NUREG/CR-3663, Vol. 1
UCRL-53500, Vol. 1

Probability of Pipe Failure in the Reactor Coolant Loops of Combustion Engineering PWR Plants

Vol. 1: Summary Report

G. S. Holman, T. Lo, and C. K. Chou

Prepared for
U.S. Nuclear Regulatory Commission

Lawrence Livermore National Laboratory

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Manuscript Completed: October 1984
Date Published: January 1985

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Prepared for
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NRC FIN No. A0133

The NUREG/CR-3663 report series, "Probability of Pipe Failure in the Reactor Coolant Loops of Combustion Engineering PWR Plants," contains three volumes:

- Volume 1: Summary
- Volume 2: Pipe Failure Induced by Crack Growth
- Volume 3: Double-Ended Guillotine Break Indirectly Induced by Earthquakes

ABSTRACT

As part of its reevaluation of the double-ended guillotine break (DEGB) as a design requirement for reactor coolant piping, the U.S. Nuclear Regulatory Commission (NRC) contracted with the Lawrence Livermore National Laboratory (LLNL) to estimate the probability of occurrence of a DEGB, and to assess the effect that earthquakes have on DEGB probability. This report describes a probabilistic evaluation of reactor coolant loop piping in PWR plants having nuclear steam supply systems designed by Combustion Engineering. Two causes of pipe break were considered: pipe fracture due to the growth of cracks at welded joints ("direct" DEGB), and pipe rupture indirectly caused by failure of component supports due to an earthquake ("indirect" DEGB). The probability of direct DEGB was estimated using a probabilistic fracture mechanics model. The probability of indirect DEGB was estimated by estimating support fragility and then convolving fragility with seismic hazard. The results of this study indicate that the probability of a DEGB from either cause is very low for reactor coolant loop piping in these plants, and that NRC should therefore consider eliminating DEGB as a design basis in favor of more realistic criteria.

CONTENTS

FIGURES

TABLES

ACKNOWLEDGMENTS

This work was funded by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, through its Mechanical/Structural Engineering Branch. Dr. J. A. O'Brien was the technical monitor for this project.

The authors wish to express their sincere appreciation to the CE plant owners for providing the design data that went into the evaluations, and especially to Drs. T. Natan and A. Cerbone of Combustion Engineering for coordinating our interaction with the plant owners. We also wish to acknowledge the support of the CE Owners Group, which funded the development of site-specific seismic hazard curves at San Onofre used in our study.

The LLNL Load Combination Program, through which this work was performed, is a multi-disciplinary effort drawing on the talents of many individuals. We would particularly like to acknowledge the contributions of B. Benda (SMA San Ramon), C.Y. Liaw (EG&G San Ramon), and Prof. W.V. Brewer (Jackson State University), who took part in the direct DEGB evaluation, and M.K. Ravindra and R.D. Campbell (SMA Newport Beach), who performed the indirect DEGB evaluations.

EXECUTIVE SUMMARY

The Code of Federal Regulations requires that structures, systems, and components important to the safety of nuclear power plants in the United States be designed to withstand appropriate combinations of effects of natural phenomena and the effects of normal and accident conditions. Designing safety-related structures, systems, and components to withstand the effects of a large loss-of-coolant accident (LOCA) is one important load requirement. Another is that these structures, systems, and components be designed to withstand the combined effects of an earthquake and a large LOCA. The double-ended guillotine break (DEGB) of the largest reactor coolant pipe has historically been postulated as a design basis accident. Instantaneous pipe severance, followed by sufficient offset of the broken ends to allow unrestricted coolant flow out of both, characterizes DEGB. Nuclear power plant designers have generally contended that the likelihood of such an accident is so low as to be considered incredible, and that its effects would bound those of less severe breaks or leaks in other piping.

The Load Combination Program, conducted as part of the LLNL Nuclear Systems Safety Program, has performed independent confirmatory research to provide NRC with a technical basis for reevaluating the DEGB design requirement. Elimination of DEGB as a design basis event would, for example, remove the need for pipe whip restraints on primary coolant piping. If the probability of an earthquake causing DEGB is sufficiently low, then seismic loads and DEGB loads -- such as jet impingement and asymmetric blowdown -- could be decoupled in plant design.

Using probabilistic techniques, we estimate the probability of DEGB in PWR reactor coolant loop piping. Two modes of complete pipe break are considered. One is DEGB induced by fatigue crack growth resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads. We refer to this as "direct" DEGB. The other mode considers DEGB resulting from seismically-induced "indirect" causes such as the failure of supports for PWR steam generators.

We have completed probabilistic analyses indicating that the probability of direct DEGB in reactor coolant loop piping is very low for Combustion Engineering PWR plants. 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 particular, the results of our evaluations indicate that:

- the best-estimate probability of direct DEGB in reactor coolant loop piping ranges from 5.5×10^{-14} to 4.5×10^{-13} events per plant year.

- the median probability of leak (through-wall crack) in reactor coolant loop piping ranges from 1.5×10^{-8} to 2.3×10^{-8} events per plant year.

- the probabilities of leak and DEGB in reactor coolant loop piping are negligibly affected by earthquakes, to the extent that direct DEGB and earthquakes can be considered independent random events.

We have also completed analyses indicating that the probability of indirect DEGB in reactor coolant loop piping is very low for Combustion Engineering 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 seismic hazard curves with a "plant level" fragility derived from the individual component fragilities.

Based on generic seismic hazard information for the eastern U.S., our evaluation of Combustion Engineering plants indicated that the median probability of indirect DEGB is about 10^{-6} events per plant year for older plants, and less than 10^{-8} events per plant year for newer plants.

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

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

- 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 how we define seismic hazard, but is nevertheless low even when earthquakes significantly greater than the safe shutdown earthquake are considered.

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

The results of these analyses compare favorably with similar analyses that we performed for Westinghouse plants located both east and west of the Rocky Mountains.

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 CE 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:

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

Volume 1 of this report series summarizes our DEGB evaluations, including the motivation for this research and potential applications of our results. Volume 2 describes in detail our investigation of pipe failure (i.e., leak or break) due to crack growth. Volume 3 provides a detailed description of our generic evaluation of indirect DEGB for all Combustion Engineering plants.

1. INTRODUCTION

1.1 Background

The Code of Federal Regulations requires that structures, systems, and components important to the safety of nuclear power plants in the United States be designed to withstand appropriate combinations of effects of natural phenomena and the effects of normal and accident conditions.[1] The U.S. Nuclear Regulatory Commission, through its regulations, Regulatory Guides, branch technical positions, and the Standard Review Plan, has required that the responses to various accident loads and loads caused by natural phenomena be considered in the analysis of safety-related structures, systems, and components.

Designing safety-related structures, systems, and components to withstand the effects of a large loss-of-coolant accident (LOCA) is one load requirement that has been implemented by the nuclear industry for many years in the design of commercial nuclear power plants. Historically, the double-ended guillotine break (DEGB) of the largest reactor coolant pipe has been postulated as a design basis accident. Instantaneous pipe severance, followed by sufficient offset of the broken ends to allow unrestricted coolant flow out of both, characterizes DEGB. Nuclear power plant designers have generally contended that the likelihood of such an accident is so low as to be considered incredible, and that its effects would bound those of less severe breaks or leaks in other piping.

Postulation of DEGB affects many aspects of plant design. The assumption of end offset maximizes the postulated rate at which reactor coolant would be lost and therefore sets the minimum makeup capacity of emergency core cooling systems (ECCS). The escaping coolant jet would induce reaction loads at pipe and component supports, as well as mechanical loads on structures and components located in its path. If unrestrained, "whipping" pipe ends could damage structures and components in the immediate vicinity of the break. Changes in containment environment -- pressure, temperature, and humidity -- could affect the ability of safety-related mechanical and electrical components to perform their intended functions during and after a LOCA, and therefore must be designed for to assure that such equipment is "blowdown resistant." Increases in pressure and temperature following a LOCA would place substantial loads on the reactor containment.

The issue of pipe whip restraints presents a particular problem for the nuclear industry. For piping systems inside of containment, current NRC requirements stipulate that breaks be assumed at terminal ends as well as at a various intermediate locations, and that suitable restraints against pipe whip be provided accordingly. Pipe whip restraints are often very complex, very massive steel structures, congesting the already cramped confines of a typical reactor containment. Not suprisingly, pipe whip restraints represent a major capital cost for a new plant. Because they must sometimes be removed for routine in-service examination of critical welds and then reinstalled, often to close tolerances, they also increase plant maintenance costs as well as personnel exposure to radiation.

Another important requirement is that safety-related structures, systems, and components be designed to withstand the combined effects of an earthquake and a large LOCA. The combination of the most severe LOCA load with safe shutdown earthquake (SSE) loads was not controversial until several years ago when the postulated LOCA and SSE loads were both increased substantially to account for such phenomena as blowdown loads on the reactor vessel and reactor internals, referred to as "asymmetric blowdown" in pressurized water reactor (PWR) plants.

As a result of this change, the combination requirement became more difficult to implement, particularly in the design of reactor pressure vessel internals and support systems. For future plants, the change brought with it the prospect of increased construction costs. Additionally, the load combination requirement raised the issue of whether design for extreme loads will result in reduced reliability during normal plant operation. For example, present seismic design methods tend to result in stiff systems and more supports when additional strength is provided for the earthquake loading. Because a stiff system is subjected to greater cyclic thermal stress than a flexible one under normal thermal operating loads, reliability is reduced under normal conditions.[2] Restriction of pipe movement at an improperly designed or improperly installed pipe whip restraint could have the same effect.

Faced with these design, cost, and safety issues, the nuclear industry requested that the NRC reconsider the DEGB design requirement, arguing on the basis of its own calculations and experimental research that DEGB was an extremely unlikely event. From a safety standpoint, costs alone can not be a justification for changing design requirements; the costs of meeting these requirements are industry's responsibility. However, for existing plants to comply with the revised loading criteria and also satisfy the combination requirement, modification is almost unavoidable. Certain plants can be feasibly modified, but other plants not feasible to modify present a difficult problem to the NRC. The NRC must either challenge the safety of continued operation without modifications, or reassess the design requirement and allow continued operation with no or only limited modifications.

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 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.

Elimination of DEGB as a design basis event for PWR reactor coolant loop piping could have far reaching consequences. If it can be shown that an earthquake will not induce DEGB, then the two can be considered independent random events whose probability of simultaneous occurence is negligibly low; thus, the design requirement that DEGB and SSE loads be combined could be removed. If the probability of a DEGB is very low under all plant conditions, including seismic events, then asymmetric blowdown loads in PWR plants could be eliminated. Reaction loads on pipe and component supports could be reduced. Jet impingement loads, as well as environmental effects due to a LOCA, could be modified accordingly. Pipe whip restraints could be eliminated altogether, as without a double-ended break, the pipe would retain at least geometric integrity. This last benefit would apply to operating plants as well as to those in design or under construction, because once removed for periodic weld inspection, pipe whip restraints would not have to be reinstalled.

1.2 Objectives

The overall objective of the LLNL Load Combination Program is to estimate the probability that a double-ended guillotine break occurs in the reactor coolant piping of light water reactor power plants. We consider two potential causes for DEGB, namely:

- fatigue crack growth at welded joints resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads;

- earthquake-induced failure of component supports or other equipment whose failure would in turn cause a reactor coolant pipe to break.

In the nomenclature of our study we refer to these two cases as "direct" and "indirect" DEGB, respectively.

1.3 Scope

The work presented in this report is a continuation of work performed in Phase I of the Load Combination Program. In Phase I we developed a probabilistic fracture mechanics methodology for estimating the likelihood of direct DEGB in the reactor coolant loop piping of PWR plants. We applied this methodology in an extensive pilot study of a single Westinghouse PWR plant, Zion Unit 1 operated by the Commonwealth Edison Company of Illinois. We also performed a limited study in which we identified the supports of the reactor pressure vessel, reactor coolant pump, and steam generators as critical components whose failure could indirectly induce DEGB, and estimated the probability that any one of these supports could fail. The resultant probability of DEGB in the reactor coolant piping was, however, not investigated in Phase I.

The Phase I investigations were documented extensively[3] and presented before the Advisory Committee on Reactor Safegaurds (ACRS) in December 1980. Following this presentation, the ACRS asked us to perform three additional studies: (1) evaluate indirect DEGB in depth, (2) assess the effect of design and construction errors on the probability of indirect DEGB, and (3) generalize the Zion study to include other PWR plants. This request forms the basis for the work reported here.

To arrive at a general conclusion about the probability of DEGB in the reactor coolant loop piping of PWR plants, LLNL has taken a vendor-by-vendor approach. For each of the three PWR vendors (Westinghouse, Babcock & Wilcox, and Combustion Engineering) our specific objectives 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.

(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.

We have completed generic evaluations of DEGB probability for plants with nuclear steam supply systems manufactured by Combustion Engineering (CE), which are reported herein, as well as for plants having nuclear steam supply systems manufactured by Westinghouse.[4] The results of these evaluations indicate that the probability of DEGB from either cause is very low, and suggest 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 our Westinghouse and Combustion Engineering evaluations, we 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 NSSS vendors, we 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.

The investigations described in this report are limited to estimating the generic probability of DEGB in reactor coolant loop piping of Combustion Engineering PWR plants. Each reactor coolant loop, of which all CE plants (with the single exception of Maine Yankee) have two, consists of three sections -- the hot leg, two cold legs, and two suction (crossover) legs -- connecting the reactor pressure vessel, one steam generator, and two reactor coolant pumps. The loops are identical, except for one which also includes the pressurizer, used to control system volume. Neither the pressurizer or the interconnecting surge line are included in the present study. The reactor coolant pipes typically have inside diameters of 30 to 42 inches, and walls that are approximately 3 to 4 inches thick. Because they are short and stiff, the pipes are supported solely by the major loop components; no additional supports are necessary. Reactor coolant loop piping is fabricated from carbon steel with stainless steel inner cladding, except for Fort Calhoun which has stainless steel piping.

To estimate the probability of direct DEGB, we only considered fatigue crack growth from the combined effects of thermal, pressure, seismic, and other cyclic loads as the mechanism leading to pipe leak or break. Hydrodynamic loads due to water hammer were not considered because they have never been observed in PWR reactor coolant loop piping. Likewise, we also excluded intergranular stress corrosion cracking (IGSCC) from consideration because stress corrosion problems have not been observed in ferritic pipe materials.

In addition to our fracture mechanics evaluation, we also present an investigation of DEGB indirectly induced by earthquakes. To estimate the probability of indirect DEGB, we considered the safety margins against seismic failure for critical components whose failure could in turn cause a reactor coolant pipe to break. By combining this information with a suitable probability distribution of ear quake intensity (seismic hazard), we were able to estimate the probability of guillotine break caused by earthquakes.

Through sensitivity studies, we also considered the effects of gross design and construction errors on the probability of indirect DEGB.

Probabilistic risk assessments of nuclear power plants have indicated that the break of a smaller pipe may be more probable, and that such a small LOCA may pose a larger overall plant risk. Nevertheless, the reactor coolant pipes are of the most immediate interest for NRC confirmatory research because their failure would generate the most severe LOCA loads. Although we have limited our present study accordingly, we believe that the methodologies and general concepts presented here could be extended to assess the probability of DEGB in other piping systems.

1.4 Probabilistic Approaches to Failure Evaluation

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. 1, 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.[5] The application of "safety margins" provides added conservatism in component design. The safety margin compensates for uncertainty associated with many factors, including:

- variability in nominal material strength, that is, actual strength may be lower than that specified in the analysis.
- degradation in material strength during plant operation, such as radiation embrittlement.

- variations in postulated loading conditions such as pressure and temperature transients.

- load conditions generally regarded as having secondary significance and which are therefore neglected in the evaluation.

- unanticipated load conditions.

- simplifications made in modeling a physical system.

- 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 a "worst-case" evaluation based on maximum stress and minimum 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:

- the margin between code allowable limits and actual failure.

- the margin between design conditions and code limits.

- the particular analytic techniques used to predict component response to applied loads.

- 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. 1), 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 of non-destructive examination (inspection interval, crack non-detection probability). Such sensitivity studies can give important information about unsound design areas and about how each parameter influences the overall reliability of a design.

The distinction between deterministic and probabilistic approaches widens as the number of parameters involved in the calculation increases. The more parameters involved, the more conservative a deterministic analysis tends to be because conservatisms embedded in each parameter add together. This problem is avoided by a probabilistic analysis.

Because of its capabilities, the probabilistic approach is seeing increased application in many engineering fields. Nevertheless, the deterministic approach still plays an important role, especially when statistical data for certain parameters is not adequate. Thus, rather than one being an alternative for the other, deterministic and probabilistic approaches complement each other for assessing design reliability.

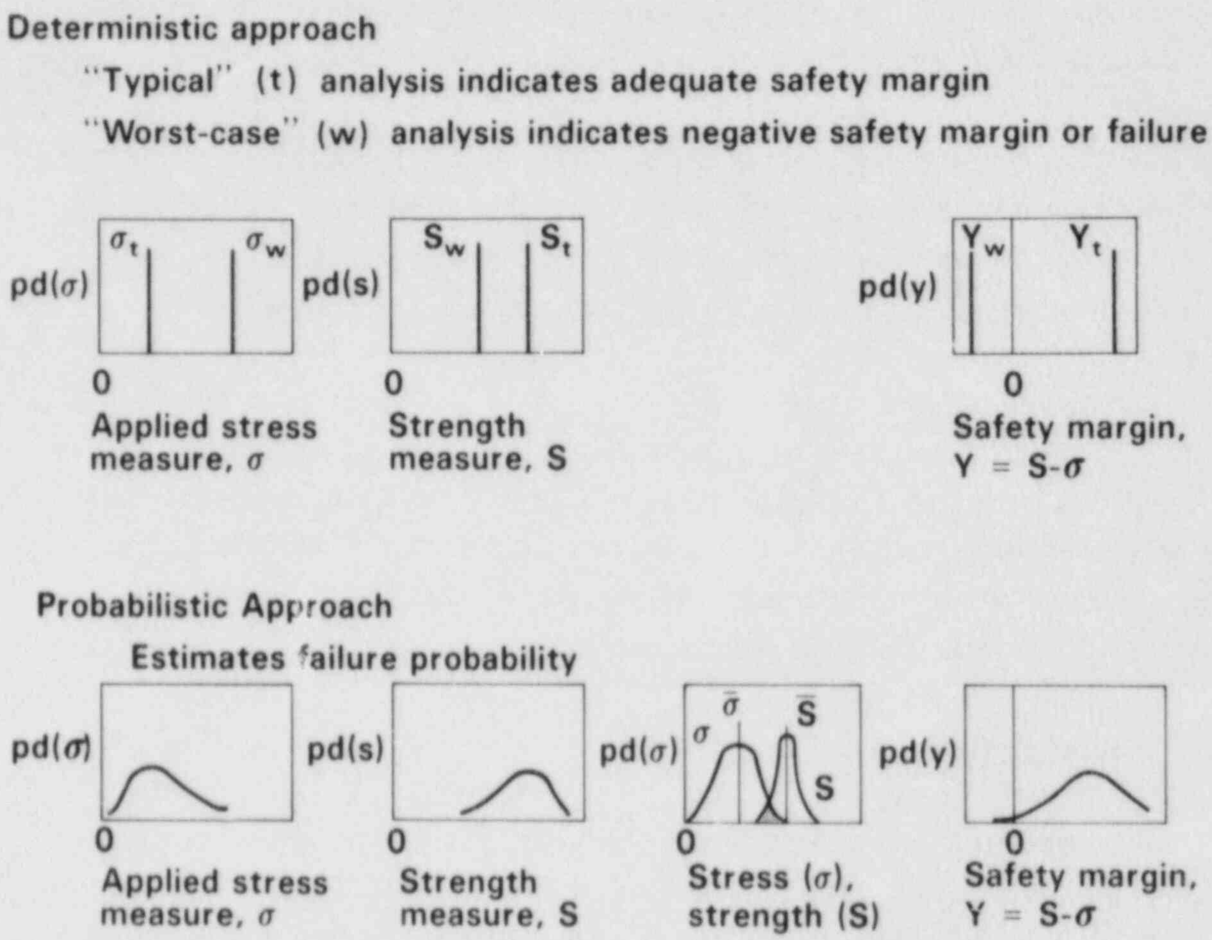

Figure 1. Comparison between probabilistic and deterministic approaches for assessing component adequacy for postulated load conditions. In the probabilistic representation, failure is possible only in the shaded region.

2. GENERAL PLANT DESCRIPTION

2.1 Plant Grouping

In the United States there are at present 15 nuclear power units (on 10 plant sites) that have a nuclear steam supply system manufactured by Combustion Engineering. These plants are divided by CE into four groups according to vintage, material used in reactor coolant loop piping, number of reactor coolant loops, and type of supports for loop components. Table 1 lists these plant groups together with various characteristics of each plant. In general, Group A includes plants of older vintage, while the Group C plants are all modern plants of Combustion Engineering's "System 80" product line. Groups B and D each contain only one plant, the former Fort Calhoun because it is the sole CE plant having stainless steel piping, and the latter Maine Yankee, the only CE plant having more than two reactor coolant loops.

In our study, we estimated the probability of direct DEGB only for plants in Groups A and C. Information for Maine Yankee was not available, and the fracture mechanics characteristics of piping at Fort Calhoun are more similar to those of Westinghouse plants than other CE plants. Fort Calhoun is therefore covered by the direct DEGB evaluation for Westinghouse plants. The plant was, however, included in our evaluation of indirect DEGB probability.

2.2 Reactor Coolant Loop Piping

All CE nuclear steam supply systems (except Maine Yankee) have two reactor coolant loops, each of which has two branches. Each branch is a loop by itself and shares with the other branch a common hot leg and a common steam generator, which are substantially larger than those of Westinghouse plants of similar power rating. The reactor coolant loop pipes are connected to loop components at both ends, and there are no intermediate supports.

Figure 2 shows the general reactor coolant loop arrangement of a typical two-loop system. The coolant flows from the reactor vessel to one of the steam generators through a hot leg with an inside diameter of 42 inches. The loop branches into two suction legs at the steam generator. A reactor coolant pump, located on each side of the steam generator, pumps the coolant back into the reactor vessel through a discharge leg. The inside diameter of the suction and discharge legs is approximately 30 inches, which is comparable to that of the crossover legs and cold legs of Westinghouse plants. The CE reactor coolant loop system is pressurized to approximately 2250 psi during operation. The coolant temperature downstream from the steam generator is approximately 550 $^{\circ}F$, while the temperature in the hot leg is some 50 to 60 $^{\circ}F$ higher.

There are typically 29 or 31 circumferential welds in each loop. Table 2 gives typical pipe dimensions at the welds for Palo Verde Unit 1, and compares these with the equivalent Westinghouse pipe dimensions. The loop piping in all CE plants except for Fort Calhoun is fabricated from SA-516 Grade 70 carbon steel, with a stainless steel inner cladding at least 1/8-inch thick. Most welds are shop welds; there are only about two field welds in each leg of the piping. The shop welds are believed to be of higher quality than the field welds; however, we made no distinction between shop and field welds in our evaluations. The welds were stress relieved, and therefore we did not include residual stresses in our analyses.

2.3 Reactor Coolant Loop Supports

The supports for the major loop components (Figs. 3 and 4) are generally composed of specially manufactured mechanical parts. Unlike the Westinghouse support system, CE systems have no standard structural steel members, thereby eliminating welding. The reactor vessel is supported by columns at the nozzles. The steam generators are supported at two elevations: the upper support consists of keys in one direction and lever-snubber arrangements in the other, the lower support is a skirt with a sliding base that allows free thermal expansion. The reactor coolant pump supports are generally of the pin-and-column type with snubbers to resist seismic loads, although early model supports have skirts and spring hangers.

TABLE 1

List of Combustion Engineering Plants with NSSS Characteristics

Group A *	Net MWe	Loops/Pumps	Pump Support Type
Calvert Cliffs 1,2	850	2/4	Spring hanger and snubber
Millstone 2	828	2/4	Spring hanger and snubber
Palisades	740	2/4	Spring hanger and snubber
St. Lucie 1,2	777	2/4	Spring hanger and snubber
Group B †			
Fort Calhoun	457	2/4	Custom design by architect-engineer
Group C			
Palo Verde 1,2,3	1270	2/4	Column and snubber
San Onofre 2,3	1100	2/4	Column and snubber
WPPSS 3	1240	2/4	Column and snubber
Waterford 3	1165	2/4	Column and snubber
Group D			
Maine Yankee **	790	3/3	Skirt

* For the DEGB evaluation of Group A plants, Combustion Engineering provided a composite plant whose parameters enveloped those of the individual plants.

† Not included in direct DEGB evaluation because of stainless steel reactor coolant loop piping.

** Not included in direct or indirect DEGB evaluation.

TABLE 2

Comparison of Typical Sizes and Operating Conditions for Westinghouse and Combustion Engineering Reactor Coolant Loop Piping

	Pressure MPa (psia)	Temperature °C (°F)	Inside Diameter cm (in)	Wall Thickness cm (in)
Westinghouse (Zion)				
Hot leg	15.4 (2235)	311 (592)	73.7 (29.0)	6.35 (2.50)
Crossover	15.4 (2235)	277 (530)	78.7 (31.0)	6.76 (2.66)
Cold leg	15.4 (2235)	277 (530)	69.9 (27.5)	6.05 (2.38)
Pipe material: Type 316 stainless steel				
Combustion Engineering (Palo Verde)				
Hot leg	15.5 (2250)	327 (621)	106 (42.0)	9.53 (3.75)
Suction	15.5 (2250)	296 (565)	76.2 (30.0)	6.35 (2.50)
Discharge	15.5 (2250)	296 (565)	76.2 (30.0)	6.35 (2.50)
Pipe material: Type SA516 Grade 70 carbon steel with 0.32 cm (0.125 in) stainless steel cladding				

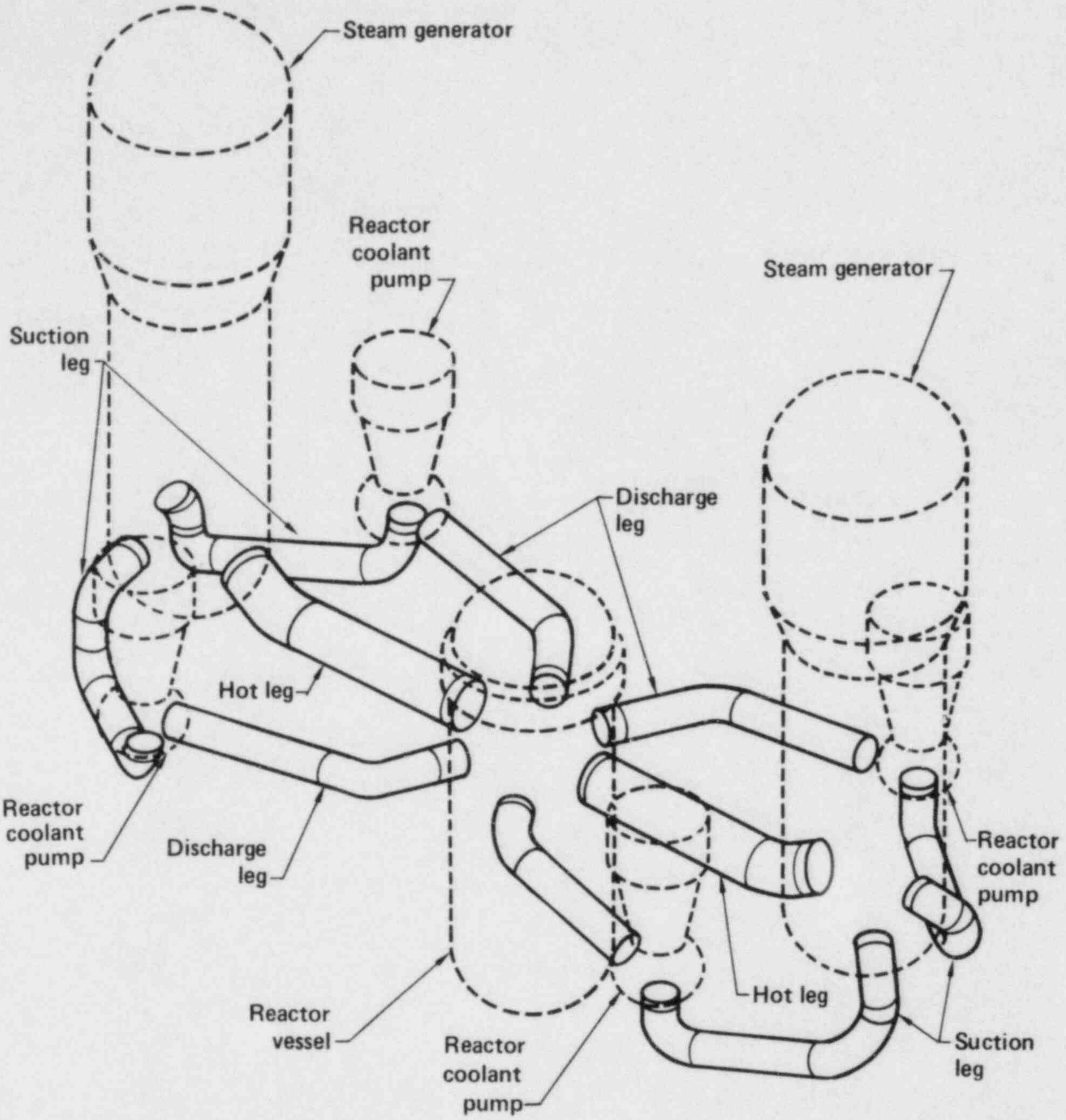

Figure 2. General arrangement of a typical Combustion Engineering two-loop nuclear steam supply system.

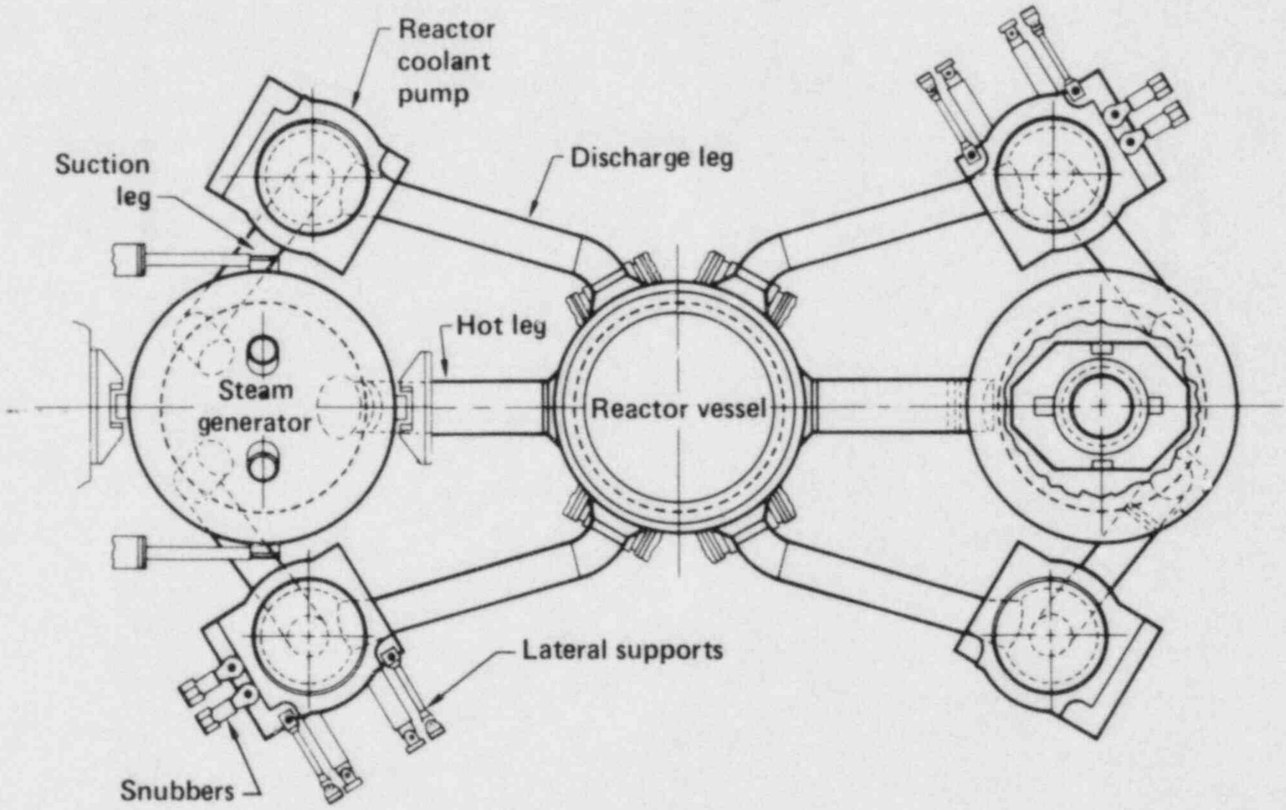

Figure 3. General arrangement of Combustion Engineering reactor coolant loop supports.

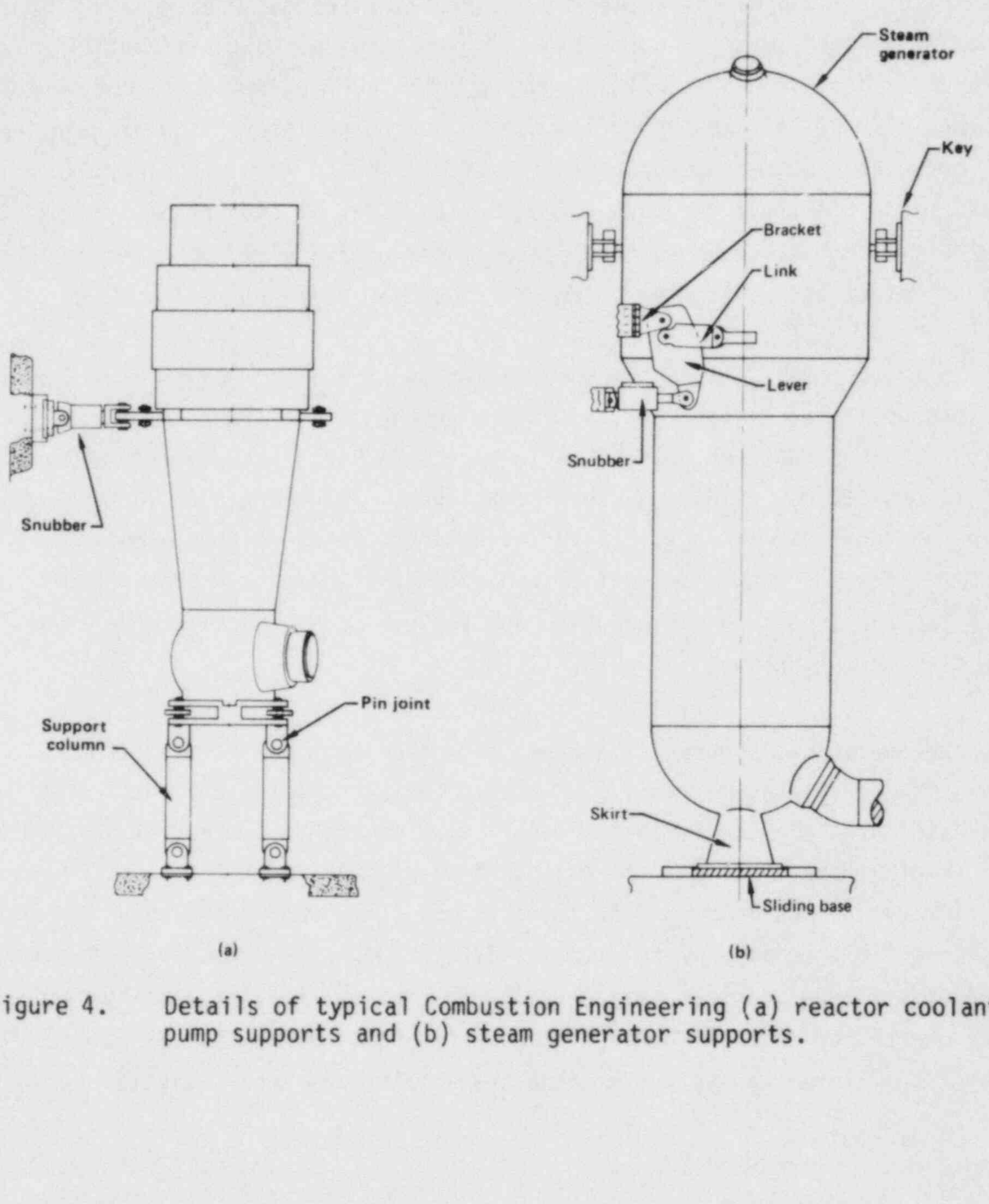

Figure 4. Details of typical Combustion Engineering (a) reactor coolant pump supports and (b) steam generator supports.

3. PIPE FAILURE INDUCED BY CRACK GROWTH

3.1 Probabilistic Fracture Mechanics Model

The postulated mechanism leading directly to a pipe failure (here defined as either leak or DEGB) is the growth of cracks at welded pipe joints. Cracks can exist before a nuclear power plant begins service -- an artifact of improper welding or heat treatment during pipe fabrication or assembly -- or can initiate during plant operation due to corrosive interaction between the pipe material and the reactor coolant. If allowed to grow unchecked, such cracks could penetrate the pipe wall, causing leaks or even break. It is therefore important to understand not only how cracks grow, but also to be able to detect and monitor existing cracks during plant operation.

To model crack growth during the lifetime of a plant and thus estimate the probability of direct DEGB, we used a probabilistic fracture mechanics approach. This approach, described in detail in Ref. 6 and in Volume 2 of this report series, allowed us to account for the randomness of load events and parameters associated with plant operation. Figure 5 is a simplified flow chart of the approach. The left column shows the analytical procedure, the right the required input information and the various simulation models used at each step of the analysis.

The analytical process is divided into two parts. The first, implemented in the PRAISE (Piping Reliability Analysis Including Seismic Events) computer code, estimates the conditional probabilities of leak and break at individual weld joints, given that a crack exists at that joint, that the plant experiences various loading conditions at any time, and that a seismic event of a specific intensity occurs at a specific time. The second part estimates the probability of "system failure", in other words, the probability that at least one of the weld joints in a pipe system fails during the lifetime of the plant. The system analysis estimates the absolute (or non-conditional) probabilities of leak and break for the entire pipe system by convolving (1) the conditional leak and break probabilities at all of the associated weld joints, (2) the non-conditional probability that at least one crack, regardless of size, exists at a weld joint, and (3) the relationship between intensity of seismically-induced ground motion and earthquake occurence rate ("seismic hazard").

Except where noted otherwise, failure probabilities in this report are presented in terms of failure events per plant-year. It is important to point out that the system failure analysis actually yields the cumulative failure probability over the entire duration of plant life (assumed to be 40 years) from which the annual failure probability was derived by assuming that system failure probabilities are uniform over the entire duration.

It is also important to emphasize that this probabilistic fracture mechanics model 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.

The following two sections discuss each part of the analysis in greater detail.

3.2 Failure Probability of a Weld Joint

For each weld joint of the piping system, we used a Monte Carlo simulation algorithm to calculate the conditional leak and DEGB probabilities at any specific time during plant life. The weld joint was subjected to a stress history associated with plant events, such as normal heatup and cooldown, anticipated transients, and the occurence of potential earthquakes.

Each replication of the simulation -- a typical PRAISE simulation may include 10,000 or more -- starts with the random selection of a sample crack size from a "stratified" sampling space (see Vol. 2, Appendix A) and then determines its conditional existence probability from appropriate distributions of crack depth and length. Fracture mechanics theory is then applied to calculate the growth of the crack and to determine if pipe failure (i.e., leak or break) occurs during the plant lifetime. As shown in Fig. 5, various factors affecting crack growth are simulated: preservice inspection using non-destructive examination (NDE) techniques, hydrostatic proof test, in-service inspections, leak detection.

Fatigue crack growth takes into account the cyclic stress history of various thermal transients and postulated seismic events. A failure criterion based on either net section stress or tearing modulus instability is applied to define when pipe failure occurs, depending on their applicability to the material characteristics and the geometric conditions of the pipe. The stress state of the plant varies as the various loading events occur throughout plant life. Therefore, we monitor or calculate the state of the cracks, considering the effects of these loading events as time progresses. The time of occurrence of these loading events can be either deterministic or stochastic. In this study, we treat the seismic events as stochastic and assume them to be describable by a Poisson process in calculating the system failure probability. Other plant transients are considered to be uniformly spaced throughout plant life.

Most of the significant plant events, such as heatup and cooldown, are more or less uniform in nature. Other events are either insignificant, or we were unable to determine a more suitable spacing. The frequencies of thermal transient events used in the analysis are based on design postulations and are considered to be conservative.

The pre-service inspection was performed once before the plant began operation, as is the actual case. Although we can also model in-service inspections, we neglected these in our analyses because inspection programs vary greatly from plant to plant and therefore cannot be modeled with reasonable confidence. Neglecting in-service inspection adds conservatism to the results.

We assessed the effect of an earthquake of specific intensity on the failure probability at each weld joint at specific times during the plant life. First we determined the probability of failure with no seismic events. Then we imposed earthquakes of specified intensity, usually expressed in terms of peak ground accelerations, on normal operating conditions. The increase in the failure probability after the earthquake was added represents the contribution of the seismic event to the failure probability. This process was repeated for a wide range of earthquake intensities.

As previously noted, the PRAISE simulation yields the conditional leak and DEGB probabilities as a function of time for a specific weld joint. This analytical process is repeated for all welds in one loop of the total reactor coolant system. The two loops of a given CE nuclear steam supply system are assumed to be identical in geometry and to have identical stress behavior at each corresponding weld joint; therefore, the corresponding joint failure probabilities are assumed identical.

3.3 System Failure Probability

The second part of the analysis estimates the non-conditional system probabilities of leak and break by combining the conditional probabilities yielded by the Monte Carlo simulation with the non-conditional crack existence probability and the seismic hazard.

The probability of pipe failure is potentially affected by both the intensity and the occurrence rate of earthquakes. In our evaluations, earthquake intensities expressed in terms of peak ground acceleration (PGA) can range from zero up to five times the safe shutdown earthquake (SSE). For this study, an earthquake is defined as ground motion with peak free field acceleration above a certain threshold value below which no significant structural damage is expected to occur. The value of this threshold acceleration is subjective; however, a sensitivity study that we performed indicated that the estimated system failure probability is not significantly affected by the choice of this parameter.

Earthquake occurrence rate is expressed in terms of "seismic hazard", defined as the probability that an earthquake will occur causing different levels of peak ground acceleration. This is usually decribed by a set of seismic hazard curves (Fig. 7) plotting exceedance probability as a function of peak ground acceleration. Our evaluation of direct DEGB in plants east of the Rocky Mountains was based on the same generic hazard curves developed for our investigations of indirect DEGB; west coast plants were evaluated using site-specific seismic hazard information.

In evaluating the probability of direct DEGB, we considered three events in which failure occurs in reactor coolant loop piping:

(1) failure occurs simultaneously with the first earthquake occurring during plant life.

(2) failure occurs prior to the first earthquake occurring during plant life.

(3) failure occurs with no earthquake occurring during plant life.

Probabilities of direct DEGB were calculated independently for each event and then combined into an overall probability that pipe failure occurs sometime during plant life. A fourth event, one or more earthquakes occurring during plant life with failure occurring after the first earthquake, was neglected because presumably the plant would be shut down for a complete inspection and repairs after the first earthquake.

3.4 Uncertainty Analyses

Two types of variability, or uncertainty, are associated with each of the parameters considered in this study. One type, random uncertainty, represents the inherent physical variation or randomness of the parameters. Modeling uncertainty, the other type, accounts for the lack of complete knowledge or detailed information about the parameters to describe them precisely.

To illustrate these two types of uncertainties, consider flow stress (the average of yield and ultimate stresses) of a specific material as an example. Because of the physical variability of materials and structures, flow stress is inherently variable. The variability, i.e., randomness, of flow stress can be described, for example, by a normal probability distribution characterized by a mean and standard deviation. Estimates of the mean and standard deviation for a specific type of material can be derived from test samples.

If the number of test samples is limited, then we would be uncertain in the estimated values of the mean and standard deviation and therefore in our description of the random variation of flow stress. This is modeling uncertainty. Also, we might have some uncertainty about how well the normal distribution describes the variability of flow stress. Perhaps another distribution, such as a log-normal distribution, would be better. This uncertainty would be another contributor to the modeling uncertainty associated with the flow stress.

There are many sources of modeling uncertainty. Some additional examples include uncertainties associated with:

- the selection of methods for modeling soil-structure interaction, such as the finite-element approach and impedance approach.
- the selection of methods for modeling structural response, such as response spectrum vs time-history analysis, two- or three-dimensional analysis, coupled vs uncoupled models of structures and equipment.
- the selection of damping values used to model various energy absorbing mechanisms in structures.
- the estimation and sampling methods used in the probability analysis, including uncertainties in the Monte Carlo simulation technique.
- the inherent randomness in parameters other than flow stress.

A deterministic value will often suffice to represent a parameter if the variation is negligible; otherwise, a distribution is required. We used appropriate distributions to describe the inherent randomness in many of the parameters. In addition, we found it necessary to quantify the modeling uncertainties for five parameters that sensitivity studies had shown were particularly important to the fracture mechanics evaluation: initial crack depth, initial crack length, thermal stress, seismic stress, and seismic hazard. Because the random uncertainties of input parameters contribute to the value of pipe failure probability, they are intrinsic to the analytic process illustrated in Fig. 5. We treated modeling uncertainties in a different manner, by defining several sets of these five parameters through Latin Hypercube sampling and then estimating the probability of failure for each set. In this way we developed a distribution about the "best estimate" probability of failure. The details of our uncertainty analyses are provided in Volume 2 of this report series.

3.5 Discussion of Results

Probability of Direct DEGB

We began our evaluation of Combustion Engineering PWR plants with a detailed reference study of the Palo Verde nuclear power plant. Similar in purpose to the pilot study in our Westinghouse evaluation but less extensive in scope, the CE pilot evaluation concentrated mainly on developing a tearing instability failure criterion for ferritic pipe materials. We also conducted extensive sensitivty studies to identify key parameters affecting the probability of DEGB, and performed uncertainty analyses to establish confidence bounds on the final DEGB probability. Thus, the pilot study served to develop and "shake down" the assessment methodology that we applied in subsequent generic studies.

After completing the Palo Verde study, we performed a generic evaluation of DEGB probability for other Combustion Engineering plants. In contrast to our Westinghouse study, where we first reviewed for each plant the important factors contributing to DEGB probability, and then grouped similar plants together, in the CE study we performed "best estimate" calculations for each of the four Group C plants as well as for a composite plant enveloping all of the Group A plants (see Table 1). Thus, we obtained plant-specific "best estimates" of DEGB probability and leak probability.

From our results we concluded that the best-estimate probability of direct DEGB is very low, ranging between 5.5×10^{-14} and 4.5×10^{-13} events per plant year (see Table 3).

To account for modeling uncertainty, we also placed distributions on the five parameters that our earlier Westinghouse pilot study (using Zion Unit 1 as pilot plant) had indicated most significantly affect the probability of DEGB: initial crack depth, initial crack length, thermal stresses, seismic stresses, and seismic hazard. We then performed uncertainty analyses to establish confidence bounds on the estimated probability of DEGB. The 10th and 90th percentile probabilities of DEGB , as well as upper and lower bound probabilities, are presented in Table 4. Based on our uncertainty analyses, a probability of 10^{-10} events per plant-year appears reasonable as an approximate upper bound on the probability of direct DEGB in CE reactor coolant loop piping.

Probability of Leak

The best-estimate probabilities of leak (Table 5) varied over a narrow range of 1.5×10^{-8} to 2.3×10^{-8} events per plant year. Uncertainty analyses yielded the values included in Table 6; from these analyses, a value of about 2×10^{-7} events per plant year appears reasonable as an approximate upper bound on the probability of leak in CE reactor coolant loop piping. The significantly higher leak probability compared to DEGB probability tends to suggest the validity of the leak-before-break concept for reactor coolant loop piping.

Effect of Earthquakes

In evaluating the probability of direct DEGB, we considered three events in which failure occurs in reactor coolant loop piping:

- failure occurs simultaneously with the first earthquake occurring during plant life (i.e., failure caused by an earthquake).
- failure occurs prior to the first earthquake occurring during plant life.
- failure occurs with no earthquake occurring during plant life.

Probabilities of direct DEGB were calculated independently for each event and then combined into an overall probability that pipe failure occurs sometime during plant life (see Table 6). It was found for both leak and DEGB that the probability of the first event -- simultaneous occurence of failure and an earthquake -- was typically one to three orders of magnitude less than that of failure occurring independently of an earthquake. The results of the study of CE plants indicated therefore that the probability of an earthquake causing direct DEGB is negligible.

TABLE 3

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

(events per plant year)

	Event[(1)]			
	Event 1	Event 2	Event 3	P[DEGB]
Palo Verde 1,2,3	6.5×10^{-16}	2.7×10^{-14}	4.3×10^{-13}	4.5×10^{-13}
San Onofre 2,3	2.4×10^{-15}	1.0×10^{-13}	1.9×10^{-15}	1.0×10^{-13}
WPPSS 3	3.3×10^{-16}	6.4×10^{-15}	5.4×10^{-14}	6.1×10^{-14}
Waterford 3	4.7×10^{-15}	4.6×10^{-15}	8.0×10^{-14}	9.0×10^{-14}
Group A [(2)] Composite	1.6×10^{-15}	2.9×10^{-15}	5.1×10^{-14}	5.5×10^{-14}
Westinghouse[(3)]	5.3×10^{-14}	3.3×10^{-12}	3.0×10^{-12}	6.3×10^{-12}

(1) Event 1: Probability of DEGB coincident with first earthquake
Event 2: Probability of DEGB prior to first earthquake
Event 3: Probability of DEGB with no earthquake
P[DEGB]: Combined probability of DEGB

(2) See Table 1 for plants included in Group A

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

TABLE 4

Uncertainty Values for Probability of Direct DEGB in Reactor Coolant Loop Piping of Combustion Engineering PWR Plants

(events per plant year)

		Confidence Limit (1)		
	Lower Bound	10%	90%	Upper Bound
Palo Verde 1,2,3	0.	4.5×10^{-17}	7.2×10^{-11}	8.4×10^{-11}
San Onofre 2,3	0.	1.1×10^{-18}	3.9×10^{-12}	1.0×10^{-12}
WPPSS 3	0.	1.3×10^{-19}	1.2×10^{-11}	2.1×10^{-11}
Waterford 3	0.	1.0×10^{-17}	3.5×10^{-11}	5.9×10^{-11}
Group A (2) Composite	0.	2.2×10^{-18}	8.8×10^{-12}	1.7×10^{-11}

(1) Upper and lower bounds are, respectively, the highest and lowest probabilities resulting from the uncertainty analysis. A confidence limit of 90% implies that there is a 90% subjective probability (confidence) the probability of direct DEGB is less than the value indicated.

(2) See Table 1 for plants included in Group A

TABLE 5

Best-Estimate Probabilities of Leak in Reactor Coolant Loop Piping of Combustion Engineering PWR Plants

(events per plant year)

	Event(1)			
	Event 1	Event 2	Event 3	P[Leak]
Palo Verde 1,2,3	4.5×10^{-11}	9.4×10^{-10}	1.4×10^{-8}	1.5×10^{-8}
San Onofre 2,3	5.8×10^{-10}	2.1×10^{-8}	3.6×10^{-10}	2.2×10^{-8}
WPPSS 3	1.0×10^{-10}	2.0×10^{-9}	1.6×10^{-8}	1.8×10^{-8}
Waterford 3	6.7×10^{-10}	9.6×10^{-10}	1.6×10^{-8}	1.8×10^{-8}
Group A (2) Composite	6.1×10^{-10}	1.3×10^{-9}	2.1×10^{-8}	2.3×10^{-8}
Westinghouse(3)	3.5×10^{-11}	6.0×10^{-8}	5.8×10^{-8}	1.2×10^{-7}

(1) Event 1: Probability of leak coincident with first earthquake
Event 2: Probability of leak prior to first earthquake
Event 3: Probability of leak with no earthquake
P[Leak]: Combined probability of leak

(2) See Table 1 for plants included in Group A

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

TABLE 6

Uncertainty Values for Probability of Leak in Reactor Coolant Loop Piping of Combustion Engineering PWR Plants

(events per plant year)

	Confidence Limit (1)			
	Lower Bound	10%	90%	Upper Bound
Palo Verde 1,2,3	3.3×10^{9}	5.1×10^{-9}	6.8×10^{-8}	1.9×10^{-7}
San Onofre 2,3	3.5×10^{-9}	5.6×10^{-9}	9.5×10^{-8}	1.4×10^{-7}
WPPSS 3	3.6×10^{-9}	4.4×10^{-9}	8.8×10^{-8}	1.5×10^{-7}
Waterford 3	4.2×10^{-9}	4.7×10^{-9}	1.0×10^{-7}	1.1×10^{-7}
Group A (2) Composite	3.4×10^{-9}	6.0×10^{-9}	8.4×10^{-8}	1.4×10^{-7}

(1) Upper and lower bounds are, respectively, the highest and lowest probabilities resulting from the uncertainty analysis. A confidence limit of 90% implies that there is a 90% subjective probability (confidence) the probability of leak is less than the value indicated.

(2) See Table 1 for plants included in Group A

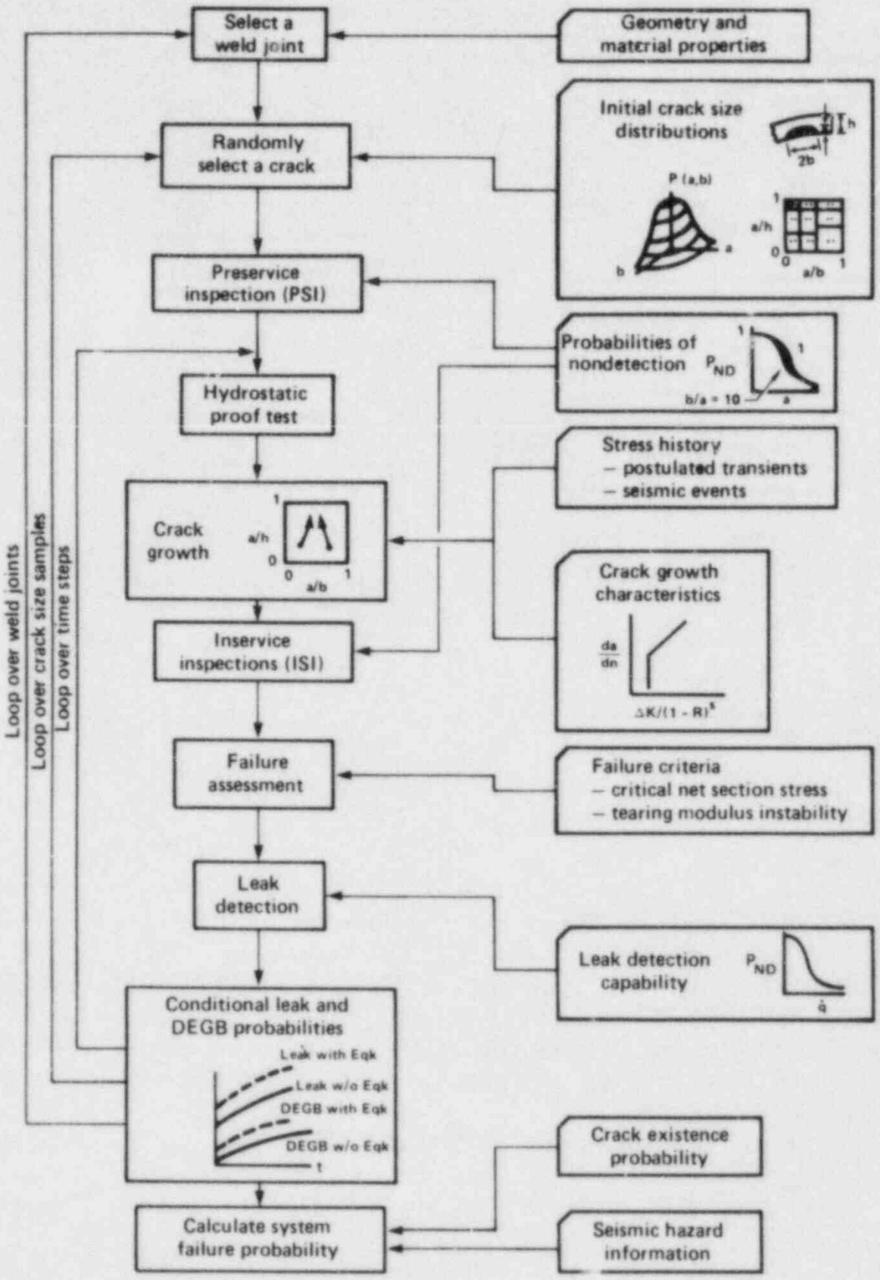

Figure 5. Flowchart of the probabilistic fracture mechanics model implemented in the PRAISE computer code.

4. DOUBLE-ENDED GUILLOTINE BREAK INDIRECTLY INDUCED BY EARTHQUAKES

4.1 Methodology

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 loop 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 three steps. First, we identify critical components and determine the seismic "fragility", or relationship between response under seismic load and probability of failure, of each. Next, we determine for each component the probability that its failure will lead to DEGB. Finally, we combine statistically, or "convolve", the probability distribution of earthquakes for a reactor site with a "plant level" fragility derived from the individual component fragilities to estimate the non-conditional probability that indirect DEGB will occur.

As we did in our evaluations of pipe failure due to crack growth, we established confidence bounds on the probability of indirect DEGB by attaching uncertainties to the parameter values, in this case seismic fragility and seismic hazard.

4.2 Component Fragility

The seismic fragility of a component is defined as the conditional probability of its failure given a peak ground acceleration level. We included in our study only those "critical" elements whose failure could contribute significantly to the probability of an indirectly-induced DEGB. Based on our experience in the Westinghouse evaluation, we identified as critical components the steam generator supports, the reactor coolant pump supports, and the reactor pressure vessel supports. For each, the modes of failure were identified and the mean capacity calculated. We also calculated the uncertainty in capacity. Loads that each equipment support would experience during a seismic event were obtained using appropriate dynamic models. The response of each critical support element to dead loads, thermal loads, and seismic loads was found. From response calculation results we estimated mean seimsic loads and their variabilities. Finally, we computed the median factor of safety against seismic failure and the logarithmic standard deviations representing randomness and modeling uncertainty.

As in our study of Westinghouse plants, we evaluated fragilities using information on equipment failure modes, design margins and seismic response supplied to us by the NSSS vendors; no new response calculations were performed. Because design calculations inherently include conservatisms to account for such effects as soil-structure interaction, modeling assumptions, structural damping, and others (see Table 7), we applied correction factors to these design margins to obtain a "best estimate" of the actual margin against failure. For each component, we then combined the probability distributions of its capacity and seismic response to obtain a "fragility curve" (Fig. 6) describing the probability of component failure as a function of peak ground acceleration.

Next, the conditional probability of DEGB given failure of each component was established. In most cases, such as for heavy component supports, we conservatively assumed that support failure always resulted in DEGB (in other words, the conditional probability of break equals one), although evidence exists suggesting that the pipe could experience extensive plastic deformation without necessarily breaking.

After multiplying each component fragility by the appropriate conditional probability of DEGB, the resultant modified fragilities were combined into a single "plant fragility" describing the probability that any component failure resulting in DEGB will occur for a given peak ground acceleration. We then convolved this result with the "seismic hazard" to yield the non-conditional probability of indirect DEGB.

4.3 Seismic Hazard

Seismic hazard is defined as the probability that an earthquake will occur causing different levels of peak ground acceleration. This is usually decribed by a set of seismic hazard curves plotting exceedance probability as a function of peak ground acceleration. These curves result from seismic hazard analyses which take into account the earthquake history of the region, zones of potential future earthquakes, and the attenuation characteristics of the regional geology to assess the ground motion hazard at a reactor site.

As part of our generic study of Westinghouse plants, we developed generic seismic hazard curves (Fig. 7) characteristic for all sites located east of the Rocky Mountains, which we also used for CE plants located in the same geographical region. We based these generic curves on six eastern and midwestern sites for which formal seismic hazard analyses had been performed. Details of how these curves were developed are provided in the final report on our Westinghouse evaluation.[4]

4.4 Discussion of Results

Probability of Indirect DEGB

Our evaluation of Combustion Engineering plants indicated that the probability of an indirect DEGB in reactor coolant loop piping is very low (Table 8). This general result is consistent with that of our evaluation of Westinghouse plants; in fact, the probabilities of indirect DEGB in CE reactor coolant loop piping are typically lower than for the Westinghouse plants.

For the earlier vintage (Group A) plants, the best-estimate probability of indirect DEGB, estimated using our generic seismic hazard curves for the area east of the Rocky Mountains, varies from 6.6×10^{-8} to 6.4×10^{-6} events per plant-year. Uncertainty analyses yielded 90th percentile probabilities (approximate upper bound values) between 1.2×10^{-6} to 5.2×10^{-5} events per plant-year. Even for the lowest capacity plant, Palisades, the probability of indirect DEGB is very low. This is particularly meaningful when it is considered that the generic seismic hazard curves are probably too conservative for the Palisades site.

For the more modern (Group C) plants, the best-estimate probability of indirect DEGB, estimated using plant-specific or generic seismic hazard curves as indicated in Table 8, is significantly lower, varying from 3.8×10^{-16} to 1.3×10^{-8} events per plant-year. The 90th percentile values range from 3.2×10^{-14} to 3.0×10^{-7} events per plant-year.

The best-estimate probability of indirect DEGB for the single Group B plant (Fort Calhoun) is 1.6×10^{-6} events per plant-year using the generic seismic hazard curves, with a 90th percentile value of 1.4×10^{-5} events per plant-year.

Effect of Seismic Hazard

In the evaluation of San Onofre Units 2 and 3, two sets of seismic hazard curves were applied. The first set, shown in Fig. 8 and denoted as SONGS Set 1, was based largely on the results of a seismic hazard evaluation performed by New Mexico Engineering Consultants and includes three curves, the upper and lower of which asymptotically approach 0.67g and 1.05g peak ground acceleration (about 1.0 and 1.5 times the SSE, respectively).[7] Because this best- estimate curve set did not include larger earthquakes and might therefore be too optimistic, a sensitivity evaluation was performed in which a set of curves was developed to include earthquakes up to five times the SSE (see Vol. 3 of this report series). The median indirect DEGB probabilities estimated using the second set of curves (denoted as SONGS Set 2) increased by about six orders of magnitude -- from 4.6×10^{-17} to 1.1×10^{-11} events per plant-year -- over those predicted using the first set. Although the probability of indirect DEGB is still very low in either case, the result does indicate that the probability of indirect DEGB is a strong function of seismic hazard. This contrasts with the results of the direct DEGB evaluations, which showed that the probability of DEGB due to crack growth is only weakly affected by earthquakes and is instead dominated by normal operating loads resulting from pressure and restraint of thermal expansion.

Comparison with Westinghouse Results

In our previous evaluation of Westinghouse plants located east of the Rocky Mountains, the best-estimate probability of indirect DEGB was estimated to be 3.3×10^{-6} events per plant-year, with a 90th percentile value of 2.0×10^{-5} events per plant-year. This result was based on the plant having the lowest seismic capacity supports of all the Westinghouse plants considered. Our evaluation of CE plants showed that with the exception of Palisades, all CE plants have lower probabilities of indirect DEGB than the lowest capacity Westinghouse plant. A comparison of reactor coolant loop support capacities indicated the following:

- for the more modern (Group C) plants, the support response factors (which are a measure of the conservatism in design loads) are comparable to those for the lowest capacity Westinghouse plant. However, the capacity factors (i.e., margin against seismic failure) are significantly larger due to such factors as different support arrangement (CE supports are tied together and to the structure at more locations) and different design stress allowables.

- for the older (Group A) plants, the response factors are typically larger than for the lowest capacity Westinghouse plant, reflecting large conservatisms in early response analysis techniques. However, the capacity factors are generally smaller. Therefore, the earlier response calculations were in general more conservative and the equipment support design (for a given load) less conservative than for later plants.

The net result in both cases is that the probability of indirect DEGB in CE plants is generally lower than that of the lowest capacity Westinghouse plant.

4.5 Design and Construction Errors

Our 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, we could treat design and construction errors probabilistically in the same way that we treat any other parameter if a distribution of errors could be established. However, since NSSS heavy component support failures are hard to find, developing a suitable distribution may not be possible. Therefore, during our Westinghouse study we performed a limited sensitivity study to determine what degree of error would be required to significantly change the probability of indirect DEGB.

In this study, we first identified plausible construction errors and estimated the corresponding reduction in the capacity of critical equipment. We then recomputed the indirect DEGB probability for Zion to determine the resultant effect on the probability of indirect DEGB. The specific errors that we considered included:

- bad workmanship in, improper material selection for, or improper installation of anchor bolts used for steam generator, RPV, and reactor coolant pump supports.
- improper installation or maintenance of steam generator support snubbers.

The sensitivity studies that we performed indicated that only extremely large construction errors could significantly increase the probability of indirect DEGB (see Fig. 9).

Although we do not represent that we can resolve the important question of design and construction errors through such a limited study alone, its results suggest that only very serious errors -- errors that would presumeably be detected by the stringent quality control procedures applied to reactor coolant piping -- could change our conclusion that indirect DEGB is a very unlikely event. Our review of CE quality control procedures (see Volume 3, Appendix A) leads us to conclude that such errors should not be a problem for reactor coolant loop piping.

Volume 3 of this series provides a more complete discussion of our sensitivity studies, including details on Combustion Engineering quality assurance and quality control procedures for reactor coolant systems.

TABLE 7

Parameters Considered in Developing Component Fragilities

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

TABLE 8

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 8 (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 used in evaluation

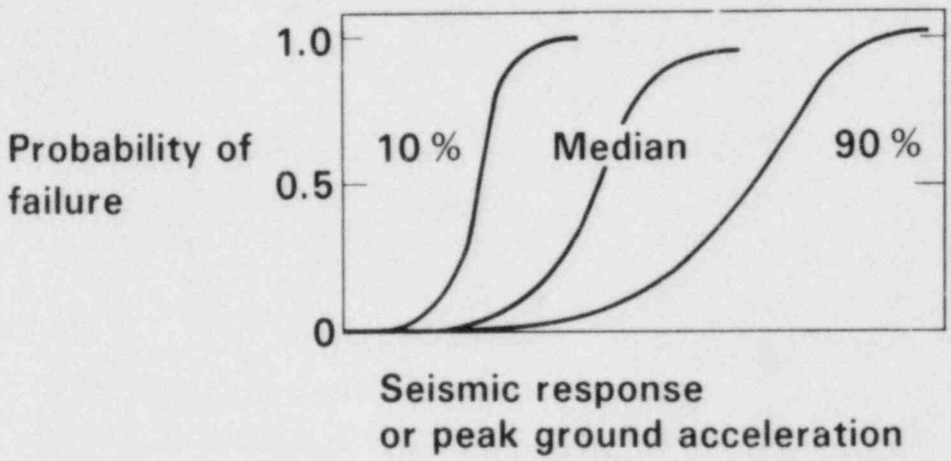

Figure 6. Typical curve set representing structural or equipment fragility.

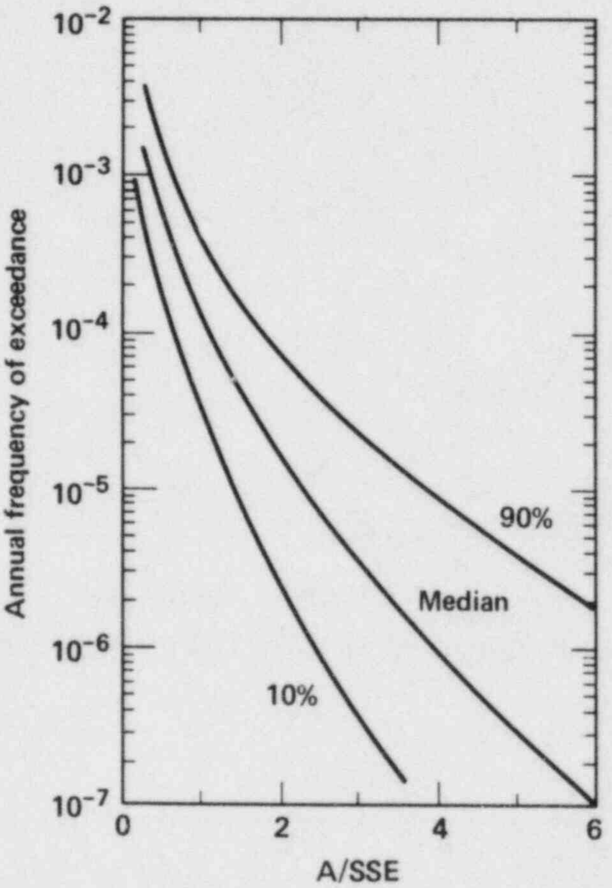

Figure 7. Generic seismic hazard curves used for estimating probability of indirect DEGB in plants located east of the Rocky Mountains.

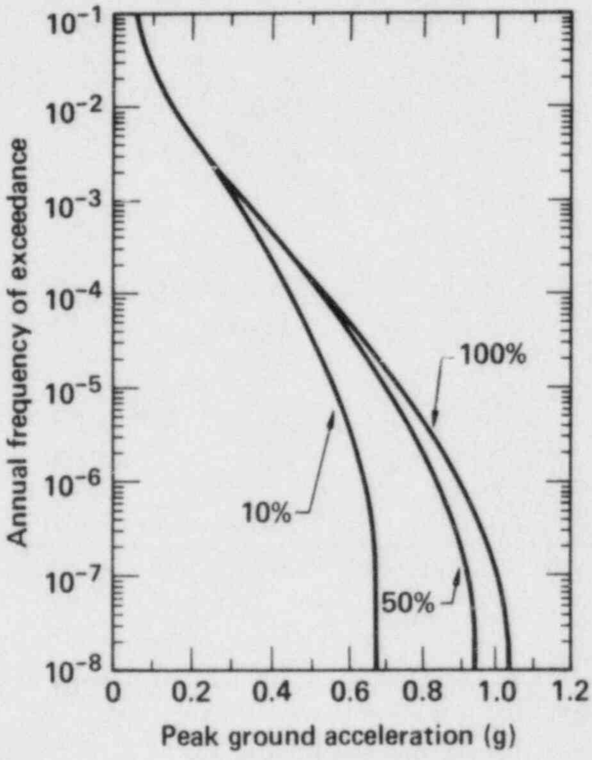

Figure 8. Site-specific seismic hazard curves used for the San Onofre nuclear power plant (from Ref. 7).

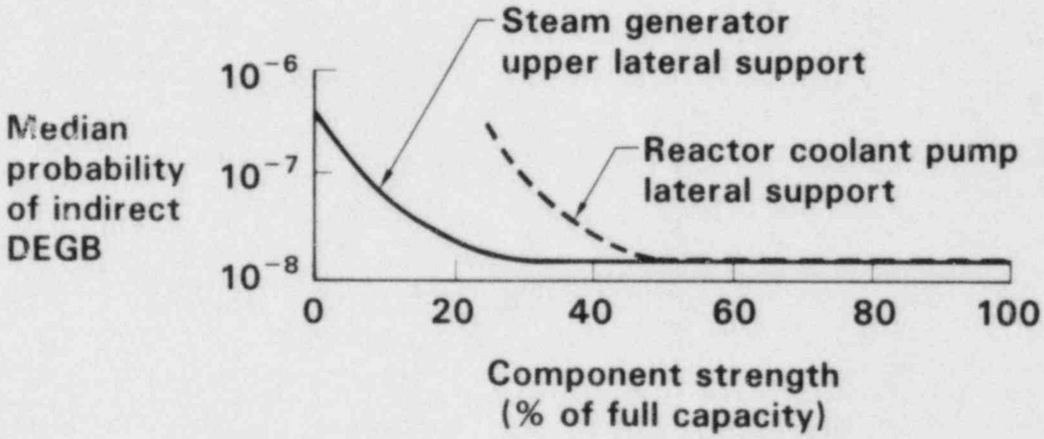

Figure 9. Typical effect of support capacity on probability of indirect DEGB (from Ref. 3, Vol. 3).

5. SUMMARY AND CONCLUSIONS

5.1 Probability of Direct DEGB in Reactor Coolant Loop Piping

We have completed probabilistic analyses indicating that the probability of direct DEGB in reactor coolant piping is very small for Combustion Engineering PWR plants both east and west 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 particular, the results of our evaluations indicate that:

- the "best estimate" probability of direct DEGB ranges from 5.5×10^{-14} to 4.5×10^{-13} events per plant year.

- the "best estimate" probability of leak (through-wall crack) ranges from 1.5×10^{-8} to 2.3×10^{-8} events per plant year. The significantly lower probability of DEGB compared to leak suggests that "leak before break" is a valid concept for CE reactor coolant loop piping.

Based on our uncertainty analyses, a probability of 10^{-10} events per plant-year appears reasonable as an approximate upper bound on the probability of direct DEGB in CE reactor coolant loop piping. The upper bound on leak probability is about 2×10^{-7} events per plant-year.

5.2 Probability of Indirect DEGB in Reactor Coolant Loop Piping

We have completed probabilistic analyses for Combustion Engineering plants 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 indicated for all plants 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.

- the best-estimate probability of indirect DEGB is about 10^{-6} events per plant year for older plants, and less than 10^{-8} events per plant year for newer plants.

- only gross design and construction errors of implausible magnitude could substantially increase the probability of indirect DEGB beyond the values predicted.

- the probability of indirect DEGB is a strong function of seismic hazard. A sensitivity study performed for the San Onofre plant, in which we used two different sets of seismic hazard curves, showed a several order of magnitude difference in indirect DEGB probability depending on how we treated earthquakes significantly larger than the SSE. This contrasts with the results of our evaluations of direct DEGB probability, which was shown to be only weakly affected by earthquakes.

The probability of DEGB due to crack growth at welded joints is five orders of magnitude or more 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.

5.3 Effect of Earthquakes on DEGB Probabilities

Our analyses have shown that the probability of direct DEGB is only very weakly affected by an earthquake. In evaluating the probability of direct DEGB, we considered three events in which failure occurs in reactor coolant loop piping:

- failure occurs simultaneously with the first earthquake occurring during plant life.
- failure occurs prior to the first earthquake occurring during plant life.
- failure occurs with no earthquake occurring during plant life.

Probabilities of direct DEGB were calculated independently for each event and then combined into an overall probability that pipe failure occurs sometime during plant life. It was found for both leak and DEGB that the probability of the first event -- simultaneous occurence of failure and an earthquake -- was one to three orders of magnitude less than that of failure occurring independently of an earthquake. This result indicates that direct DEGB and a safe shutdown earthquake can be considered independent random events, and that the probability of their simultaneous occurence during plant life is negligibly low.

We have identified earthquake as the only credible cause of indirect DEGB; the probability of indirect DEGB therefore also expresses the probability that DEGB and an earthquake simultaneously occur. For the lowest capacity CE plant (Palisades), the 90th percentile probability is 5.2×10^{-5} events per reactor year. Therefore, 5×10^{-5} events per plant-year appears to be a reasonable upper bound generically applicable to older CE plants, compared to an upper bound value of 3×10^{-7} for newer plants. Not surprisingly, we found that seismic hazard had a significant effect on the estimated probability of indirect DEGB.

In developing the indirect DEGB results, we conservatively assumed that failure of any critical support unconditionally led to DEGB. In other words, no credit was taken for large inelastic deformation of the pipe that might occur resulting in only partial break or no break at all. Furthermore, the wide spread of uncertainty in the generic seismic hazard curves, combined with the assumption of a 0.15g minimum SSE, is expected to cover all sites in the eastern and midwestern U.S. Using the generic curves in lieu of site-specific seismic hazard information may be overly conservative for certain sites; we believe, for example, that this may be true for Palisades. In those instances where site-specific seismic hazard curves were actually used for an individual plant, the estimated probability of indirect DEGB was generally lower than when the generic curves were used for that plant.

5.4 Reliability of Heavy Component Supports

If the probability of DEGB is determined to be acceptably low, then the current regulatory requirement that SSE and pipe rupture loads be combined in the design of reactor coolant loop piping could be eliminated. Given that future reactors may not be designed for this load combination, a question may arise concerning the reliability of heavy component supports.

Interestingly, the results of our indirect DEGB evaluation imply that the reliability of heavy component supports is as much a function of the particular analysis techniques used in plant design as it is of load combination. In our study of eastern and midwestern Westinghouse plants, we selected two "lower bound" (lowest seismic capacity) plants for detailed evaluation of component seismic fragilities. For one of these plants, an older plant not designed for the SSE and DEGB load combination, we actually predicted a slightly lower best-estimate probability of DEGB than we did for the more modern plant that had been designed for both SSE and DEGB loads (2.4×10^{-6} compared to 3.3×10^{-6} events per plant year, respectively). The older plant had high seismic margins because of relatively conservative analytical techniques used in its design (three-dimensional uncoupled response spectrum analysis). The newer plant, on the other hand, was designed using more sophisticated analytical techniques (three-dimensional coupled time-history response analysis). Although this plant was designed for combined SSE and DEGB loads, reduced conservatism in the analysis methods used yielded a DEGB probability similar to that of the older plant.

The lesser degree of refinement in the design methods for the older plant was, not surprisingly, evidenced by a somewhat larger uncertainty in its DEGB probability.

It can be argued that eliminating the requirement to combine SSE and DEGB loads in the design of component supports will result in "less conservative" support designs. Load definition is certainly one way of introducing conservatism into an analysis. However, many other factors also contribute to the degree of conservatism in a component design, including:

- the particular analytic techniques used to predict component response, such as two- or three-dimensional analysis, time-history or response spectrum analysis, coupled or uncoupled analysis, and the various combinations thereof.
- input data, that is, selection of parameters such as damping values.
- application of safety factors to calculated results to "insure" conservatism.

Just what constitutes a "conservative" analysis is therefore open to discussion. We can, for example, perform best-estimate calculations, using state-of-the-art modeling and realistic response characteristics (damping, for example) to determine response to conservative design-basis loads. Or we can use less sophisticated analysis techniques, and introduce conservatism through the input parameters (again, such as damping) that we select. The example previously discussed illustrates a case where two different approaches to component design yield predicted reliabilities that are remarkably similar.

From this comparison we can conclude that component support reliability should not be judged solely on the basis of whether or not SSE and DEGB loads are combined. Instead, support reliability should be evaluted in terms of adequate margin against failure, with the definition of "adequate" taking into consideration a wide range of parameters as was done in developing component fragilities for our indirect DEGB evaluation. As was discussed earlier, probabilistic analysis techniques are particularly well-suited for this purpose.

5.5 Combination of Seismic and LOCA Effects

As we noted in Section 1.1, postulation of pipe break can affect many aspects of plant design. Because a loss of coolant accident could have long-term as well as short-term effects, we may not necessarily be able to decouple all seismic and LOCA effects even though the events themselves may not occur simultaneously. For example, in its specifications for environmental qualification of mechanical and electrical equipment, Kraftwerk Union (KWU) divides a LOCA in containment into three time regimes:

- a short-term regime (0 to 3 hours after break), in which peak pressure and temperature are reached approximately 10 sec after break, affecting structures as well as those components that would be required either at the time of or immediately following a pipe break.

- an intermediate-term regime (3 to 24 hours after break), which addresses equipment that would be required during the initial recovery phase following a LOCA.

a long-term regime (over 24 hours after break), addressing in particular corrosion effects on components either required indefinitely or that would be restarted after extended shutdown for later plant reactivation. The maximum period of interest is defined on a component-specific basis, but is generally on the order of several months to a year.

The short-term regime includes the most dynamic effects associated with a LOCA -- pipe whip, jet impingement, decompression waves -- which would result in the most severe LOCA loads. If DEGB were eliminated as a design basis event, then pipe whip could be similarly eliminated, as without a double-ended break the pipe would retain geometric integrity.

Experimental research, in particular full-scale blowdown testing at the HDR facility in West Germany, has shown that loads due to jet impingement and decompression waves in effect coincide with the blowdown event.[7] Therefore, if DEGB and earthquake can be considered as independent random events, loads associated with jet impingement and decompression waves could likewise be decoupled from seismic loads.

This may not be the case, however, for other LOCA effects acting over longer or later time periods. Testing at HDR has shown that containment pressure and temperature peak during blowdown, then fall to lower, albeit still elevated, quasi-steady values that can persist for several hours after blowdown. Although pressures throughout the containment tend to be fairly uniformly distributed, thermal convection causes long-term temperatures in the upper containment to be generally higher than at lower levels. The resultant temperature gradients have been found to produce non-trivial global thermal stresses in the HDR steel containment. The HDR experience has been that the fictive pressure derived from pressure and thermal stresses is lower than the containment design pressure. Nevertheless, for commerical plants having steel containments, it might not be unreasonable to combine pressure and thermal loads with seismic loads in evaluating containment response, if an earthquake were postulated to occur shortly -- say within 24 hours -- after blowdown.

In addition to the magnitude of seismic loads, the deciding factors here would be (1) magnitude and duration of the post-LOCA temperature and pressure in containment, which would depend on break characteristics, and (2) the probability that an earthquake occurs during the time period of interest. According to our generic hazard curves for the eastern and midwestern U.S., the median probability of an earthquake larger than one SSE occurring within any given 24-hour period is about 4.1×10^{-7}, with an upper bound of about 1.4×10^{-6}.

Assuming that the probability of a double-ended break is judged to be sufficiently low so that we can regard DEGB and earthquakes as independent random events, we can draw the following conclusions regarding coupling of seismic and LOCA effects:

- eliminating DEGB as a design basis event would allow pipe whip to be disregarded altogether.

- the most highly dynamic LOCA effects -- jet impingement and decompression waves -- coincide with the blowdown event; therefore, the resultant loads could be decoupled from seismic loads.

- longer-term LOCA effects, such as containment stresses resulting from elevated pressures and temperatures following blowdown, would possibly need to be considered in combination with seismic loads.

The results of our investigation indicate that a decoupling of DEGB and SSE, and with it modification of related design criteria, is warranted for CE reactor coolant loop piping. We recommend however that the strength of component supports, currently designed for the combination of SSE plus DEGB, not be reduced. This recommendation is based on our finding that seismically induced support failure is the weak link in the DEGB evaluation. 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.

5.6 Replacement Criteria

The results of our evaluation of CE and Westinghouse reactor coolant loop piping have shown that a seismically induced DEGB is very unlikely. Therefore, SSE and DEGB can be considered independent random events whose probability of simultaneous occurence is negligibly low, and the design requirement that DEGB and SSE loads be combined should be removed. Our study further indicates that the probability of DEGB in reactor coolant loop piping is sufficiently low under all 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 as a design basis event would not, of course, remove the need to design for the effects of a postulated pipe break. What would change is the basis for plant design against a LOCA. As a result, a suitable replacement for reactor coolant loop DEGB would have to be identified to address various aspects of plant design, including, but not necessarily limited to:

- whipping of broken pipe ends and the need for pipe whip restraints.
- containment pressurization resulting from pipe break, which affects the volume and overall design of the containment structure.
- coolant discharge rate, which in turn sets the minimum make-up capacity of emergency core cooling systems.

- external loads on the reactor vessel and loads on RPV internals resulting from decompression waves.

- jet impingement loads on structures and equipment in the immediate break vicinity.

- reaction loads at support locations.

- global environmental effects -- pressure, temperature, humidity -- affecting the performance of mechanical and electrical equipment important to safety.

- local environmental effects affecting equipment performance.

Except for pipe whip, which could be disregarded altogether, elimination of reactor coolant loop DEGB as a design basis would require that suitable replacement criteria be developed to address these aspects of plant (and not piping) design.

One approach to replacing DEGB, implemented by West Germany in the Guidelines for Pressurized Water Reactors set by its Reactor Safety Commission (RSK), postulates a reduced break in reactor coolant loop piping.[8] For LOCA issues associated specifically with the reactor coolant loops, the RSK guidelines define a replacement pipe break with a flow area 10% that of the affected piping and a break opening time of 15 ms. The postulated reduction in break flow reduces blowdown loads on reactor pressure vessel internals, reaction loads on pipe and component supports, jet impingement loads, and eliminates pipe whip entirely. However, the RSK guidelines retain DEGB as a basis for areas affecting <u>overall</u> plant design: discharge capacity of emergency core cooling systems, containment design pressures, and environmental conditions influencing the performance of safety-related mechanical and electrical equipment.

Although practical to apply in a regulatory sense, the RSK approach is inherently inconsistent, a fact recognized by its authors but accepted for regulatory convenience. This inconsistency is particularly evident in the dual manner in which the DEGB criterion is applied, but is unavoidable if a reactor coolant loop break is to remain the design basis event. For example, if reactor coolant loop DEGB were totally eliminated in favor of a 10% break, then main steam line DEGB would most likely become the governing design basis event for plant design (in particular, containment sizing) due to its greater severity compared to the reduced reactor coolant loop break.

It is clear that replacement criteria for plant design must go beyond simply defining an alternative break size for reactor coolant loop piping. In the development of comprehensive replacement criteria, two factors will require consideration:

- the failure type (i.e., DEGB, partial break, leak) postulated for each piping system whose failure would have a potentially significant impact on overall plant safety, and

- assuming that a failure occurs, what the relative effect of each system failure on overall plant safety is.

Once prescribed, a given type (and size) of failure would have associated with it a probability of occurrence that could, in principle, be evaluated in a manner similar to that used to evaluate the DEGB probabilities discussed in this report. This result would then provide input to a probabilistic risk assessment from which the contribution to overall plant safety could be determined.

Two piping systems are presently of greatest interest as bases for PWR plant design: reactor coolant loops and main steam lines. If reactor coolant loop DEGB were eliminated as a design basis event and not replaced by an alternate break, then main steam line DEGB would most likely become the governing design basis event for plant design. If a reactor coolant loop break of reduced size -- defined by as yet unspecified criteria -- were postulated instead, the effect of this break on plant design would have to be compared against that of the main steam line break to determine which would become the governing design basis event.

In the near term, evaluations such as the one presented in this report provide NRC with a technical basis for reviewing specific piping systems on a case-by-case basis. The results of the present study are applicable to reactor coolant loop piping; a similar evaluation of recirculation, main steam, and feedwater piping in Mark I BWR plants is in progress. Equivalent results could be obtained for other key systems such as surge lines and other piping connected to the reactor coolant pressure boundary, and PWR main steam lines.

Any NRC rulemaking action defining general replacement criteria, however, will have to be based on a more comprehensive approach integrating many technical disciplines and addressing various elements in plant design. In our opinion, general replacement criteria can only developed after the following four-step assessment is performed:

(1) Determine causes of pipe failure in order to assess the likelihood of a pipe break.

(2) Establish the break size and its potential effects on the various aspects of plant design.

(3) Define an acceptable level of safety requirement.

(4) Define criteria for regulating the postulation of pipe break.

Such an approach would be a very powerful one, in that the criteria themselves would have considered the effect of various break sizes on plant design. It is clear, however, that the such replacement criteria will require careful development and objective review to assure their intended generic applicability.

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NRC FORM 335 (7 77) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1 REPORT NUMBER (Assigned by DDC)
NUREG/CR-3663, Vol. 1
UCRL-53500, Vol. 1

4 TITLE AND SUBTITLE (Add Volume No., if appropriate)
Probability of Pipe Failure in Reactor Coolant Loops of Combustion Engineering PWR Plants
Vol. 1: Summary Report

2 (Leave blank)

3 RECIPIENT'S ACCESSION NO.

7 AUTHOR(S)
G. S. Holman, T. Lo and C. K. Chou

5 DATE REPORT COMPLETED
MONTH October YEAR 1984

9 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Lawrence Livermore National Laboratory
Post Office Box 808, L-46
Livermore, California 94550

DATE REPORT ISSUED
MONTH January YEAR 1985

6 (Leave blank)

8 (Leave blank)

12 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

10 PROJECT/TASK/WORK UNIT NO

11 CONTRACT NO
A0133

13 TYPE OF REPORT
Technical

PERIOD COVERED (Inclusive dates)

15 SUPPLEMENTARY NOTES

14 (Leave blank)

16 ABSTRACT (200 words or less)

As part of its reevaluation of the double-ended guillotine break (DEGB) as a design requirement for reactor coolant piping, the U.S. Nuclear Regulatory Commission (NRC) contracted with the Lawrence Livermore National Laboratory (LLNL) to estimate the probability of occurrence of a DEGB, and to assess the effect that earthquakes have on DEGB probability. This report describes a probabilistic evaluation of reactor coolant loop piping in PWR plants having nuclear steam supply systems designed by Combustion Engineering. Two causes of pipe break were considered: pipe fracture due to the growth of cracks at welded joints ("direct" DEGB), and pipe rupture indirectly caused by failure of component supports due to an earthquake ("indirect" DEGB). The probability of direct DEGB was estimated using a probabilistic fracture mechanics model. The probability of indirect DEGB was estimated by estimating support fragility and then convolving fragility with seismic hazard. The results of this study indicate that the probability of a DEGB from either cause is very low for reactor coolant loop piping in these plants, and that NRC should therefore consider eliminating DEGB as a design basis in favor of more realistic criteria.

17 KEY WORDS AND DOCUMENT ANALYSIS

17a DESCRIPTORS

double-ended guillotine break (DEGB)

17b IDENTIFIERS/OPEN-ENDED TERMS

18 AVAILABILITY STATEMENT
Unlimited

19 SECURITY CLASS (This report)
Unclassified

20 SECURITY CLASS (This page)

21 NO OF PAGES

22 PRICE
$

NRC FORM 335 (7 77)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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