NUREG/CR-4792
UCID-20914
Vol. 1

PROBABILITY OF FAILURE IN BWR REACTOR COOLANT PIPING

Vol. 1: Summary Report

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Prepared for
U.S. Nuclear Regulatory Commission

Lawrence
Livermore
National
Laboratory

Legacy: no

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

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Manuscript Completed: December 1988
Date Published:

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Prepared for
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN No. A0133

Report NUREG/CR-4792, **Probability of Failure in BWR Reactor Coolant Piping**, is published in the following four volumes:

- Vol. 1: Summary Report
- Vol. 2: Pipe Failure Induced by Crack Growth and Failure of Intermediate Supports
- Vol. 3: Probabilistic Treatment of Stress Corrosion Cracking in 304 and 316NG BWR Piping Weldments
- Vol. 4: Guillotine Break Indirectly Induced by Earthquakes

This report is Volume 1 of the report series.

ABSTRACT

This report summarizes a probabilistic reliability evaluation of BWR reactor coolant piping performed for the U.S. Nuclear Regulatory Commission (NRC) by the Lawrence Livermore National Laboratory (LLNL). In this evaluation, LLNL estimated the probability of a double-ended guillotine break (DEGB) in the main steam, feedwater, and recirculation loop piping of a representative Mark I BWR plant. Two causes of pipe break were considered: crack growth at welded joints, and the earthquake-induced failure of supports for piping and components. A probabilistic fracture mechanics model, including intergranular stress corrosion cracking (IGSCC) in Types 304 and 316NG stainless steels, was used to estimate the probability of crack-induced pipe break. The probability of pipe break indirectly caused by support failure was estimated by applying reliability techniques to supports for "heavy components", such as the reactor pressure vessel, as well as to conventional pipe supports such as spring hangers and snubbers. Our probabilistic fracture mechanics evaluation found that the probability of crack-induced DEGB in main steam, feedwater and, if IGSCC is not a factor, recirculation piping is very low. In IGSCC-susceptible Type 304SS piping, stress corrosion dominates the probability of DEGB due mainly to cracks that initiate during the first few years of plant life; replacing Type 304 piping with IGSCC-resistant Type 316NG lowers DEGB probabilities by several orders of magnitude. We also found that the probability of pipe break caused by seismically-induced support failure is low regardless of whether "heavy component" supports or conventional pipe supports are being considered.

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ACKNOWLEDGEMENTS

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

The authors wish to express their sincere appreciation to the Carolina Power and Light Company for allowing us to use the Brunswick Steam Electric Plant as our case study and for providing the required input data for our probabilistic evaluations. We also wish to express our appreciation to the Electric Power Research Institute and to the Argonne National Laboratory for providing most of the experimental data on which we based the probabilistic model of intergranular stress corrosion cracking applied in this evaluation.

The LLNL Load Combination Program, through which the reported work was performed, is a multi-disciplinary effort drawing on the talents of many individuals. We would particularly like to acknowledge the contributions of the following participants in the BWR evaluation:

- T. Lo (LLNL), who developed the support reliability methodology which we applied in our evaluation of pipe break due to failure of "intermediate" pipe supports, and N.C. Tsai (NCT Engineering), who developed the support fragility descriptions applied in this evaluation

- S.E. Bumpus and D.J. Chinn (LLNL), who actually performed most of the actual reliability analyses in our evaluations of crack-induced pipe failure

- D.O. Harris and D.D. Dedhia (Failure Analysis Associates), who developed the advanced IGSCC model applied in our recirculation loop evaluations

- W.J. Shack (Argonne National Laboratory) and S.J. Bush (Review & Synthesis Associates), for their valuable consultation during development of the IGSCC model

- G.S. Hardy, R.D. Campbell, and M.K. Ravindra (formerly of Structural Mechanics Associates), who performed the "heavy component" support failure segment of our "indirect DEGB" evaluations.

EXECUTIVE SUMMARY

This report summarizes probabilistic analyses of BWR reactor coolant piping performed by the Lawrence Livermore National Laboratory (LLNL) for the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. Using a general "reliability" approach together with specific computational tools developed during prior evaluations of PWR reactor coolant loop piping, LLNL estimated the probability of a double-ended guillotine break (DEGB) in the main steam, feedwater, and recirculation piping of a BWR plant. For these piping systems, the results of our investigations provide NRC with one technical basis on which to:

(1) reevaluate the current general design requirement that DEGB be postulated in the design of nuclear power plant structures, systems, and components against the effects of postulated pipe breaks. Recent NRC rulemaking actions, based in large part on the results of our PWR evaluations, now provide a means for eliminating dynamic effects of reactor coolant loop breaks (e.g., pipe whip, jet impingement) as a basis for PWR plant design.

(2) determine if an earthquake could induce a DEGB and thus reevaluate the design requirements that pipe break loads be combined with those resulting from a safe shutdown earthquake (SSE). Recent deviations from the NRC Standard Review Plan, for example, now allow decoupling of SSE and DEGB loads for PWR reactor coolant loop piping.

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

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

Background Information

Over the past several years, LLNL has completed generic reliability evaluations of reactor coolant loops in PWR nuclear steam supply systems manufactured by Westinghouse, Combustion Engineering, and Babcock & Wilcox. In these evaluations, we performed the following:

(1) estimated the probability of direct DEGB taking into account such contributing factors as the initial size (depth and length) of pre-existing fabrication flaws, pipe stresses due to normal operation and sudden extreme loads (such as earthquakes), the crack growth characteristics of pipe materials, and the capability to

detect cracks or to detect a leak if a crack were to penetrate the pipe wall. For this purpose, we developed a probabilistic fracture mechanics model using Monte Carlo simulation techniques, implemented in the PRAISE (Piping Reliability Analysis Including Seismic Events) computer code.

(2) estimated the probability of indirect DEGB by identifying critical 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 then combining this result with the probability that an earthquake occurs producing a certain level of excitation ("seismic hazard").

(3) for both causes of DEGB, performed sensitivity studies to identify key parameters affecting the probability of pipe break. We also performed uncertainty studies to quantify how uncertainties in input data affect the uncertainty in the final estimated probability of pipe break.

The results of these evaluations consistently indicated that the probability of a DEGB in PWR reactor coolant loop piping is extremely small, about 1E-7 events per reactor-year from indirect causes, and less than 1E-10 events per reactor-year from direct causes. It was also found that thermal stresses dominated the probability of direct DEGB, and that earthquakes contributed only negligibly. These results suggested 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 — warranted reevaluation for PWR reactor coolant loop piping. Details of these investigations have been extensively documented in numerous NUREG reports published by the NRC.

The overall objectives and approach of the BWR study described in this report were the same except that additional potential failure mechanisms were added. We limited our investigation to Mark I plants, which have recirculation piping historically susceptible to the effects of intergranular stress corrosion cracking (IGSCC). Although all of our evaluations have been generally similar, two important aspects distinguish the BWR study from the earlier PWR evaluations:

(1) the susceptibility of certain BWR stainless steels to IGSCC required that we develop an appropriate probabilistic model of corrosion phenomena. Stress corrosion is generally not perceived as a problem in PWR primary loop piping and was therefore not considered in our earlier evaluations.

(2) the greater complexity and flexibility of BWR recirculation piping compared to PWR primary loops required that we incorporate conventional supports (e.g., snubbers, spring hangers) for piping and light loop components into the evaluation.

Our "direct DEGB" evaluation focussed on development of an IGSCC model and on application of this model to a "representative" BWR recirculation system. We also applied our more conventional piping reliability techniques to estimate probabilities of crack-induced leak and break in main steam and feedwater piping. Our "indirect DEGB" evaluation of pipe break caused by seismically-induced support failure considered not only "heavy component" supports (the main focus of the equivalent PWR evaluations), but the more complex issues associated with failure of conventional "intermediate" supports as well.

Pipe Failure Due to Crack Growth

Using the Brunswick Mark I BWR plant as a case study, we completed probabilistic fracture mechanics analyses indicating that the probability of direct DEGB is very low for BWR main steam and feedwater piping, and for BWR recirculation loop piping if stress corrosion cracking is not a factor. 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 to postulated earthquakes. We also considered other factors, such as the capability to detect cracks by non-destructive examination and the capability to detect pipe leaks, in our analyses.

The best-estimate lifetime system leak and DEGB probabilities for the Brunswick major coolant piping systems are rather low and fall within narrow ranges. For the three piping systems considered, the system leak probabilities vary between 2.4E-6 and 3.8E-5 over the life of the plant, or between about 6.0E-8 and 1.0E-6 per reactor year. The DEGB probabilities behave similarly, ranging from 4.0E-11 to 1.5E-10 over the lifetime of the plant, or about 1.0E-12 to 3.8E-12 per reactor-year. These results are similar to those estimated for PWR reactor coolant piping, both in absolute magnitude and in the differential (three or more orders of magnitude) between the probabilities of leak and break.

We also performed a rigorous uncertainty study, the results of which showed that even at the 90th-percentile confidence level, the highest lifetime probabilities are 1.2E-3 and 5.0E-8 for the leak and the DEGB, respectively, or about 3.0E-5 and 1.3E-9 per reactor-year. Both of these probabilities are for the feedwater line, and are still comparable to our earlier PWR results despite the increased complexity of the BWR systems considered.

To address the effect of stress corrosion cracking (SCC) on the probability of failure in BWR recirculation loop piping, we developed an advanced SCC model for the PRAISE code. This semi-empirical model is based on experimental and field data compiled from several sources. Using probabilistic techniques, the model addresses various stress corrosion phenomena, including crack growth, crack initiation, and linking of multiple cracks. The model also considers the effect of residual stresses in addition to cyclic stresses resulting, for example, from normal plant operation.

The model covers not only the Type 304 stainless steel found in older BWR recirculation piping, but also Type 316NG ("nuclear grade"), a low-carbon alloy widely regarded as an SCC-resistant replacement for Type 304. Crack growth rates and times-to-initiation for each material are correlated against "damage parameters" which consolidate the separate influences of several individual parameters. The damage parameters are multiplicative relationships among various terms which individually describe the effects of the various phenomena on SCC behavior, including (1) environment, specifically coolant temperature, dissolved oxygen content, and level of impurities, (2) applied loads, including both constant and variable loads to account for steady-state operation and plant loading or unloading, respectively, (3) residual stresses, and (4) material sensitization.

After completing our development work, we applied the model to the recirculation loop piping in the Brunswick BWR plant. We estimated the leak and DEGB probabilities both for an existing recirculation loop and for a proposed "replacement" configuration having fewer weld joints than the original system. We also investigated the relative effects of Types 304 and 316NG stainless steel on the estimated probabilities of leak and break. From the results of these evaluations, we observed that:

- recirculation loops fabricated from Type 304 stainless steel are predicted to leak after about 10 years of operation, a result consistent with some field observations. If the 304SS material is replaced by 316NG and the existing loop configuration is retained, the system leak probability at ten years is effectively zero. The end-of-life system leak probability with 316NG (i.e. after another 30 years of operation) is about 5E-1 per loop, or about 2E-2 per loop-year assuming "worst case" applied stresses and no ISI. The replacement configuration, with its fewer welds, reduces leak probabilities still further.

- for recirculation loops fabricated from Type 304 stainless steel, the system probability of DEGB is about 1E-2 after ten years of operation (or about 1E-3 per loop-year), increasing to about 2E-2 by the end of plant life. For recirculation loops fabricated from Type 316NG, the system probability of DEGB is zero for the first 30 years of operation, even under "worst case" applied stresses and no ISI, and on the order of 1E-4 per loop-year or less over the final ten years of plant life.

In all cases, the 316NG appears to owe its corrosion resistance mainly to the fact that (1) fewer cracks initiated than in the 304SS material, and (2) those that did initiate typically did so later in plant life. Once a crack initiates, however, its subsequent growth rate is not significantly affected by material type. We also found that system failure concentrates at bypass welds (if present) and at riser welds. Further investigation of the relative susceptibility of recirculation loop welds to crack-induced failure is currently in progress for NRC.

We present these results as "best-estimate" values, although many of the input parameters used in the analyses can be regarded as conservative. Due to the excessive computer time requirements of the stress corrosion calculation (compared, for example, to that for thermal fatigue only), we were unable to perform extensive uncertainty analyses within the time and resources available to us. The results of analyses performed during model development, however, indicate that the probability of SCC-induced pipe failure (either leak or DEGB) is most sensitive to the description of residual stress assumed. Therefore, we would expect the uncertainty in the estimated failure probabilities to be significant, owing to variations in plant-to-plant residual stresses.

Pipe Failure Due to Support Failure

We have developed detailed approaches for evaluating how support failures caused by earthquakes would contribute to the overall probability of piping system failure. Two different approaches are used, depending on the type of support considered:

(1) a "support reliability" approach for heavy component supports, that is, supports whose failure could reasonably be expected to cause pipe failure under all circumstances, and

(2) a more rigorous "piping reliability" approach, where the effect of support failure is incorporated directly into a probabilistic fracture mechanics evaluation of the piping to determine if and under what conditions support failure would cause a pipe to break.

In evaluating the probability of "indirect DEGB", we first identified critical components whose failure could plausibly result in a pipe break; in our BWR study, seismically-induced failure of supports for piping and components was determined to be the most likely cause of an indirect pipe break. We then developed a "fragility" description for each critical support relating its probability of failure given the occurrence of an earthquake of given peak ground acceleration. Finally, we estimated the non-conditional probability of support failure ("support reliability") by convolving the support fragilities with an appropriate description of seismic hazard. "Seismic hazard" relates the probability of an earthquake exceeding a given level of peak ground acceleration. Both approaches incorporated these same basic steps; the piping reliability approach, however, goes one step further by explicitly estimating the probability of pipe failure given support failure.

The only critical "heavy component" supports requiring consideration in our BWR evaluation were those making up the reactor support structure. These included the lower support structure at the base of the reactor pressure vessel, as well as the lateral stabilizers at the top of the vessel. We also considered many other plausible causes of indirect DEGB unrelated to earthquakes, such as crane failure and pump flywheel missiles, but determined these to be of negligible significance compared to support failure.

We found that the probability of indirect DEGB due to failure of heavy component supports was about 2E-8 events per reactor-year, with a 90th-percentile value (confidence limit) of 5E-7 per reactor-year. Our results further indicated that the "star" stabilizer at the top of the reactor pressure vessel, which restrains the RPV against lateral motion in the event of an earthquake, was the primary contributor to failure rather than the main support structure at the bottom of the vessel.

This result was comparable to those from our PWR reactor coolant loop evaluations, in which we investigated the seismically-induced failure of RPV, steam generator, and reactor coolant pump supports. For Westinghouse plants, the median probability of indirect pipe break was about 1E-7 per reactor-year for plants east of the Rocky Mountains (based on generic seismic hazard curves), and about 3E-6 per reactor-year for plants on the more seismically active west coast; "upper bound" (i.e. 90% confidence level) probabilities were typically about one order of magnitude higher. Equivalent results for Combustion Engineering and Babcock & Wilcox reactor coolant loop supports were comparable to the Westinghouse results.

Our evaluations of "indirect" DEGB caused by heavy component support failure assumed that support failure unconditionally resulted in pipe break. This assumption was regarded as conservative, but nevertheless resulted in very low DEGB probabilities. To have assumed that failure of a snubber or a constant-load support would similarly cause a DEGB in BWR recirculation loop piping would have been unreasonably conservative; in other words, a simple "support reliability" evaluation would no longer suffice. We therefore developed a more sophisticated approach to incorporate the effect of support fragility into the probabilistic fracture mechanics evaluation, which we used to investigate the effect of support failure on the probability of DEGB. As discussed earlier, the need to incorporate support failure in the fracture mechanics evaluation blurs the distinction between "direct" and "indirect" DEGB, leading us back towards a more integrated approach for estimating the probability of pipe break.

The "piping reliability" approach that we applied to the recirculation loop in our study comprised four steps:

(1) Identify critical supports and support failure combinations, then estimate support fragilities. Our study considered not only "conventional" pipe supports (e.g. spring hangers, snubbers), but the supports for the recirculation loop pump as well.

(2) Calculate structural responses (e.g. pipe stresses) for each combination of support failure. We considered 15 cases of support failure, in addition to the "no support failure" case.

(3) Estimate the conditional pipe failure probabilities at weld joints for all support failure combinations.

(4) Estimate the non-conditional system probability of pipe break for all support failure combinations.

The results of this study indicated that the maximum probability of recirculation loop DEGB due to failure of intermediate supports is about 3.0E-6 per plant lifetime (or about 7.5E-8 per reactor-year) when earthquakes up to five times the SSE are considered in the seismic hazard description. If we only consider earthquakes up to twice the SSE, the lifetime probability of DEGB drops to about 1.7E-11, or about 4.3E-13 per reactor-year.

These probabilities were low enough to allow failure of "intermediate" supports -- and the redistribution of weld joint stresses that would result -- to be omitted from our subsequent detailed evaluations of pipe break due to crack growth.

Application of Reliability Analysis in Licensing Assessments

The usefulness of probabilistic evaluations in regulatory applications has already been demonstrated through recent NRC rulemaking actions based in large part on the results of LLNL piping reliability studies. Most notable among these were revisions in General Design Criterion 4 which, first for PWR reactor coolant loop piping and later for piping systems in general, provide a mechanism for eliminating the dynamic effects of postulated pipe breaks from the plant design basis.

The analytic techniques developed and applied during the work described in this report include a means of assessing the reliability of reactor coolant piping subject to the effects of stress corrosion cracking. Future applications of reliability analysis in licensing assessments include the following:

- developing specific licensing criteria. The NRC recently published criteria, including guidelines for in-service inspection, which form the basis for future licensing decisions concerning austenitic BWR piping susceptible to IGSCC. Probabilistic evaluations like the one discussed in this report could conveniently be applied to more fundamentally define just what constitutes an "acceptable" piping inspection program.

- assessing the effectiveness of recommended inspection schedules, relative to alternate inspection schemes (e.g. more or less frequent inspection, greater or lesser extent of inspection).

- assessing the effectiveness, either relative or absolute, of such measures as inductive heating stress improvement (IHSI), hydrogen water chemistry and weld overlay, which enhance the performance of piping susceptible to stress corrosion cracking.

Our probabilistic fracture mechanics techniques have proven to be well-suited for such applications.

We have also developed reliability approaches for evaluating how support failures caused by earthquakes would contribute to the overall probability of piping system failure. The "support reliability" and "piping reliability" approaches described in this report have been applied to various reactor coolant piping systems in both PWR and BWR plants. The results of these evaluations have typically indicated that the likelihood of pipe break due to seismically-induced support failure is small, not only for the large, stiff piping found in PWR primary systems, but for more complex, more flexible piping systems as well. From the standpoint of addressing specific regulatory issues associated with piping behavior, a reliability approach also yields the following:

- the relative contribution of various failure scenarios to the overall likelihood of pipe system failure, in other words, the "safety significance" of each failure scenario.

- the relative "safety significance" of individual supports, in other words, identification of those supports whose failure would most serious affect system integrity.

- an assessment of system failure based on realistic failure criteria reflecting the actual behavior of the piping, rather than on simple exceedance of code allowable limits.

The general reliability techniques that we developed in this study could be applied to any piping system, given input data equivalent to that which we applied for recirculation loop piping. In principle, the support reliability techniques could, with appropriate modifications, also be applied to non-piping systems (such as cable trays and their supports) to, for example, investigate the relative influence of individual supports and support failure scenarios on overall system reliability.

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.

As Fig. 1.1 illustrates, postulation of DEGB potentially 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 has presented a particular problem for the nuclear industry. For piping systems inside of containment, NRC requirements stipulate that breaks be postulated at terminal ends as well as 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 surprisingly, design, fabrication, and installation of pipe whip restraints represent major capital expenses 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. Many experts believe that pipe whip restraints and jet impingement barriers may actually decrease the reliability of piping systems by limiting access to pipe welds, therefore reducing the effectiveness of in-service inspection and thus increasing pipe stresses caused by restraint of thermal expansion.

Another important requirement has been 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 became controversial 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 or "annulus pressurization" in boiling water reactor (BWR) 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 could result in reduced reliability during normal plant operation. For example, present seismic design methods tend to result in stiff piping systems and more supports when additional strength is provided for the earthquake loading. Because a stiff piping system is subjected to greater cyclic thermal stress than a flexible one under normal thermal operating loads, reliability may be reduced under normal conditions [2]. Unanticipated 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 cannot justify changing design requirements; the costs of meeting these requirements are the industry's responsibility. However, for older operating plants to comply with the more recent 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 must reassess the design requirement and allow continued operation with no or only limited modifications.

Until recently, no basis acceptable to the NRC staff existed for excluding DEGB from the plant design basis. Over the past few years, however, extensive deterministic fracture mechanics research has supported the "leak before break" concept for nuclear power plant piping. The fundamental premise of leak-before-break is that the materials used in nuclear power plant piping (particularly piping connected to the reactor coolant pressure boundary) are sufficiently tough that even large through-wall cracks resulting in coolant leak rates well in excess of those detectable by present leak detection systems would remain stable and not result in a DEGB.

The Lawrence Livermore National Laboratory, through its Nuclear Systems Safety Program (NSSP), has performed probabilistic "reliability" analyses of PWR and BWR reactor coolant loop piping for the NRC Office of Nuclear Regulatory Research. Specifically, LLNL has estimated the probability of a double-ended guillotine break (DEGB) in the reactor coolant loop piping of 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 general requirement that DEGB be assumed in the design of nuclear power plant structures, systems, and components.

(2) determine if an earthquake could induce a DEGB, and thus reevaluate the 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 reactor coolant loop DEGB as a design basis for PWR plants could have far-reaching consequences. If it can be shown, for example, that an earthquake will not induce DEGB, then the two can be considered independent random events whose probability of simultaneous occurrence is negligibly low; thus, the design requirement that DEGB and SSE loads be coupled could be removed. If the probability of a reactor coolant loop DEGB is very low under <u>all</u> plant conditions, not only seismic events, then the dynamic effects of a pipe break could be eliminated altogether. 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 maintain 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 or other routine maintenance activities, pipe whip restraints would not have to be reinstalled. A value-impact assessment performed by LLNL indicated that exclusion of reactor coolant loop DEGB from the design basis of operating plants and plants currently under construction would save tens of millions of dollars and reduce radiation exposure to plant personnel by tens of thousands of man-rem [3]. For future plants, the selective exclusion of breaks in other piping systems could conceivably save up to $100 million per unit in design and construction costs.

The work presented in this report is a continuation of work performed in the NSSP Load Combination Program. In Phase I we developed a probabilistic fracture mechanics methodology for estimating the likelihood of "direct" DEGB — pipe break caused by crack growth at welded joints — in the reactor coolant loop piping of PWR plants. We applied this methodology in an extensive "pilot" study of Zion Unit 1, a four-loop Westinghouse PWR plant operated by the Commonwealth Edison Company of Illinois. We also performed a limited study in which we identified the supports for the reactor pressure vessel, reactor coolant pumps and steam generators as those "critical" components whose failure could indirectly cause a DEGB, and then estimated the probability that any one of these supports could fail. The resultant probability of "indirect" DEGB in the reactor coolant piping was not, however, explicitly estimated in Phase I.

The Phase I investigations were documented extensively [4] and presented before the Advisory Committee on Reactor Safeguards (ACRS) in December 1980. Following this presentation, the ACRS asked us to perform three additional studies to (1) evaluate indirect DEGB in depth, (2) assess the effect that design and construction errors might have on the probability of indirect DEGB, and (3) generalize the Zion study to include other PWR plants, not only those with Westinghouse nuclear steam supply systems, but those with Combustion Engineering (C-E) and Babcock & Wilcox (B&W) reactors as well. This request formed the basis of our subsequent PWR evaluations.

To arrive at a general conclusion about the probability of DEGB in the reactor coolant loop piping in all PWR plants, LLNL has taken a vendor-by-vendor approach. For each of the three PWR vendors in the United States we:

(1) estimated 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 were to penetrate the pipe wall.

(2) estimated 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 ground motion ("seismic hazard").

(3) for both causes of DEGB, performed sensitivity studies to identify key parameters contributing to the probability of pipe break (for B&W plants, indirect DEGB only).

(4) for both causes of DEGB, performed uncertainty studies to determine how uncertainties in input data affected the final estimated probability of pipe break (for B&W plants, indirect DEGB only).

We previously completed detailed generic evaluations of reactor coolant loop DEGB for plants with Westinghouse and C-E nuclear steam supply systems [5,6], and (at NRC request) a somewhat less rigorous evaluation — in that the probability of "direct" DEGB was not explicitly estimated — of reactor coolant loop DEGB for B&W nuclear steam supply systems [7]. In our evaluations of Westinghouse and C-E reactor coolant loop piping, 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. Each pilot study thus served to develop and "shake down" the assessment methodology that was later applied in a generic study for that vendor. In each generic study 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 reactor coolant piping manufactured by that vendor. Thus, the generic evaluations were "production" applications of the assessment methodology.

In our evaluation of B&W reactor coolant loop piping, we followed a simplified approach. Our evaluation of direct DEGB did not include explicit probabilistic fracture mechanics analyses. Instead, a comparison of relevant plant data, mainly reactor coolant loop stresses, for one representative plant with equivalent information for C-E and Westinghouse plants inferred that the probability of direct DEGB should be similarly low. Our indirect DEGB study comprised detailed evaluations of two reference plants only; for each of the remaining B&W plants, we estimated probabilities of indirect DEGB based on selected information characterizing component support strength.

In general, the results of our PWR evaluations indicated that the probability of a DEGB from either cause is extremely low under all plant conditions, about 1E-7 events per reactor-year from indirect causes and less than 1E-10 per reactor year from direct causes. We also found that thermal stresses dominated the probability of direct DEGB, and that earthquakes contributed only negligibly. These results suggested that reactor coolant loop DEGB — and with it such related issues as coupling of DEGB and SSE loads, asymmetric blowdown, and the need to install pipe whip restraints, warranted reevaluation as a design requirement for all PWR plants, regardless of NSSS supplier.

Based on the results of the probabilistic evaluations performed by LLNL, the NRC has revised the requirements of 10 CFR 50, General Design Criterion 4 (GDC-4), "Environmental and Missile Design Bases", to exclude dynamic effects associated with postulated pipe ruptures from the plant design basis. Under the proposed rule changes, the direct dynamic effects of pipe rupture are: missile generation; pipe whipping; pipe break reaction forces; influence of discharging fluids, including jet impingement forces, decompression waves within the ruptured pipe, and pressurization in vessel cavities, subcompartments, and compartments.

This rulemaking began with publication of an "interim" rule change applicable only to PWR reactor coolant loop piping [8], followed by a "broad-scope" rule change [9] applicable to any piping system, subject to acceptance criteria along the lines of those recommended by the NRC Piping Review Committee, Pipe Break Task Group, in Vol. 3 of NUREG-1061 [10]. In its evaluation, the Task Group recommended deterministic fracture mechanics analyses that would be required and other criteria that would have to be met before a piping system could be considered a "leak before break" candidate. The Task Group also recommended several instances in which leak-before-break should not be applied, such as when operating experience has indicated a particular susceptibility to failure from the effects of corrosion (e.g. stress corrosion cracking, erosion-corrosion), water hammer, or low- and high-cycle (i.e. thermal, mechanical) fatigue. The effect of stress corrosion cracking on piping reliability, particularly on BWR recirculation piping, is a key focal point of our BWR study.

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 of DEGB, namely:

- crack growth at welded joints driven by the combined effects of thermal, pressure, seismic, and other loads, coupled with other contributory effects such as stress corrosion.
- seismically-induced failure of piping and component supports, or of other equipment whose failure could, in turn, plausibly cause a reactor coolant pipe to break.

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

1.3 Scope of the Present Evaluation

In the present evaluation we extend to BWR reactor coolant piping the techniques developed and experience gained in our earlier evaluations of PWR reactor coolant loops. Although the overall objectives and approach of the BWR study are generally the same, evaluation of BWR reactor coolant piping presents a special challenge in at least two important regards:

(1) the susceptibility of certain BWR stainless steels to stress corrosion cracking required that we develop of an advanced probabilistic model of corrosion phenomena. Stress corrosion is generally not perceived as a problem in PWR primary loop piping and was therefore not considered in our earlier evaluations.

(2) the greater complexity and flexibility of BWR reactor coolant piping compared to PWR primary loops required that conventional supports (e.g. snubbers, spring hangers) for piping and light loop components be incorporated in the evaluation.

The present evaluation is based on a single Mark I BWR plant, the Brunswick Steam Electric Plant operated by the Carolina Power and Light Company. The decision to focus our efforts on Mark I plants was based on the historical susceptibility of their recirculation piping to stress corrosion cracking. Probabilities of DEGB due to both "direct" causes (crack growth at welded joints) and to "indirect" causes (seismically-induced support failure) were estimated for the following reactor coolant piping systems:

- recirculation loop piping. For the Brunswick recirculation loops we estimate the probability of direct DEGB due to crack growth under normal and postulated accident loads, both with and without the effects of intergranular stress corrosion cracking. Two loop configurations, the original and a proposed replacement, are considered. The IGSCC evaluation also considers two different pipe materials, the original and an IGSCC-resistant replacement.

  The "indirect DEGB" evaluation estimates the probability of pipe break caused by the seismically-induced failure of supports. Both "heavy component" supports, in particular the reactor support structure, and conventional pipe supports such as spring hangers and hydraulic snubbers are considered.

- main steam line piping. For the Brunswick main steam lines we estimate the probability of direct DEGB due to crack growth under normal and postulated accident loads; effects of corrosion are not considered. The evaluation considers one of the four lines. The "indirect DEGB" evaluation considers the probability of pipe break due to RPV support structure failure; an evaluation of "intermediate" support failure, equivalent to that for the recirculation piping, is not included.

- main feedwater piping. For the Brunswick main feedwater lines we estimate the probability of direct DEGB due to crack growth under normal and postulated accident loads; effects of corrosion are not considered. The evaluation considers one of the two lines. The "indirect DEGB" evaluation considers the probability of pipe break due to RPV support structure failure; an evaluation of "intermediate" support failure, equivalent to that for the recirculation piping, is not included.

With the exception of our IGSCC evaluation, all of these 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. The results of the IGSCC evaluation are presented on a "best estimate" basis with only limited attention given to questions of sensitivity and uncertainty. Unexpectedly high computer time requirements prevented us from performing detailed studies of uncertainty within practical resource constraints.

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. 1.2, 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 [11]. 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. 1.2), 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 thus more realistically weight the relative effects of frequent vs infrequent load events on overall reliability. Uncertainties due to lack 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 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.

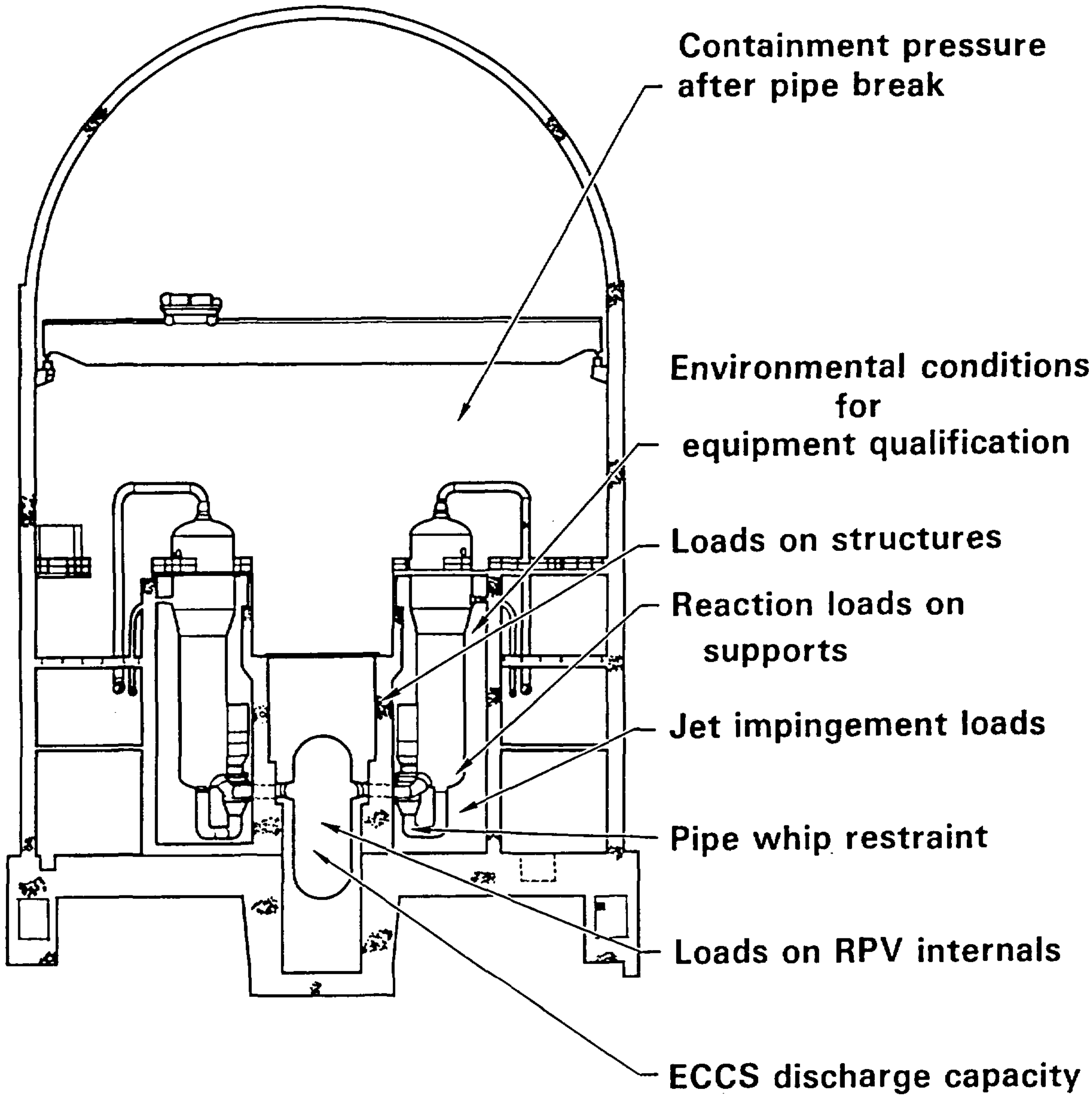

Figure 1.1. Selected aspects of plant design potentially affected by the DEGB design requirement.

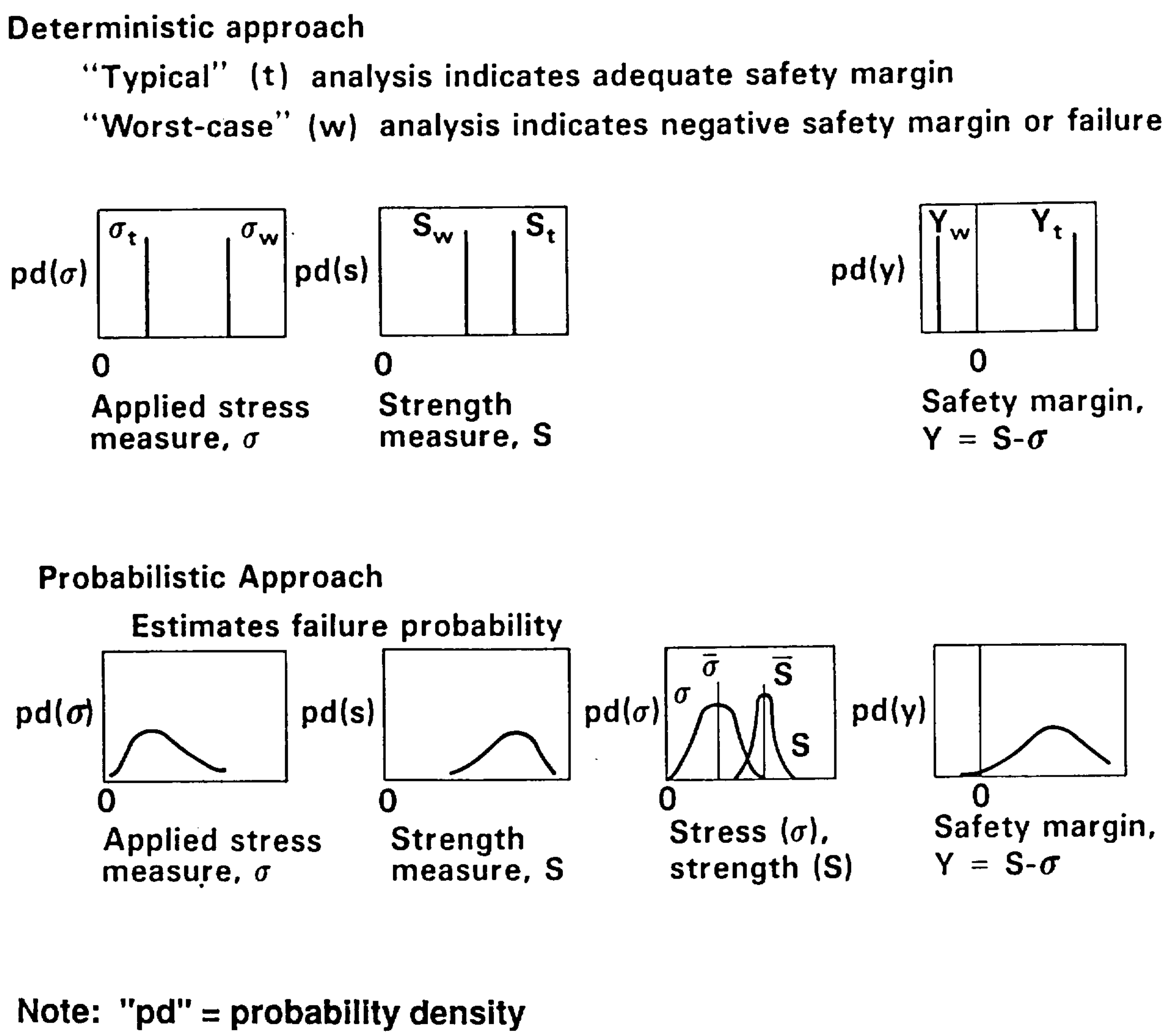

Figure 1.2. Comparison of probabilistic and deterministic approaches for assessing component adequacy under postulated load conditions. In the probabilistic representation, failure is possible only under conditions represented by the shaded region.

2. GENERAL PLANT DESCRIPTION

2.1 General Discussion

All operating commercial boiling water reactors in the United States are supplied by the General Electric Company (GE). Except for early prototype designs, GE reactors have evolved through six generations: the BWR-1, a natural circulation reactor; the BWR-2, the first to use forced coolant circulation as a means of controlling reactor thermal output; and the BWR-3 through BWR-6, a series of generally similar reactors using hydraulic "jet pumps" instead of electrically-driven mechanical pumps for coolant circulation. The jet pumps, which are located in the downcomer between the RPV wall and the core shroud, are venturis through which all reactor coolant must pass before entering the reactor core. The jet pumps themselves have no moving parts; instead, a small amount of reactor coolant is drawn from the RPV lower plenum and, by means of external electrical pumps, is injected into the upstream end of the pumps. This coolant "recirculation" induces a venturi effect which controls the rate at which the entire coolant inventory flows through the reactor core. By adjusting the rate of coolant recirculation, the reactor operator can closely control the thermal output of the reactor; consequently, BWR nuclear power plants tend to have better electrical load-following characteristics than their PWR counterparts.

Containment designs for these reactors have similarly evolved through three generations "Mark I" through "Mark III". Although there is no unique relationship between reactor model and containment design, the Mark I containment has historically been associated with the BWR-3 and BWR-4 reactor models, the Mark II containment with the BWR-5, and the Mark III with the BWR-6 reactor. The Mark III/BWR-6 combination, incidentally, represented the first major effort to develop a BWR "standard plant" design.

At NRC direction, we focussed our BWR evaluation on plants categorized by the Mark I containment design. Except for early BWR prototypes (e.g., Humboldt Bay), the Mark I containment was the first developed for the General Electric series of production boiling water reactors. A typical Mark I containment (Fig. 2.1) comprises a steel drywell, shaped like an inverted light bulb, connected to a toroidal steel wetwell filled approximately half-full with water (the "pressure suppression pool"). In the event of a pipe rupture inside of the drywell, steam would flow through large vent pipes into a ring header inside the torus, and finally discharged into the suppression pool via downcomers attached to the ring header. The resultant condensation of the steam would keep pressure loads on the containment within allowable limits. Nuclear steam supply systems paired with a Mark I containment are typically designed around either a BWR-3 or BWR-4 reactor, the first GE production models to utilize jet pumps for reactor coolant recirculation.

We selected the Brunswick Steam Electric Plant operated by the Carolina Power and Light Company (CP&L) as the case study for our evaluation. Located near Wilmington, North Carolina, Brunswick has two nominally identical units, each rated at 790 MWe, that have been in commercial operation since 1975 and 1977. The Brunswick containment (see Fig. 2.2) is atypical in that the drywell is reinforced concrete (rather than steel) and is not light-bulb shaped, having instead the cylinder-cone shape more characteristic of the later Mark II containment; Brunswick does, however, retain the toroidal suppression pool characteristic of the Mark I, as well as the BWR-4 reactor model.

2.2 Reactor Coolant Piping

We focussed our investigation on three major piping systems of the Brunswick plant: the recirculation loops, the primary (or main) steam lines, and the main feedwater lines.

Two recirculation piping systems at the Brunswick Plant are studied in this report: an existing system and a proposed replacement system. The existing system is made of Type 304 stainless steel, which was found to be susceptible to intergranular stress corrosion cracking (IGSCC) in many BWR plants. It was the intention of the replacement system to solve this problem by using the less IGSCC-susceptible Type 316NG stainless steel (ASME SA-358 Class 1 Nuclear Grade).

2.2.1 Reactor Recirculation System (Existing)

The existing recirculation piping system of the Brunswick Plant comprises two loops linked together at the header by a pair of equalizer valves. Fig. 2.3 shows the plan and elevation views of the recirculation system. These two loops (Loops A and B) are the mirror image of each other on two sides of the reactor pressure vessel except that a shutdown supply line of the residual heat removal (RHR) system is connected to the suction line of Loop B. The 24-inch shutdown return branches of the RHR system are connected to the discharge lines just below the header.

For each loop, the coolant flows out of the reactor vessel via a 28-inch diameter suction pipe and flows into a 28-inch discharge line due to the action of a recirculation pump between these two pipes. A suction valve is located on one side of the pump and a discharge valve on the other. A 22-inch header downstream from the 28-inch discharge line distributes the coolant to five 12-inch risers, which return the coolant to the reactor vessel. The header and the risers are part of the discharge piping. However, for convenience, we will call only the 28-inch piping downstream from the recirculation pump as the discharge line in this study. A 4-inch diameter bypass line with a bypass valve is connected to the discharge line on either side of the discharge valve. There are 51 circumferential welds in the piping. Table 2.1 shows the dimensions, the material types, and the number of welds for each pipe section. Only the Loop B elevation is shown in this figure. The pipe material is SA-240 Type 304 stainless steel except the bypass lines, which are SA-376 Type 304 stainless steel.

The suspension system of recirculation Loop B shown in Fig. 2.4 includes four variable spring hangers for the pipes and three constant support hangers for the pump. To provide restraint against postulated earthquake loads yet allow thermally-induced motion during normal operation, the loop has ten seismic snubbers; these are designated as SSB1 to SSB6, two SSB9s, and two SSB12s. The two SSB9s are located on the suction line just below the tee connection to the shutdown supply line of RHR system. The SSB12 snubbers support the discharge piping under the branch tee to the shutdown return line. The recirculation pump is supported by SSB1 through SSB3 while the pump motor, located on top of the pump, is supported by SSB4 through SSB6. Loop A has the same snubber arrangement except that one SSA10 replaces the two SSB9s of Loop B and is at a much lower elevation.

2.2.2 Reactor Recirculation System (Replacement)

The basic layout of the replacement recirculation loops closely resembles the existing system. Loop A corresponds to the existing Loop B, and Loop B corresponds to the existing Loop A. The replacement piping system is made of Type 316NG stainless steel (SA-358 Class 1, Nuclear Grade) and is structurally simpler than the existing system, having fewer weld joints (30 per loop compared to 51). It also eliminates the bypass lines which had been observed in several BWR plants to be particularly susceptible to stress corrosion cracking. The replacement system also has no equalizer valves; consequently, the two loops are structurally independent of each other. Figure 2.4 shows the plan and elevation views of the system. The dimensions, material types, and the weld information are presented in Table 2.2.

2.2.3 Main Steam Lines

The main steam or primary steam piping system consists of four 24-inch diameter carbon steel pipes designated as Lines A, B, C, and D. Line A and Line B are on one side of the reactor pressure vessel while Lines C and D are on the other side as shown in Fig. 2.5. These two groups are nearly the mirror images of each other about a vertical plane through the center of the reactor vessel. Line A corresponds to Line D and Line C corresponds to Line B. The material type is A106 Grade B. Each primary steam line originates from the reactor vessel upper cylindrical shell. Lines A and D have two safety-relief valves each, while Lines B and C have four. These safety-relief valves are provided with discharge piping to a pressure suppression chamber called the torus.

Downstream from the steam line header for the safety-relief valves, each of the primary steam lines has an isolation valve before passing through the drywell wall via a penetration assembly, which consists of head fittings, guard pipe and bellow to protect the integrity of the containment. In this study, we evaluate welds of the primary steam line inside the drywell and the welds before the first isolation valve outside the drywell. Therefore, no description of the primary steam lines beyond the drywell is given here. The pipe dimensions, material type and the weld numbers are presented in Table 2.3.

2.2.4 Feedwater Lines

Figure 2.6 shows the general arrangement of the Brunswick main feedwater system. It consists of two branches designated Loops A and B. These two branches are the mirror images of each other about a vertical plane passing through the axis of the reactor pressure vessel. There is no structural connection between these two branches. Each branch has an 18-inch diameter feedwater line, which penetrates the drywell wall. It has an isolation valve on either sides of the drywell wall and a penetration assembly to maintain containment integrity. The 18-inch line splits out into two 12-inch lines which connect to the upper cylindrical shell of the reactor vessel. The relevant information about the system is presented in Table 2.3 along with the information about the primary steam lines.

2.3 Reactor Vessel Support System

In our evaluations of PWR reactor coolant piping, we identified various components whose failure could plausibly result in a reactor coolant loop DEGB. These included reactor pressure vessel supports, steam generator supports and supports for the reactor coolant pumps. A BWR plant, of course, has no steam generator; therefore, our BWR study focussed mainly on the components of the reactor vessel support system. These components include the following (see Fig. 2.2):

- the primary containment structure, or "drywell"; note that possible failure of the reactor containment building was not within the scope of this study.
- the concrete RPV pedestal at the bottom of the drywell.
- the steel lower support structure for the RPV.
- the sacrificial shield wall.
- the "star truss" stabilizer which pins the top of the sacrificial shield wall to the containment building structure (see Fig. 2.7).
- the RPV stabilizer, a strut-type support assembly designed to transfer horizontal accident loads (either earthquake loads or pipe break loads) from the RPV to the shield wall.

We also identified as "critical" components the supports for the recirculation loop piping and recirculation pumps discussed in the previous section. These supports were considered in a separate evaluation as described in Section 5 of this report.

Table 2.1. Pipe properties of the existing recirculation loops.

	Suction	Discharge	Header	Riser	Bypass
Nominal size (in)	28	28	22	12	4
Outside diam (in)	28.169	28.519	22.003	12.706	4.500
Wall thickness (in)	1.151	1.326	1.038	0.631	0.337
Material type	SA-240 Type 304 SS	SA-240 Type 304 SS	SA-240 Type 304 SS	SA-240 Type 304 SS	SA-376 Type 304 SS
Welds (per loop)	10	6	5	20	10

Table 2.2. Pipe properties of the replacement recirculation loops.

	Suction	Discharge	Header	Riser	Bypass
Nominal size (in)	28	28	22	12	n/a
Outside diam (in)	28.000	28.000	22.000	12.750	n/a
Wall thickness (in)	1.209	1.390	1.750	0.688	n/a
Material type	SA-358 Type 316 SS	SA-358 Type 316 SS	SA-358 Type 316 SS	SA-358 Type 316 SS	n/a
Welds (per loop)	11	5	2	12	n/a

Table 2.3. Pipe properties of the primary steam lines and the main feedwater lines.

	Main Steam*	Feedwater	
Nominal size (in)	24	18	12
Outside diam (in)	24.000	18.000	12.750
Wall thickness (in)	1.218	1.375	0.843
Material type	SA-106 Type B Seamless	SA-333 Grade 6	SA-333 Grade 6
Welds (per line)	16	13	16

* Branch "A"

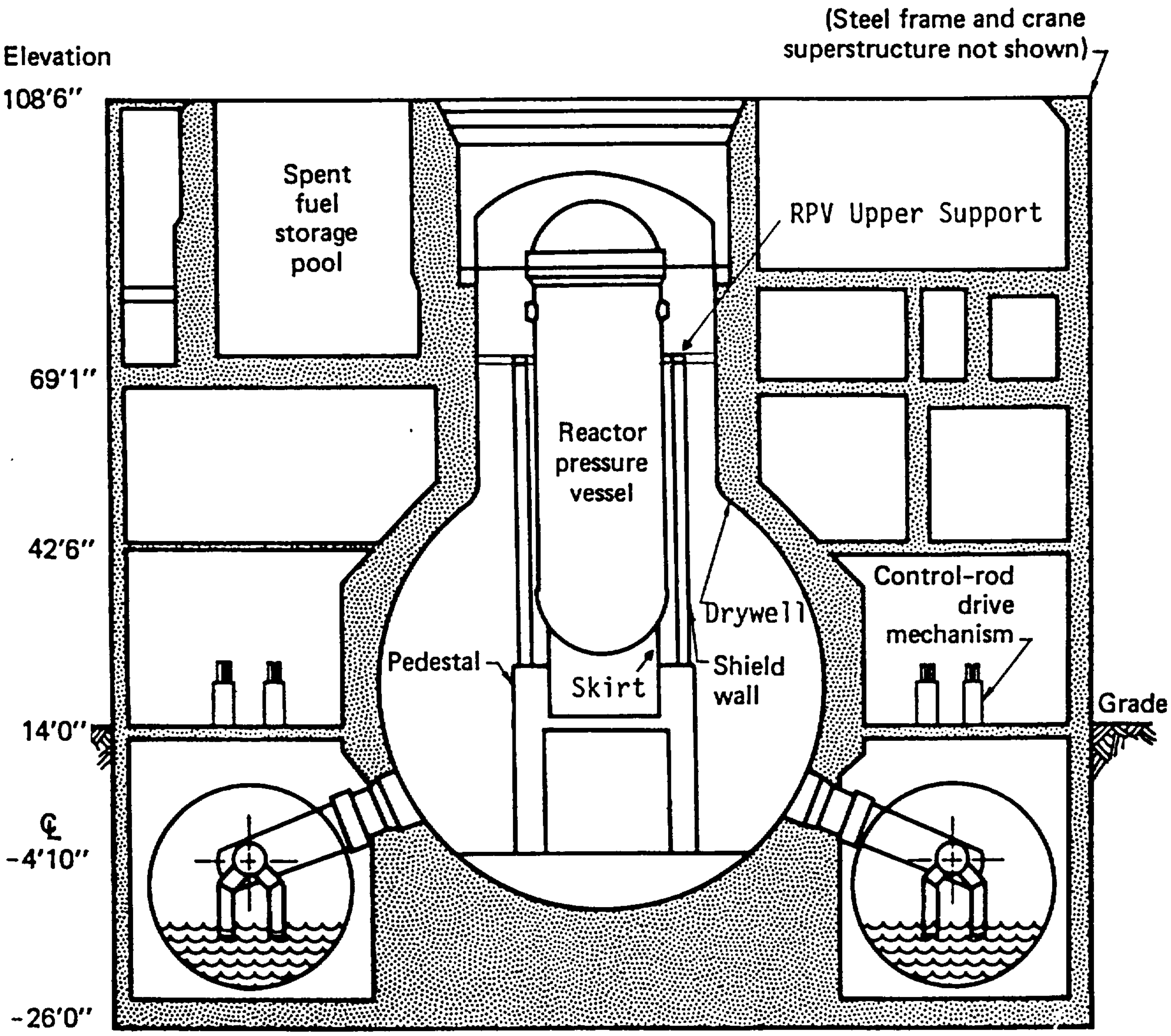

Figure 2.1. Typical Mark I containment. Note the characteristic "inverted light bulb" shape of the drywell.

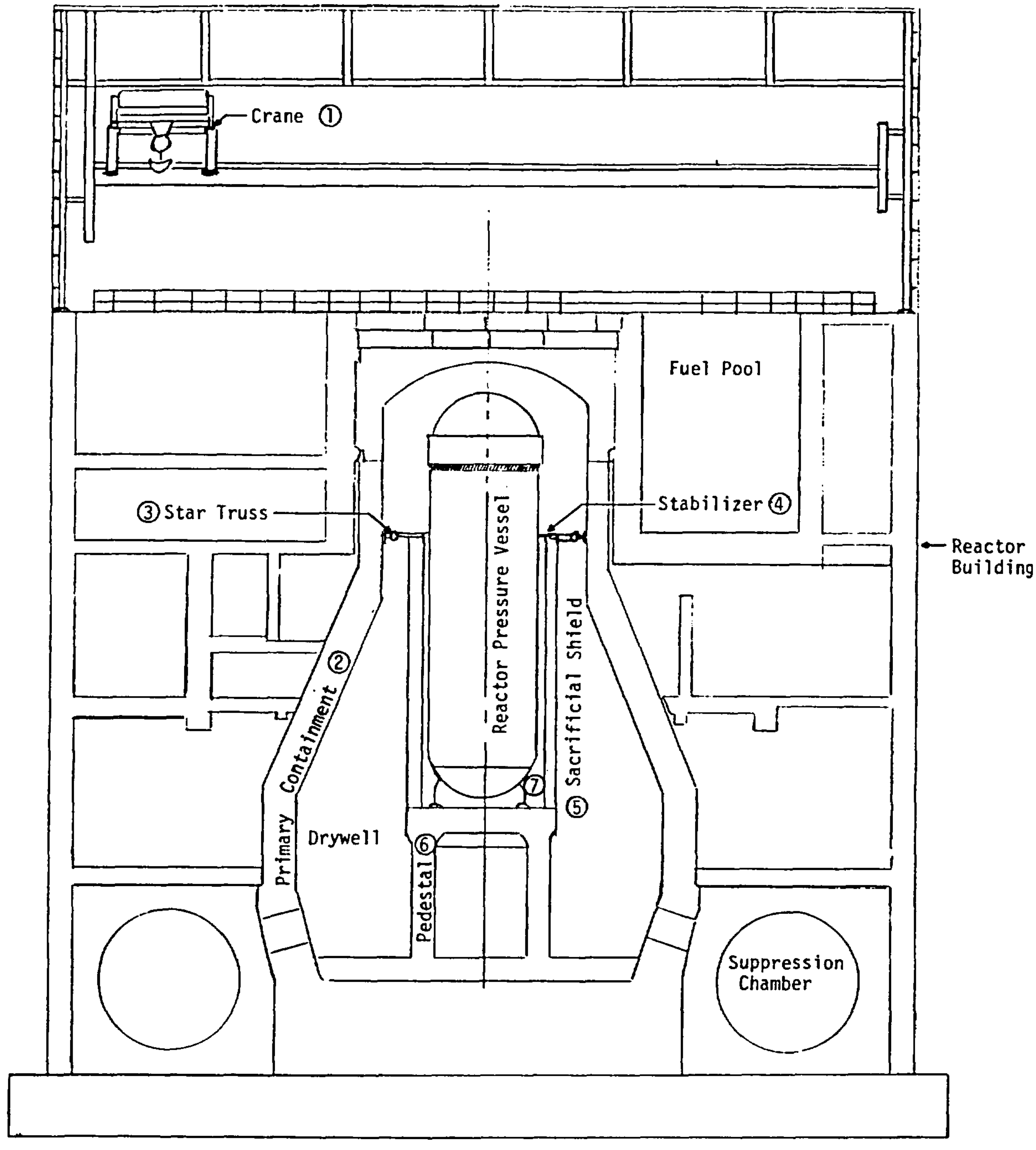

Figure 2.2. Brunswick containment structure and reactor vessel support system.

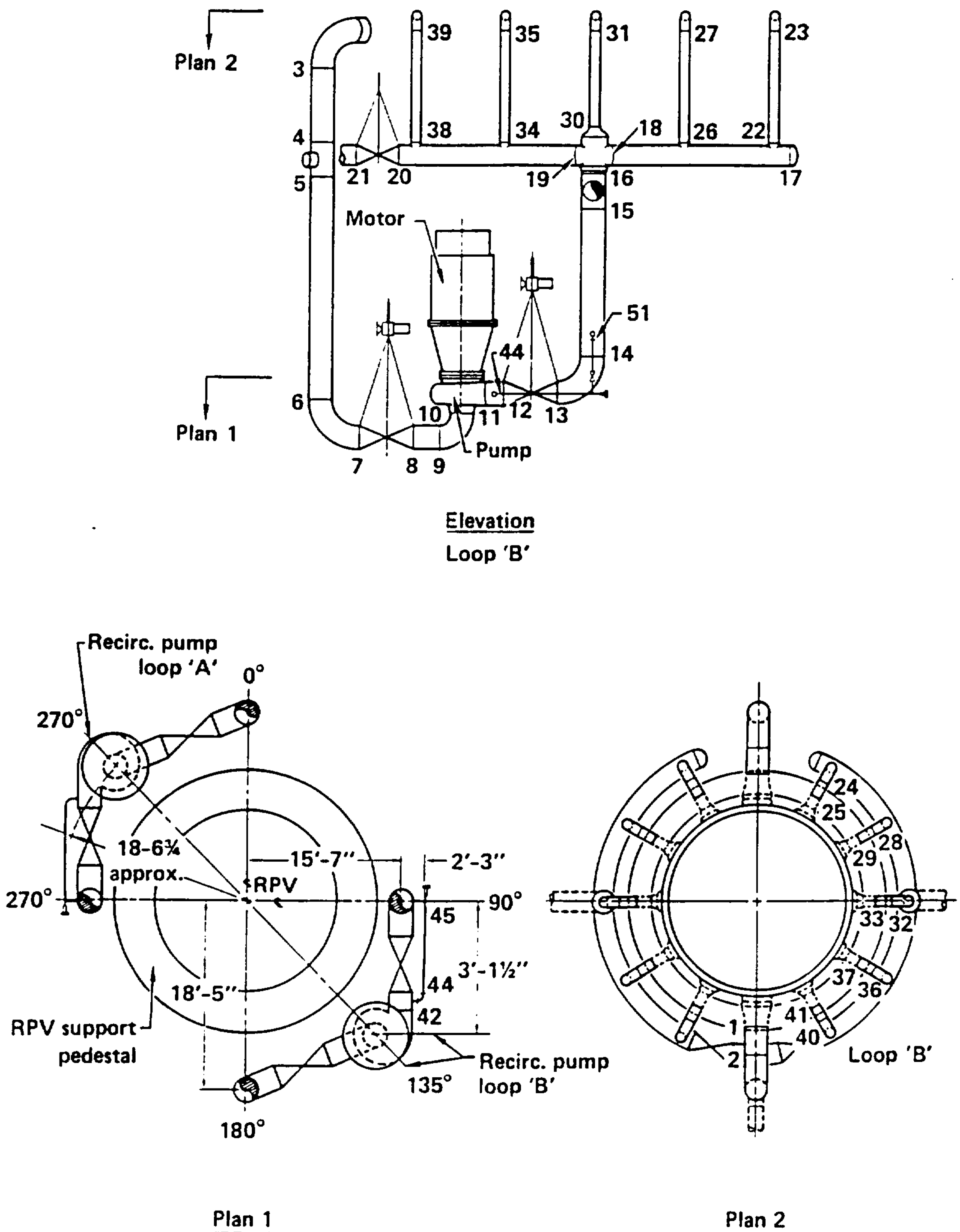

Figure 2.3. Existing Brunswick recirculation loops.

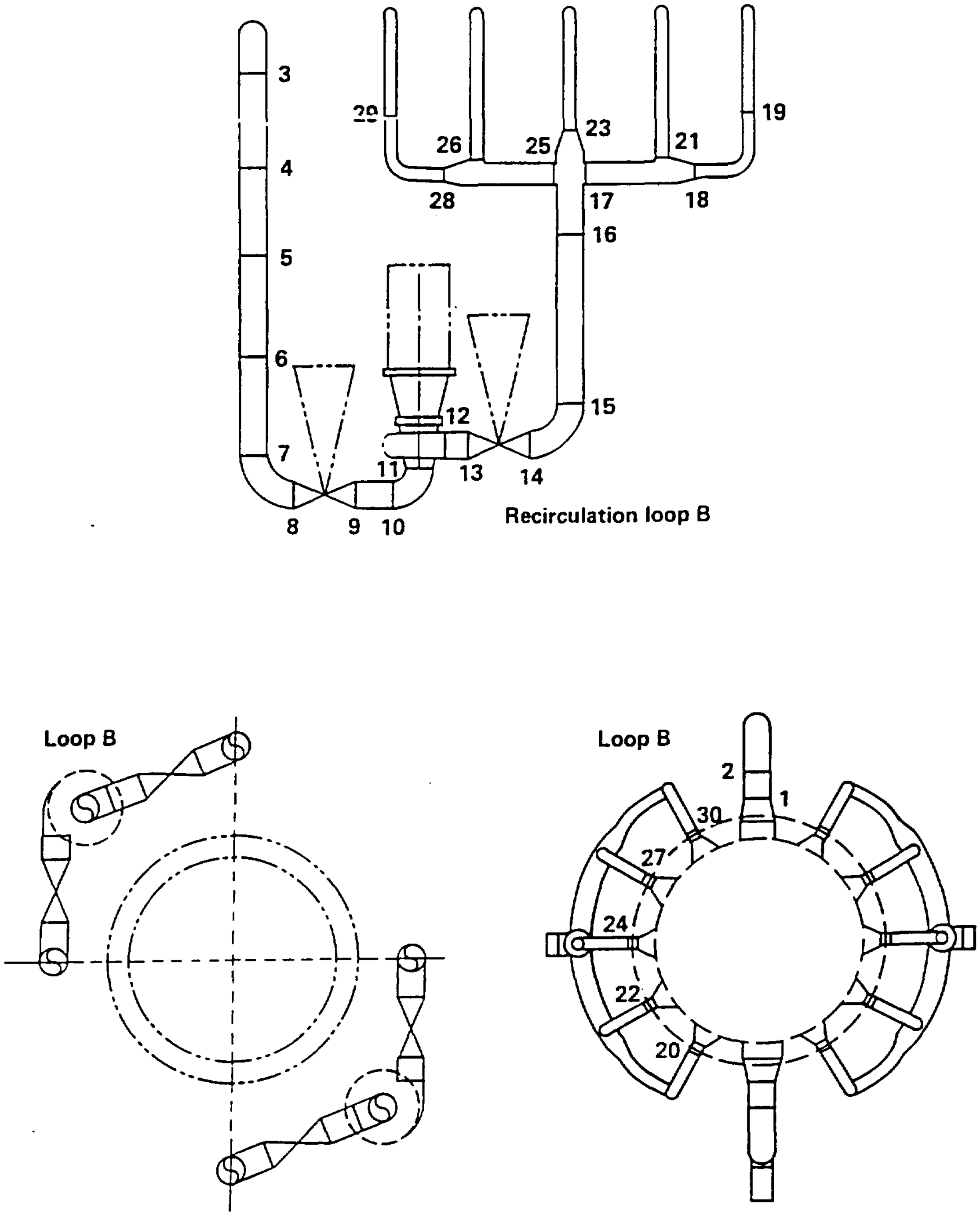

Figure 2.4. Layout of Brunswick replacement reactor recirculation system.

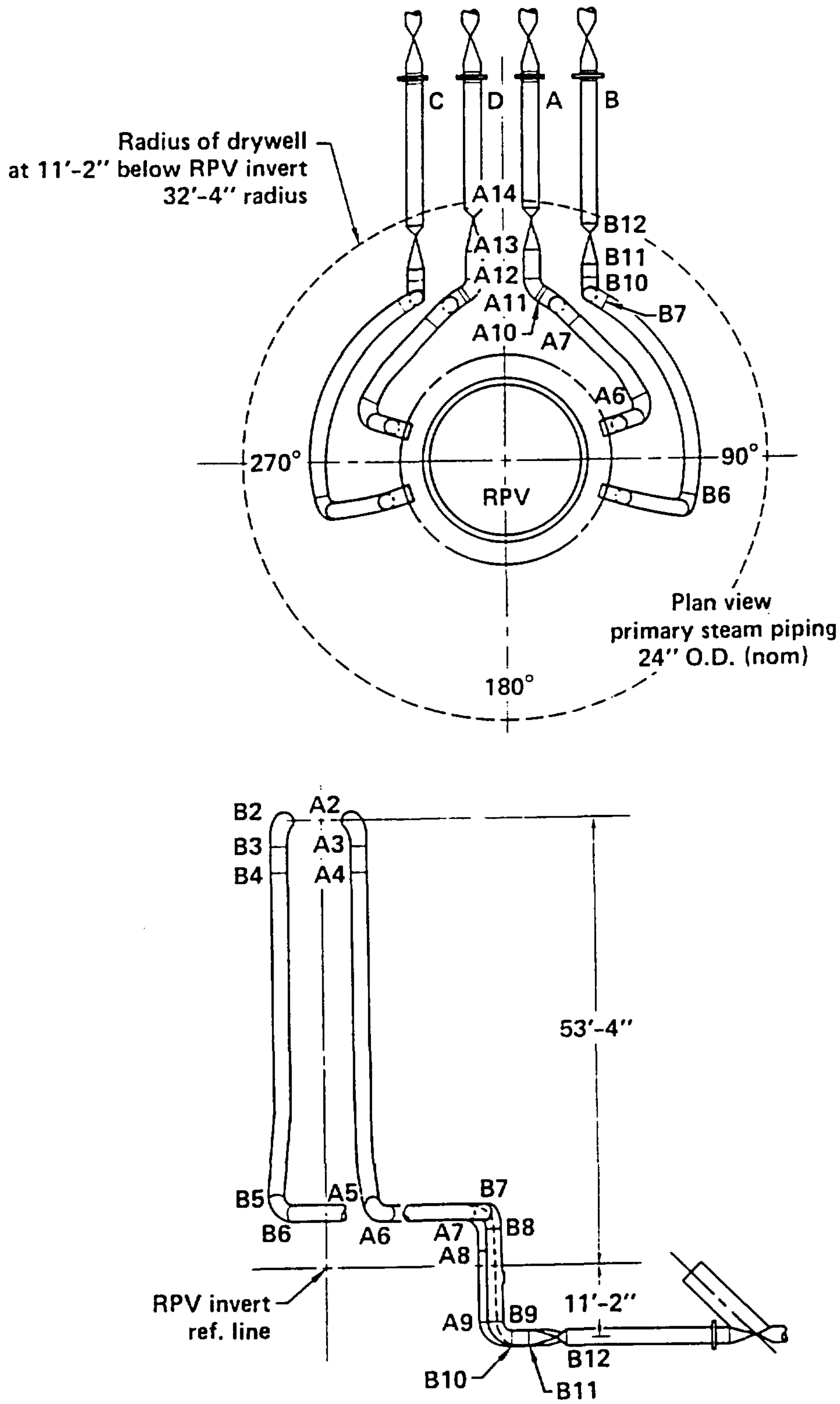

Figure 2.5. Brunswick main steam lines.

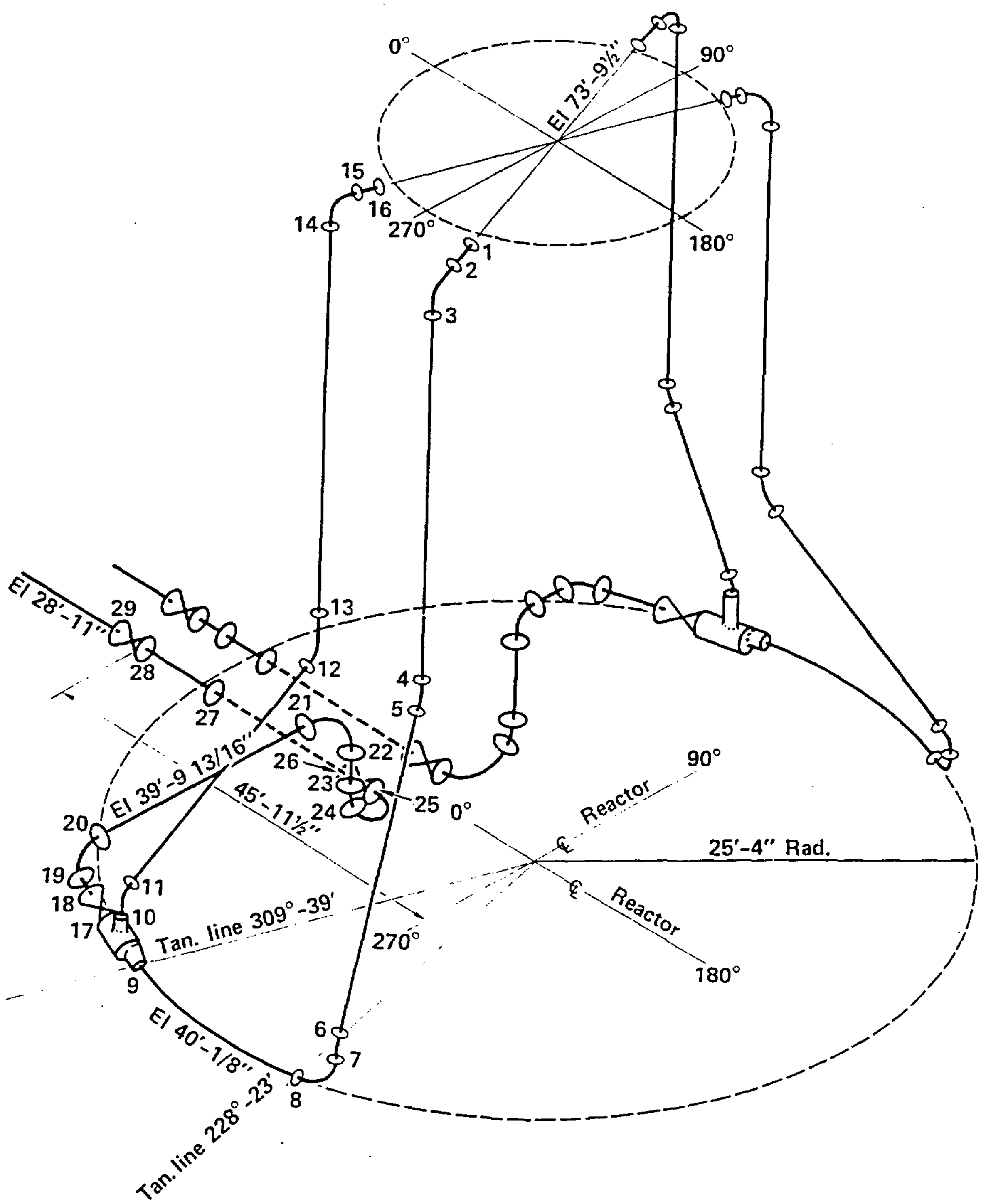

Figure 2.6. Brunswick main feedwater lines.

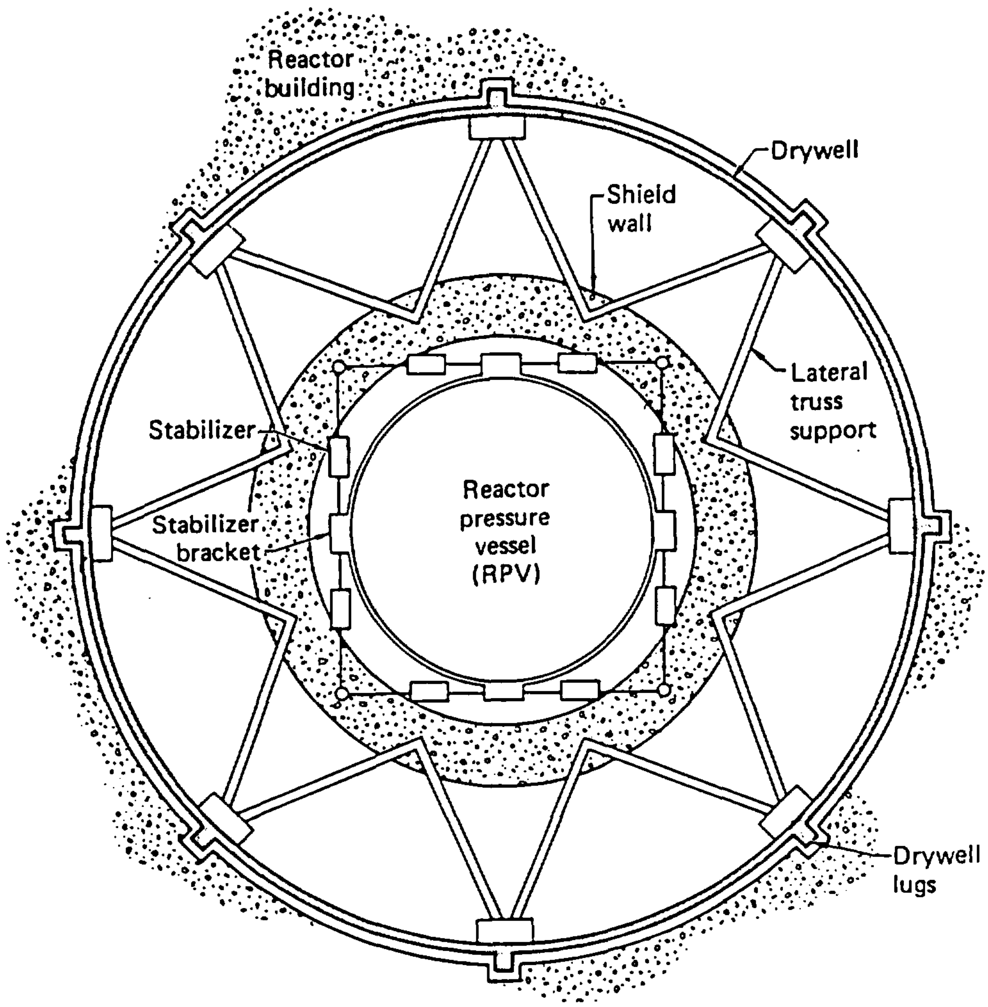

Figure 2.7. Schematic diagram of the drywell truss system.

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.

As part of our pilot study of reactor coolant loop leak and DEGB in Westinghouse PWR plants [4], we developed a probabilistic fracture mechanics approach to explicitly estimate the probability of a direct DEGB in austenitic (i.e. stainless steel) piping. We later expanded the model to include carbon steels and similarly applied it to estimate the probability of a direct DEGB in the reactor coolant loop piping of Combustion Engineering PWR plants. Further enhancements, in particular an advanced model of stress corrosion cracking, allowed us to perform the BWR evaluation described in this report.

The basic approach, described elsewhere in detail [4,6,12], not only consolidates the separate effects of many (and often deterministically unrelated) factors influencing DEGB probability, but also allows us to account for the randomness of load events and other parameters associated with plant operation. Figure 3.1 illustrates schematically how probabilities of leak and DEGB are estimated. The left column shows the analytical procedure, the right the required input information and the various simulation models used in 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 as a function of time the conditional probabilities of leak and break at individual weld joints, given (1) that a crack exists at that joint, (2) that the plant experiences various loading conditions at any time, and (3) that an earthquake of a specific intensity occurs at a specific time. The second part estimates the absolute (or non-conditional) probabilities of leak and break for the entire piping system by convolving (1) the conditional leak and break probabilities at each 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 occurrence rate ("seismic hazard"). Thus, a complete evaluation of a piping system involves several PRAISE runs (i.e. one for each weld), the results of which are then consolidated in a "systems analysis" performed by a separate post-processor.

Except where noted otherwise, we typically present failure probabilities in terms of events per reactor-year (in this study, the term "plant-year" is sometimes used equivalently). It is important to point out that the system analysis actually yields the cumulative failure probability over the entire duration of plant life (here taken as 40 years), from which the annual failure probability is estimated by assuming that system failure probabilities are uniform throughout plant life.

It is also important to emphasize that this probabilistic fracture mechanics approach is *not* a probabilistic risk assessment utilizing event tree and fault tree analysis. Rather, the procedure incorporates deterministic (either empirical or analytic) models into a probabilistic "framework". This framework 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 features of the analysis in greater detail.

3.1.1 Failure Probability of a Weld Joint

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

Each replication of the simulation — and a typical PRAISE simulation may include 10,000 or more — begins with a pre-existing flaw having length and aspect ratio randomly selected from a two-dimensional sampling space. The conditional probability that a flaw exists having this specific size is in turn related by appropriate distributions. If only pre-existing cracks are considered (i.e. no additional cracks initiate during plant operation), "stratified sampling" can be applied to select initial crack samples from only those sizes that can potentially lead to pipe break. This technique allows us to reliably estimate very low failure probabilities (less than one in a million) from only a few thousand replications of the Monte Carlo simulation.

Fatigue crack growth is then calculated using a Paris growth model, to which are applied stresses associated with normal operating conditions and postulated seismic events. The influence of such factors as non-destructive examination (NDE) and leak detection is also considered through the inclusion of appropriate statistical distributions (e.g., probability of crack non-detection as a function of crack size).

Fatigue crack growth takes into account the cyclic stress history of various thermal transients and postulated seismic events. Leak occurs when (and if) a crack grows through the pipe wall, break when a failure criterion based on either net section stress or tearing modulus instability (depending on material characteristics and pipe geometry) is exceeded. The stress state of the piping system 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 our past evaluations, we treated the seismic events as stochastic and assumed to to be describable by a Poisson process in calculating the system failure probability. Other plant transients were 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 do not conveniently lend themselves to more suitable spacing. The frequencies of thermal transient events used in past analyses were based on design postulations and were considered to be conservative.

The pre-service inspection (see Fig. 3.1) is performed once before the plant begins operation, as is typical for real plants. Although we can also model in-service inspections, these were not included in our evaluations because ISI programs vary from plant to plant and thus cannot be generically modeled with any reasonable confidence. The relationship between DEGB probability and ISI interval was studied in detail in a separate evaluation [13], which indicated that the influence of ISI is relatively minor compared to that of the pre-service inspection. In any case, disregarding ISI adds conservatism to the results.

The effect that earthquakes of specific intensity have on the failure probability at each weld joint at specific times during plant life is also assessed. First, the probability of failure without earthquakes is estimated. Earthquakes of specified intensity are then imposed on normal operating conditions, usually in terms of peak ground accelerations. The increase in failure probability after the earthquake is added therefore represents the contribution of the seismic event to the failure probability. This process is repeated for a wide range of earthquake intensities.

As mentioned earlier, 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 a piping system, for example, one reactor coolant loop. In our PWR evaluations, all loops in a given plant were assumed geometrically identical; therefore, the failure probabilities at corresponding weld joints were assumed identical.

3.1.2 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 (the probability that any crack, regardless of size, exists in a given volume of weld material) and the seismic hazard. As noted earlier, this "systems analysis" is not part of the PRAISE code, but is instead performed using a separate post-processing code.

The probability of pipe failure is potentially affected by both the intensity and the occurrence rate of earthquakes. In our past evaluations, earthquake intensities expressed in terms of peak free field ground acceleration (PGA) ranged from zero to five times that of the safe shutdown earthquake (SSE). We also defined a threshold PGA value below which no crack growth is assumed to occur. The value of this threshold acceleration is subjective; however, sensitivity studies that we performed indicated that the estimated system failure probability is not significantly affected by 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 described by a set of seismic hazard curves which plot exceedance probability as a function of peak ground acceleration. Our evaluations of PWR plants east of the Rocky Mountains were based on generic seismic hazard curves for this region that were developed as part of our investigations of indirect DEGB (see Fig. 3.2). West coast plants were evaluated using site-specific seismic hazard information; the small number of plant sites and widely varying seismic conditions did not allow a generic characterization of seismic hazard to be made without assigning a large degree of uncertainty.

In evaluating the probability of a direct DEGB, we considered three events causing pipe break:

Event 1: break and an earthquake occur simultaneously, in other words, the earthquake *causes* pipe break.

Event 2: the pipe breaks independently of any earthquakes occurring during plant life.

Event 3: the pipe breaks even though no earthquake at all occurs during plant life.

Probabilities of direct DEGB were estimated independently for each case and then combined into an overall probability that pipe break occurs sometime during plant life. We consistently found in our PWR studies that the probability of the first case — earthquake and pipe break occur simultaneously — was typically one to three orders of magnitude less than that of pipe break occurring independently of an earthquake. These results implied that direct DEGB and a safe shutdown earthquake can be considered as independent random events whose probability of simultaneous occurrence is vanishingly small.

3.1.3 Uncertainty Analyses

Two types of variability, or uncertainty, are associated with each of the parameters considered in our DEGB evaluations. One type, random uncertainty, represents the inherent physical variation or randomness of the parameters. Modeling uncertainty, the other type, accounts for any lack of knowledge or detailed information about the parameters that may be necessary to describe them precisely.

To illustrate these two types of uncertainties, consider the flow stress (the average of yield and ultimate stresses) of a particular material as an example. Because of the physical variability of materials and structures, flow stress is inherently variable. The variability, or "randomness" of flow stress can be described, for example, by a normal probability distribution characterized by a mean value and a standard deviation. Estimates of the mean and standard deviation for a specific material can be derived from laboratory tests. 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. We might also be uncertain about how well the normal distribution describes the variability of flow stress; perhaps a log-normal distribution would be better. This uncertainty would also contribute to the overall modeling uncertainty associated with flow stress.

There are many sources of modeling uncertainty associated with estimating the probability of DEGB in piping. Some additional examples include uncertainties associated with:

- the selection of methods for modeling soil-structure interaction, such as the finite-element approach and the impedance approach.
- the selection of methods for modeling structural response, such as response spectrum vs time-history analysis, two- or three-dimensional analysis, or 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, such as sampling error in the Monte Carlo simulation technique.
- the inherent randomness in physical parameters other than flow stress.

A deterministic value will often represent a parameter adequately if the variation is negligible; otherwise, a suitable distribution (including uncertainty bounds or "confidence" limits) is required. In our evaluations, we developed distributions to describe the inherent randomness in many parameters. Some distributions were generated from plant-specific data supplied by the NSSS vendors, others were based on generic information. We also quantified modeling uncertainties for the five key parameters that sensitivity studies had shown were most important to the fracture mechanics evaluation: initial crack depth, initial crack length (aspect ratio), thermal stress, seismic stress, and seismic hazard. Because random uncertainties of input parameters contribute to the value of pipe failure probability, they are intrinsic to the analytic process illustrated in Fig. 3.1. Modeling uncertainties were treated 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 manner, we developed a distribution about the "best estimate" probabilities of DEGB and leak (i.e. those based on the "best estimate" values of these five parameters).

Volume 2 of this report series and the final reports from our evaluations of Westinghouse and C-E reactor coolant loop DEGB [5,6] offer further details on how these uncertainty analyses were performed.

3.2 Probability of Direct DEGB in PWR Reactor Coolant Piping

The results of our Westinghouse and C-E evaluations indicated that the probability of a direct DEGB is very low for reactor coolant loop piping supplied by either vendor. 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 postulated seismic events. Other factors, such as the capability to detect cracks by non-destructive examination and the capability to detect leaks, were also considered. In particular, the results of these evaluations indicated that:

- the "best-estimate" probability of direct DEGB in reactor coolant loop piping is on the order of 1E-12 to 1E-11 per reactor-year for Westinghouse plants, and 1E-14 to 1E-12 per reactor-year for C-E plants. Based on extensive uncertainty analyses, a direct DEGB probability of 1E-10 per reactor-year appears to be a reasonable upper bound for either vendor, regardless of whether plants are located east of the Rocky Mountains or on the more seismically active west coast.

- similarly, a leak probability of about 4E-7 per reactor-year appears to be a reasonable upper bound for either vendor, regardless of location. The significantly higher probability of leak compared to that of DEGB supports the concept of "leak before break" in reactor coolant loop piping.

- the probability of direct DEGB is typically lower than that of indirect DEGB by several orders of magnitude, indicating that the seismically induced failure of heavy component supports is a much more plausible mode of failure than crack growth at individual weld joints.

The development of these failure probabilities considered all plant conditions, including earthquakes, and took into account both the magnitude and the frequency of plant loads. Our analyses showed, for example, that the probability of a direct DEGB (and the probability of leak as well) is only very weakly affected by an earthquake. We found instead that normal operating loads, in particular stresses caused by restraint of thermal expansion during system heatup and cooldown, were the dominant contributors to pipe failure.

We found these results to be highly consistent for both vendors despite significant differences between the two in reactor coolant loop configuration and piping materials. Largely on the basis of this general finding and the low probabilities of direct DEGB estimated in the earlier evaluations, the NRC requested that we perform no probabilistic fracture mechanics analyses of B&W reactor coolant loop piping. Rather than explicitly estimate leak and DEGB probabilities, we instead compiled for a representative plant the necessary input data for such an analysis and then reviewed this data for conformity with that used in our earlier evaluations. This review identified no substantial differences that would infer probabilities of direct DEGB in B&W reactor coolant loop piping significantly greater (i.e. by many orders of magnitude) than those estimated for Westinghouse and C-E reactor coolant loops.

3.3 Probability of Failure in BWR Reactor Coolant Piping

Using the probabilistic approach previously discussed and neglecting IGSCC, we estimated the probabilities of leak and DEGB in the recirculation, main steam, and main feedwater piping of the Brunswick BWR plant. We performed two types of analyses: a best-estimate analysis and an uncertainty analysis. The former yields a single point estimate of the probability of failure (i.e. leak or DEGB) based on the "best estimate" values of those parameters treated as random variables. The latter takes into account modeling uncertainty in addition to the randomness of the parameters, and thus provides an uncertainty distribution for the estimated leak and DEGB probabilities.

In this probabilistic piping reliability analysis, we considered the following specific piping systems:

- Loop B of the existing and the proposed replacement recirculation systems
- Branch A of the main steam system

- the feedwater line located in the third and fourth quadrants of the reactor cavity

The selection is arbitrary because, within a given system, each branch closely resembles any other. Therefore, the probabilistic results for one branch should be representative for its respective system.

Tables 3.1 and 3.2, respectively, provide estimated leak and DEGB probabilities for the piping systems considered. Note that these values are cumulative probabilities over the design lifetime of the plant, taken to be 40 years. Tables 3.1 and 3.2 also present the probability values associated with each of the three events that comprise the overall system failure probability. As discussed in Section 3.1.2, Event 1 represents the pipe failure due to earthquake, Events 2 and 3 pipe failure induced by causes other than earthquake.

It is important to note that while best-estimate models were used for many parameters, some other parameters were based on conservative assumptions. Therefore, the best-estimate analysis actually yields conservative results. Note also that the recirculation loop results do not reflect the influence of intergranular stress corrosion cracking; pipe failure due to IGSCC is discussed in Section 4 of this report. Stress corrosion was also disregarded in the evaluations of main steam and feedwater piping because the materials used are not susceptible to IGSCC effects.

The best-estimate lifetime system leak and DEGB probabilities for the existing Brunswick major coolant piping systems are rather low and fall within narrow ranges. The system leak probabilities vary between 2.4E-6 and 5.4E-5 over the life of the plant, or between about 6.0E-8 and 1.4E-6 per reactor-year. The DEGB probabilities similarly range from 1.0E-11 to 7.0E-11 over the lifetime of the plant, or about 2.5E-13 to 1.8E-12 per reactor-year. These results are similar to those estimated for PWR reactor coolant piping, both in absolute magnitude and in the differential (three or more orders of magnitude) between the probabilities of leak and break.

Note, however, that the seismically-induced DEGB probabilities (Event 1) for the existing and the replacement recirculation loops are higher than the probabilities due to other causes. This situation differs from our previous results for PWR reactor coolant loop piping, and from the results for other piping systems in this study.

This can be attributed to the fact that BWR recirculation loops typically have a relatively complex arrangement comprising piping of various lengths and diameters. In this regard, note that in either the existing or the replacement loop configurations, the probability of failure is dominated by the smallest piping section. As shown in Tables 3.3 and 3.4, pipe size has a significant influence not only on per-weld failure probability (which may vary by several orders of magnitude), but on the relative contribution of earthquakes to the probability of failure. In the existing recirculation loop considered, for example, the probability of seismically-induced DEGB in the bypass line (a 3-inch diameter pipe) is some 30 times higher than that due to other causes; this result reflects the high seismic stresses that would be induced in this line, particularly by earthquakes much larger than the safe shutdown earthquake.

To account for modeling uncertainty, we also placed distributions on seven parameters which we determined most significantly affected the probability of direct DEGB. Five of these -- initial crack depth, initial crack aspect ratio (i.e. crack length), thermal stresses, seismic stresses, and seismic hazard -- were similarly considered in our earlier PWR evaluations. In our BWR study we added two other parameters, the non-conditional crack existence probability (used in the system analysis), and the probability of crack detection by non-destructive in-service inspection.

Tables 3.5 and 3.6, respectively, show the probabilities of leak and DEGB corresponding to the 10th, 50th, and 90th-percentile values ("confidence limits") for each system considered. The best-estimate failure system probabilities are also presented in these tables for comparison. We used the Latin Hypercube sampling technique on a set of 20 samples to estimate the distributions of leak and DEGB probabilities due to modeling uncertainty.

Note that the distributions of the leak and DEGB probabilities are rather wide for each of these systems, ranging over several orders of magnitude. Note also that the best-estimate failure probabilities are not necessarily close to the medians of the uncertainty distributions. While some are rather close, there is a factor of 2.8 for the leak on the feedwater line, and a factor of 8.8 for the DEGB on the replacement recirculation loop.

Nevertheless, it is important to note that all of the estimated failure probabilities are low, even at the 90th-percentile confidence level. The highest lifetime probabilities at this level are 1.2E-3 and 5.0E-8 for the leak and the DEGB, respectively, or about 3.0E-5 and 1.3E-9 per reactor-year. Both of these probabilities are for the feedwater line, and are still comparable to our earlier PWR results despite the increased complexity of the BWR systems considered.

Volume 2 of this report series discusses in detail our direct DEGB evaluations of BWR recirculation, main steam, and feedwater piping, excluding the effects of stress corrosion cracking.

Table 3.1. Best-estimate leak probabilities of major coolant piping systems.

	Leak Probability[1]				
	Event 1	Event 2	Event 3	Event 2+3	Event 1+2+3
Recirculation Loop B (Existing)[2]	.453E-7	.332E-6	.678E-5	.711E-5	.715E-5
Recirculation Loop B (Replacement)[2]	.331E-6	.179E-5	.358E-4	.376E-4	.380E-4
Main steam line[3] (Branch A)	.870E-8	.145E-6	.223E-5	.238E-5	.239E-5
Feedwater line[3]	.238E-7	.333E-6	.507E-5	.540E-5	.543E-5

Notes:

1. Cumulative over 40-year plant lifetime. See Section 3.1.2 for event definitions.

2. Effects of stress corrosion cracking are not included in these recirculation loop analyses.

3. Effects of stress corrosion cracking routinely disregarded in evaluations of main steam and feedwater piping.

Table 3.2. Best-estimate DEGB probabilities of major coolant piping systems.

	DEGB Probability[1]				
	Event 1	Event 2	Event 3	Event 2+3	Event 1+2+3
Recirculation Loop B (Existing)[2]	.144E-09	.256E-12	.958E-11	.984E-11	.154E-09
Recirculation Loop B (Replacement)[2]	.805E-11	.584E-13	.219E-11	.225E-11	.103E-10
Main steam line[3] (Branch A)	.454E-12	.408E-11	.652E-10	.693E-10	.698E-10
Feedwater line[3]	.411E-11	.884E-12	.353E-10	.362E-10	.403E-10

Notes:

1. Cumulative over 40-year plant lifetime. See Section 3.1.2 for event definitions.
2. Effects of stress corrosion cracking are not included in these recirculation loop analyses.
3. Effects of stress corrosion cracking routinely disregarded in evaluations of main steam and feedwater piping.

Table 3.3. Lifetime leak probabilities of dominant welds within each section of the piping systems.

		Leak Probability[1,2]			
		Due to Earthquake		Due to Other Causes	
Recirculation Loop B (Existing)[3]	Bypass	#42	.19E-7	#51	.14E-5
	Riser	#30	.38E-8	#26	.42E-6
	Header	#19	.15E-9	#19	.53E-7
	Discharge	#16	.10E-9	#15	.33E-7
	Suction	#4	.12E-9	#4	.35E-7
Recirculation Loop B (Replacement)[3]	Riser	#23	.69E-7	#18	.38E-5
	Header	#17	.02E-9	#17	.85E-8
	Discharge	#12	.10E-9	#12	.19E-7
	Suction	#1	.06E-9	#10	.23E-7
Main steam line[4] (Branch A)		#5	.16E-8	#2	.21E-6
Feedwater line[4] (by diameter)	12-inch	#16	.55E-8	#2	.83E-6
	18-inch	#27	.12E-8	#26	.82E-7

Notes:

1. Cumulative over 40-year plant lifetime.
2. "#" denotes number of dominant weld.
3. Effects of stress corrosion cracking are not included in these recirculation loop analyses.
4. Effects of stress corrosion cracking routinely disregarded in evaluations of main steam and feedwater piping.

Table 3.4. Lifetime DEGB probabilities of dominant welds within each section of the piping systems.

		DEGB Probability[1,2]			
		Due to Earthquake		Due to Other Causes	
Recirculation Loop B (Existing)[3]	Bypass	#42	.14E-09	#45	.50E-11
	Riser	#30	.94E-12	#30	.25E-12
	Header	#19	.47E-15	#19	.67E-14
	Discharge	#16	.13E-15	#11	.85E-15
	Suction	#4	.14E-16	#1	.23E-36
Recirculation Loop B (Replacement)[3]	Riser	#24	.27E-11	#18	.18E-11
	Header	#17	.46E-15	#17	.23E-14
	Discharge	#12	.28E-15	#14	.57E-15
	Suction	#1	.22E-17	#1	.26E-37
Main steam line[4] (Branch A)		#5	.76E-13	#2	.97E-11
Feedwater line[4] (by diameter)	12-inch	#16	.99E-12	#16	.85E-11
	18-inch	#27	.64E-12	#27	.14E-11

Notes:

1. Cumulative over 40-year plant lifetime.

2. "#" denotes number of dominant weld.

3. Effects of stress corrosion cracking are not included in these recirculation loop analyses.

4. Effects of stress corrosion cracking routinely disregarded in evaluations of main steam and feedwater piping.

Table 3.5. Leak probabilities at the 10th, 50th, and 90th percentiles of the uncertainty distribution.

	Leak Probability[1]			
	10%	50%	90%[2]	Best Estimate
Recirculation Loop B[3] (Replacement)	6.4E-7	4.0E-5	4.0E-4	3.8E-5
Main steam line[4] (Branch A)	3.2E-7	3.7E-6	5.4E-4	2.4E-6
Feedwater line[4]	7.0E-7	1.9E-5	1.2E-3	5.4E-5

Notes:

1. Cumulative over 40-year plant lifetime.

2. A confidence limit of 90% implies that there is a 90% subjective probability ("confidence") that the probability of leak is less than the value indicated.

3. Effects of stress corrosion cracking are not included in these recirculation loop analyses.

4. Effects of stress corrosion cracking routinely disregarded in evaluations of main steam and feedwater piping.

Table 3.6. DEGB probabilities at the 10th, 50th, and 90th percentiles of the uncertainty distribution.

	DEGB Probability[1]			
	10%	50%	90%[2]	Best Estimate
Recirculation Loop B[3] (Replacement)	6.0E-14	8.0E-12	1.0E-09	7.0E-11
Main steam line[4] (Branch A)	2.0E-13	1.2E-11	5.5E-09	1.0E-11
Feedwater line[4]	4.5E-13	6.0E-11	5.0E-08	4.0E-11

Notes:

1. Cumulative over 40-year plant lifetime.
2. A confidence limit of 90% implies that there is a 90% subjective probability ("confidence") that the probability of DEGB is less than the value indicated.
3. Effects of stress corrosion cracking are not included in these recirculation loop analyses.
4. Effects of stress corrosion cracking routinely disregarded in evaluations of main steam and feedwater piping.

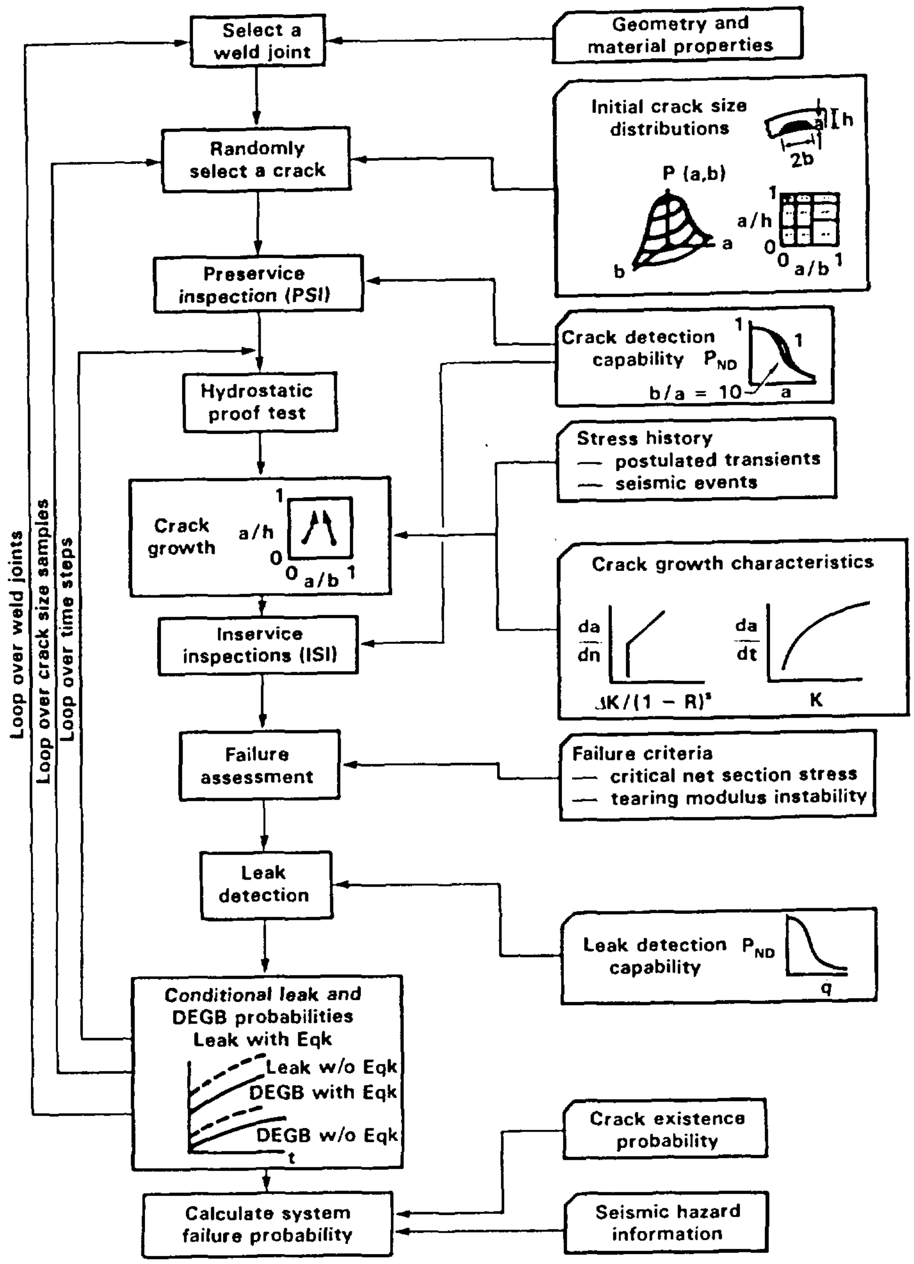

Figure 3.1. Flowchart of the probabilistic fracture mechanics model implemented in the PRAISE computer code.

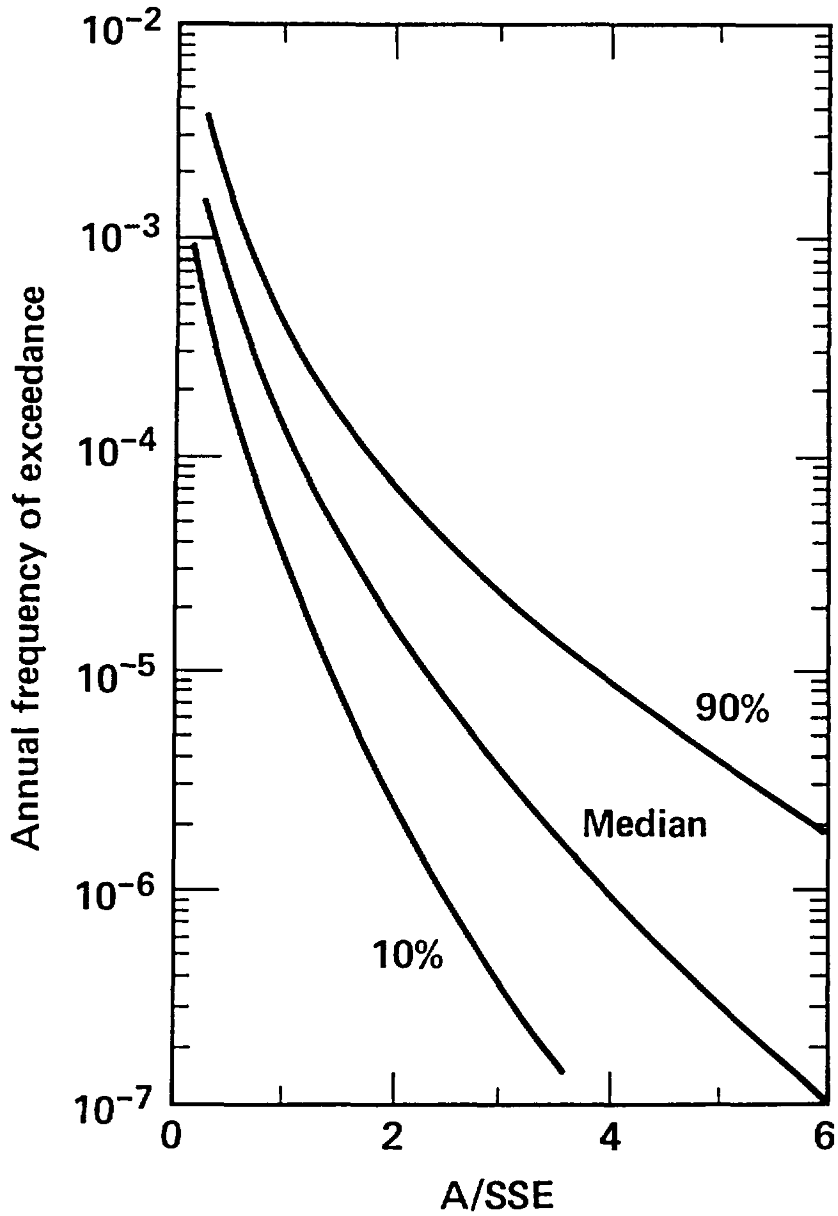

Figure 3.2. Generic seismic hazard curves used for estimating the probability of reactor coolant piping DEGB for plants east of the Rocky Mountains. For a given plant, these curves normalize peak ground acceleration A to the SSE acceleration at the plant site.

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4. PROBABILISTIC TREATMENT OF STRESS CORROSION CRACKING

4.1 General Discussion

Recirculation piping in older BWR plants, particularly those characterized by the General Electric Mark I containment design, has been found in recent years to be susceptible to intergranular stress corrosion cracking. Stress corrosion cracking occurs in stainless steel piping (in the Mark I plants, Type 304) when the "appropriate" (in an adverse sense) conditions of "sensitization", i.e. material properties conducive to IGSCC that result from prolonged exposure to high temperatures during welding, environment and stress are met. Our PWR evaluations did not consider IGSCC because history has not shown it to be a problem in PWR reactor coolant loop piping. Such is not the case, however, for BWR piping.

Stress corrosion crack growth can occur under constant loading conditions, and is therefore very different from crack growth driven by the cyclic loading discussed in the previous section of this report. Our original probabilistic fracture mechanics model incorporated a simple model of stress corrosion cracking, based on the assumption that crack growth velocity in either the radial or circumferential direction is controlled by the value of stress intensity factor K (for a given material and environment) at the crack tip. This model described crack kinetics by a simple functional relationship between crack growth rate and stress intensity factor (see Fig. 4.1). The original model did not, however, account for variations in coolant environment (most notably, temperature and dissolved oxygen content); crack growth rates were considered applicable under typical operating reactor conditions in the presence of water with 0.1 to 0.3 ppm of dissolved oxygen. As such, it did not account for higher levels of dissolved oxygen (typically in the range of 8 ppm) that occur, for example, during plant startup and which have been shown to aggravate stress corrosion cracking. The model also did not allow for crack initiation during plant operation, which adds new cracks at a given weld location and has been observed to occur in the field, or for variations in the susceptibility (or resistance) of different materials to "sensitization" during the welding process.

4.2 Probabilistic Model of Stress Corrosion Cracking

As part of our BWR study we developed an advanced IGSCC model for the PRAISE code. This semi-empirical model, described in detail by Vol. 3 of this report series, is based on laboratory and field data compiled from several sources. Using probabilistic techniques, the model addresses the following IGSCC phenomena:

- **crack initiation**, including the effects of environment, applied loads, and material type (i.e., sensitization). Crack location, time of initiation, and velocity upon initiation are all defined by appropriate distributions based on experimental data.

"Initiated" cracks are considered separately from pre-existing cracks until one of the following two criteria are satisfied: (1) the crack attains a depth of 0.1 inch, or (2) the velocity of the crack estimated according to the Paris growth law exceed the initiation velocity. Beyond this point, "initiated" and "fracture mechanics" cracks are treated identically.

- **crack growth rate**, including effects of environment, applied loads, and material type.
- **multiple cracks**. Because our earlier evaluations were based on pre-existing flaws only, each Monte Carlo replication included one crack only. Inclusion of crack initiation requires that multiple cracks be considered during each replication.
- **crack linking**. Treating multiple cracks requires that their potential linkage into larger cracks be considered. This is done using linkage criteria specified in Section XI of the ASME Boiler and Pressure Vessel Code.

The model covers not only the Type 304 stainless steel (304SS) found in most Mark I recirculation piping, but Type 316NG "nuclear grade" steel as well, a low-carbon alloy widely regarded as an IGSCC-resistant replacement for Type 304. Crack growth rates and times-to-initiation for each material are correlated against "damage parameters" which consolidate the separate influences of several individual parameters. The damage parameters are multiplicative relationships among exponential terms which individually describe the effects of the various phenomena on IGSCC behavior, including:

- **environment**, specifically coolant temperature, dissolved oxygen content, and level of impurities.
- **applied loads**, including both constant and variable loads to account for steady-state operation and plant loading or unloading, respectively.
- **residual stresses**. Steady-state pipe loads due to welding residual stresses are considered in addition to fatigue loads.
- **material sensitization**, including material type and degree of sensitization.

Figures 4.2 and 4.3 show, respectively, times-to-initiation and crack growth rates for 304SS, the material on which the initial development of the model was based. The solid curved lines in Fig. 4.3 show crack growth rates predicted by the earlier IGSCC model in PRAISE for oxygen concentrations of 0.2 ppm (typical during plant operation) and 8 ppm (typical during startup); the relatively close agreement implies that the earlier model gave reasonable crack growth rates despite its much simpler approach.

The damage parameters in the 304SS model were based on the results of both constant-load (CL) and constant extension rate (CERT) IGSCC laboratory tests. Many other factors were considered during initial model development, but were later excluded from consideration either because they were judged to be of secondary influence for 304SS, or because suitable operating data was not available to exercise them in a plant-specific evaluation. The model also assumes that growth rates and times-to-initiation measured under intentionally harsh laboratory conditions can be extrapolated to the relatively benign conditions found in actual reactors. We regarded this assumption as conservative, noting, for example, that some experimental observations [14] suggest levels of stress intensity factor below which stress corrosion cracking is effectively arrested or at least significantly reduced. Our original simplified model of IGSCC allowed for such "threshold" behavior (such as that shown in Fig. 4.3), the present advanced model does not.

Although the present model was developed for 304SS, adapting the correlation scheme for 316NG was a relatively straightforward matter of defining new damage parameters based on appropriate laboratory data; the basic functional form of the model was otherwise left unchanged. Two features unique to the 316NG model are, however, noteworthy:

- where both CERT and CL data were available for 304SS, only CERT data was available for 316NG. These data were used to define constant-load growth rates and times-to-initiation in 316NG under the assumption that the creep behavior of both alloys is similar.

- as noted earlier, three conditions are necessary for IGSCC in austenitic steels: stress, environment, and sensitization. In 304SS, whenever stress corrosion cracking occurs in laboratory tests intended to simulate operating BWR conditions, it is most often intergranular. In 316NG, however, CERT specimens fail by transgranular stress corrosion cracking (TGSCC), whereas IGSCC *is* observed in fracture mechanics specimens. Since the relative influence of environment and loading on TGSCC in 316NG appears similar to that of IGSCC in 304SS, the available TGSCC data were used to predict intergranular cracking in 316NG.

Residual stress is treated as a random variable in the Monte Carlo simulation. Distributions of residual stress as a function of distance from the inner pipe wall were developed from experimental data for three categories of nominal pipe diameter. For large lines (20 to 26 inches), residual stresses took the form of a damped cosine through the wall as based on data collected by General Electric and Argonne National Laboratory (see Fig. 4.4). The nominal tensile stress at the inner pipe wall is about 40 ksi. For intermediate-diameter (10 to 20 inches) and small-diameter (less than 10 inches) lines, a linear distribution was assumed through the pipe wall with respective inside wall stresses of 9.3 ksi and 24.4 ksi. The outside wall stress is calculated on the basis of axial force equilibrium.

The 304SS model was benchmarked by comparing predicted leak rates under nominal BWR applied load conditions against actual leak and crack indication data made available to us by the NRC Office of Nuclear Reactor Regulation (NRR). During benchmarking we quickly ascertained that residual stress was the parameter most influencing the predicted leak rates, and we therefore opted to "tune" the model on this basis. A variety of schemes were considered before we settled on adjusting the stress magnitude (using a multiplication factor) to bring the model into agreement with the field data. Figure 4.5 compares predicted leak rates against field data for various adjustment factors, Figure 4.6 the number of NDE indications with depth a greater than 10 and 50 percent of the wall thickness h, based on the optimum stress adjustment factor. As Fig. 4.5 shows, surprisingly large reduction factors had to be applied to bring the model into line with the field data, suggesting that factors other than residual stress may be more influential than we first concluded.

Calculations performed during final development of the 316NG model revealed several interesting characteristics of its behavior compared to that of the less-resistant 304SS. For example, we performed analyses both for initiated cracks and for pre-existing cracks, the latter case reflecting only the effect of stress corrosion on crack growth and not only the addition of new "initiated" cracks to the overall population. Figure 4.7 shows a typical set of results from these analyses, in this case cumulative leak probabilities for an intermediate-diameter weld. Two observations are significant here:

- at any given time, the estimated leak probability in 304SS is some two to three orders of magnitude higher than in 316NG.
- the time required to reach a given leak probability is about six times as long in 316NG as it is in 304SS.

These results also show that where leak in 304SS is *always* dominated by initiated cracks (i.e., resulting from stress corrosion), in 316NG the initiated cracks dominate the probability of leak only after about 12 years. Once cracks are present, however, growth rates are nominally the same in either material. Consequently, the predicted difference in behavior between the two materials is due to differences in the times-to-initiation and in the number of initiated cracks, rather than differences in their "fracture mechanics" characteristics.

Volume 3 of this report series discusses the features, development and benchmarking of the stress corrosion model in greater detail. The following sections describe its application to the recirculation piping in a representative Mark I BWR plant.

4.3 Probability of Failure in BWR Recirculation Loop Piping

After we completed development of the stress corrosion model, we applied it to the recirculation loop piping in an actual Mark I BWR plant. We estimated the leak and DEGB probabilities both for an existing recirculation loop (shown in Fig. 2-3), and for a proposed "replacement" loop (Fig. 4.8) fabricated from 316NG. Aside from its use of the more corrosion-resistant material, the replacement loop differs from the original by having fewer weld joints (30 compared to 51) and by eliminating entirely the pump bypass line (see Table 4.1).

During development of the IGSCC model, we found that its complexity greatly increased computer time requirements for its execution (up to three CPU hours per weld for the 20000 to 50000 Monte Carlo replications typical of our analyses) compared to our earlier PWR reactor coolant loop assessments. In order to keep the computational effort within practical bounds, we grouped the welds in the BWR pilot plant recirculation piping, taking those welds with the highest applied loads in each group. We then estimated the leak and DEGB probabilities at each of these representative welds and performed a systems analysis assuming that these leak and DEGB probabilities applied to all welds in the respective group. We followed a similar procedure for the proposed replacement system.

Practical considerations aside, the assumption of "worst case" stress conditions for each weld group offers reasonable assurance that the results of the analysis will be conservative. This conservatism is further enhanced by the fact that we did not include in-service inspection (ISI) in our evaluations (although PRAISE has this capability), nor did we consider how such IGSCC mitigating measures as weld overlay or inductive heating stress improvement (IHSI) might influence the estimated failure probabilities. Our main objective was to investigate the relative behavior of different material types under otherwise nominally identical conditions.

Figure 4.9(a) and Figure 4.9(b) show, respectively, cumulative per-loop system leak and DEGB probabilities estimated by PRAISE for the existing loop configuration (i.e. including bypass piping). Results are given both for the original 304SS material and for the Type 316 nuclear grade. In the 304SS piping, leak is predicted to occur after about ten years of operation (i.e. the cumulative probability of leak approaches one). While it is important to keep in mind the conservatism of the analysis, this result is nonetheless reasonably consistent with some field observations. The corresponding probability of DEGB is on the order of 1E-2 after 10 years (or about 1E-3 per year), increasing only slightly (by about a factor of two) over the remaining 30 years of plant life.

If the 304SS is replaced with 316NG while keeping the original piping configuration (a fictitious intermediate step between the existing loops in our pilot plant and the replacement system actually proposed), corresponding leak and break probabilities are nominally zero after 10 years of operation. The probability of leak first exceeds 1E-4 after about 12 years, increasing to about 5E-1 at the end of plant life. Two DEGB events (out of 25000 Monte Carlo replications) were predicted in the riser weld, the first of which occurred at about 30 years; all other weld groups experienced no DEGB events over the entire 40 years of plant life. The resultant end-of-life system break probability is about 2E-3 per loop, or about 2E-4 per loop-year; keep in mind that this result assumes (1) no "threshold" behavior, (2) no ISI over the 30-year period, (3) worst-case applied stresses, and that (4) all risers in the system behave identically. For the "replacement" loop configuration actually proposed, the end-of-life DEGB probability falls to about 1E-3 per loop (1E-4 per loop-year), due to fewer welds in the new system (Fig. 4.10).

The bar charts in Fig. 4.11 show the relative contribution each weld type makes to the overall system probabilities of leak and DEGB; note that Fig. 4.11 does not depict the number of predicted failures, which were far fewer in the 316NG material than in the 304SS. In the existing loop configuration, about 80 percent and 20 percent of the breaks, and about 65 percent and 25 percent of the leaks, occurred at riser welds and bypass line welds, respectively. The remaining leaks predicted (about 10 percent of the total) were distributed, in descending order, among header, discharge line, and suction line welds.

In the proposed replacement system, virtually all leaks occurred in riser welds. System break resulted solely from riser DEGB as discussed above, which Fig. 4.11(b) reflects.

The relative contribution of different weld types is further illustrated by Fig. 4.12, which shows weld-by-weld leak probabilities for the existing loop configuration. Note in particular that the per-weld leak probabilities differ by up to one order of magnitude at 10 years, and by almost two orders of magnitude by the end of plant life. Note also that while the per-weld leak probabilities for riser and bypass piping behave similarly over time, the larger number of riser welds (20 compared to 10) and their somewhat higher per-weld leak probability are reflected in their dominant overall contribution to the probability of system leak (Fig. 4.11).

Figure 4.13 compares riser per-weld leak probabilities for 304SS and 316NG piping, in both cases based on the original loop configuration. Note the probability of leak in the 304SS weldment exceeds 1E-4 after only about 3 years of operation, while in Type 316NG this threshold is crossed only after some 15 years. The reason for this difference is clear from Fig. 4.14, which shows the total number of riser crack initiations in our evaluation (one weld, 25000 Monte Carlo replications) in both the 304SS and 316NG materials. Note that cracks initiate in 304SS within the first year of operation; by the time the first initiation occurs in the 316NG (about four years), nearly 1000 cracks have initiated in the less resistant material. The ratio of 316NG initiations to 304SS initiations falls to less than 100 at ten years, and to less than five by the end of plant life (see Fig. 4.15). By this time, however, piping in an actual plant would have gone through one or more ISI cycles.

Although the results presented here are only for the representative riser weld (i.e. the dominant contributor to the probability of system failure), they are characteristic of what we observed for the other welds considered. In all cases, the 316NG appears to owe its corrosion resistance mainly to the fact that (1) fewer cracks initiated than in the 304SS material, and (2) those that did initiate typically did so later in plant life. Once a crack initiates, however, its subsequent growth rate is not significantly affected by material type.

4.4 Discussion of Results

As part of our evaluations of reactor coolant piping for the Nuclear Regulatory Commission, we developed an advanced probabilistic model of stress corrosion cracking which we applied to the recirculation loops of a Mark I BWR plant. Based on the results of these evaluations, we were able to make the following general observations:

- if stress corrosion is not a factor, thermal fatigue is the main cause of pipe failure. Furthermore, the probability of break is similar to that in PWR reactor coolant loop piping (on the order of 1E-10 per reactor-year or lower). As for PWR reactor coolant loop piping, earthquakes contribute only negligibly to the probability of direct DEGB.

- when stress corrosion is a factor, corrosion-induced failure clearly dominates. Furthermore, the probability of pipe failure is dominated by residual stresses (i.e. uniform loads) rather than by stresses induced by applied loads. Our analyses further indicated that failure probability is very sensitive to the particular description of residual stress assumed in the analysis. This result may offer insight into field observations where nominally identical recirculation loops (e.g., in terms of configuration, materials, applied loads) may exhibit stress corrosion cracking in one plant and not in another. Such differences may be at least partly attributable to plant-to-plant differences in residual stresses caused by welding and "fit up" during pipe assembly.

- recirculation loops fabricated from Type 304 stainless steel are predicted to leak after about 10 years of operation. Although this result is based on conservative "worst case" stress assumptions and on the assumption of no in-service inspection over this period, it is also consistent with some field observations.

  If the 304SS material is replaced by 316NG and the existing loop configuration is retained, the system leak probability at ten years (a "typical" ISI interval) is effectively zero. The end-of-life system leak probability (i.e. after another 30 years of operation) is about 5E-1 per loop, or about 2E-2 per loop-year assuming "worst case" applied stresses and no ISI.

- for recirculation loops fabricated from Type 304 stainless steel, the system probability of DEGB is about 1E-2 after ten years of operation (or about 1E-3 per loop-year), increasing to about 2E-2 by the end of plant life. Again, these results reflect "worst case" applied stresses and no ISI.

- for recirculation loops fabricated from Type 316NG stainless steel, the system probability of DEGB is zero for the first 30 years of operation, even under "worst case" applied stresses and no ISI. In our evaluation we predicted only two riser breaks (out of 25000 Monte Carlo replications), none in other weld types, which implies a per-loop DEGB probability on the order of 1E-4 per year or less over the final ten years of plant life, zero up to that time. Routine ISI over plant life could be expected to substantially lower the "late-life" probability of DEGB though early detection of potentially troublesome cracks.

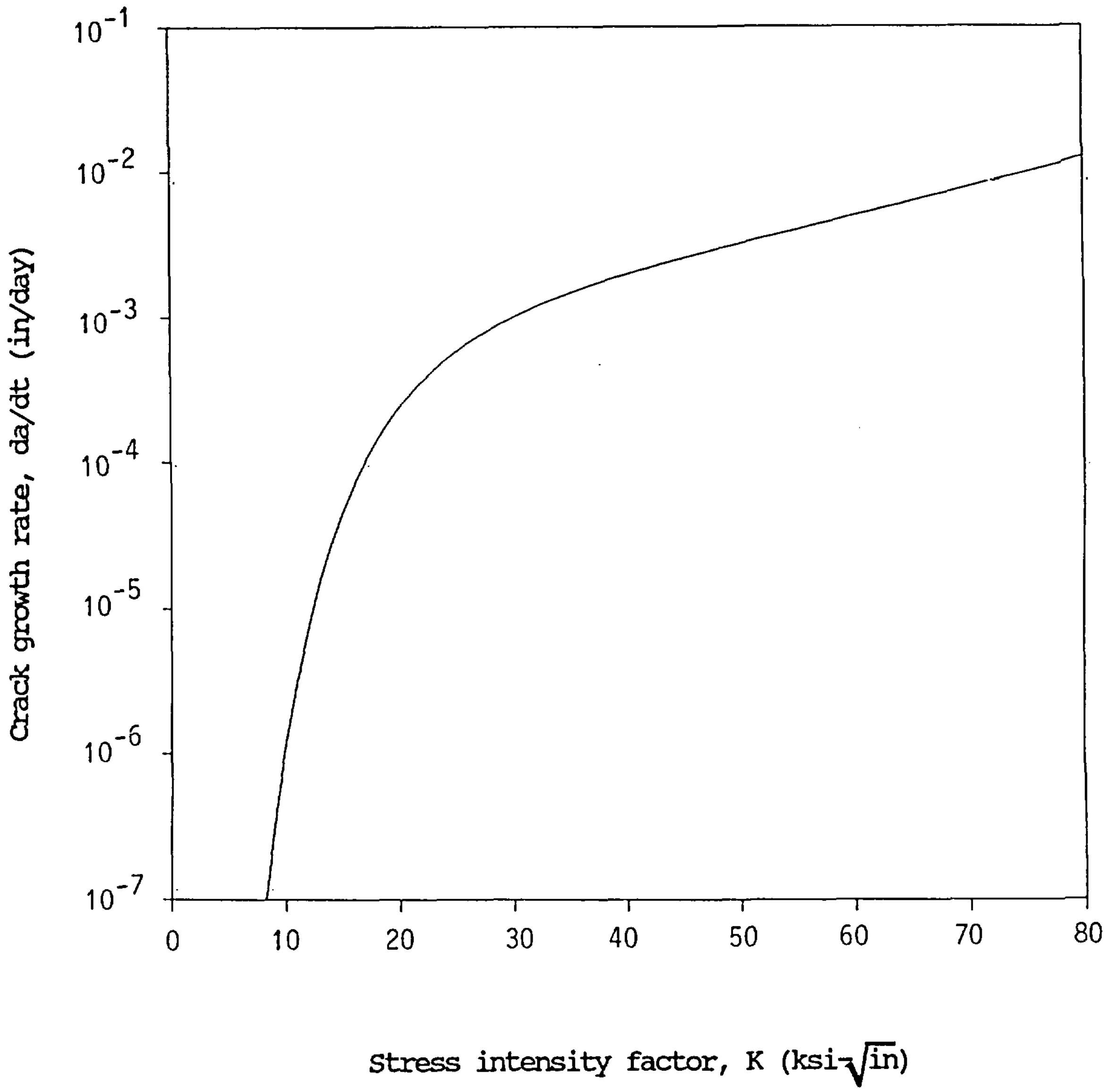
  Note that for 316NG, our "intergranular" stress corrosion model was actually based on laboratory data for transgranular stress corrosion cracking; we were unable to find suitable IGSCC data. Consequently, we would expect corrosion-induced cracking to more likely be transgranular rather than intergranular, and the probability of failure induced by "IGSCC" to actually be less than implied by our evaluations.

- for the replacement Type 316NG loop configuration, comprising fewer welds (30 compared to 51) and eliminating the bypass line altogether, the end-of-life leak and break probabilities drop by about a factor of two. Interestingly, the time-dependence of the system leak and break probabilities does not change significantly, reflecting the observation that the risers, rather than the bypass piping, dominate the probability of system failure.

- where failure in Type 304 piping is always dominated by initiated cracks (i.e., resulting from stress corrosion), in 316NG the initiated cracks dominate the probability of leak only after about 12 years. Once cracks are present, growth rates are nominally the same in either material. Consequently, the predicted difference in behavior between the two materials is due to differences in the number of initiated cracks and their later times-to-initiation, rather than how these cracks would grow once initiated.

Table 4.1. Pipe diameters, number of welds in existing and replacement recirculation loop configurations for BWR pilot plant.

Weld Group	Diameter (in)	Welds/loop (existing)	Welds/loop (replacement)
Discharge	26	10	11
Suction	26	6	5
Header	20	5	2
Risers	12	20	12
Bypass	3	10	0
Total per loop		51	30



$$\log \frac{da}{dt} = \begin{cases} -39.90 + 64.32(\log K) - 38.17(\log K)^2 + 7.81(\log K)^3 & K < 100 \\ -1.46 & K > 100 \end{cases}$$

Figure 4.1. IGSCC crack growth rate in 304SS as a function of stress intensity factor, as originally implemented in the PRAISE computer code.

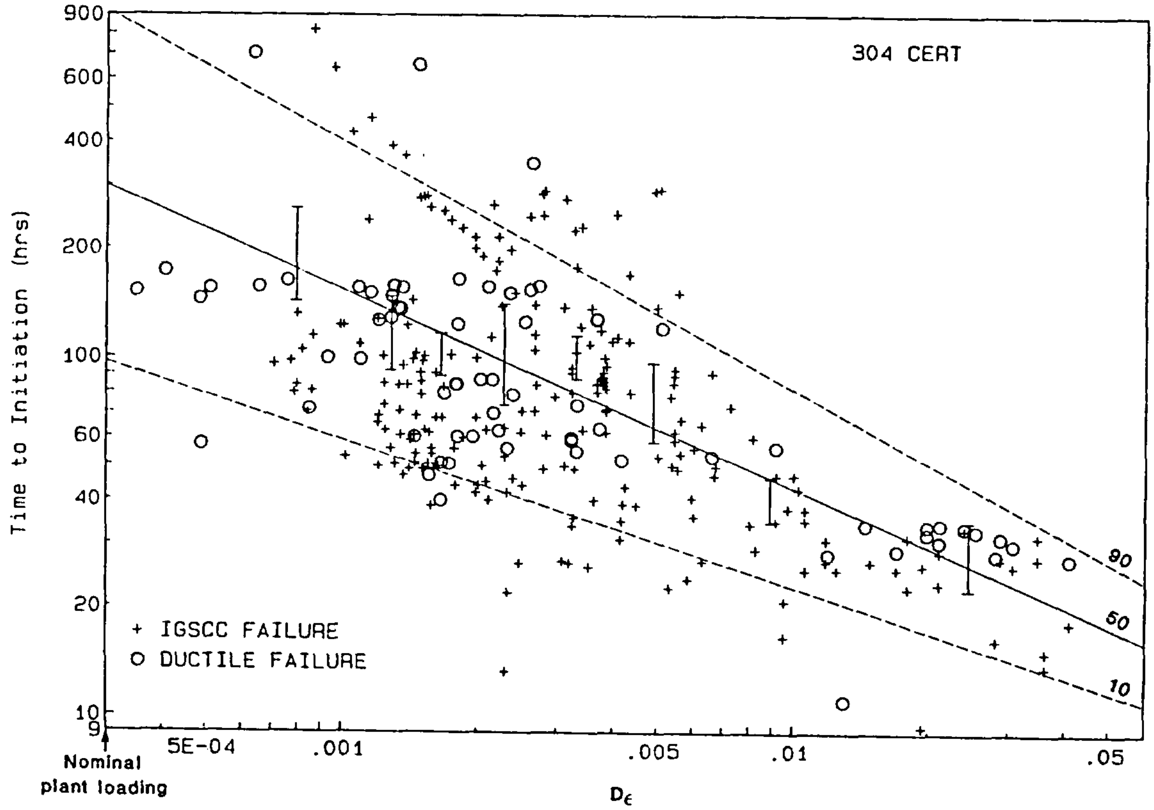


Figure 4.2. Time-to-initiation for IGSCC cracks in 304SS as a function of damage parameter, plant loading/unloading.

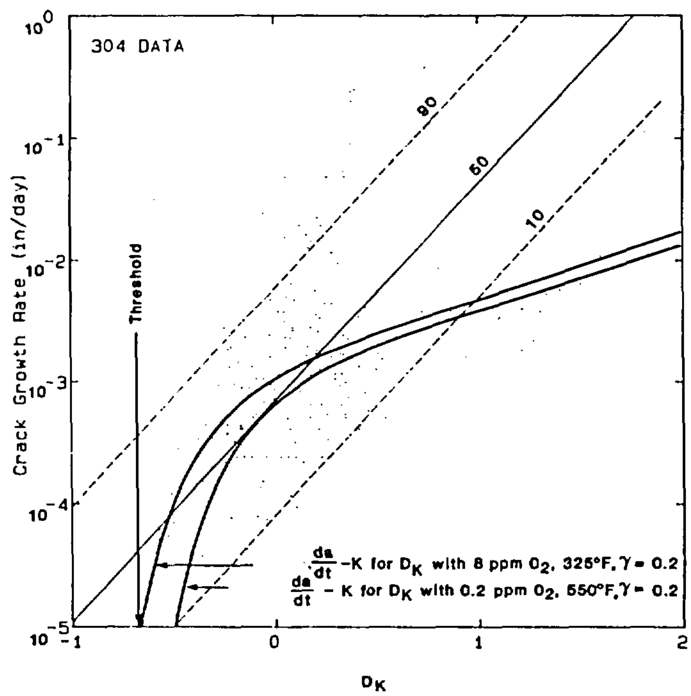


Figure 4.3. IGSCC crack growth rate in 304SS as a function of damage parameter, steady-state operation. The solid lines represent crack growth rates predicted by the earlier IGSCC model in PRAISE.

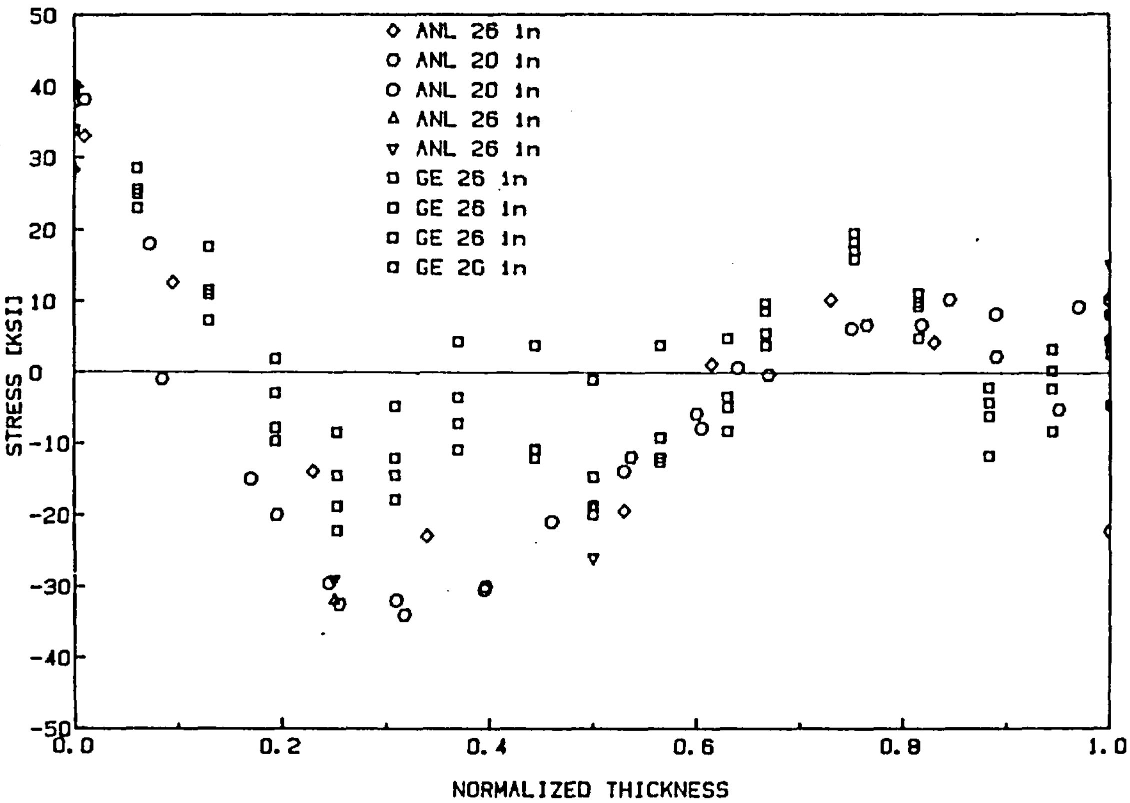


Figure 4.4. Residual stress data for large-diameter (20 to 26 inches) piping as a function of distance from the inner wall.

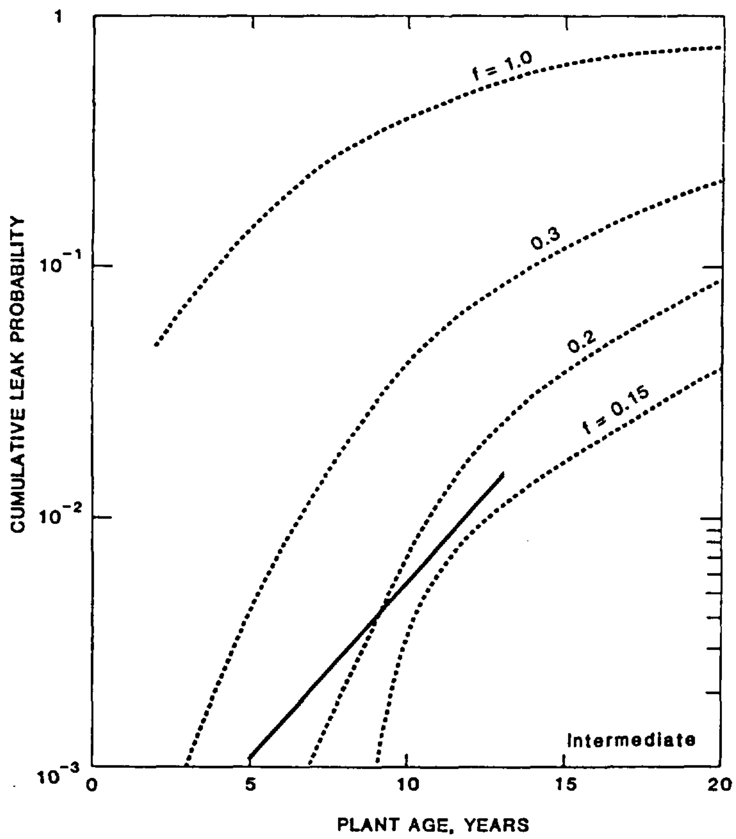
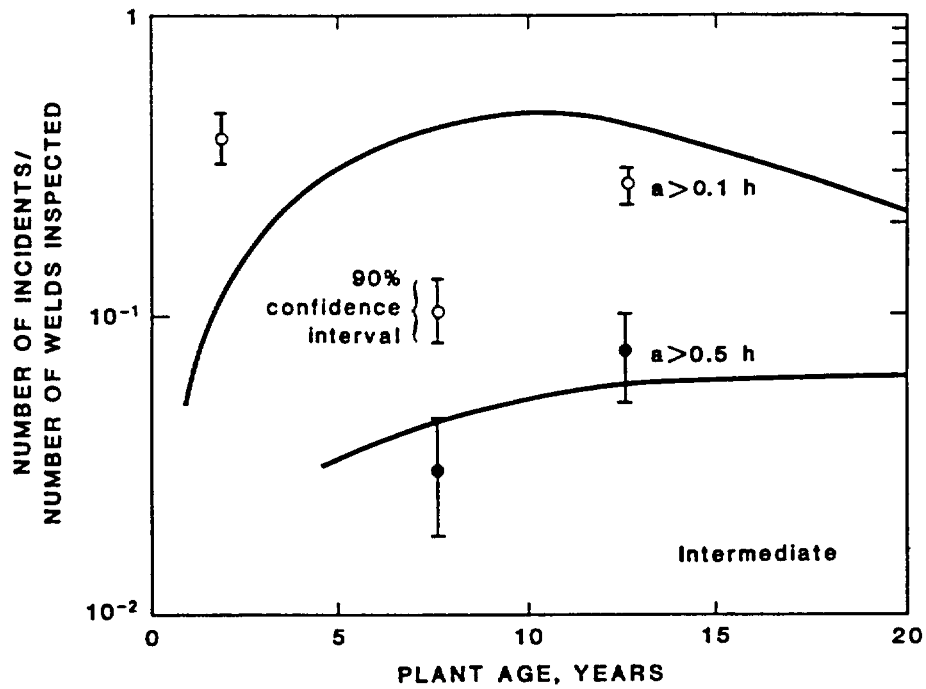


Figure 4.5. Comparison of leak probabilities derived from field data with leak probabilities estimated by PRAISE for various values of residual stress adjustment factor "f" (304SS).



●, o field data (a = crack depth, h = wall thickness) with 90% confidence interval

—— crack size distributions estimated by PRAISE for a > 0.1h (top) and a > 0.5h (bottom)

Figure 4.6. Comparison of crack indications derived from field data with PRAISE results based on optimum value of residual stress adjustment factor (304SS).

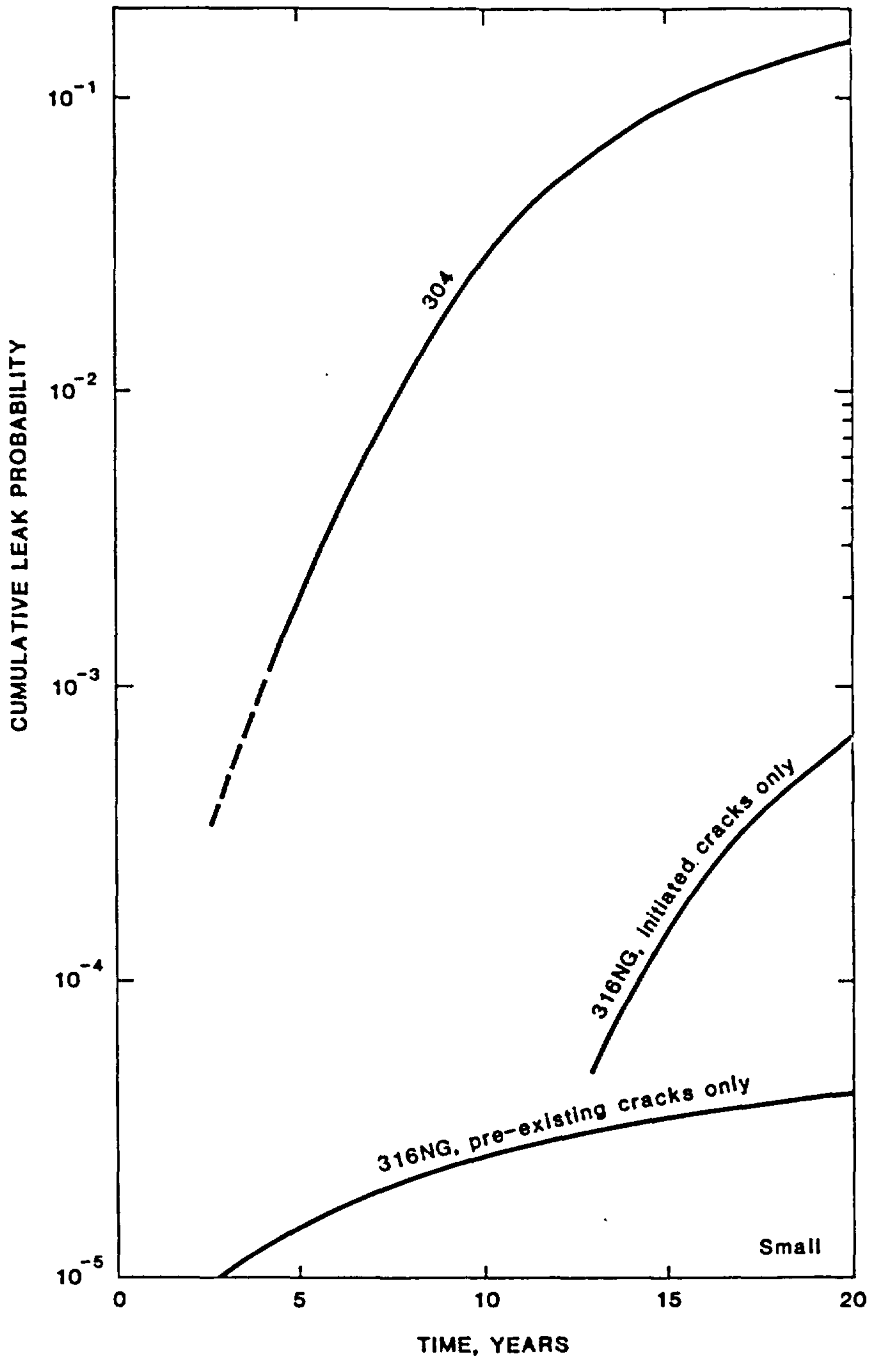


Figure 4.7. Cumulative leak probability as a function of time for small-diameter weldments fabricated from Types 304 and 316NG stainless steel.

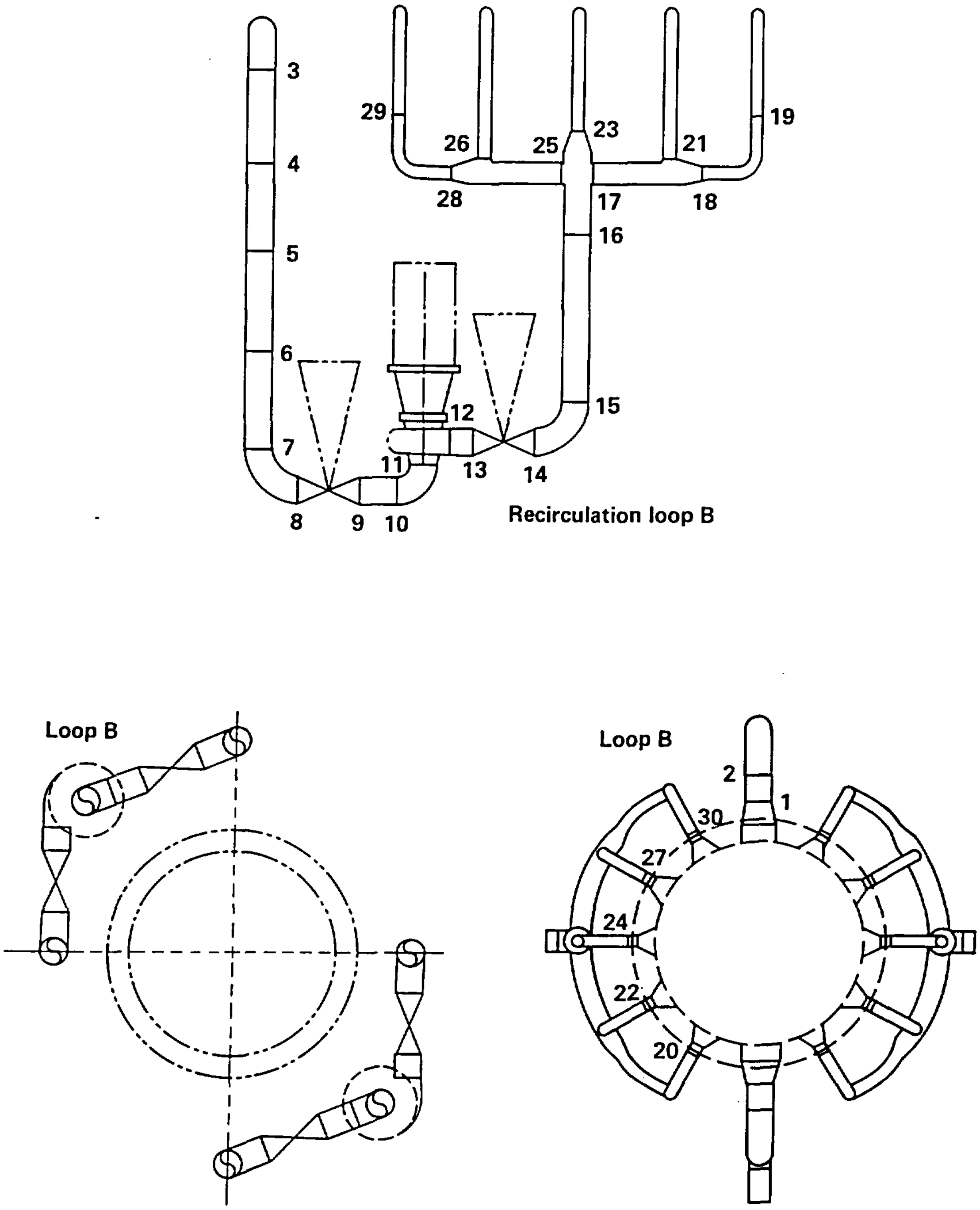


Figure 4.8. Replacement recirculation loop configuration proposed for BWR pilot plant.

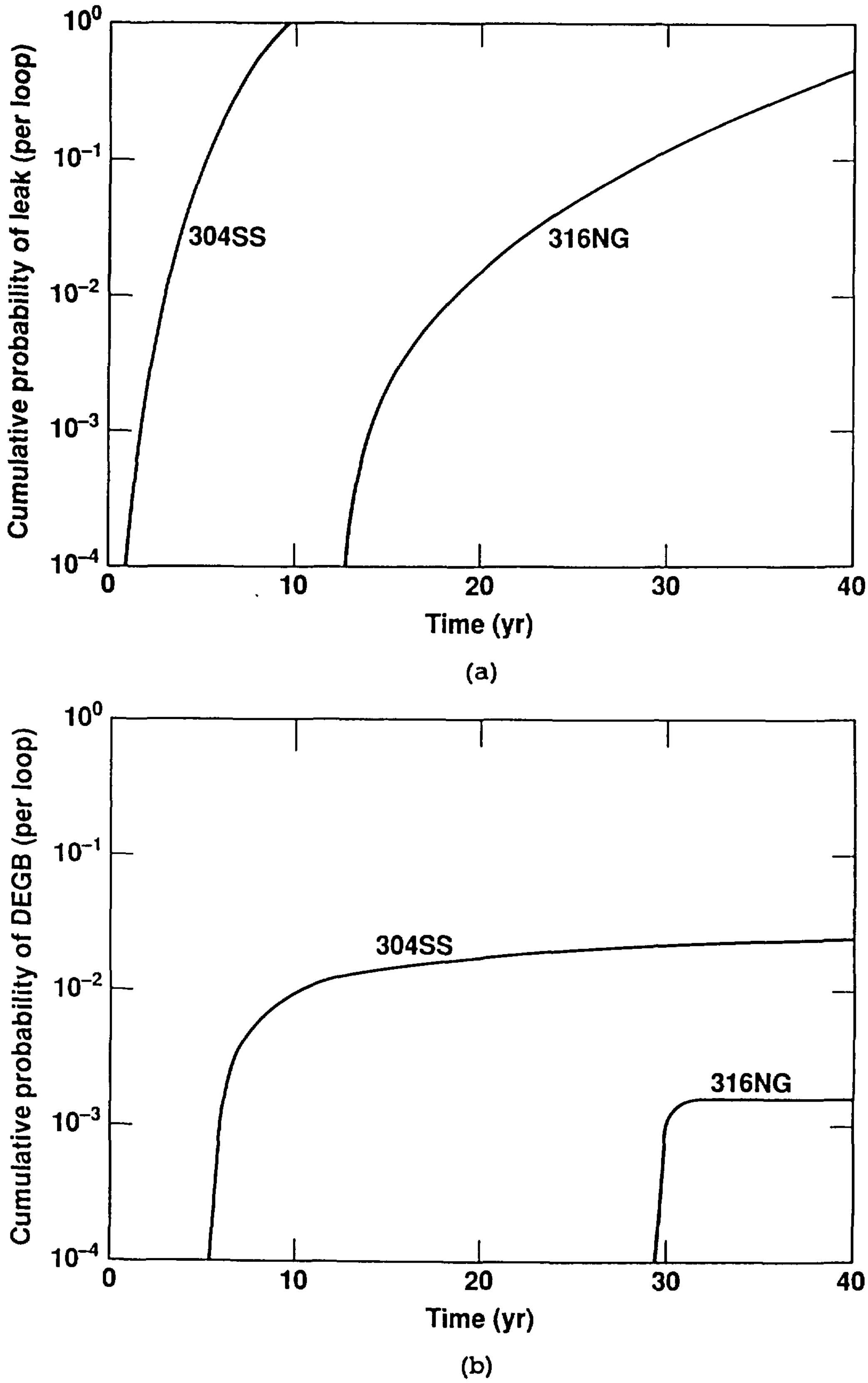


Figure 4.9. Cumulative system probabilities of (a) leak and (b) DEGB for one pilot plant recirculation loop (existing configuration).

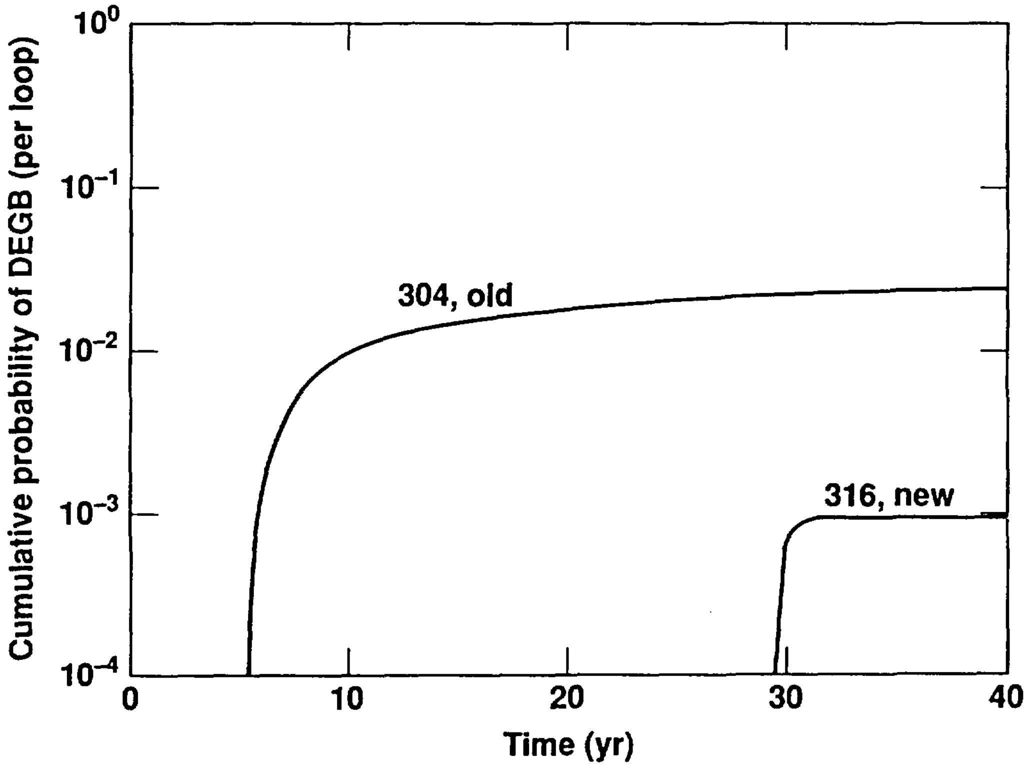


Figure 4.10. Comparison of cumulative DEGB probabilities between existing recirculation loop and proposed replacement configuration.

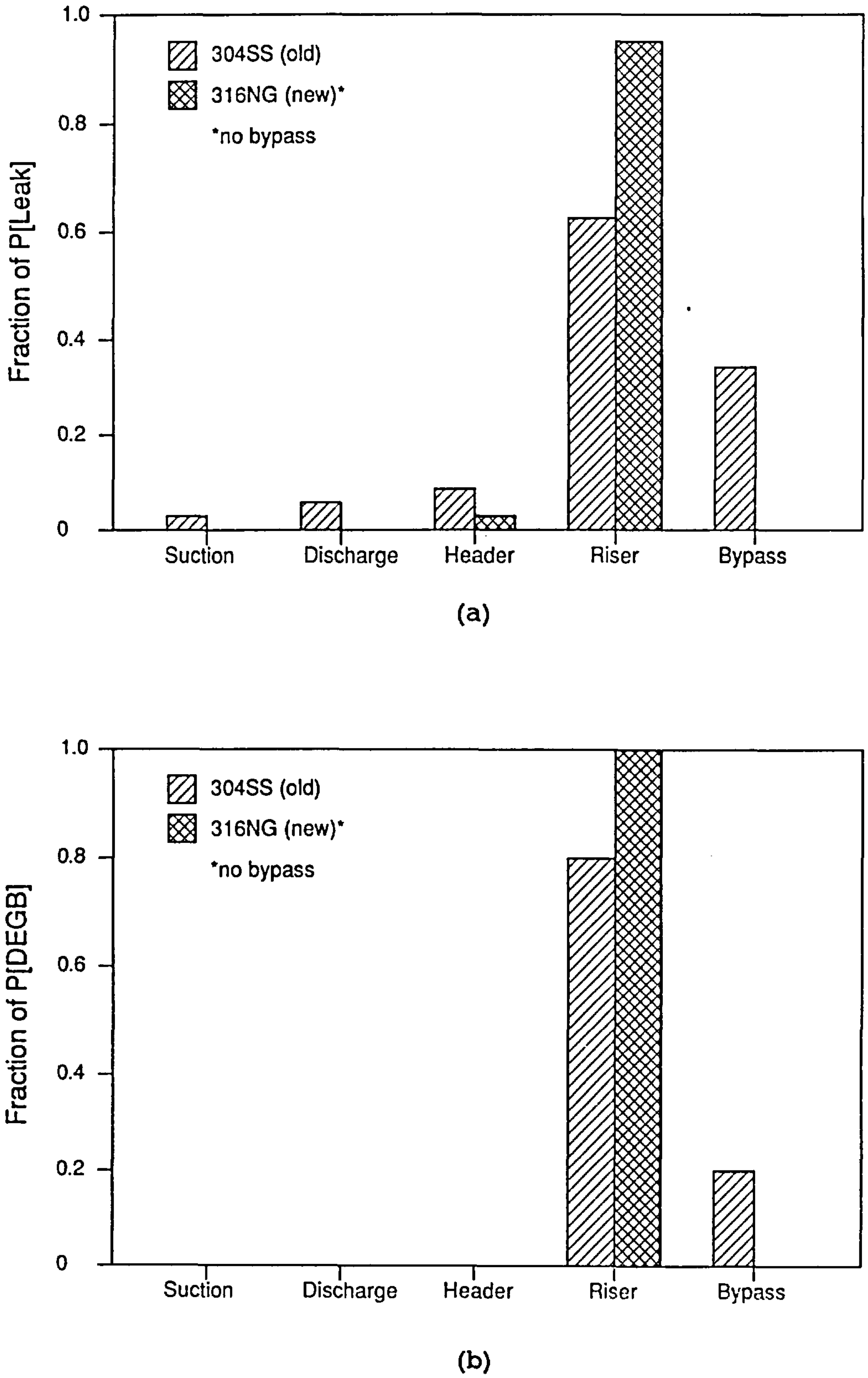


Figure 4.11. Relative contribution of various weld types to system probabilities of (a) leak and (b) DEGB, existing recirculation loops and proposed replacement configuration.

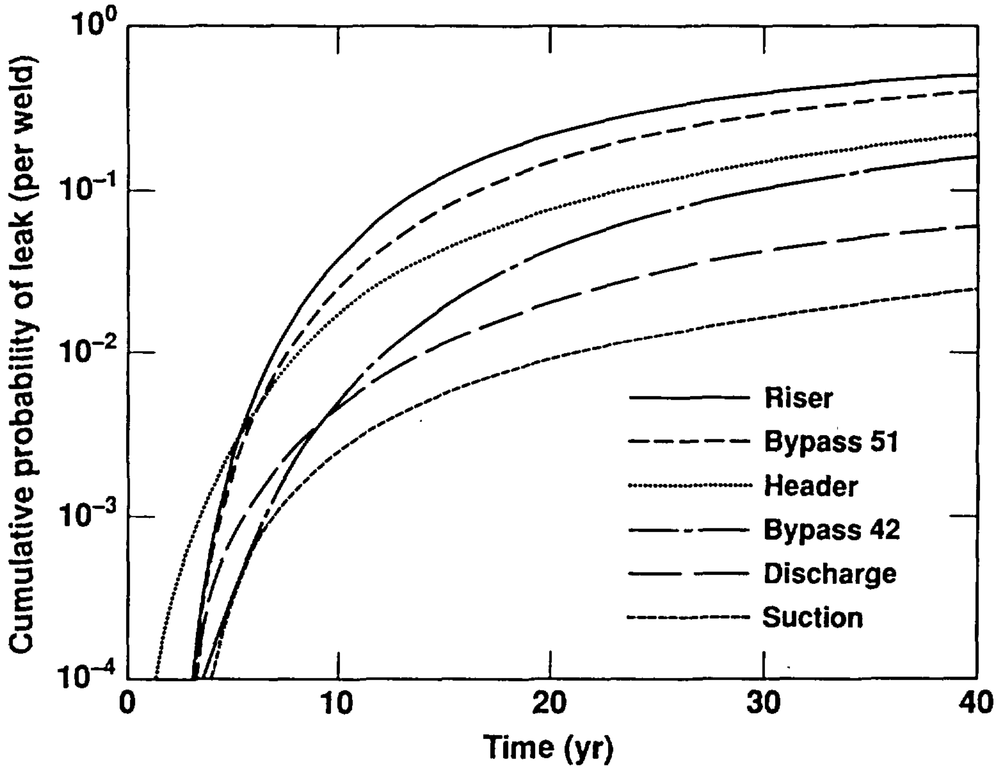


Figure 4.12. Cumulative probabilities of leak for indicated welds (existing loop configuration).

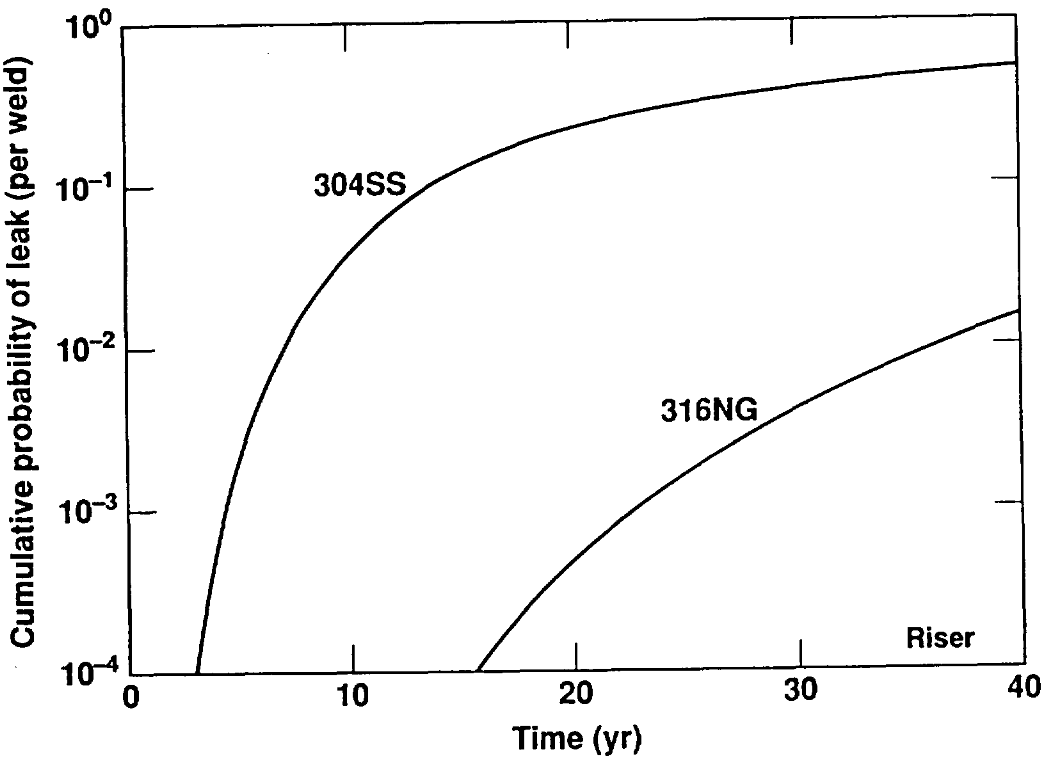


Figure 4.13. Riser weld cumulative probability of leak, for Type 304 and Type 316NG stainless steel (existing loop configuration).

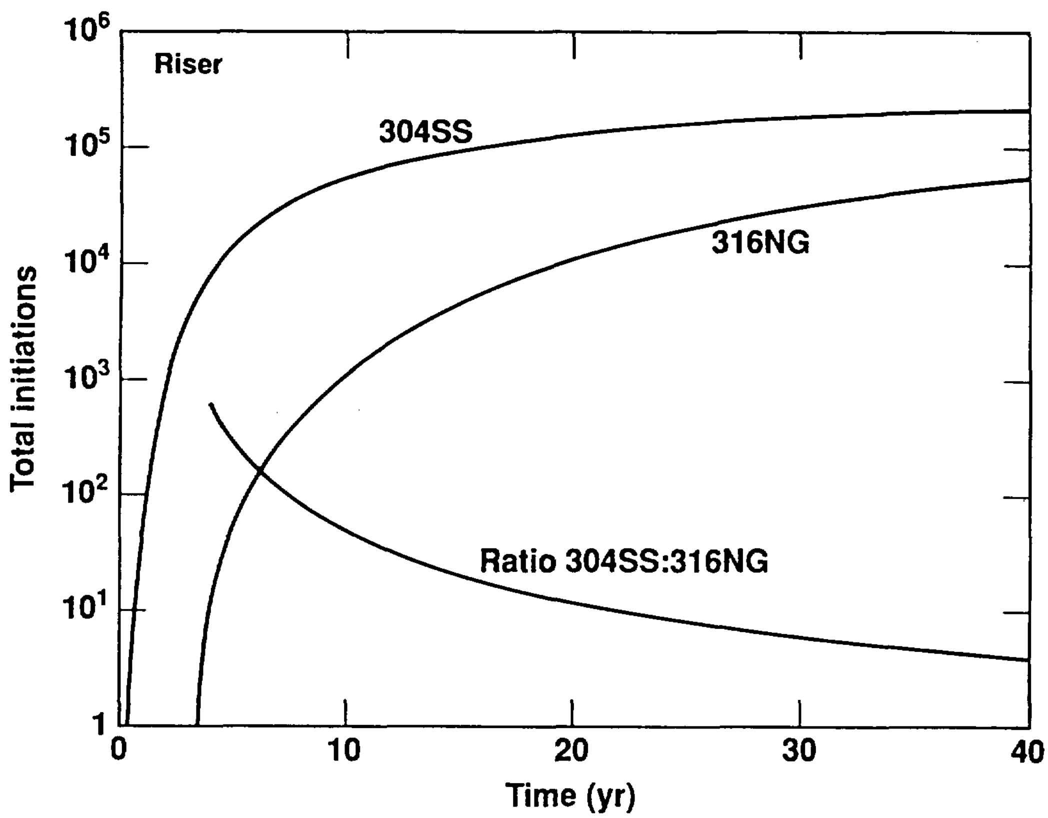


Figure 4.14. Total riser weld initiations for Type 304 and Type 316NG material, existing loop configuration (25,000 Monte Carlo replications).

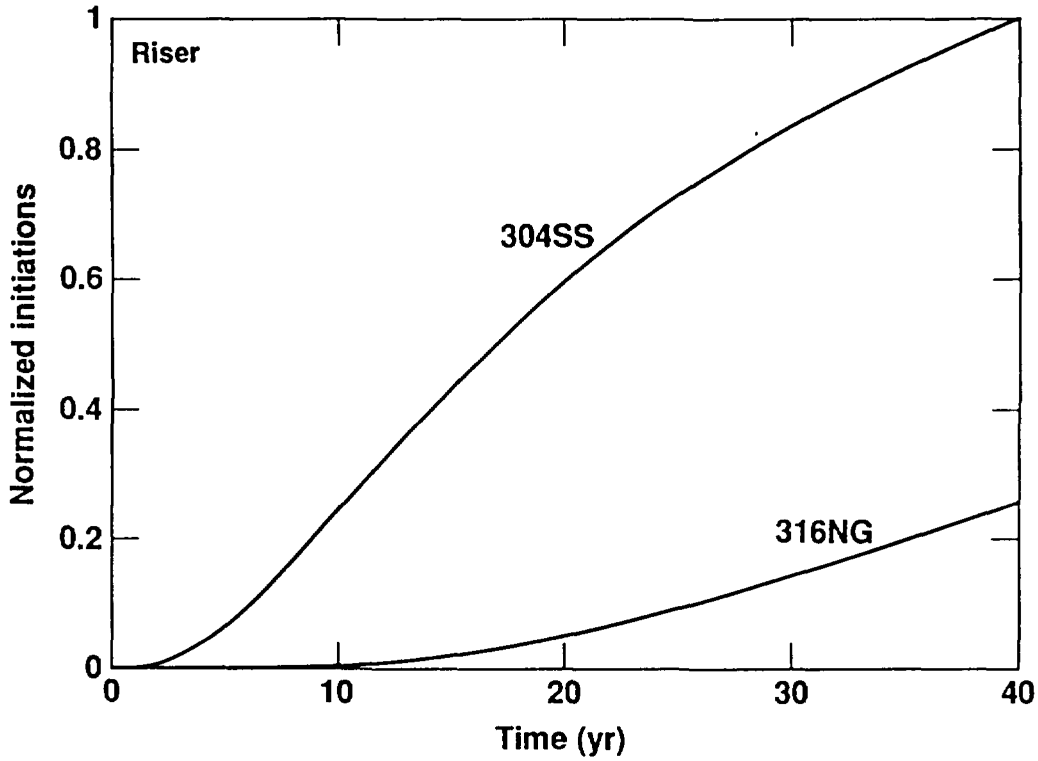


Figure 4.15. Cumulative distribution of riser weld initiated cracks, normalized to Type 304 lifetime total.

5. GUILLOTINE BREAKS INDIRECTLY CAUSED BY SEISMICALLY-INDUCED FAILURES

5.1 General Discussion

If earthquakes and pipe breaks are considered as purely random events, the probability of their simultaneous occurrence is negligibly low. However, if an earthquake could cause DEGB, then the probability of simultaneous occurrence would be significantly higher. Our studies of direct DEGB, both past and present, have generally concluded that earthquakes are 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 in turn would cause a reactor coolant pipe to break.

In our evaluations of indirect DEGB in PWR reactor coolant loop piping, we focussed our attention on the "heavy component" supports in the nuclear steam supply system, namely those for the reactor pressure vessel, the steam generators, and the reactor coolant pumps. As part of our PWR pilot study [4], we also considered in depth many other plausible causes of indirect DEGB unrelated to earthquakes (e.g. crane failure, pump flywheel missiles). We concluded that non-seismic causes of indirect DEGB were of negligible significance compared to heavy component support failure.

Our BWR study also considered the possibility of "heavy component" support failure causing a reactor coolant piping DEGB. Because, the BWR piping which we addressed is generally more complex than the PWR reactor coolant loops, we also included the seismically-induced failure of "intermediate" supports -- pipe supports and supports for light loop components -- in our overall evaluation.

5.2 Pipe Break Caused by Failure of "Heavy Component" Supports

As shown schematically in Fig. 5.1, evaluating the probability of indirect DEGB involves the following three general steps:

(1) identify "critical" components whose seismically-induced failure could induce a DEGB. The critical "heavy component" supports that we considered in our BWR indirect DEGB evaluation were limited to those comprising the reactor support structure. These included the lower support structure at the base of the RPV as well as the lateral stabilizers at the top of the vessel. A BWR, of course, has no steam generators, and the failure of coolant pump supports was considered separately in the "intermediate" support failure evaluation described in the following section.

As described in Vol. 2 of this report series, we also considered various non-seismic causes of indirect DEGB which, as in our PWR evaluations, we were able to exclude from further consideration.

(2) for each critical component, estimate the conservatism and the uncertainty in the calculated structural responses for various loading conditions, such as dead weight, thermal expansion, pressure, and seismic loads. Identify significant failure modes and, based on the structural responses, develop a "fragility" description for each. The fragility description relates the probability of structural failure conditioned on the occurrence of an earthquake of given peak ground acceleration. Combine the fragilities for individual components into an overall "plant level" fragility description.

(3) calculate the non-conditional probability of indirect DEGB by convolving the plant level fragility with an appropriate description of seismic hazard. As discussed earlier, "seismic hazard" relates the probability of occurrence of an earthquake exceeding a given level of peak ground acceleration.

In our evaluations we assumed that "heavy component" support failure would unconditionally result in pipe break. Because of this conservative assumption, the "indirect DEGB" evaluation becomes a "support reliability" evaluation. The following discussion, excerpted from Ref. 15, describes how this evaluation is performed.

5.2.1 Methodology

The probability of indirect DEGB, P[DEGB], can be mathematically expressed by:

$$P[\text{DEGB}] = \int_0^\infty P[\bigcup_{i=1}^{n} (C_i < R_i) \mid A = a]\, f_A(a)da \qquad (5\text{-}1)$$

where:

- $\cup$ = "union" symbol
- C_i = capacity of structural element "i" (e.g., RPV support, steam generator support)
- R_i = random variable representing response of structural element "i" to peak ground acceleration a
- $f_A(a)da$ = frequency of occurrence of an earthquake with peak ground acceleration between a and a+da

Equation (5-1) is written assuming that there is perfect knowledge about the values of the parameters that define the probability terms. Since there is uncertainty in these parameter values, a subjective probability distribution of the probability of induced DEGB will be obtained by appropriately varying the parameter values.

The first term within the integral of Eq. (5-1) is the conditional probability of occurrence of DEGB due to structural failures for a given peak ground acceleration, a. It is defined as the probability of failure of at least one of the structural elements which can lead to DEGB of the RCL piping. Therefore, the focus in this study is only on those structural elements within the containment whose failure can result in DEGB. Among these, some elements may have large margins of safety against seismic failure and thus may not contribute significantly to the probability of DEGB. Therefore, critical elements are defined as those whose failure could contribute significantly to the probability of indirectly-induced DEGB. For PWR plants, these were identified as the steam generator supports, the reactor coolant pump supports, and the reactor pressure vessel supports [4].

The conditional probability of DEGB is evaluated by treating the failure events of individual structural elements as statistically independent. This gives a conservative upper bound on the probability of DEGB. Also, if one of the structural elements has a very high conditional probability of failure compared to other elements, the upper bound is a good approximation to the actual P[DEGB].

5.2.2 Seismic Fragility

The conditional probability of failure of a structural element for a given peak ground acceleration is called the seismic "fragility" of the element (Fig. 5.2). The fragility evaluation in our DEGB evaluations was accomplished using information on plant design bases (see Table 5.1) and by appropriately extrapolating the responses calculated at the design analysis stage to the failure levels of the structural elements.

Evaluation of the fragility is simplified by defining a random variable called the ground acceleration capacity. The ground acceleration capacity, denoted by A_C, is expressed as:

$$A_C = F * A_{SSE} \tag{5-2}$$

where F is the factor of safety on the design basis earthquake (usually the safe shutdown earthquake) and A_{SSE} is the peak ground acceleration at the safe shutdown earthquake. The factor of safety is defined as a ratio of the seismic capacity of the structural element C_i to the response, R_i, of the element due to the SSE. Since C_i and R_i are random variables, the factor of safety F is also a random variable.

The factor of safety F is modeled as a log-normally distributed random variable with the parameters, median F and logarithmic standard deviation β_F. Two basic types of variability are identified in describing the factor of safety: one that represents the inherent randomness and one that represents the uncertainty in the parameter value, e.g. the median. These variabilities are quantified by the logarithmic standard deviations $\beta_{F,R}$ and $\beta_{F,U}$, respectively. Essentially, $\beta_{F,R}$ represents the variability due to randomness of earthquake characteristics for the same peak ground acceleration and to the randomness of the structural response parameters which relate to these characteristics. The dispersion represented by $\beta_{F,U}$ is due to such factors as:

(1) Our lack of understanding of structural material properties such as strength, inelastic energy absorption capacity and damping, and

(2) errors in calculated response due to use of approximate modeling of the structure and equipment, and inaccuracies in mass and stiffness representations.

For equipment supports, the factor of safety can be modeled as the product of three variables:

$$F = F_C * F_{RS} * F_{RE} \quad (5\text{-}3)$$

where the capacity factor F_C for the equipment support is a product of a strength factor F_S and an inelastic energy absorption factor F_μ. The strength factor F_S represents the ratio of ultimate strength to the stress calculated for A_{SSE}. The inelastic energy absorption factor (ductility) accounts for the fact that an earthquake represents a limited energy source and structures or components are generally capable of absorbing substantial amounts of energy beyond yield without loss of function.

The structural response factor F_{RS} recognizes that in the design analyses, the structural response was computed using specific (and often conservative) deterministic response parameters for the structure. Because many of these parameters are random (often with a wide variability) the actual response may differ substantially from the design response calculated for a given peak ground acceleration. The more significant factors include variability in (1) ground motion and associated ground response spectra for a given peak free-field ground acceleration, (2) soil-structure interaction, (3) energy dissipation (damping), (4) structural modeling, (5) method of analysis, (6) combination of modes, and (7) combination of earthquake components.

The equipment response factor F_{RE} depends upon the response characteristics of the equipment (in this case, the nuclear steam supply systems including RPV, steam generators, reactor coolant pumps, and their supports) and is influenced by the same variables as those listed for structural response.

For each variable affecting the factor of safety, the median value as well as the associated random and modeling uncertainties β_R and β_U are estimated.

With the overall factor of safety F estimated as described above, the ground acceleration capacity of the structural element is then calculated using Eq. (5-3):

$$A_C = F_C * F_{RS} * F_{RE} * A_{SSE} \tag{5-4}$$

$$\beta_{A,R} = (\beta^2_{C,R} + \beta^2_{RS,R} + \beta^2_{RE,R})^{1/2} \tag{5-5}$$

$$\beta_{A,U} = (\beta^2_{C,U} + \beta^2_{RS,U} + \beta^2_{RE,U})^{1/2} \tag{5-6}$$

The ground acceleration capacity of each equipment support was modeled as the lowest capacity in all credible failure modes. This is a realistic assumption since the failure modes are highly correlated due to common structural material, method of fabrication, and correlation of input motion. Again, if the structural element of one of the failure modes has a very low capacity compared to other modes, this assumption leads to a good approximation of the probability distribution of the capacity.

5.2.3 Seismic Hazard

The remaining term within the integral of Eq. (5-1), $f_A(a)da$, is the annual probability that the peak ground acceleration at the plant site is between a and a+da. This is generally referred to as "seismic hazard" and is usually described by a set of curves plotting annual exceedance probability as a function of peak ground acceleration. The uncertainty in the hazard description is represented by assigning each curve a subjective weighting factor (or confidence limit). Our BWR evaluation and most of our PWR evaluations were based on generic hazard curves which we developed for the eastern United States (Fig. 5.2); west coast PWR plants were evaluated on the basis of site-specific seismic hazard information.

5.2.4 Probability of Pipe Break

Applying this methodology to PWR reactor coolant loop piping, we found for all vendors that the probability of indirect DEGB was very small. For Westinghouse plants, the median probability of indirect pipe break was about 1.0E-7 per reactor-year for plants east of the Rocky Mountains (based on generic seismic hazard curves), and about 3.0E-6 per reactor-year for plants on the more seismically active west coast; "upper bound" (i.e. 90% confidence level) probabilities were typically about one order of magnitude higher. Equivalent results for Combustion Engineering and Babcock & Wilcox reactor coolant loops were comparable to the Westinghouse results.

In our BWR evaluation, we found that the probability of indirect DEGB due to heavy component support failure was about 2E-8 events per reactor-year, with a 90th-percentile value (confidence limit) of 5E-7 per reactor-year. We found that the "star" stabilizer at the top of the reactor pressure vessel, which restrains the RPV against lateral motion in the event of an earthquake, was the primary contributor to failure rather than the main support structure at the bottom of the vessel.

Volume 4 of this report series discusses our BWR "heavy component" support evaluations in greater detail.

5.3 Pipe Break Caused by Failure of "Intermediate" Pipe Supports

5.3.1 General Discussion

Reactor coolant loops in PWR plants typically have small length-to-diameter ratios and, because of their stiffness, are supported solely by the major loop components (reactor pressure vessel, reactor coolant pumps, and steam generators); therefore, no additional supports are needed. However, recirculation loop piping in BWR plants is longer and smaller-diameter (typically 12 to 26 inches, although some systems include piping as small as 3 to 4 inches in diameter), and requires additional support from spring- or constant-load hangers. This piping may also have numerous snubbers to reduce stresses in the event that an earthquake occurs. Each recirculation loop at our BWR pilot plant, for example, has a snubber pair each on the inlet and outlet lines, as well as a snubber triplet at the top and at the bottom of the recirculation pump.

The potential effect of intermediate support failure on estimating the probability of direct DEGB is two-fold:

(1) support failure would redistribute applied stresses at weld joints, in turn affecting crack growth rates as well as the failure criteria used to define when pipe break occurs.

(2) accounting for stress redistribution would require an individual probabilistic fracture mechanics evaluation for each support failure scenario, dramatically increasing the computational effort involved. For example, even if only four supports were addressed, sixteen separate PRAISE runs would be required to cover all possible combinations and permutations of support failure.

Our evaluations of "indirect" DEGB caused by heavy component support failure assumed that support failure unconditionally resulted in pipe break. This assumption was regarded as conservative, but nevertheless resulted in very low DEGB probabilities. To assume that failure of a snubber or a constant-load support would similarly cause a DEGB in BWR recirculation loop piping would be unreasonably conservative; in other words, a simple "support reliability" evaluation would no longer suffice. We therefore developed a more sophisticated approach to incorporate the effect of support fragility into the probabilistic fracture mechanics evaluation, which we used to investigate the effect of support failure on the probability of DEGB. Note that the need to incorporate support failure in the fracture mechanics evaluation blurs the distinction between "direct" and "indirect" DEGB, and leads us back to a more integrated approach for estimating the probability of pipe break.

5.3.2 Methodology

Incorporating the effect of support failure on pipe failure probability is clearly a complicated problem demanding an accordingly complex analytic approach. Many questions can be asked:

(1) What is the failure probability of a support for a given earthquake?

(2) When does this support failure occur?

(3) What is the response of the piping once a support fails?

(4) What is the effect of this new response on the pipe failure probability?

(5) If there is more than one support, how many supports will fail during an earthquake?

(6) What is the failure sequence of these supports?

(7) What is the piping response in such a scenario of multiple support failure? How is that going to affect the pipe failure probability?

It is difficult to answer these questions. The problem is further complicated by the fact that these questions are interrelated. For example, the support failure probability (Question 1) is affected by the piping and support responses (Question 3), which is in turn affected by the failure sequence of the supports (Question 6). Obviously, it is beyond our capability to address all of these questions. In this study, we made the following assumptions to simplify the problem to a manageable level.

(1) All support failures occur at the same time and at the beginning of an earthquake. In other words, the piping system experiences the full duration of the earthquake for any given combination of support failures. This assumption is conservative. Thus, timing and the sequence of support failures in an earthquake are not considered. This assumption greatly reduced the complexity of the problem to a manageable size.

(2) The supports experience the same stress distribution as if no support failure occurred during an earthquake. This allows the regular fragility development method to be applied to develop one fragility curve for each support for all levels of earthquake intensity. This assumption also allows one single set of in-structure response spectra or one single set of floor time-histories to be used in all seismic analyses.

(3) The failure events of the supports are statistically independent of each other. The probability that certain supports will fail together in an earthquake is the product of their individual failure probabilities.

With these three assumptions, we are ready to perform the complicated, even though much simplified, assessment of pipe failure probability with the effects of seismically-induced support failure. Thus, the probability of pipe failure can be expressed as:

$$P[PF] = P[PF|\text{no } SF] * P[\text{no } SF] + \sum_{i=1}^{N} \{P[PF|SFi] * P[SFi]\} \qquad (5\text{-}7)$$

where N represents the total number of support failure combinations, and SFi represents the "i"th combination of support failure. For example, a piping system with four supports will have a total of 15 support failure combinations (excluding the case of no supports failing): four cases of single-support failure, six cases of two-support failure, four cases of three-support failure, and one case of four-support failure.

To describe the general methodology as represented by Eq. (5-7), a flow chart is shown in Fig. 5.3. The analysis can be summarized in four major tasks.

(1) Estimate support fragilities.

(2) Calculate structural responses for all support failure combinations.

(3) Estimate the conditional pipe failure probabilities at weld joints for all support conditions.

(4) Perform system failure analyses for all support failure combinations.

In the first task, fragilities of the supports are estimated. The values of P[no SF|a] and P[SFi|a] for a given earthquake peak ground acceleration level, a, can be calculated from the fragility curves of the supports. For each earthquake intensity, P[SFi|a] is simply the product of individual support failure probabilities of the "i"th combination scenario as stated in assumption number 3. P[no SF] is equal to 1.0 minus the sum of all cases of support failure probabilities.

In cases of support failure, the seismic responses of a piping system are different from that of the system without support failure. The structural responses for each case should be estimated separately depending upon the specific support failure combination. This estimation is the second task in assessing the effects of support failure. The regular seismic analysis process can be used starting with preparing the seismic analysis model, followed by either response spectrum or time history analysis, and ending with the calculated seismic stresses at each weld joint.

Once the seismic stresses are calculated, a probabilistic fracture mechanics analysis is then followed for each case of support failure combination. This analysis is the third task. This analysis yields the conditional failure probabilities at weld joints conditioned on the occurrence of an earthquake of specific intensity and the occurrence of a specific support failure scenario.

The probabilistic fracture mechanics methodology described earlier is a rather complicated procedure and the study of support failure effects does not warrant this level of sophistication. A simplified procedure was developed in this study to estimate the values of P[PF|no SF] and P[PF|SFi] and is described in Vol. 2 of this report series.

A system failure analysis, the last task in Fig. 5.3, can be performed to fold in the various results, such as the P[PF|SFi,a], P[SFi|a], and the seismic hazard curves to calculate the probability of failure of a piping system for each support failure scenario. The probability of overall system failure, including all support failure scenarios, can therefore be obtained as simply the sum of the system failure probabilities of the cases according to Eq. (5-7).

5.3.3 Support Fragility

Three kinds of pipe supports were used in the the recirculation loops of the pilot plant; these supports are representative of those used in most vital piping in nuclear power plants (except PWR reactor coolant loops). These pipe supports are the rigid supports (or anchors), the spring hangers, and the hydraulic snubbers. The hangers and the snubbers are by themselves supported by structural members. These structural members are, by the requirements of manufacturers' design specification, much stronger than the hangers and the snubbers. Therefore, there is no need to examine the failure mode of the supports due to failure of these structural members in estimating overall support fragility.

The reactor pressure vessel provides a rigid support for the recirculation loops since the reactor vessel is massive and the recirculation loop comes out and returns back to the reactor vessel to form a loop. Failure of reactor vessel supports would most likely induce the recirculation loop to fail. The conditional failure probability of the piping (given that the reactor vessel supports have failed) can be assumed to be unity in this case. This scenario is the same as for the PWR reactor coolant loops. Therefore, the falling down of the reactor pressure vessel is addressed in the same fashion as the earlier indirect pipe failure and is documented in Vol. 2 of this report series. Here we focus our attention only on the cases where the conditional failure probability is not necessarily equal to unity.

Spring hangers are used to support the dead weight of the piping system; the snubbers are used to resist the seismic loads during an earthquake event. Two kinds of spring hangers were used. Constant spring hangers support the recirculation pumps, and variable spring hangers support the the coolant pipes. Hanger failure is not considered in this support failure analysis as discussed in the following paragraph.

The stiffness of the spring hangers is much less than the stiffness of the piping and the active snubbers. During an earthquake, movement of the piping system is mainly restricted by the snubbers and the rigid supports of the piping system. The increase in load in the hangers is expected to be insignificant compared to the snubbers. This expectation implies that there will be no significant difference in the hanger failure probabilities during operation or during an earthquake. On the other hand, the load in snubbers is zero at all times except during a seismic event, during which the load can be very high depending on the earthquake intensity. If a spring hanger did not fail before the earthquake, it is unlikely to fail before the snubbers fail during an earthquake. Therefore, it is reasonable to neglect hanger failure during an earthquake in this study.

Figure 5.4 shows the pilot plant recirculation loops, Fig. 5.5 the corresponding pipe support arrangement. The two loops A and B are essentially identical; in this demonstration analysis, we chose to study Loop B.

There are nine snubbers in four natural groups for the Loop B of the recirculation system, with bore diameters ranging from 3.25 to 6 inches. One snubber supports the suction line. Two are in the discharge line at the same location except in different orientations. There are three snubbers each for the pump motor at the top and the casing at the bottom of the recirculation pump. Each group as a unit provides support for a specific part of the piping system. It is reasonable to assume that if one in the group fails, the other snubbers in the same group would also lose their function. This conservative assumption simplifies the problem and makes it easier to handle than considering all nine snubbers as individual supports.

In this study the fragilities of all nine snubber supports were estimated first; then, the fragilities of the four support groups were calculated based on the assumption that the support (or the snubber) failure events are statistically independent (Fig. 5.6). If any one of the snubbers in a specific group fails, the whole group is assumed to have lost its function.

The particular snubbers used in the subject recirculation system offer an optional relief valve which, if installed, will protect the snubber from being damaged if the dynamic load exceeds the load limit set for the relief valve. The relief valve opens when the load reaches the preset limit so the hydraulic pressure will not continue to build up inside the cylinder. When the load reduces, the valve closes and the snubber is ready to take more load. Thus, a snubber functions like an elastic-plastic axial load member. The snubbers in the recirculation loops are equipped with such a relief valve and are set to open at 133% of the rated load. However, the test results indicated that the minimum valve opening load is actually 160% of the rated load. Under a very high earthquake load, it is possible that the load on the snubbers may exceed this valve opening load. In this situation, the snubbers behave like a non-linear structural member with a large energy absorption capability due to its plasticity effect. Note that unless specified otherwise, all snubber fragilities are based on no relief valve being present.

Many failure modes of the subject snubbers were identified in our evaluation. Based on dynamic test results, the governing failure mode is the tensile failure of the threads at the piston rod end nut inside the cylinder. All of these failure modes (including the thread failure) showed higher capacity than the valve opening load. The minimum capacity of these failure modes is still about a factor of 1.8 or more than the relief valve opening load. This result is consistent with the relief valve design concept of protecting the snubber assembly from being overstressed. The snubber with relief valve does not simply fail when the relief valve capacity is reached; the snubber just goes into "plastic" deformation. It would be grossly conservative to consider the relief valve opening load as the fragility level of the snubber. Therefore, the nut thread failure will be considered in this study as the best-estimate failure mode. However, fragility estimates based on a relief valve opening load were also developed for the purpose of a sensitivity study.

With the fragilities of the four support groups developed, the next step is to calculate the failure probabilities at different earthquake intensity levels for the various support failure scenarios or combinations. As stated earlier there are 15 cases of support failure scenarios for a system with four individual supports. These combinations are presented in Table 5.2. Table 5.3 shows the failure probabilities at different earthquake levels for various support failure combinations.

If we were to follow the indirect DEGB approach adopted in our evaluation of RCLs by assuming P[PF|SF]=1.0, summing the probabilities of the 15 cases of support failure (Cases 2 through 16 in Table 5.3) would yield the probability of the recirculation loop DEGB indirectly caused by failure of intermediate supports, i.e. lifetime indirect DEGB probabilities of 2.4E-10, 3.2E-5, 4.3E-3, 5.9E-2 and 3.0E-1 for seismic hazard cutoffs of 1, 2, 3, 4, and 5 times the SSE, respectively. Some of these indirect DEGB probabilities are very high and may not be realistic judging from the current state (i.e. no DEGB having occurred) of the recirculation loops in the United States. To more realistically assess how support failure would actually affect the likelihood of pipe failure, we evaluated seismic responses for each failure case and then performed a series of probabilistic fracture mechanics analyses.

5.3.4 Seismic Responses Given Support Failure

To study the pipe failure induced by earthquake, we started with the calculation of the seismic stresses due to one earthquake level. Fifteen cases of seismic analysis of the recirculation loop B were performed and the corresponding pipe stresses were obtained using the response spectrum approach. Each case corresponds to one case of failure scenario of the support groups. Cases 2 to 16 of Table 5.2 show these combinations. Also included in Table 5.2 is Case 1, a support failure case in which no support failure occurs. All sixteen cases of seismic analysis were based on the OBE and a subsystem damping of 0.005. The seismic stresses due to other earthquake levels were estimated using the results of the design analyses and a series of response factors. In general, the suction line has the lowest average stress, and the risers have the highest. The discharge line has slightly higher average stress than the suction line. The discharge line is stiffer than the suction line because it has slightly thicker wall thickness and is shorter in length even though both lines have the same outside diameter.

To get a general idea about the stress situation in the pipe if several supports failed during an earthquake, the ratios of the maximum normal stresses (on the pipe cross section) for various support failure to the normal stress for no-support-failure case were calculated at individual weld joints. These stress ratios are presented in Fig. 5.7. As this figure shows, the seismic stress increases significantly if the supports fail during an earthquake. However, many more supports failing in an earthquake does not necessarily generate much higher stresses in the piping system. The implication is that the support-failure cases with large number of support failures will most likely contribute little to the overall system failure probability because the probability of so many supports failing in a seismic event is very low.

5.3.5 Simplified Analysis Method

In principle, accounting for stress redistribution caused by the failure of intermediate supports would require a separate PRAISE (or equivalent) calculation for each support failure scenario ("case"), dramatically increasing the computational effort associated with a probabilistic fracture mechanics assessment. For the four support groups identified in our study, sixteen separate PRAISE runs would have been required to cover all possible combinations and permutations of support failure (including the case of no failure). As part of our study, we performed sensitivity calculations to determine the relative contribution of each support failure case to the overall system probability of DEGB. In order to minimize computational effort, we developed a simplified analysis method based on modified versions of the standard pre- and post-processing routines used by PRAISE. These routines, normally used, respectively, to develop the stratified sampling space used by PRAISE and to perform the "systems analysis", execute much faster than PRAISE itself. Improved computational efficiency comes at the expense of accuracy in the probabilistic results; however, because we were addressing only relative effects in these sensitivity calculations, we concluded that the simplified analyses were sufficient for our purposes. Details on the particular techniques used in these simplified analyses are provided in Vol. 2 of this report series.

Following this simplified approach, the conditional failure probability for each weld joint was calculated for all 15 cases of the support failure scenario. The conditional failure probabilities of the weld joint for the no-support-failure case were not obtained in this fashion even though the same method applies, because they are already available from a rigorous PRAISE analysis of "direct" DEGB probability without support failure.

5.3.6 Probability of Pipe Break

The conditional pipe failure probabilities of individual weld joints for each case of the failure scenario were calculated following the simplified methodology described above using the pipe stresses obtained from the seismic analyses along with other operating stresses due to dead weight, pressure, and thermal expansion. System failure probability analyses were performed for each of these fifteen cases. These system failure probabilities were then combined with that of Case 1 following Eq. (5-7) to obtain the overall probability of seismically induced system failure.

The seismic hazard curve used in the system analysis was a generic curve based on an SSE of 0.16g peak ground acceleration. Because no seismicity data is available at very high levels of earthquake intensity (above one SSE), there is the question about how far the seismic curve should be extrapolated or truncated. That is, there exists a large modeling uncertainty in seismic hazard curves in the high earthquake intensity level.

To study the effect of different levels of extrapolation or truncation of the seismic hazard curve, several system failure analyses for various levels of truncation were performed. Five truncation levels were considered: 1, 2, 3, 4, and 5 times the SSE. The truncated seismic hazard curves are shown by the dashed lines in Fig. 5.8. The system failure probabilities are shown in Table 5.4 for the case when the effect of the relief valve is neglected. Table 5.5 presents the same results for the case in which the relief valve opening load is considered as the failure level of the snubbers.

In these tables, the probability of system failure for each of the fifteen support-failure scenarios (Cases 2 through 16) are presented for various seismic hazard truncations along with the probability of system failure for the no-support-failure scenario (Case 1). Note in Tables 5.4 and 5.5 that the total probability of system failure is a straight sum of the individual failure probabilities because each of the 15 support failure scenarios, as well as the no-support-failure case, are statistically independent of one another.

Tables 5.4 and 5.5 that the overall ("total") probability of system failure decreases as the seismic hazard curve is truncated at lower levels. The maximum probabilities of overall system failure are 4.8E-4 and 4.3E-6 per plant lifetime, respectively, for the cases with and without the relief valve at the seismic hazard curve truncation level of five times the SSE.

At first glance, these results appear to contradict the purpose of the snubber relief valve, i.e. to protect the snubbers against extreme seismic loads and thereby reduce the likelihood of overall pipe system failure. It is important to consider, however, that with the relief valve, snubber "failure" — defined as opening of the relief valve — would only be momentary, i.e. snubber function would be recovered as soon as the seismic load dropped below the snubber load limit. Without the relief valve, snubber "failure" would be just that — permanent loss of function — and therefore the corresponding fragility, based on structural capacity rather than a pre-set load limit, is accordingly higher than for a snubber with the relief valve. The issue of momentary vs permanent loss of function was <u>not</u> accounted for in estimating the respective probabilities of system failure; instead, for computational convenience, we treated relief valve "failure" as if it led to permanent loss of snubber function. How (or even <u>if</u>) momentary loss of snubber function would actually manifest itself as a pipe stress begins to address the time-dependent character of the seismic loads; evaluating this effect was beyond the scope of the current study. It seems reasonable, however, to expect that such pipe stresses would not act long enough to cause the pipe to fail, and that the actual probability of system failure would not only be significantly lower than that estimated above, but would also be lower than that for the same system equipped with snubbers having no relief valves.

The following paragraphs briefly summarize the findings of this study, which can be used to reduce the amount of work needed to assess accurately the effects of seismically induced support failures.

(1) The maximum probabilities of overall system failure are 2.0E-4 and 3.0E-6 per plant lifetime, respectively, for the cases with and without the relief valve at the seismic hazard curve truncation level of five times the SSE (or about 5.0E-6 and 7.5E-8 per

reactor-year, respectively, if a 40-year plant lifetime is assumed). These probability levels can be considered as the upper bound values. They are not very high probability values considering the fact that the case associated with the relief valve very conservatively assumes the valve opening load limit to be the "failure" (or fragility) level of the supports.

If we only consider earthquakes up to twice the SSE, the lifetime probabilities of failure with and without the relief valve drop to 7.7E-8 and 1.7E-11, respectively (or about 1.9E-9 and 4.3E-13 per reactor-year, respectively).

(2) The seismic stresses of cases when many supports fail during an earthquake are not significantly higher than those cases in which only one or two supports fail. It is unlikely that these cases will have any significant contribution to the overall system failure because the probability of so many supports failing in an earthquake is very small.

(3) The welds which have high seismic stress in a no-support-failure case are most likely the dominate welds for the overall system failure probability. The welds with low seismic stress in Case 1 may have higher rates of stress increase for the with-support-failure cases from Case 1. However, the higher rate may still not make them major contributors to overall system failure.

(4) The shape of the seismic hazard curve has a major effect on the overall system failure probability. Seismic hazard curves which do not extend far beyond the one SSE level indicate that evaluation of the no-support-failure case might be sufficient. Otherwise, the with-support-failure cases dominate. Following these observations, the effects of support failure may be assessed with the evaluation of a few carefully selected welds and support failure combinations using the methodology presented in this section.

Besides the extrapolation or truncation of seismic hazard curves, a large modeling uncertainty also exists in the support fragilities. To study the effect of this uncertainty in support failure fragility, we considered two levels of uncertainties (90% and 10%) on the modeling uncertainty distribution of the support fragility. Table 5.6 shows the results for the 90% on the uncertainty distribution in the same format as Tables 5.4 and 5.5, which represent the median of the support fragility; as in Tables 5.4 and 5.5, the individual "SF" system failure probabilities will not necessarily sum to the total "SF" probability given. Not surprisingly, the overall system failure probability is again heavily dependent on the truncation level of the seismic hazard curve. At 90% on the modeling uncertainty distribution, the maximum overall system failure probability reaches 7.3E-4 per plant lifetime at the seismic hazard truncation level of 5SSE. It is close to the maximum probability of 3.0E-4 calculated for the case when the relief valve opening load was considered as the fragility level (see Table 5.5). This probability level is still not high.

Volume 2 of this report series describes our BWR "intermediate" support evaluations in greater detail.

Table 5.1. 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 sites) • Fundamental frequency of internal structure if uncoupled analysis was performed • Interface spectra for NSSS points of connection to structure if uncoupled analysis was performed • Input ground spectra resulting from synthetic time history applied to structural model
NSSS Response
• Method of analysis (e.g., time-history or response spectrum) • Modeling of NSSS and structure (i.e. 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 5.2. Support failure combinations considered for recirculation loop B.

Case No.	No. of failed supports	Group 1 SB4,SB5,SB6 Pump Motor	Group 2 SB1,SB2,SB3 Pump Casing	Group 3 SB10 Suction	Group 4 SB12,BB12 Discharge
1	0 SF				
2	1 SF	x			
3	1 SF		x		
4	1 SF			x	
5	1 SF				x
6	2 SF	x	x		
7	2 SF	x		x	
8	2 SF		x	x	
9	2 SF	x			x
10	2 SF		x		x
11	2 SF			x	x
12	3 SF	x	x	x	
13	3 SF	x	x		x
14	3 SF	x		x	x
15	3 SF		x	x	x
16	4 SF	x	x	x	x

Table 5.3. Probability of support failure at various levels of earthquake intensity.

Case	SF*	Maximum Earthquake Level					
		0.5 SSE	1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
1	0	.1000E+01	.1000E+01	.1000E+01	.9957E+00	.9417E+00	.7264E+00
2	1	.1729E-16	.1879E-09	.2100E-04	.2381E-02	.2833E-01	.1207E+00
3	1	.2289E-17	.4206E-10	.8191E-05	.1315E-02	.2007E-01	.1018E+00
4	1	.2432E-26	.3912E-17	.6131E-10	.1177E-06	.9775E-05	.1767E-03
5	1	.3247E-18	.9362E-11	.2791E-05	.5616E-03	.9920E-02	.5618E-01
6	2	.3958E-34	.7902E-20	.1720E-09	.3130E-05	.5686E-03	.1229E-01
7	2	.4204E-43	.7351E-27	.1288E-14	.2802E-09	.2769E-06	.2132E-04
8	2	.5566E-44	.1645E-27	.5022E-15	.1547E-09	.1962E-06	.1799E-04
9	2	.5613E-35	.1759E-20	.5860E-10	.1337E-05	.2810E-03	.6779E-02
10	2	.7432E-36	.3937E-21	.2286E-10	.7383E-06	.1991E-03	.5720E-02
11	2	.7894E-45	.3663E-28	.1711E-15	.6610E-10	.9697E-07	.9926E-05
12	3	.9624E-61	.3092E-37	.1055E-19	.3683E-12	.5558E-08	.2171E-05
13	3	.1285E-52	.7398E-31	.4800E-15	.1758E-08	.5640E-05	.6903E-03
14	3	.1365E-61	.6882E-38	.3593E-20	.1574E-12	.2747E-08	.1198E-05
15	3	.1807E-62	.1540E-38	.1401E-20	.8689E-13	.1946E-08	.1011E-05
16	4	.3125E-79	.2894E-48	.2943E-25	.2069E-15	.5514E-10	.1220E-06
Total P[SF]		.199E-16	.239E-09	.320E-04	.426E-02	.594E-01	.304E+00

* number of support failures for case indicated. Case 1 represents the probability that no supports fail.

Table 5.4. Best-estimate seismically induced pipe failure probability (without considering relief valve) and the effects of seismic hazard curve extrapolation.

Case	Failed Supports	Maximum Earthquake Level				
		1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
1	0	.5971E-11	.7837E-11	.8882E-11	.9403E-11	.9754E-11
2	1	.2289E-20	.1184E-14	.5064E-12	.1303E-10	.6170E-09
3	1	.2139E-20	.6142E-14	.1273E-10	.9298E-08	.8913E-06
4	1	.1153E-28	.3215E-21	.7304E-18	.8469E-16	.3060E-14
5	1	.1957E-19	.9124E-11	.1521E-06	.7618E-06	.1684E-05
6	2	.1070E-29	.4974E-18	.4923E-12	.4169E-08	.2147E-06
7	2	.1162E-37	.1282E-24	.1222E-18	.3140E-15	.1211E-11
8	2	.5697E-37	.6615E-23	.8322E-10	.1063E-10	.3116E-09
9	2	.9460E-29	.128E-14	.3445E-09	.1659E-07	.1343E-06
10	2	.4688E-29	.3737E-14	.1909E-09	.1162E-07	.1106E-06
11	2	.1556E-37	.5982E-23	.1217E-13	.5388E-11	.1708E-09
12	3	.5456E-48	.1082E-29	.2183E-21	.6912E-17	.1299E-13
13	3	.5578E-40	.3524E-22	.3993E-12	.3107E-09	.1180E-07
14	3	.2756E-46	.3475E-25	.3991E-16	.1491E-12	.1942E-10
15	3	.5060E-47	.6237E-26	.2204E-16	.1054E-12	.1631E-10
16	4	.1137E-58	.1034E-34	.4881E-24	.1413E-17	.1919E-12
Total P[DEGB]		.597E-11	.170E-10	.153E-06	.804E-06	.305E-05

Table 5.5. Best-estimate seismically induced pipe failure probability (considering relief valve) and the effects of seismic hazard curve extrapolation.

Case	Failed Supports	Maximum Earthquake Level 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
1	0	.5971E-11	.7661E-12	.7807E-11	.7725E-11	.7723E-11
2	1	.8537E-16	.8601E-13	.4645E-10	.3140E-09	.4685E-08
3	1	.7044E-14	.6165E-11	.5519E-08	.4314E-06	.9499E-05
4	1	.4109E-21	.8925E-17	.1309E-13	.2275E-12	.1967E-11
5	1	.1617E-14	.1501E-08	.3136E-04	.5385E-04	.6106E-04
6	2	.1210E-18	.3489E-12	.1723E-07	.4012E-05	.2111E-04
7	2	.1453E-25	.2518E-17	.1787E-12	.1656E-10	.4999E-08
8	2	.6532E-23	.1793E-14	.5753E-07	.1378E-05	.2970E-05
9	2	.2755E-19	.1500E-09	.5972E-05	.2096E-04	.2959E-04
10	2	.1217E-17	.6045E-08	.1732E-04	.3994E-04	.4789E-04
11	2	.4474E-25	.2712E-15	.3845E-07	.6351E-06	.1881E-05
12	3	.2157E-29	.2089E-18	.1293E-12	.1652E-10	.5534E-09
13	3	.5135E-24	.4062E-13	.2594E-05	.1642E-04	.2542E-04
14	3	.2814E-29	.1123E-14	.1085E-07	.3476E-06	.1457E-05
15	3	.4604E-28	.2782E-14	.2888E-07	.5587E-06	.1805E-05
16	4	.3676E-35	.3285E-20	.5362E-13	.1328E-09	.1005E-06
Total P[DEGB]		.597E-11	.770E-07	.574E-04	.139E-03	.203E-03

Table 5.6. The effects of uncertainty in estimating support fragility on the seismically induced pipe failure probability (90% on the uncertainty distribution of the support fragility).*

Case	Failed Supports	Maximum Earthquake Level				
		1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
1	0	.5971E-11	.7826E-11	.8667E-11	.8692E-11	.8630E-11
2	1	.7160E-17	.2048E-12	.1863E-10	.1769E-09	.3576E-08
3	1	.9488E-17	.1550E-11	.6646E-09	.1566E-06	.6281E-05
4	1	.8799E-24	.1249E-17	.5221E-15	.1862E-13	.2755E-12
5	1	.1140E-15	.2939E-08	.1042E-04	.2275E-04	.2909E-04
6	2	.1442E-22	.2137E-13	.8870E-09	.8669E-06	.9661E-05
7	2	.2709E-29	.8499E-19	.3005E-14	.8291E-12	.5764E-09
8	2	.1891E-28	.6442E-17	.2969E-09	.3929E-07	.2181E-06
9	2	.1684E-21	.7053E-10	.8141E-06	.5345E-05	.1083E-04
10	2	.1189E-21	.3024E-09	.6652E-06	.4997E-05	.1063E-04
11	2	.6826E-29	.7472E-17	.5436E-09	.2570E-07	.1562E-06
12	3	.5561E-36	.1801E-21	.2738E-15	.3024E-12	.4243E-10
13	3	.4344E-29	.4878E-15	.4605E-07	.1496E-05	.5574E-05
14	3	.3715E-34	.7422E-17	.6277E-10	.8530E-08	.9609E-07
15	3	.9720E-35	.1957E-17	.5059E-10	.8147E-08	.9687E-07
16	4	.6712E-43	.5549E-24	.3917E-16	.1286E-11	.5471E-08
Total P[DEGB]		.597E-11	.332E-08	.120E-04	.357E-04	.726E-04

* no snubber relief valve

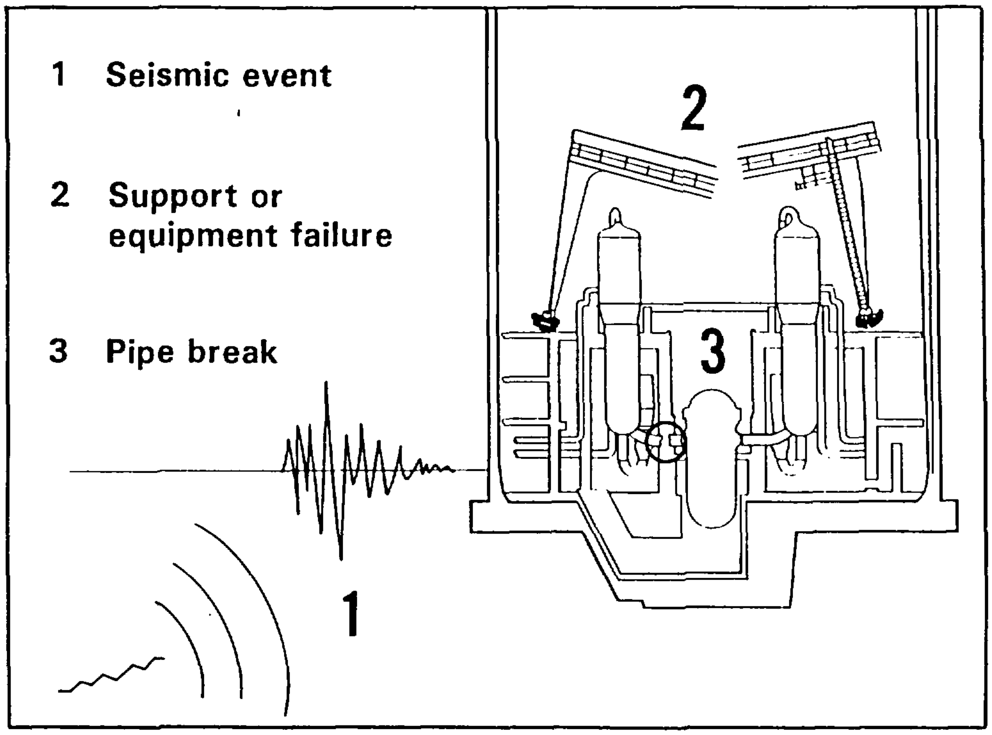


Figure 5.1. General approach for estimating probability of "indirect" DEGB due to seismically-induced failure of heavy component supports.

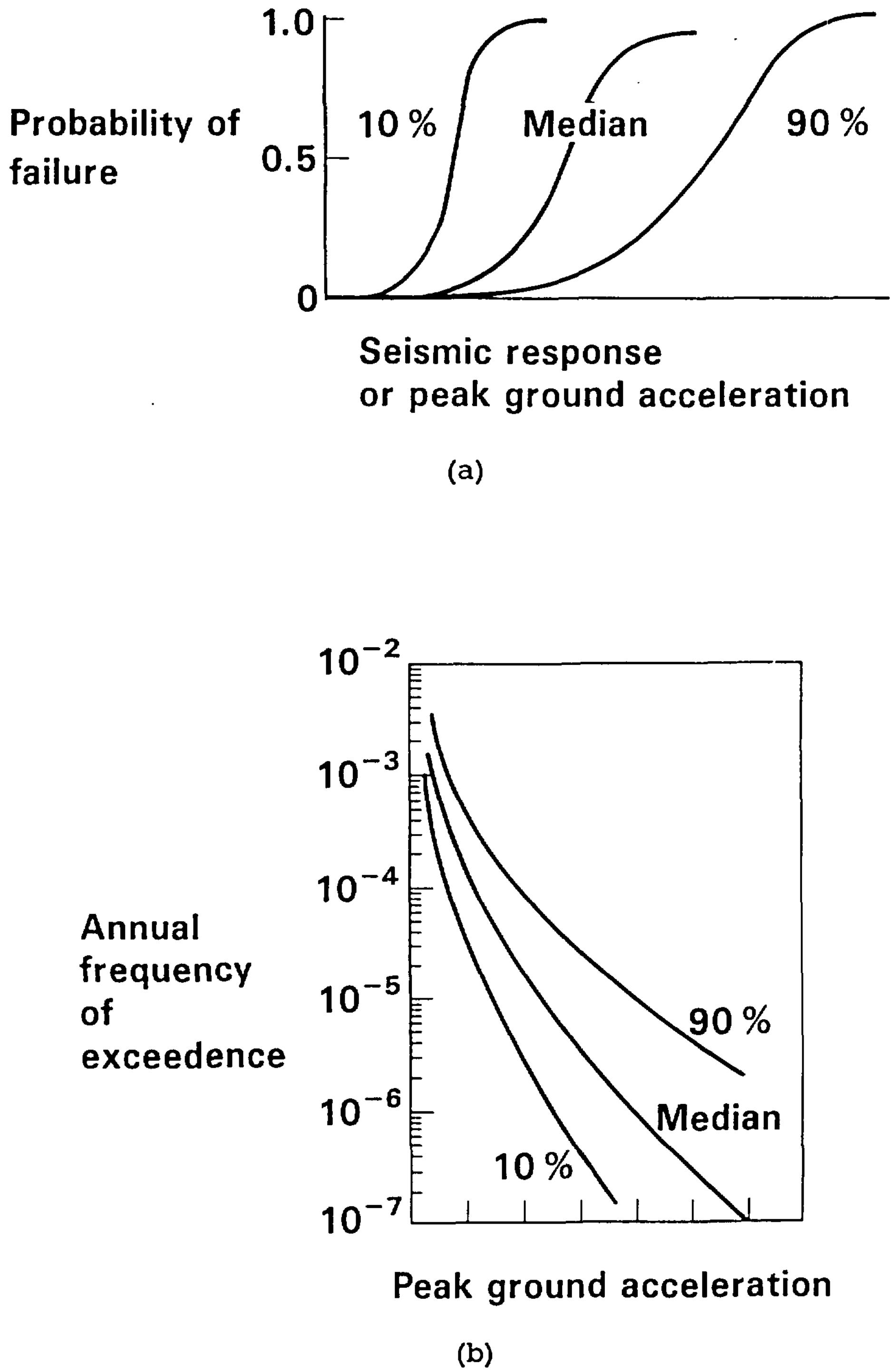


Figure 5.2. Typical descriptions of component fragility (a) and seismic hazard (b).

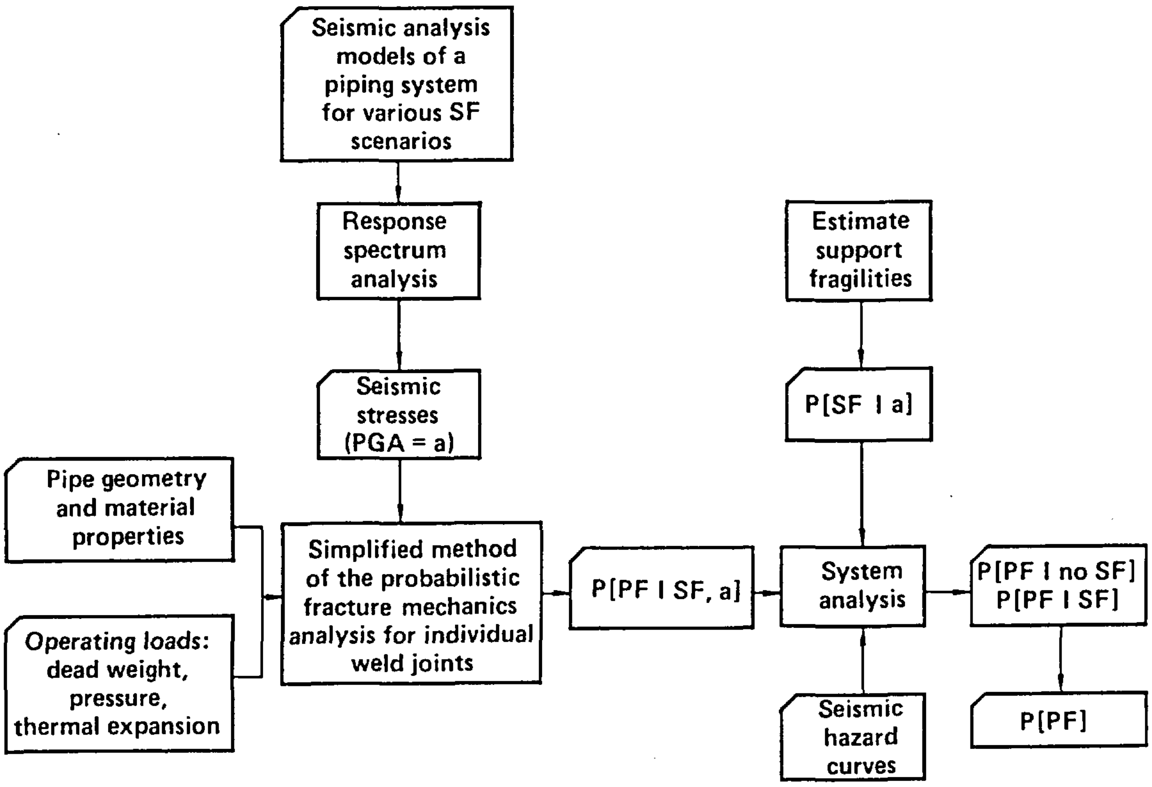


Figure 5.3. General approach used to study seismically-induced system failure considering the effects of support failure.

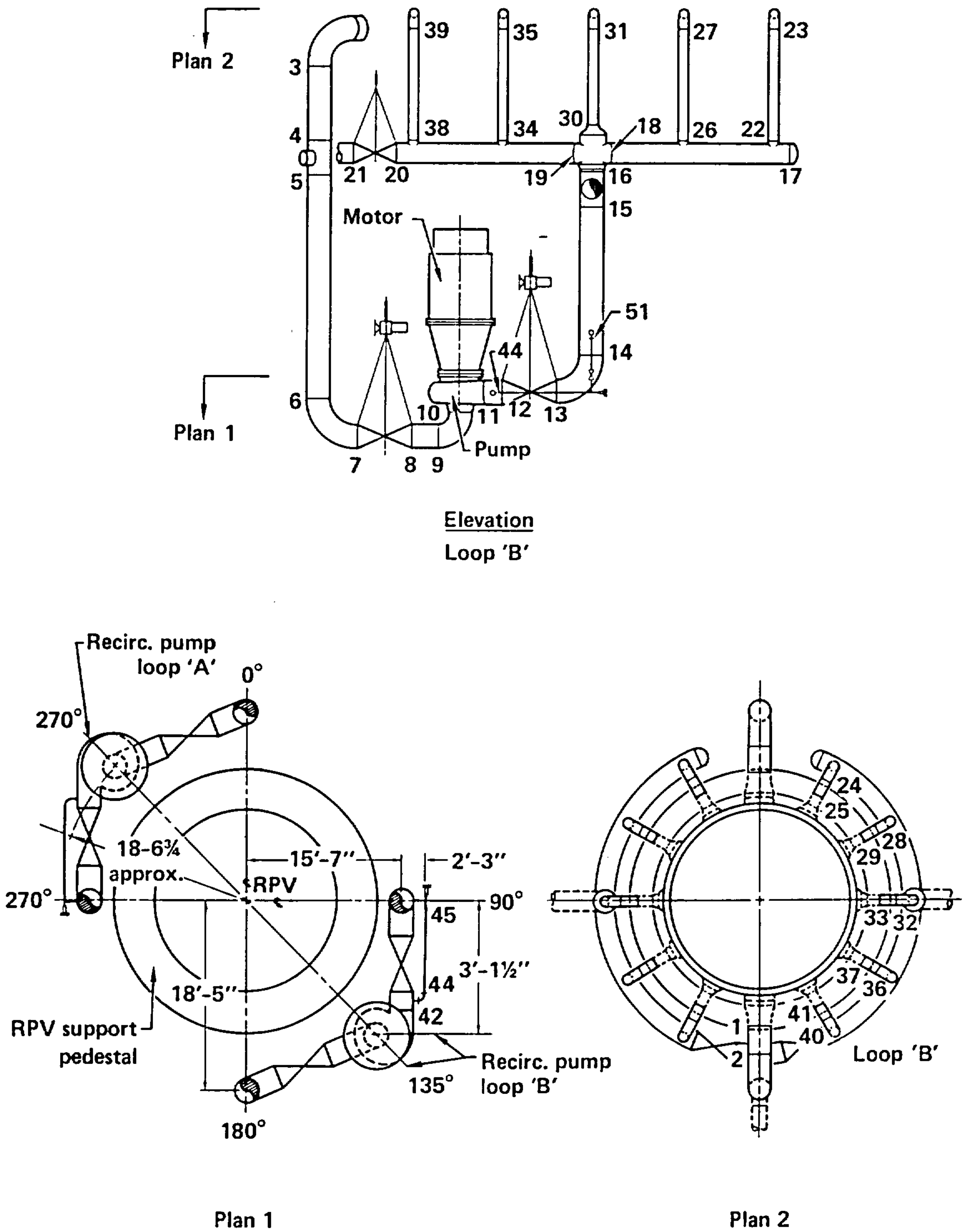


Figure 5.4. Brunswick recirculation system.

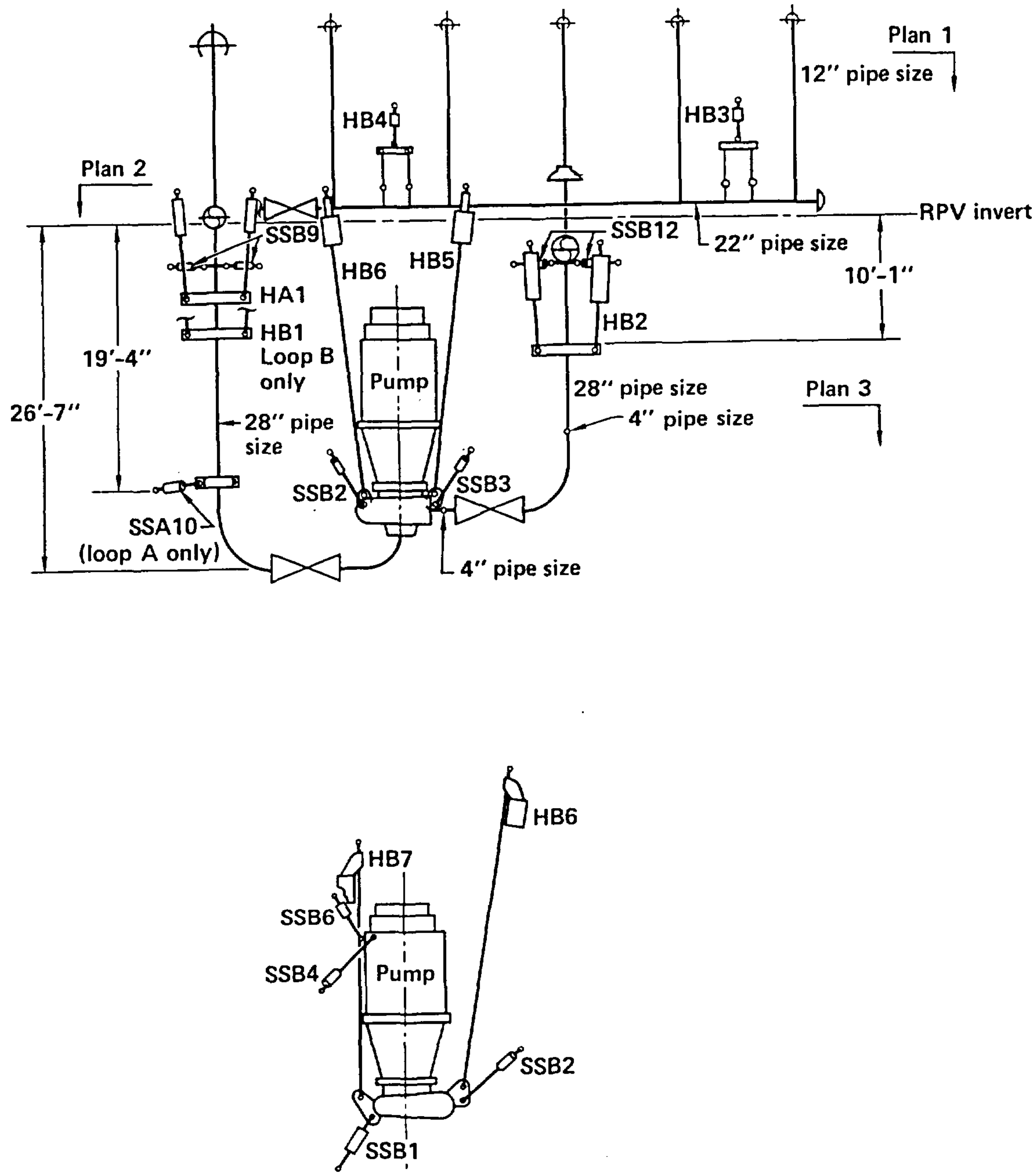


Figure 5.5. Supports of the Brunswick recirculation system.

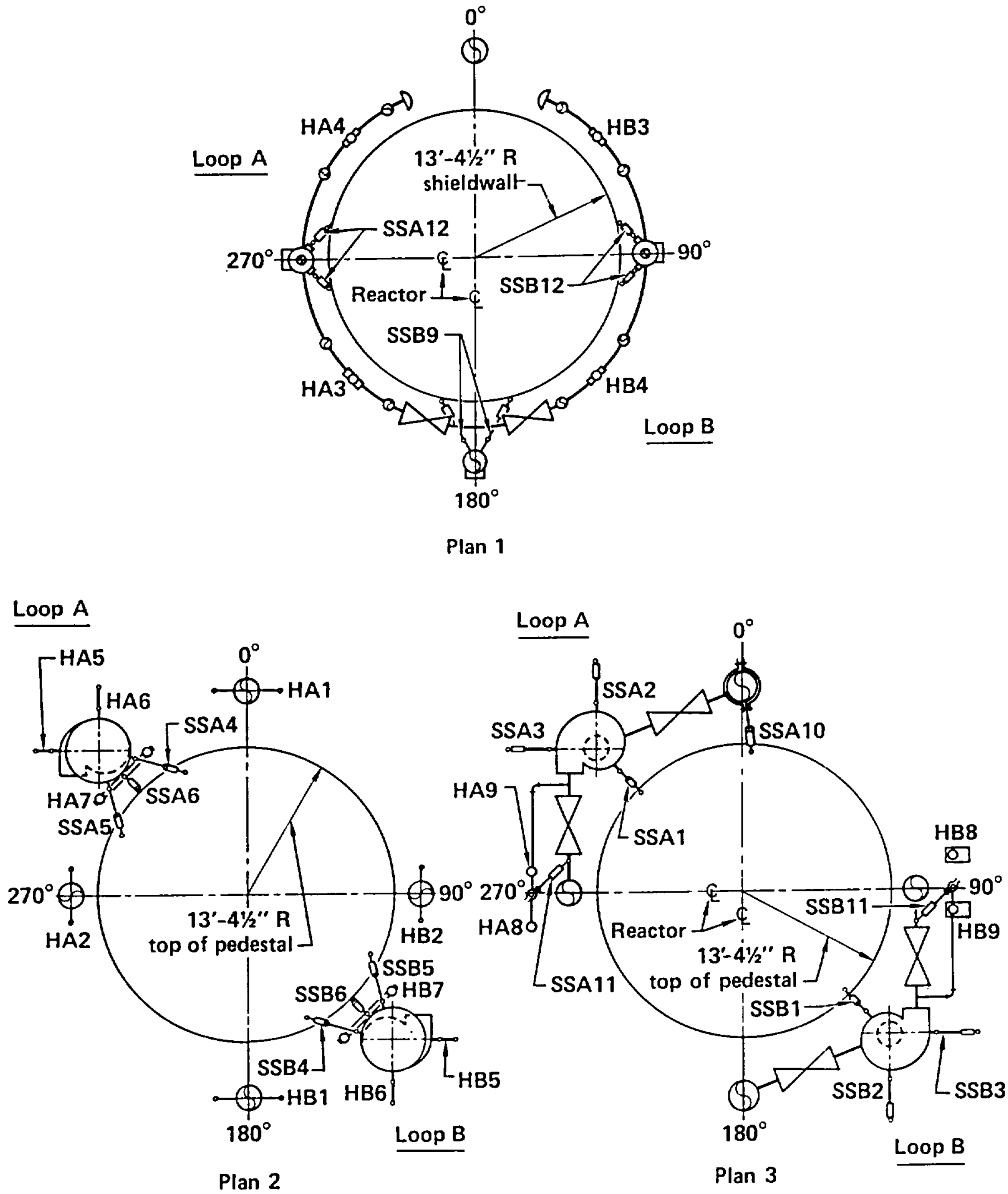
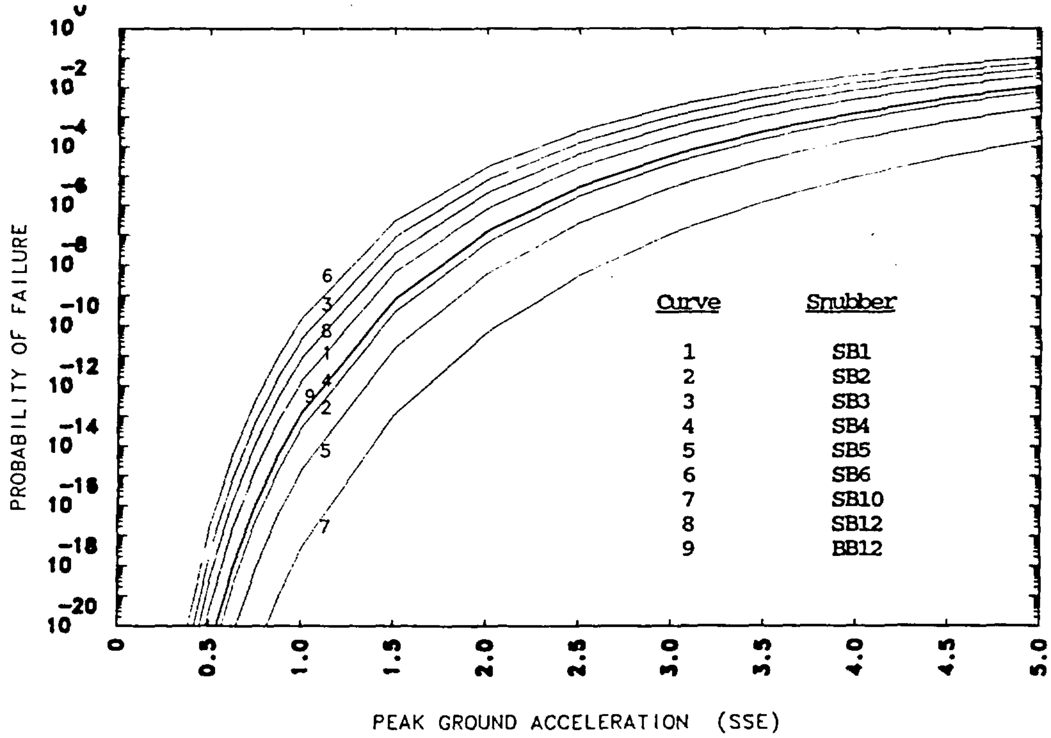
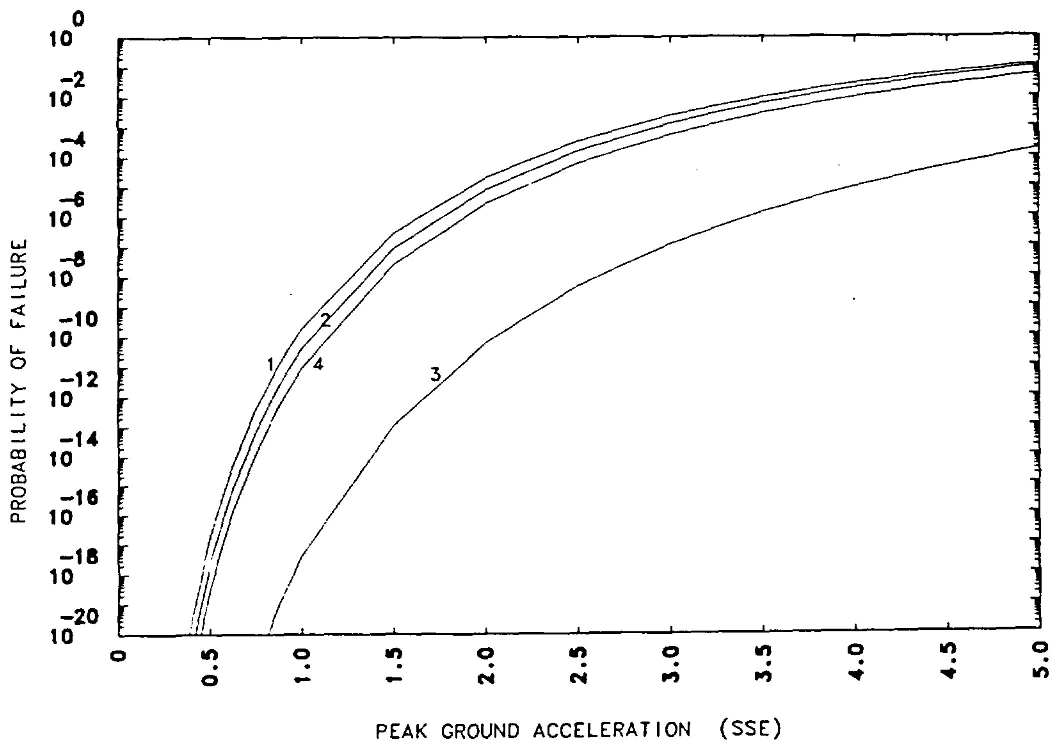


Figure 5.5 (cont.). Supports of the Brunswick plant recirculation system.

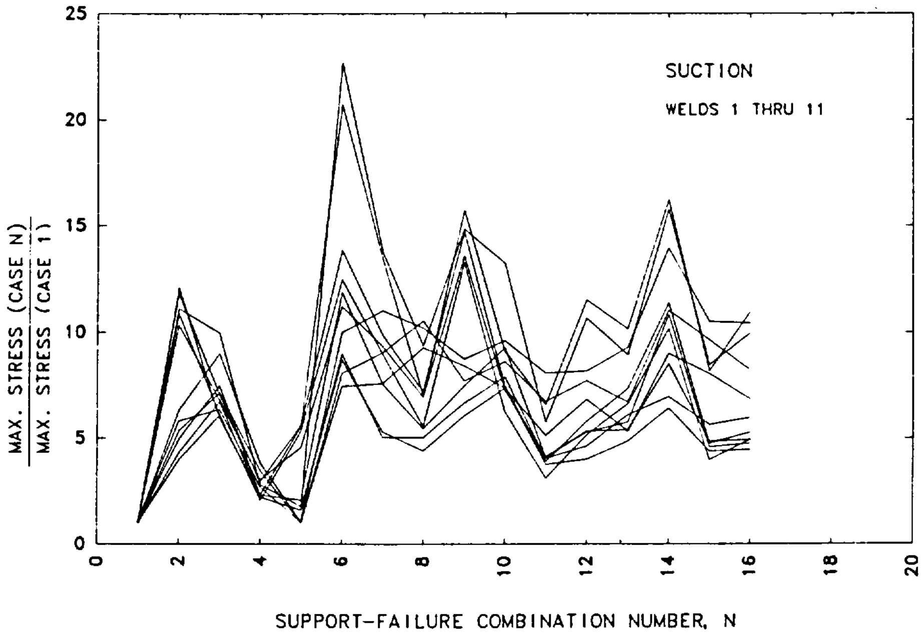


(a)

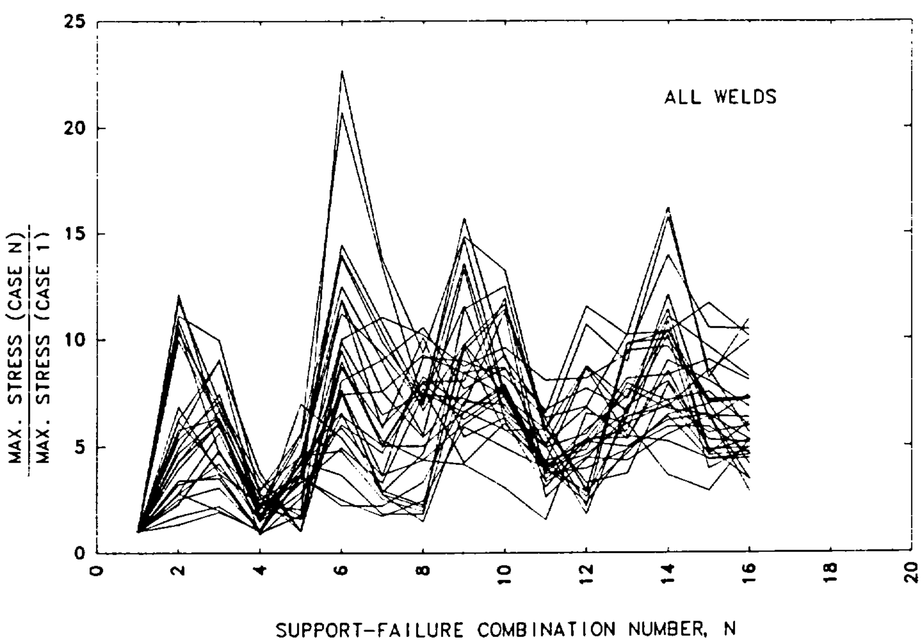


(b)

Figure 5.6. Failure probabilities of (a) nine individual pipe supports and (b) four support groups.



(a)



(b)

Figure 5.7. Increase in seismic stresses at (a) weld joints of the suction line and (b) all recirculation loop weld joints, due to support failure.

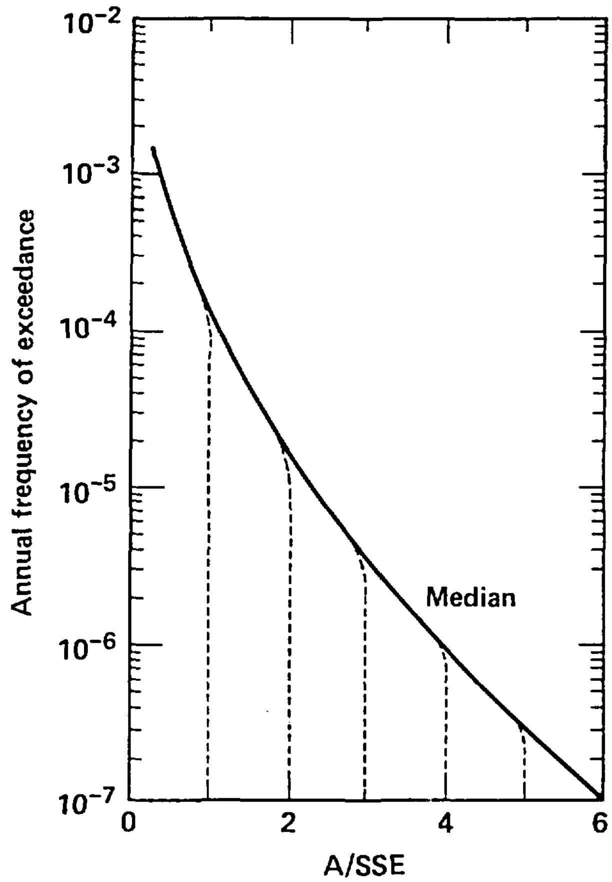


Figure 5.8. Seismic hazard curve used in intermediate support failure evaluation; dashed lines show truncations at indicated PGA levels.

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6. SUMMARY AND CONCLUSIONS

6.1 Pipe Failure Due to Direct Causes

Using the Brunswick Mark I BWR plant as a case study, we completed probabilistic analyses indicating that the probability of direct DEGB is very low for BWR main steam and feedwater piping, and for BWR recirculation loop piping if stress corrosion cracking is not a factor. 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 to postulated earthquakes. We also considered other factors, such as the capability to detect cracks by non-destructive examination and the capability to detect pipe leaks, in our analyses.

The best-estimate lifetime system leak and DEGB probabilities for the Brunswick major coolant piping systems are rather low and fall within narrow ranges. For the three piping systems considered, the system leak probabilities vary between 2.4E-6 and 3.8E-5 over the life of the plant, or between about 6.0E-8 and 1.0E-6 per reactor-year. The DEGB probabilities behave similarly, ranging from 4.0E-11 to 1.5E-10 over the lifetime of the plant, or about 1.0E-12 to 3.8E-12 per reactor-year. These results are similar to those estimated for PWR reactor coolant piping, both in absolute magnitude and in the differential (three or more orders of magnitude) between the probabilities of leak and break.

We also performed a rigorous uncertainty study, the results of which showed that even at the 90th-percentile confidence level, the highest lifetime probabilities are 1.2E-3 and 5.0E-8 for the leak and the DEGB, respectively, or about 3.0E-5 and 1.3E-9 per reactor-year. Both of these probabilities are for the feedwater line, and are still comparable to our earlier PWR results despite the increased complexity of the BWR systems considered.

In order to address the effect of stress corrosion cracking (SCC) on the probability of failure in BWR recirculation loop piping, we developed an advanced SCC model for the PRAISE code. This semi-empirical model is based on experimental and field data compiled from several sources. Using probabilistic techniques, the model addresses various stress corrosion phenomena, including crack growth, crack initiation, and linking of multiple cracks. The model also considers the effect of residual stresses in addition to cyclic stresses resulting, for example, from normal plant operation.

The model covers not only the Type 304 stainless steel found in older BWR recirculation piping, but also Type 316NG ("nuclear grade"), a low-carbon alloy widely regarded as an SCC-resistant replacement for Type 304. Crack growth rates and times-to-initiation for each material are correlated against "damage parameters" which consolidate the separate influences of several individual parameters. The damage parameters are multiplicative relationships among various terms which individually describe the effects of the various phenomena on SCC behavior, including (1) environment, specifically coolant temperature, dissolved oxygen content, and level of impurities, (2) applied loads, including both constant and variable loads to account for steady-state operation and plant loading or unloading, respectively, (3) residual stresses, and (4) material sensitization.

After completing our development work, we applied the model to the recirculation loop piping in an actual Mark I BWR plant. We estimated the leak and DEGB probabilities both for an existing recirculation loop and for a proposed "replacement" configuration having fewer weld joints than the original system. We also investigated the relative effects of Types 304 and 316NG stainless steel on the estimated probabilities of leak and break. From the results of these evaluations, we observed that:

- recirculation loops fabricated from Type 304 stainless steel are predicted to leak after about 10 years of operation, a result consistent with some field observations. If the 304SS material is replaced by 316NG and the existing loop configuration is retained, the system leak probability at ten years is effectively zero. The end-of-life system leak probability (i.e. after another 30 years of operation) is about 5E-1 per loop, or about 2E-2 per loop-year assuming "worst case" applied stresses and no ISI. The replacement configuration, with its fewer welds, reduces leak probabilities still further.

- for recirculation loops fabricated from Type 304 stainless steel, the system probability of DEGB is about 1E-2 after ten years of operation (or about 1E-3 per loop-year), increasing to about 2E-2 by the end of plant life. For recirculation loops fabricated from Type 316NG, the system probability of DEGB is zero for the first 30 years of operation, even under "worst case" applied stresses and no ISI, and on the order of 1E-4 per loop-year or less over the final ten years of plant life.

In all cases, the 316NG appears to owe its corrosion resistance mainly to the fact that (1) fewer cracks initiated than in the 304SS material, and (2) those that did initiate typically did so later in plant life. Once a crack initiates, however, its subsequent growth rate is not significantly affected by material type. We also found that system failure concentrates at bypass welds (if present) and at riser welds. Further investigation of the relative susceptibility of recirculation loop welds to crack-induced failure is currently in progress for NRC.

We present these results as "best-estimate" values, although many of the input parameters used in the analyses can be regarded as conservative. Due to the excessive computer time requirements of the stress corrosion calculation (compared, for example, to that for thermal fatigue only), we were unable to perform extensive uncertainty analyses within the time and resources available to us. The results of analyses performed during model development, however, indicate that the probability of SCC-induced pipe failure (either leak or DEGB) is most sensitive to the description of residual stress assumed. Therefore, we would expect the uncertainty in the estimated failure probabilities to be significant, owing to variations in plant-to-plant residual stresses.

6.2 Pipe Failure Due to Indirect Causes

We have developed detailed approaches for evaluating how support failures caused by earthquakes would contribute to the overall probability of piping system failure. Two different approaches are used, depending on the type of support considered:

(1) a "support reliability" approach for heavy component supports, that is, supports whose failure could reasonably be expected to cause pipe failure under all circumstances, and

(2) a more rigorous "piping reliability" approach, where the effect of support failure is incorporated directly into a probabilistic fracture mechanics evaluation to determine if and under what conditions support failure would cause a pipe to break.

In evaluating the probability of "indirect DEGB", we first identified critical components whose failure could plausibly result in a pipe break; in our BWR study, seismically-induced failure of supports for piping and components was determined to be the most likely cause of an indirect pipe break. We then developed a "fragility" description for each critical support relating its probability of failure given the occurrence of an earthquake of given peak ground acceleration. Finally, we estimated the non-conditional probability of support failure ("support reliability") by convolving the support fragilities with an appropriate description of seismic hazard. "Seismic hazard" relates the probability of an earthquake exceeding a given level of peak ground acceleration. Both approaches incorporated these same basic steps; the piping reliability approach, however, goes one step further by explicitly estimating the probability of pipe failure given support failure.

6.2.1 Pipe Break Caused by "Heavy Component" Support Failure

The only critical "heavy component" supports that we considered in our BWR evaluation were those making up the reactor support structure. These included the lower support structure at the base of the reactor pressure vessel, as well as the lateral stabilizers at the top of the vessel. We also considered many other plausible causes of indirect DEGB unrelated to earthquakes, such as crane failure and pump flywheel missiles, but determined these to be of negligible significance compared to support failure.

We found that the probability of indirect DEGB due to failure of heavy component supports was about 2E-8 events per reactor-year, with a 90th-percentile value (confidence limit) of 5E-7 per reactor-year. Our results further indicated that the "star" stabilizer at the top of the reactor pressure vessel, which restrains the RPV against lateral motion in the event of an earthquake, was the primary contributor to failure rather than the main support structure at the bottom of the vessel.

This result was comparable to those from our PWR reactor coolant loop evaluations, in which we investigated the seismically-induced failure of RPV, steam generator, and reactor coolant pump supports. For Westinghouse plants, the median probability of indirect pipe break was about 1E-7 per reactor-year for plants east of the Rocky Mountains (based on generic seismic hazard curves), and about 3E-6 per reactor-year for plants on the more seismically active west coast; "upper bound" (i.e. 90% confidence level) probabilities were typically about one order of magnitude higher. Equivalent results for Combustion Engineering and Babcock & Wilcox reactor coolant loop supports were comparable to the Westinghouse results.

6.2.2 Pipe Break Caused by Failure of "Intermediate" Pipe Supports

Our evaluations of "indirect" DEGB caused by heavy component support failure assumed that support failure unconditionally resulted in pipe break. This assumption was regarded as conservative, but nevertheless resulted in very low DEGB probabilities. To have assumed that failure of a snubber or a constant-load support would similarly cause a DEGB in BWR recirculation loop piping would have been unreasonably conservative; in other words, a simple "support reliability" evaluation would no longer suffice. We therefore developed a more sophisticated approach to incorporate the effect of support fragility into the probabilistic fracture mechanics evaluation, which we used to investigate the effect of support failure on the probability of DEGB. As discussed earlier, the need to incorporate support failure in the fracture mechanics evaluation blurs the distinction between "direct" and "indirect" DEGB, leading us back towards a more integrated approach for estimating the probability of pipe break.

The "piping reliability" approach that we applied to the recirculation loop in our study comprised four steps:

(1) Identify critical supports and support failure combinations; estimate support fragilities. Our study considered not only "conventional" pipe supports (e.g. spring hangers, snubbers), but the supports for the recirculation loop pump as well.

(2) Calculate structural responses (e.g. pipe stresses) for each combination of support failure. We considered 15 cases of support failure, in addition to the "no support failure" case.

(3) Estimate the conditional pipe failure probabilities at weld joints for all support failure combinations.

(4) Estimate the non-conditional system probability of pipe break for all support failure combinations.

The results of this study indicated that the maximum probability of recirculation loop DEGB due to failure of intermediate supports is about 3.0E-6 per plant lifetime (or about 7.5E-8 per reactor-year) when earthquakes up to five times the SSE are considered in the seismic hazard description. If we only consider earthquakes up to twice the SSE, the lifetime probability of DEGB drops to about 1.7E-11, or about 4.3E-13 per reactor-year.

These probabilities were low enough to allow failure of "intermediate" supports -- and the redistribution of weld joint stresses that would result -- to be omitted from our subsequent detailed evaluations of pipe break due to crack growth.

6.3 Current and Future Applications

6.3.1 Piping Reliability Assessments

The NRC Office of Nuclear Reactor Regulation (NRR) recently published NUREG-0313, Rev. 2, which describes methods acceptable for controlling the susceptibility of BWR reactor coolant piping to intergranular stress corrosion cracking [16]. Although the NRR staff prefers replacement of sensitive piping with piping fabricated from IGSCC-resistant materials such as Type 316NG, enhancement of existing piping by appropriate combinations of repair (e.g., weld overlay, IHSI), prevention (e.g., hydrogen water chemistry), and augmented ISI is also an acceptable option for plant licensees. For example, the NRR guidelines specify various inspection intervals and sample sizes, depending on IGSCC mitigating measures that have been applied to an affected piping system, but do not define the specific welds that must be inspected.

The results of our recirculation loop evaluation indicated that the likelihood of pipe failure (i.e. leak or break) can vary widely among the weld joints in a piping system. Consequently, the specific welds selected at any given inspection could have a significant influence on system safety. As part of a new project for the NRC Office of Nuclear Regulatory Research, we are using the PRAISE computer code, and in particular our probabilistic model of stress corrosion cracking, to establish an inspection priority for BWR recirculation loop welds on the basis of calculated leak rates for the "representative" Mark I BWR plant in our earlier evaluation. Although not intended in itself to define an "acceptable" piping inspection program, it will provide NRR with one technical basis for reviewing utility responses to NUREG-0313, Revision 2.

The usefulness of probabilistic evaluations in regulatory applications has already been demonstrated through recent NRC rulemaking actions based in large part on the results of LLNL piping reliability studies. Although not a part of our present work, future licensing assessments related to the issue of stress corrosion cracking might conceivably include the following:

- developing specific licensing criteria. It is presumed that the criteria now included in NUREG-0313, Revision 2, will provide the basis for future NRR licensing decisions pertaining to BWR piping susceptible to IGSCC. Probabilistic evaluations like the one discussed in this paper could conveniently be applied to more fundamentally define just what constitutes an "acceptable" piping inspection program.

- assessing the effectiveness of the recommended inspection schedules relative to alternate inspection schemes (e.g. more or less frequent inspection, greater or lesser extent of inspection).

- assessing the effectiveness, either relative or absolute, of various measures for enhancing the performance of piping susceptible to stress corrosion cracking.

In principle, our probabilistic approach could be applied without modification to the first two of these activities. The approach could also be applied to the third given appropriate PRAISE code modifications, such as the capability to change residual stress patterns, coolant conditions, and pipe geometry at selected times during plant life to model, respectively, IHSI, hydrogen water chemistry, and weld overlay. Such capability would be a powerful tool for future licensing assessment and should be considered for further development and application.

6.3.2 Support Reliability Assessments

In the course of our evaluations of DEGB probability in reactor coolant piping, we have applied our "piping reliability" and "support reliability" approaches to various reactor coolant piping systems in both PWR and BWR plants. The results of our evaluations have typically shown that the likelihood of pipe break due to seismically-induced support failure is small, not only for the large, stiff piping found in PWR primary systems, but for more complex, more flexible piping systems as well. From the standpoint of addressing specific regulatory issues associated with piping behavior, a reliability approach also yields the following:

(1) the relative contribution of various failure scenarios to the overall likelihood of pipe system failure, in other words, the "safety significance" of each failure scenario.

(2) the relative "safety significance" of individual supports, in other words, identification of those supports whose failure would most serious affect system integrity.

(3) an assessment of system failure based on realistic failure criteria reflecting the actual behavior of the piping, rather than on simple exceedance of code allowable limits.

The general reliability techniques that we developed in this study could be applied to any piping system, given input data equivalent to that which we applied for recirculation loop piping. In principle, the support reliability techniques could, with appropriate modifications, also be applied to non-piping systems (e.g., cable trays and their supports) to, for example, investigate the relative influence of individual supports and support failure scenarios on overall system reliability.

6.4 Other Recommended Future Work

6.4.1 PRAISE Code Development

Our experience with PRAISE during the BWR evaluation indicated that inclusion of the SCC model dramatically increased computer time requirements for estimating DEGB probabilities. Where in our PWR evaluations a complete reactor coolant loop assessment -- including system analysis -- typically required about 90 CPU minutes on the CDC 7600 computer (or about five minutes per weld), estimating the DEGB probability *for a single weld* subjected to SCC took up to three CPU *hours*. The increased time demand can be traced to two sources:

(1) the SCC model itself. In our earlier evaluations we were considering only one pre-existing ("fabrication") flaw per replication of the Monte Carlo simulation, and *not* allowing for crack initiation after plant operation had begun. Although PRAISE-B did include a rudimentary SCC model, only crack propagation was considered. The advanced SCC model developed for the BWR evaluation considers both propagation *and* initiation; consequently, up to 30 cracks may be present during a single replication of the simulation, increasing computational effort as well as code overhead.

(2) the inability to use stratified sampling. Because in our PWR evaluations we were considering only single flaws, we were able to use "stratified sampling" to eliminate very small flaws (which were pre-determined not to cause leak or break) and very large flaws (for which failure would be certain) from the initial crack size distributions sampled by PRAISE. Consequently, we were able to reliably predict extremely low failure probabilities with relatively small sample sizes. However, if SCC cracks are allowed to initiate after plant operation has begun, even very small flaws must be carried through the analysis because we must consider the possibility of small cracks linking together to create a single large crack and therefore cannot presently use any kind of stratification techniques. Note that this not only increases the number of replications necessary to predict DEGB probabilities, but also increases the total number of cracks considered. A typical BWR weld calculation using 30,000 replications, for example, may end up tracking the growth of as many as one million cracks to obtain the one or two breaks necessary to estimate DEGB probability. Note also that the low number of failures relative to the total number of cracks considered also increases the sampling error in the estimated DEGB probabilities. Fortunately, this has not been

the case for *leak* probabilities, as many more through-wall cracks not causing break are encountered.

As part of the our NUREG-0313 study, we are converting PRAISE to the CRAY computer. This activity, motivated by the LLNL decision to retire its CDC 7600 computers, is currently limited to converting PRAISE to scalar CRAY operation only. The resulting improvements in code efficiency -- CDC 7600 codes typically run some two to five times faster on the CRAY -- are not, however, significant compared to the increased CPU requirements of the SCC model.

In order to substantially improve code efficiency, we recommend that the Monte Carlo simulation algorithm in PRAISE be "vectorized" to make optimum use of the capabilities of the CRAY computer. Vectorization of appropriate variables in the algorithm would allow certain calculations to be made in parallel ("pipelined") rather than sequentially, significantly reducing execution time. Potential vectorization schemes for PRAISE include, but are not necessarily limited to, one or more of the following:

- vectorize on welds within a given piping system. Weld pipelining would also require that the systems analysis, currently done external to PRAISE, be integrated into PRAISE directly; this would not, however, in itself entail major effort.

- vectorize on replications of the Monte Carlo simulation for a given weld. In principle, this should be fairly straightforward when only pre-existing flaws are considered, somewhat more complex if multiple cracks (i.e. a pre-existing flaw plus "initiated" cracks) are considered per replication.

- vectorize on multiple cracks within a given replication. This would apply only to the SCC model when considering cracks that initiate after plant operation has begun; PRAISE as currently written allows only one pre-existing flaw per Monte Carlo replication.

It is reasonable to expect that vectorization would result in "order of magnitude" -- one, perhaps two -- reductions in CPU time requirements, which would make more practical reliability evaluations involving stress corrosion cracking.

A more fundamental way to improve code accuracy and efficiency would be to directly estimate crack growth and failure probabilities through a Markov model. The present version of PRAISE uses Monte Carlo simulation to sample from the distributions that describe various parameters used in the analysis. Although we have found this approach adequate for our past purposes, we also recognize that simulation methods have two basic limitations, namely

(1) to estimate very low failure probabilities, a large number of replications is required. Although this problem can be reduced through stratified or importance sampling, much more substantial gains in computational efficiency could be achieved through the use of purely analytical solution techniques. This would be particularly important in cases where the simultaneous growth of multiple cracks is considered, or where several types of transient events had to be included (e.g., thermal transients other than heatup/cooldown).

(2) sampling error introduces uncertainty into the estimated failure probability. The degree of uncertainty depends largely on the number of simulated failures compared to the total number of samples or simulations. The Markov process, being a direct analytical solution technique, has no sampling error.

Crack growth according to the Paris law implemented in PRAISE has already been shown to be a Markov process, indicating that this approach is feasible. It is proposed that PRAISE be modified to compute the probability *distribution* of crack size over the operating life of the system. The distribution of crack size will result in an estimate of failure probability. Because the Markov solution is an analytical (rather than a simulation) process, sampling error will be eliminated, improving the accuracy of the estimated failure probabilities. Computer time requirements would be reduced substantially, thereby increasing the extent to which sensitivity and uncertainty analyses could be performed. The Markov process could also be used to *directly* estimate how sensitive computed failure probabilities are to specified variations in input parameters.

Implementation of Markovian solution techniques, coupled with the advances in small computers over the past few years, could also make a PC version of PRAISE a practical reality.

6.4.2 Technology Transfer

Originally developed in 1979, PRAISE has been continually upgraded to meet the changing demands of the various assessments which we have performed for the NRC. The version of the code used in our evaluation of Westinghouse reactor coolant loop piping ("PRAISE-B"), together with an extensive documentation package, was made publicly available in 1983 through release to the National Energy Software Center (NESC) at the Argonne National Laboratory [17]. Since then, we have modified the code extensively; significant modifications include the tearing instability failure criterion for carbon steels (which supplements the net section stress criterion used for austenitic materials) and the advanced probabilistic model of stress corrosion cracking in BWR stainless steels.

In order to make this improved capability available to the public, the NRC in late 1987 requested that we release an updated version of the code ("PRAISE-C") to the NESC. We submitted the code to the NESC in January 1988 [18]; at this time the NRC staff decided that existing documentation was sufficient for release purposes and therefore did not instruct us to prepare a comprehensive user manual specifically for PRAISE-C. Consequently, "user manual" documentation for PRAISE itself is presently spread over three NUREG reports [19,20,21], while associated theoretical information is similarly dispersed in three volumes [12,20,22]. Yet another volume [23] is dedicated to the methodology used to consolidate the results of individual PRAISE runs into "system" failure probabilities — the core of any piping system assessment.

Although in principle the advanced capability of PRAISE-C is now available to outside users, two factors make this latest release a less than totally satisfactory "technology transfer" exercise:

- code documentation is not conveniently available to the outside user. Just to understand the mechanics of PRAISE itself, the new user must now digest at least five technical reports describing code theory and operation.

- application of PRAISE in actual piping system assessments is not convenient for the new user. The only documentation specifically devoted to this topic is one of a nine-volume report now over eight years old. Furthermore, no software tools for either crack sample stratification or for post-processing PRAISE results in a "system" analysis are available to the outside user as part of the code release package.

We regard it of no small significance that virtually all of the code-related questions that we have answered since PRAISE-B was released have been about application of PRAISE, rather than about the mechanics of the code itself.

Even before release of PRAISE-B, we placed considerable effort into code application and into streamlining the overall piping assessment process. To this end, we developed several pre- and post-processing routines for (1) consolidating PRAISE results into "system analyses", (2) preparing the stratified sampling spaces for stainless steel and carbon steel piping (STRADA, ROLLIN), and (3) generating crack tip stress intensity factors for thermal transients other than heatup and cooldown (TIFFANY). These small codes simplify application of PRAISE, but have never been formally documented or included as part of any past code release package.

Updated documentation, plus preparation and execution of suitable sample problems, is *necessary* before the latest release of PRAISE can truly be regarded as a "technology transfer" achievement. We therefore recommend that the following be prepared and submitted to the NESC as a supplement to the PRAISE-C release package:

- comprehensive documentation of the PRAISE code itself. This will require consolidating and updating existing documentation, as well as generating new material as appropriate. We recommend that this information be presented in two separate volumes: a theoretical volume describing the PRAISE assessment methodology, and a "user's manual" describing code operation.

- formal documentation of the pre- and post-processing routines for PRAISE. We recommend that this information -- both theoretical background and user instructions -- be placed in a single volume to supplement the above two.

- suitable sample problems for inclusion in the appropriate volumes above.

Preparation of this material would entail significant effort but would also make it more convenient for new users to apply the code in actual piping system assessments. This material would not only benefit public (e.g., industry) users of the code, but would also aid the NRC staff in reviewing licensee submittals based on PRAISE calculations.

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