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Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plant

Volume 1: Summary Report

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ABSTRACT

As part of its reevaluation of the double-ended guillotine break (DEGB) of reactor coolant loop piping as a design basis event for nuclear power plants, 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 Westinghouse. 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 and 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 event in favor of more realistic criteria.

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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), 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 event. 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 small for Westinghouse PWR plants located east of the Rocky Mountains. These analyses calculated the growth of as-fabricated surface flaws at welded joints, taking into account loads on the piping due to normal operating conditions and seismic events. Other factors, such as the capability to detect cracks by non-destructive examination and the capability to detect pipe leaks, were also considered. In particular, the results of our evaluations indicate that:

- the median probability of direct DEGB in reactor coolant loop piping is about 4.4×10^{-12} events per plant-year, with an upper bound of about 1.2×10^{-10} events per plant-year.

- the median probability of leak (through-wall crack) in reactor coolant loop piping is about 1.1×10^{-7} events per plant-year, with an upper bound of about 2.0×10^{-7} events per plant-year.

We estimated the probabilities of leak and direct DEGB for two west coast plants, Trojan and Diablo Canyon, using site-specific seismic hazard information. The results of these evaluations indicated that:

- for Trojan, the estimated median and 90th percentile probabilities of direct DEGB were 2.2×10^{-13} and 1.0×10^{-9} events per plant-year, respectively. The estimated median and 90th percentile probabilities of leak were 5.9×10^{-8} and 1.5×10^{-7}, respectively. These values are comparable to the corresponding generic probabilities of DEGB and leak for plants east of the Rocky Mountains.

- for Diablo Canyon, the estimated median probability of direct DEGB was 2.3×10^{-11} events per plant-year, based on a seismic hazard curve derived from three independent seismc hazard evaluations of the plant site. The estimated median probability of leak was 3.8×10^{-8} events per plant-year.

These evaluations indicated that the probability of an earthquake causing a direct DEGB in reactor coolant loop piping is negligibly small. This result applies both to west coast plants and to plants east of the Rocky Mountains. The sole instance where the simultaneous occurrence of earthquake and DEGB contributed non-negligibly to the probability of direct DEGB was Diablo Canyon, and then only for earthquakes significantly larger than the safe shutdown earthquake. In general, normal operating loads due to pressure and restraint of thermal expansion, and not seismic events, contribute most to pipe failure caused by crack growth.

We have also completed analyses indicating that the probability of indirect DEGB in reactor coolant loop piping is very small for Westinghouse 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 46 Westinghouse plants east of the Rocky Mountains yielded a median probability of indirect DEGB of 1.0×10^{-7} events per plant-year, with a 90th percentile value of 7.0×10^{-6} events per plant-year.

We also estimated the probabilities of indirect DEGB for two west coast plants, San Onofre Unit 1 and Diablo Canyon, using site-specific seismic hazard information consolidated from a variety of independent seismic hazard studies. For Diablo Canyon, the best-estimate probability of indirect DEGB in the reactor coolant loop piping is 1.7×10^{-6} events per plant-year, with a 90th percentile value of 2.2×10^{-5} events per plant-year. For San Onofre, the best-estimate probability of indirect DEGB in the reactor coolant loop piping is 5.4×10^{-8} events per plant-year, with a 90th percentile value of 9.5×10^{-7} events per plant-year. These values are comparable to those for the lowest seismic capacity plants east of the Rocky Mountains.

In general, the results of our evaluation indicate that the probability of DEGB in the reactor coolant loop piping of Westinghouse 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 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.

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

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 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 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 evaluations of DEGB in the reactor coolant loop piping of Westinghouse PWR plants, including the motivation for this research and potential applications of our results. Volume 2 describes in detail our investigation of direct DEGB for plants east of the Rocky Mountains. Volume 3 provides a detailed description of our generic evaluation of indirect DEGB for plants east of the Rocky Mountains, as well as site-specific evaluations for two west coast sites. Volume 4, which can be considered an addendum to Volume 2, documents our evaluation of direct DEGB in west coast 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 event. 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 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.

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 Westinghouse, which are reported herein, as well as for plants having nuclear steam supply systems manufactured by Combustion Engineering.[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 Westinghouse PWR plants. Each reactor coolant loop, of which most Westinghouse plants have four, consists of three sections -- the hot leg, cold leg, and crossover -- connecting the reactor pressure vessel, one steam generator, and one reactor coolant pump. 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 outside diameters of 30 inches or more, and walls that are approximately 2.5 inches thick. Because they are short and stiff, the pipes are supported solely by the major loop components; no additional supports are necessary.

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 earthquake 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.

1.2 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 probabilities of DEGB due to both direct causes (crack growth at welded joints) and indirect causes (failure of supports for heavy loop components) are estimated for reactor coolant loop piping in Westinghouse PWR plants. Included in this evaluation are:

- plants east of the Rocky Mountains (46 plant sites evaluated). These evaluations were based on generic seismic hazard curves for this region.

- plants located on the west coast. Direct DEGB results are presented for Trojan and for Diablo Canyon Units 1 and 2. Indirect DEGB results are presented for San Onofre Unit 1 and for Diablo Canyon Units 1 and 2. Site-specific seismic hazard information was used in these evaluations.

All evaluations included separate analyses to quantify uncertainty in the estimated probabilities of DEGB and to investigate the sensitivity of the results to certain key parameters.

1.4 Probabilistic Approaches to Failure Evaluation

Over the past several years, probabilistic analysis techniques have gained increased acceptance as a method of generating useful technical information on which to base regulatory decisions affecting 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 uncertainty, 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.

In the deterministic approach, uncertainties are usually addressed by making conservative assumptions about the parameters used in the analysis. 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 probability distribution. Thus, variations in strength and stress about their nominal 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 deterministic 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 as variable, 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 failure. Uncertainties due to lace of precise knowledge about each distribution can be carried through the analysis to estimate the uncertainty in the predicted probability of failure.

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 probability of failure.

The distinction between deterministic and probabilistic approaches widens as the number of parameters involved in the calculation increases. The more parameters involved, the more uncertain (and usually more conservative) a deterministic analysis tends to be because uncertainties 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, and will continue to play, an important role in design. The probabilistic approach, on the other hand, is a powerful tool for evaluating the individual and combined effects of factors influencing the behavior of structures, systems, and components, and therefore provides an important technical basis for regulatory decisions related to safety. Thus, rather than one being an alternative for the other, deterministic and probabilistic approaches complement each other for assessing design reliability.

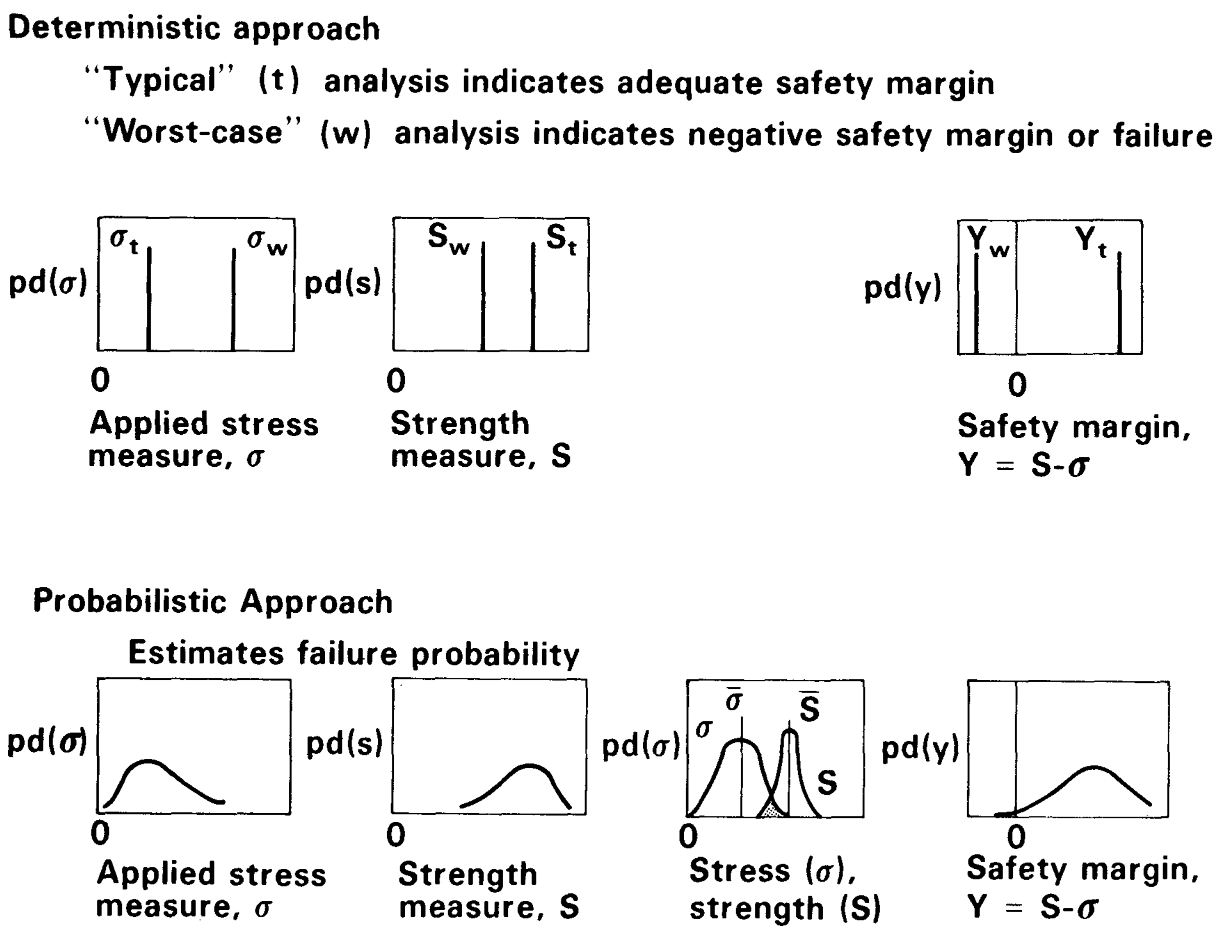

Figure 1. Comparison between probabilistic and deterministic approaches for assessing component adequacy for postulated load conditions. In the probabilistic representation, failure is possible only in the shaded region.

2. GENERAL PLANT DESCRIPTION

2.1 Reactor Coolant Loop Piping

Westinghouse supplies PWR nuclear steam supply systems having two, three, or four loops. The primary components of a typical four-loop Westinghouse NSSS, the most common configuration, are the reactor pressure vessel (RPV), the four steam generators, and the four reactor coolant pumps (Fig. 2). Each reactor coolant loop consists of a hot leg from the RPV to the steam generator, a crossover pipe connecting the steam generator and the coolant pump, and a cold leg between the coolant pump and the RPV. All four loops are identical, except that one is connected to a pressurizer which controls primary system volume. Neither the pressurizer nor the interconnecting "surge line" are included in the present study. Table 1 gives the nominal piping dimensions and operating parameters for each leg in the Zion nuclear power plant. The girth-welded butt joints common to each loop were evaluated for the probability of a directly induced LOCA; the actual number of weld joints may vary slightly from plant to plant.

The primary piping, nozzles, and fittings are fabricated from various grades of cast and wrought Type 316 stainless steel. In this study, no attempt was made to differentiate the mechanical properties in these different components. The ASME Code requirements for the minimum specified room temperature are yield and ultimate strengths of 30 ksi (207 MPa) and 70 ksi (483 MPa), respectively. The code allowable stress at the operating temperature ranges from 11.8 ksi (81 MPa) to 16.8 ksi (116 MPa), depending on whether pipe fittings or nozzle material are considered.

2.2 Reactor Coolant Loop Supports

Figure 3 presents a schematic elevation of the Zion Unit 1 containment, showing the locations of the major reactor coolant loop components. The reactor pressure vessel is supported on four alternate vessel nozzles, each of which has a seat which bears on a shoe. Each steam generator is restrained laterally by an upper support and lower support, and vertically at its lower support feet by four columns which are pinned at both ends. The reactor coolant pump is supported laterally by two structural steel struts and a tension tie rod; the pump is supported vertically by three pin-ended columns.

2.3 Overhead Crane

The overhead crane inside the containment is mounted on wheels which travel over a circular track at the operating floor level. The crane is 66 feet high, and has 3 hooks with lifting capacities of 35 tons, 230 tons, and 460 tons. The Zion crane is an atypical design; most other PWR plants have overhead cranes that travel on a rail mounted on the containment wall near the dome.

Table 1

Primary Piping Dimensions and Operating Parameters
for the Zion Nuclear Power Plant

	Hot Leg	Cold leg	Crossover
Operating Pressure, psi (Mpa)	2235 (15400)	2235 (15400)	2235 (15400)
Temperature, °F (°C)			
Designed	592 (311)	530 (277)	530 (277)
Recorded	588 (309)	540 (282)	540 (282)
Outside diameter, in (cm)	34.0 (86.4)	32.3 (82.0)	36.3 (92.2)
Thickness, in (cm)	2.50 (6.4)	2.38 (6.0)	2.66 (6.8)
Length, in (m)	151 (3.83)	223 (5.66)	97 (2.46)

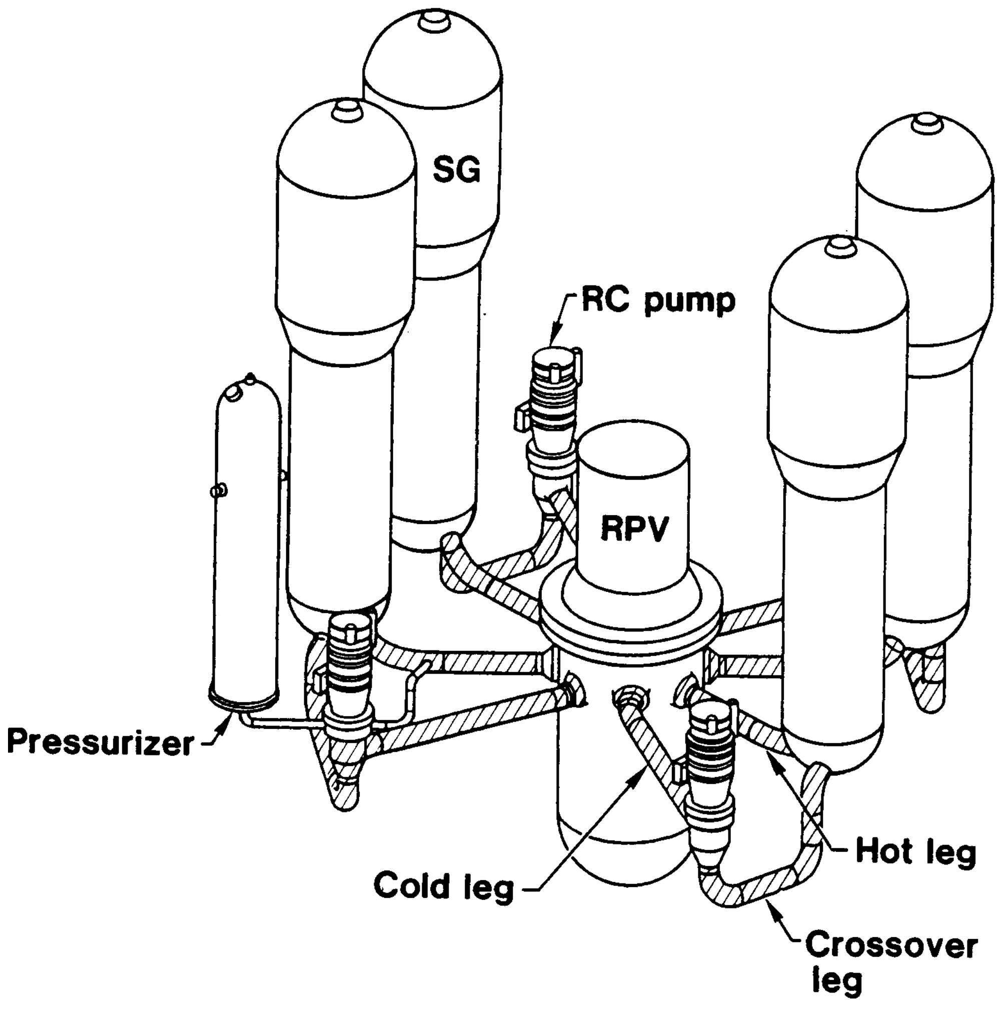

Figure 2. Typical general arrangement of a four-loop Westinghouse PWR nuclear steam supply system.

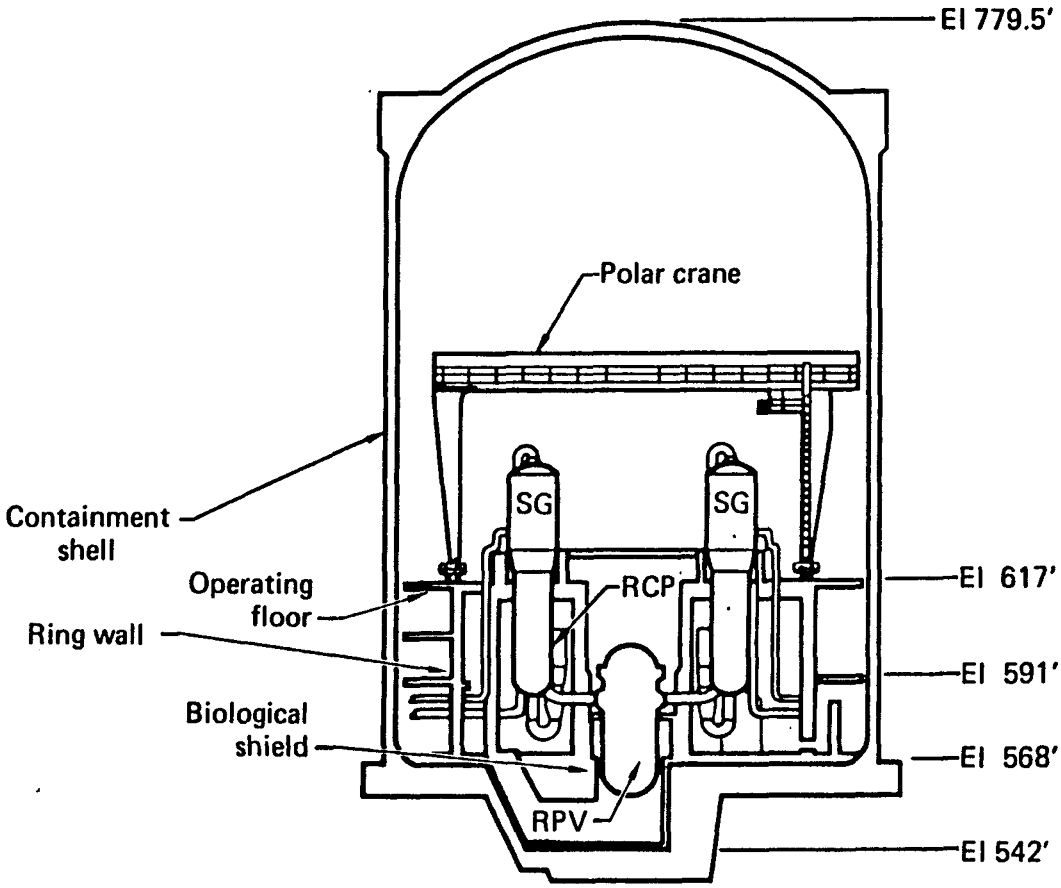

Figure 3. Schematic elevation of the Zion Unit 1 containment.

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 imperfect 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 4 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. 4, 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. All loops of a given Westinghouse 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 exceeding a specified level 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. 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 (Fig. 5). West coast plants were evaluated using site-specific seismic hazard information; the small number of plant sites and widely varying seismic conditions do not allow a generic characterization of seismic hazard to be made without assigning a large degree of uncertainty.

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 (i.e., the earthquake causes failure).

(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 <u>random</u> uncertainties of input parameters contribute to the value of pipe failure probability, they are intrinsic to the analytic process illustrated in Fig. 4. We treated <u>modeling</u> 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 - Plants East of the Rocky Mountains

We began our study of Westinghouse PWR plants with a "pilot" study, using Zion Unit 1 as our pilot plant. The pilot plant provided a basis for developing our fracture mechanics assessment methodology, as well as for conducting extensive sensitivty studies to identify key parameters affecting the probability of DEGB. We also conducted 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 Zion pilot study, we performed a generic evaluation of DEGB probability for other Westinghouse plants, beginning with plants located east of the Rocky Mountains. We first reviewed for each plant the important factors contributing to DEGB probability, and then grouped similar plants together, avoiding the need to perform a separate analysis for each plant. In this study, we performed "best estimate" calculations for each of 17 sample plants (33 plant units), obtaining 17 point estimates of DEGB probability as well as 17 point estimates of leak probability. These point estimates described "best estimate" distributions of DEGB probability (Fig. 6) and leak probability (Fig. 7). The median values (50% confidence limit) of these distributions provide generic point estimates of DEGB and leak probabilities characteristic of all plants east of the Rocky Mountains.

From our results we concluded that the median probability of direct DEGB is very low for the eastern plants -- about 10^{-12} events per plant-year (see

Table 2). Even for the sample plant with the highest probability of failure, the median probability of direct DEGB was still only about 10^{-11} events per plant-year (Table 3). These results were obtained using the generic seismic hazard curves developed as part of our evaluation of indirect DEGB for plants located east of the Rocky Mountains (see Section 4.3).

We also placed distributions on the five parameters that our Zion pilot study indicated most significantly affect the probability of DEGB -- initial crack depth, initial crack length, thermal stresses, seismic stresses, and seismic hazard -- and performed uncertainty analyses to establish confidence bounds on the probability of DEGB. The 90% statistical confidence limit for the sample plant with the highest probability of direct DEGB was less than 10^{-9} events per plant-year.

The median probability of leak was about 1.1×10^{-7} events per plant-year, with a 90th percentile value of 2.4×10^{-7} events per plant-year for the sample plant with the highest probability of direct DEGB. The much higher probability of leak as compared to DEGB suggests that "leak-before-break" is a valid concept for reactor coolant loop piping.

The results of our generic study of eastern Westinghouse plants indicated therefore that the probability of an earthquake causing direct DEGB is negligible. Furthermore, sensitivity analyses made during our Zion pilot study indicated that even when very large earthquakes -- up to five times the intensity of the safe shutdown earthquake -- were assumed, the probability of DEGB increased only slightly over that calculated assuming no earthquake at all.

3.6 Discussion of Results - West Coast Plants

3.6.1 Trojan

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

3.6.2 Diablo Canyon

Our evaluation of the Diablo Canyon nuclear power plant, located in Southern California near San Luis Obispo, was unique in that the simultaneous occurence of earthquake and pipe failure contributed non-neglibibly to the overall probability of direct DEGB, compared to other loads.

As we had done for other Westinghouse plants, we first obtained a best-estimate probability of direct DEGB in the reactor coolant loop piping at Diablo Canyon. We modeled earthquake occurrence by statistically combining seismic hazard curves presented by Cornell[7] which reflected the results of independent seismic hazard evaluations performed for the site by Blume,[8] Trifunac and Anderson,[9] and Ang and Newmark[10] (Fig. 8). Because none of the curves presented considered peak ground accelerations above 1.2g (about 1.5 times the 0.75g safe shutdown earthquake), we extrapolated our curve to five times the SSE assuming a quadratic relationship in log-log space between occurrence rate and PGA. Using this seismic hazard description, we estimated the median probability of direct DEGB to be 2.5×10^{-11} events per plant-year, about one order of magnitude higher than the median probability of DEGB for plants east of the Rocky Mountains. This result mainly reflects the higher frequency and intensity with which earthquakes occur on the west coast compared to the region east of the Rocky Mountains.

A close examination of conditional failure probabilities for Diablo Canyon (Table 5) shows that contrary to our past experience, a point is reached at which the simultaneous occurrence of DEGB and earthquake dominates the total probability of direct DEGB. Our PRAISE results show that this occurs when peak ground acceleration reaches a level between 0.75g and 2.25g, or between one and three times the SSE level for the Diablo Canyon site. This is indicated in Table 5 by the transition from Events 2 and 3 (earthquake and DEGB do not occur simultaneously) to Event 1 (DEGB and earthquake occur simultaneously) as the dominant failure event. A plot of the data in Table 5 (Fig. 9) shows that this transition occurs at about 1.3 times the SSE ground acceleration. Note that the failure probabilities given in Table 5 are estimated over a 40-year plant lifetime, and are conditioned upon (i.e., assume) the occurrence of an earthquake with the level indicated; the effect of earthquake occurrence rate (seismic hazard) is not included.

Table 5 shows that as earthquake level increases, the conditional probability of leak approaches that of DEGB. The physical implication here is that for very large earthquakes the resultant stresses in the pipe become so large that fatigue crack growth is of less importance. Instead, as the ultimate strength of the unflawed pipe is approached, pipe break occurs. The DEGB probability therefore becomes more strongly dependent on earthquakes than was the case in our other evaluations. Note, however, that even at five times the SSE, the conditional DEGB probability is still only 0.33×10^{-5} events during plant life, or less than 10^{-7} events per plant-year, assuming that an earthquake of this intensity occurs.

We performed a limited uncertainty analysis on the estimated direct DEGB probability, and found that the conditional probability of Event 1 (simultaneous occurrence of DEGB and earthquake) is a strong function of the seismic response factor used in the calculation. Recall that this factor is that by which we reduce vendor SSE stresses to account for conservatisms inherent in design procedures. In our analyses, the seismic load factor is a random variable with an estimated distribution. The 50th percentile value of this distribution was used in our original evaluation; we subsequently performed additional sensitivity calculations using the 10th and 90th percentile values on the distribution. As shown in Table 6, using the 90th percentile value (i.e., about 1.28 standard deviations off the mean) caused the conditional probability of failure increases to 2.0×10^{-2} during plant life if an earthquake of five times the SSE occurs.

When interpreting these results and those of the sensitivity evaluations that follow, it is important to keep in mind that five times the SSE at Diablo Canyon is 3.75g, or 25 times the minimum SSE assumed for plants east of the

Rocky Mountains. Furthermore, recall that in our evaluations stresses for earthquakes larger than the SSE are estimated by linearly extrapolating the SSE stresses. The high conditional DEGB probability given above is hardly surprising in light of the very high stresses implied by this conservative assumption coupled with a seismic response factor one-and-a-quarter standard deviations off of the median value.

The conditional probability of leak, also included in Tables 5 and 6, follows the same general trend, but less dramatically. An interesting result in Table 6 is that beyond the transition point, leak and DEGB probabilities approach and eventually equal one another. This result contrasts with earlier findings that leak probability was several orders of magnitude higher than DEGB probability, and implies that "leak before break" would not apply for Diablo Canyon if an earthquake significantly larger than the 0.75g SSE were to occur. Such behavior would be consistent with exceedance of ultimate strength, and not fatigue crack growth, being the cause of failure.

Even though for very large earthquakes the conditional probability of failure can be high, the extremely low probability that such large earthquakes actually occur offsets the high conditional probability, keeping the non-conditional failure probability low. This implies that seismic hazard plays a more significant role in estimating the probability of direct DEGB for Diablo Canyon than it did for other Westinghouse plants that we evaluated. We therefore performed extensive sensitivity calculations to assess the effect that certain seismic hazard assumptions we had made in our original evaluation of Diablo Canyon had on the estimated probability of direct DEGB.

Effect of Seismic Hazard Curve PGA Limit

Our first sensitivity study investigated the effect of limiting the level of peak ground acceleration in the seismic hazard curve to earthquakes less than five times the SSE, in other words, reducing the contribution of very large earthquakes to the probability of failure. We modified our original seismic hazard curve (which was truncated at five times the SSE) to create four new curves which were asymptotically limited to one, two, three, and four times the SSE level (Fig. 10). We then estimated the probability of direct DEGB corresponding to each of the new curves. The results summarized in Table 7 indicate that the non-conditional probability of direct DEGB decreases by about two orders of magnitude -- from 2.5×10^{-11} to 2.0×10^{-13} events per plant-year -- when the original seismic hazard curve is asymptotically limited to one SSE. This decrease reflects the reduced contribution of earthquakes greater than the SSE. The leak probabilities, on the other hand, are essentially unaffected by the upper limit of the seismic hazard curve.

A plot of the data in Table 7 shows that for this particular seismic hazard curve the simultaneous occurrence of earthquake and DEGB dominates the total failure probability when the upper limit of PGA exceeds about two-and-a-half times the SSE level. However, this transition could vary for different representations of seismic hazard for the plant site. We therefore performed additional sensitivity analyses to investigate how the probability of direct DEGB varied for different individual seismic hazard curves.

Results for Individual Seismic Hazard Curves

Diablo Canyon is located in an area of high seismicity, and the precise seismic hazard at the plant site has been, and continues to be, a subject of much controversy. Our original calculations were based on a seismic hazard curve that we derived from three independent -- and substantially different -- site-specific seismic hazard evaluations (Fig. 8). Because our previous experience had shown that the simultaneous occurrence of earthquake and DEGB contributed only negligibly to the overall probability of direct DEGB, we considered this a reasonable representation of seismic hazard despite the differences. However, when the results of our original evaluation indicated the increased importance of earthquake effects, we performed another series of sensitivity calculations in which we estimated the probability of direct DEGB for each of the three consultant curves individually. We applied each curve in two different ways as follows:

- as originally presented, that is, with peak ground acceleration cut off at 1.2g or less (the Blume curve presented by Cornell was limited to 1.1g), depending on the individual curve.

- as extrapolated by Structural Mechanics Associates (SMA) for estimating the probability of indirect DEGB. In these evaluations, each curve was extrapolated log-linearly to five times the SSE peak ground acceleration.

The median DEGB and leak probabilities estimated using each of these curves are given in Tables 8 and 9, respectively, and compared against those obtained using corresponding forms of the LLNL curve. The DEGB probabilities estimated using the original (i.e., unextrapolated) curves range from 2.0×10^{-13} to 3.8×10^{-13} events per plant-year -- less than a factor of two variation -- and bound the value obtained by truncating the original LLNL curve at 1.2g. The DEGB probabilities for the extrapolated curves range from 2.5×10^{-12} to 1.5×10^{-11} events per plant-year, all of which are exceeded by that obtained using the original LLNL curve (2.5×10^{-11} events per plant-year). This bounding effect reflects the higher rates of occurrence for large earthquakes yielded by the LLNL extrapolation scheme. In any case, the variation is no more than one order of magnitude, indicating that the overall probability of failure is relatively insensitive to the particular seismic hazard curve selected from among those used in our evaluation, despite the relatively wide variation among the individual curves and our lack of a firm basis for extrapolating these curves for very large earthquakes.

We recognize that the seismic hazard information upon which we based our Diablo Canyon evaluation is now some eight years old, and furthermore that information for earthquakes significantly larger than the SSE is as good as non-existent. In the intervening period, new relevant strong ground motion prediction information has been generated, and new information about the

Hosgri fault itself has come to light. These issues will be revisited in the near future by a new seismic PRA for Diablo Canyon, which PG&E will perform for NRC.

It is outside our scope to pass judgement on what constitutes the "best" description of seismic hazard for any particular site. Instead, we rely on information generated by recognized seismic experts and then assess the effect that variations in this information have on our probabilistic results. Based on the sensitivity studies that we performed, we are confident that our results provide a reasonable representation of the relationship between direct DEGB and seismic events for the Diablo Canyon plant.

3.6.3 San Onofre

San Onofre Unit 1, operated by Southern California Edison Company, was not included in this evaluation for the following reasons:

- the site and plant parameters considered in estimating the probability of direct DEGB are bounded by those for Diablo Canyon and Trojan; therefore, the probability of direct DEGB should be similar.

- the probability of indirect DEGB in the reactor coolant loop piping of all other Westinghouse plants that we have evaluated has typically been several orders of magnitude higher than that of direct DEGB. This was also true in our evaluations of Combustion Engineering plants, including San Onofre Units 2 and 3. We expect that the same holds true for San Onofre Unit 1.

It is therefore reasonable to expect that the general conclusions drawn for Diablo Canyon and Trojan are applicable to San Onofre Unit 1 as well. The evaluations of San Onofre Units 2 and 3 are detailed in the documentation of our Combustion Engineering study.[4]

TABLE 2

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

(events per plant-year)

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

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

(2) Generic seismic hazard curves for sites east of the Rocky Mountains were used (Fig. 5).

(3) Plant-specific seismic hazard curves used.

TABLE 3

Probabilities of Direct DEGB and Leak for Sample Plant with Highest DEGB Probability (Plants East of the Rocky Mountains)

(events per plant-year)

	Confidence Limit[1]		
	10%	50%	90%
DEGB	$< 1.0 \times 10^{-15}$	1.0×10^{-11}	7.5×10^{-11}
Leak	1.4×10^{-8}	6.0×10^{-8}	2.4×10^{-7}

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

(2) Generic seismic hazard curves for sites east of the Rocky Mountains were used (Fig. 5).

TABLE 4

Effect of Earthquakes on Best-Estimate Probabilities of DEGB and Leak for Sample Plant with Highest DEGB Probability(1) (Plants East of the Rocky Mountains)

	Event (2)			P[PF](3)
	Event 1	Event 2	Event 3	
DEGB	2.1×10^{-12}	1.3×10^{-10}	1.2×10^{-10}	2.5×10^{-10}
Leak	1.4×10^{-9}	2.4×10^{-6}	2.3×10^{-6}	4.7×10^{-6}

(1) Probability of failure (DEGB or leak) during 40-year plant life. Generic seismic hazard curves (Fig. 5) used in evaluation.

(2) Event 1: Probability of failure coincident with first earthquake
Event 2: Probability of failure prior to first earthquake
Event 3: Probability of failure with no earthquake

(3) P[PF]: Total probability of pipe failure

TABLE 5

Effect of Seismic PGA Level on the Median Probability of Failure[1] in Reactor Coolant Loop Piping at Diablo Canyon

Seismic PGA Level	Event[2]			P[PF][3]
	Event 1	Event 2	Event 3	
DEGB				
1 x OBE	0.25×10^{-12}	0.59×10^{-12}	0.10×10^{-10}	0.11×10^{-10}
1 x SSE (0.75g)	0.23×10^{-11}	0.59×10^{-12}	0.10×10^{-10}	0.12×10^{-10}
3 x SSE (2.25g)	0.21×10^{-8}	0.59×10^{-12}	0.10×10^{-10}	0.21×10^{-8}
5 x SSE (3.75g)	0.33×10^{-5}	0.59×10^{-12}	0.10×10^{-10}	0.33×10^{-5}
Leak				
1 x OBE	0.35×10^{-8}	0.91×10^{-7}	0.20×10^{-5}	0.20×10^{-5}
1 x SSE (0.75g)	0.84×10^{-8}	0.91×10^{-7}	0.20×10^{-5}	0.21×10^{-5}
3 x SSE (2.25g)	0.78×10^{-7}	0.91×10^{-7}	0.20×10^{-5}	0.22×10^{-5}
5 x SSE (3.75g)	0.35×10^{-5}	0.91×10^{-7}	0.20×10^{-5}	0.56×10^{-5}

(1) Conditional probability of failure (DEGB or leak) during 40-year plant life, assuming occurrence of an earthquake with the indicated peak ground acceleration. Seismic hazard is not included.

(2) Event 1: Probability of failure coincident with first earthquake
Event 2: Probability of failure prior to first earthquake
Event 3: Probability of failure with no earthquake

(3) P[PF]: Total probability of pipe failure

TABLE 6

Sensitivity of Event 1 Failure Probability to Variations in Seismic Load for Diablo Canyon Reactor Coolant Loop Piping

Seismic PGA Level	Probability Level of Seismic Load Factor[1]		
	10%	50%	90%
	DEGB[2]		
1 x OBE	0.24×10^{-13}	0.25×10^{-12}	0.30×10^{-11}
1 x SSE	0.16×10^{-12}	0.23×10^{-11}	0.30×10^{-10}
3 x SSE	0.34×10^{-10}	0.21×10^{-8}	0.84×10^{-5}
5 x SSE	0.15×10^{-8}	0.33×10^{-5}	0.20×10^{-1}
	Leak[2]		
1 x OBE	0.12×10^{-8}	0.35×10^{-8}	0.11×10^{-7}
1 x SSE	0.30×10^{-8}	0.84×10^{-8}	0.24×10^{-7}
3 x SSE	0.26×10^{-7}	0.78×10^{-7}	0.85×10^{-5}
5 x SSE	0.72×10^{-7}	0.35×10^{-5}	0.20×10^{-1}

(1) Corresponds to the indicated percentile on the estimated distribution of seismic load factor.

(2) Conditional probability that earthquake and failure (leak or DEGB) occur simultaneously during 40-year plant life, assuming occurrence of an earthquake with the indicated peak ground acceleration. Seismic hazard is not included.

TABLE 7

Sensitivity of Median Direct DEGB Probability
to Upper Limit of Seismic Hazard Curve PGA

PGA Upper Limit	Probability of Direct DEGB[1]			
	Event[2]			P[PF][3]
	Event 1	Event 2	Event 3	
1 x SSE (0.75g)	0.42×10^{-12}	0.41×10^{-12}	0.73×10^{-11}	0.81×10^{-11} (2.0×10^{-13})
2 x SSE (1.50g)	0.25×10^{-11}	0.41×10^{-12}	0.73×10^{-11}	0.11×10^{-10} (2.8×10^{-13})
3 x SSE (2.25g)	0.13×10^{-10}	0.41×10^{-12}	0.73×10^{-11}	0.20×10^{-10} (5.0×10^{-13})
4 x SSE (3.00g)	0.90×10^{-10}	0.41×10^{-12}	0.73×10^{-11}	0.98×10^{-10} (2.5×10^{-12})
5 x SSE (3.75g)	0.97×10^{-9}	0.41×10^{-12}	0.73×10^{-10}	0.98×10^{-9} (2.5×10^{-11})

(1) Probability of DEGB during 40-year plant life using modified LLNL seismic hazard curves shown in Fig. 10. Values in parantheses are annual probabilities of DEGB.

(2) Event 1: Probability of failure coincident with first earthquake
Event 2: Probability of failure prior to first earthquake
Event 3: Probability of failure with no earthquake

(3) P[PF]: Total probability of pipe failure

TABLE 8

Median Probabilities of Direct DEGB in Diablo Canyon Reactor Coolant Loop Piping for Various Seismic Hazard Curves[1]

Seismic Hazard Curve	PGA Limit of Seismic Hazard Curve	
	Cut-Off	Extrapolated
LLNL [2]	0.94×10^{-11} (2.4×10^{-13})	0.98×10^{-9} (2.5×10^{-11})
Trifunac and Anderson [3]	0.15×10^{-10} (3.8×10^{-13})	0.60×10^{-9} (1.5×10^{-11})
Newmark and Ang [3]	0.86×10^{-11} (2.2×10^{-13})	0.32×10^{-10} (8.0×10^{-13})
Blume [4]	0.81×10^{-11} (2.0×10^{-13})	0.10×10^{-9} (2.5×10^{-12})

(1) Probability of DEGB during 40-year plant life using indicated seismic hazard curve from Fig. 8. Annual DEGB probability in parantheses.

(2) LLNL curve developed from consultant curves presented by Cornell.[7] Extrapolation by LLNL to 3.75g (five times SSE) assumes quadratic behavior in log-log space.

(3) Original curves as presented by Cornell.[7] Extrapolation by SMA from 1.2g to 3.75g assumes linear behavior in log-normal space.

(4) Original curve presented by Cornell limited to 1.0g. Extrapolation by SMA to 1.2g and 3.75g by SMA assumes linear behavior in log-normal space.

TABLE 9

Median Probabilities of Leak in Diablo Canyon Reactor Coolant Loop Piping for Various Seismic Hazard Curves[1]

Seismic Hazard Curve	PGA Limit of Seismic Hazard Curve	
	Cut-Off	Extrapolated
LLNL [2]	0.15×10^{-5}	0.15×10^{-5}
	(3.8×10^{-8})	(3.8×10^{-8})
Trifunac and Anderson [3]	0.14×10^{-5}	0.14×10^{-5}
	(3.5×10^{-8})	(3.5×10^{-8})
Newmark and Ang [3]	0.15×10^{-5}	0.15×10^{-5}
	(3.5×10^{-8})	(3.5×10^{-8})
Blume [4]	0.15×10^{-5}	0.15×10^{-5}
	(3.5×10^{-8})	(3.5×10^{-8})

(1) Probability of leak during 40-year plant life using indicated seismic hazard curve. Annual leak probability in parantheses.

(2) LLNL curve developed from consultant curves presented by Cornell.[7] Extrapolation by LLNL to 3.75g (five times SSE) assumes quadratic behavior in log-log space.

(3) Original curves as presented by Cornell.[7] Extrapolation by SMA from 1.2g to 3.75g assumes linear behavior in log-normal space.

(4) Original curve presented by Cornell limited to 1.1g. Extrapolation by SMA to 1.2g and 3.75g by SMA assumes linear behavior in log-normal space.

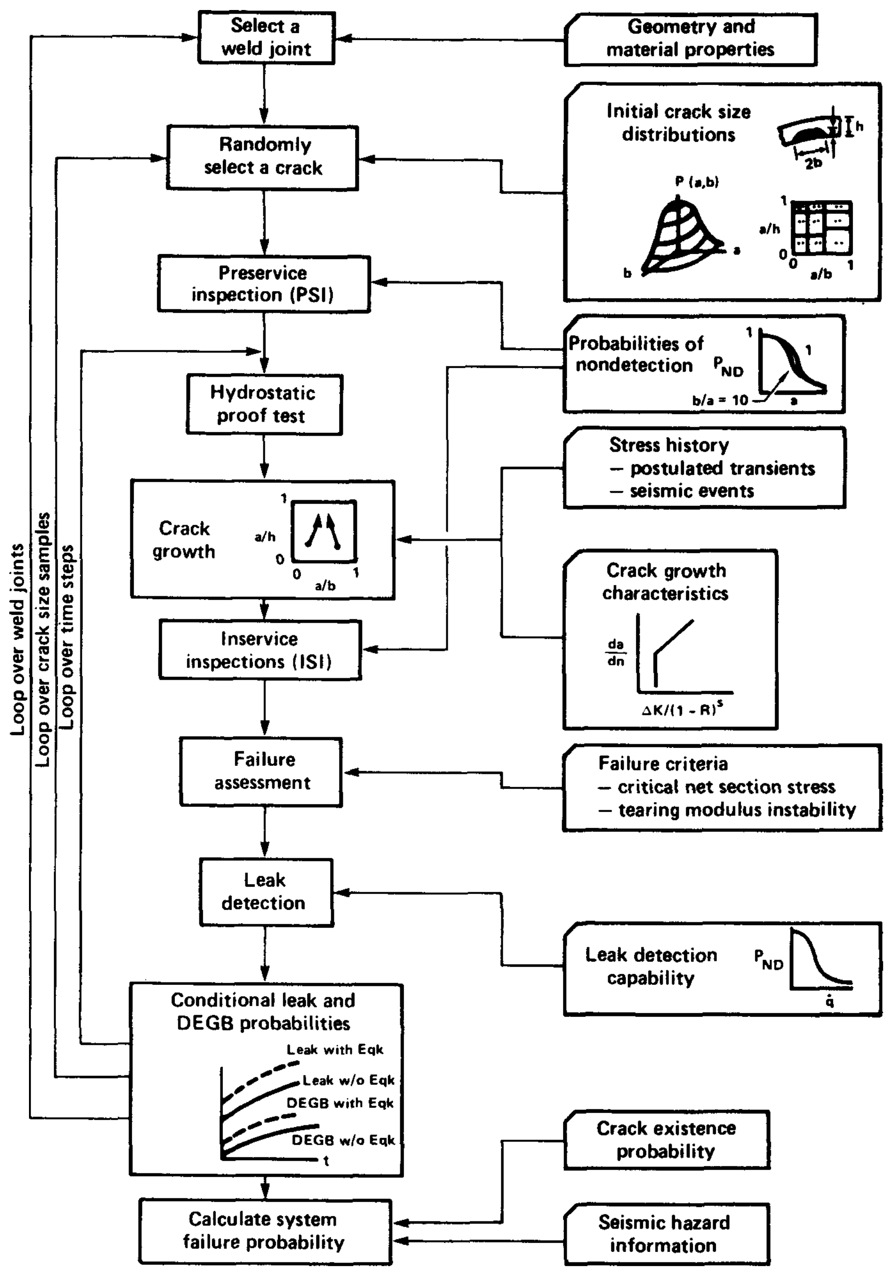

Figure 4. Flowchart of the probabilistic fracture mechanics model implemented in the PRAISE computer code.

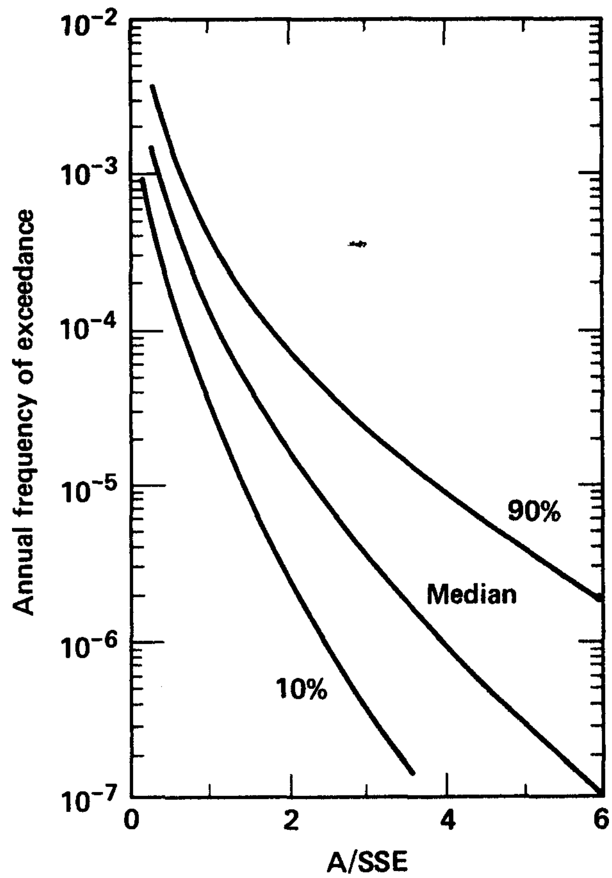

Figure 5. Generic seismic hazard curves used in evaluation of plants east of the Rocky Mountains.

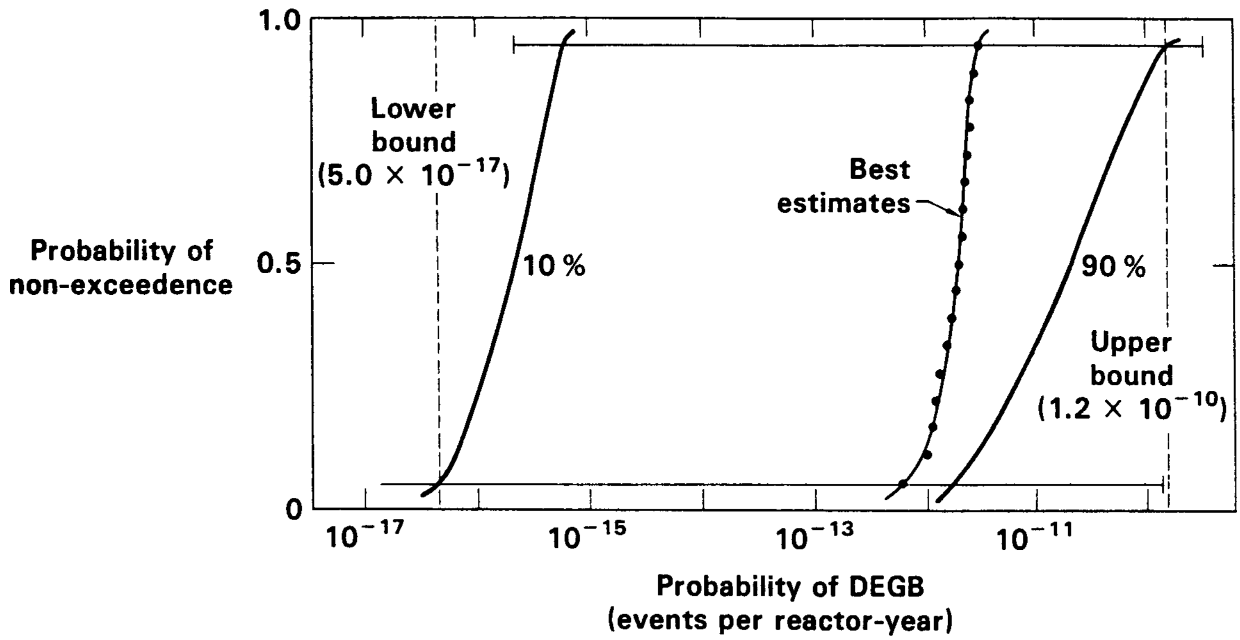

Figure 6. Empirical cumulative distribution of the probability for a direct DEGB in reactor coolant loop piping (plants east of the Rocky Mountains).

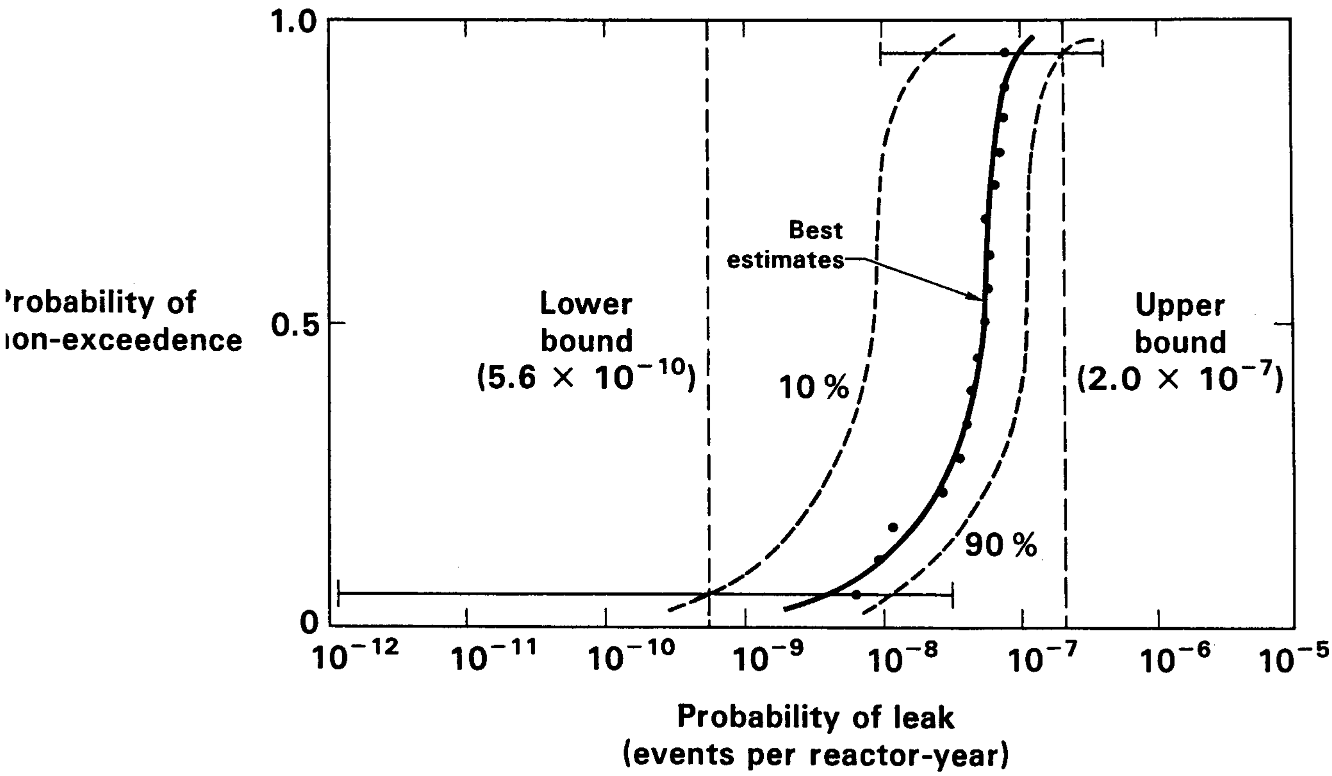

Figure 7. Empirical cumulative distribution of the probability for a leak in reactor coolant loop piping (plants east of the Rocky Mountains).

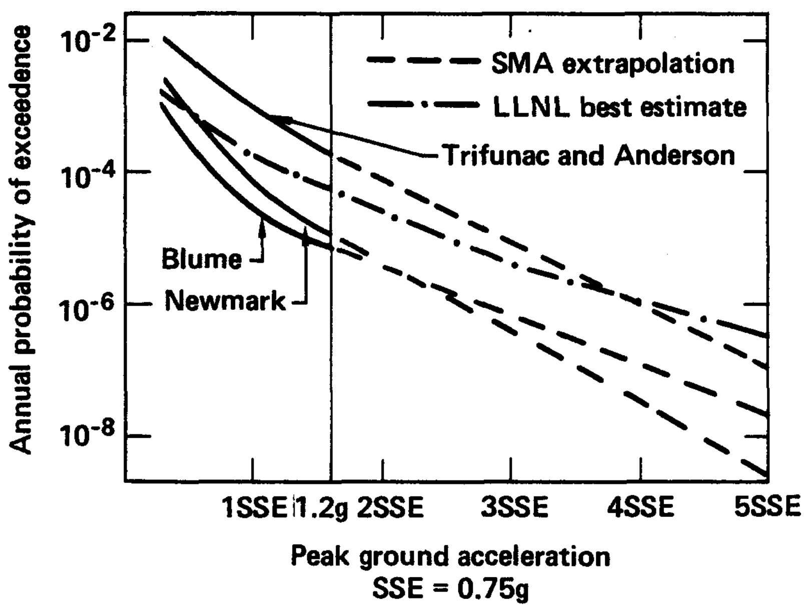

Figure 8. Site-specific seismic hazard curves used to estimate probability of direct DEGB at Diablo Canyon.

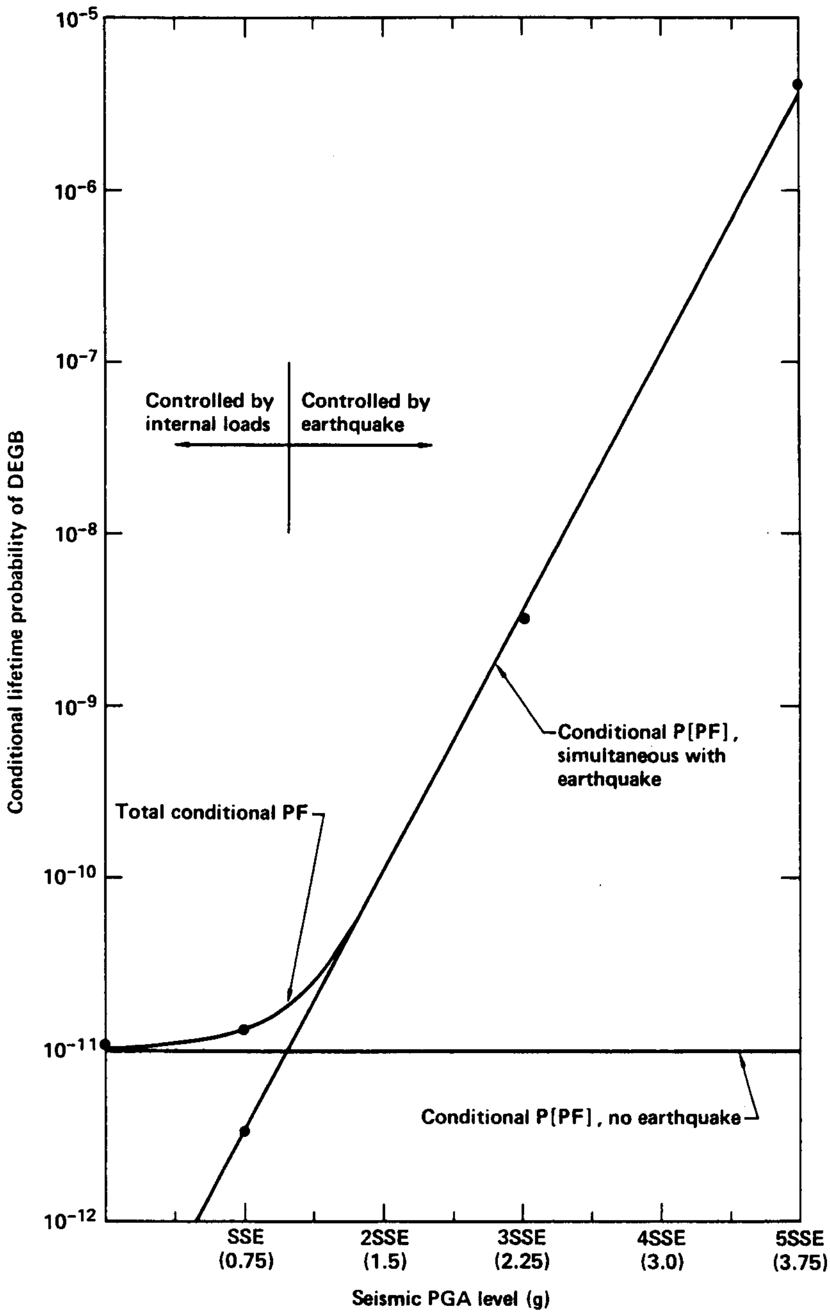

Figure 9. Comparison of conditional DEGB probabilities with and without occurrence of earthquake for Diablo Canyon reactor coolant loop piping.

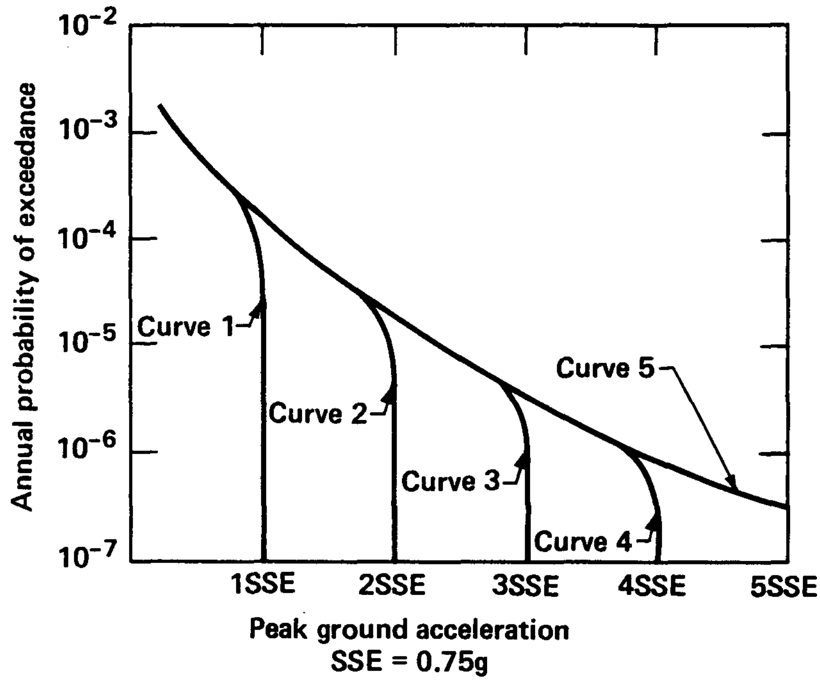

Figure 10. Modified seismic hazard curve for Diablo Canyon for investigating sensitivity of DEGB probability to maximum peak ground acceleration.

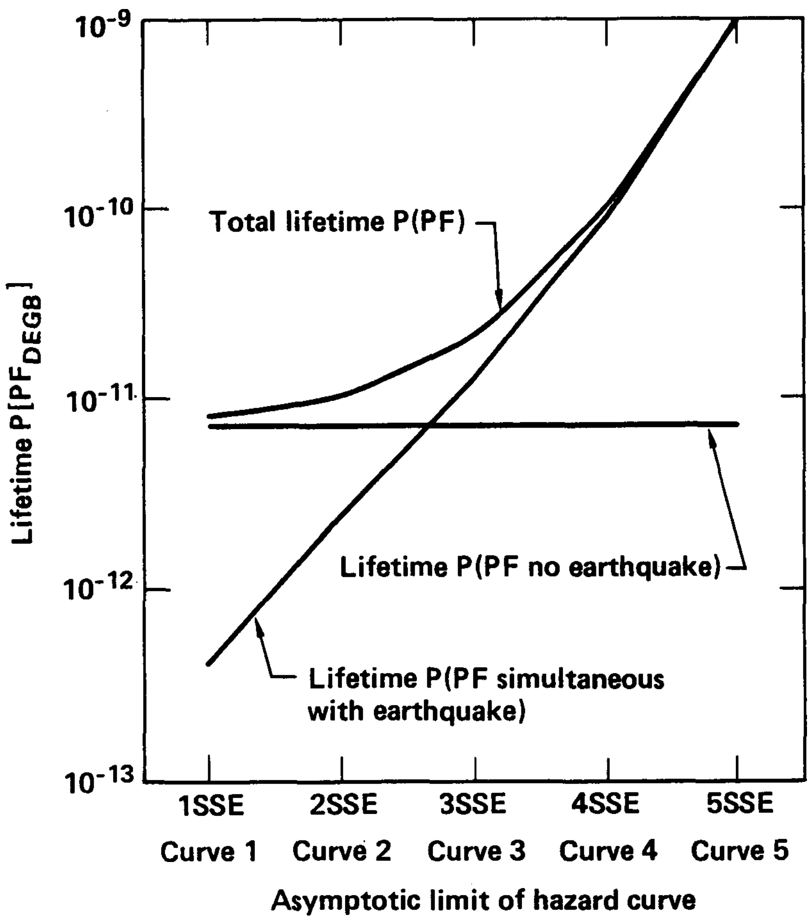

Figure 11. Comparison of non-conditional direct DEGB probabilities over 40-year plant life for Diablo Canyon reactor coolant loop piping, for seismic hazard curves shown in Fig. 10.

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 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 Grouping of Plants

Westinghouse provided data on the seismic design parameters and SSE design margins for reactor coolant loop supports for a total of 53 units.

These units were designed for various zero period peak ground accelerations and response spectra, ranging from scaled El Centro earthquake spectra to the Regulatory Guide 1.60 spectra, and including about 19 variations in between. Twenty-two units were analyzed for all three components of the safe shutdown earthquake. Most were designed using response spectrum techniques, except for five using time-history analysis and three using static analysis; most were designed using uncoupled models of the reactor coolant loop and the structure. Damping values for the piping and the structure also varied widely.

We classified the total population of Westinghouse plants into the following five groups:

(1) units with primary equipment (RPV, steam generator, reactor coolant pump) supports designed by Westinghouse.

(2) units with primary equipment supports designed by the architect-engineer.

(3) units located west of the Rocky Mountains.

(4) units with primary equipment supports designed by static analysis.

(5) units for which the utility owners did not participate in this study.

Our generic study of Westinghouse plants included plants from the first two groups. A representative plant from each of these groups was selected for a detailed analysis of equipment support fragilities. Since the generic treatment of these plants had to be conservatively biased, the plants with the lowest seismic capacity in each group were selected. This experience was subsequently used in defining equipment support fragilities for each of the remaining 44 plants in the first two groups. Seismic hazard was defined by generic hazard curves.

West coast plants (Group 3) were evaluated on a site-specific basis. Equipment support fragilities were developed for three of the four plants in this group -- San Onofre Unit 1 and Diablo Canyon Units 1 and 2. The fourth plant, Trojan, was not considered due to a lack of sufficiently detailed seismic hazard information for the plant site. In developing the support fragilities for Diablo Canyon, we considered not only information supplied to us by Westinghouse, but also the results of a seismic reevaluation performed by the Pacific Gas and Electric Company for an earthquake on the Hosgri fault having a peak ground acceleration of 0.75g.[11] For San Onofre we used the results of a similar seismic reevaluation performed by the plant owners.[12] Seismic hazard was defined on a site-specific basis.

Plants in the last two groups were not included in the evaluation.

4.3 Component Fragility

The seismic fragility of a component is defined as the conditional probability of its failure given a peak ground acceleration level. In our study of Westinghouse plants we included only those "critical" elements whose failure could contribute significantly to the probability of an indirectly-induced DEGB. A pilot study that we performed for the Zion nuclear power plant identified the steam generator supports, the reactor coolant pump supports, the reactor pressure vessel supports, and the overhead crane as critical equipment. 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.

In our generic 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 response, modeling assumptions, structural damping, and others (see Table 10), we applied correction factors to these design margins to obtain a uniform margin against failure for plants designed according to different methods. For each component, we then combined the probability distributions of its capacity and seismic response to obtain a "fragility curve" (Fig. 12) describing the probability of component failure as a function of peak ground acceleration.

Next, the conditional probability of pipe break given failure of each component was established. For example, assume that failure of the overhead crane causes the trolley to be released. However, if the trolley does not impact against a reactor coolant pipe, the RPV, or a steam generator, the crane failure does not result in DEGB. Based solely on containment building layout, the Zion crane trolley would only have about a 14% probability of causing DEGB if it fell; if the conditional probability of DEGB is considered over time, it is even less because during normal operation the crane is parked so that the trolley can not fall on critical equipment. 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 <u>any</u> 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.

4.4 Seismic Hazard

Seismic hazard relates the probability that an earthquake will occur causing ground motion exceeding a specified level. This is usually decribed by a set of seismic hazard curves (Fig. 5) 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 characteristic for all sites located east of the Rocky Mountains. We based these generic curves on six eastern and midwestern sites for which formal seismic hazard analyses have been performed. Two of these analyses -- Zion and Indian Point -- have already been published. The remaining four have not yet been published and the associated plants are not specified here in order to preserve their anonymity. As discussed in Vol. 3 of this report series, the seismic hazard curves from each of the six sites were normalized by dividing the peak ground acceleration by the larger of the SSE or 0.15g -- currently thought to be the acceptable minimum SSE in most parts of the eastern and midwestern U.S. -- to assure that each site was weighted equally. At each site, the normalized curves were pooled together as one population and the original subjective probability assigned to each curve was divided by six. The total set of curves was then condensed into the final generic curve set (median, upper bound, and lower bound) represented by Fig. 5.

Two plants on the more seismically active west coast -- Diablo Canyon and San Onofre Unit 1 -- were assessed using site-specific seismic hazard curves, owing to the low number of plants and the wide varability in seismic hazard among individual reactor sites.

For Diablo Canyon, we developed a set of site-specific hazard curves by consolidating the results of separate studies by three seismology consultants: Ang and Newmark, Blume, and Trifunac and Anderson.[7] We assumed that these curves, shown in Fig. 13, bound the probability of earthquake occurence at the plant site and assigned them equal subjective probabilities in accounting for uncertainty. However, because these curves did not include peak ground accelerations above 1.2g (about 1.5 times the SSE), we extrapolated them log-linearly to five times the SSE.

For the San Onofre site, we first developed a set of seismic hazard curves by consolidating the results of various independent studies including that performed by New Mexico Engineering Consultants for our evaluation of Combustion Engineering plants.[13] The curves asymptotically approach 0.67g, 0.93g and 1.05g peak ground acceleration, respectively. Since the curves lie reasonably close together, we chose to use only the upper and lower bound curves in our indirect DEGB evaluation, assigning them equal subjective probabilities in our uncertainty analysis. These curves are denoted as SONGS Set 1 in Fig. 14.

The asymptotic termination of these curves at 1.05g -- about 1.5 times the SSE peak ground acceleration -- is not universially accepted by seismologists. Therefore, to avoid neglecting the potential effect that earthquakes significantly greater than the SSE may have on indirect DEGB probability, we formed a second set of curves by combining the upper bound curve from the first set with the results of a U.S. Geological Survey study[14] and the study by Ang and Newmark for the Diablo Canyon site,[9] extrapolating the latter two log-linearly for peak ground accelerations beyond 0.8g. This curve set is denoted as SONGS Set 2 in Fig. 14.

4.5 Discussion of Results

The details of our evaluation of indirect DEGB in the reactor coolant loop piping of Westinghouse PWR plants are included in Vol. 3 of this report series. The following summary gives the key results of these evaluations.

Plants East of the Rocky Mountains

Our generic study of 46 Westinghouse plants east of the Rocky Mountains indicated that indirect DEGB in reactor coolant loop piping is a very unlikely event for plants located in this region. The median probability for these plants was about 10^{-7} events per plant-year, with 10th and 90th percentile values of 2.0×10^{-9} and 7.0×10^{-6} events per plant-year, respectively. As part of this study, two "lower bound" plants were selected, one that had originally been designed for combination of SSE and DEGB loads, and one that had been designed for the SSE alone, and analyzed in greater detail. Even for these two plants, the 90th percentile probability of indirect DEGB is on the order of 10^{-5} events per plant-year.

West Coast Plants

Our site-specific evaluation of San Onofre Unit 1 and Diablo Canyon Units 1 and 2 indicated that indirect DEGB in reactor coolant loop piping is also a very unlikely event for west coast plants. In particular, we found that:

- for Diablo Canyon, the best-estimate probability of indirect DEGB in the reactor coolant loop piping is 1.7×10^{-6} events per plant-year, with a 90% percentile value of 2.2×10^{-5} events per plant-year.

- for San Onofre, the best-estimate probability of indirect DEGB in the reactor coolant loop piping is 5.4×10^{-8} events per plant-year, with a 90th percentile value of 9.5×10^{-7} events per plant-year.

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. We therefore 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. 15).

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 presumably 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.

Volume 3 of this series provides a more complete discussion of our sensitivity studies, including a detailed description of Westinghouse quality assurance and quality control procedures for reactor coolant systems.

These values are slightly more than an order of magnitude higher than the corresponding generic probabilities for plants east of the Rocky Mountains, and are similar to the indirect DEGB probabilities estimated for the two lower bound eastern plants.

As part of our San Onofre evaluation, we developed two sets of seismic hazard curves: one in which maximum peak ground acceleration asymptotically approached 1.5 times the SSE, and another which included earthquakes much larger than the SSE. The median indirect DEGB probabilities predicted using the second set of curves increased by about two orders of magnitude -- from 5.4×10^{-8} to 4.7×10^{-6} events per plant-year -- over those predicted using the first set. This result indicates, not surprisingly, that the probability of indirect DEGB is strongly dependent on seismic hazard. This is in contrast to the results of our evaluations of direct DEGB probability, which, except for Diablo Canyon, was shown to be only weakly affected by earthquakes.

Because we did not have sufficiently detailed seismic hazard for the Trojan site, we felt that an evaluation of indirect DEGB in the reactor coolant loop piping at Trojan would have little meaning. However, Trojan is of similar vintage as Diablo Canyon and is located in one of the less seismically active areas on the west coast; it is therefore reasonable to expect that the results for the two California plants are representative for Trojan as well.

4.6 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

TABLE 10

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 11

Annual Probabilities of Indirect DEGB for Westinghouse PWR Plants

(events per plant-year)

	Confidence Limit[(1)]		
	10%	50%	90%
Plants East of the Rocky Mountains[(2)]			
Lowest Seismic Capacity Plants			
Designed for SSE + DEGB	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}
Designed for SSE alone	1.0×10^{-7}	2.4×10^{-6}	2.0×10^{-5}
All 46 Eastern Plants	2.0×10^{-9}	1.0×10^{-7}	7.0×10^{-6}
West Coast Plants			
San Onofre Unit 1[(3)]			
SONGS Set 1	3.1×10^{-10}	5.4×10^{-8}	9.5×10^{-7}
SONGS Set 2	1.3×10^{-7}	4.7×10^{-6}	4.9×10^{-5}
Diablo Canyon Units 1,2[(4)]	4.0×10^{-7}	1.7×10^{-6}	2.2×10^{-5}

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

(2) Generic seismic hazard curves used in evaluation (Fig. 5).

(3) Site-specific seismic hazard curves used in evaluation. See text and Fig. 14 for definition of "Set 1" and "Set 2" seismic hazard curves.

(4) Site-specific seismic hazard curves used in evaluation (Fig. 13).

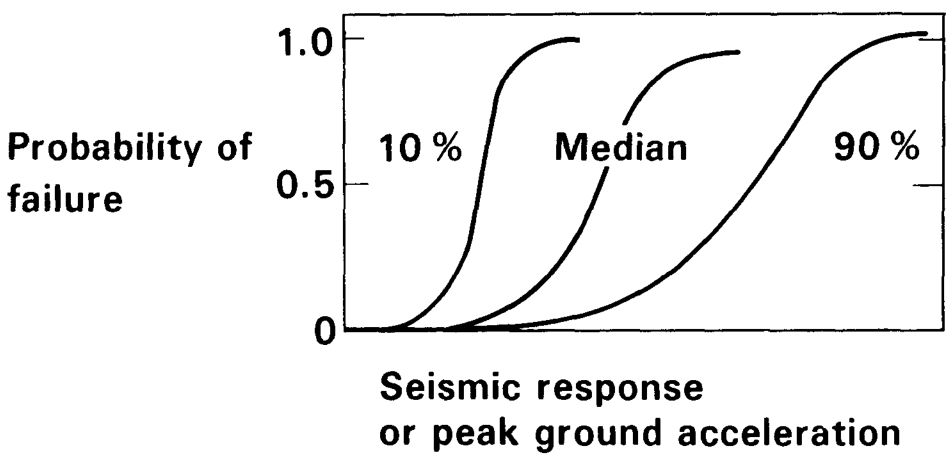

Figure 12. Typical curve set representing structural or equipment fragility.

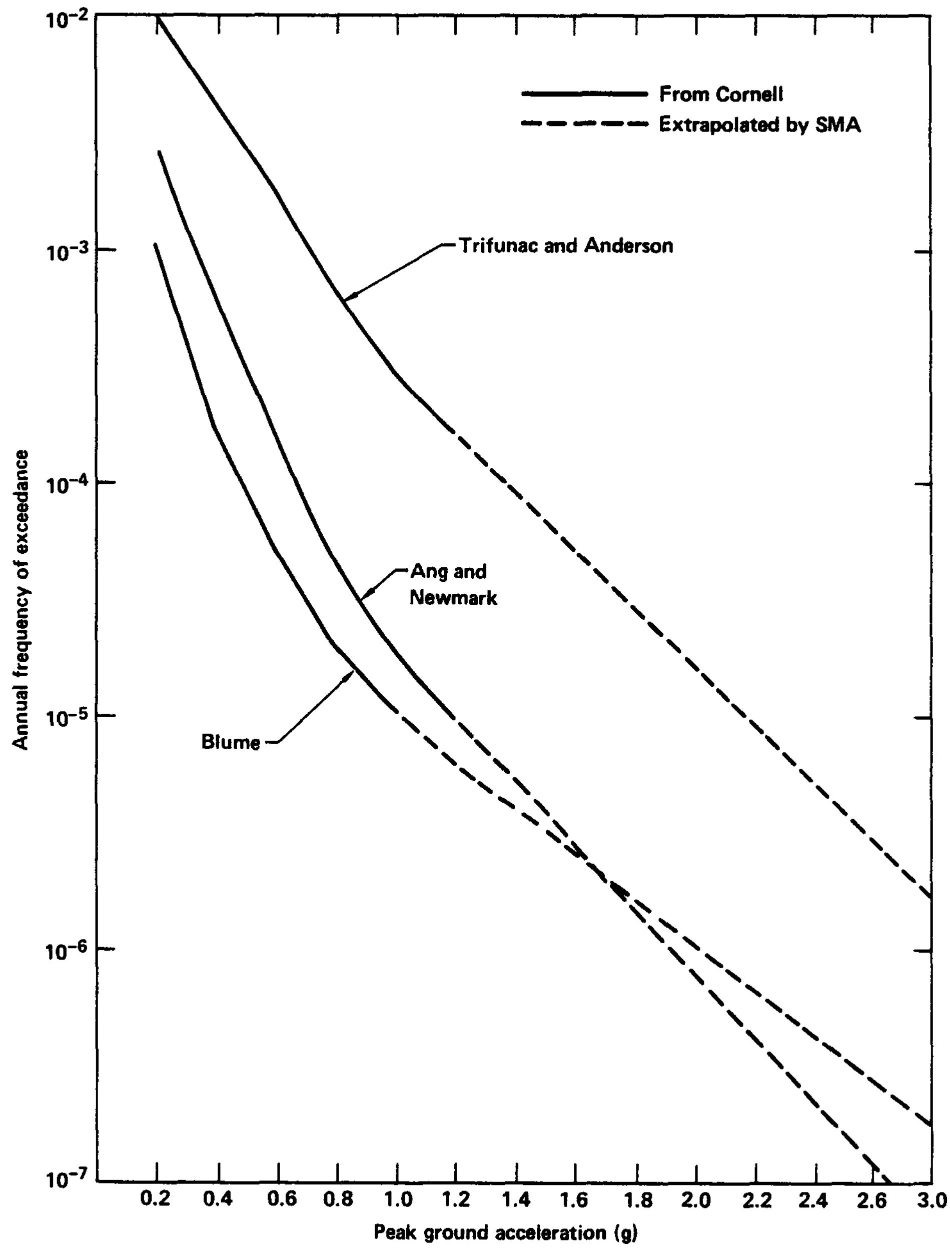

Figure 13. Seismic hazard curves used for estimating probability of indirect DEGB at the Diablo Canyon nuclear power plant.

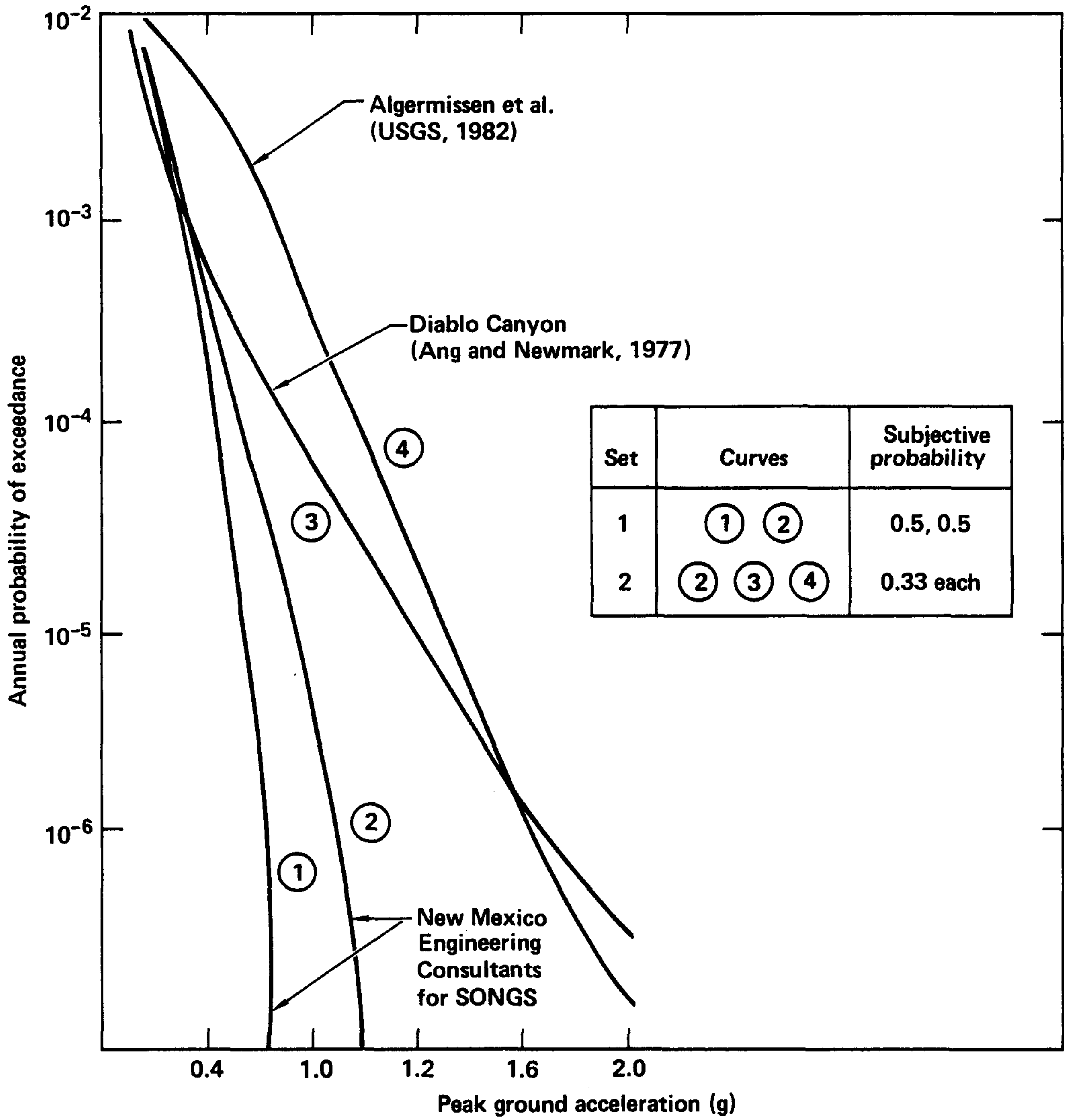

Figure 14. Seismic hazard curves used for estimating probability of indirect DEGB at the San Onofre nuclear power plant.

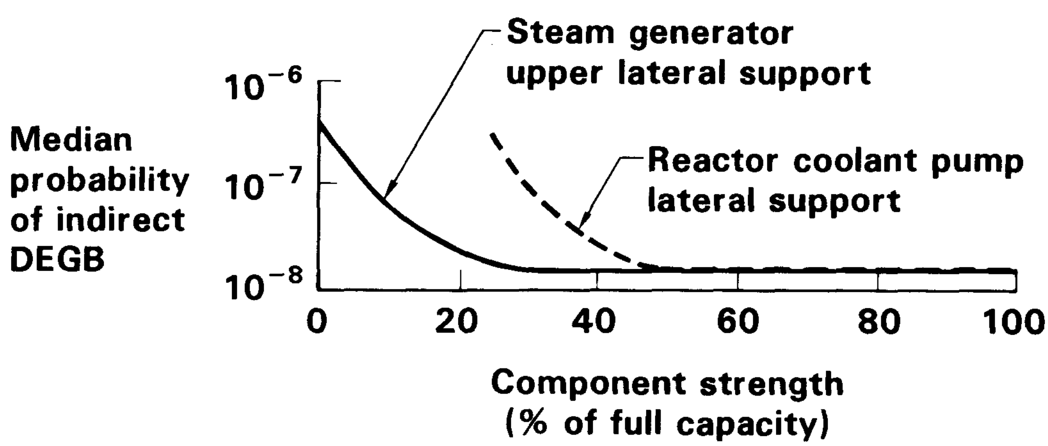

Figure 15. Typical effect of support capacity on probability of indirect DEGB.

5. SUMMARY AND CONCLUSIONS

5.1 Probability of Direct DEGB in Reactor Coolant Loop Piping

Plants East of the Rocky Mountains

We completed probabilistic analyses indicating that the probability of direct DEGB in reactor coolant piping is very small for Westinghouse PWR plants located east of the Rocky Mountains. These analyses calculated the growth of as-fabricated surface flaws at welded joints, taking into account loads on the piping due to normal operating conditions and seismic events. Other factors, such as the capability to detect cracks by non-destructive examination and the capability to detect pipe leaks, were also considered.

In particular, the results of our evaluations for 17 sample plants (33 plant units) indicate that:

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

- the "best estimate" probability of leak (through-wall crack) ranges from 1.3×10^{-8} to 1.5×10^{-7} events per plant-year, with a median value of 1.1×10^{-7} events per plant-year. The significantly greater probability of break compared to DEGB supports the concept of "leak before break" in PWR reactor coolant loop piping.

- uncertainty analyses indicated that the 90th percentile values of DEGB and leak probabilities for the sample plant with the highest probability of direct DEGB are 7.5×10^{-11} and 2.4×10^{-7} events per plant-year, respectively.

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

West Coast Plants

Plant-specific evaluations were performed for reactor coolant loop piping at two west coast plants: Trojan and Diablo Canyon. For Trojan, the median probability of direct DEGB was 2.2×10^{-13} events per plant-year, with 10th and 90th percentile values of 2.6×10^{-17} and 1.0×10^{-9} events per plant-year, respectively. The estimated median probability of leak was 5.5×10^{-8} events per plant-year, with 10th and 90th percentile values of 2.0×10^{-8} and 1.5×10^{-7} events per plant-year, respectively. These values are comparable to corresponding generic DEGB and leak probabilities for plants east of the Rocky Mountains. As in our generic evaluations, we found that normal operating loads, such as stresses due to pressure and thermal expansion, were the dominant contributors to pipe failure; earthquakes had a negligibly small effect.

For Diablo Canyon, the simultaneous occurrence of earthquakes and pipe break made a non-negligible contribution to the overall probability of direct DEGB. Using seismic hazard curves that we derived from three independent seismic hazard evaluations of the plant site, we estimated the probability of direct DEGB to be 2.5×10^{-11} events per plant-year, about one order of magnitude higher than the median value for plants east of the Rocky Mountains. We found that earthquakes less than or equal to the SSE had a negligible effect on conditional failure probabilities (i.e., assuming that a specified level of earthquake occurs), but that for earthquakes above this level, the simultaneous occurrence of earthquake and DEGB dominated the conditional probability of failure. However, the extremely low probability that such large earthquakes actually occur offsets the high conditional DEGB probabilities, keeping the overall DEGB probability low.

When comparing probabilities of direct DEGB for west coast plants with those for plants east of the Rocky Mountains, it is important to keep in mind that the west coast evaluations were made using site-specific seismic hazard information while the eastern plants were evaluated using generic seismic hazard curves. 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 United States; using the generic curves in lieu of site-specific seismic hazard information may be overly conservative for certain sites having particularly low seismicity. Therefore, the probabilities of direct DEGB for these eastern sites may actually be lower -- and the difference compared to west coast values accordingly greater -- than the median value estimated using the generic hazard curves.

5.2 Probability of Indirect DEGB in Reactor Coolant Loop Piping

Plants East of the Rocky Mountains

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

The results of our analyses indicated for Westinghouse plants east of the Rocky Mountains that:

- the critical components whose failure would result in DEGB were the reactor pressure vessel supports, the reactor coolant pump supports, and the steam generator supports. For the Zion Unit 1 plant used in our pilot study, the overhead crane in the containment building was also a critical component due to its atypical design. More typical crane designs, supported on rails mounted to the containment structure near the dome, did not contribute significantly to the probability of indirect DEGB.

- the estimated median probability of indirect DEGB (50th percentile value) is 1.0×10^{-7} events per plant-year, with a 90th percentile value of 7.0×10^{-6} events per plant-year.

- the median probability of indirect DEGB for one "lower bound" (i.e., lowest seismic capacity) plant designed for the combination of safe shutdown earthquake (SSE) and DEGB loads was 3.3×10^{-6} events per plant-year, with a 90th percentile value of 2.3×10^{-5} events per plant-year.

- the median probability of indirect DEGB for another lower bound plant designed for SSE alone (no DEGB loads) was 2.4×10^{-6} events per plant-year, with a 90th percentile value of 2.0×10^{-5} events per plant-year.

<u>West Coast Plants</u>

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

- the median probability of indirect DEGB in the Diablo Canyon reactor coolant loop piping is 1.7×10^{-6} events per plant-year, with a 90%

confidence limit of 2.2×10^{-5} events per plant-year. These values are about the same as those for the lowest seismic capacity plants east of the Rocky Mountains.

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

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

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

5.3 Conclusions and Recommendations

In general, the results of our evaluation indicate that the probability of DEGB in the reactor coolant loop piping of Westinghouse plants is extremely low under all plant conditions, including earthquakes. This is the case both for DEGB caused by crack growth at welded joints ("direct" DEGB) and DEGB resulting from the seismically-induced failure of heavy component supports ("indirect" DEGB). The probability of direct DEGB is typically four to five orders of magnitude lower than that of indirect DEGB, clearly identifying indirect causes as the dominant mechanism leading to DEGB in reactor coolant loop piping. Our results further indicate that:

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

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

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

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.

6. RESPONSE TO NRC QUESTIONS

The NRC, in its letter of September 16, 1983, requested that certain key information related to resolution of Task Action Plan B-6, "Loads, Load Combinations, and Stress Limits," be emphasized and highlighted in reporting the results of our reactor coolant loop piping investigations.[15] The letter posed the following questions:

(1) What probabilities do you estimate for direct and indirect seismically induced pipe rupture in reactor coolant loop piping, and what are the limitations on these estimates? More specifically, what is the likelihood that an earthquake event will occur simultaneously with a pipe rupture event in reactor coolant loop piping?

(2) What statements would you make on the reliability of heavy component supports, given that future reactors may not be designed against the combination of pipe rupture in reactor coolant loop piping and SSE?

(3) What statement would you offer regarding the application of these results to combinations of SSE and short-term LOCA effects (decompression waves and associated thermal transients in piping, pipe whip, jet impingement) and long-term LOCA effects (containment and compartment pressurization)?

These issues are addressed in the following three sections, respectively. A fourth section discusses issues associated with the definition of alternate bases for plant design, given that the double-ended guillotine break of a reactor coolant loop pipe were eliminated as a design basis event.

6.1 Effect of Earthquakes on DEGB Probabilities

Direct DEGB

Our analyses have generally shown that the probability of direct DEGB is only very weakly affected by an earthquake. We found for both leak and DEGB that the probability of failure caused by an earthquake was two or more orders of magnitude lower than that of failure occurring independently of an earthquake (i.e., due to all loading conditions, including stresses resulting from pressure and restraint of thermal expansion). For the sample plant with the highest probability of direct DEGB (2.5×10^{-10} events during 40-year plant life), the probability of an earthquake causing direct DEGB was about 2.1×10^{-12} events during plant life. This result implies that direct DEGB and a safe shutdown earthquake can be considered independent random events whose probability of simultaneous occurence during plant life is negligibly low.

The sole exception to this general result was Diablo Canyon, for which the simultaneous occurrence of earthquake and DEGB contributed non-neglibibly to the overall probability of failure when the earthquake exceeded twice the SSE level. Given the occurrence of an earthquake five times the 0.75g SSE, the median conditional probability of direct DEGB is 0.33×10^{-5} events during plant life (40 years). Increasing the seismic load, as we did in one of our sensitivity studies, increases this value further. For example, combining a seismic response factor corresponding to the 90th percentile on the distribution (i.e., about 1.3 standard deviations off the mean) with the occurrence of an earthquake of five times the SSE increases the conditional probability of direct DEGB to 2.0×10^{-2} events during plant life. These results do not, however, include the effect of seismic hazard. When seismic hazard is considered, the non-conditional probability of direct DEGB decreases dramatically, to about 2.5×10^{-11} events per plant-year, due to the very low probability that earthquakes larger than the SSE actually occur.

When interpreting these results, it is important to keep two key factors in mind:

- the peak ground acceleration associated with an SSE at Diablo Canyon is 0.75g, or about five times the minimum SSE acceleration assumed in our generic hazard curves for plants east of the Rocky Mountains. Five times the SSE (the maximum PGA considered in our seismic hazard curves) at Diablo Canyon is 3.75g, or 25 times the minimum SSE assumed for plants east of the Rocky Mountains.

- in our evaluations, stresses for earthquakes larger than the SSE are estimated by linearly extrapolating the SSE stresses, an admittedly conservative assumption that was made for analytic convenience.

The high conditional DEGB probability given above is hardly surprising in light of the massive stresses implied by this conservative assumption, particularly when they are coupled with a seismic response factor one-and-a-quarter standard deviations off of the median value.

Given the increased importance of seismic effects and also recognizing that the precise seismic hazard at the Diablo Canyon site has been, and continues to be, a controversial subject, we performed an extensive series of sensitivity calculations to assess the effect that various seismic hazard assumptions had on our median probability of direct DEGB in the Diablo Canyon reactor coolant loop piping. The results of these sensitivity calculations indicated that:

- the non-conditional probability of direct DEGB estimated using the LLNL seismic hazard curve decreases by about two orders of magnitude when the upper limit on peak ground acceleration in the seismic hazard curve is reduced from five to one SSE. The leak probabilities are essentially unaffected by changing the upper limit.

- the non-conditional probabilities of direct DEGB estimated for each of the three seismic hazard curves from which our composite curves were derived varied by less than a factor of two, and were all exceeded by those estimated using the LLNL curve.

From these sensitivity evaluations we conclude that the probability of direct DEGB at Diablo Canyon is relatively insensitive to the particular seismic hazard curve selected from those used in our evaluation. Instead, it depends more on the PGA level to which the seismic hazard curve extends. Defining a "best" upper limit of PGA is outside the scope of the work discussed in this report, and is more appropriately addressed by detailed seismic hazard evaluations.

Indirect DEGB

We have identified earthquake as the only credible cause of indirect DEGB; thus, the probability of indirect DEGB also expresses the probability that DEGB and an earthquake simultaneously occur. For the lowest capacity plant east of the Rocky Mountains, the estimated 90th percentile probability is 2.0×10^{-5} events per plant-year. The 90th percentile probability generically applicable to all plants in this region is 7.0×10^{-6} events per plant-year, compared to corresponding values of 2.2×10^{-5} and 9.5×10^{-7} events per plant-year for Diablo Canyon and San Onofre Unit 1, respectively. Not surprisingly, seismic hazard had a significant effect on the estimated probability of indirect DEGB.

Given the higher magnitude and higher frequency of earthquakes on the west coast, the relatively close agreement between the results for west coast plants and plants east of the Rocky Mountains may at first seem surprising. When interpreting these results, however, it is important to keep in mind that both the intensity (seismic loads) and occurrence rate (seismic hazard) of earthquakes are considered to estimate the probability of indirect DEGB. We base support fragility on vendor-supplied margins against seismic loads.

Assuming that an SSE occurred, plants of similar configuration (but possibly of different design, if SSE loads were different) would be expected to have similar conditional probabilities of indirect DEGB because the Code requires specified margins to be maintained for SSE loads regardless of site location. The differences between the results for eastern and western plants can be therefore attributed mainly to higher seismic hazard, which increases the probability of indirect DEGB simply because earthquakes are more likely to occur.

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, as was discussed earlier, using the generic curves in lieu of site-specific seismic hazard information may be overly conservative for certain sites. We are therefore confident in the low probabilities of DEGB yielded by our indirect DEGB evaluation.

6.2 Reliability of Heavy Component Supports

If the probability of DEGB is determined to be acceptably low under all plant conditions, including seismic events, 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 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 is, not surprisingly, evidenced by the 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 subject to debate. 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.

6.3 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.[16] Thus, 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 Westinghouse 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.

6.4 Replacement Criteria

The results of our evaluation of 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.[17] 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) 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, the relative effect of each system failure on overall plant safety.

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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