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Large LOCA-Earthquake Combination Probability Assessment— Load Combination Program Project 1 Summary Report

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ABSTRACT

This report summarizes work performed for the U.S. Nuclear Regulatory Commssion (NRC) by the Load Combination Program at the Lawrence Livermore National Laboratory to establish a technical basis for the NRC to use in reassessing its requirement that earthquake and large loss-of-coolant accident (LOCA) loads be combined in the design of nuclear power plants. A systematic probabilistic approach is used to treat the random nature of earthquake and transient loading to estimate the probability of large LOCAs that are directly and indirectly induced by earthquakes. A large LOCA is defined in this report as a double-ended guillotine break of the primary reactor coolant loop piping (the hot leg, cold leg, and crossover) of a pressurized water reactor (PWR). Unit 1 of the Zion Nuclear Power Plant, a four-loop PWR-1, is used for this study.

To estimate the probability of a large LOCA directly induced by earthquakes, only fatigue crack growth resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads is considered. Fatigue crack growth is simulated with a deterministic fracture mechanics model that incorporates stochastic inputs of initial crack size distribution, material properties, stress histories, and leak detection probability. Results of the simulation indicate that the probability of a double-ended guillotine break, either with or without an earthquake, is very small (on the order of 10^{-12}). The probability of a leak was found to be several orders of magnitude greater than that of a complete pipe rupture. A limited investigation involving engineering judgment of a double-ended guillotine break indirectly induced by an earthquake is also reported.

FOREWORD

This report describes work performed for the U.S. Nuclear Regulatory Commssion (NRC) by the Load Combination Program at the Lawrence Livermore National Laboratory since March 1979. It represents a milestone effort by the NRC to develop a technical basis for reassessing the requirement that loads resulting from an earthquake and a large LOCA be combined in the design of nuclear power plants. The report contains a complete description of the technical issues, the approach taken to clarify these issues, and major results, conclusions, and recommendations. Details about the models and computational procedures, which are not presented in this summary report, will be published in a subsequent final report on Project I. The work was performed for the Mechanical Engineering Research Branch and the Metallurgy and Materials Research Branch of the Office of Nuclear Regulatory Research. The NRC technical monitor is J. O'Brien.

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INTRODUCTION

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] Historically, the U.S. Nuclear Regulatory Commission (NRC)--through Regulatory Guides, regulations, branch technical positions, and the Standard Review Plan--has required that the responses to various accident loads and loads caused by natural phenomena be combined in the analysis of safety-related structures, systems, and components.

Designing safety-related structures, systems, and components to withstand the combined effects of an earthquake and a large loss-of-coolant accident (LOCA) is one such load combination requirement that has been implemented by the nuclear industry for more than ten years in the design of commercial nuclear power plants. The combination of the most severe LOCA load with safe shutdown earthquake (SSE) loads was not controversial until about five years ago when the postulated LOCA and SSE loads were both increased by a factor of two or more to account for such phenomena as asymmetric blowdown in PWRs and techniques to better define the loading were developed.

As a result of this change, the combination requirement became more difficult to implement, particularly in the design of reactor pressure vessel internal and support systems. For future plants, the change brought with it the prospect of increased construction costs. Additionally, the load combination requirement raises the issue of whether design for extreme loads will result in reduced reliability during normal plant operation. For example, present seismic design methods tend to result in stiff systems and more supports when additional strength is provided for the earthquake loading. Because a stiff system is subjected to greater cyclic thermal stress than a flexible one under normal thermal operating loads, reliability is reduced under normal conditions.

Faced with these design, cost, and safety issues, the nuclear industry petitioned the NRC to reconsider its design requirement. From a safety viewpoint, costs should not be a factor in changing design requirements. The costs of meeting design requirements are industry's responsibility. However, for existing plants to meet the revised loading definition and also satisfy the combination requirement, modification is almost unavoidable. Certain plants can be feasibly modified, but other plants are not feasible to modify, and they present a difficult problem to the NRC. The solution can be either to allow continued operation without modifications, challenge the safety of continued operation without modifications, or solicit a technical basis for reassessing the design requirement.

Industry's approach to the problem has been to assure safety by justifying that the combination of events is unlikely. For the NRC to accept industry's justification and change the design requirement, independent confirmatory research is necessary. One such study has already been made for the NRC by Battelle Columbus Laboratories.[2] The Battelle researchers used a deterministic approach to assess the likelihood of a break occurring in a cold leg pipe of a PWR plant. The work reported herein assesses the hot leg, cold leg, and crossover of a PWR plant. The approach used is probabilistic. Both efforts will be considered by the NRC in assessing the nuclear power industry's submittal and request.

OBJECTIVE

The objective of this study is to estimate the probability that a large LOCA and an earthquake occur simultaneously. This information will be used by the NRC to reassess the requirement that earthquake and LOCA loads be combined. If a LOCA and an earthquake are independent events, the probability of simultaneous occurrence is expected to be very low. However, if an earthquake can cause a LOCA, the probability of simultaneous occurrence may be significant. Thus, this assessment considers only LOCAs that are directly and indirectly induced by earthquakes in addition to normal and abnormal plant operating loads.

SCOPE

In Phase I, we limit our investigation to determining the probability of a large LOCA induced by earthquakes for a pressurized water reactor (PWR) plant. A large LOCA is defined as a double-ended guillotine break of the primary reactor coolant loop piping--the hot leg, cold leg, and crossover. Such pipes typically have outside diameters of 30 inches or more, and have walls that are approximately 2.5 inches thick. This evaluation is limited to the rupture of such large pipes because they will generate the most severe LOCA loads, which, when combined with SSE loads, present the design and retrofit problems discussed above.

We recognize that the break of a smaller pipe may be more probable, and that such a small LOCA may pose larger risks to the plant. However, for Phase I we limit our scope to the large LOCA defined above in order to address the immediate NRC need for confirmatory research. We believe that the models and computational procedures developed for the large LOCA can be extended to the assessment of smaller LOCAs during subsequent phases of our study.

Only fatigue crack growth resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads was considered as the mechanism leading to complete pipe rupture as a direct consequence of earthquakes. The water hammer mechanism was not considered because it has never been observed in PWR primary systems. Likewise, stress corrosion is another plausible mechanism, but it was excluded from consideration because the coolant water chemistry in PWRs is such that stress corrosion problems have not been observed.

A limited investigation involving engineering judgment of a double-ended guillotine break indirectly induced by an earthquake is also reported. Earthquake-induced indirect causes such as falling cranes, mechanical, electrical and structural failure, as well as fire, explosion, and missiles are considered. The emphasis of this work was to identify sources and establish the ground work for a more thorough evaluation in a subsequent phase. Therefore, the preliminary results presented in this report are limited.

Unit 1 of the Zion Nuclear Power Plant (Zion Unit 1) was selected as the demonstration plant for this study. The results and conclusion are applicable only to Zion Unit 1 at this time; that is, no attempt has been made to extend our findings to other plants during Phase I. However, the methodology developed for the evaluation is an advanced computational tool. It can be applied in future evaluations to the break of reactor coolant pipes, large or small, to other PWR and boiling water reactor (BWR) plants, or to general piping reliability assessments. Such studies are clearly beyond the scope of the work reported here, but they may be part of future phases of the program.

APPROACH

The current practice of considering these dynamic events acting concurrently has generally been based on conservative engineering judgment that has not addressed the fact that the postulated LOCA and earthquake loads are random events. Amplitude, duration, frequency content, time of occurence, and time-phase relationship are random and stochastic in nature. Thus, a systematic probabilistic assessment is necessary before a technical basis for an appropriate combination requirement can be developed.

A multiphase, systematic probabilistic approach was used to treat the random nature of earthquakes and LOCAs. In reaching our Phase I objective, we took the following steps:

- Considered many mechanisms that can lead to a pipe failure as a direct result of an earthquake, but concluded that fatigue crack growth resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads has the highest potential to lead to complete pipe rupture. In particular, the water hammer effect was not considered because it has never been observed in PWR primary systems.

- Modeled fatigue crack growth with a deterministic fracture mechanics model that incorporated stochastic inputs of initial crack size distribution, material properties, stress histories, and leak detection probability.

- Developed structural models to calculate seismic stresses and nonseismic stresses induced in the piping by dead weight, pressure, thermal expansion, and transients. Steady-state vibrational stresses and residual stresses were also considered but found to be insignificant for growing fatigue cracks.

- Calculated the probabilities of pipe leaks and breaks during the plant's life by inputing to the fracture mechanics model the results of the stress analysis and estimates of the crack size distribution, material properties, and crack and leak detection probabilities.

- Estimated the probability of a directly induced large LOCA based on the site-specific seismic hazard.

- Performed limited sensitivity studies on the inputs to the stress analyses, fracture mechanics model, and estimation procedure.

- Used engineering judgment to estimate the probability of a large LOCA induced indirectly by earthquakes. Structural, mechanical, and electrical failures, as well as explosions, fires, and missile incidents caused by earthquakes were considered. The emphasis of this work was to identify sources and establish the ground work for a more thorough evaluation in a subsequent phase. Therefore, the preliminary results presented in this report are limited.

MAJOR ASSUMPTIONS

Assumptions are necessary to simplify the complex assessment. These assumptions and simplifications reflect engineering judgment that is based on information available from the literature and from preliminary analyses. The assumptions for the directly induced LOCA problem are listed below, with the most basic assumption listed first.

- Failure results from fatigue crack growth of crack-like defects that are confined to the girth-welded butt joints. Note that these cracks are modeled in the circumferential orientation because of its reference to the postulated double-ended pipe break. Longitudinal cracks, though important to the leak assessment, will not result in such a guillotine break and are, therefore, not considered.

- Crack size distribution can be adequately characterized, despite the lack of data on large cracks. The tail of the crack size distribution (i.e., the large cracks that are present from the onset of crack growth) plays a major role in determining the probability of a LOCA. Unfortunately, there are no data on large cracks; thus, crack distribution data generated from smaller crack sizes must be extrapolated into this regime. Several distributions were considered,[3,4,5] and the most conservative distribution (that of Marshall, Ref. 3) was used.

- Based on observed pipe cracks, the crack shape is semielliptical; the shape is maintained during crack growth; and the length-to-depth aspect ratio is two or more. Only surface cracks are evaluated. Based on equivalent crack surface area and crack length, a surface semielliptical flaw has a stress intensity 1.4 to 2.5 times that of the embedded flaw. When this is cast in terms of fatigue behavior, we find that the surface flaw will grow at a rate of 4 to 50 times that of the subsurface flaw.

- No distinction is made between the shop or field welds for either material properties or crack size distribution. Data on fatigue and tensile properties of welded and unwelded 316 stainless steel show little variation in mechanical behavior.[6,7,8,9]

- The initial crack size distribution is independent of location within a joint and from joint to joint.

- At most, one crack is modeled in each joint, and crack interaction is not considered. Multiple cracks are ignored because the probability of having exactly one crack in a joint is approximately 0.09, whereas the probability of having two or more cracks in the same joint is less than 5×10^{-3}. These approximations are based on the probability of having a crack in a unit volume equal to $10^{-4}/\text{in.}^3$ (an adaptation from Refs. 10 and 11) and a weld volume of approximately 10^3 in.^3 per weld joint.

- Subcritical crack growth results from fatigue. Stress corrosion cracking is not considered because it has not been observed in PWR primary coolant piping.

- A Paris model describes crack growth.[6,12,13,14] Crack retardation stemming from pulses of high-stress cycles is not considered, nor is the enhanced fatigue behavior resulting from large values of stress intensity. The effects of material variation, weld properties, and environment are accounted for in the distribution of the coefficient to the Paris equation. When an inside surface crack grows through the pipe wall, its length on the outside surface is conservatively set equal to the inside length. Therefore, the minimum through-wall crack is at least twice the pipe thickness.

- Only cracks in the circumferential orientation are considered. Probable crack locations that are not considered in this study include longitudinal welds in elbows and pipe sections, nozzle corners, bimetallic transition joints (i.e., the reactor pressure vessel to safe-end welds), and base metal. Except for the bimetallic transition joint, these crack locations will not result in the double-ended pipe break. Data for fatigue of the bimetallic joint are unavailable. However, if the 316 stainless steel fatigue relation is a reasonable approximation of the bimetallic property, then the system failure probabilities should not change significantly.

- Stresses used in this evaluation include pressure, thermal expansion, dead weight, operating and abnormal transients, and seismic events. Conservative values of design temperature and pressure changes and frequency are used in calculating stresses. Vibration stress magnitudes are very low, and for crack sizes of interest result in a stress intensity below the threshold value. Residual stresses are found to be compressive on or very near the inside surface, and the stress intensities evaluated from them are negative. Residual stresses are not included because (1) the negative stress intensity would retard crack growth, and (2) there is a certain amount of uncertainty associated with residual stresses and how they redistribute as a function of crack growth.

- Design, construction, and assembly errors are not estimated. The primary piping is subjected to high quality assurance testing and if gross design or construction errors exist, they would be found during system checkout and hydrotesting.

- Support stiffnesses and structural damping values are estimated based on the best available data. All component supports and snubbers are assumed to be in good working order. The soil model and seismic hazard curve are both based on site-specific estimates.[15,16]

- Transients occur as a stationary Poisson process reflecting the Zion Unit 1 operating history for its seven years of operation. Earthquakes characterized by a peak horizontal ground acceleration less than 0.07 g are assumed to affect crack growth negligibly. The low stresses calculated bear this out. The maximum free-field peak acceleration that can be generated at the Zion site from the available data base is 0.85 g (5 x SSE).

- Both leak and large LOCA failures are considered. Cracks which grow through the piping wall lead to leaks of greater than 2 gpm based on the minimum crack length of twice the wall thickness. These leaks are assumed to be detected and repaired. The crack size distribution is set equal to the initial size distribution after weld repair.

- A large LOCA can occur if the load-controlled stresses are high enough to sever the pipe. It is assumed that dead weight, pressure, and seismic stresses are load controlled. Leaks that occur and lead to a LOCA during a seismic event are not evaluated--only the final result (leak or LOCA) after the earthquake is considered.

- Pipe severance results from net section plastic instability, which occurs when the net section stress exceeds the average flow stress; $\sigma_{flow} = (\sigma_y + \sigma_u)/2$. The flow stress is assumed to be normally distributed. Approximation of the J-integral and the applied tearing yield considerably larger crack sizes.

- The simulation is ended when the first operating basis or larger earthquake occurs.

SUMMARY OF RESULTS

Results of the simulation indicate that the probability of a large LOCA, either with or without an earthquake, is very small. The probability of a large LOCA induced by plant transients given the condition that no earthquake occurs during the 40-year plant life is estimated to be 1.6×10^{-12}. Seismic stresses tend to increase this probability; that is, the probability of a large LOCA induced by plant transients and earthquakes is estimated to be 1.8×10^{-12}, assuming that the plant is shut down after an operating basis or larger earthquake. The probability of an earthquake and an earthquake-induced large LOCA occurring simultaneously during the plant's life is about 4×10^{-13}. These findings are consistent with other studies of this nature.

These results are best estimates based on the approach taken and the assumptions made. A limited sensitivity study on input variables was performed. Input variables were assigned incredible values to determine how the results vary for the following cases:

- No preservice inspection
- No preservice hydrostatic proof test
- No leak detection
- Modified Marshall initial crack distribution
- Variation of aspect ratio of the length and the depth of the initial crack
- Increase the 5 x SSE seismic stress by a factor of 3.

The limited sensitivity study indicates that leak detection capability, preservice inspection, and proof testing tend to reduce the probability of a LOCA by less than an order of magnitude. Increasing the percentage of cracks with large aspect ratio increases the large LOCA probability significantly; however, it has little influence on the leak probability.

Results of the simulation indicate that an earthquake has almost no effect on the leak probability, which is estimated to be 8×10^{-7} during the plant's life.

In view of the extremely low probabilities reported above, it can be concluded that a reactor coolant loop pipe break as a result of fatigue crack growth is highly unlikely. The probability that an earthquake and a LOCA induced by an earthquake occur simultaneously is less. These conclusions imply only that a reactor coolant loop pipe is unlikely to break as a direct result of the combined loads from normal plant operation and an earthquake. Other external earthquake-induced sources--such as fires, explosions, missiles, and structural, mechanical, or electrical failures--can potentially cause a pipe break as an indirect result of an earthquake. A limited investigation of such LOCAs indirectly induced by an earthquake revealed that most sources do not result in a LOCA which falls within our definition of a large LOCA. Several scenarios of possible sources were identified, and engineering judgment was used to estimate the probabilities of occurrence for the scenarios. A more refined assessment is needed if the indirect scenarios are to become part of the technical basis for decision-making.

PROBABILITY OF A LARGE LOCA DIRECTLY INDUCED BY AN EARTHQUAKE

PLANT DESCRIPTION

The Unit 1 pressurized water reactor (PWR) is one of two PWR units operated by the Commonwealth Edison Company at its Zion Nuclear Power Plant near Zion, Illinois. Designed and constructed by the Westinghouse Electric Corporation, Sargent & Lundy Engineers, and the Commonwealth Edison Company, it began operation in 1971.

The primary components of the four-loop PWR nuclear steam supply system (NSSS) supplied by Westinghouse are the reactor vessel, the four steam generators, and the four reactor coolant pumps (Fig. 1). The primary loop piping, which includes the hot leg (reactor pressure vessel to steam generator), the crossover pipe (steam generator to coolant pump) and the cold leg (reactor coolant pump to the reactor pressure vessel) of each of the four loops provides the basis for the current study. Table 1 gives the nominal piping dimensions and operating parameters for each leg. The 14 girth-welded butt joints common to each loop (shown in Loop 1, Fig. 1) were evaluated in the fracture mechanics model.

The primary piping, nozzles, and fittings are made of various grades of cast and wrought 316 stainless steel.[17] In this study, no attempt has been made to differentiate the mechanical properties in these different components. The ASME code requirements for the minimum specified room temperature yield and ultimate strengths are 30 ksi (207 MPa) and 70 ksi (483 MPa), respectively. The code allowable stress at the operating temperature

TABLE 1. Primary piping dimensions and operating parameters, from Ref. 17.

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

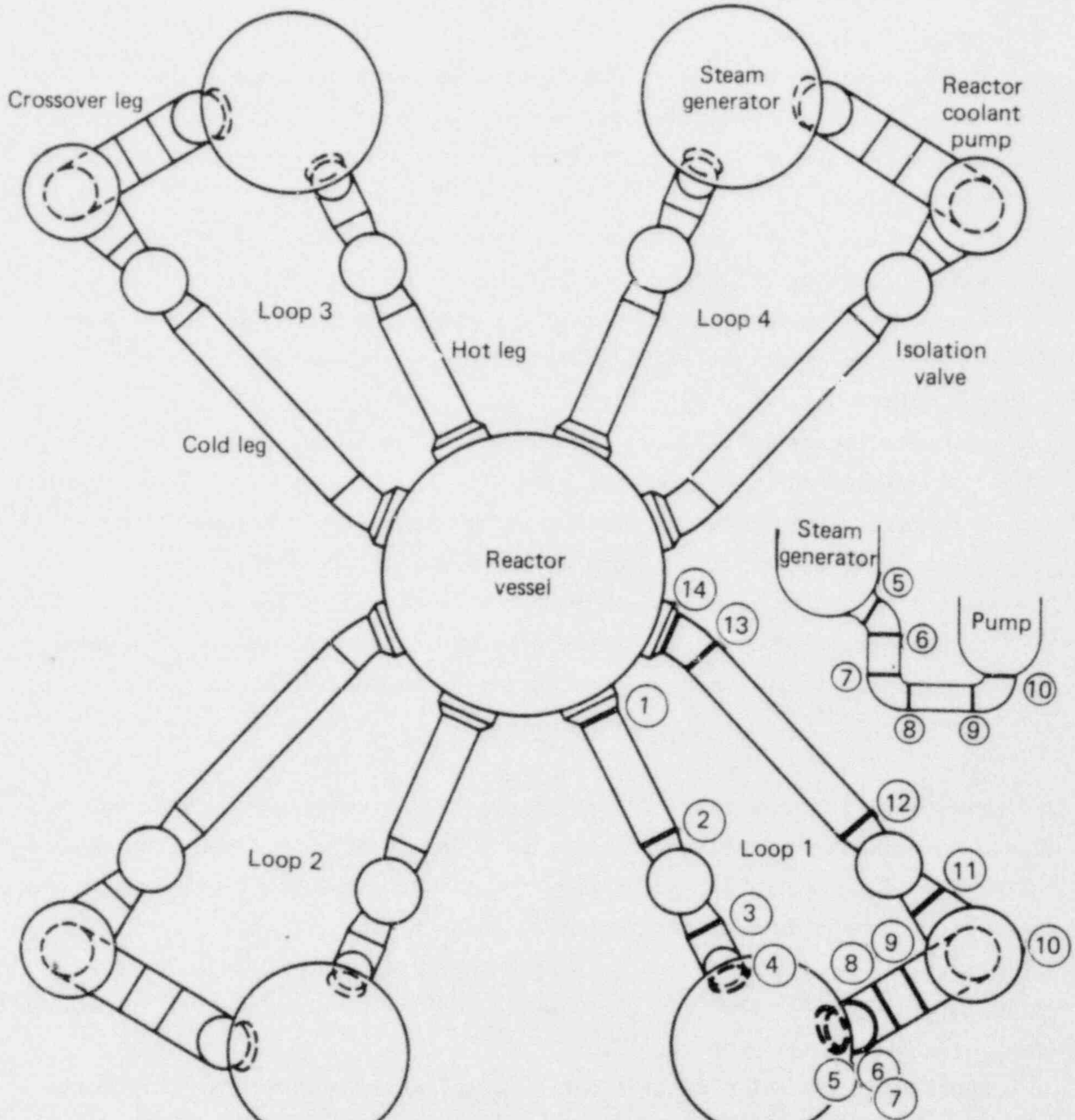

FIG. 1. Schematic of the four-loop PWR nuclear steam supply system at Zion Unit 1. Shown in Loop 1 are the 14 circumferential girth-welded butt joints common to each loop that were analyzed in the fracture mechanics model.

ranges from 11.8 ksi (81 MPa) to 16.8 ksi (116 MPa), depending on whether piping fittings or nozzle material are considered.[18]

LOAD AND STRESS ANALYSES

The approach selected to estimate fatigue crack growth and pipe fracture requires estimates of the pipe stresses arising from all relevant loads during the plant's lifetime. We calculated loads and stresses, given the following assumptions:

- All responses are linear.
- Stresses arising from each load are scaled according to the random variation of that load and added to other stresses, which act simultaneously.
- Seismic loads occur during normal operation at full power, and no other transients occur at the same time.
- Fatigue crack growth is controlled by the cyclic stresses resulting from both normal operating transients and seismic vibrations.
- Failure occurs at the girth-welded butt joints in the hot leg, crossover, and cold leg of the primary piping. As such, the nodal points of the structural model are made to correspond to the weld joints.

Figure 2 is a flow diagram showing how the stresses are calculated. Since the objective of this work is to perform a realistic simulation, best estimate loads and material properties are used for the calculations whenever possible. If such data are unavailable, conservative values for loads and properties are used instead. Many parameters, such as damping, support stiffness, and soil properties, are varied to estimate the sensitivity of the results to those parameters.

Based upon nominal pipe sections, the operating pressure of 2250 psi produces a maximum axial stress in the pipe of 7120 psi--considerably less than the value allowed by the ASME code at the operating temperature. The combination of dead weight and restraint of thermal expansion produces an additional stress of about 8500 psi at the most highly stressed portion of the pipe, the hot leg to pressure vessel joint. Figure 3 shows how the maximum stress in that joint varies with increasing seismic loads.

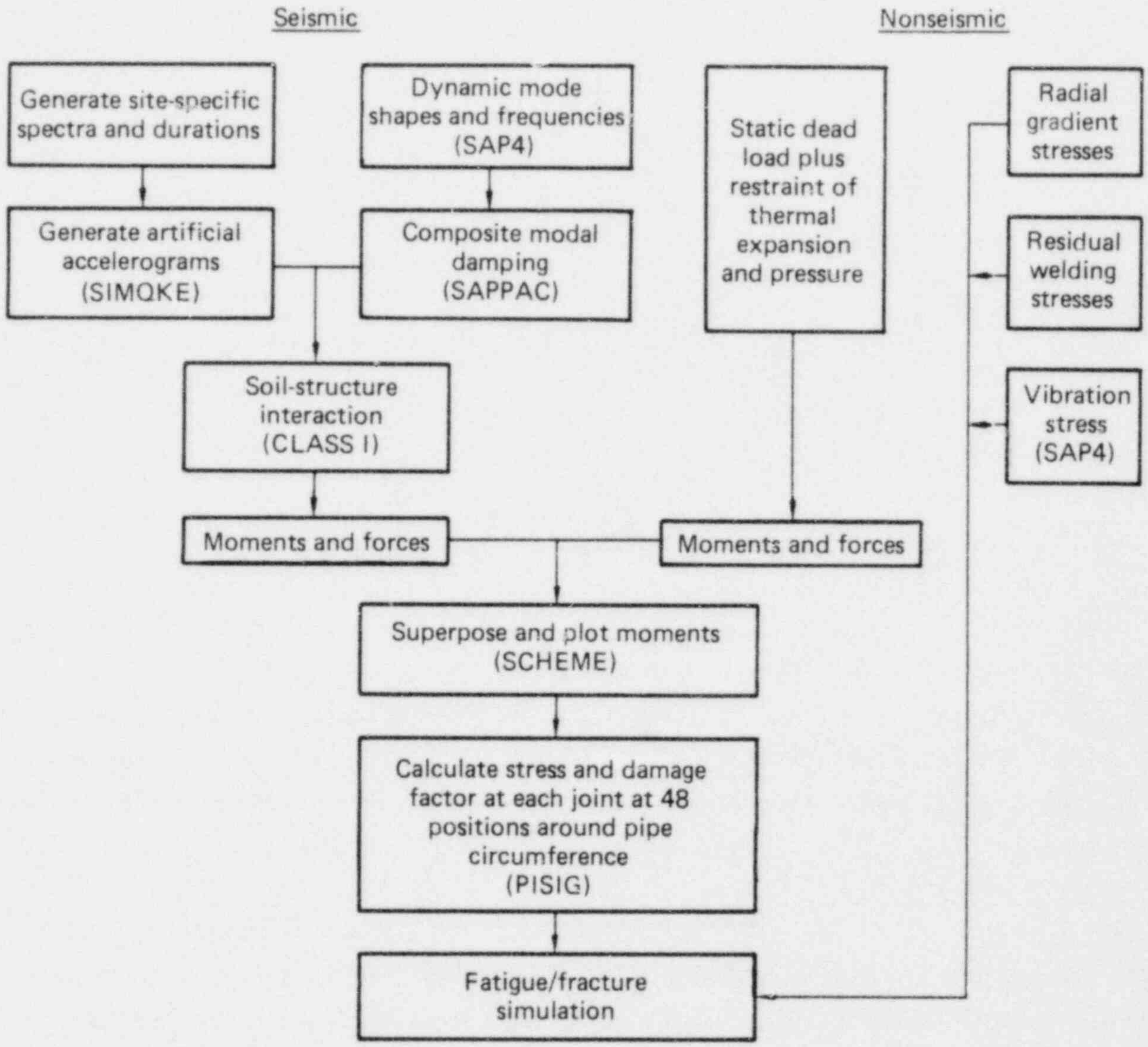

FIG. 2. Flow chart of the stress calculations.

Nonseismic Stresses

For the pipe stress analyses, our intention is to use data that best reflect actual plant operating conditions. Useful data for Zion Unit 1 are available for heat up, cool down, unit loading at 5% per minute, and unit unloading at 5% per minute. The recorded temperatures for the hot-standby and

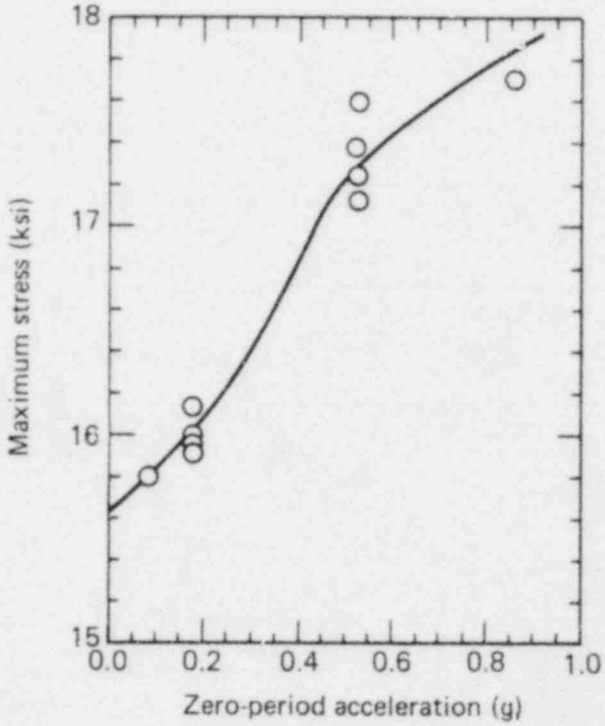

FIG. 3. Variation of maximum stress in the hot leg to pressure vessel joint (joint 1, Fig. 1) with peak horizontal ground acceleration. Note that each plot point represents a different randomly generated artificial accelerogram.

full-power conditions are 548.5°F (both hot and cold legs) and 540°F (cold leg) and 588°F (hot leg), respectively. These temperatures are used for the thermal expansion and thermal radial gradient stress analyses. The design temperature for the hot-standby condition is 547°F; the design temperatures for the full-power condition are 592°F and 530°F for the hot and cold legs, respectively.[17]

The primary stresses in the primary piping during normal operation result from thermal expansion, dead weight, and pressure. Thermal expansion stress is defined as the stress induced by limiting the thermal expansion and contraction of the piping system. Dead weight stress is caused by the piping system's own weight; pressure stress is produced by the internal pressure loading of the piping system.

The thermal expansion and dead weight stresses are computed using the SAP4 computer program.[19] The pipes, reactor pressure vessel, steam generators, reactor coolant pumps, pressurizer, valves and component supports are represented by a combination of pipe, beam and truss elements. All four loops are modeled in the analysis. The input temperatures for the thermal expansion analysis are 588°F for the hot leg, and 540°F for the cold leg and crossover leg. For the surge line, the average temperature between the hot leg and the pressurizer (635°F) is used. Results are calculated in terms of maximum principal stresses for Loop 1.

During the thermal transient condition, the changes in fluid temperature create a nonuniform temperature distribution across the pipe, which induces stress variations in the pipe wall. For clarity, these stresses are referred to as thermal radial gradient stresses in this report. The thermal radial gradient stresses are evaluated for the hot leg, cold leg, and surge line. Because of the similarity in temperature, the resulting thermal gradient stresses for the cold leg are also used to evaluate the crossover leg.

Because poor time resolution in the operational data made it difficult to determine what the temperatures were during a transient, the temperatures for the cold leg and hot leg at full power are used along with the Westinghouse design transients to calculate stresses for the thermal radial stress analyses. The stress analysis of some of the thirteen thermal transients evaluated in this study indicate that minimum and maximum stresses at the pipe surface tended to occur close to or at the time limit shown by the Westinghouse curves.[20] Since the transient had not actually terminated by this time, the curves were extrapolated to a sufficient time to see a 5% change in the inside surface thermal stress. The only exception to this procedure is the steam line break transient, where the eventual temperature is 212°F for both hot and cold legs. The extrapolation for this case is made linearly to 212°F. Temperatures and stresses through the wall were calculated analytically.

Vibration stresses in the piping system are generally caused by vibration of the rotating equipment, by flow-induced vibration (such as vortex shedding, cavitation, or flashing), or by fluid pressure pulses. The importance of vibration stresses depends on the stress magnitude. Because there may be many cycles, even moderate vibrational stresses can be detrimental to the piping system.

To predict vibration stresses, we performed dynamic time history analyses and compared the results to the preoperational test data from the Indian Point Nuclear Plant, which has the same thermal output and system configuration as Zion Unit 1.[21] Flow-induced vibration was not included in this study. A simple harmonic forcing function based on the rotating mass, eccentricity, and rotational speed of the reactor coolant pump was used for the time history analysis. The effect of the upper restraints on the steam generator was also investigated. The time history analysis was performed using the SAP4 code with 3% of critical damping. The bending mode of the steam generator was found to be the first mode. Its modal frequency was 7.2 Hz with the upper restraint included and 2.4 Hz with the upper restraint excluded; both values agreed well with the test results.[21] On the other hand, calculated deflections were much lower than those found during preoperational testing. To account for this difference, the resulting stresses were scaled by the ratio of the measured to the analytical deflections. The resulting stresses were very low (i.e., a few hundred psi) and did not influence the fatigue growth.

Seismic Stresses

Fatigue crack growth and pipe fracture can occur as a consequence of seismic events. Earthquakes and the stresses induced by earthquakes are assumed to occur during normal plant operation at 100% power. No other transients occur at the same time. Note that the best estimate seismic stresses shown in Fig. 3 are a small fraction of the total operating loads.

The analytical model for the seismic stress analysis includes a reactor coolant loop model, reactor building model, and a soil model. The reactor model includes all major components affecting the seismic response of a four-loop PWR-1 Westinghouse nuclear steam supply system (NSSS)--the components, supports, and the interconnecting piping. The major components include one reactor pressure vessel, four steam generators, four reactor coolant pumps, and one pressurizer. Major piping includes the hot leg, crossover, and cold leg. This model is a reduced version of the NSSS model used for the nonseismic stress analysis. A detailed description of this model is given in Ref. 22. The two components of the reactor building are the prestressed concrete containment shell and the reactor building internal structures. These two structures share a common foundation; otherwise, they are structurally separated. The containment shell is represented by a 3-D, 12-mass stick model. Beam elements connect the lumped masses. The internal concrete structure is represented by a 3-D finite element model composed mainly of plate elements to simulate concrete walls and slabs. The NSSS model is connected to the internal concrete structure mainly at the operating floor and at the foundation level.

The dynamic soil stiffness parameters (shear modulus and material damping) required for the soil-structure interaction analysis were estimated using a site-specific soil model as input to the SHAKE computer program.[23] The soil was idealized into layers of constant soil properties which include unit weight, layer thickness and description, low-strain shear modulus, and shear wave velocity values. The basis for these data was reported in Ref. 15.

Seismic inputs were generated for a variety of peak accelerations from one times OBE to five times SSE. For each given peak acceleration, an earthquake with a specific spectrum and duration was randomly selected from a base of 5000 potential earthquakes at the Zion site. Artifical accelerograms were generated using the computer program SIMQKE[24]. Three statistically independent artifical accelerograms were generated from each target spectrum for two horizontal and one vertical direction, respectively. The ratio of peak vertical acceleration to peak horizontal acceleration was established based upon an analysis of earthquake-motion data.

The soil-structure interaction analysis was carried out using the CLASSI code.[25] The formulation of CLASSI is generally categorized as the substructure technique (or 3-step solution technique). The soil-impedance function matrices and structural dynamic (modal) properties were separately determined. The structural modal properties, in this case, were determined by SAP4. Steam generator sway was found to dominate the first few modes. Reactor coolant pump motion became important in the higher modes. The dynamic soil stiffness and damping input to CLASSI for establishing the impedance function were taken from the SHAKE analysis. The output from CLASSI is the three-dimensional moment time history at the selected primary piping joints.

Dead weight and thermal expansion moments were combined with the seismic response to evaluate the fatigue damage and maximum stress. The fatigue damage calculation is based upon a Paris-type fatigue crack growth relation as discussed in the modeling section. The fatigue damage factor, S', and maximum stress, σ_{max}, are computed at 48 locations around the circumference of the joints under investigation, and the maximum value is used in the crack growth analysis.

Best-estimate, state-of-the-art techniques have been used to estimate the seismic stresses at the selected joints. The results indicate that at the SSE level, the seismically induced stresses are small compared to normal operating stresses, and their effects have only a small contribution to crack growth. At this level, only the joint connecting the hot leg to the pressure vessel (joint 1, Fig. 1) is stressed beyond one-half yield. A limited sensitivity study was performed to evaluate the variability of these results with changes in damping, seismic input, soil-structure interaction, and dynamic analysis method. Compared to the results using the best-estimate input parameters, we found that a reduction in damping, use of the Regulatory Guide 1.60 (Ref. 26) design spectra, and elimination of soil-structure interaction effects (i.e., fixed base) can all significantly increase the joint stresses.

Use of a two-step response spectrum analysis technique, as compared to the simple one-step analysis employed in the best-estimate study, increased the stresses by over a factor of two. The two-step method is inherently conservative since it uses the enveloped floor response spectrum curve; whereas, the majority of the supports are near the foundation level. Separate seismic inputs occurring at the different supports could provide the most accurate analysis; however, this is beyond the current scope.

The stresses obtained in this analysis were significantly less than those predicted by Westinghouse for the design of the NSSS.[27] By inputing conservatively low values for support stiffnesses and using the R. G. 1.60 response spectrum as used by Westinghouse, we were able to reproduce the original design results. The low value of stresses calculated using the best-estimate input parameters appears to be a more realistic estimate of the actual piping stresses during a seismic event.

FRACTURE MECHANICS MODEL

The probability of a leak or guillotine break in the primary piping as a result of a range of earthquakes is estimated using a fracture mechanics based model. Crack growth is modeled using a Paris-type fatigue formulation. The initial crack size distribution models semielliptical surface cracks oriented in the circumferential direction. Guillotine failure of the pipe is governed by elastic-plastic material behavior. Failure is restricted to the girth-welded butt joints shown in Fig. 1. There are four in each hot and cold leg and six in each crossover pipe for a total of 56 possible pipe fracture locations in the four loops.

The probability of leak and pipe break is estimated as a function time for the 40 years of the plant's life. Crack growth and pipe fracture occur only during plant operation as a result of cyclic stresses (thermal expansion, thermal radial gradient, vibration, and seismic) and static stresses (dead weight, pressure, and residual).

The calculational procedure for the estimation of leak and guillotine pipe break uses a deterministic fatigue crack growth relation combined with random initial crack size distribution, crack detection probability, material properties, stress histories, and leak detection probability. Figure 4 shows a schematic representation of the calculation procedure.

The fracture mechanics model incorporates many of the advances developed in earlier models, such as those summarized in Refs. 10, 11, and 28-32. The major extension is the consideration of elliptical defects (which require two dimensions for their description), rather than crack geometries that can be characterized by a single dimension (such as a complete circumferential crack). Hence, a bivariate, rather than a univariate, crack size distribution is required. This added dimension contributes significantly to the difficulty of this problem. However, the model is more realistic, and pipe leaks can be discerned from complete pipe severances. The other major extensions over previous models are detailed consideration of stress gradients through the pipe wall and consideration of leak detection.

Given that a crack exists, the initial size distribution of thumbnail-shaped cracks is characterized by a bivariate distribution. The crack depth, a, and length, 2b, describe the semielliptical flaw shape. Based on a very limited data base, various distributions have been developed to describe the distribution of flaw depth. The exponential distribution developed by Marshall is assumed to be the best estimate of the crack shape distribution in nuclear piping.[3]

No data were available on the distribution of the crack aspect ratio b/a. We assume, based on observations, that b/a is greater than or equal to one, and use a truncated lognormal distribution to describe its distribution.[33] The mode (most likely value) was chosen to be b/a = 1, and 1% of the cracks were assumed to have an aspect ratio greater than 10.

The crack population distribution after preservice inspection is obtained by multiplying the flaw size distribution by the probability that a flaw exists and that it is not detected. The probability that a flaw exists in a unit volume of weld is taken to be $10^{-4}/\text{in.}^3$ This value is based on values from earlier studies that are modified to account for the fact that only weld material is analyzed.[10,11] The detection probability is a function of crack size.[10,11]

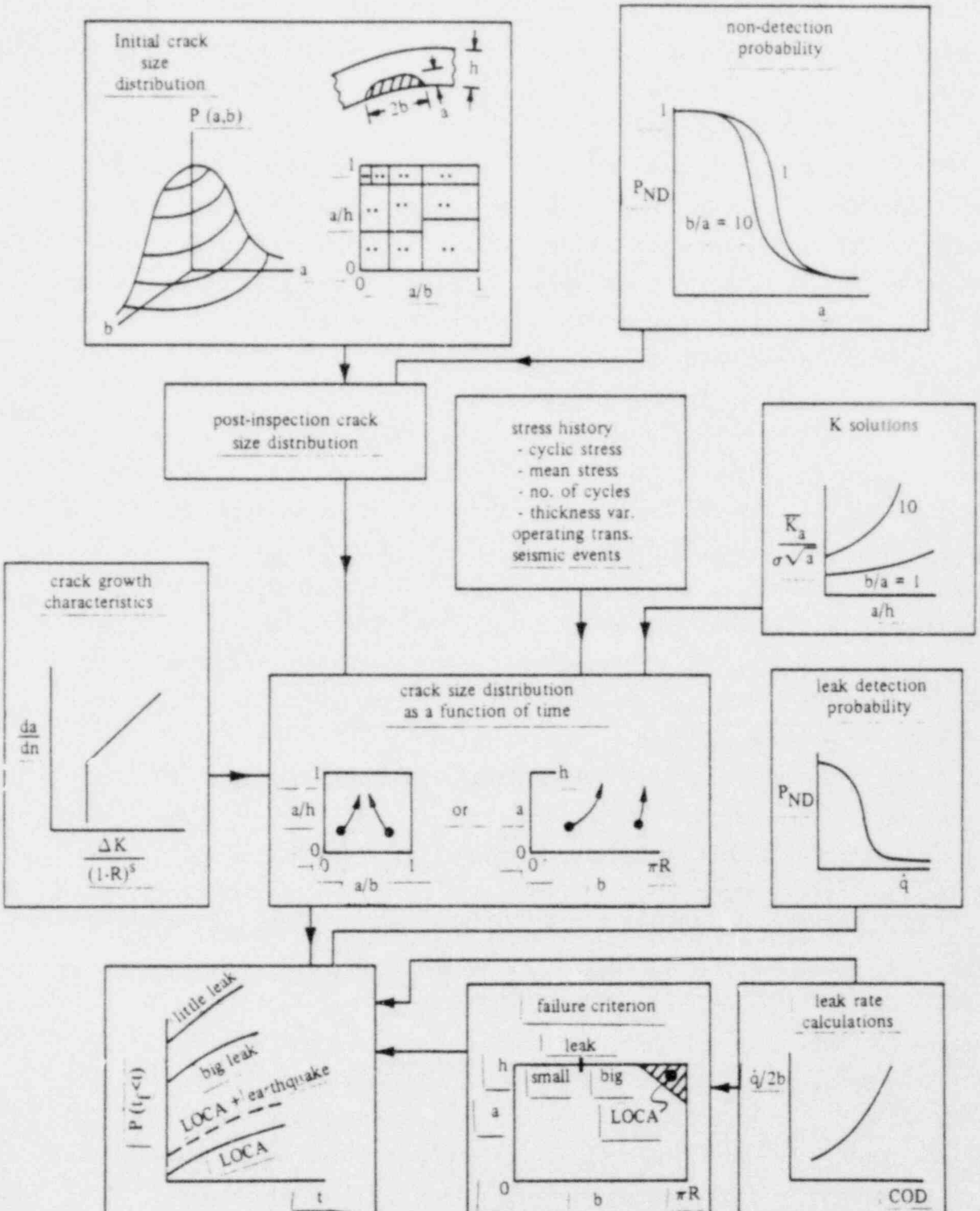

FIG. 4. Schematic representation of the various steps in the analysis shows their interrelationships.

The fatigue model for crack growth employs a Paris-type growth rate equation, i.e., $(da/dn) = C\ (K')^4$, where the exponent of 4 was evaluated from the available data, and the coefficient C is lognormally distributed. The distribution of C is based on data from the literature.[6,7,14,34] Its mean value is 9.14×10^{-12} with a standard deviation of 3.01×10^{-11} for K' in units of ksi-in$^{1/2}$ and da/dn in inch/cycle. The expression for K' accounts for the mean stress level by $K' = \Delta K/(1 - R)^{1/2}$ where R is the ratio of minimum to maximum stress and ΔK is the cyclic stress intensity factor. Below a threshold value of K' fatigue crack growth is not observed.[14,35,36] The fatigue crack growth rate is set to zero when K' drops below the value of 4.6 ksi-in$^{-1/2}$.

The semielliptical shape of the fatigue crack is assumed to be maintained during crack growth. The size and aspect ratios of the cracks are governed by weighted stress intensity solutions K_a and K_b for the depth and length, respectively.[37,38] The stress intensities are calculated using boundary integral equation techniques (Refs. 39 and 40) for each class of transient or seismic event (e.g., thermal radial gradient or uniform stress distribution) as functions of time, and the maximum change in K for each cycle is used in the fatigue growth evaluation. To reduce the computation, the stress intensity solution for each transient is evaluated, made nondimensional with respect to flaw size, aspect ratio and stress level, and stored for subsequent use. When a given transient is to be simulated, the magnitude of its stress (or temperature variation) is multiplied by the corresponding stress intensity for the crack size under investigation. In this respect, a seismic event is simply considered as a special class of transient.

The crack size distribution is updated as a result of fatigue. Crack growth can lead to a leak or LOCA or, as in the case of most cracks, result in neither during the plant's life. The leak rate calculation and the probability of leak detection are employed with the current crack size distribution and pipe fracture criterion to estimate the probability of leak or LOCA as a function of time. The minimum leak size is found to be greater than 2 gpm given that the crack length is taken to be at least two times the flaw depth. Since the required minimum leak detection is 2 gpm, we assume all leaks are found and repaired.[41] Crack shapes which can lead to the double-ended pipe break without leaking are thus limited to very deep, complete, or nearly complete circumferential cracks. Other large cracks lead to a leak and are detected and repaired prior to growing into a LOCA.

The austenitic piping materials employed at Zion Unit 1 are very tough and ductile.[42] The failure of such materials will be governed either by a tearing instability,[43,44] or by stress in excess of critical net stress on the uncracked portion of the pipe cross section. The lengths and diameters of the pipe considered along with the material tearing resistance indicate that a tearing instability will not occur in the pipes considered.[44] However, failure will occur if a critical net section stress is exceeded, because of the inability of the remaining cross-section to transmit the applied loads.[45] Only load-controlled stresses are included.

The flow stress of the material, taken as the average of the yield and ultimate strength σ_{flo} can be considered as a random variable rather than a deterministic value. Information is available in the literature on the statistical distribution of tensile properties of austenitic stainless steel at operating reactor conditions.[8,9,46,47] The mean flow stress at the reactor temperature was calculated to be 44.9 ksi, with a corresponding standard deviation of 1.37 ksi. The flow stress was found to be approximately normally distributed.

SIMULATION OF PIPE FAILURE

A computer code (PRAISE) was developed to simulate the life history of a reactor primary coolant system including all random elements that dictate a possible pipe rupture. These elements are crack distribution, preservice and inservice inspections, loads and stresses, and leak detection. Each weld joint was randomly simulated through all life history replications. The proportion of replications without LOCAs provided estimates for weld joint reliability. Individual weld joint reliabilities were combined to bound system reliability. Stratified sampling obviated the need for many replications to simulate LOCAs, which are rare events. Figure 5 is a flow chart that illustrates the simulation procedure.

The probability of LOCA is very small. If all random processes are simulated according to their actual probabilities, too many replications are needed to get one LOCA, and the statistical confidence interval on the probability estimate is large. Our simulation used a stratified sampling technique that forced rare events to occur more frequently than in real life, and the LOCA probability estimator was adjusted to correct for the stratification.

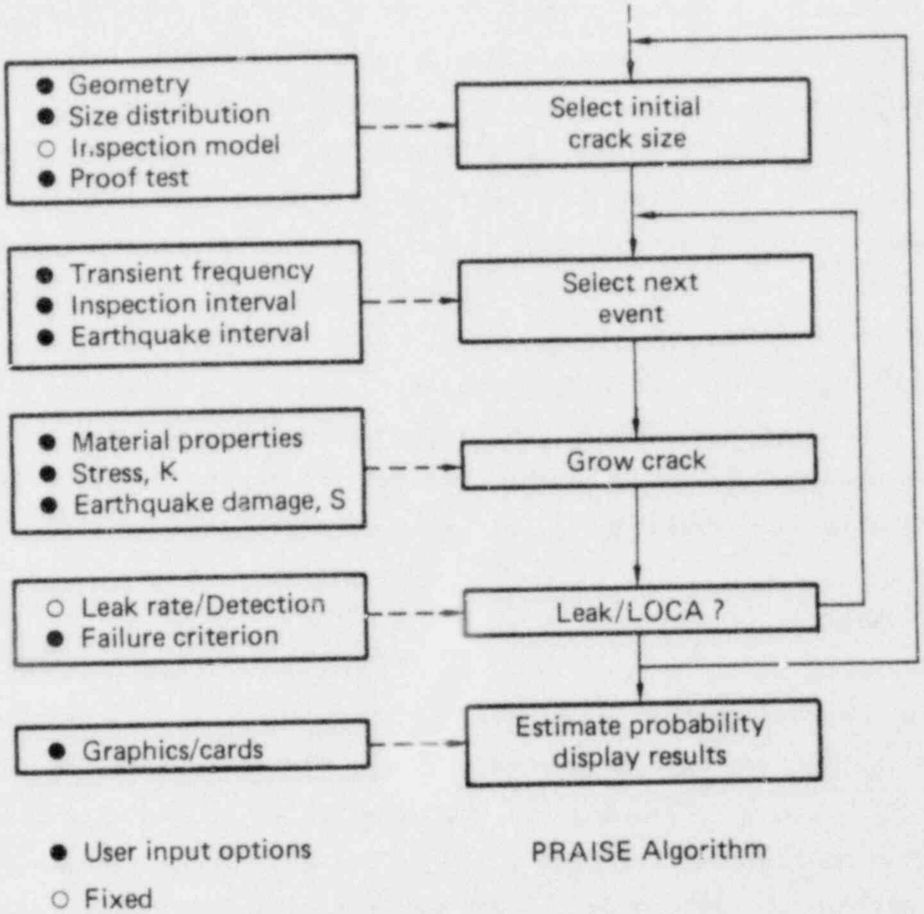

FIG. 5. Flow chart showing the PRAISE algorithm and the inputs required at each step.

Inputs were the crack occurrence rate, weld joint volumes, crack length and depth distribution, crack and leak detection probabilities as functions of crack size, transient occurrence rates, seismic hazard curve, transient and

IMAGE EVALUATION TEST TARGET (MT-3)

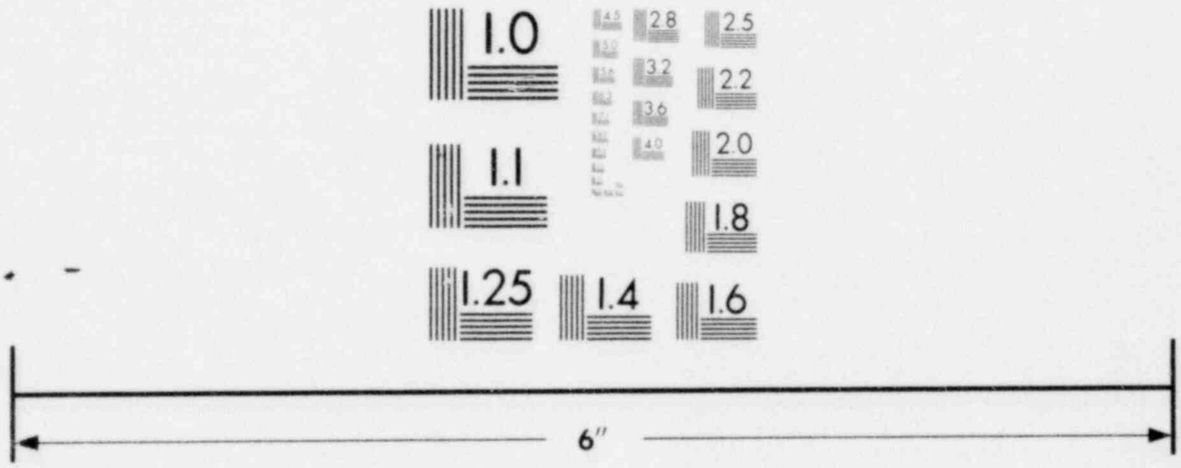

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

earthquake stresses, and crack growth model parameters. Input parameter estimates were obtained from published reports and from operating data on Zion Unit 1.

The computer code simulated cracks in weld joints and preservice inspection. Surviving cracks grew because of simulated piping stresses. The code simulated times to each transient and to the first earthquake. Stress damage resulting from the transient or earthquake made surviving cracks grow. If a leak occurred and was detected, the crack was assumed to be repaired. The replication terminated when a LOCA occurred or when plant lifetime was reached. All weld joints were simulated through the same set of transient and earthquake events.

Simulation outputs were probabilities of LOCAs in all joints as functions of time. The probability of primary-coolant-loop pipe rupture was estimated to equal the sum of the failure probabilities of all joints. This is accurate if all probabilities are small and joint failures are independent events.

DISCUSSION OF RESULTS

Estimates for the probability of a leak and guillotine pipe break for the joint between the hot leg and the pressure vessel (joint 1, Fig. 1) are presented in Figs. 6 and 7. This joint was selected to demonstrate typical results from the PRAISE computer code.

For the primary coolant loop, the probability of a large LOCA, given no earthquake during the life of the plant, was found to be 1.6×10^{-12}. Since the probability of no earthquake occurring during the plant life is 0.76, the probability of a large LOCA and no earthquake is 1.2×10^{-12}.

A seismic event tends to increase the LOCA probability. If we include the occurrence of seismic events, the probability that a large LOCA occurs is 1.8×10^{-12}, assuming that the plant is shut down after an operating basis or larger earthquake. Also, the probability of the simultaneous occurrence of an earthquake and a large LOCA is approximately 4×10^{-13}. The probability of a leak during the plant's life is estimated at 8×10^{-7}.

Of particular interest to this study is the guillotine pipe break of any of the joints located within the reactor cavity. This includes the four hot-leg-to-reactor-pressure-vessel joints (location 1, Fig. 1) and the four cold-leg-to-reactor-pressure-vessel joints (location 14, Fig. 1). These joints are important since their failure will result in the recently

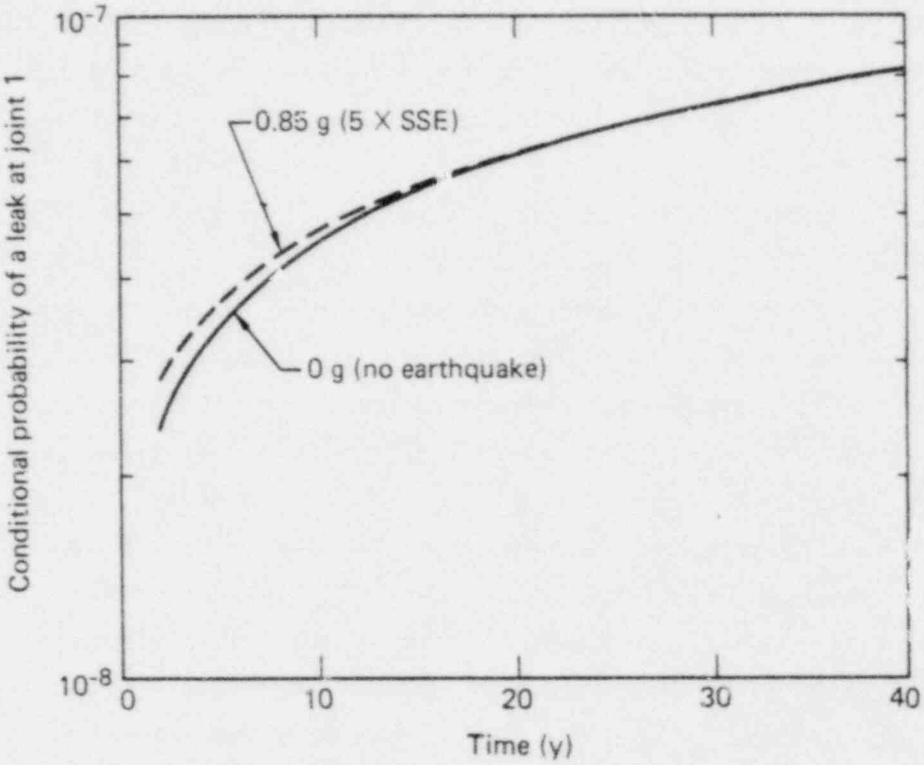

FIG. 6. The conditional probability of a leak at joint 1 of Fig. 1 during the 40-yr plant lifetime.

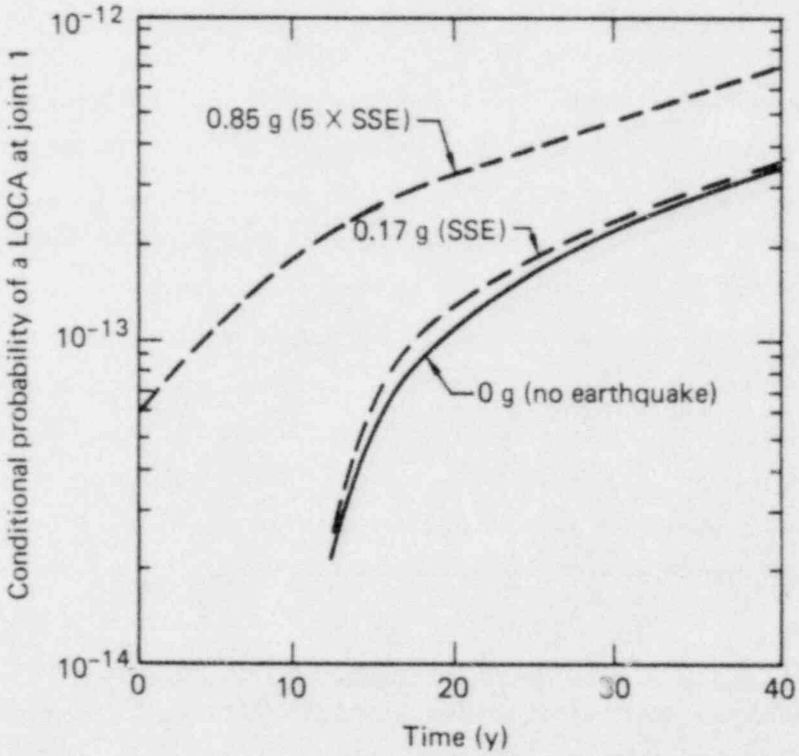

FIG. 7. The conditional probability of a large LOCA at joint 1 of Fig. 1 during the 40-yr plant lifetime for three values of peak horizontal ground acceleration.

identified asymmetric blowdown force. The probability of a leak or large LOCA at these joints is simply a subset of the reactor coolant loop failure probability. Including seismic events, the single joint LOCA probabilities are 4 x 10^{-13} and 10^{-14} for locations 1 and 14, respectively. The probability of a large LOCA and no earthquake occurring is 2.7 x 10^{-13} for joint 1 and 1.8 x 10^{-16} for joint 14.

The probability of pipe fracture was estimated from simulation. Some of the inputs (that is, models and model parameters) are not precisely known. When using the outputs for making decisions, the subjectivity of the inputs must be recognized. The two major sources of variation associated with the outputs of the simulation experiment and hence the probability analyses are uncertainty about the inputs and random variation due to simulation.

Uncertainty in the inputs is due to lack of precise knowledge about the models or the values of the parameters of the models (e.g., knowing which crack size distribution). Therefore, it is necessary to make subjective judgments about the inputs. The state-of-knowledge about the inputs can vary considerably. Knowledge can vary from that based on a statistical analysis of data from testing, to parameters chosen only to demonstrate extremes. In any case, uncertainty about inputs can be expressed by specifying probabilities that the inputs will assume values over a range of potential values. This is often stated as a probability distribution. Consequently, the outputs of the probability analyses vary. This variation can be described by a probability interval--an interval of values for the parameter of interest to which is associated a subjective probability.

A limited sensitivity study on input variables was performed for locations 1 and 14 (Fig. 1). Input variables were assigned incredible values to determine how the results vary for the following cases:

- No preservice inspection
- No preservice hydrostatic proof test
- No leak detection
- Modified Marshall initial crack distribution
- Variation of aspect ratio of the length and the depth of the initial crack
- Increase the 5 x SSE seismic stress by a factor of 3.

The limited sensitivity study indicates that leak detection capability, preservice inspection, and proof testing tend to reduce the probability of a LOCA by less than an order of magnitude.

Variation in the crack size distribution can have a significant effect on the probability of a LOCA. The Marshall distribution, as employed in this study, is believed to represent a somewhat conservative--but believable--crack size distribution. Unbelievable crack size distributions were also used and were found to increase the probability of a LOCA from one to two orders of magnitude. The crack aspect ratio also has a pronounced effect on pipe break probability. Increasing the percentage of cracks with a large length-to-depth ratio rapidly increases the probability of a large LOCA.

Both the seismic and nonseismic stresses, calculated using the best estimate inputs with state-of-the-art structural, building, and soil models, were significantly below the design stresses. Since the crack growth rate is a function of the fourth power of stress, the low stress values have a large effect on reducing the failure probability. The sensitivity analysis of the stress calculation can account for the differences between best estimate and design stresses.

Random variation is inherent in any sampling such as a simulation experiment. This variation is usually expressed by the variance of the output and by constructing interval estimates (e.g., confidence intervals) of the parameters of interest (in this case, the probability of system failure.)

For the event combination assessment, many of the inputs (e.g., crack size distribution, frequency of earthquakes and transients, stress magnitude and cycles, soil, structure, and piping stiffness and damping) are uncertain, even for the Zion plant used in this analysis. Therefore, the single value produced by the analysis is a point estimate based on the judgment made in choosing the inputs for the analysis. It is important that the variation associated with this estimate also be recognized.

The best estimate results of this analysis were compared with other estimates of pipe and pressure vessel failure probability. Westinghouse reported that the probability of a LOCA caused by an SSE is 2×10^{-11} per year.[48] This work was based upon some conservative assumptions; i.e., quarter thickness flaws, and considered only the SSE. Our results ($\approx 10^{-12}$ for any earthquake less than 5 SSE during the plant's life) compare well with the Westinghouse estimates.

Battelle has recently completed a cold leg integrity analysis.[2] They concluded that leaks will occur and be detected prior to a large break. The Battelle results further suggested that leaks should have occurred in operating PWR plants. This is contrary to operating experience. Our leak probability estimate is consistent with observations that leaks have not occurred.

A report by the Advisory Committee on Reactor Safeguards on the integrity of reactor vessels for light water power reactors, indicated that the failure rates per vessel year were about 10^{-6}.[49] This failure rate encompasses the complete range of failure from degradation through leaking to catastrophic failure. If we consider the PWR piping to be a pressure vessel, this failure probability would be compared to the probability of a leak. As such, we agree within two orders of magnitude. Periodic inspection reduces this probability bringing them closer to the current estimates.

The failure mechanisms in the PWR primary piping are believed to be more limited than in PWR secondary piping, BWR piping, or non-nuclear systems for a number of reasons. First, there is little or no evidence of stress corrosion in the PWR primary system. Further, the simplicity of the primary system should minimize or eliminate thermal fatigue as a cracking mechanisms. The toughness of piping material, joint radiography and ultrasonic nondestructive testing, and hydro tests should provide early warning of gross damage, such as incomplete welding, if gross errors existed.

In light of the various comparisons to other works and the limited sensitivity study, the best estimate analyses provide a useful and realistic estimate of the probability of leak and complete pipe break of the primary piping in Zion Unit 1.

PROBABILITY OF A LARGE LOCA INDIRECTLY INDUCED BY AN EARTHQUAKE

In addition to direct pipe fracture as a result of fatigue, a guillotine pipe fracture may be caused by an indirect means; that is, a seismic event may cause a failure in some component which in turn causes a primary piping failure. Electrical and structural failures, explosions, missiles, and fires were examined as potential causes for an indirectly induced LOCA. However, only those event sequences leading to the guillotine break of the primary coolant piping are considered in the event combination assessment.

Seismically induced electrical and mechanical failures, fires, explosions, or missiles outside of containment could cause a transient which can lead to a LOCA. An event tree was developed to evaluate the possibility. However, transient initiated LOCAs will not result in the postulated guillotine break of the primary coolant piping or similar loading.

Seismic events which can lead to an indirectly induced double-ended pipe break include both missiles and structural failures inside containment. A missile capable of striking and rupturing the primary pressure boundary may be generated inside containment by failure of the overhead crane and associated hooks, a flywheel of the reactor coolant pumps, control rod drive mechanisms, valve motor operators, secondary pipe whip, and containment dome rupture. Each source was analyzed and engineering judgment applied to estimate their probability of occurence. Given that the missile does occur, the potential targets are analyzed to determine the sequence of events and their probability that the missile will cause double ended rupture.

The reactor coolant pump flywheel and overhead crane missiles yield the highest probability of causing a guillotine pipe break. For a flywheel missile to be created, the seismic stress must be sufficient to stress the flywheel to failure. This may be due to pump seismic acceleration, vibration, bearing failure, etc. The resulting flywheel missile must then penetrate its casing and cause a failure in the primary piping. Similarly, a crane or lifting hook missile is generated due to failure of the supports or crane assembly. The crane assembly must also rotate to a position over the primary piping and the missile must strike and penetrate the primary piping. In either case, a series of failures must occur to result in a pipe break.

Closely linked to the missile type failure induced LOCA is that of structural failure inside containment as a direct consequence of the earthquake. Failure of the supports associated with the reactor pressure vessel, steam generator, reactor coolant pump, and spray-header pipe hangers, as well as structural failure of the containment dome, can lead to a double-ended pipe break.

The probability of structural failure is assessed by first calculating the design safety factor at the SSE load level. From this value of the safety factor, we estimate the earthquake magnitude (and standard deviation) for which the factor of safety is one; i.e., the earthquake magnitude which will just cause failure in the given component. The probability of failure of any one component is computed by integrating the probability density function of the annual maximum peak ground acceleration at the site times the probability of failure of any one structural component as a function of peak acceleration. The probability density function of the annual maximum peak ground acceleration is obtained from the derivative of the site-specific and seismic hazard curve. The probability of structural failure of any one of the sources is calculated to be 5×10^{-5}. This probability must then be combined with the probability that a structural failure causes a guillotine pipe break. Additional work will be performed to assess this probability.

The structural failure probability as a precursor to a pipe break does not exclude the directly induced LOCA concurrent with the indirectly induced event. Since the earthquake levels required to cause the indirect LOCA are high, the probability of a direct guillotine break may not be insignificant. The direct and indirect events should be separated into mutually exclusive events so that their probabilities can be added to estimate the probability of a LOCA. Another conservative assumption is embedded in the assessment of the safety factor based on the design stresses rather than the actual operating stresses. The design stresses often overestimate the effect of the earthquake. Further, while component failure may correspond to the reduction of the factor of safety to one, this does not generally correspond to complete failure of the structural member. Often, with a safety factor based on yield, structural failure corresponds to component yielding. Even if yielding does occur, the component will not lose total function. Even if we assume that structural failure results in loss of load-carrying capacity, we must then estimate the probability that failure will lead to other failures which in turn result in guillotine pipe break.

The discussion presented for the assessment of indirect LOCA probability did not consider pipe stresses or pipe degradation due to prior cracking. Event scenarios can be construed in which an earthquake causes a failure or transient in a "secondary" system or component which in turn yields high stresses in the primary piping.

Such events may include slosh in the steam generator, water hammer in the feedwater line, or bolting failure of the various supports. In such cases, high stresses may be introduced into the primary piping, resulting in the postulated guillotine pipe break. Unlike the previously considered events, the primary piping is not directly acted upon by a structural failure or a missile. Sequences which yield these high stresses require further investigation.

In consideration of both approaches--the event-tree analysis and the structural safety factor approach--the probability of an indirectly induced double-ended pipe break in the primary coolant loop appears to be higher than that for a directly induced LOCA. However, we caution that such an estimate is preliminary and that additional study is needed.

CONCLUSIONS

A comprehensive methodology has been developed to estimate the probability that a large LOCA and an earthquake occur simultaneously. This methodolology is the result of an extensive study that encompasses a wide spectrum of the engineering sciences--namely, seismicity, structural mechanics, fracture mechanics, and statistical mathematics. The developed methodology is used to provide an estimate of the probability of a pipe break with and without earthquakes. The results, as presented and discussed in this report, indicate that a double-ended guillotine break caused by fatigue crack growth in the primary coolant loop of the Zion Unit 1 PWR is extremely unlikely. The NRC needs such information to reassess the design requirement that the effects of a large LOCA and an earthquake are to be considered simultaneously.

This methodology also provides, as a by-product, the leak probability for a PWR primary pipe. More importantly, the developed methodology can be applied to the break of reactor coolant pipes, large or small, of other PWR and BWR plants, or to general piping reliability assessments. For example, this approach can be used to determine postulated pipe break locations based on probability.

Our results are based on assumptions that are clearly stated and fully discussed in this report. These assumptions were absolutely necessary to begin the technically complex and difficult work. They reflect our best judgment based on technical information available from the literature and some preliminary analyses. They have been reviewed by technical experts in the related fields, and they will be subject to further review.

As expected, our results are subject to uncertainty resulting from variation of input data as well as the accuracy of the analytical model. Limited sensitivity studies were conducted to observe the influence on the results of variations in many input elements. A time-consuming systematic uncertainty evaluation, however, was beyond the scope of this study.

The current study also includes a limited investigation of a possible pipe break indirectly induced by earthquakes. Budget and scheduling restraints prevented a comprehensive evaluation of this complex issue in Phase I. Nevertheless, the work reported herein establishes a basis for a more thorough evaluation in a subsequent phase.

RECOMMENDATIONS

The results of this study indicate that a double-ended guillotine break as a direct consequence of an earthquake is extremely unlikely. Although this low probability is plausible evidence for decoupling the combined effect of a LOCA and an earthquake in nuclear power plant designs, we cannot recommend decoupling now because of uncertainties involved with LOCAs that are indirectly caused by earthquakes. More importantly, the lack of a probability-based decoupling criterion prevents such a recommendation.

Whether or not to consider the combined effect of a large LOCA and a seismic event in nuclear power plant design is a very serious decision involving the public safety. We recommend continued study with emphasis on the following activities:

- Perform systematic sensitivity studies
- Perform an uncertainty evaluation
- Refine the indirectly induced LOCA evaluation
- Establish a probability-based decoupling criterion
- Assess generalizing the results to other PWR or BWR plants.

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This report summarizes work performed for the U.S. Nuclear Regulatory Commission (NRC) by the Load Combination Program at the Lawerence Livermore National Laboratory to establish a technical basis for the NRC to use in reassessing its requirement that earthquake and large loss-of-coolant accident (LOCA) loads be combined in the design of nuclear power plants. A systematic probabilistic approach is used to treat the random nature of earthquake and transient loading to estimate the probability of large LOCAs that are directly and indirectly induced by earthquakes. A large LOCA is defined in this report as a double-ended guillotine break of the primary reactor coolant loop piping (the hot leg, cold leg, and crossover) of a pressurized water reactor (PWR). Unit 1 of the Zion Nuclear Power Plant, a four-loop PWR-1, is used for this study.

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