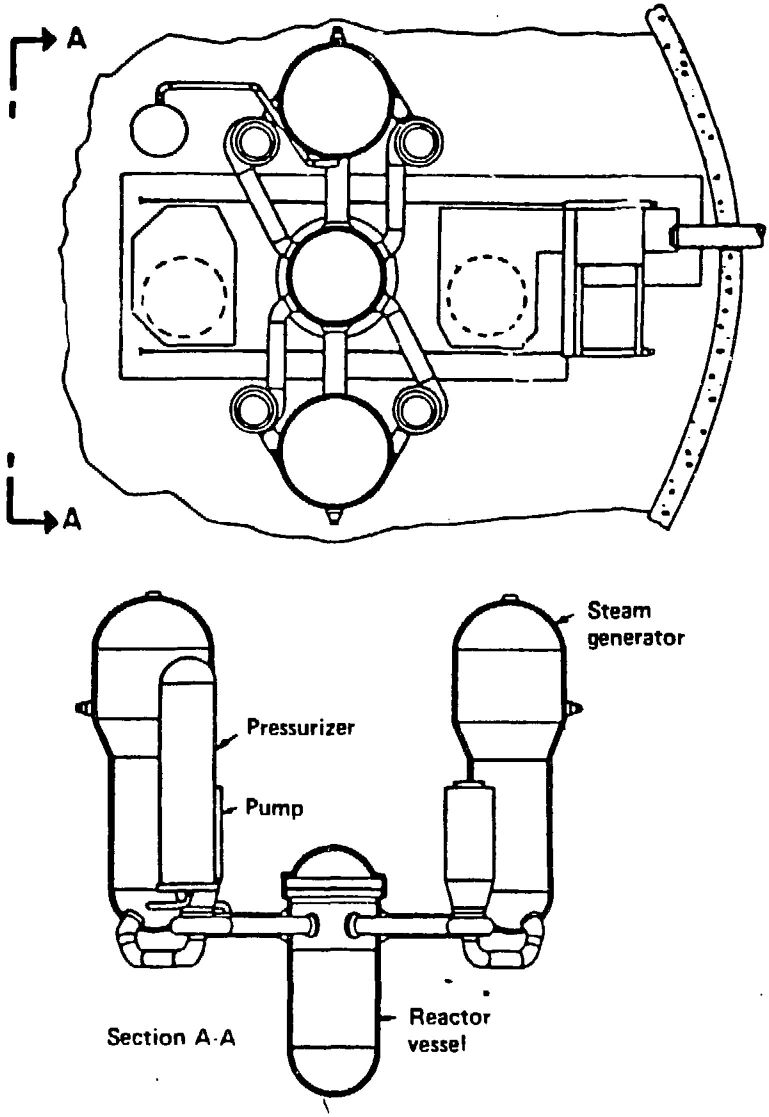

Fig. 2. Typical reactor coolant loop piping arrangement in a Combustion Engineering pressurized water reactor nuclear steam supply system.

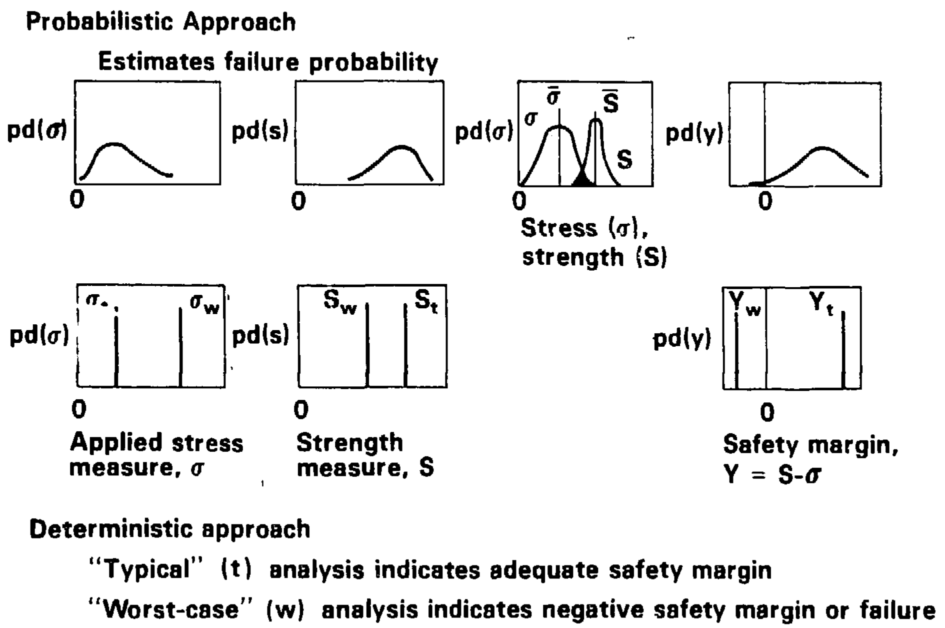

Fig. 3. Schematic representation of deterministic and probabilistic techniques for evaluating design adequacy. In the probabilistic approach, failure is possible in the cross-hatched region.

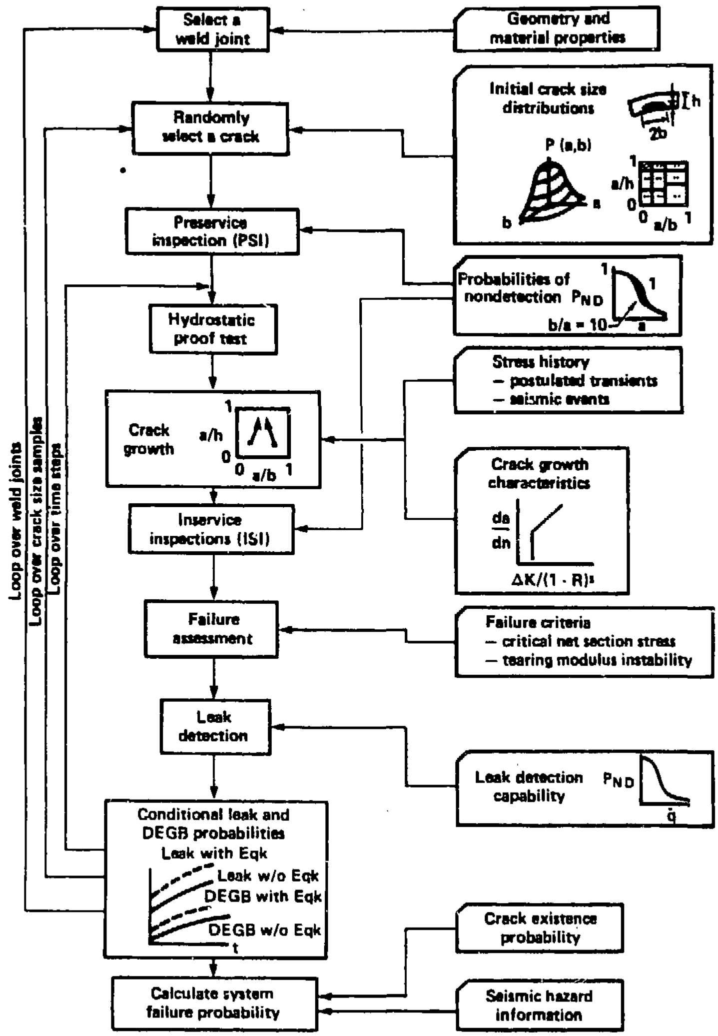

Fig. 4. Schematic representation of the probabilistic fracture mechanics simulation model implemented in the PRAISE computer code.

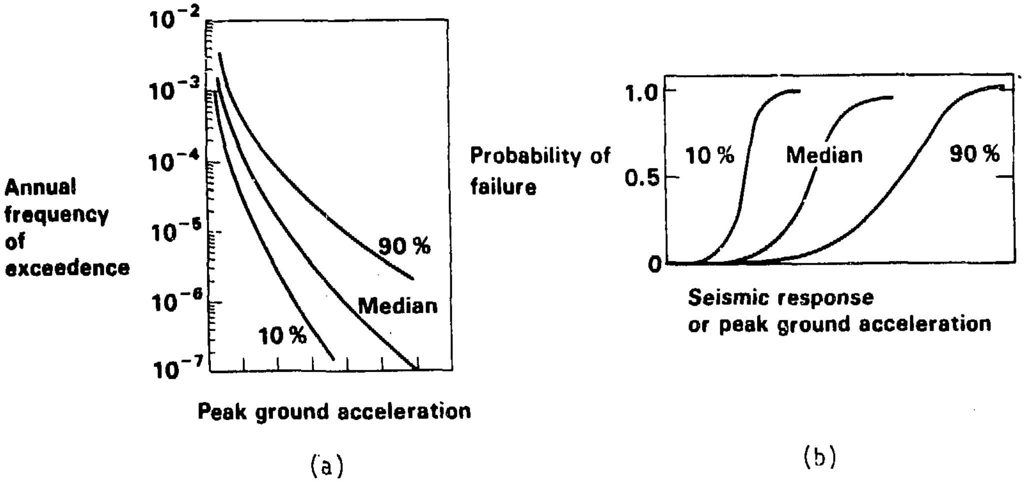

Fig. 5. Typical descriptions of seismic hazard (a) and fragility (b).