

Load Combination Program

Progress Report No. 5

Manuscript Completed: July 1980
Date Published: September 1980

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Prepared for
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN No. A0126

8010300499

FOREWORD

Starting April 1, 1980, Load Combination activities at the Lawrence Livermore National Laboratory were officially separated from the Seismic Safety Margins Research Program (SSMRP) and became an independent program. The first quarterly progress report for Load Combination activities was provided as part of the SSMRP Quarterly Progress Report No. 3, published April 15, 1979, under the number ME79-208. The second quarterly progress report was also published under the ME series, (ME79-209), July 15, 1979. The third quarterly progress report was issued as a NUREG report under the number NUREG/CR-1120, and published October 15, 1979. The fourth Load Combination activities progress report was published April 15, 1980 as part of NUREG/CR-1120, Vol. 2. All previous progress reports are available either at Lawrence Livermore National Laboratory (ME series) or at USNRC (NUREG series).

This quarterly progress report, No. 5 to the Load Combination Program is issued under the new NUREG number.

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ABSTRACT

This document is a progress report on the Load Combination Program (LCP) covering the period April 1, 1980 through June 30, 1980. The report gives a general description of the program by project and tasks, together with financial summaries, technical reports generated, and meeting attendance. Two appendixes which discuss technical subjects are also included.

LOAD COMBINATION PROGRAM (PHASE I)
FIN A0133, A0362 and A0363

GENERAL DESCRIPTION

Personnel

NRC Branch Chief: J.E. Richardson
NRC Program Manager: J. O'Brien/M. Vagins
Contractor: Lawrence Livermore National Laboratory (LLNL)
LLNL Program Manager: C. K. Chou
LLNL Project I Leader: S. C. Lu
LLNL Project II Leader: M. W. Schwartz

Program Dates and Cost

Starting date: March 1979
Phase I completion date: August 1980
Ending date: 1985
Phase I cost: Total budget for Phase I is \$1,020,000. Of that amount, \$820,000 is sponsored by the Mechanical Engineering Research Branch (MERB). The remaining \$200,000 is supported by the Metallurgy and Materials Research Branch (MMRB) for the reliability of piping system evaluation which is part of Project I -- Event Decoupling (LOCA-SSE).

Justification

NRR User Request No. 76-5, dated June 16, 1977

INTRODUCTION

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, 10CFR50, requires that structures, systems, and components important to the safety of nuclear power plants be designed to withstand combinations of effects of natural phenomena and the effects of normal and accident conditions.

Historically, the NRC has required that the structural/mechanical responses due to various accident loads and loads caused by natural phenomena be combined in the analysis of structures, systems, and components important to safety. This requirement appears in the various regulatory documents such as regulatory guides, regulation revisions, and branch technical positions, as well as in the standard review plan.

The postulated accident loads and loads caused by natural phenomena, such as earthquakes, are random events. Amplitude, duration, frequency content, time of occurrence, and time-phase relationship are random and stochastic in nature. Without a systematic probability assessment, it is almost impossible to come up with a rationale for an appropriate combination of events. Requirements to consider dynamic events acting concurrently have, therefore, been based on judgement tending towards conservatism because no comprehensive probabilistic study and unified philosophy exists on which to base better-founded decisions. Consequently, safety margins between various systems and from plant to plant vary widely. Trying to follow the inconsistent and fragmented load combination requirements is frustrating to the nuclear industry.

A unified approach to load combination is needed by the NRC as well as the nuclear industry. This approach must evolve from a rational procedure to address the following basic issues:

1. What loads need to be combined?
2. If certain loads must be combined, how can this be done?
3. What factors should be applied to load effects or responses?
4. What are appropriate service levels, load categories, and stress limits?
5. How can pipe break locations be postulated?

The Load Combination Program undertaken by Lawrence Livermore National Laboratory (LLNL) since March 1979 represents a milestone effort funded by the NRC to resolve these difficult and important issues. Our program adopts a probabilistic approach to develop the desired reliability methodology for load combination. Three consecutive LLNL reports address themselves to the program. First, the Load Combination Project Work Plan covers the work scope of Phase I (UCID-18126, July 10, 1979). Second, Load Combination Methodology Development Interim Report I presents some intermediate results of Phase I (UCID-18149, January 1980). Third, the Load Combination Program Phase II Work Plan (UCID-18659, April 15, 1980) describe Phase II of the program, which has as its objective the satisfaction of the long-term research requirements and the short-term licensing needs.

Objectives

Overall Objectives

1. Develop a methodology for appropriate combination of dynamic loads for nuclear power plants under normal plant operation, transients, accidents, and natural hazards. The methodology is to be based on the probabilistic assessment of the reliability of components, systems, and structures.
2. Establish design criteria, load factors, and component service levels for appropriate combinations of dynamic loads or responses to be used in nuclear power plant design.

LCP (Phase I)

3. Determine the reliability of typical piping systems, both inside and outside the containment structure, and provide the NRC with a sound technical basis for defining the criteria for postulating pipe breaks. (It is anticipated that NRC Regulatory Guide 1.46, *Protection Against Pipe Whip Inside Containment*, will be revised in accordance with the findings.)
4. Determine the probabilities of a large LOCA induced directly and indirectly by a range of earthquakes.

Phase I Objectives

1. Assess the contribution to safety resulting from the combination of a large LOCA and SSE, and the cost incurred due to this requirement.
2. Determine the probability of a large LOCA induced either directly or indirectly by a range of earthquakes.
3. Develop a framework for the load combination methodology (LCM).
4. Develop the preliminary computational chain for combining responses.
5. Continue the work in connection with the reliability of piping systems (see Overall Objective 3 above) into Phase II.

Phase I activities are scheduled to be completed at the end of August 1980.

Task Description

Project I: Event Decoupling (LOCA-SSE)

Task 1: Safety Margins and Cost Assessment – This task is to assess the contribution to safety margins that result from the requirement to design for simultaneous large LOCA and SSE, and to determine the cost incurred due to this requirement.

Task 2: Probability of a Large LOCA Induced by Earthquakes – This task is to determine the probability of a large LOCA induced directly and indirectly by a range of earthquakes for a selected Westinghouse four-loop, PWR-1 plant. The probability of a directly induced LOCA is related to pipe rupture of the primary loop piping caused by pipe crack growth when the pipe is subjected to the combined effects of thermal, pressure, seismic, and other cyclic loads throughout the lifetime of the plant. In Phase I, we limit the assessment to a large LOCA, defined as a double-ended guillotine break at the primary loop piping (hot leg, cold leg, and crossover lines). The probability of a large LOCA induced indirectly by earthquakes is evaluated for structural, mechanical, and electrical failures, and explosions, fires, and missile incidents caused by earthquakes.

Task 3: Reliability of Piping Systems – This task develops a probabilistic model employing fracture-mechanics to compute the reliability of piping systems. The first-year effort of the task is integrated in Task 2, the probability of a large LOCA induced by earthquakes. The second-year effort will be applied to generate technically sound bases for postulating pipe break locations.

Project II: Load (Response) Combination Methodology Development

Task 1: Methodology Development – This task is to develop a methodology for appropriately combining generic dynamic responses. Both component and system reliability methodologies are used. The component reliability will be used to determine proper load combinations for mechanical or structural component design for a given target limit state. Also, this methodology can be used by the NRC to evaluate component reliability for current design and to compare the component reliability results of using SRSS (square root of the sum of the square) and ABS (absolute sum) response combination methods. The system reliability methodology takes into account system functionality to determine the proper target limit states for components under various plant conditions.

Task 2: Methodology Demonstration – This task is to demonstrate the feasibility and applicability of the load combination methodology.

Load Combination Program Phase I Schedule
Program Management

TASK	FY 79										FY 80										FY 81															
	F	M	A	M	J	J	A	S	O	N	D	J	F	M	A	M	J	J	A	S	O	N	D	J	F	M										
Program planning and management	001			002																003	004	005	006	007												
	▽			▽																	▽	▽		▽	▽											

Load Combination Program Phase I Schedule
 Project I - Event Decoupling (LOCA-SSE)

TASK		FY 79					FY 80					FY 81															
		F	M	A	M	J	J	A	S	O	N	D	J	F	M	A	M	J	J	A	S	O	N	D	J	F	M
Task I.1	Safety margins and cost assessment						101 ▽					102 ▽	103 ▽	104 ▽													
Task I.2 - LOCA-SSE Decoupling	Subtask I.2.1 - Directly Induced LOCA						105 ▽	106 ▽																			
	1.2.1.1 Fracture mechanics and material science																										
	1.2.1.2 Loads and stresses											109 ▽															
	1.2.1.3 Probability of large LOCA directly induced by earthquakes																										
	Subtask I.2.2 - Indirect	1.2.2 Probability of large LOCA indirectly induced by earthquakes											110 ▽	111 ▽	112 ▽												
Subtask I.2.3 - Combine	1.2.3 Combined probability of large LOCA directly and indirectly induced by earthquakes																										
Task I.3	Reliability of piping systems																										

Load Combination Program Phase I Schedule
Project II - Load Combination Methodology Development

TASK		FY 79										FY 80										FY 81				
		F	M	A	M	J	J	A	S	O	N	D	J	F	M	A	M	J	J	A	S	O	N	D	J	F
Task II.1	Response combination: develop detailed methodology and approach							201		202		203	204	205												
								▽		▽		▽	▽	▽												
Task II.2	Response combination: methodology demonstration													207	209	211										
														206	210	212										213
													▽	▽	▽	▽	▽	▽	▽	▽	214					

Load Combination Program
Program Management
 Milestone

Milestone	Target Date	Completed Date
001 Kickoff Meeting for Phase I Activities (9.1)*	3-1-79	3-1-79
002 Phase I Work Plan Completed (9.2)*	5-15-79	5-25-79
003 Preliminary Phase II Work Plan Completed (New)	4-15-80	4-15-80
004 Phase II Work Plan Clarification Meeting (New)	5-21-80	5-21-80
005 Peer Review Panel Formulation (New)	6-1-80	6-10-80
006 First Peer Review Panel Meeting (New)	9-15-80	
007 ACRS Meeting (New)	8-29-80	

*Milestone number given in the April 15, 1980 progress report, NUREG/CR-1120, Vol. 2

Load Combination Program
 Project I – Event Decoupling (LOCA-SSE)
 Milestone

Task	Milestone	Target Date	Completed Date
Task I.1	Safety Margins and Cost Assessment		
	101 Safety margins and cost assessment subcontractor on board (9.6)*	8-15-79	10-1-79
	102 Safety margins assessment (9.7)*	2-20-80	3-1-80
	103 Cost assessment (9.8)*	3-15-80	4-30-80
	104 Task completion and final report (9.9)*	4-28-80	
Task I.2	LOCA-SSE Decoupling		
Subtask I.2.1	Probability of large LOCA directly induced by earthquakes		
	I.2.1.1 Fracture Mechanics and Material Science		
	105 Reliability of piping system joint work scope completed (9.10)*	5-15-79	4-24-79
	106 Subcontractor on board (9.11)*	7-16-79	8-9-79
	107 Fracture mechanics work completed (9.12)*	7-18-80	
	108 Task completed (9.13)*	8-15-80	
	I.2.1.2 Loads and Stresses		
	109 Primary loop piping model subcontractor on board (9.14)*	8-17-79	8-17-79
	110 Transient time history data collection completed (9.15)*	3-14-80	3-14-80
	111 Primary loop piping model completed and finite element model for complex geometry development begun (9.16)*	2-15-80	2-15-80
	112 Structure building response obtained from SSMRP Project IV and finite element models completed (9.17)*	4-1-80	5-15-80
	113 Stress calculations completed for input to probabilistic fracture mechanics model (9.18)*	7-4-80	

*Milestone number given in the April 15, 1980 progress report, NUREG/CR-1120, Vol. 2

Load Combination Program
Project I – Event Decoupling (LOCA-SSE)
 Milestone (Continued)

Task	Milestone	Target Date	Completed Date
	114 Task completed (9.19)*	7-18-80	
I.2.1.3	Probability of Large LOCA directly induced by earthquakes		
	115 Probabilistic model development begun (9.20)*	7-2-79	8-9-79
	116 Seismic hazard curves obtained from SSMRP (9.21)*	5-15-80	
	117 Probabilistic model completed, uncertainty propagation methodology developed (9.22)*	7-11-80	
	118 Stress and fracture mechanics input obtained (9.23)*	7-18-80	
	119 Probability of LOCA directly induced by earthquakes calculated (9.24)*	8-15-80	
Subtask I.2.2	Probability of large LOCA indirectly induced by earthquakes		
	120 Structural, mechanical, and electrical failure mode identification begun (9.25)*	9-14-79	9-14-79
	121 Study of fire, missile, and explosion caused by earthquakes begun (9.16)*	2-15-80	2-15-80
	122 Calculation of probability of LOCA indirectly induced by earthquakes begun (9.27)*	7-11-80	
	123 Probability of LOCA induced indirectly by earthquakes calculated (9.28)*	8-15-80	
Subtask I.2.3	Combined probability of large LOCA induced directly and indirectly by earthquakes		
	124 Work planning begun (9.29)*	2-1-80	2-15-80
	125 Inputs obtained (9.30)*	8-15-80	
	126 Task completed (9.31)*	8-29-80	
Task I.3	Reliability of piping systems Integrated with Subtask I.2.1		

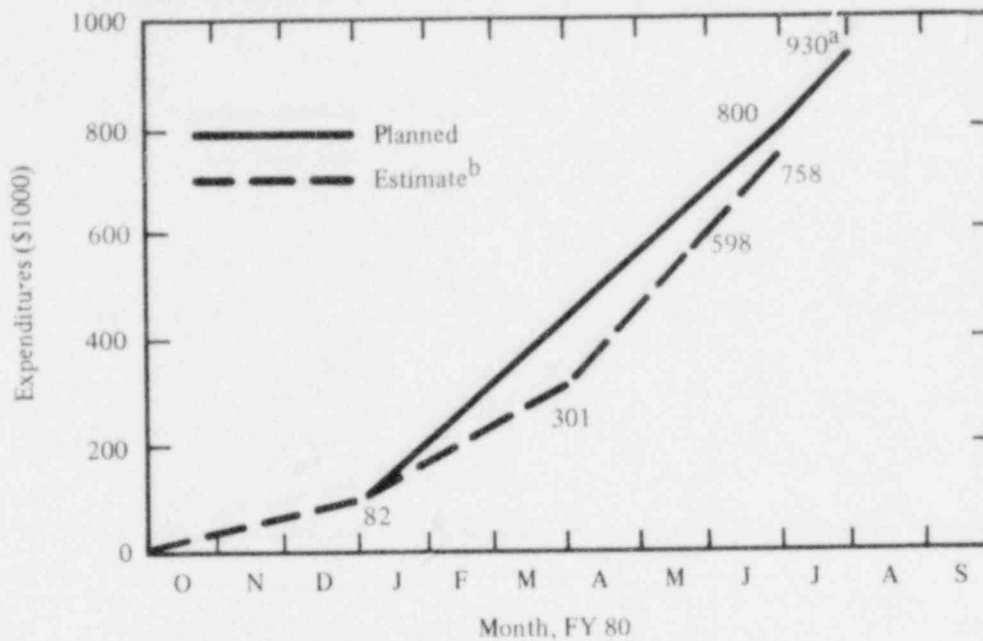
*Milestone number given in the April 15, 1980 progress report, NUREG/CR-1120, Vol. 2

Load Combination Program
Project II – Load Combination Methodology Development
 Milestone

Task	Milestone	Target Date	Completed Date
Task II.1	Load (Response) Combination Methodology Development		
201	Response (load) combination methodology development subcontractor on board (9.32)*	8-22-79	8-20-79
202	Review outputs from NRC Task B-6 and other research programs completed (9.33)*	11-2-79	Suspended
203	Detailed approach and methodology for dynamic response combination completed (9.34)*	1-15-80	1-15-80
204	Meeting with technical community (9.35)*	1-25-80	Suspended
205	Approach and methodology finalized (9.36)*	1-30-80	2-8-80
Task II.2	Load (Response) Combination Methodology Demonstration		
206	Set of loads and specified system selected (NRC) (9.37)*	3-10-80	3-10-80
207	Loading data developed (9.38)*	4-11-80	6-1-80
208	Dynamic load characters studied (9.39)*	5-9-80	6-30-80
209	Required modeling (structure, subsystem, etc.) completed and stochastic process of loads studied (9.40)*	5-9-80	6-30-80
210	System event/fault trees completed and structural and subsystem dynamic characters studied (9.41)*	5-9-80	6-30-80
211	Computer runs for ABS, SRSS, and other identified methods completed and dynamic response characters studied (9.42)*	6-20-80	
212	Component and piping failure evaluated (9.43)*	7-25-80	
213	System reliability determined from event/fault trees for ABS, SRSS, and other response combination methods (9.44)*	8-8-80	
214	Comparison of results completed and recommendations issued (9.45)*	8-29-80	

*Milestone number given in the April 15, 1980 progress report, NUREG/CR-1120, Vol. 2

Expenditures for FY 80, Load Combination Program (FIN A0133, A0362, and A0363)



Load Combination Program
Phase I Budget Summary FIN A0133, A0362 and A0363

	Budgeted Amounts (\$1,000)	Expenditures (\$1,000)
Total budget for Phase I (through June 1980)	1,000 ^c	
Funds in support of SSMRP	-94	
Total actual budget for Phase I	906	
Expenditures:		
Prior Year (FY (79)		106
Current Year (FY 80)		758
Additional funds received from NRC ^a		
Program Execution for the month of July 1980	100	
Peer Review Panel	30	
Total funding expenditures	1,036	864
Balance of funds not expended	172 ^d	

a. Additional \$130,000 received from NRC in May for the support of the Peer Review Panel (\$30,000), and for Program Execution (\$100,000) for July 1980.

b. Current FY 80 expenditures plus subcontract obligations through June 30, 1980, totals \$1,054,000.

c. Received \$80,000 from SSMRP A0126 rather than expected \$100,000.

d. Additional \$214,000 required to carry program through September 1980.

LOAD COMBINATION PROGRAM

A. Accomplishments

1. Program Management

On April 1, 1980, load combination activities at the Lawrence Livermore National Laboratory (LLNL) were officially separated from the Seismic Safety Margins Research Program (SSMRP) and became an independent program. Dr. C. K. Chou was appointed Program Manager. The program has two projects remaining for Phase I, which is scheduled to be completed at the end of August 1980. Dr. Steve Lu was appointed Project Leader for Project I, Event Decoupling (LOCA-SSE), and Dr. Martin Schwartz was designated Project Leader for Project II, Load Combination Methodology Development. The projects are described in detail in the subsections that follow.

We have completed the Phase II work plan. It calls for the start of Phase II on July 1, 1980, with completion by the end of December 1981, and will have a special feature of utilizing intermediate results from long-term research to help make short-term licensing decisions. We believe that in order to conduct the systematic probabilistic assessment, develop and demonstrate reliability methodology, evaluate and improve component reliabilities, and develop a more rational design criteria, five or more years of long-term research effort are required. However, the intermediate products from the long-term development can definitely be used to make better decisions on short-term licensing requirements. Because of this, a special project is planned in Phase II to integrate long-term results into short-term decision making. The plan, "Load Combination Program Phase II Work Plan," UCID-18659, which includes the special project description, was issued for NRC and ACRS comments on April 15, 1980. A clarification meeting was held at Bethesda, Maryland, on May 21, 1980 for NRR and RES. We presented our philosophy, technical approach and work detail in the meeting and answered questions.

We have received formal instructions from the NRC to form a Load Combination Peer Review Panel. Five members from the fields of metallurgy and material science, risk and reliability, structural and mechanical analyses, and mathematical analysis were selected for the panel. Invitations have been accepted by all the candidates, and contract negotiations have been started. We anticipate that all contracts will be signed by the first week of August 1980. Because most of the Review Panel members are from the academic field, the first Review Panel meeting will not be held during the summer. At this time, we plan to have the first meeting in September 1980.

A meeting to report to the ACRS Subcommittee on Combination of Dynamic Loads has been requested. A preliminary schedule has been set for the end of August 1980. We believe by that time both Phase I projects will be close to completion. Meaningful results and a good technical approach are expected to be presented to the ACRS.

2. Project I - Event Decoupling (LOCA-SSE)

When Project IX activity was separated officially from the SSMRP on April 1, 1980, it was divided, in turn, into two projects as previously discussed. Project I, for the remaining period of Phase I, has essentially the same scope of work, the continuation of Tasks I, II and IV, as they were defined originally in the Phase I work plan. The activities of these tasks are Safety Margins and Cost Assessment, Probability of LOCA Induced by Earthquake, and Reliability of Piping System, respectively. In this progress report, however, the tasks have been identified as Task I.1, Safety Margins and Costs Assessment, and Task I.2, Decoupling of Large LOCA and Earthquake (Zion-1). To provide continuity between progress reports and to avoid confusion, prior task identifications used in the April 15, 1980 SSMR Progress Report are provided as cross-references in the descriptions that follow.

Task I.1, Safety Margins and Cost Assessment (Task IX.3, April 15, 1980 Progress Report)

– The draft report on this task was received the first week of May from Dr. John Stevenson of Woodward-Clyde Consultants, the subcontractor for the task. Dr. Stevenson's report has been reviewed by the Load Combination Program technical staff and their review comments will be incorporated in the final report.

Task I.2, Decoupling of Large LOCA and Earthquake (Zion-1)

a. **Subtask I.2.1, Probability of Directly Induced Large LOCA** – As shown on the LCP Phase I work plan, this subtask is divided into three separate tasks identified as I.2.1.1 through I.2.1.3.

1.2.1.1 Fracture Mechanics, Material Science, and Reliability of Piping Systems (Tasks IX.4 and IX.6, April 15, 1980 Progress Report) – Science Application, Inc. (SAI) is the subcontractor for this work, while the technical staff of LLNL, in the areas of probability fracture mechanics and stress calculations, are heavily involved in the development of the probabilistic fracture mechanics model. The probabilistic model is completed as well as a computer code to simulate single weld joint crack growth and the probability of failure. A test run to calculate a single joint failure rate by using stress input, generated from Zion original design seismic ground motion and design transients data, is near completion. Data collection for the fatigue growth of austenitic piping steel is complete. Data from Ford, Bamford, French, Shahinian, and G.E. have been used to determine the crack growth rate distribution. The state space for fatigue crack growth in piping is developed. The probability of cracks existing in various regions of state space has been generated from the existing data base. Leak rate calculations to predict the size of leak for a given size of crack have been developed. Techniques for calculating stress intensity factors for arbitrary crack sizes (depth and length), and for evaluating stresses are now available for circumferential interior surface cracks in the piping weldments.

The analysis of all weld joints in the primary piping, including a stochastic stress history based on Zion operating experience, is underway. The stress history forecasting model is developed, and should be available soon.

1.2.1.2 Loads and Stresses (Task IX.4B, April 15, 1980 Progress Report) – The development of the Zion 1 reactor cooling loop models for computing stresses due to seismic excitations, as well as weight and thermal loadings, was completed by the subcontractor (Sargent and Lundy). The models are described in detail in a draft report which has undergone technical review by LLNL as well as Westinghouse. All review comments and suggestions will be reflected in a final report to be published in August.

The plant transient forecast is underway. The forecast is based on Zion operating records, EPRI study results, and FSAR design requirements. A comparison study between the forecasted and the design transients is part of the effort and the results will be included in a future report, a first draft of which will be available in August 1980.

Thermal and weight stresses are being calculated by Science Application, Inc. (SAI). Thermal expansion stress analyses for the reactor coolant loop piping system and thermal gradient stress calculations for the hot leg and cold leg were completed for the model, with equipment support stiffness generated by Sargent and Lundy. At the end of May, after three months of negotiations with Westinghouse, with the assistance of Commonwealth Edison, a new set of equipment support stiffness data, generated by Westinghouse, was forwarded to LLNL. The model was then modified. We are rerunning the stress analyses for the new model. The results between two different sets of support stiffness data will be compared to

help understand the effect of the support system on the piping stress results. The calculation for the surge line is underway.

Detailed thermal gradient stresses in the two nozzles at the surge line terminations are being evaluated at LLNL. A finite element mesh was generated for the hot leg nozzle. Thermal stress history for the reactor trip transient has been obtained using finite element computer codes TACO and NIKI2D. Stress analyses for other operating transients are underway.

Evaluation of vibrational and residual stresses has also started at SAI. The calculation of residual stresses is undertaken by Dr. Rybicki of the University of Tulsa through an SAI subcontract.

Seismic stress analyses are conducted at LLNL. Simulated ground motion time histories at the Zion site are being generated with consultation services provided by Dr. N. C. Tsai of NCT Engineering, Inc. A computer code to automatically link the soil structure interaction analysis, the containment building floor response calculation, and the RCL piping stress analysis was to have been available from the SSMRP, but has not been completed. Without this linkage code, an alternative procedure had to be adopted. This procedure is capable of computing the containment building floor response and the seismic stresses in the RCL piping system, but is not able to consider the soil structure interaction which is an integral element of the linkage code.

Additional efforts related to computer code modification, debugging, and verification have been required in order to establish the alternative analytical procedure. One of these efforts is the modification of SAP4 to provide the option of generating time-history responses of relative displacement, velocity, and absolute acceleration at selected nodes, as well as the response spectra of the nodal acceleration time-history records. Another deals with the debugging and verification of SAPPAC, a finite element computer code developed at LLNL for dynamic analysis of a structural or piping system subjected to multiple support input motion. Also, the manpower required to manually drive the analytical sequence will be increased significantly. The impact on the project schedule created by adopting the alternative procedure is yet to be determined.

1.2.1.3 Probability of Large LOCA Directly Induced by Earthquakes (Task IX.4C, April 15, 1980 Progress Report) – The probabilistic model was completed and presented to NRC/RES in February 1980. Continuation of technical development is currently underway at LLNL.

b. **Subtask 1.2.2, Probability of Large LOCA Indirectly Induced by Earthquakes (Task IX.4D, April 15, 1980 Progress Report)** – Work has been started at Sargent and Lundy to evaluate the probability of large LOCA caused by structural failures indirectly induced by earthquakes. Science Application, Inc. has been the other subcontractor working on indirectly induced LOCA caused by other sources, i.e. mechanical and electrical failures, explosions, fires, missiles, etc. Two reports are being generated. The draft of the first report, describing all possible scenarios by which LOCA can be indirectly caused, has been presented to LLNL. The second report, giving estimates on the probability for each of the scenarios, will be available at the completion of this task in August 1980.

c. **Subtask 1.2.3, Combined Probability of Large LOCA Directly and Indirectly Induced by Earthquakes (Task IX.4E, April 15, 1980 Progress Report)** – Some preliminary concepts have been developed for evaluating the combined probability of directly and indirectly induced large LOCA's. More refinement is under development.

3. Project II – Load Combination Methodology Development

The objective of this project is to develop load combinations to be used in nuclear component design, and to provide NRC with a sound, technically based evaluation tool for reviewing the existing design. Numerical values of load and resistance factors in these load combinations are being selected so that the components will achieve desired reliabilities. The scope of Phase I of this project includes formulation and demonstration of the methodology to derive load combinations based on component reliability.

The methodology consists of selecting a set of load combinations with load and resistance factors, designing a wide class of components, and performing a reliability analysis to evaluate the limit state probabilities. These steps are iterated for different values of load and resistance factors until acceptable component limit state probabilities are achieved. The procedure could also be used for evaluating the reliability of existing design criteria.

Our original approach to load combination methodology required that, for each set of load and resistance factors, the components under all design situations be analyzed and designed accordingly. This calls for a large number of costly structural system analyses and piping subsystem analyses. Also, the design of a piping subsystem generally requires several iterations of analysis for different arrangements of pipe restraints. A simulation technique has been developed to avoid this prohibitively expensive analyses. Component designs are generated from the influence coefficients data developed from existing designs.

The system of computer programs being developed as part of the methodology includes DESIGN PROGRAM, LOAD AND RESPONSE ANALYSIS PROGRAMS, LIMIT STATE PROBABILITY ANALYSIS PROGRAM, and LOAD COMBINATION PROGRAM. The first three programs have been completed. The LOAD COMBINATION PROGRAM will be completed by July 7, 1980.

The methodology will be illustrated with two examples. The first example is the Essential Service Water (ESW) Piping in a PWR subjected to seismic and hydraulic transient loads, in addition to self-weight and pressure. The procedure for evaluating the limit state probability of existing design criteria and for developing load combinations will be illustrated. Data on influence coefficients, loads and component fragilities have been collected. Preliminary results will be available by July 21, 1980. The work on the ESW line was undertaken primarily to provide an early check on computer program adequacy.

The second example is the SR/V line between the drywell floor and the quencher in a Mark II BWR plant. This piping is a relatively simple system because there are only 13 rigid supports, including two anchors, and there are no snubbers, valves or other in-line equipment. The loading condition, on the other hand, is comprehensive, and the following loads are included in the demonstration of load combination methodology:

1. Weight
2. Thermal expansion
3. Seismic – OBE, SSE
4. SRV steamhammer
5. SRV building response
6. SRV drag (radial and tangential on submerged portion)
7. Condensation – oscillation response
8. Condensation – oscillation drag
9. Chugging response
10. Chugging drag
11. Thermal transients
12. Pressure loads

The influence coefficient data has been collected and analyzed. The seismic load analysis of the structural system is underway. Data on the pool dynamic loads and thermal transients will be collected by August 1, 1980. This illustration is more extensive than the first example, and includes

the treatment of one load event initiating other loads on the component. It is expected to be completed by the first week in September 1980.

During this reporting period, the J.H. Wiggins Company made progress in the development of an approach for the allocation of failure probabilities to components of nuclear power plant systems subjected to combined loadings. The approach requires the use of event trees to model system interactions as a result of an initiated event. After allocating limit state probabilities at the system level, the methodology requires the use of fault trees to allocate to the component service level. A simple system was defined which will allow a numerical example of the approach to be presented.

B. Next Quarter

Load Combination Program Phase I activities, both Event Decoupling and Load Combination Methodology Projects, are scheduled to be completed by the end of August 1980. We anticipate that our Phase I results will be presented to the NRC for review and comment during the next quarter. We plan to have the first Peer Review Panel meeting during the next quarter. It is likely that the technical presentation meeting with the ACRS Subcommittee on Combination of Dynamic Loads will be held in the next quarter as well.

C. Concerns

1. Technical

a. The results of calculations for various weld joints, with realistic stress histories, will provide information on the sizes of cracks required to cause piping failures. Such results will then indicate how important crack initiation is, and whether or not extended effort is necessary to evaluate initiation in the pipe crack growth model. Additionally, the effect of subsurface and base material defects probably will not be as significant as anticipated because of the strong possibility that surface defects will dominate the failure probability.

b. The temperature design transients for the surge line are not given in the Westinghouse Design Transient report nor in Zion records. However, the flow directions and flow rates are given. The surge line temperature transients are determined using the hot leg temperature, the system pressure, and the pressurizer surge rate. Specifically, for flow going into the pressurizer, the hot leg temperature history is used. For flow going out of the pressurizer, the saturated temperature at the reactor system pressure is used.

c. Because the computer code to link automatically the soil structure interaction analysis, the containment building floor response calculation, and the RCL piping stress analysis has not been made available from SSMRP, an alternative procedure for handling seismic stress history calculations has been under development. The alternative procedure, if adopted, will not be able to consider soil structure interaction (SSI) because SSI is an integrated element of the linkage code to be provided by SSMRP. We plan to generate seismic stresses for the fixed base condition (without SSI) first. When SSI becomes available to us, we plan to rerun some of the computations, if time and budget permit. The comparison between the effect of pipe failure due to SSI and with no SSI can be very meaningful additional information.

d. As we stated in the last quarterly progress report, budget and schedule restraints have limited both the detail and the depth of the evaluation of indirectly induced LOCA

during Phase I. The outcome is that the confidence intervals for directly and indirectly deduced LOCA are of different size and nature. Therefore, confidence intervals for the combination of both direct and indirect LOCA are difficult to evaluate. We propose to estimate "equivalent" samples for all LOCA events and estimate probabilities and CI's from those samples. The proposal is provided for review and comment in the technical appendix of this report.

e. Load Combination Methodology Development and Demonstration – As we stated in the last quarterly progress report, the component reliability methodology to determine proper load combination for mechanical or structural component design has been developed. This methodology can be used by NRC to evaluate component reliability for the current design. However, in order to make use of the methodology more practical and attractive, it is necessary that it require very little computation and cover as many variations as possible. To achieve this objective we have chosen the influence coefficients, obtained from various existing designs from industry, to optimize the load combination requirement. This process, if successful, can make the methodology a very efficient tool and can generate design load combination requirements simpler than the current criteria used by industry. If the convergence is not as expected, we may have to subdivide the component into a few service levels, as it is now used in industry. In addition to a more complex demonstration on the SRV lines, we have also conducted evaluations on two simpler examples. The purpose is to understand the effect of component reliability due to the various influence coefficients. We plan to carry this process through Phase II. A very simple description of our methodology is provided for review and comment in the technical appendix of this report.

2. Schedule

As we stated in the last quarterly progress report, the increased scope in the load combination methodology demonstration and the delay in obtaining the necessary computer models and seismic inputs from the SSMRP caused a two-month delay in completion of Phase I. The current target schedule is the end of August 1980. At this time, our estimate is that load combination methodology demonstration results will be developed for review and presented to the ACRS at that time. In regard to the Event Decoupling (LOCA-SSE) effort, because seismic input and linkage codes are not available from the SSMRP, we may not have complete results by the end of August 1980.

The first Peer Review Panel meeting is planned for the second week of September because of the fact that most panel members are not available during the summer. It is most desirable that the ACRS meeting be held after we have a chance to hear the comments from the Peer Review Panel. However, at this time the ACRS meeting is scheduled for the end of August 1980. We suggest that if it is possible, the ACRS meeting be rescheduled for a later time.

3. Cost

In the last quarterly progress report, April 1980, we indicated that an additional \$200,000 would be required to carry the Load Combination Program through September 1980. At that time, we were under the impression that only \$94,000 had been taken in support of the SSMRP and that this would make our actual requirement \$294,000. Since that reporting time, we discovered that \$114,000 had actually been taken in support of the SSMRP, resulting in our now requiring \$314,000 to carry the program through September 1980. During the month of May we received \$130,000 from the NRC, \$30,000 for the Peer Review Panel and \$100,000 for program execution. By adding these additional funds, the total budget for FY 1980 is \$930,000. Total expenditure at the end of June 1980 was \$758,000. The balance is \$172,000 which includes \$30,000 for the Peer

Review Panel. Therefore, the remaining \$142,000 is just enough for the month of July. (We are at the peak of the spending at this time, approximately \$160,000 per month, which includes LLNL and subcontractors.) If no additional funds are available, the program activities have to be completely stopped July 31, 1980. By July 5, 1980, a decision has to be made whether we should immediately stop all subcontract work (approximately \$110,000/month) and utilize the money to support in-house manpower through September, or whether we should start transferring in-house manpower and completely close the program July 31, 1980. We request that immediate instructions be given by the NRC.

REPORTS GENERATED BY THE LCP

1. C. K. Chou, K. Vepa, L. George, and P. D. Smith, "Seismic Safety Margins Research Program - Project VII - Load Combination Project - Work Plan," July 10, 1979, UCID-18126.
2. C. K. Chou, R. W. Mensing, P. D. Smith, K. Vepa, J. D. Collins, C. A. Cornell, R. P. Kennedy, and M. K. Ravindra, "Seismic Safety Margins Research Program Load Combination Project - Task 3, Load Combination Methodology Development Interim Report I," January 31, 1980, UCID-18149.
3. "Zion Station Reactor Coolant Loop Model Development," Preliminary, February 1980, Prepared by Sargent and Lundy.
4. C. K. Chou, S. C. Lu, and M. W. Schwartz, "Load Combination Program, Phase II Work Plan," Draft, April 15, 1980, UCID-18659.
5. "Secondary Caused LOCA's," Interim Report, Draft, June 1980, Prepared by Science Applications, Inc.
6. Quarterly Progress Reports, April 15, July 15, October 15, 1979 and April 15, 1980.
7. "Safety and Cost Assessment of the Effects of Simultaneous Occurrence of Large LOCA and SSE Loads," Preliminary, April 30, 1980, Prepared by Dr. John Stevenson of Woodward-Clyde Consultants.

REPORTS TO BE GENERATED BY THE LCP DURING THE REMAINING PHASE I

1. "Zion Station Reactor Coolant Loop Model," Final Report, Prepared by Sargent and Lundy. Scheduled August 1980.
2. "Zion Station Operating Transient Forecasting Report," Preliminary, Prepared by Lawrence Livermore National Laboratory. Scheduled August 1980.
3. "Safety and Cost Assessment of the Effects of Simultaneous Occurrence of Large LOCA and SSE Loads, Final Report," Prepared by Woodward-Clyde Consultants. Scheduled September 1980.
4. "Load Combination Methodology Demonstration Interim Report II," Prepared by Lawrence Livermore National Laboratory. Scheduled September 1980.
5. "Probability of LOCA Induced Indirectly by Earthquakes," Preliminary Report, Prepared by Science Applications, Inc. Scheduled September 1980.
6. "Load Combination Methodology Development and Demonstration, Phase I Final Report," Prepared by Lawrence Livermore National Laboratory. Scheduled November 1980.
7. "LOCA - Earthquake Event Decoupling, Phase I Final Report," Prepared by Science Application, Inc. Scheduled November 1980.

Meeting Attendance Summary

LCP (Phase I)

Date	Meeting	Location	Attending	Comments
4/1/80	Load Combination Methodology Committee	Chicago, IL	CKC, MWS, RM, PDS	
4/4/80	SAI	Palo Alto, CA	LLG	Crack growth simulation
4/9/80	SAI	Palo Alto, CA	LLG	Transient data analysis
4/9/80	Pipe Crack Study Group	Bethesda, MD	CKC	
4/10/80	S. Hanauer, R. Bosnaek, R. Mattu, J. O'Brien of NRC	Bethesda, MD	CKC	Load combination activities review and Phase II planning
4/17/80	D. Harris of SAI	Palo Alto, CA	CKC	Discuss next year's activities
4/29/80	SAI Technical Progress Report on Indirectly Induced LOCA	LLNL	RAL, JEW, FMG, CKC, SCL	Progress report on Indirect LOCA Task
4/29/80	Dr. J. D. Collins on System Reliability	Los Angeles, CA	MWS	
5/1/80	SAI	Palo Alto, CA	LLG, CD	Pipe fracture simulation
5/5/80	Dr. C. A. Cornell	LLNL	MWS, CKC	Technical meeting
5/12/80	NCI, Inc.	LLNL	GLG, RAL, SCL	Seismic input and dynamic modeling
5/14/80	SAI	LLNL	CKC, CD, LLG, RAL, SCL	Progress report on thermal stress
5/19/80	NCI, Inc.	LLNL	RAL, SCL	Seismic input
5/20/80	Pipe Crack Study Group	Silver Spring, MD	CKC, LLG	

Meeting Attendance Summary

Date	Meeting	Location	Attending	Comments
5/21/80	Load Combination Program Phase II Work Plan Presentation to NRC	Bethesda, MD	CKC, MWS, SCL, PDS, LLG	
5/21/80	Load Combination Methodology Technical Meeting with C. A. Cornell, M. K. Ravindra	Bethesda, MD	CKC, MWS, PDS, SCL, LLG	
5/22/80	SAI	LLNL	CD, RAL	Thermal stress analysis
5/23/80	Meeting with Prof. Shinozuka of Columbia University	New York, NY	CKC	Peer Review Panel
5/19/80	Meeting with M. K. Ravindra on Load Combination Methodology	Chicago, IL	MWS	
6/17/80	SAI	Palo Alto, CA	LLG	Transient analysis and multiple joint simulation
6/20/80	NCT, Inc.	LLNL	RAL, SCL	Seismic input
6/21/80	SAI	Palo Alto, CA	CD	Surge line thermal stress analysis

CKC = C. K. Chou
 LLG = L. L. George
 MWS = M. W. Schwartz
 RM = R. Mensing
 RAL = R. A. Larder
 JEW = J. E. Wells
 FMG = F. M. Gilman
 PDS = P. D. Smith
 SCL = S. C. Lu
 CD = C. Dutton
 GLG = G. L. Goudreau

TECHNICAL APPENDIXES

Appendix A, Event Decoupling (LOCA-SSE), and Appendix B, Load Combination Methodology, are included in this progress report to provide more detailed technical information on key activities of the Load Combination Program. The discussions of both A and B reflect Projects I and II of the Program, respectively, but both the technical approach followed and the results reported are preliminary in nature. Therefore, it should be emphasized that the information presented is incomplete in its current form and has been provided for review and evaluation by NRC's Technical Staff. We at LLNL are looking forward to receiving comments and suggestions from NRC at an early date following that review so that they can be incorporated into the Load Combination Program.

APPENDIX A EVENT DECOUPLING (LOCA-SSE)

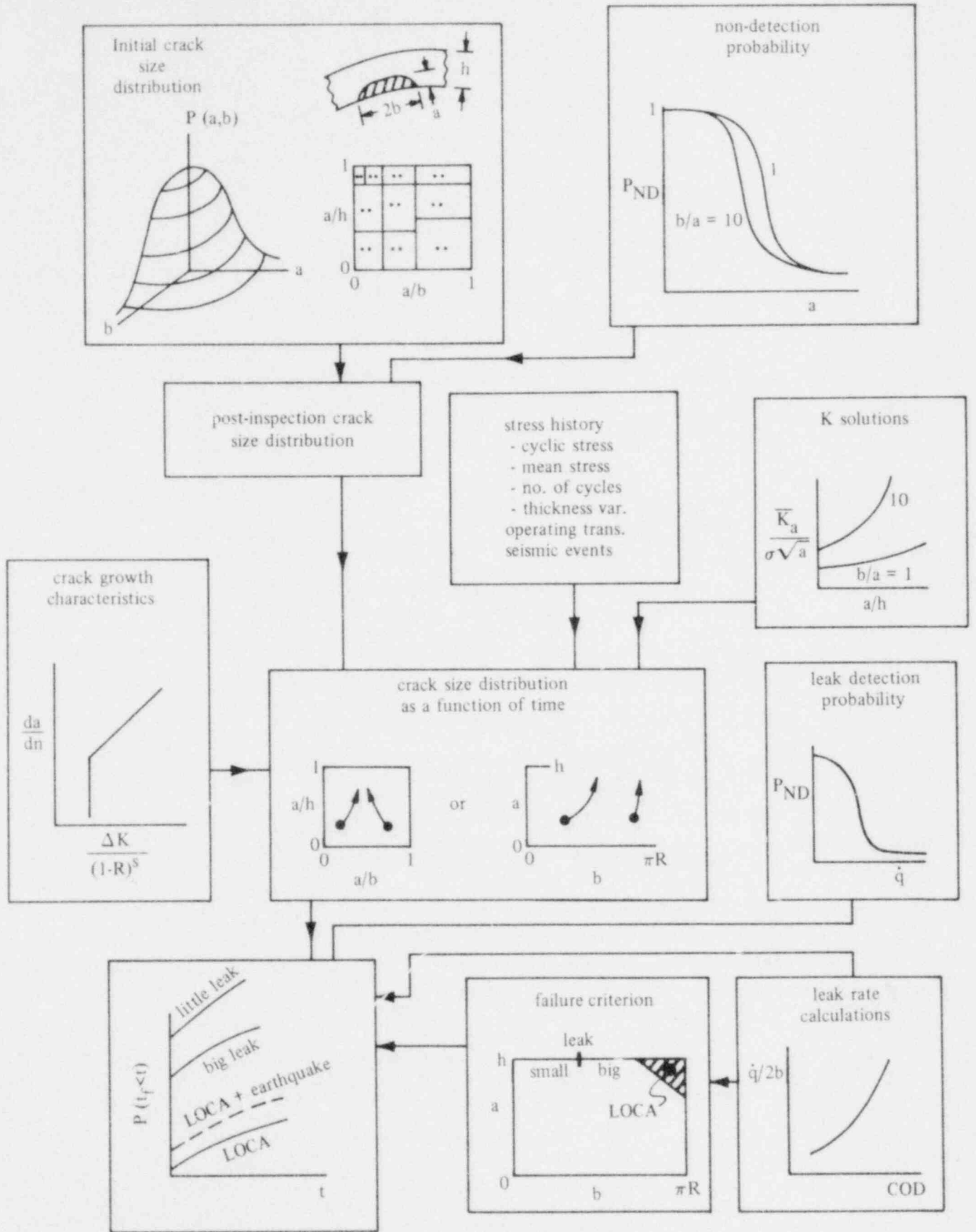
A.1 Preliminary Fracture Mechanics Calculation

This section of Appendix A presents the methodology for calculation and preliminary results of fatigue crack growth, leak, and failure (i.e., LOCA) analyses for the primary coolant loop piping system of Zion 1. The method employs a deterministic fracture mechanics model for fatigue crack growth, combined with random initial crack size, detection probability, material properties (including fatigue, yield and ultimate strength, and fracture), stress histories (both amplitude of stress and number of cycles), and leak detection probability. Figure A.1-1 shows the calculation methodology for evaluation of failure probability.

The initial length and depth distribution of thumbnail-shaped (i.e., semi-elliptical) cracks is characterized using a bivariate distribution as shown in Figure A.1-2 for a 15-inch I.D., 2.5-inch wall thickness section of pipe. While some limited data exists on the distribution of crack depth, a/h on the figure, the depth-to-width parameter, a/b is less well defined in the analysis. The assumption is made that the crack length (i.e., $2b$) must be greater than or equal to the crack depth, a . As can be seen in Figure A.1-2, the vast majority of crack sizes (99.99% of all cracks) are contained within two regions of the state space. Because such cracks are generally small, we have found that they do not contribute significantly to the actual failure probability. On the other hand, the cracks characterized in the upper left region of the state space of Figure A.1-2 are large in size. Although they have a low probability of occurrence, they will generally result in failure, thereby contributing to failure probability. The statistical modeling of cracks will employ a stratified sample emphasizing the regions around the diagonal from $a/b = 0$, $a/h = 0$, to $a/b = 1$, $a/h = 1$, since the result of growth of these cracks is unknown. Once the crack size characteristics are determined, the model is then adjusted to the probability that a flaw exists and is not detected.

The fatigue model for crack growth employs a Paris-type growth rate equation modified for the mean stress level, i.e., $\frac{da}{dn} = C(K')^4$ where the exponent of 4 was evaluated from the available data, and the coefficient, C is assumed to be log normally distributed (see Figure A.1-3). The expression for K' accounts for the mean stress by: $K' = \Delta K / (1-R)^{1/2}$ where R is the ratio of minimum to maximum stress. The distribution of C is based on the data from five sources (see Figure A.1-3); its mean value is 9.14×10^{-12} with a standard deviation of 3.01×10^{-11} for K' in units of $\text{ksi-in}^{1/2}$ and da/dn in inch/cycle . The threshold value of K' – below which fatigue crack growth is not observed – is $4.6 \text{ ksi-in}^{1/2}$.

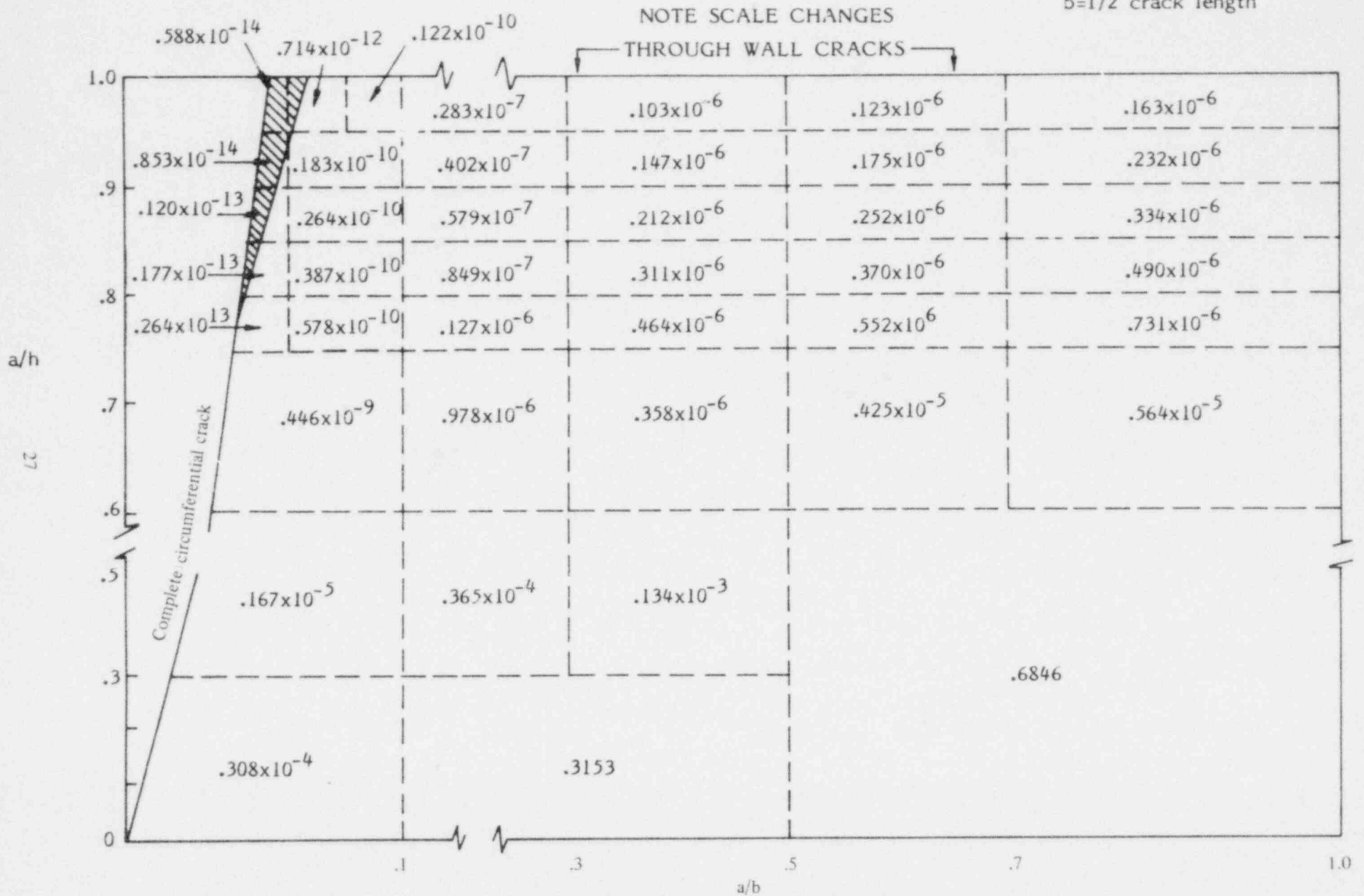
The semi-elliptical shape of the fatigue crack is assumed to be maintained during crack growth. The size and aspect ratios of the cracks are governed by weighted stress intensity solutions, \bar{K}_a and \bar{K}_b , for the depth and length, respectively. The stress intensities are calculated employing an "energy" weighted average which uses the stress intensity solution along the crack front (i.e., $K^2(\phi)$ is proportional to the energy release rate) for a given infinitesimal area of crack extension, $dA(\phi)$. For a given transient,



Calculation procedure

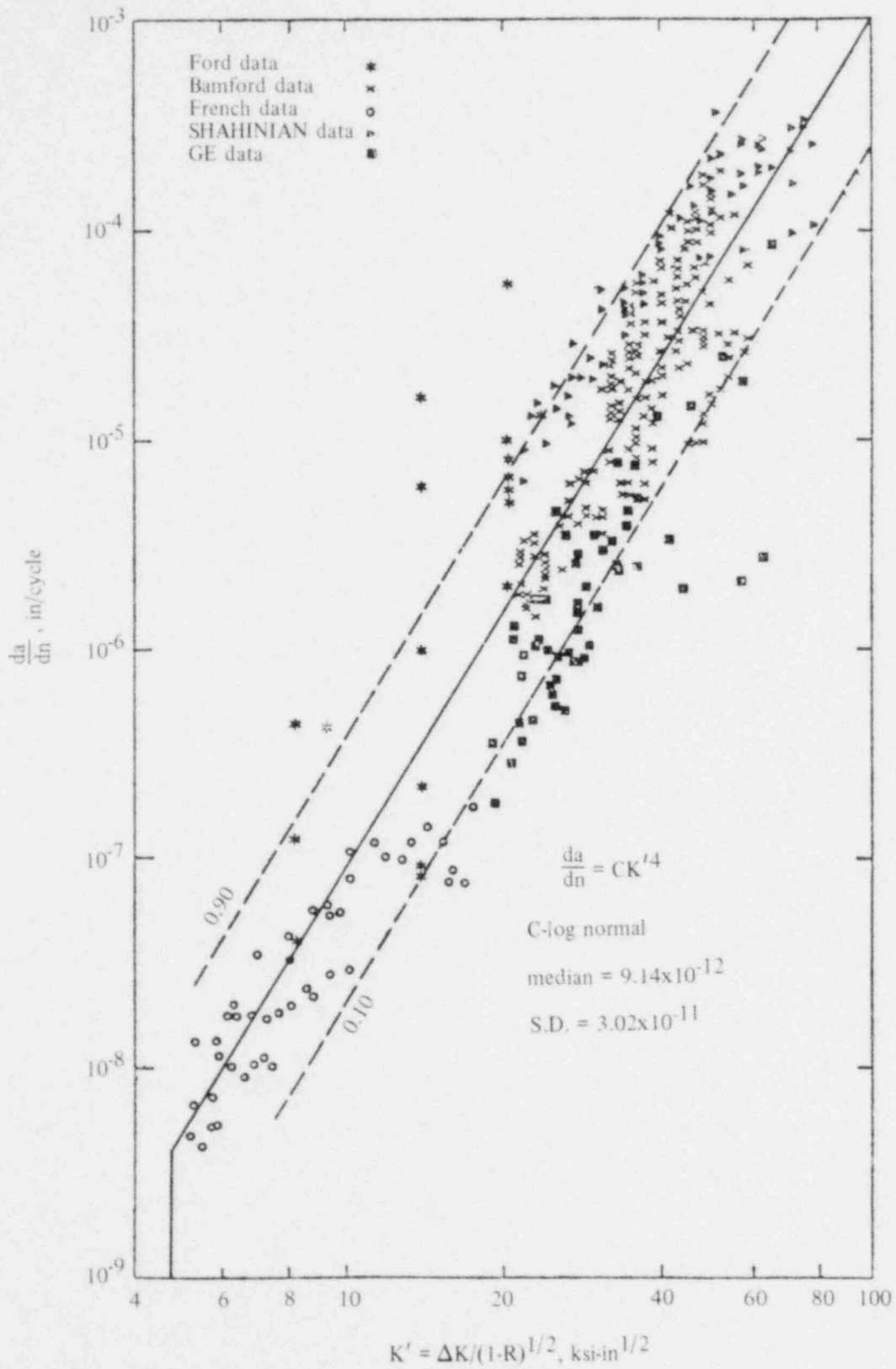
Figure A.1-1

$R_{in} = 15$ inches $h = 2.5$ inches $a =$ crack depth
 $b = 1/2$ crack length



The probability of a crack existing in various regions of the size distribution state space assuming that a crack exists

Figure A.1-2



Fatigue growth data for austenitic piping steels with associated Paris crack growth equation

Figure A.1-3

\bar{K}_a and \bar{K}_b are evaluated as functions of time, and the maximum ΔK is used in the fatigue growth evaluation. To reduce the computation, the stress intensity solution for the given transients will be evaluated, made non-dimensional with respect to flaw size, aspect ratio and stress level, and stored for subsequent use.

The weighted stress intensity solutions are used in the fatigue growth model to extend the cracks, thereby changing the crack size distribution as a function of time. Figure A.1-4 shows two alternative presentations of crack size state space, demonstrating the crack growth path leading to a leak and eventually a LOCA, for a given loading history. The leak rate calculations and the probability of leak detection impact on the failure criteria and probability of failure, respectively. The probability of a LOCA or leak is evaluated as a function of time as shown in Figures A.1-5 and A.1-6. Seismic events can be simulated as a particular transient at a given time to assess their influence on the probability of a leak or LOCA at that time. The single earthquake can be prescribed as in the example in Figures A.1-5 and A.1-6, or drawn from an appropriate distribution.

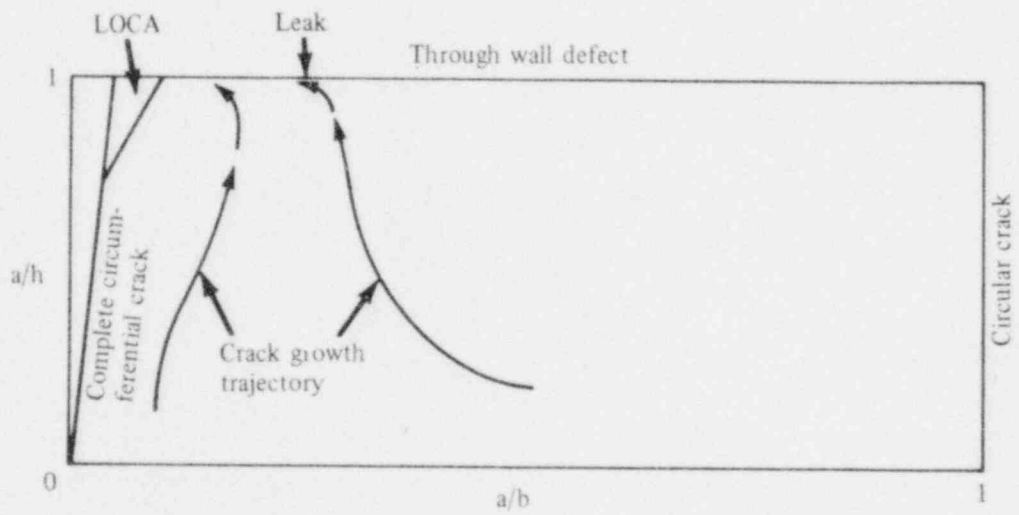
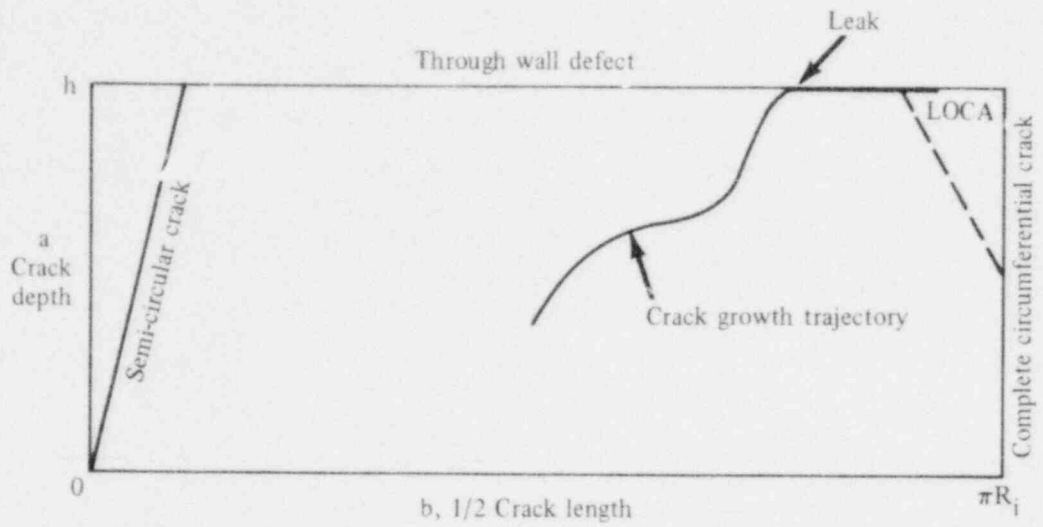
It should be noted that these results were obtained from preliminary information available at the time the calculation was performed. For instance, only one transient (plant heat-up and cool-down at six-month intervals) was considered. In addition, stress histories were estimated from old calculations. Therefore, the calculations presented herein serve to demonstrate the analytical procedure and should not be interpreted for any other purposes.

A.2 Reactor Coolant Loop Model

The model described below 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 (RPV), four steam generators (SG), four reactor coolant pumps (RCP), and one pressurizer. Each loop includes three sections of pipe. Coolant flows from the RPV to the SG through the hot leg, then through the crossover leg to the RCP, and finally through the cold leg back to the RPV. The surge line pipe connects the pressurizer to the system as a branch from one of the hot legs.

The model has 319 nodes and 1,561 degrees of freedom. Node numbers 1-59 are in loop one, 101-159 in loop two, 201-259 in loop three, and 301-359 in loop four, while nodes 360-416 are in the surge line and pressurizer. Other nodes are used for the RPV and its supports.

All element numbers in the model are shown in rectangular boxes. Element numbers preceded by the letter "B" indicate beam elements. The letter "T" indicates truss elements. A number with no letter indicates a pipe element. In most cases, there is an increment of 50 for pipe elements in the four loops of the model. Pipe elements 1-50 are in loop one, 51-100 in loop two, 101-150 in loop three, 151-200 in loop four, and 201-240 are in the surge line and pressurizer. Other members are used for the RPV and its supports.



Alternative presentations of state space for fatigue crack growth in reactor piping

Figure A.1-4

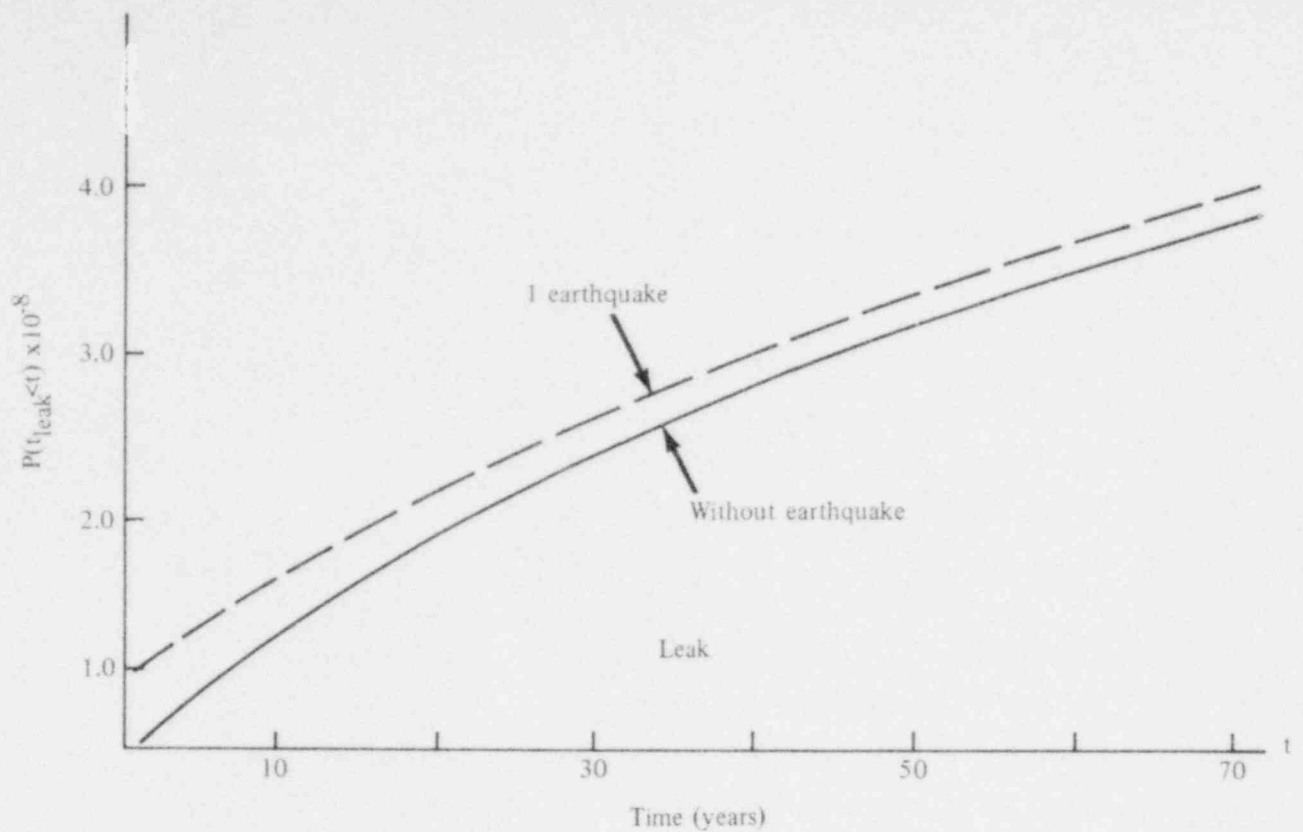
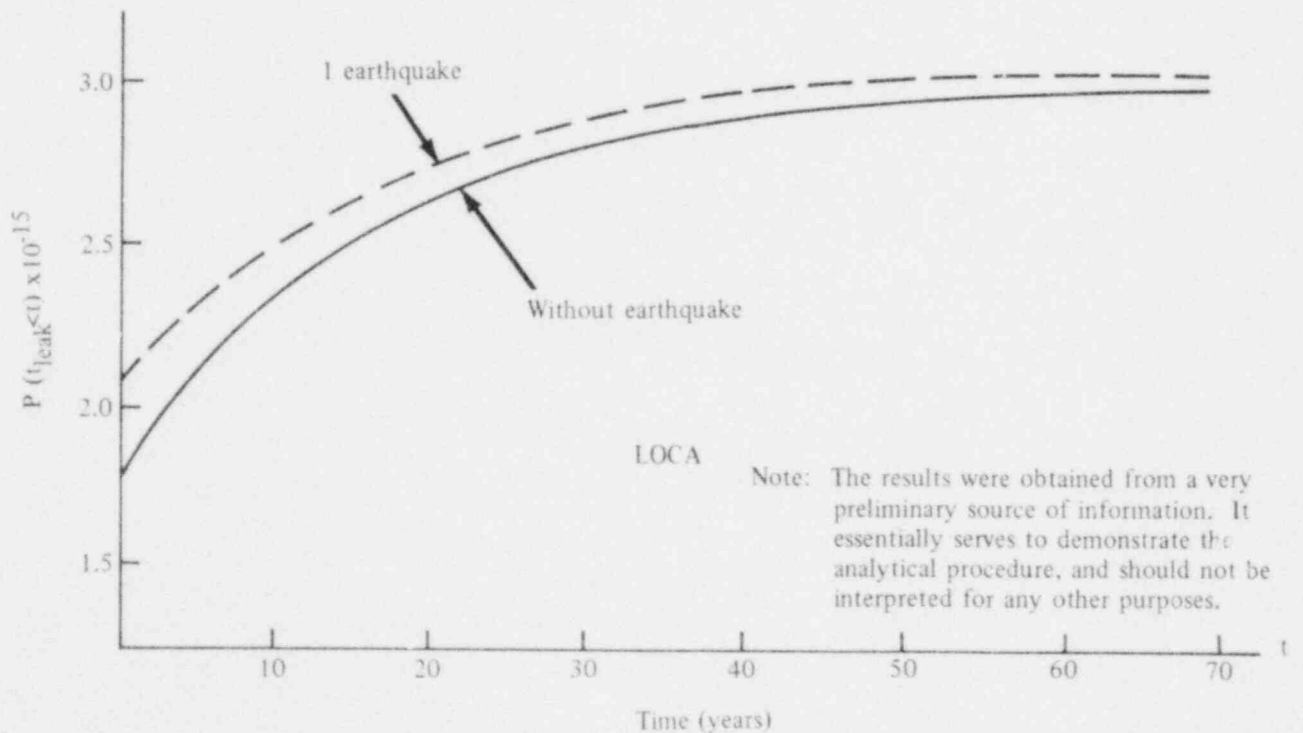


Figure A.1-5 Probability of leak as function of time with and without earthquake
(See Figure A.1-6 for transient and earthquake loading conditions)



Transient: Heatup - Cooldown - At 6 months intervals

$$\sigma_{up} = 28 \text{ ksi} \quad \sigma_{low} = 4 \text{ ksi} @ 1 \text{ cycle}/6 \text{ months}$$

Earthquake $\Delta\sigma = 8 \text{ ksi}$ (peak to peak) 50 cycles

Probability of LOCA as function of time with and without earthquake

Figure A.1-6

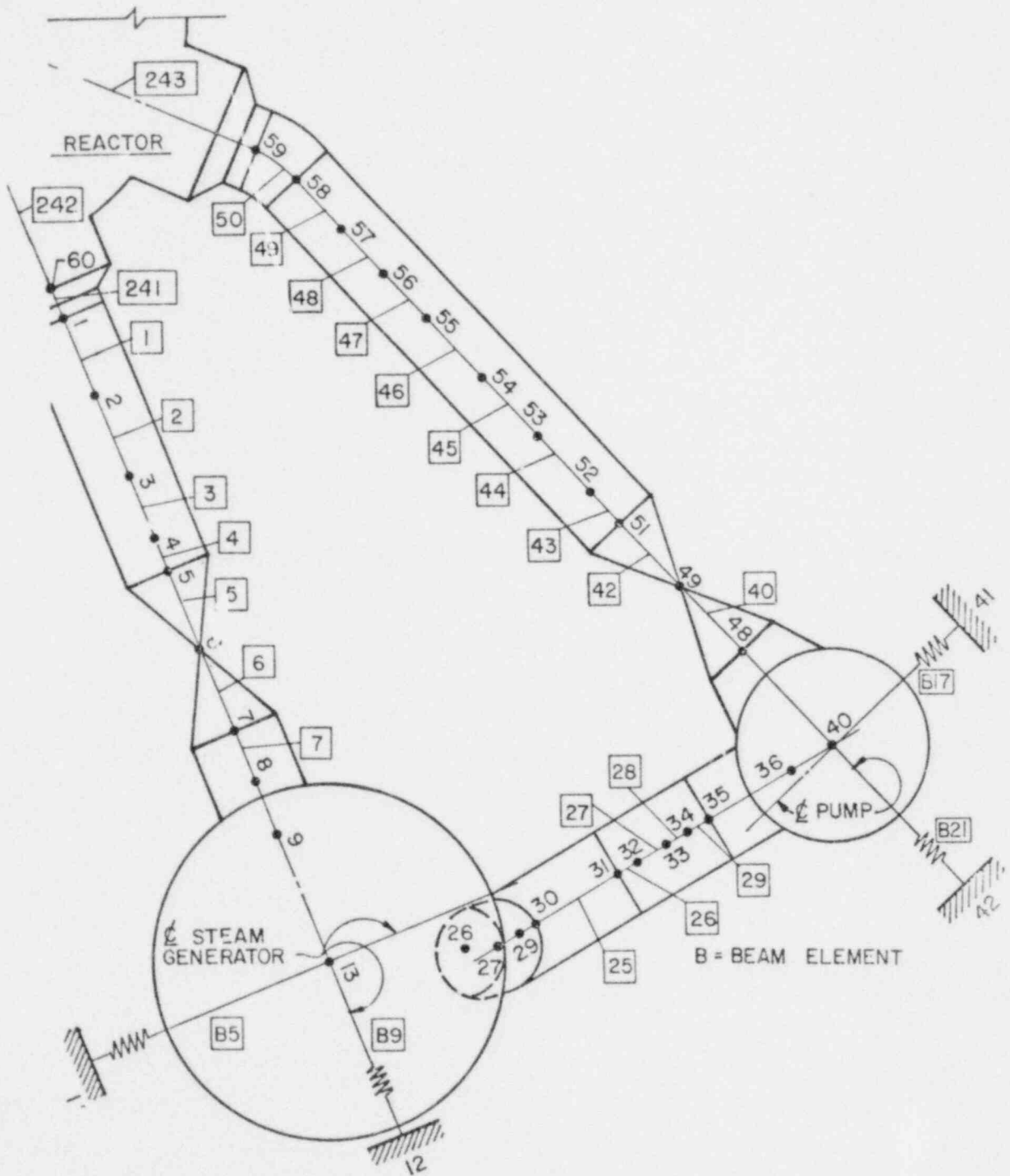
The model contains all main reactor coolant loop piping and the pressurizer surge line. The surge line connects the pressurizer to the system and is attached to a nozzle in the hot leg of loop number four. None of the other 84 branch lines are included in the model. However, nodes are provided in the model at each of the branch line nozzle attachment points for all lines one inch or greater in diameter.

Figure A.2-1 provides a typical plan view of the first 59 nodes of the model, which represent Loop No. 1. The following assumptions and limitations have been used in the development of the model:

1. Stiffness calculations for component and pipe supports consider the flexibility of the supports, including special auxiliary steel members. Primary structural concrete and steel members are assumed to be rigid, however, compared to support flexibility.
2. Local flexibilities of the components at their nozzles and support legs are not considered in the model, unless they were represented in the original component models.
3. Main loop pipe break restraints are not included in the model because the gaps between the restraints and the crossover leg are assumed to be larger than movement occurring under seismic excitation.
4. Stress levels in all component supports are assumed to remain within the elastic limits of the steel materials. This permits the stiffness properties of an entire support system to be represented by a few SAP IV beam and truss elements with equivalent stiffness properties.
5. Because the pressurizer has the same volume as the Westinghouse 1800 ft³ pressurizer at the Byron Station, the pressurizer model has been taken from information contained in the Westinghouse Electric Corporation letter for the Byron Station, dated November 18, 1976.
6. Coolant in the reactor coolant system is assumed to be at an average temperature of 585°F and at a pressure of 2235 psi. Therefore, a coolant density of 45 pounds per cubic foot is used.
7. It is assumed that the model will be input to a version of the SAP IV computer program which contains an internal band-width minimizer. Therefore, for the user's convenience, nodes in the model have been numbered consecutively around each piping loop.

A.3 Thermal Transient Stresses in Surge Line Nozzles

In the analysis of the nozzle at the surge line connection to the hot leg, the three-dimensional geometry has been approximated as a two-dimensional, axisymmetric one, where the hot leg pipe wall is taken as a flat disk. For comparison purposes, a finite element mesh has been generated for a straight pipe connection, i.e., a nozzle with no reinforcement or thermal shield. The thermal boundary conditions have been set as a perfectly insulated boundary at the outer pipe wall, and as a force-convection boundary condition at the inner wall, with a time-dependent heat transfer coefficient based on the flow-rate history in the pipe under consideration. The nozzle temperature time history for the reactor trip, from the Westinghouse design transients, has been obtained using TACO, a transient, nonlinear, finite element, conduction code developed at LLNL. The temperature time history was then used to obtain the thermal



Reactor coolant loop model – Loop 1

Figure A.2-1

stress time history for the connection by utilizing NIKE2D, an implicit, finite-deformation, finite element code for stress analysis, also developed at LLNL. Preliminary results indicate that, for this simplified geometry, significant thermal stresses are generated during the reactor trip, with typical maximum values for the r-z principal stress and loop stress on the order of 10 ksi.

A finite element mesh (shown by Figure A.3-1) was generated for the actual geometry at the surge line - hot leg nozzle, including the thermal shield. SLIC, an interactive mesh generator, was used for this task, and the thermal stress history for the reactor trip was also obtained for this connection using TACO and NIKE2D. A preliminary comparison of the results for the straight pipe and "real" connection indicates that, while the thermal shield affects the details of stress distribution, particularly in the branch pipe behind the shield, the magnitude and location of the maximum thermal stresses are approximately the same for the two connection geometries. The temperature and stress distributions, after 80 seconds in the reactor trip, are illustrated by the contour plots, Figures A.3-2 and A.3-3.

A.4 Seismic Input

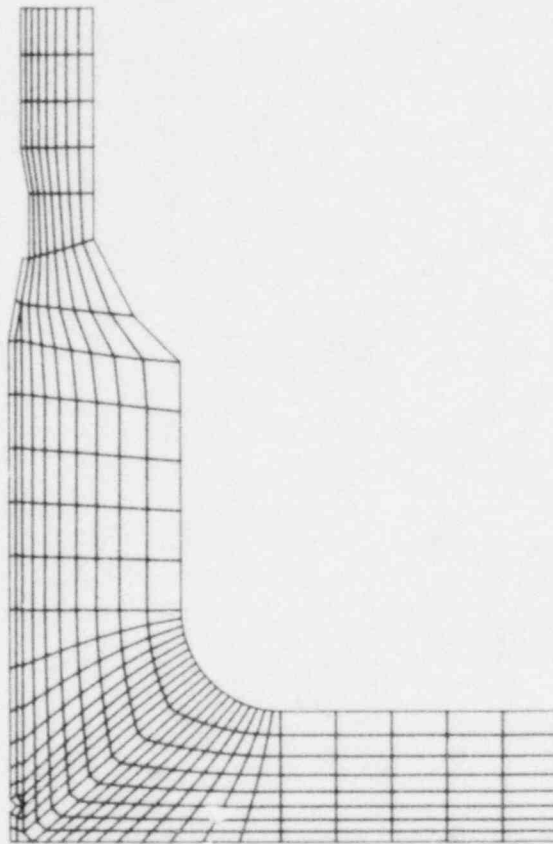
Simulated spectra representative of site-specific events at the Zion site are generated and based on the following assumptions and limitations:

1. They are based on the Gupta-Nuttli intensity attenuation.
2. The damping ratio is five per cent of critical damping.
3. The coefficients of correlation between frequencies needed in the simulation might be improved upon through additional analysis.
4. The sample provided is limited to 40 spectra per PGA band, even though for small PGA a much larger number of spectra were simulated. They were stored in sequential order of simulation and no sorting scheme was applied.
5. The duration was simulated independently of PGA, using the Trifunac and Brady equation with a constant sigma of 10.67 seconds. The large sigma and independence from PGA introduces great variations in the duration. A few acceleration spectra at SSE level (0.17g) are shown on Figure A.4-1.

A.5 Indirectly Seismic Induced Large LOCA

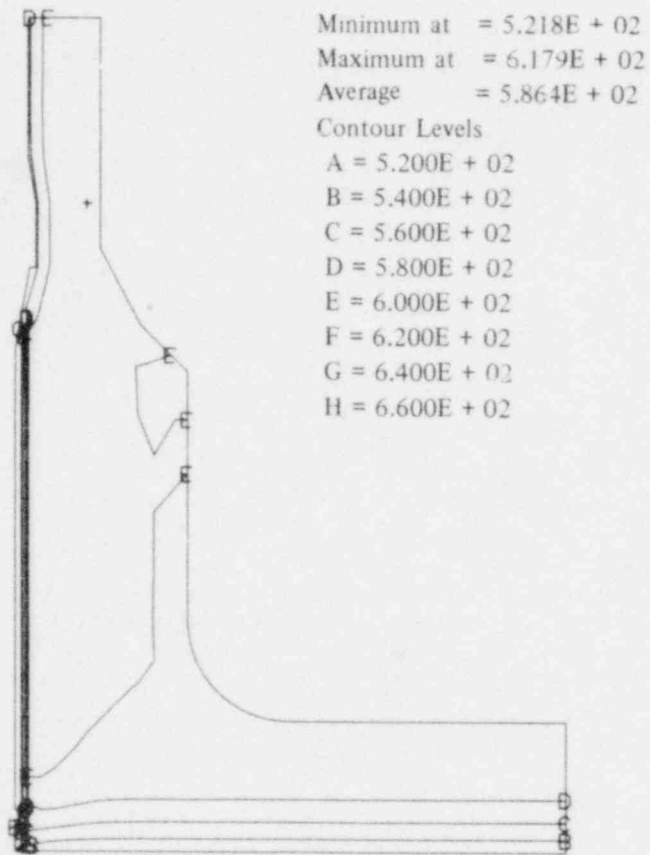
1. Introduction

The purpose of this Project I Subtask is to estimate the probability of an indirect LOCA. The procedure is to determine whether electrical failures, explosions, missiles, or fires can induce a LOCA. If it is possible, then a search for the sources of these four events is initiated. Finally, the mechanics of how the earthquake could cause the source are analyzed.



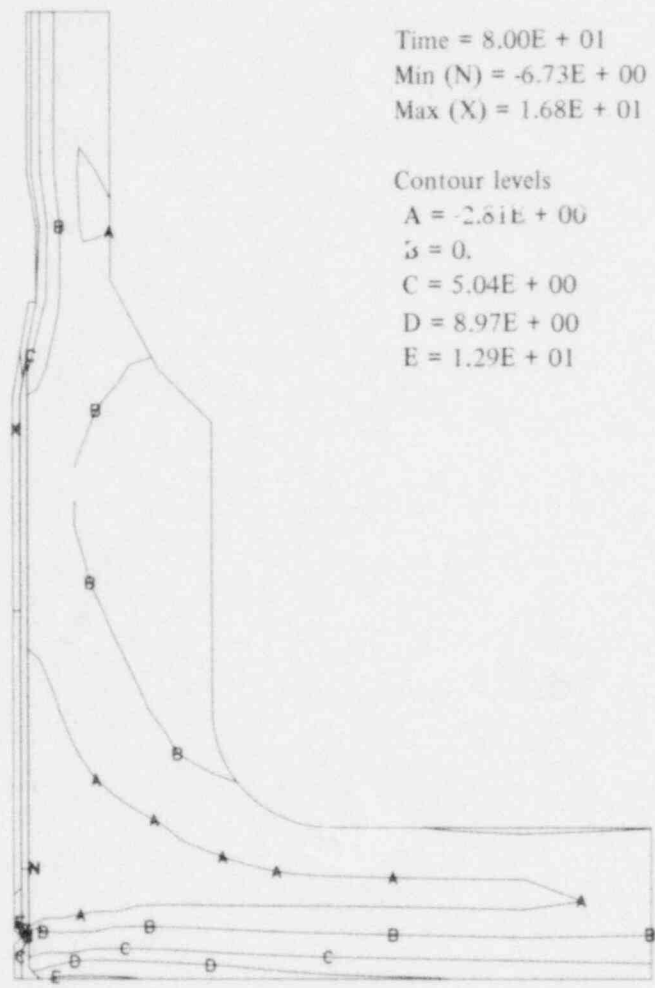
Reactor trip – hot leg connection mesh plot

Figure A.3-1



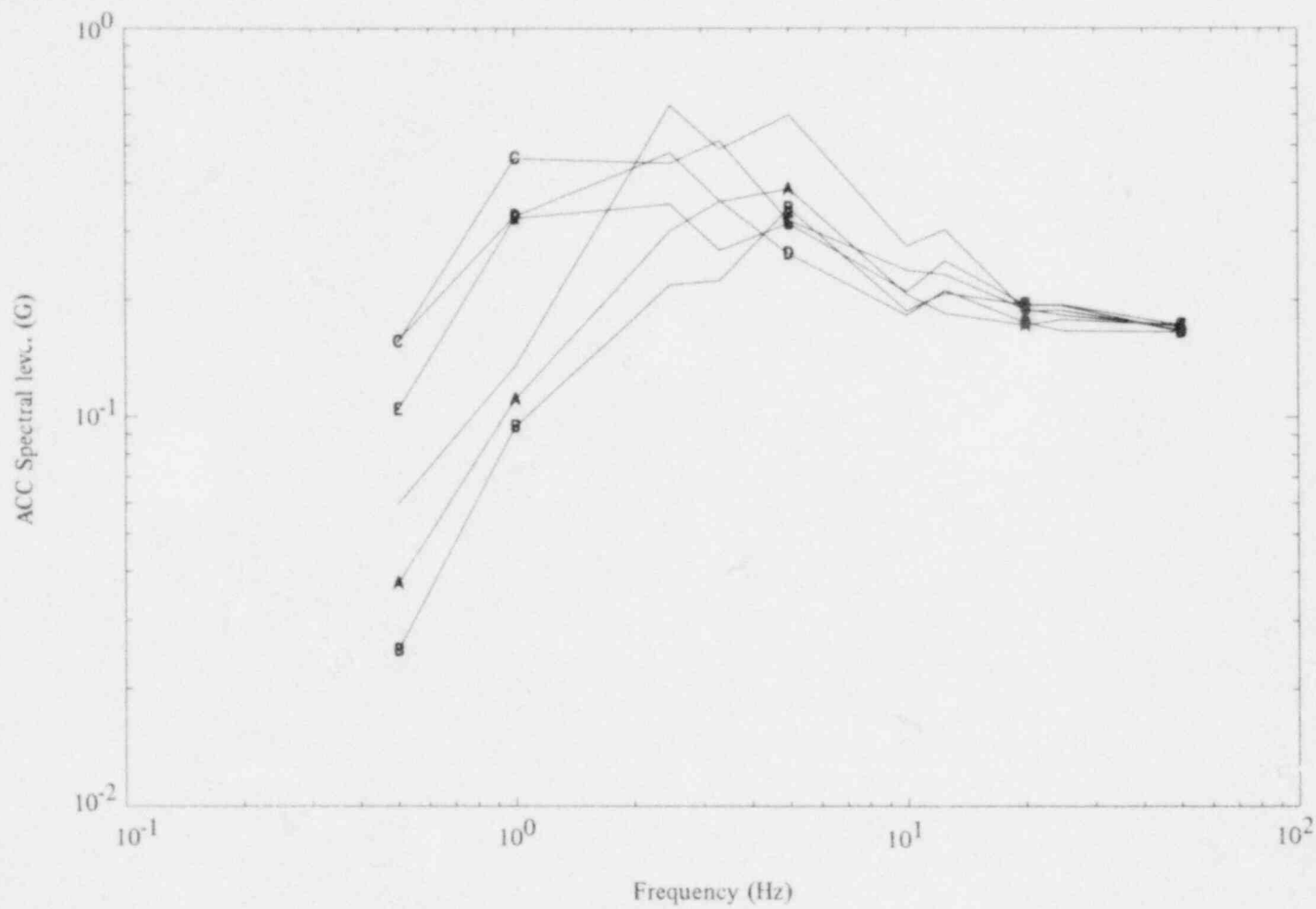
Reactor trip – hot leg connection
Isoplot at time $8.0000E + 01$

Figure A.3-2



Thermal stresses for reactor trip – hot leg connection
 contours of Hoop stress

Figure A.3-3



Acceleration spectra at SSE for 5% damping

Figure A.4-1

A LOCA is defined as any of the following ruptures:

- a. Primary piping including pipes from the emergency core cooling and residual heat removal headers, the pressurizer spray and surge line, and the bypass pipe between RC stop valves.
- b. Valve rupture of the primary loop isolation valves or the check valves on the emergency core cooling or residual heat removal injection line, located on the boundary of the primary system.
- c. Steam generator tube.
- d. Reactor coolant pumps.
- e. Reactor vessel.
- f. Pressurizer, including the power-operated relief valves and the safety relief valves.

After a preliminary analysis, it was concluded that there are no sources for a fire hot enough to melt the primary piping inside the containment. It is believed, however, that a fire can cause a secondary system outside of the containment to fail and that this can lead to a primary system overpressure. Such events or sequences have been previously defined in the SSMRP transient event trees. However, for this study, it will be necessary to define the initiator in terms of seismically-induced electrical failures, explosions, missiles or fires.

Electrical failures within the primary system can induce a LOCA directly through spurious signals to open either a PORV or one of the motor-operated valves at the suction head of the RHR pump from the hot leg of loop A. However, such an event can be considered a transient such as that mentioned in the discussion on fires above. As a result of these preliminary findings, it was decided to categorize the secondary LOCA's into missiles, explosions, and transients, where the transients can be induced by fire or electrical failures.

2. Missiles

Potential missile sources inside the containment that could strike the six components, causing a LOCA are:

- a. The overhead crane (six to 11 feet above the steam generators).
- b. The control rod drive mechanism.
- c. The flywheel of reactor coolant pumps (approximately five feet from steam generator).
- d. A valve motor operator.
- e. Secondary pipe whip.
- f. Jet forces from a high pressure steam or feedwater line.
- g. Containment dome rupture.

3. Seismically Induced Transients

The primary coolant system was analyzed for seismically-induced electrical failures which can cause a LOCA. The only potential event found is a seismically-induced, spurious electrical signal which can inadvertently open a power-operated relief valve (PORV). If the valve does not close or is not blocked, the result is a LOCA. There are also several events which can be seismically-induced outside the primary system and affect the PORV's. It has been concluded that all such events can be analyzed simultaneously as transient events.

A transient is defined as an event requiring plant shutdown. Transients can be categorized into three groups:

- a. Loss of flow of reactor coolant.
- b. Loss of heat sink.
- c. Reactivity related transients.

These groups can be further subdivided. The Electric Power Research Institute (EPRI) considered 41 pressurized water reactor transients in their report titled, "ATWS: A Reappraisal, Part II, Frequency of Anticipated Transients," EPRI NP801, dated July 1978. Time did not permit an analysis of whether or not and how seismically-induced electrical failures, missiles, explosions, or fires could initiate each potential transient. If it is assumed that an earthquake will initiate a requirement to shutdown, we can investigate the potential seismically-induced electrical failures and fires which occur in the transient mitigating systems.

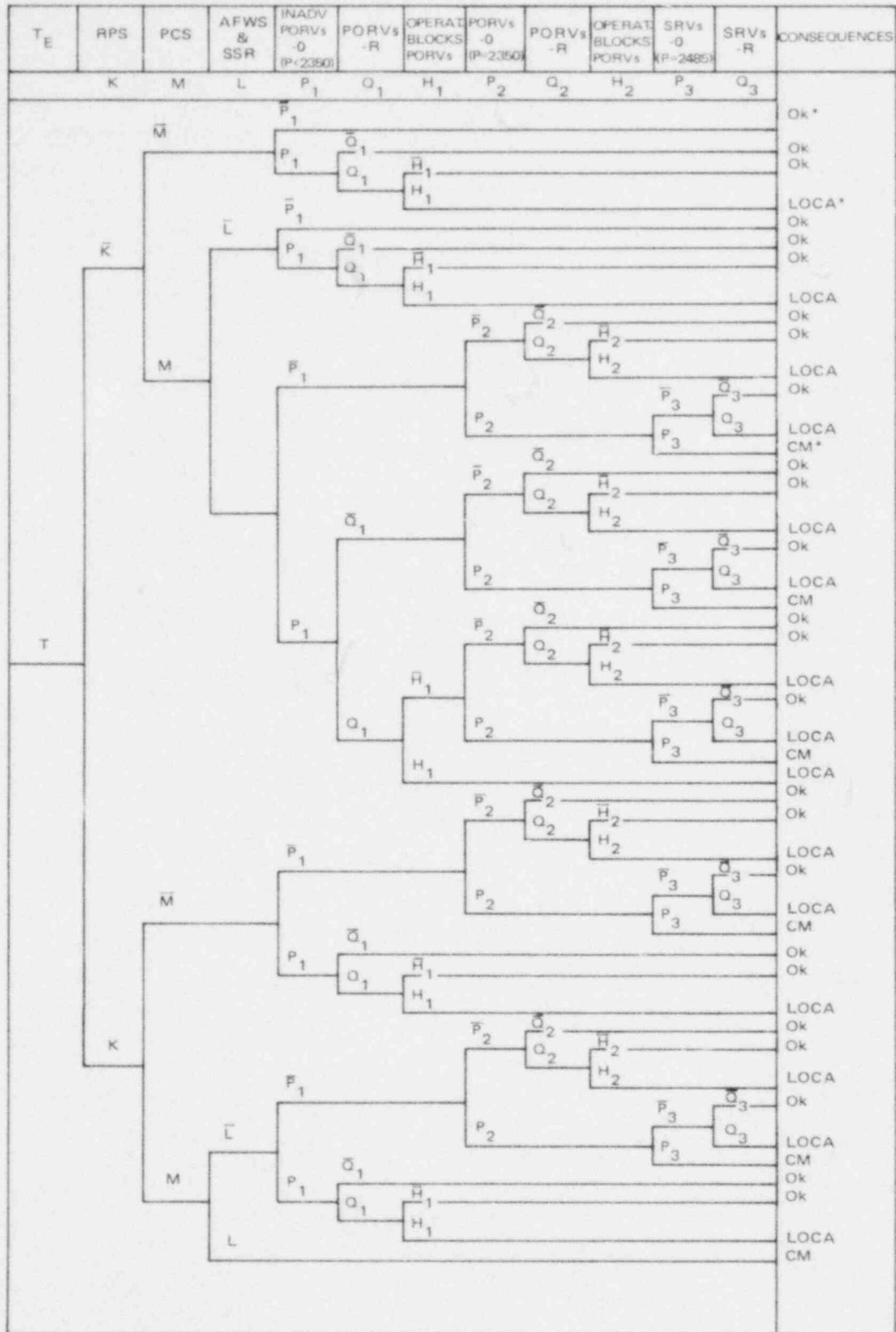
The LOCA of concern is an open PORV or a safety relief valve that will not close. An event tree has been developed to determine the sequence of events which can result in such a LOCA. This event tree is shown in Figure A.5-1. The first column in the event tree, T_E , represents the seismically-induced event requiring immediate plant shutdown. It has been assumed that an earthquake sufficient to cause a spurious PORV opening will also initiate a requirement to shutdown. This sequence is the first sequence on the event tree that leads to a LOCA. In this case, there has been a successful reactor trip, (\bar{K}); successful decay heat removal, (\bar{M}); and one or more PORV's have been inadvertently opened, (P_1), by a seismically-induced spurious signal; the PORV's fail to reclose automatically on low pressure, (Q_1); and the operator does not, or cannot, close the PORV block valves, (H_1).

4. Explosions

- a. Explosions Inside Containment

Three sources of explosion inside the containment have been established. They are as follows:

- (1) Hydrogen buildup.
- (2) Pressurizer relief tank (nitrogen) - (P=100 psig, T=340°F, Vol.=13,500 gal.)
- (3) Accumulator tank (water and boric acid) - (P=700 psig, T=300°F, Vol.=10,000 gal., mix=2,500 ppm).



Ok = No potential release of radioactivity or LOCA
 LOCA = Loss of coolant accident
 CM = Core melt with overpressure causing LOCA or vessel rupture

Event Tree for Indirect LOCA Induced by Electrical Failures

Figure A.5-1

b. Explosions Outside of Containment

The explosion sources identified outside of containment are as follows:

- (1) The 150,000-gallon main fuel oil tank (192 feet from the containment wall).
- (2) The 400,000-gallon refueling water storage tank (five feet from the containment wall) (P=static pressure plus sloshing, T=135°F).
- (3) The compressor room (four feet from the containment wall).
- (4) The boron injection tank (two feet from the containment wall (P=2,735 psig, T=300°F, Vol.=900 gal., mix=12 wt. boron).
- (5) Hydrogen, nitrogen and liquid nitrogen storage (156 feet from containment wall).
- (6) The 50,000 gallon diesel fuel oil tanks (26 feet from the containment wall and approximately seven feet from feedwater piping).
- (7) The boiler room (115 feet from the containment wall).

5. Summary

This analysis has attempted to determine all potential seismically-induced fires, explosions, missiles, and electrical failures which can induce a LOCA. The systematic approach taken of locating sources for these events, with respect to the location of the primary coolant system, has led to several potential scenarios. Some are much more likely to occur than others. The next step in this analysis will be ordering these scenarios probabilistically.

A.6 Probability Estimation of LOCA in Event of Earthquake

This section explains how to estimate the probability of a LOCA in the event of an earthquake, and how to construct a confidence interval based on the estimate. The estimate is needed for the decoupling decision which is based on the probability of a LOCA and an earthquake. If the estimated probability of a LOCA and an earthquake is sufficiently small, and if we are confident in the estimated probability, then decouple.

The estimation is illustrated for the event pipe fracture. Pipe fracture is a subset of LOCA composed of two events so estimation of pipe fracture shows the procedure without complication.

The following definition of events is used:

- PLOCA = LOCA which is a reactor cooling loop (RCL) pipe fracture,
- PDLOCA = directly induced RCL pipe fracture,
- PILOCA = indirectly induced RCL pipe fracture,
- EQ = earthquake of detectable magnitude,
- DLOCA = directly induced LOCA,
- $\overline{\text{EVENT}}$ = complement of EVENT, and
- \cap = Boolean "and" operator.

The object is to estimate the probability of a pipe fracture given an earthquake and construct a confidence interval on it. See Figure A.6-1 for a Venn diagram showing all events.

The probability $P[\text{PLOCA}/\text{EQ}]$ is the sum of probabilities of the mutually exclusive events,

$$P[\text{PLOCA}/\text{EQ}] = P[\text{PDLOCA}/\text{EQ}] + P[\text{PILOCA} \cap \overline{\text{DLOCA}}/\text{EQ}].$$

Note the definition of $\text{PILOCA} \cap \overline{\text{DLOCA}}$, indirectly induced pipe fracture and no direct LOCA, restricts the event to initiating events classed as transients in Section A.5.

This notation will be used for probabilities:

$$\begin{aligned} P &= P[\text{PLOCA}/\text{EQ}], \\ P_1 &= P[\text{PDLOCA}/\text{EQ}], \\ P_2 &= P[\text{PILOCA} \cap \overline{\text{DLOCA}}/\text{EQ}], \\ 1-\alpha_i &= \text{confidence in estimate of } p_i, i = 1, 2, \\ 1-\alpha &= \text{confidence in estimate of } p, \text{ and} \\ \hat{p} &= \text{an estimate of any probability.} \end{aligned}$$

The data in Section A.1 of this appendix provides \hat{p}_1 and $1-\alpha_1$, and Section A.5 data provides \hat{p}_2 and, eventually, $1-\alpha_2$. The objective is to produce \hat{p} and confidence intervals on p for arbitrary $1-\alpha$. The illustration assumes the normal approximation to the binomial is adequate.

Since PDLOCA and $\text{PILOCA} \cap \overline{\text{DLOCA}}$ are mutually exclusive events,

$$\hat{p} = \hat{p}_1 + \hat{p}_2.$$

That is, the estimate of $P[\text{PLOCA}/\text{EQ}]$ is the sum of the two estimates from Section A.1 and A.5.

Given confidence intervals on p_1 and p_2 , variances are estimated by solving the confidence interval formula

$$\hat{p}_i \pm Z_{\alpha_i/2} \sqrt{\text{var } p_i}$$

for $\text{var } \hat{p}_i$, where $Z_{\alpha_i/2}$ is the lower percentile of the standard normal random variable. Let U_i denote the upper limit on the confidence interval of width $1-\alpha_i$. The solution is

$$\text{var } \hat{p}_i = \left(\frac{\hat{p}_i - U_i}{Z_{\alpha_i/2}} \right)^2$$

Assuming \hat{p}_1 and \hat{p}_2 are independent,

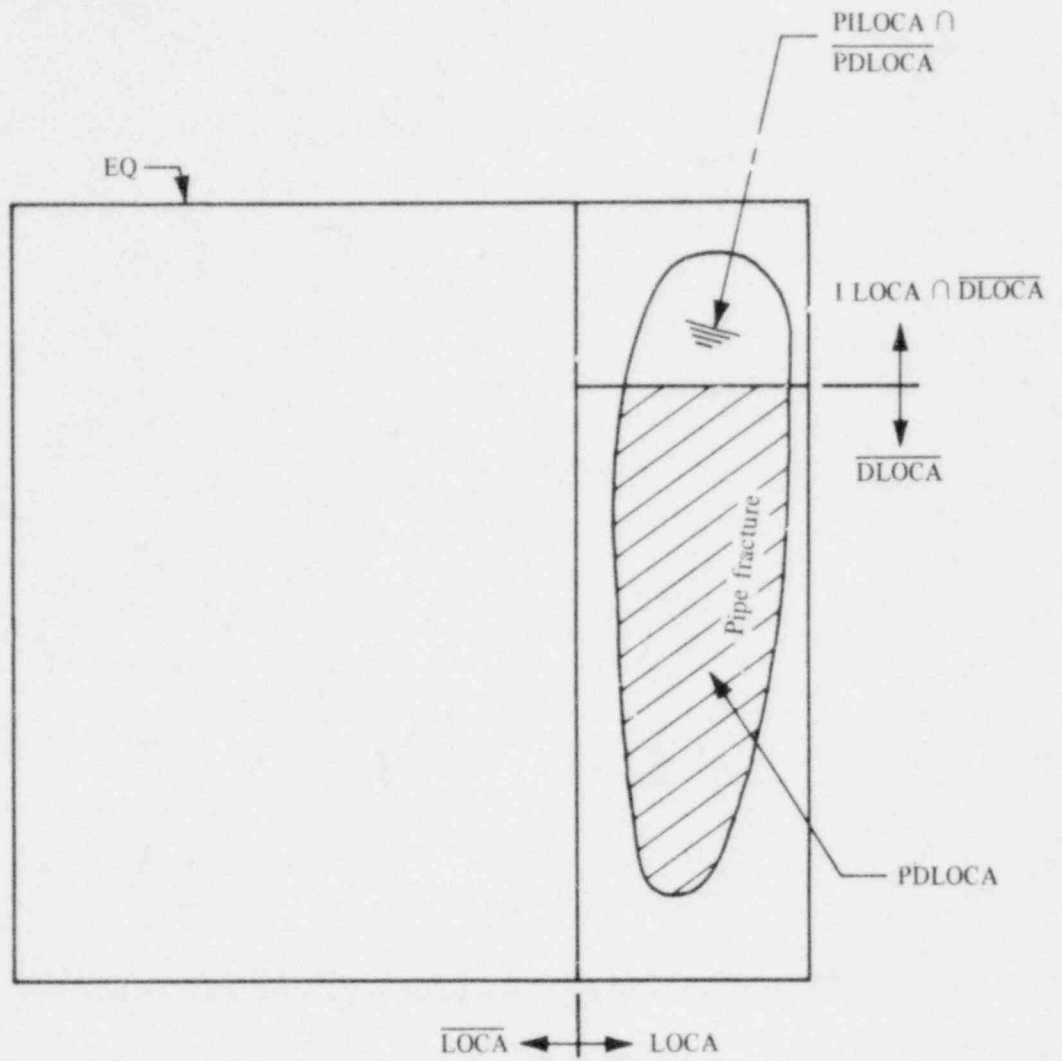
$$\text{var } \hat{p} = \text{var } \hat{p}_1 + \text{var } \hat{p}_2,$$

so an estimate of the confidence interval on p is

$$\hat{p} \pm Z_{\alpha/2} \sqrt{\text{var } \hat{p}}$$

which is

$$\hat{p}_1 + \hat{p}_2 \pm Z_{\alpha/2} \sqrt{\left(\frac{\hat{p}_1 - U_1}{Z_{\alpha_1/2}} \right)^2 + \left(\frac{\hat{p}_2 - U_2}{Z_{\alpha_2/2}} \right)^2}$$



Venn diagram of events contributing to $P[\text{LOCA}/\text{EQ}]$

Figure A.6-1

LCP (Phase I)

For example, assume $\alpha_1 = \alpha_2 = \alpha = .05$, $\hat{p}_1 = 10^{-7}$, $\hat{p}_2 = 10^{-11}$, $U_1 = 10^{-6}$, and $U_2 = 10^{-8}$. Then $\hat{p} = 10^{-7}$. The value of $Z_{\alpha/2} = -1.96$, $\text{var } \hat{p}_1 \cong 2.1 \times 10^{-11}$, $\text{var } \hat{p}_2 \cong 2.6 \times 10^{-15}$, and the upper confidence limit on \hat{p} is 9.0×10^{-6} . If $U_2 = 10^{-4}$ and all other inputs remain the same, then the upper confidence limit is .001.

Estimation of $P[\text{LOCA}/\text{EQ}]$, including events which are not pipe fractures, will be done later. Construction of a confidence interval $P[\text{LOCA} \cap \text{EQ}]$ will also be done later.

APPENDIX B LOAD COMBINATION METHODOLOGY

B.1 Introduction

The objectives of the load combination methodology development effort are to establish criteria for selecting appropriate loading combinations, and to develop methods for combining dynamic responses. The specific approach to development of load combination methodology makes use of a load factor format expressed by

$$\phi R = \gamma_1 C_1 L_1 + \gamma_2 C_2 L_2 + \dots + \gamma_n C_n L_n$$

where L_1, L_2, \dots, L_n are the loads to be combined; C_1, C_2, \dots, C_n , are a set of influence coefficients that transform each load into a particular structural response; $\gamma_1, \gamma_2, \dots, \gamma_n$ are a set of load factors which correspond to a selected or target limit state probability; R is the resistance of the structure; and ϕ a factor associated with resistance that reflects the dispersion of the material properties. The goal of this approach is the determination of an "optimal" set of load factors corresponding to a given target limit state probability, P_T .

As mentioned in the first interim report, (UCID-18149, January 31, 1980), the proposed methodology combines two steps; a "design" step followed by an "evaluation" step. For a given design format, these two steps are iterated to define the optimal set of load and resistance factors which constitute a design rule. Optimization is determined with respect to an appropriate measure of closeness of the evaluated component linear state probability, P , based upon a trial set of load factors, to the target component limit state probability, P_T . The measure of closeness is evaluated over all possible design situations such as component type, reactor type, geographical location, etc. Thus, the load combination developed for a specific component limit state probability should be applicable to a wide range of design situations.

The procedure outlined above requires that, for each set of load factors selected, the components under all design situations be analyzed and designed to obtain the influence coefficients required by the design format. This calls for a large number of costly structural subsystem and component analyses. In the case of a piping subsystem, for example, several iterations of analyses are required for different arrangements of pipe restraint. An alternative approach, now being pursued, makes use of influence coefficients derived from existing designs. By studying a number of existing designs, the frequency distribution of influence coefficients can be obtained. Representative design situations can be simulated by sampling from these distributions. The design format equation is then expressed in terms of the responses obtained from these statistically derived influence coefficients. The derivation of load factors, using this scheme, is expected to be less expensive and, consequently, more practical.

In the case of a piping subsystem, the procedure is to divide a pipe into a large number of nodes and determine the influence coefficient at each node that corresponds to a particular load. Thus at each node there will be a set of influence coefficients, each corresponding to a particular load in the combina-

tion. The influence coefficients for any particular load constitute a set of random variables from which a histogram may be constructed. The histogram describes the statistical distribution of influence coefficients over the pipe line. Thus, influence coefficients derived from existing designs will reflect good current design practice.

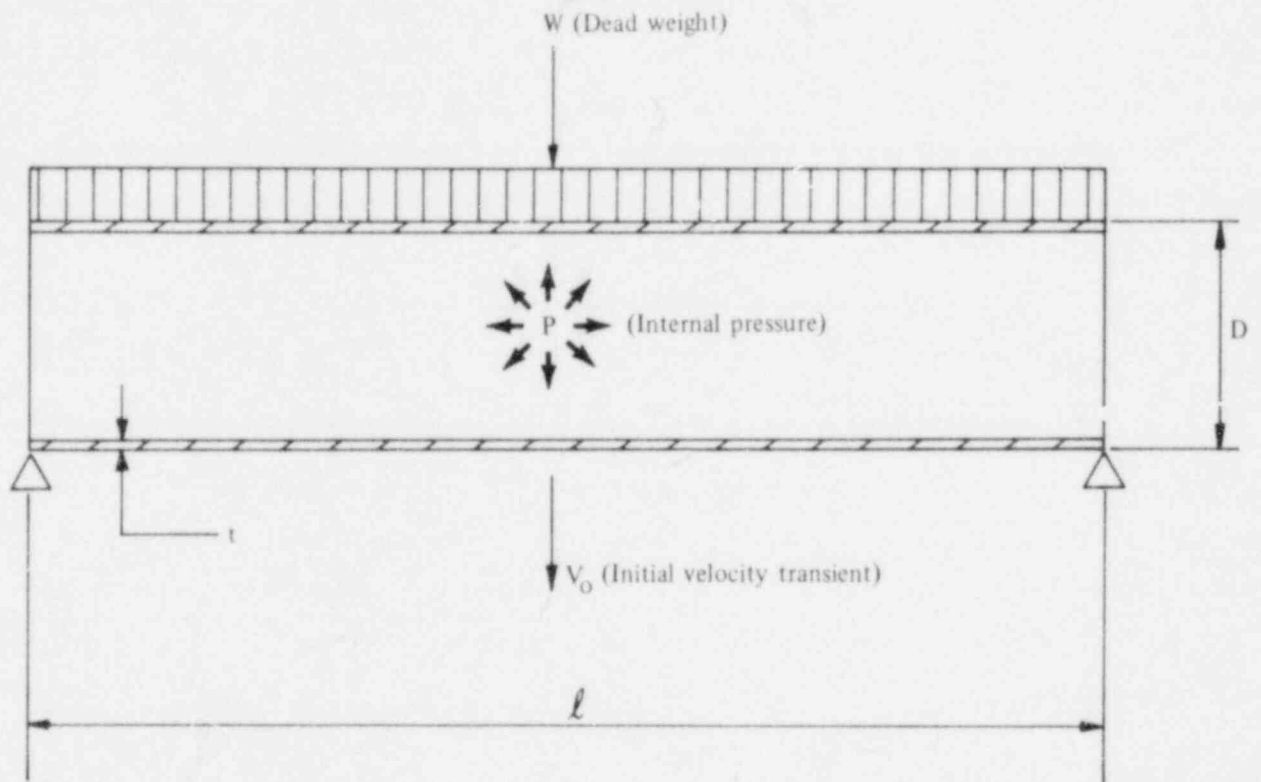
The influence coefficient distributions can now be applied to a design problem. The piping configuration is divided into a large number of nodes, and, at each node, a set of influence coefficients is selected by random sampling from the existing distributions. An initial set of load factors is chosen, and the diameter or pipe thickness at each node is determined so that the design format equation is satisfied, and the code-allowable stresses are not exceeded. We now subject the pipe to reliability analyses to determine the limit state probability of the pipe as it has actually been designed.

Where the loads are dynamic, the sum of the loads in combination will vary in accordance with their magnitude, duration, mean occurrence rate and characteristic time histories. A dynamic load combination tree is constructed and consists of a number of possible loading combinations of varying magnitude and probability of occurrence. For each load combination in the tree, the probability distribution of maximum combined response will be calculated, using the upcrossing rate derived from response time histories of the individual responses. The maximum limit state probability of each node is obtained by convolution of the combined peak response and the resistance distributions of the most critical load combination at the node. The limit state probability of the component is associated with the largest nodal limit state probability. The difference between actual and target limit state probabilities is noted and the entire procedure is repeated with new sets of load factors. These are chosen so that the target limit state probability is approached as closely as possible.

B.2 Derivation of Load Factors – A Simple Example

While our goal is to provide load factors that will eventually be used for design and evaluation of nuclear power plant components, we feel compelled at this time to provide a simple, hypothetical example that will serve to clarify the proposed methodology. This is a "text book" example which uses a highly idealized geometry and a convenient set of hypothetical values to enhance understanding, at the expense of realism. We have chosen, purely for illustrative purposes, a section of simply supported pipe subjected to a simultaneous load combination involving dead weight, internal pressure, and an initial velocity transient. (See Figure B-1.) What we seek to do is find sets of load factors, for the load combination, that correspond to different levels of specified or target limit state probabilities. The target limit state probability is a measure of the level of reliability one wishes to derive into a nuclear power plant component.

The underlying assumption of our approach is that all the parameters are random variables. For some parameters, such as pipe thickness or diameter, the dispersion of their values may be so small that their neglect will mean little loss of accuracy. For others, they may be so large that special probabilistic techniques may be needed to deal with them effectively. In our simple problem we will assume that only



Load combination configuration for hypothetical example

Figure B-1

the initial velocity transient, length, and internal pressure are independent random variables; that they are normally distributed and that their standard deviations are small enough so that variance propagates linearly.

The procedure to be followed is essentially that described in Section B.1 of this appendix. The specific steps are the same as those outlined in the presentation to NRC in Bethesda on May 21, 1980.

1. We start with the selection of a design format

$$\phi R = \gamma_1 C_1 v_0 + \gamma_2 C_2 \ell + \gamma_3 C_3 p$$

where

- v_0 = initial velocity transient, in/sec.
- ℓ = pipelength, in.
- p = internal pressure, lb/in²
- C_1 = influence coefficient which transforms the initial velocity into a bending stress, lb/in²/in/sec
- C_2 = influence coefficient which transforms the pipe length into a bending stress, lb/in²/in
- C_3 = influence coefficient which transforms the internal pressure into an axial membrane stress, lb/in²/lb/in².
- R = resistance of the pipe in terms of some design limit stress
- ϕ = resistance factor which reflects the stress categories in the load combination, and
- $\gamma_1, \gamma_2, \gamma_3$ = are the load factors which correspond to v_0, ℓ, p respectively.

2. We now evaluate the influence coefficients which take the form

$$C_1 = \frac{2}{\pi} \left[\frac{E}{Lg} \left(16 \Delta_s + \frac{4D}{t} \Delta_w \right) \right]^{1/2}$$

$$C_2 = \ell \left(\frac{\Delta_s}{D} + \frac{\Delta_w}{4t} \right)$$

$$C_3 = \frac{D}{4t}$$

where

- E = modulus of elasticity of pipe material
- g = acceleration due to gravity
- Δ_s = density of pipe material
- Δ_w = density of fluid.
- t = pipe thickness
- D = pipe diameter

3. For an assumed set of load factors, we now solve the design format equation for the pipe thickness, t , that results in a stress equal to or less than a code allowable limit stress. For this calculation the following values were used.

$$\begin{aligned} E &= 30,000,000 \text{ lb/in}^2 \\ g &= 386 \text{ in/sec}^2 \\ \Delta_s &= 0.3 \text{ lb/in}^3 \\ \Delta_w &= 0.036 \text{ lb/in}^3 \\ R &= 15600 \text{ lb/in}^2 \\ \phi &= 1.5 \text{ for combined primary membrane and bending stress} \\ p &= 2200 \text{ lb/in}^2 \\ V_o &= 12 \text{ in/sec} \end{aligned}$$

$$\gamma_1 = \gamma_2 = \gamma_3 = 1.00 \text{ initially.}$$

4. Once the thickness of the pipe has been determined, we have a design which can be analyzed probabilistically. The limit state probability is expressed by

$$P = 1 - \psi(K)$$

where ψ is the Gaussian function for argument K .

$$K = \frac{\bar{S} - \bar{s}}{[\sigma^2(S) + \sigma^2(s)]^{1/2}}$$

\bar{S} = the mean failure governing strength

\bar{s} = the mean failure governing stress

$\sigma^2(S)$ = the variance of the strength

$\sigma^2(s)$ = the variance of the stress

$$\bar{s} = C_1 \bar{v}_o + C_2 \bar{\ell} + C_3 \bar{p}$$

$$\text{and, } \sigma^2(s) = \left(\frac{\delta s}{\delta v_o}\right)^2 \sigma^2(v_o) + \left(\frac{\delta s}{\delta \ell}\right)^2 \sigma^2(\ell) + \left(\frac{\delta s}{\delta p}\right)^2 \sigma^2(p)$$

The values assumed for determining component limit state probability are as follows:

$$\begin{aligned} \bar{S} &= 30,000 \text{ lb/in}^2 \\ \sigma(S) &= 0.05S = 1500 \text{ lb/in}^2 \\ \sigma(v_o) &= 0.10 v_o = 1.2 \text{ in/sec.} \\ \sigma(\ell) &= 0.01 \ell \\ \sigma(p) &= 285 \text{ lb/in}^2 \end{aligned}$$

We now specify a target limit state probability, P_T , for the pipe component, and note the difference between this probability and the computed limit state probability. Since we want the load factors to represent as broad a class of pipe configurations as possible, we perform the calculation for the large combination of pipe diameters and length displayed in Table B-1. Associated with each unique combination of length and diameter is a pipe thickness that limits the stress to the code allowable for the applied load. A limit state probability is also associated with each combination of length and diameter. From this array we set up an objective function

$$f(\gamma) = \sum_{i=1}^N \sum_{j=1}^N [P_{ij} - P_T]^2$$

which we proceed to minimize by adjusting the load factors using a quasi-Newton algorithm. The load factors which evolve from this process assure that a pipe design, using those factors, will have a limit state probability very close to the target value. Consequently, for each specified target limit state probability, there will be a corresponding set of design load factors. These load factors are summarized in Table B-2 for the entire array of pipe lengths and diameters considered.

To validate the accuracy of the load factors obtained in this example, we display in Tables B-3 through B-5, the actual limit state probabilities and pipe thicknesses that result from using the appropriate set of load factors. We see that, for each combination of pipe length and diameter, the actual limit state probability is close to the specified target. In addition, we see how and to what extent the pipe thicknesses increase as the target limit state decreases.

This methodology can be extrapolated to the design of real components of greater complexity, can incorporate multiple dynamic loads, and can cope with stochastic variables whose distribution is other than normal. Its usefulness lies in the fact that designs may be executed using deterministic methods, in connection with a set of load factors, that assure a desired level of component reliability. In addition, the methodology contains within it the steps to compute component limit state probabilities directly. This can be used as an evaluation tool for existing designs. This was done, in fact, in this simple example to illustrate the actual limit state probabilities of each design. The same can be done for any nuclear power plant component.

POOR ORIGINAL

Table B-1

LIMIT STATE PROBABILITIES CORRESPONDING TO

GAMMA1 = 1.0000 GAMMA2 = 1.0000 GAMMA3 = 1.0000

LENGTH (IN)	PIPE DIAMETER (IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	6.4808E-03	6.7001E-03	6.8608E-03	6.9836E-03	7.0804E-03	7.1587E-03	7.2234E-03	7.2777E-03
240.	3.4988E-03	4.0227E-03	4.4407E-03	4.7802E-03	5.0605E-03	5.2956E-03	5.4952E-03	5.6668E-03
360.	1.1982E-03	1.6550E-03	2.0855E-03	2.4780E-03	2.8315E-03	3.1484E-03	3.4323E-03	3.6872E-03
480.	2.7646E-04	4.7459E-04	7.0925E-04	9.6374E-04	1.2253E-03	1.4853E-03	1.7384E-03	1.9813E-03
600.	6.1998E-05	1.1490E-04	1.9380E-04	2.9770E-04	4.2302E-04	5.8508E-04	7.1915E-04	8.8101E-04

PIPE THICKNESS CORRESPONDING TO

GAMMA1 = 1.0000 GAMMA2 = 1.00 GAMMA3 = 1.0000

LENGTH (IN)	PIPE DIAMETER (IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	3.3987E-01	4.0480E-01	4.6974E-01	5.3469E-01	5.9964E-01	6.6451E-01	7.2957E-01	7.9454E-01
240.	3.8931E-01	4.5323E-01	5.1747E-01	5.8191E-01	6.4647E-01	7.1112E-01	7.7584E-01	8.4060E-01
360.	4.9081E-01	5.4888E-01	6.0927E-01	6.7098E-01	7.3351E-01	7.9659E-01	8.6005E-01	9.2378E-01
480.	6.9500E-01	7.2721E-01	7.7214E-01	8.2359E-01	8.7882E-01	9.3644E-01	9.9587E-01	1.0560E+00
600.	1.1858E+00	1.0919E+00	1.0751E+00	1.0903E+00	1.1218E+00	1.1627E+00	1.2095E+00	1.2603E+00

Table B-2 Load factors corresponding to a range of target limit state probabilities

Target limit state probability	γ_1	γ_2	γ_3
10^{-2}	0	1.044	1.365
10^{-3}	0	1.095	1.493
10^{-4}	1.19	0.866	1.093

POOR ORIGINAL

Table B-3

LOAD FACTORS CORRESPONDING TO 1.0E-02 TARGET LIMIT STATE

GAMMA1 = 0.0000 GAMMA2 = 1.0439 GAMMA3 = 1.3846

LIMIT STATE PROBABILITIES CORRESPONDING TO 1.0E-02 TARGET LIMIT STATE

LENGTH (IN)	PIPE DIAMETER (IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	1.0372E-02	1.0402E-02	1.0424E-02	1.0440E-02	1.0453E-02	1.0463E-02	1.0472E-02	1.0479E-02
240.	9.9133E-03	1.0002E-02	1.0070E-02	1.0123E-02	1.0166E-02	1.0202E-02	1.0231E-02	1.0257E-02
360.	9.5088E-03	9.5809E-03	9.6587E-03	9.7310E-03	9.7955E-03	9.8521E-03	9.9018E-03	9.9455E-03
480.	9.7818E-03	9.5536E-03	9.4861E-03	9.4851E-03	9.5127E-03	9.5521E-03	9.5956E-03	9.6398E-03
600.	1.1888E-02	1.0634E-02	1.0039E-02	9.7399E-03	9.5879E-03	9.5148E-03	9.4852E-03	9.4811E-03

PIPE THICKNESS CORRESPONDING TO 1.0E-02 TARGET LIMIT STATE

LENGTH (IN)	PIPE DIAMETER (IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	3.3295E-01	3.9706E-01	4.6117E-01	5.2530E-01	5.8943E-01	6.5356E-01	7.1770E-01	7.8184E-01
240.	3.7265E-01	4.3609E-01	4.9974E-01	5.6352E-01	6.2738E-01	6.9131E-01	7.5527E-01	8.1927E-01
360.	4.5117E-01	5.1091E-01	5.7209E-01	6.3410E-01	6.9663E-01	7.5952E-01	8.2286E-01	8.8597E-01
480.	5.9788E-01	6.4284E-01	6.9484E-01	7.5060E-01	8.0862E-01	8.6808E-01	9.2853E-01	9.8969E-01
600.	8.9889E-01	8.8529E-01	9.0567E-01	9.4171E-01	9.8627E-01	1.0360E+00	1.0891E+00	1.1445E+00

POOR ORIGINAL

Table B-4

LOAD FACTORS CORRESPONDING TO 1.0E-03 TARGET LIMIT STATE

GAMMA1 = 0.0000 GAMMA2 = 1.0951 GAMMA3 = 1.4927

LIMIT STATE PROBABILITIES CORRESPONDING TO 1.0E-03 TARGET LIMIT STATE

LENGTH(IN)	PIPE DIAMETER(IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	1.0680E-03	1.0739E-03	1.0782E-03	1.0814E-03	1.0840E-03	1.0860E-03	1.0877E-03	1.0891E-03
240.	9.8022E-04	9.9681E-04	1.0097E-03	1.0199E-03	1.0281E-03	1.0350E-03	1.0407E-03	1.0456E-03
360.	9.0629E-04	9.1928E-04	9.3334E-04	9.4653E-04	9.5837E-04	9.6884E-04	9.7809E-04	9.8625E-04
480.	9.5244E-04	9.1327E-04	9.0192E-04	9.0198E-04	9.0099E-04	9.1408E-04	9.2193E-04	9.2987E-04
600.	1.3392E-03	1.1035E-03	9.9726E-04	9.4520E-04	9.1913E-04	9.0667E-04	9.0177E-04	9.0122E-04

PIPE THICKNESS CORRESPONDING TO 1.0E-03 TARGET LIMIT STATE

LENGTH(IN)	PIPE DIAMETER(IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	3.6430E-01	4.3442E-01	5.0456E-01	5.7470E-01	6.4485E-01	7.1501E-01	7.8517E-01	8.5533E-01
240.	4.0819E-01	4.7753E-01	5.4712E-01	6.1687E-01	6.8671E-01	7.5662E-01	8.2658E-01	8.9656E-01
360.	4.9580E-01	5.6079E-01	6.2748E-01	6.9515E-01	7.6344E-01	8.3215E-01	9.0114E-01	9.7035E-01
480.	6.6248E-01	7.0983E-01	7.6534E-01	8.2559E-01	8.8652E-01	9.5318E-01	1.0190E+00	1.0857E+00
600.	1.0169E+00	9.9010E-01	1.0067E+00	1.0428E+00	1.0895E+00	1.1425E+00	1.1995E+00	1.2593E+00

POOR ORIGINAL

Table B-5

LOAD FACTORS CORRESPONDING TO 1.0E-04 TARGET LIMIT STATE

GAMMA1 = 1.1934 GAMMA2 = 0.8660 GAMMA3 = 1.0929

LIMIT STATE PROBABILITIES CORRESPONDING TO 1.0E-04 TARGET LIMIT STATE

LENGTH(IN)	PIPE DIAMETER(IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	1.1083E-04	1.1191E-04	1.1271E-04	1.1331E-04	1.1379E-04	1.1417E-04	1.1448E-04	1.1475E-04
240.	9.5317E-05	9.8146E-05	1.0037E-04	1.0216E-04	1.0362E-04	1.0483E-04	1.0586E-04	1.0673E-04
360.	8.3369E-05	8.5369E-05	8.7596E-05	8.9725E-05	9.1665E-05	9.3405E-05	9.4956E-05	9.6339E-05
480.	9.0851E-05	8.4621E-05	8.2783E-05	8.2734E-05	8.3474E-05	8.4560E-05	8.5785E-05	8.7042E-05
600.	1.5789E-04	1.1573E-04	9.8065E-05	8.9697E-05	8.5553E-05	8.3563E-05	8.2757E-05	8.2633E-05

PIPE THICKNESS CORRESPONDING TO 1.0E-04 TARGET LIMIT STATE

LENGTH(IN)	PIPE DIAMETER(IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	3.9188E-01	4.8726E-01	5.4265E-01	6.1805E-01	6.9346E-01	7.6887E-01	8.4429E-01	9.1971E-01
240.	4.4007E-01	5.1456E-01	5.8934E-01	6.6429E-01	7.3935E-01	8.1448E-01	8.8968E-01	9.6491E-01
360.	5.3895E-01	6.0844E-01	6.7789E-01	7.5046E-01	8.2375E-01	8.9751E-01	9.7180E-01	1.0459E+00
480.	7.2381E-01	7.7239E-01	8.3107E-01	8.9504E-01	9.6215E-01	1.0313E+00	1.1017E+00	1.1731E+00
600.	1.1333E+00	1.0915E+00	1.1023E+00	1.1388E+00	1.1868E+00	1.2422E+00	1.3024E+00	1.3659E+00

LLNL:1980/7

NRC FORM 335 (7-77)		U.S. NUCLEAR REGULATORY COMMISSION BIBLIOGRAPHIC DATA SHEET		1. REPORT NUMBER (Assigned by DDCI) NUREG/CR-1624 UCID-18674	
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Load Combination Program Progress Report No. 5				2. (Leave blank)	
7. AUTHOR(S) C. K. Chou, S. C. Lu, M. W. Schwartz, J. C. Dutton, L. L. George, F. M. Gilman, R. A. Larder, R. D. Streit				3. RECIPIENT'S ACCESSION NO.	
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Lawrence Livermore National Laboratory P.O. Box 808 Livermore, California 94550				5. DATE REPORT COMPLETED MONTH July YEAR 1980	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) U.S. Nuclear Regulatory Commission Nuclear Regulatory Research Washington, D. C. 20555				DATE REPORT ISSUED MONTH September YEAR 1980	
13. TYPE OF REPORT Technical				6. (Leave blank)	
15. SUPPLEMENTARY NOTES				7. (Leave blank)	
16. ABSTRACT (200 words or less) <p style="text-align: center;"> This document is a progress report on the Load Combination Program (LCP) covering the period April 1, 1980 through June 30, 1980. The report gives a general description of the program by project and tasks, together with financial summaries, technical reports generated, and meeting attendance. Two appendixes which discuss technical subjects are also included. </p>				8. (Leave blank)	
17. KEY WORDS AND DOCUMENT ANALYSIS				9. (Leave blank)	
17a. DESCRIPTORS				10. PROJECT/TASK/WORK UNIT NO.	
17b. IDENTIFIERS/OPEN-ENDED TERMS				11. CONTRACT NO. FIN A0126, A0133, A0362, A0363	
18. AVAILABILITY STATEMENT Unlimited		19. SECURITY CLASS (This report) Unclassified		21. NO. OF PAGES	
		20. SECURITY CLASS (This page) Unclassified		22. PRICE \$	