Probabilistic Safety Analysis V

Keywords:
- Risk
- Probability-Consequence
- ATWS
- Fuel Cycle Risk
- Seismic Hazard
- Engineered Safeguards

EPRI

EPRI NP-1275
Project 1233-1
Interim Report
December 1979

MASTER

Prepared by
Science Applications, Inc.
Palo Alto, California

ELECTRIC POWER RESEARCH INSTITUTE

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Research Project 1233-1

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Prepared by

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EPRI PERSPECTIVE

PROJECT DESCRIPTION AND OBJECTIVES

This is a status report of Research Project (RP) 1233-1, the Probabilistic Safety Analysis study. The basic objective of this project and report (and its four predecessors issued under RP217-2 and RP767-1) has been the development of probabilistic risk and safety analysis codes and methods and their application to real situations of importance to the industry.

PROJECT RESULTS

This document details nine major activities carried out during the year; included are the following:

- Two workshops in probabilistic analysis for utility personnel.
- A review of the H. Lewis Committee report on WASH 1400 (NP-1130). Comparison with EPRI's reviews of 1975 (EPRI 217-2-1 and 217-2-3) shows that Lewis found little that had not already been said.
- Further developments in and evaluation of methods, and a start in creating an understanding of which tools and needs go together.
- A reevaluation of the WASH 1400 pressurized water reactor (PWR) scram system. This data based analysis shows that the WASH 1400 analysis was conservative by a factor of about 10 and that the boiling water reactor (BWR) and PWR scram reliabilities are comparable.
- Three sensitivity analyses on the importance of the engineered safety features (ESF) and of steam explosions were started; two were completed. These show that steam explosions are not and cannot be significant from a risk viewpoint until the probability of a steam explosion (given a loss-of-coolant accident or transient) becomes a significant fraction of unity (greater than 0.1 to 0.25). The ESF studies show that only some features can be validated from a public risk viewpoint, but that others, while not effective at reducing public risk, can significantly reduce the risk of severe plant damage.
- A description of the present state of EPRI's seismic hazard studies. Two major reports have been accepted for publication in the <u>Bulletin of the Seismological Society of America</u>. These

describe the definition of a new magnitude indicator and the existence of a universal frequency magnitude shape.

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ABSTRACT

This report summarizes work performed during the first year of RP1233-1. This two-year project is devoted to continued development of probabilistic risk assessment methods and their application within the industry. Two risk assessment workshops are described. A summary of a recent EPRI report comparing the Lewis Committee review of the Reactor Safety Study with earlier EPRI reviews reveals general agreement. Progress in fault tree methodology and code development is summarized. Two analyses related to anticipated transients without scram are reported. Sensitivity studies to determine the economic and risk reduction value of LWR safety features are documented. Two new consequence codes, one to predict time-dependent post-LOCA containment conditions and one to calculate radiation doses to internal organs, are described. The status report of the EPRI fuel cycle risk assessment is summarized. Progress in establishing earthquake frequency-magnitude relations and development of a model to predict earthquake strong motion are described. Finally, rapid response efforts following the accident at Three Mile Island are reported.

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SUMMARY

This interim report summarizes work performed during the first year of EPRI contract RP1233-1 on probabilistic safety analysis for nuclear power plants. The primary goal of this study and its predecessors has been the continued development of probabilistic risk assessment methodology and its application to real situations of concern to the nuclear power industry.

EPRI sponsorship of risk assessment methodology development began in 1974 with RP217-2 and continued through RP767-1. Thus the activities documented in this report represent the results of an on-going program dedicated to the formulation of standardized procedures in safety and reliability analysis. Section 1 of this report outlines the background of this on-going process.

In an initial effort to inform potential users about risk assessment methods and tools, two workshops were conducted during the past year. The first of these workshops was a one-day presentation to the members of the EPRI Nuclear Safety and Analysis Task Force in San Diego, California. The second was a four-day workshop held at EPRI headquarters for representatives of member utilities. These presentations are described in Section 2.

Section 3 presents a summary of a recent EPRI report (NP-1130), "A Comparison of the EPRI and Lewis Committee Reviews of the Reactor Safety Study" (RSS). The comparison revealed that the earlier EPRI review, which addressed specific items in the RSS, covered nearly all the more general issues raised by the Lewis Committee and that there was generally substantial agreement on those issues. An exception, discussed in detail in NP-1130 and summarized in Section 3, was the relative conservativeness of the RSS results.

Activities in fault tree methodology and code development are described in Section 4. A methodology for evaluating common-cause failures in large fault trees is being developed. Also, efforts have been initiated to expand the WAM code series with a new code that will calculate system mean time to failure. New versions of GO and CAT were acquired. CAT was modified to supplement the

capabilities of the WAM series. The methodology of the GO code was subjected to a comparison with those of SETS and the WAM codes. This comparison will form the basis of an instructional guide to system reliability and fault tree analysis.

Two analyses conducted on the subject of anticipated transients without scram (ATWS) are presented in Section 5. A NRC Staff Report, NUREG-0460, suggested that the number of reactor coolant system (RCS) relief valves be increased for PWRs. An analysis of the accident sequences in the RSS involving RCS valves was performed. It was found that adding valves to the RCS, in order to decrease the probability of failure of relief valves to open, resulted in increasing the probability of failure of the valves to close. A second ATWS analysis involved a reevaluation of the PWR reactor protection system (RPS) fault tree in light of recently updated component failure data. The new results show a decrease in the probability of RPS failure on demand of about one order of magnitude.

Section 6 summarizes three sensitivity studies conducted for light water reactors (LWRs). Present designs for commercial nuclear power plants include engineered safety features (ESF) intended to mitigate the consequences of a loss-of-coolant accident (LOCA). Sensitivity studies are being performed to determine the value of such ESF systems in terms of economics and risk reduction. The first phase of this effort has focused on ESF for pressurized water reactors (PWRs). The general approach taken in this study was to develop a model which included all core melt accident sequences defined in the RSS, classified according to release categories. Six cases, ranging from one with no containment, containment functions or Emergency Core Cooling System (ECCS), to one with all systems considered in the RSS, were quantified. In general, the results showed probabilities shifting from more severe to less severe release categories as more ESF systems were added to the scenario. However, in some instances, the model was forced to add systems in an order that deviated from the order presecribed by the system event trees. This deviation could have caused some accident sequences to be canceled out.

A parallel sensitivity study for boiling water reactors (BWRs) was initiated later in the year with the development of a model similar to that constructed for the PWR phase. Finally, a separate sensitivity study focusing on the potential of a vapor explosion occurring in conjunction with a postulated core melt was also conducted. The results of this limited investigation indicated that the probability of a vapor explosion in a PWR must be greater than 0.1 in order to affect societal risk.

Consequence analysis efforts during the past year have focused on the continued development and expansion of computer codes. A new containment behavior code package, INCOR, will merge the features of CONTEMPT-LT, BOIL, and INTER to provide input to the CORRAL code. INCOR will predict time-dependent containment conditions following a LOCA. Another new code, INRAD, has been assembled to calculate radiation doses to internal organs. INRAD incorporates the features of an improved International Commission on Radiological Protection Task Group Lung Model (TGLM) and a gastrointestinal tract model. Finally, in conjunction with the LWR sensitivity studies, the capabilities of the Calculations of Reactor Accident Consequences (CRAC) code have been extended to generate intermediate health effects for whole body and thyroid doses 50 miles from the plant site. A supplementary program, CRAC-FINAL, was developed to stage CRAC results from a tape. These consequence code development activities are described in Section 7.

The status of the EPRI fuel cycle risk assessment is summarized in Section 8. The primary purpose of this project is to complete the estimated radiological risk of nuclear power generation by addressing risks presented by the supporting fuel cycle. Routine risk from mining and milling and accident risks from reprocessing, mixed-oxide fuel fabrication, transportation of recovered material, and waste disposal were investigated and reported in five draft documents. These drafts were then modified in response to peer review, and a separate EPRI status report was issued (EPRI NP-1128). Present results indicate that the total fuel-cycle contribution to risk is about 1% of reactor accident risk. Thus nuclear power accident risk is approximately that of the power plant itself, which in turn is about 0.5% of natural background radiological risk.

Seismic hazard research during the past year is reported in Section 9. This work was carried out in three general areas, the first of which was the publication of progress in earthquake frequency-magnitude (f-M) relations studies for large regions. A universal shape regularity was observed that agrees with data subsets taken from eight separate geographical regions. This initial success in establishing f-M relations to large areas has led to a second investigation to determine if the universal f-M shape is applicable to regions small enough for estimation of local seismic hazard. Preliminary results indicate that regional tectonic zone data agree with the universal shape. Likewise, preliminary estimates of seismic activity obtained using the universal curve with historical data show good agreement with geologic estimates. A third investigation involved the development of a model, based on mathematical representations of surface waves and reduction of data from the NRC strong-motion data base, to predict the probability of earthquake strong motion at a specific site. The code computes ground shaking at a desired location due to a single earthquake or to a stochastic array of energy sources with magnitudes, depths, and locations that follow empirical distributions. It is anticipated that the results of the f-M research and the strong-motion study will eventually be combined to provide a method for estimating the probability of earthquake acceleration at a given location.

In the aftermath of the event at Three Mile Island, the EPRI probabilistic analysis group offered assistance to the task force evaluating the reliability of critical systems. These activities are reported in Section 10. In particular, the availability and reliability of off-site electric power at the site, the TMI2 Decay Heat Removal (DHR) system, and a proposed modification of a temporary emergency feedwater system alignment were assessed and recommendations for action were made. In the case of off-site power, it was concluded that dedicating a single power line to TMI2 would not significantly increase power availability. Fault tree analysis of the DHR system indicated that the positions of certain valves could significantly reduce system reliability. It was recommended that motor-operated valves be placed in DHR-mode position to minimize the total number of position changes required for activation in an emergency. The modified temporary emergency feedwater system involved using a new temporary crossover and diesel-driven pumps to divert water from the secondary side of the steam generators. This system was also subjected to fault tree analysis. It was concluded that the possibility of installation error was much more likely than any additional reliability the modification would offer. A separate analysis, prepared as background for the evaluation of the situation at Three Mile Island, involved a comparison of core parameters, RCS volumes, RCS overpressure protection, and steam generator characteristics for five PWR nuclear steam supply system designs.

Section 1

INTRODUCTION AND BACKGROUND

In 1974 EPRI contracted with Science Applications, Inc. (SAI) to establish a dedicated center for probabilistic analysis. The primary goal of RP217-2 was to create a functioning risk and reliability assessment group to supply the EPRI Nuclear Power Division with expertise in probabilistic safety methods.

The basic thrust of this two-year project was the continued development of probabilistic risk and safety analysis tools and methods and their application to real situations of importance to the utility nuclear power industry. These activities were subsequently carried forward into 1978 through RP767-1. RP1233-1, which represents the fifth and sixth years of on-going research dedicated to probabilistic analysis associated with nuclear power, began in April of 1978.

Probabilistic risk and safety analysis requires careful examination of both the consequences and the likelihood of abnormal plant or process operation. Combining these two kinds of evaluation, interpreting them in both absolute and relative measures, and defining the measures themselves, require knowledge of a broad spectrum of engineering disciplines. Standardized methods of analysis, such as procedures, computer codes, and data collection, serve to simplify this complex process. The development of such a standardized process has been a continuing goal of the dedicated group.

A major effort of the first year's activities was spent in analyzing and criticizing the draft version of the Reactor Safety Study (WASH-1400) (1-1). Most of the original project staff members had previously participated in the Reactor Safety Study. The results of this work were published in two documents, EPRI 217-2-1 (1-2) and EPRI 217-2-3 (1-3). A summary of these documents may be found in the first annual report, EPRI 217-2-4 (1-4).

The final report for RP217-2 (1-5) summarizes work accomplished during the first two years. A detailed sensitivity/perturbation analysis was conducted for the WASH-1400 PWR and BWR. The computer codes utilized in WASH-1400 were examined in detail, and a new family of codes, the WAM series, was developed to evaluate plant risk both quantitatively and qualitatively. An evaluation of the risk due to anticipated transients without scram (ATWS) was initiated which subsequently resulted in the publication of a series of EPRI reports on the subject (1-6,1-7,1-8). A systematic means of gathering actual plant failure data was developed, and work began on safety analysis verification techniques.

A summary of activities during the first year of RP767-1 can be found in EPRI NP-749 (1-9). In continued ATWS research, a careful analysis of shutdown history was undertaken to determine the probability of shutdown system failure, the likelihood of occurrence of various anticipated transients, and the attendant risk involved. Two codes, WAM-BAM and WAM-CUT, were developed for fault tree evaluations and sensitivity studies. Consequence analysis of postulated radiological releases and their impacts was undertaken and modifications made to consequence codes from WASH-1400. A brief study was made of the probability of failure and key failure modes of a decay heat removal system. The groundwork was laid for an EPRI/SAI/industry comprehensive assessment of radiological risks from external fuel cycle operations that support nuclear power plants, and a quantification of seismic effects was initiated with an investigation into earthquake frequency-magnitude relations.

The second year of RP767-1 is summarized in EPRI NP-1039 (1-10). The WAMCUT computer code was documented in a user's manual (1-11). Fault tree methodology was utilized for two reliability assessments of heat removal systems: 1) a reevaluation of the WASH-1400 BWR, and 2) an analysis of the Brookhaven National Laboratory critique of the Clinch River Breeder Reactor system. A separate fault tree analysis assessed seismic risk associated with the Diablo Canyon nuclear power plant. In source term and consequence analysis efforts, a general purpose computer program was developed from the CONTEMPT systems analysis code to provide time-dependent, post-LOCA containment conditions. In ATWS studies, the SEARCH computer code was designed to store, retrieve, and analyze the ATWS data collected since 1975. Five draft documents were assembled in the EPRI Fuel Cycle Risk Assessment on uranium mining and milling, transportation, spent fuel reprocessing, mixed oxide fuel fabrication, and waste disposal. And, finally, in seismic studies a technique based on a universal world curve for earthquake frequency vs. magnitude was developed to determine return periods for large seismic events.

This report documents the first year's efforts on Research Project 1233-1, and the fifth year of the on-going activities of the EPRI probabilistic analysis group. One month before the close of this year, the accident at Three Mile Island nuclear power plant occurred. The expertise and results of the dedicated group were placed at the disposal of EPRI and the Nuclear Regulatory Commission in a rapid response to the situation. Preliminary efforts regarding Three Mile Island are described at the end of this report.

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1-10 Probabilistic Safety Analysis IV. Palo Alto, Calif.: Electric Power Research Institute, April 1979. EPRI NP-1039.

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Section 2

RISK ASSESSMENT WORKSHOPS

EPRI TASK FORCE WORKSHOP

On June 26, 1978, the EPRI Probabilistic Reactor Safety Studies group presented a one-day tutorial workshop to the members of the EPRI Nuclear Safety and Analysis Department Task Force in San Diego, California. The purpose of this presentation was to review the basic components of risk assessment.

The agenda included:

- A review of mathematical concepts involved in risk assessment
- Discussions of fault trees and event trees
- A discussion of applicable computer codes
- Examples of the application of these risk assessment methods and tools to specific problems, including:
  - Anticipated transients without scram (ATWS)
  - Seismic response of a nuclear power plant

The text and visual materials from this one-day workshop have been published in EPRI NP-79-1-LD, "A Risk Methodology Presentation."

UTILITY WORKSHOP

The following September a four-day risk assessment workshop was held at EPRI headquarters in Palo Alto. This expanded program was attended by 21 individuals from 19 EPRI member-utilities. Table 2-1 outlines the program of the utility workshop. Briefly, the course began with an overview of risk methodology. Three initial presentations reviewed probability concepts, data and data applications, and fault trees and event trees as plant response models. The next two sessions dealt in detail with fault tree construction and computer evaluation techniques. A summary of selected risk assessment studies was then presented, and the last two sessions reviewed two specific applications: ATWS and seismic events.

It is anticipated that another utility workshop will be presented in 1979. Practical applications, rather than theory, will be emphasized.

Table 2-1

RISK ASSESSMENT WORKSHOP
SEPTEMBER 18-21, 1978
PALO ALTO, CALIFORNIA

INTRODUCTION

COURSE INTRODUCTION AND OVERVIEW

INTRODUCTION TO RELIABILITY AND SAFETY ANALYSIS

- Definitions of Reliability and Safety Terminology
- Probability Concepts
    - Definitions of Probability
    - Sample Space Axioms of Probability; Conditional Probability
    - Combining Probabilities
    - Bayes Equation
    - Difference Between "Classical" and Bayes Statistics
- Logic
    - Set Theory
    - Boolean Equations
    - Identities
    - AND and OR Operations

DISTRIBUTION FUNCTIONS

- Distribution of Random Variables
    - Continuous and Discrete Distributions

Table 2-1 (continued)

- Moments and Generating Functions
- Combining Distributions

• Examples of Failure Rate Estimation
  - Chi-Squared
  - Binomial

RELIABILITY MODELS

• Fault Trees
• Event/Decision Trees
• Markov Models
• Comparison of Modeling Types

INTRODUCTION OF FAULT TREE AND EVENT TREE CONSTRUCTION

• Definition of Fault Trees and Event Trees
• Uses in Reactor Safety Study
• Development of Event Trees; Advantages and Disadvantages
• Development of Fault Trees; Advantages and Disadvantages

FAULT TREE AND EVENT TREE CONSTRUCTION TECHNIQUES

• General Approach
• Event Modeling
  - Components
  - Operator Interface
  - Maintenance
  - Dependent Events
• Common Mode Representations

Table 2-1 (continued)

- Data Base Usage
- Meaning of Parameters

EXAMPLE PROBLEM AND DATA APPLICATION

CAT CODE USAGE FOR AUTOMATIC FAULT TREE CONSTRUCTION

- Basic CAT Code Methodology (Decision Theory)
- Development of Decision Tables with Examples
- CAT Code Mechanics
- CAT Input Requirements with Examples
- Applications to Power Systems
- Overview, Summary and Discussion

FAULT TREE/EVENT TREE EVALUATION TECHNIQUES AND RELATED COMPUTER CODES

- Evaluation Techniques
    - Direct Evaluation
    - Indirect Evaluation
    - Error Bound Propagation
    - Cut Sets
- Use of Fault Tree Evaluation Codes
    - Fault Tree/Event Tree Input
    - Data Input
    - WAMBAM
    - WAMCUT
    - SPASM
    - FRANTIC
    - WAMDRAW

SPECIAL TOPICS IN FAULT TREE AND EVENT TREE ANALYSIS

- Sensitivity Studies
- Seismic Events

Section 3

A COMPARISON OF THE EPRI AND LEWIS COMMITTEE REVIEWS OF THE REACTOR SAFETY STUDY

The draft version of the Reactor Safety Study (RSS), also known as WASH-1400 (3-1), was issued in August 1974. Approximately one year later the EPRI probabilistic safety analysis group released a comprehensive review (3-2) and a critique (3-3) of the draft report. Both of these publications were prepared by SAI in response to a request for peer review. The final version of WASH-1400 (3-4) was released in October 1975. In 1977 an *ad hoc* review group (the Lewis Committee) was formed to review the final version. Their report (3-5) was released in September 1978.

This section summarizes a recently published report (3-6) comparing the Lewis Committee report, which reviewed the final version of WASH-1400, to the earlier EPRI reports, which dealt with the draft version. Several topics addressed in the Lewis Report were not addressed in the earlier EPRI reports, mainly due to the fact that they were completed before certain events had occurred (e.g., the Browns Ferry fire, March 1975). Such items were excluded from this comparison.

The comparison revealed that the earlier EPRI work addressed specific items within the RSS. This detailed critique covered nearly all the more general technical issues raised by the Lewis Committee. There is generally substantial agreement regarding the issues addressed. Among the exceptions is the relative conservativeness of the RSS results. After carefully examining the calculations, EPRI concluded that the RSS results were conservative. The Lewis committee, however, expressed the belief that the error bounds were "greatly understated."

Although there can be no clean calculation of the effect of events which have not occurred, the authors of the earlier EPRI work used judgment to quantify the items believed not completely treated in WASH-1400. In the comparison report those items still applicable to the final version of the RSS were combined mathematically and judgmentally to determine their effect on the WASH-1400 results. It was shown that the median values for core melt probability were most likely a factor of 12 less than stated in WASH-1400 and that the uncertainty was indeed larger than stated in WASH-1400.

However, it is also possible to use the expected commercial date of operation of the reactor population to infer the core melt probability for the remainder of the century. This assumption leads to the perception that there may be no upside error in the WASH-1400 calculation, and a tentative conclusion that the upside error in WASH-1400 is maximally less than a factor of 4. Appendixes C and D of the comparison report (3-5) are devoted to this approach.

AREAS OF GENERAL AGREEMENT

In terms of a general overview of the Reactor Safety Study, the EPRI critique stated that WASH-1400 "represents perhaps the most comprehensive work yet performed on nuclear reactor safety from a probabilistic viewpoint [and] as such... will provide a foundation for all future work in the United States in this area" (3-3). The Lewis Committee agreed that the study was "a substantial advance over previous attempts to estimate the risks of the nuclear option. The methodology has set a framework that can be used more broadly to assess choices involving both technical consequences and impacts on humans" (3-5).

General agreement was also found between the two reviews in the following subject areas:

- Risk assessment methodologies
  - Use of event trees and fault trees
  - Limits on completeness
  - Need to include failure rate variability
- Statistical issues
  - Use of the geometric mean, or "Square Root Bounding Model"
  - Log-normal distribution

- Need for clarification and definition for choice of models, assumptions, and methods
- Use of an "average site" instead of site-specific and plant-specific data
- Inappropriateness of "smoothing" technique

- Event completeness
- Common-cause failures
- Human factors
- Scrutability
- Earthquakes
- Acceptable levels of risk for nuclear energy
- Sabotage and war
- Influence of design defects and quality assurance failures
- Calculations of population doses from releases of radionuclides
    - Sensitivity analysis
    - Evacuation model

TOPICS ADDRESSED BY ONE REVIEW ONLY

It was found that the Lewis Committee report specifically addressed four issues not included in the EPRI reviews:

- Use of the median instead of the mean of log-normal distributions
- Narrow spectrum of experience of RSS team
- Lack of a specific definition of risk
- Underestimation of risk from anticipated transients without scram (ATWS)

It should be noted, however, that the ATWS issue was addressed in considerable detail and reported in several documents published separately from the basic EPRI reviews (3-7,3-8,3-9,3-10,3-11,3-12).

Conversely, the EPRI review documents addressed several issues not raised by the Lewis Committee. The EPRI critique (3-3) pointed out that the reactors analyzed in WASH-1400 were licensed under much less stringent regulations than present-day reactors and that the effect of initiating events at multi-unit sites was not considered. Other topics addressed by the EPRI team, but not by the Lewis Committee, included:

- Accident sequences and accident definition
  - Spent fuel pool accident
  - Anticipated transients
  - BWR accident sequences
  - Core behavior in the soil and water table
- Consequences
  - Mitigating effects of medical action
  - Costs of land damage, loss of assets, and relocation
  - Transient and small pipe break consequences
  - Sensitivity to conservative assumptions
- Data assessment
  - Normal operation data
  - Mixing of data from plant startup and early life with plant midlife data
  - Maintenance data

CONSERVATISM AND CONFIDENCE BOUNDS

Several of the findings of the Lewis Committee addressed the question of whether the results of the RSS were conservative or nonconservative. The committee concluded that:

> We are unable to determine whether the absolute probabilities of accident sequences in WASH-1400 are high or low, but we believe that the error bounds on those estimates are, in general, greatly understated.(3-5)

The earlier EPRI review attempted to quantify the various factors involved and concluded that the RSS was "quite conservative" in terms of the potential contributors to risk that were considered. However, the EPRI critique (3-3) also identified 27 items which were felt to be inadequately treated in the draft of WASH-1400. Each of the 27 items was evaluated for its impact on the risk relation (both probability of core melt and consequence) and where such an impact could be quantified, the result was recorded. Of the 27, only item 12 (sabotage) was felt to be unquantifiable. Some of these results were based on technical insight and related technical experience and as such are the end-points of sound technical judgment. Appendix A of the comparison report (3-6) presents a discussion of judgment in terms of assessment of uncertainty.

The EPRI critique of WASH-1400 (3-3) assumed that, given sufficient time (many of these accidents would take hours to have impact on the public), knowledgeable people at the plant could alleviate the accident consequences or divert a potentially bad sequence into a safe shutdown.

With some care, this portion of the EPRI critique can be used to estimate the total shift in the WASH-1400 point estimate and uncertainty bounds and hence to address the concern raised by the Lewis Committee. Table 3-1 indicates the results of this process.

On the original figures given in WASH-1400, uncertainties were "estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities." These uncertainty bands for one case are sketched in Figure 3-1. Also shown there is the range of uncertainty as indicated in Table 3-2. The median shifts and uncertainty factors in Table 3-2 were obtained by combining values from Table 3-1 as if combining log-normal distributions, expressed in equation form as:

$$UF = \exp\left[\sqrt{(\ln UF_1)^2 + (\ln UF_2)^2}\right] \qquad (3\text{-}1)$$

Using the results in Table 3-2, an updated estimate of the median values for the accident effects can be created. Such an update is shown in Figure 3-1, which contains the expected range of probabilities for early fatalities for both the original WASH-1400 estimate and for this update. The error band is wider in the update, but the whole band has been shifted downward so that the new upper bound is less than the old upper bound.

Table 3-1

REDUCED TABLE OF ESTIMATED VARIATIONS

Item Identified From RSS		Probability		Consequences	
		Median Change	Uncertainty	Median Change	Uncertainty
Pt. 1	Learning Curve	-2	--	none	--
Pt. 2	Single vs. Multi-unit Site	--	--	--	--
Pt. 3	Mitigating Medical Treatment	none	--	--	--
Pt. 4	Demographic Changes With Time	--	--	--	--
Pt. 5 & 9	Site Specific Population and Meteorology Variations	none	--	--	± 10
Pt. 6 & 7	Scaling 2 to 100 Plants and Evacuation	--	± 3	none	--
Pt. 8	Land Damage	N/A	N/A	N/A	N/A
Pt. 10 & 11	Category Smoothing and Overlooked Sequences	--	--	--	--
Pt. 12	Sabotage	?	?	?	?
Pt. 13 & 14	Operator Behavior and Human Error	-3	--	--	--
Pt. 15	Core Meltthrough	--	--	--	--
Pt. 16	Partial Core Melt	none	--	-5	--
Pt. 17 & 18	Break Location and Steam Explosion	-2	--	none	--
Pt. 19	Spent Fuel Storage	none	--	+2	--
Pt. 20	Non-LOCA sequences	--	--	--	--
Pt. 21	Containment Plateout	none	--	-2	--
Pt. 22	PWR LPIS Fixed	--	--	--	--
Pt. 23, 25-27	Fault Tree Analyses Errors, Partial System Success, Partial Component Success, NRC Definition of Operability	--	--	--	--
Pt. 24	Analysis Inconsistencies	--	± 10	none	--

NOTE: -- indicates insignificance

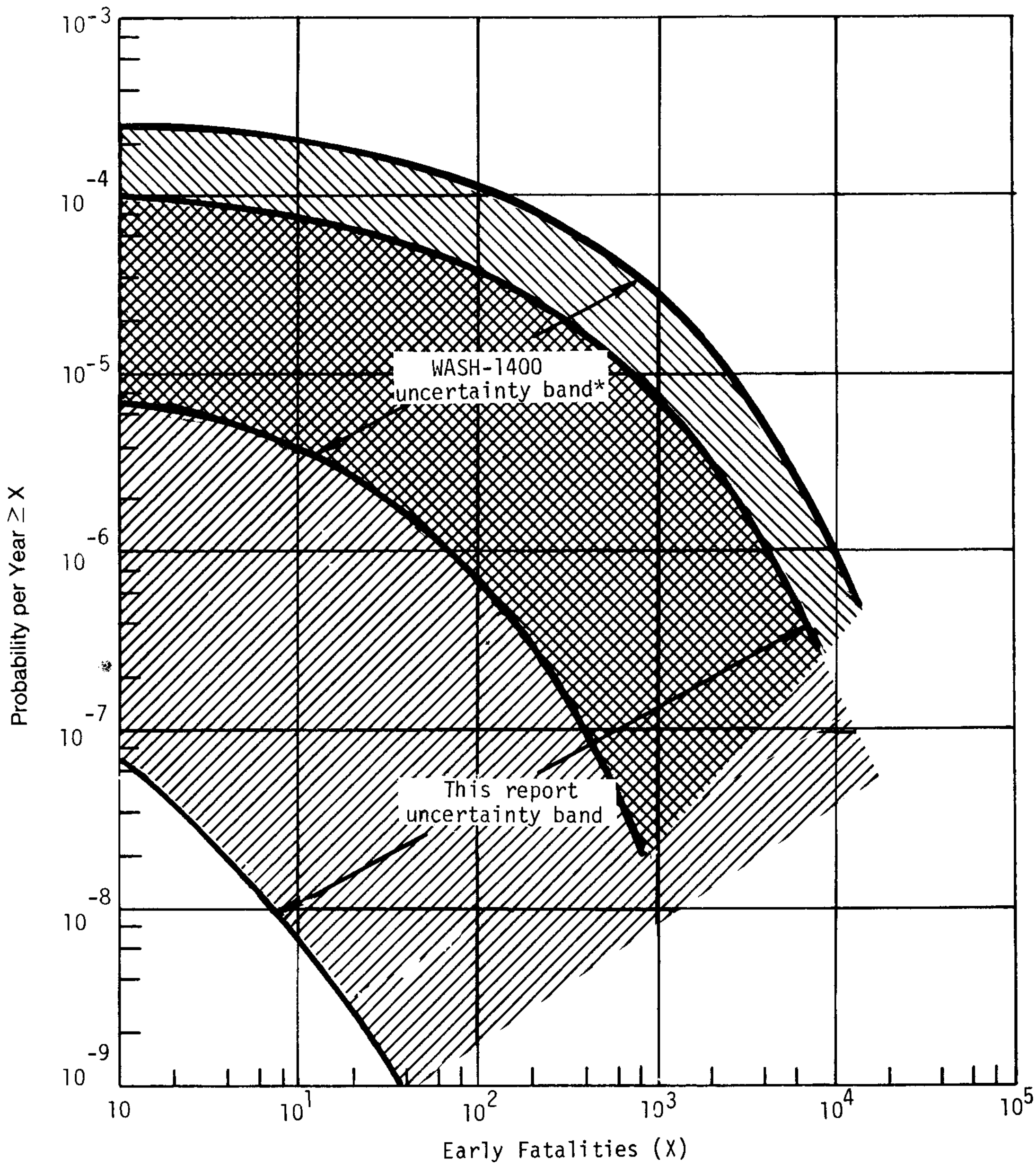

Figure 3-1. Probability Distribution for Early Fatalities Per Year For 100 Reactors

Table 3-2

SHIFTS IN ACCIDENT FREQUENCY AND CONSEQUENCES

	Accident Frequency	Accident Consequence
WASH-1400 Uncertainty	5	5
Multiplicative shift in median	1/12	1/5
Increase in multiplicative uncertainty factor	13	10
Total multiplicative uncertainty including WASH-1400 (see Eq. 4-1)	20	15

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Section 4

FAULT TREE METHODOLOGY AND CODE DEVELOPMENT

A common-cause evaluation methodology is being developed for analyzing large fault trees. This new methodology applies efficient reduction techniques to fault trees that have been expanded to include common-cause failures. An attempt is also being undertaken to expand the WAM code series with a separate code to calculate system mean time to failure (MTTF).

Revisions of two codes (4-1,4-2,4-3,4-4) have been acquired for supplemental fault tree analysis. The methodology of one of these codes, GO, has been subjected to a comparison with that of the WAM codes (4-5,4-6) and of SETS (4-7). It is anticipated that this comparison will form the basis of a guide to system reliability and fault tree analysis technology.

COMMON CAUSE EVALUATION METHODOLOGY FOR LARGE FAULT TREES

Common-cause applications to fault tree methodology have historically been limited in size and scope. This limitation has been due to the relatively restricted capacities and speeds of the computer codes available for fault tree analysis. These fault tree codes, which require minimal cut-sets as input, are not practical for modeling a large, real-world system. Codes that do not require minimal cut-sets as input are usually limited to identifying only single common-cause events that lead to system failure. However, in some cases it is important to identify double common-cause events, e.g., for fires in adjacent rooms (4-8). It may also be necessary to identify common-cause events combined with random failure events that together lead to system failure.

The evaluation methodology now being developed extends common-cause failure analysis by including a descriptive cause set, which is a description of the manner in which a common-cause event leads to system failure. A descriptive cause set is a minimal cut-set of a fault tree in which the basic events have been transformed to represent both random failure and failure from common-cause event(s). These minimal cut-sets contain descriptions of each component by identifying both the failed component and the type (cause) of its failure, which may be either common-cause or random failure.

The structure for the event transformation is shown in Figure 4-1. A fault tree component is replaced with logic that combines a unique component identifier with the common-cause events to which the component is susceptible. The descriptive cause sets, which result from Boolean reduction of the fault tree containing the transformed events, are filtered to leave only the cut-sets containing:

- A single common-cause event and its associated failures
- A single random component failure plus a single common-cause event and its associated component failures
- Double common-cause events and their associated component failures

From these results fault tree analysis can identify the components that should be protected to assure a system's invulnerability to the common-cause event(s) identified.

Because the number of cut-sets generated is greatly increased, the event transformation greatly complicates the fault tree model. The new methodology utilizes a variety of algorithms to minimize the number of terms to be reduced at any one time.

An important factor of this approach is the efficiency and flexibility of the SETS computer code developed at Sandia (4-7). SETS provides sufficient versatility and speed to analyze large fault trees. A computer program will be written to take advantage of the capabilities of the SETS code. This program will consist of a preprocessor, which performs the event transformation, and a SETS program generator, which in turn implements modularization algorithms to force SETS to solve the fault tree in the most efficient manner.

Three runs are required to generate the descriptive cause sets. Two initial runs generate single and double common-cause events that can lead to system failure. The third run generates descriptive cause sets that contain the

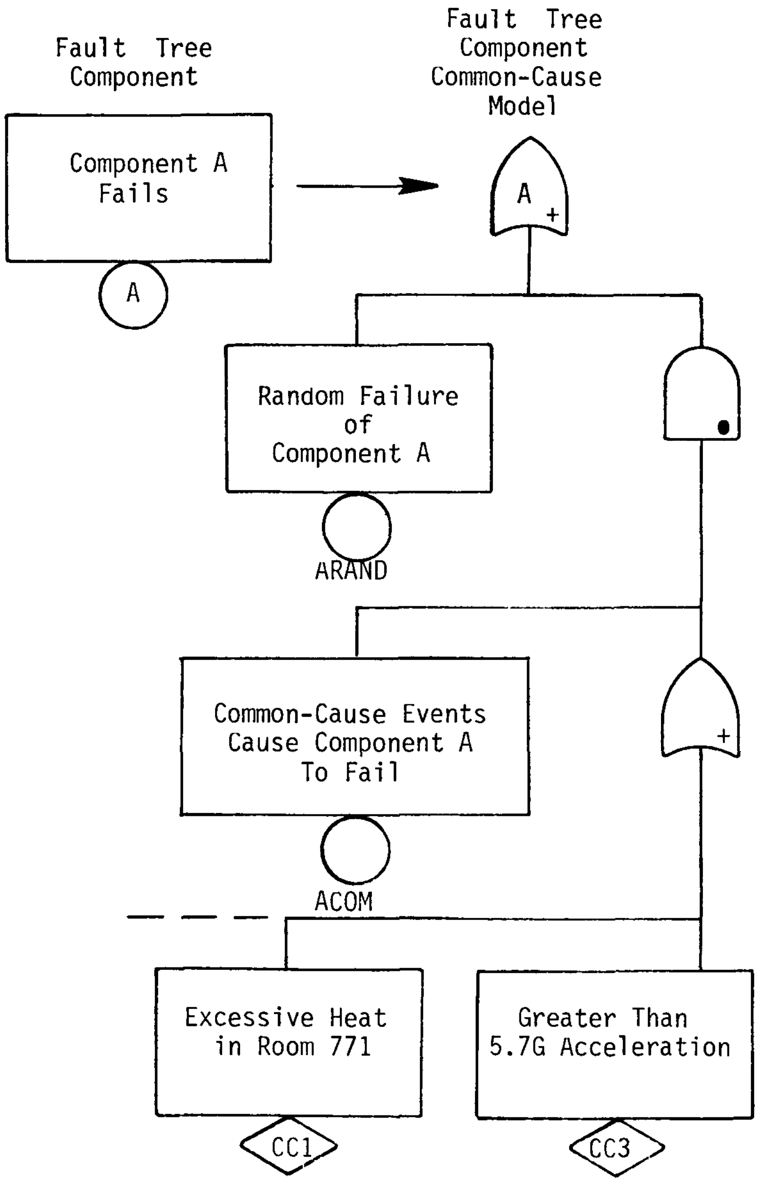

Figure 4-1. Example Fault Tree Component Common-Cause Transformation

common-cause events of interest. In the initial two runs the tree is pruned of all events that do not involve those common-cause event or random failures being investigated. This technique allows the event transformation to take place for only a small portion of the events appearing in the fault tree.

In order to improve the efficiency of the computer runs, the modularization scheme in the program generator will be used to break the tree up into subtrees that can be individually reduced. The fault tree is then rebuilt and reduced in stages, allowing SETS to eliminate terms as quickly as possible in order to pare the problem down to a reasonable level. The use of this program by a skilled fault tree analyst should facilitate the satisfactory solution of large common-cause failure problems.

This methodology was developed using a fault tree for shutdown heat-removal-system analysis for the Clinch River Breeder Reactor (CRBR). This fault tree, which is quite large (1000 components), provided a good exercise for the reduction capabilities of this methodology. The results showed good agreement with those for the more limited CRBR fire analysis (4-8). The new methodology solved the more complex problem with relative ease, establishing that it is possible to obtain actual failure mechanisms along with the significant common-cause events for large, real-world fault trees.

Future work in common-cause failure methodology will focus on program development. A user's guide will be prepared for publication.

MEAN TIME TO FAILURE (MTTF) METHODOLOGY

On a time-permitting basis, several methods have been investigated for calculating system MTTF for a system made up of components with constant failure and repair rates. The ultimate purpose of this project is to develop a code for calculating MTTF that will be included in the WAM fault tree evaluation code series.

Methods investigated to date include graph theoretic approaches and matrix techniques for evaluating simultaneous equations. It was concluded that a signal-flow graph theory method offered the more promising approach.

A computer code was written for a limited case that did not allow for repair rates. However, the basic graph theory method used could not be extended to the more complex case involving repair rates. Therefore, this method was no longer pursued.

Within the next months an investigation of simulation methods will be undertaken. Although a simulation approach will most likely require increased computer run time, it will offer several advantages over the other methods investigated. These advantages include the flexibility to calculate not only constant but also variable repair and failure rates, such as time-dependent or fixed rates. Additional information, such as mean time to repair and variance in time to failure, can be obtained.

CODE ACQUISITION

Two new codes have been acquired for supplemental system analysis.

GO

GO (4-1,4-2,4-3) is a code package consisting of six separate programs. When run successively, GO1, GO2, and GO3 perform basically the same function as WAM-BAM. Input is in the form of a GO chart. The output from GO3 is the unavailability associated with each signal in the GO chart.

The remaining three programs, FF0, FF1, and FF2, comprise the fault-finder portion of the GO package, which is essentially equivalent to WAM-CUT. However, the GO fault-finder routine is limited to finding fault sets of up to order four only.

CAT

The CAT code (4-4) is an automatic fault tree construction program which has been made compatible to the WAM codes. A new routine was added to CAT to create a file containing the generated fault tree in WAM format. CAT now produces input decks for the entire WAM series.

IBM Revisions of WAM Codes

The WAM-BAM and WAM-CUT codes are now fully operational on an IBM system. Northeast Utilities had originally created these revisions and made them available for the EPRI probabilistic group's use. These revisions have been given to the EPRI Code Center for distribution.

COMPARISON OF SYSTEM RELIABILITY AND FAULT TREE CODE METHODOLOGIES

Three of the best and fastest codes available for system reliability and fault tree analysis are GO (4-1,4-2,4-3), the WAM series (4-5,4-6), and SETS (4-7). This effort was undertaken to outline the capabilities of each code's methodology as a guide to choosing the best code or combination of codes for the solution of various types of reliability and/or fault tree analysis problems.

The GO Package

The GO code (4-1,4-2,4-3) provides the ability to analyze systems in which multistate components contribute to an undesired system state. GO can be used to analyze time-dependent system states and to model events requiring signals in a specified sequence. GO methodology uses symbols that are analogous to hardware, without requiring a complete knowledge of a failure mechanism, as is the case for fault trees. Probabilities for various system states can be produced with one computer run.

GO appears to be able to handle small and medium-sized problems with little difficulty, but larger problems have been known to exceed the code's capacity. Large problems should be modeled by a fault tree for each undesired state. GO has the ability to generate fault paths for simple systems of one to four components. However, the limits have been reached on small example problems. GO also has the ability to model a fault tree, but capacity and speed are much smaller and slower than for either the WAM codes or SETS.

The GO methodology should be used if the analyst is interested in small problems that have multiple time points involving multistate components. Reliability diagrams are more easily transformed into the GO symbology than into fault tree diagrams. Therefore, it may be advantageous to use the GO code to evaluate problems formulated with reliability block diagrams, especially if the analyst is unfamiliar with fault tree methodology.

The WAM Series

The codes in the WAM series offer the best means of obtaining system statistics for fault tree analysis. These codes have been developed to generate rapid, accurate solutions for large, complex fault trees.

The two main codes in the series are WAM-BAM and WAM-CUT. These codes are documented in two EPRI publications (4-5,4-6), available from the EPRI Code Center.

WAM-BAM can quickly and efficiently solve a complex fault tree, generating single-point probabilities or the availability of any gate in the tree. It is especially useful for obtaining quick answers to complicated problems. However, WAM-BAM alone does not provide information about dominant fault paths, which is valuable in terms of fault tree verification, as well as for an understanding of the problem solution itself. If dominant fault paths or cut-sets are desired, WAM-CUT should be used.

WAM-CUT lists up to 2000 cut-sets for any gate in a fault tree. Since even the largest fault tree seldom has more than 100 cut-sets within 10% of the cut-set most likely to occur, WAM-CUT will show all the cut-sets that make a significant contribution. WAM-CUT can also provide an output that can be used to generate a complete probability distribution from the various types of distributions for the basic events. Thus WAM-CUT should be used whenever a reliability problem can be put into a fault tree format and numerical solutions are required.

When no probabilistic data are available, WAM-CUT can be run to generate qualitative information about a fault tree. In this case the solution is much more easily obtained than with SETS. However, there is a sacrifice of both efficiency and capacity with WAM-CUT.

WAM-CUT should be used for all quantitative fault tree analysis problems. If it proves to be difficult to arrive at a final answer, WAM-BAM can be used to obtain a numerical result. The WAM-BAM result, however, is merely a number; every effort should be made to use WAM-CUT to obtain the dominant fault paths. WAM-CUT should also be used for a first attempt at a qualitative solution of a fault tree, for example identifying single, double, and triple event cut-sets. If the problem proves to be untenable using WAM-CUT, SETS can be used in a further attempt to arrive at an acceptable result.

The SETS Code

The SETS computer program (4-7) is basically a Boolean equation manipulation tool. Boolean identities are applied to a system of equations usually represented in the form of a fault tree. SETS manipulates the gate and component names to produce cut-sets or prime implicants (in the case of equations, with negated elements). The input to SETS can be either in the form of a fault tree or in the form of Boolean equations. These Boolean equations are combined, expanded, reduced, and factored as the analyst directs to yield a reduced representation of the fault tree. This representation can be listed as cut-sets or in a factored form. Solutions can be derived for any gate or equation within the set of equations. The discussion on the BWR accident sequence model in Section 5 illustrates SETS reduction capabilities.

Additional flexibility is derived from the abilities to store intermediate solutions, to set elements to the empty set or universal set, to truncate a solution on the number of terms in a cut-set, and to redefine gates and components. SETS has the flexibility to perform common-cause analysis (as discussed in the section on common-cause methodology), as well as analysis of event trees and reduced cut-sets.

SETS should be used when a problem is in the form of Boolean equations or when a qualitative solution is required that WAM-CUT is not able to generate. An additional package is available for quantifying the results generated by SETS. The trade-off for SETS's flexibility is a complicated set of instructions that are not easily understood without a thorough knowledge of Boolean algebra and its applications. While the solution of very large fault trees is possible, much analyst interaction is required.

The SETS code, as well as a user's manual (4-7), is available from the Argonne National Laboratory Code Center.

Summary of Applications

Together, these three codes have provided the capability to quickly and efficiently solve all the problems encountered thus far by the EPRI probabilistic analysis group. Six classes of problems have been identified:

- System reliability assessment
- Quantitative fault tree analysis
- Qualitative fault tree analysis
- Boolean equation manipulation
- Multiple time point reliability calculations
- Common-mode failure analysis

System Reliability Assessment. The GO code provides an excellent tool for reliability assessments from a reliability block diagram. It provides analogs for all block diagram symbols. The fault-finder option makes it possible to verify the model accuracy. Analysts unfamiliar with both fault tree analysis and GO methodology have commented that the GO methodology was easier to learn. However, the GO methodology can become quite complicated if feedback loops and large interdependent systems are involved. In this case fault tree analysis should be performed using WAM-CUT as an evaluation tool.

Quantitative Fault Tree Analysis. Quantitative analysis of fault tree diagrams is best performed with WAM-CUT. The truncation schemes used by WAM-CUT provide a quick solution to most problems.* In addition, WAM-CUT lists the cut-sets most likely to cause the undesired event. These cut-sets have proven to be invaluable for fault tree model verification. WAM-CUT can generate an output which can be used to obtain the reliability function from the probability distributions of the components. If for some reason WAM-CUT is unable to generate an answer, WAM-BAM will solve the tree at any gate and produce the first and second moments. Use of WAM-BAM is not advised unless necessary because of the difficulty in verifying the accuracy of the fault tree. However, WAM-BAM is useful in performing sensitivity analysis subsequent to a WAM-CUT run.

*The numerical results can be first and second moments or failure rate and duration time (lambda-tau calculations).

If a tree is too large for WAM-CUT and WAM-BAM, SETS may have the capacity to handle the problem.

Qualitative Fault Tree Analysis. Qualitative fault tree analysis is most easily performed by using WAM-CUT without entering component probability values. WAM-CUT provides an easy, efficient means for generating cut-sets for use in qualitative analysis. However, WAM-CUT does have limited capacity, which may require the use of SETS.

Boolean Equation Manipulation. SETS can factor, reduce, invert, filter, and redefine Boolean equations. These operations can be used to generate prime implicants of a Boolean equation, as well as to convert an equation into a more useful tool for performing sensitivity analysis.

Multiple Time Point Reliability Calculations. Sometimes an analyst is interested in more than just success or failure of a system. For instance, system reliability at different times may be sought. The GO code provides a method for modeling multistate components such as switches. A switch can fail to open or close, or it can operate prematurely. A signal may arrive at a certain time during a sequence of events. These situations can be modeled by GO. The GO code provides the ability to define and quantify all the system states in one analysis run.

Common-Mode Failure Analysis. Although this is a difficult problem, SETS can identify single and double common-mode failures and generate descriptions of cut-sets affected by common-mode failures. This application is discussed in more detail in the previous section on common-mode failure methodology.

Future Work

The comparison documented here will form the basis of a broader effort that will integrate the three reliability/fault tree analysis methodologies. The result will be a generalized system reliability/fault tree analysis technology. A basic instruction text for applying this technology to various problems such as those outlined here will also be developed.

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Section 5

ANTICIPATED TRANSIENTS WITHOUT SCRAM

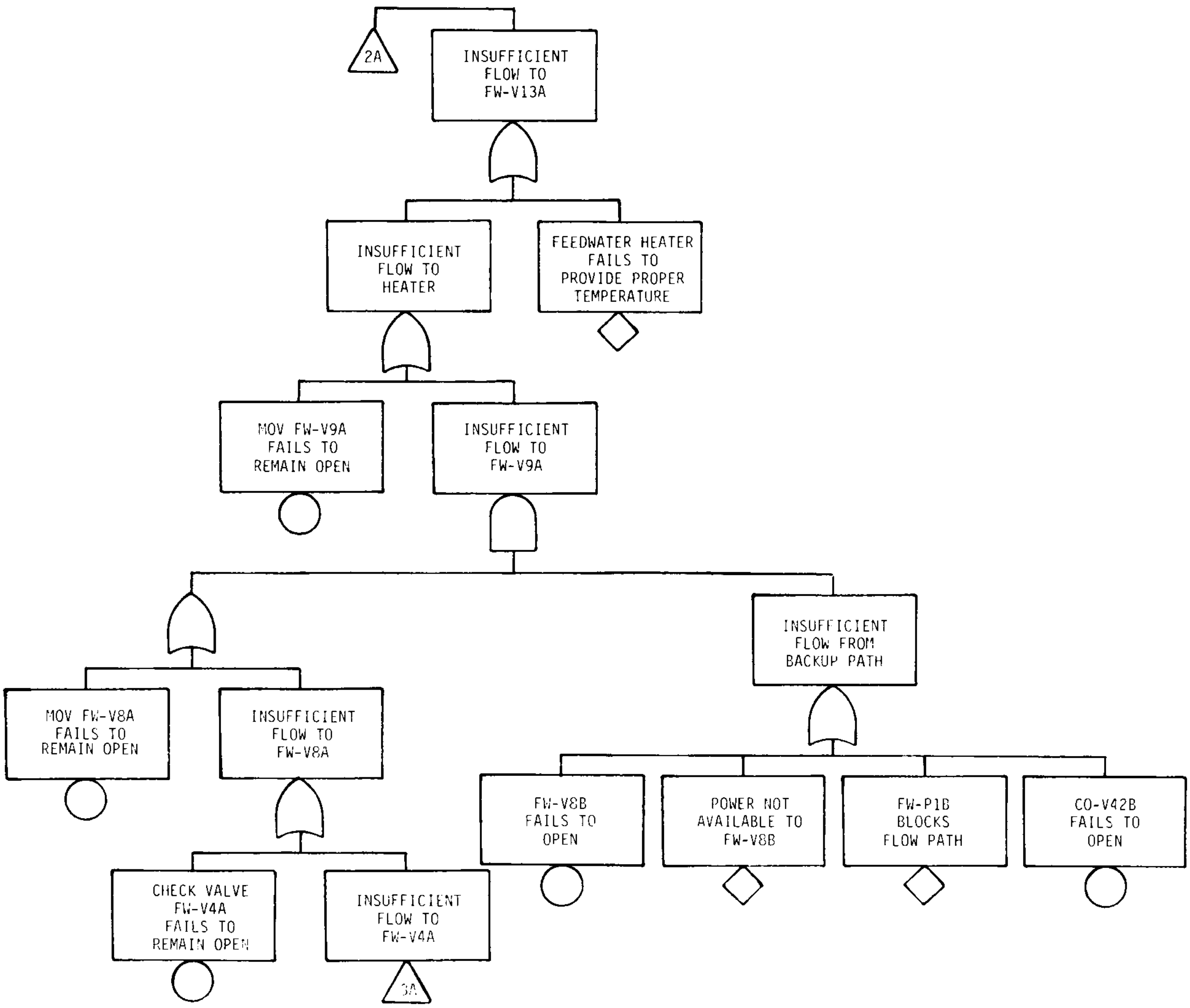
A Nuclear Regulatory Commission (NRC) Staff Report on anticipated transients without scram (ATWS), NUREG-0460 (5-1), was issued in December of 1978. This report suggested that the number of reactor coolant system (RCS) relief valves should be increased for pressurized water reactors (PWRs). This subject was subsequently discussed in several EPRI presentations by G.S. Lellouche to the NRC Advisory Committee on Reactor Safety (ACRS). To provide background material for this presentation, an analysis was performed of the reactor protection system (RPS) accident sequences in the Reactor Safety Study (RSS) (5-2) that involved RCS relief and safety valves. In addition, a reevaluation of the PWR reactor protection system fault tree in the RSS was conducted in light of recently updated component failure data.

RPS ACCIDENT SEQUENCE ANALYSIS

The PWR transient event tree shown in the RSS (5-3) is reproduced in Figure 5-1. As indicated in the figure, there are 12 core melt accident sequences. Four of these accident sequences imply successful RPS operation, while another, TKML, does not involve failure of the safety and relief valves. These five sequences were therefore not of interest to this analysis. The seven remaining sequences are as follows:

TKQ	TKP
TKQU	TKMP
TKMQ	TKMLP
TKMQU	

Sequence TKQU involves one more system failure than TKQ. There was no interest for this analysis in the success or failure of U. Therefore, the left column of the sequences listed above was limited to TKQ and TKMQ. A similar argument was used to drop the TKMLP sequence from the right column. Thus the four sequences investigated were: TKP, TKMP, TKQ, and TKMQ.



CM = Core Melt Sequences

Figure 5-1. Figure I 4-14 From The RSS. PWR Transient Event Tree.

Definitions of Accident Sequence Components

The letter T represents the number of anticipated transients per reactor-year, which the RSS estimated to be ten (5-4). (Three of these transients are related to main feedwater interruption, indicated by the two sequences with M in them.)

The letter K represents the reactor protection system (RPS). The RPS serves to trip the reactor control rods and terminate core power. The RSS assessed RPS unavailability at 3.6×10^{-5} per demand (5-4).

The letter M represents those portions of the power conversion system that provide main feedwater to the steam generators. The RSS assessed the probability of main feedwater interruption at three events per year (5-4).

The letter P represents a failure of RCS safety and relief valves to open. The operation of the safety and relief valves serves to limit RCS pressure levels. The valves are designed to open when RCS pressure exceeds preset levels. Failure of one or more of the three pressurizer safety valves could significantly increase RCS overpressure, thus increasing the likelihood of a RCS rupture. The failure of one of the three pressurizer safety valves to open was assessed in the RSS as 3×10^{-5} per demand (5-4).

The letter Q represents failure of these same valves to close after the RCS pressure level returns to below the valve set pressure. In the PWR, if the valves fail to reclose, they provide a path for coolant loss, causing a small RCS loss-of-coolant accident (LOCA). The RSS assessed this event at 1×10^{-2} per demand (5-4).

Sequence Evaluation

NUREG-0460 (5-1) suggested that more relief valves should be added to PWRs. The purpose of this analysis was to evaluate the increase or decrease in risk due to the addition of more valves. To establish a probabilistic line, the four sequences of interest were evaluated in terms of the RSS data presented in the definitions above, as follows:

$$\overset{T}{(7)}\ \overset{K}{(3.6\times10^{-5})}\ (1\times10^{-5} + \overset{P}{1\times10^{-5}} + 1\times10^{-5}) = 7.56\times10^{-9} \text{ per year} \quad (5\text{-}1)$$

$$\overset{TM}{(3)}\ \overset{K}{(3.6\times10^{-5})}\ (1\times10^{-5} + \overset{P}{1\times10^{-5}} + 1\times10^{-5}) = 3.24\times10^{-9} \text{ per year} \quad (5\text{-}2)$$

$$\overset{T}{(7)}\ \overset{K}{(3.6\times10^{-5})}\ (3.34\times10^{-3} + \overset{Q}{3.34\times10^{-3}} + 3.34\times10^{-3}) = 2.5\times10^{-6} \text{ per year} \quad (5\text{-}3)$$

$$\overset{TM}{(3)}\ \overset{K}{(3.6\times10^{-5})}\ (3.34\times10^{-3} + \overset{Q}{3.34\times10^{-3}} + 3.34\times10^{-3}) = 1.1\times10^{-6} \text{ per year} \quad (5\text{-}4)$$

If one more safety relief valve were to be added to the system, the success requirements would remain the same. Three valves must open (TKP sequences); however, all four must close (TKQ sequences). With this in mind, P was reevaluated as follows:

$$P = 6(1 \times 10^{-5})\ (1 \times 10^{-5}) = 6 \times 10^{-10} \quad (5\text{-}5)$$

instead of 3×10^{-5} (four things taken two at a time result in six combinations), and

$$Q = 4(3.34 \times 10^{-3}) = 1.34 \times 10^{-2} \quad (5\text{-}6)$$

instead of 1×10^{-2}. Incorporating these values into the accident sequences yields the following:

$$\overset{T}{(7)}\ \overset{K}{(3.6\times10^{-5})}\ \overset{P}{(6\times10^{-10})} = 1.5\times10^{-13} \text{ per year} \quad (5\text{-}7)$$

$$\overset{TM}{(3)}\ \overset{K}{(3.6\times10^{-5})}\ \overset{P}{(6\times10^{-10})} = 6.48\times10^{-14} \text{ per year} \quad (5\text{-}8)$$

$$\overset{T}{(7)}\ \overset{K}{(3.6\times10^{-5})}\ \overset{Q}{(1.34\times10^{-2})} = 3.4\times10^{-6} \text{ per year} \quad (5\text{-}9)$$

$$\overset{TM}{(3)}\ \overset{K}{(3.6\times10^{-5})}\ \overset{Q}{(1.34\times10^{-2})} = 1.4\times10^{-6} \text{ per year} \quad (5\text{-}10)$$

The accident sequences involving P decrease by a factor of 2×10^{-5} per year. However, those involving Q increase by a factor of 1.3.

With the addition of two more valves, success for P will be three out of five valves opening, and for Q all five must close. Thus,

$$P = 10(1 \times 10^{-5})^3 = 1 \times 10^{-14} \tag{5-11}$$

(five things taken two at a time result in ten combinations), and

$$Q = 5(3.34 \times 10^{-3}) = 1.67 \times 10^{-2} \tag{5-12}$$

Incorporating these values into the accident sequences yields the following:

$$\overset{T}{(7)}\ \overset{K}{(3.6\text{x}10^{-5})}\ \overset{P}{(1\text{x}10^{-14})} = 2.5\text{x}10^{-18}\text{ per year} \tag{5-13}$$

$$\overset{TM}{(3)}\ \overset{K}{(3.6\text{x}10^{-5})}\ \overset{P}{(1\text{x}10^{-14})} = 1.1\text{x}10^{-18}\text{ per year} \tag{5-14}$$

$$\overset{T}{(7)}\ \overset{K}{(3.6\text{x}10^{-5})}\ \overset{Q}{(1.67\text{x}10^{-2})} = 4.21\text{x}10^{-6}\text{ per year} \tag{5-15}$$

$$\overset{TM}{(3)}\ \overset{K}{(3.6\text{x}10^{-5})}\ \overset{Q}{(1.67\text{x}10^{-2})} = 1.8\text{x}10^{-6}\text{ per year} \tag{5-16}$$

Again, the accident sequences involving P decrease by a factor of 3.3×10^{-10} per year, as compared to the base case. Those involving Q increase by a factor of 1.68 as compared to the base case.

The RSS concluded that the accident sequences containing P should be discarded since their probabilities were negligible in comparison to those for other accident sequences (5-4). In this analysis the process of adding valves to drive these probabilities down further resulted in the probabilities for sequences containing Q being simultaneously driven up. The two accident sequences containing Q were, however, retained in the RSS. When the containment failures were included in this analysis, they accounted for five out of the twelve PWR transient-dominant accident sequences. The basic TKQ and TKMQ accident sequences in themselves lead to a core melt; in addition, they are equivalent to a small-small LOCA. The RSS did not evaluate them in this fashion because their probabilities were in the range of 1×10^{-6} per year.

The RSS found that a random pipe break causing a small-small LOCA has a probability of 1×10^{-3} per year, which is three orders of magnitude more likely than TKQ (5-4).

The worth of these accident sequences in terms of risk is reflected in the risk curve in Figure 5-2. Accident sequences containing RPS failure account for about 0.3% of the risk.

REEVALUATION OF REACTOR PROTECTION SYSTEM (RPS) FAULT TREE

Figure 5-3 shows the reduced PWR RPS fault tree used as the basis for a previously published EPRI evaluaton (5-5) and the original RSS evaluation (5-6). This fault tree was reevaluated for the ACRS presentation to include additional data regarding failures of some of the components.

Updated Component Data

Data searches conducted by the Nuclear Safety Information Center found all failures in the RPS circuit breakers and SCRAM logic. The results for circuit breakers (search made through 1976) were as follows:

- Closed, fail to open
  - 3 failures at Robinson 2 in October 1971
  - 1 failure at Robinson 2 in December 1973
- Bypass breaker open, fails by closing - no failures

The SCRAM logic results (search made through 1977) revealed that there were no failures of the command for the breakers to open.

To determine the number of opportunities for these events to occur required three data items:

- Number of days each PWR plant has been in operation since commercial startup, listed in Table 5-1
- Number of scrams per year, calculated as approximately 8.8 from a previous report (5-7)
- Frequency of SCRAM system (logic and breakers) testing, which is estimated at once per month

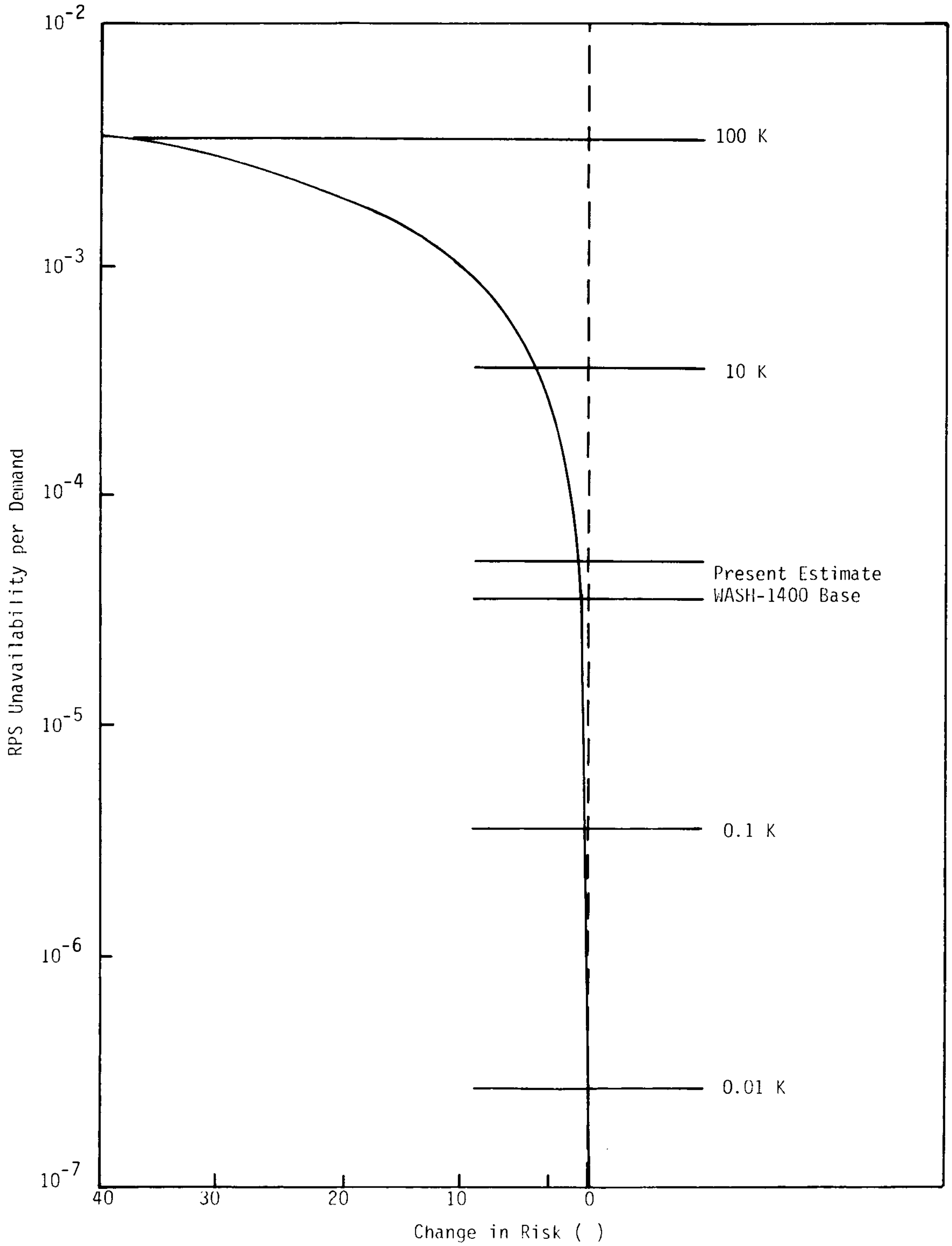


Figure 5-2. % Change in Risk With Change in RPS Unavailability
Source: EPRI NP-265 (5-5)

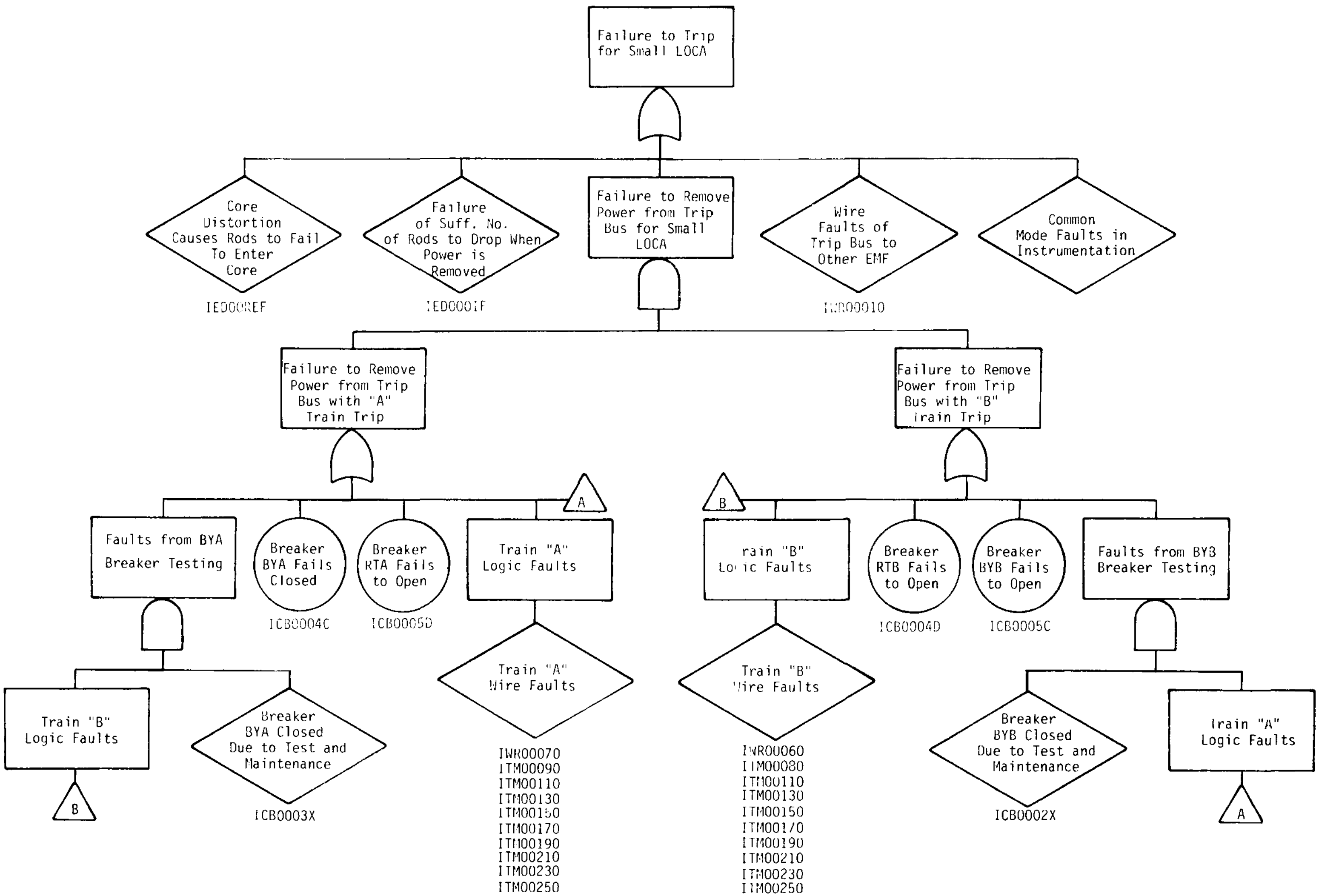


Figure 5-3. Reactor Protection System Reduced Fault Tree
Source: RSS, Appendix II (5-6)

Table 5-1

PWR PLANT-DAYS

Plant	Date Began Commercial Operation	Days From Commercial Operation to 1/1/77	Days From Commercial Operation to 1/1/78
Arkansas 1	12/19/74	744	1109
Beaver Valley 1	10/1/76	92	457
Calvert Cliffs 2	5/8/75	604	969
Calvert Cliffs 2	4/1/77	--	275
Cook 1	8/27/75	493	858
Crystal River 3	3/13/77	--	294
Davis-Besse 1	11/20/77	--	42
Farley 1	12/1/77	--	31
Ft. Calhoun	9/26/73	1193	1558
Ginna	7/15/70	2362	2727
Haddam Neck	1/1/68	3288	3653
Indian Pt. 1	10/62	4399*	4399*
Indian Pt. 2	8/73	1235	1600
Indian Pt. 3	8/30/76	124	489
Kewaunee	6/74	931	1296
Maine Yankee	12/28/72	1465	1830
Millstone	12/26/75	372	737
Oconee 1	7/15/73	1266	1631
Oconee 2	9/9/74	845	1210
Oconee 3	12/16/74	747	1112
Palisades	12/31/71	1828	2193
Point Beach 1	12/21/70	2203	2568
Point Beach 2	10/1/72	1553	1918

*Plant not operating since 10/31/74

Table 5-1 (continued)

Plant	Date Began Commercial Operation	Days From Commercial Operation to 1/1/77	Days From Commercial Operation to 1/1/78
Prairie Island 1	12/16/73	1112	1477
Prairie Island 2	12/21/74	742	1107
Rancho Seco	4/17/75	625	990
Robinson 2	3/7/71	2127	2492
Salem 1	6/30/77	--	185
San Onofre 1	1/1/68	3288	3653
St. Lucie 1	12/21/76	11	376
Surry 1	12/22/72	1471	1836
Surry 2	5/1/73	1341	1706
Three Mile Island 1	9/2/74	852	1217
Trojan	6/20/76	226	591
Turkey Point 3	12/14/72	1479	1844
Turkey Point 4	9/7/73	1212	1577
Yankee Rowe	7/61	5649	6014
Zion 1	12/31/73	1097	1462
Zion 2	9/17/74	837	1202
TOTAL DAYS		47,813	60,685

Thus, for the SCRAM circuit breakers "closed, fail to open," the number of opportunities to fail was estimated by:

$$N = \Bigg[(47813 \text{ plant days}) \left(\frac{1 \text{ year}}{365 \text{ plant days}}\right) \cdot \left(8.8 \, \frac{\text{SCRAMs}}{\text{year}}\right) + (47813 \text{ plant days}) \left(\frac{1 \text{ month}}{30 \text{ plant days}}\right) \left(1 \frac{\text{test}}{\text{month}}\right)\Bigg] \times (2 \text{ breakers per plant}) \tag{5-17}$$

$$= \left[\left(\frac{47813}{365}\right) 8.8 + \left(\frac{47813}{30}\right)\right] (2) = 5493$$

For the SCRAM bypass breakers "open, fails by closing," the number of opportunities was the same for the circuit breakers.

Since SCRAM logic data cover an additional year, the,number of opportunities to fail was evaluated as:

$$N = \Bigg[(60685 \text{ plant days}) \left(\frac{1 \text{ year}}{365 \text{ plant days}}\right) \left(8.8 \, \frac{\text{SCRAMs}}{\text{year}}\right) + (60685 \text{ plant days}) \left(\frac{1 \text{ month}}{30 \text{ plant days}}\right) \left(1 \frac{\text{test}}{\text{month}}\right)\Bigg] \times (2 \text{ logic trains per plant}) \tag{5-18}$$

$$= 6971$$

Table 5-2 presents a summary of the data used in the fault tree analysis.

Table 5-2

SUMMARY OF DATA USED IN ANALYSIS
OF RPS FAULT TREE

Event	Number of Occurrences	Number of Opportunities
SCRAM breakers		
Closed, fail to open	4	5493
Open, fail to close	0	5493
SCRAM logic fails	0	6971

Results of Fault Tree Analysis

Since the data obtained above cover entire logic fault, a single component can replace the list of Train "A" and Train "B" wire faults previously used to estimate logic train failure on demand. Table 5-3 compares the results with those from the Reactor Safety Study (5-6) and the previous EPRI analysis (5-5). The new results represent the best estimate of PWR RPS unavailability on demand, based on the latest available plant data.

Table 5-3
COMPARISON OF RPS FAULT TREE EVALUATIONS

	WASH-1400	EPRI NP-265	THIS EVALUATION
Mean		6.4×10^{-5}	5.1×10^{-6}
Median	3.6×10^{-5}	5.1×10^{-5}	4.2×10^{-6}
95 Percentile	1.0×10^{-4}	1.5×10^{-4}	1.1×10^{-5}
5 Percentile	1.3×10^{-5}	1.9×10^{-5}	1.7×10^{-6}

REFERENCES

5-1 Anticipated Transients Without Scram for Light Water Reactors. U.S. Nuclear Regulatory Commission, December 1978. NUREG-0460.

5-2 Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. U.S. Nuclear Regulatory Commission, October 1975. NUREG-75/014 (WASH-1400).

5-3 Reactor Safety Study, Appendix I, Accident Definitions and Use of Event Trees. U.S. Nuclear Regulatory Commission, October 1975. NUREG-75/014 (WASH-1400).

5-4 Reactor Safety Study, Appendix V, Quantitative Results of Accident Sequences. U.S. Nuclear Regulatory Commission, October 1975. NUREG-75/014 (WASH-1400).

5-5 ATWS: A Reappraisal, Part II, Evaluation of Societal Risks Due to Reactor Protection System Failure. Vol. 3, PWR Risk Analysis. Palo Alto, Calif.: Electric Power Research Institute, August 1976. EPRI NP-265.

5-6 Reactor Safety Study, Appendix II, Fault Trees. U.S. Nuclear Regulatory Commission, October 1975. NUREG-75/014 (WASH-1400).

5-7 ATWS: A Reappraisal, Part III, Frequency of Anticipated Transients. Palo Alto, Calif.: Electric Power Research Institute, July 1978. EPRI NP-801.

Section 6

SENSITIVITY STUDIES

The designs for commercial nuclear power plants in the United States presently include engineered safety features (ESF) intended to mitigate the consequences of a loss-of-coolant accident (LOCA). A sensitivity study was initiated during the past year to determine the value of such ESF systems in terms of economic and risk reduction factors. The first phase of this effort has focused on ESF for pressurized water reactors (PWRs). A parallel study was initiated later in the year for boiling water reactors (BWRs).

A separate sensitivity study focusing on the potential of a vapor explosion occurring in conjunction with a postulated core melt was also conducted. This study provided data for a paper presented at a meeting on fuel-coolant interaction in nuclear reactor safety.

PWR SENSITIVITY STUDY

The general approach taken in this study was to develop a model which would include all the core melt accident sequences defined in the Reactor Safety Study (RSS) (6-1). These accident sequences were then categorized according to the release categories specified in the RSS. The model was quantified assuming that various engineered safety features did not exist. Once determined, the release category probabilities were input to the Calculation of Reactor Accident Consequences (CRAC) code (6-2) in order to determine the consequences.

The resulting consequences reported here were those that can be related to the cost-benefit ratio defined in Appendix I of 10CFR Part 50 ($1,000 per total man-rem) (6-3). On an economic as well as risk reduction basis, any change in these consequences resulting from the addition of an ESF was then evaluated.

PWR Accident Sequence Model

Before performing the sensitivity study, a model was developed using fault tree methodology. For each core melt release catgory defined by the RSS, a fault tree was constructed which ORed all the accident sequences assigned to that category. This procedure included both dominant and nondominant accident sequences. Within a given accident sequence, the initiating event, the system failures and successes, and the containment failure modes were ANDed together. Thus, each initiating event, system failure or success, and containment failure mode was made equivalent to a component input in the fault tree.

The potential accident sequences considered resulted from the following initiators:

- Vessel rupture (R)
- Large LOCA (A)
- Small LOCA (S1)
- Small-small LOCA (S2)
- Transient (T)

Effects of System Success. System successes, as well as system failures, were included in the accident sequences. This procedure was necessary in order to compensate for the fact that, at various stages of the study, several engineered safety features were assumed not to exist. For example, the RSS assigned accident sequence AD-α to release category 3. However, accident sequence ACD-α was assigned to release category 1. The only difference between the two is that containment spray injection system C has been considered failed in ACD-α. The fact that C was considered operative in AD-α caused this sequence to move from release category 1 to release category 3. Including the containment spray success in the AD-α accident sequence caused this sequence to be eliminated once it was assumed that the containment spray did not exist. Thus the same accident sequence was not included twice in the analysis.

Hot and Cold Releases. Although the models were developed for release categories 1 through 7, it was necessary to subdivide release category 1 into 1A, cold release, and 1B, hot release. The results to be compared were the accumulative consequences 50 miles from the site. It was possible that the difference between a cold release and a hot release at this distance could affect the results. The RSS, in defining release category 1, discussed these two types of releases.

For release category 1B, it was assumed that the steam explosion would rupture the upper portion of the reactor vessel and breach the containment barrier, with the result that a substantial amount of radioactivity might be released from the containment in a puff over a period of about ten minutes. Due to the sweeping action of gases generated during containment-vessel meltthrough, the release of radioactive materials would continue at a relatively low rate thereafter. Because the containment would contain hot pressurized gases at the time of failure, a relatively high release rate of sensible energy from the containment could be associated with this scenario.

For release category 1A, the RSS considered potential accident sequences that would involve the occurrence of core melting and a steam explosion after containment rupture due to overpressure. The rate of release would be lower, although still relatively high.

The high-energy release (category 1B) would result in smaller consequences near the plant as compared to the low-energy release (category 1A) (6-2). This comparison is illustrated in Figure 6-1 (Figure VI 13-26 in the RSS, repeated here for clarity).

Cases Considered. Once the models were completed and verified, six cases were quantified:

- Case I — No containment, no emergency core cooling system (ECCS), and no containment post-accident heat or radioactivity removal systems
- Case IA — ECCS, but no containment or containment functions
- Case II — Containment, but no ECCS or containment functions
- Case IIIA — Containment and containment functions, but no ECCS
- Case IIIB — Containment and ECCS, but no containment functions
- Case IV — All systems considered in the RSS

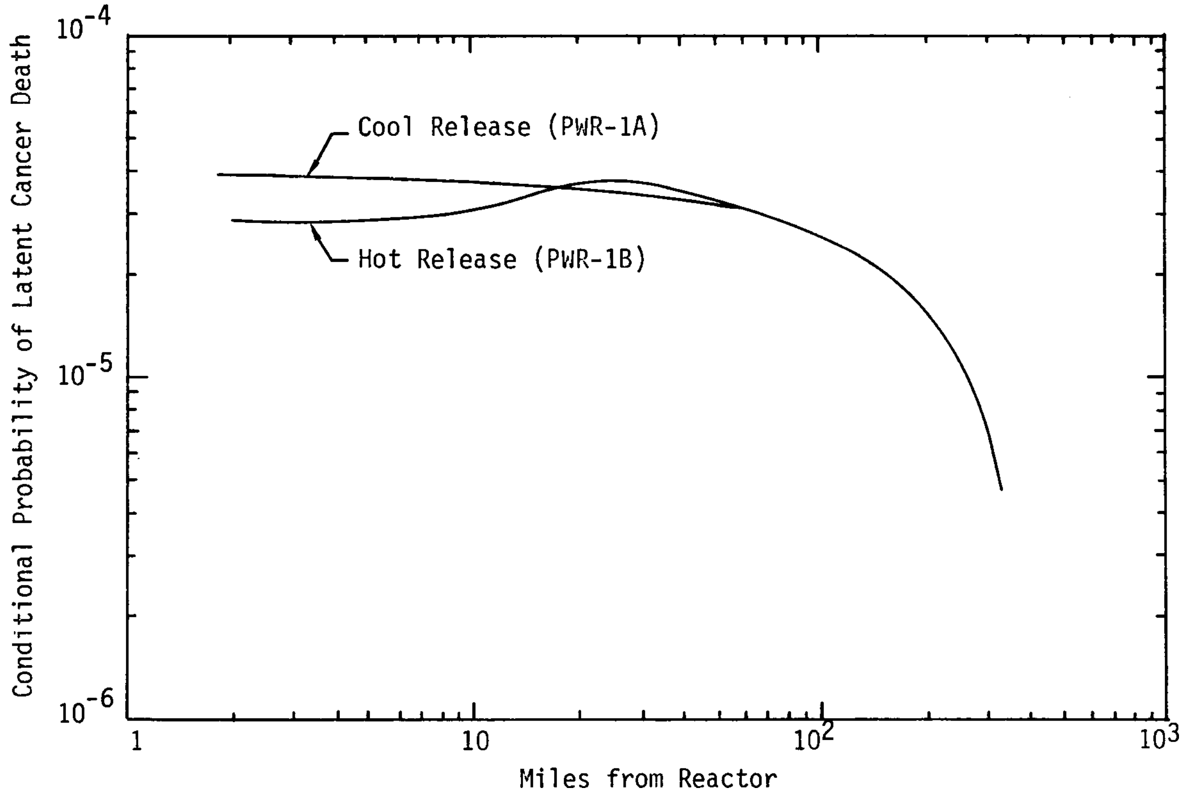


Figure 6-1. Figure VI 13-26 from RSS (5-2). Conditional Probability of Latent Cancer Death Given a Category 1A or 1B Release (Absolute Mortality Probabilities are Approximately 10^{-6} Per Reactor-year Stated Times One.)

For these cases three consequence-mitigating functions were considered:

- Containment
- Containment functions
- ECCS

Containment was defined as the building surrounding the primary system of the pressure vessel, steam generators, pressurizers, and circulation pumps. The containment functions of post-accident heat and radioactivity removal were defined as the containment spray injection and recirculation systems, including the heat exchangers in the containment spray recirculation system used for containment heat removal. The ECCS was defined as the proper combination of high- or low-pressure injection subsystems and accumulators to reflood the core for the various sized LOCAs.

The method used to assume that a system would not exist was to set its probability of failure or unavailability equal to one. The exceptions were Cases I and IA, where no containment was assumed to exist.

Containment Rupture Probability. The RSS defined five failure modes for containment:

- Vessel steam explosion resulting in containment rupture (α).
- Containment failure resulting from inadequate isolation of containment openings and penetrations (β)
- Containment failure due to hydrogen burning (γ)
- Containment failure due to overpressure (δ)
- Containment vessel meltthrough (ε)

In the RSS α was evaluated in the following manner:

> When conditions exist which will lead to a core melt, 1/10 of the time a vessel steam explosion occurs and 1/10 of these times the vessel steam explosion induces a containment failure. Thus the probability of alpha was determined to be 0.01. (6-4)

In this study, alpha was assigned the probability of the vessel steam explosion (0.1) when containment was assumed not to exist. The remaining 0.9 probability of containment failure was assigned to beta (β) (containment leakage). The other failure modes of containment defined in the RSS were assumed not to exist for this study's case where containment does not exist.

When one of the three consequence-mitigating functions was assumed to exist, the unavailabilities defined in the RSS for the systems that make up those functions were applied.

Probabilistic Results

The results of the quantifications for each of the six cases studied are shown in Table 6-1. In Case I, where it was assumed that there are no containment, no vessel functions, and no containment functions, the probability of a core melt is dominated by the LOCA accident sequences. The probability is

Table 6-1

RELEASE CATEGORY PROBABILITIES FOR
PWR SENSITIVITY STUDY

CASE NUMBER	INITIATING EVENTS	RELEASE CAT. 1A (COLD)	RELEASE CAT. 1B (HOT)	RELEASE CAT. 2	RELEASE CAT. 3	RELEASE CAT. 4	RELEASE CAT. 5	RELEASE CAT. 6	RELEASE CAT. 7	SYSTEM ASSUMPTION
CASE I	R	$1.00x10^{-8}$	0	$9.00x10^{-8}$	0	0	0	0	0	No Containment
	A	$1.00x10^{-5}$	0	$9.00x10^{-5}$	0	0	0	0	0	
	S1	$3.00x10^{-5}$	0	$2.70x10^{-4}$	0	0	0	0	0	No Vessel Functions
	S2	$1.00x10^{-4}$	0	$9.00x10^{-4}$	0	0	0	0	0	
	T	$1.07x10^{-6}$	0	$9.63x10^{-6}$	0	0	0	0	0	No Containment Functions
	TOTAL	$1.41x10^{-4}$	0	$1.27x10^{-3}$	0	0	0	0	0	
CASE IA	R	$1.00x10^{-8}$	0	$9.00x10^{-8}$	0	0	0	0	0	No Containment
	A	$5.71x10^{-8}$	0	$5.14x10^{-7}$	0	0	0	0	0	
	S1	$2.85x10^{-7}$	0	$2.57x10^{-6}$	0	0	0	0	0	Vessel Functions
	S2	$8.65x10^{-7}$	0	$7.81x10^{-6}$	0	0	0	0	0	
	T	$1.07x10^{-6}$	0	$9.63x10^{-6}$	0	0	0	0	0	No Containment Functions
	TOTAL	$2.29x19^{-6}$	0	$2.06x10^{-5}$	0	0	0	0	0	
CASE II	R	0	$1.00x10^{-9}$	$9.90x10^{-8}$	0	0	0	0	0	Containment
	A	0	$1.00x10^{-6}$	$1.62x10^{-5}$	$1.00x10^{-6}$	0	0	$8.20x10^{-5}$	0	
	S1	0	$3.00x10^{-6}$	$4.86x10^{-5}$	$3.00x10^{-6}$	0	0	$2.46x10^{-4}$	0	No Vessel Functions
	S2	$1.00x10^{-5}$	0	$1.62x10^{-4}$	$1.00x10^{-5}$	0	0	$8.20x10^{-4}$	0	
	T	0	$1.07x10^{-7}$	$8.56x10^{-6}$	0	0	0	$2.03x10^{-6}$	0	No Containment Functions
	TOTAL	$1.00x10^{-5}$	$4.10x10^{-6}$	$2.35x10^{-4}$	$1.40x10^{-5}$	0	0	$1.15x10^{-3}$	0	

Table 6-1 (cont.)

CASE NUMBER	INITIATING EVENTS	RELEASE CAT. 1A (COLD)	RELEASE CAT. 1B (HOT)	RELEASE CAT. 2	RELEASE CAT. 3	RELEASE CAT. 4	RELEASE CAT. 5	RELEASE CAT. 6	RELEASE CAT. 7	SYSTEM ASSUMPTION
CASE IIIA	R	0	2.40×10^{-12}	2.38×10^{-10}	0	0	0	0	0	Containment
	A	0	4.81×10^{-9}	1.62×10^{-10}	1.00×10^{-6}	4.97×10^{-10}	1.99×10^{-7}	1.07×10^{-8}	9.88×10^{-5}	
	S1	0	7.20×10^{-9}	5.40×10^{-10}	3.00×10^{-6}	1.49×10^{-9}	5.98×10^{-7}	3.21×10^{-8}	2.97×10^{-4}	No Vessel Functions
	S2	2.40×10^{-8}	1.00×10^{-10}	1.18×10^{-6}	1.00×10^{-5}	1.70×10^{-10}	1.99×10^{-6}	2.07×10^{-6}	9.87×10^{-4}	
	T	0	3.02×10^{-8}	2.42×10^{-6}	7.70×10^{-8}	6.04×10^{-10}	1.54×10^{-9}	5.74×10^{-7}	7.62×10^{-6}	Containment Function
	TOTAL	2.40×10^{-8}	4.22×10^{-8}	3.60×10^{-6}	1.40×10^{-5}	2.70×10^{-9}	2.79×10^{-6}	2.69×10^{-6}	1.39×10^{-3}	
CASE IIIB	R	0	1.00×10^{-9}	9.90×10^{-8}	0	0	0	0	0	Containment
	A	0	5.71×10^{-8}	7.01×10^{-8}	9.94×10^{-5}	0	0	4.59×10^{-7}	0	
	S1	2.97×10^{-6}	2.85×10^{-8}	2.24×10^{-9}	2.94×10^{-4}	0	0	1.38×10^{-6}	0	Vessel Functions
	S2	1.00×10^{-5}	0	4.25×10^{-6}	9.81×10^{-4}	0	0	4.59×10^{-6}	0	
	T	0	1.07×10^{-7}	8.56×10^{-6}	0	0	0	2.03×10^{-6}	0	No Containment Functions
	TOTAL	1.30×10^{-5}	1.94×10^{-7}	1.30×10^{-5}	1.37×10^{-3}	0	0	8.48×10^{-6}	0	
CASE IV	R	0	2.41×10^{-12}	2.35×10^{-10}	1.00×10^{-9}	0	0	0	9.90×10^{-8}	Containment
	A	1.84×10^{-10}	2.34×10^{-11}	1.88×10^{-10}	3.68×10^{-8}	2.83×10^{-12}	3.70×10^{-9}	9.76×10^{-10}	1.84×10^{-6}	
	S1	5.50×10^{-10}	9.84×10^{-11}	5.18×10^{-10}	1.21×10^{-7}	1.47×10^{-11}	1.34×10^{-8}	2.94×10^{-9}	6.62×10^{-6}	Vessel Functions
	S2	2.58×10^{-8}	1.00×10^{-10}	1.20×10^{-8}	2.74×10^{-6}	3.13×10^{-12}	4.28×10^{-8}	6.90×10^{-8}	2.13×10^{-5}	
	T	0	3.02×10^{-8}	2.42×10^{-6}	7.70×10^{-8}	6.04×10^{-10}	1.54×10^{-9}	5.74×10^{-7}	7.62×10^{-6}	Containment Function
	TOTAL	2.65×10^{-8}	3.04×10^{-8}	2.44×10^{-6}	2.98×10^{-6}	6.25×10^{-10}	6.14×10^{-8}	6.46×10^{-8}	3.74×10^{-5}	

clustered in release categories 1A and 2. This clustering is due primarily to the fact that, if a LOCA occurs, the core will rapidly melt and there will be a large release of the core inventory, with no decontamination factors assumed.

In Case IA, where the ECCS was assumed to exist, the probability of a core melt accident sequence was reduced by almost two orders of magnitude in both release categories 1A and 2. There is a radioactivity gap release (release category 1B) from the fuel rods, but its risk can be assumed to be quite small in comparison to the risk from category 1A for this case. The reduction in probability for both release categories 1A and 2 is due to the fact that the ECCS is assumed to exist and to be capable of preventing core melt.

Release category 8 (a noncore-melt category) increases in probability as categories 1A and 2 decrease. There is some doubt regarding the meaning or significance of this. The event trees from which the accident sequences were formed imply a specific order of systems and their interrelationshps. When an ECCS is assumed to exist before the existence of a containment and a containment spray injection system, the specific order of the event trees has been altered. Accident sequences in less severe core melt release categories may have been forced out of existence as a result of the assumption of no containment and no containment functions. Also, it was assumed for this study that an ECCS with no containment would be accompanied by either an adequate supply of water or a heat removal capability.

In Case II, where the containment was added, the probability of vessel rupture, large LOCA, and small LOCA shifts from release category 1A to release category 1B. The addition of containment has therefore increased the probability of a hot release. In release categories 1A and 1B, the 0.1 probability of containment failure resulting from a vessel steam explosion is obvious from the order of magnitude difference in probabilities (see Table 6-1). The addition of containment has caused nearly an order-of-magnitude drop in release category 2. Probabilities begin to appear in categories 3 and 6 due to the addition of other containment failure modes, such as hydrogen burning, overpressure, and meltthrough, to vessel steam explosion and containment leakage. The total probability remains equal to that of Case I. With the assumed lack of systems to reflood the core and remove heat from containment, a high-risk release is still dominant.

In Case IIIA, with the addition of the containment functions (containment spray injection, recirculation, and heat removal), there is a considerable drop in the LOCA-initiated accident sequence probabilities for categories 1A, 1B, and 2. This drop in probability has actually been shifted to the lesser-consequence release categories, as expected. The total probability of a core melt has remained the same at 1.41×10^{-3}, which is even more indicative of the shifting to lesser-consequence release categories.

In Case IIIB, the ECCS has been added to the containment, but there are no containment functions. The lack of post-accident heat removal for the containment causes a shifting of accident sequence probabilities from those of Cases II and IIIA to higher-consequence categories, while total core melt probability remains the same for these cases. Sequences contributing to melt-through accidents in categories 6 and 7 are replaced by overpressure failures in category 3 and overpressure followed by a steam explosion accident characterized by a cold release in category 1A. This shift is caused by a failure to control the pressure transient within containment following a LOCA-initiated accident, resulting in containment failure, ECCS pump cavitation, and eventual core melt. Case IIIB, like Case IA, involves the addition of systems in an order that deviates from the order prescribed by the event trees. As a result, the correctness of this case should be verified before conclusions are drawn.

Case IV includes the addition of the sets of systems from both Cases IIIA and IIIB, bringing the plant back to the configuration analyzed in the RSS. Thus this case contains the combined results of Cases IIIA and IIIB. Comparing Case IV to Case II, there is a considerable drop in the probabilities of release categories 1 and 2, with a shifting to category 3. Category 4 has dropped, with an increase in category 5, and so on. Generally, there is a shifting of probabilities to lesser-consequence categories. The total probability of a core melt has dropped to 3.75×10^{-5}, compared to 1.41×10^{-3} in Case I. Note that this drop is for core melt release categories only. Had the noncore-melt release categories been included in Table 6-1, the shifting would be seen to continue into the realm of the noncore-melt release categories.

Cases IA and IIIB were included in the probabilistic results because of their potential significance and applicability to a plant that removes heat by both a containment cooling system and an emergency core recirculation system. This design prevents some of the system interrelationships unique to the Surry plant analyzed in the RSS and therefore defined by the base event trees used in this study.

Consequences

Once the probabilities for the release categories were determined, they were input to the CRAC code (6-2) in order to calculate the consequences involved in each of the six cases. The consequences calculated were total man-rem and total thyroid-rem within 50 miles of the site of the potential accident. These consequence data were evaluated specifically so that the economic conversion factors defined in 10CFR, Part 50, Appendix I (6-3) could be used for the final comparison of the six cases. These factors are described in 10CFR, Part 50, Appendix I as:

> ...the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit. (6-3)

It should be noted that there are some limitations to the consequence results presented here. These results are intended to be consistent with the above regulatory definition. If the consequences were to be calculated out to 500 miles from the site and the remainder of the inventory averaged out over the rest of the world for a 40-year period, the results would be much more severe. Furthermore, it is not clear if the two consequences considered, total man-rem and total thyroid-rem, would be truly representative of the dose factors associated with other consequences.

Table 6-2 shows the results of the CRAC code calculations for each of the six cases. There is a continuous drop in risk with the addition of safety systems. The exception is for Case IIIB. Compared with the results of Case I, those for Case IIIB show some improvement. However, evaluating the ECCS

Table 6-2

RESULTS OF CRAC CODE CALCULATIONS
FOR PWR SENSITIVITY STUDY

CONSEQUENCE	CASE I	CASE IA	CASE II	CASE IIIA	CASE IIIB	CASE IV
Total man-rem	1.7×10^4	2.3×10^3	1.9×10^3	2.48×10^2	7.54×10^3	3.39×10^1
Total thyroid	8.36×10^4	1.45×10^4	8.45×10^3	1.92×10^2	2.54×10^3	1.31×10^2

alone requires comparing Case IIIB with Case II, revealing an increase in risk. This increase in risk is suspect, however, in light of the fact that the model was forced to assume some systems in an order that was not consistent with the order of the event tree.

Future Work

The next phase of the PWR sensitivity study will involve a comparison of the consequences defined by the parameters indicated in 10CFR, Part 50, Appendix I (6-3) with the more severe consequences expected from a large high-energy release. This comparison will include some system and plant value-impact factors in addition to the value-impact of total man-rem. A report including this additional information will be released within the next year.

BWR ACCIDENT SEQUENCE MODEL

The BWR portion of the ESF sensitivity study began with a modeling effort similar to that undertaken for the PWR portion. The complete BWR event trees from the RSS (6-5) and the corresponding accident sequences (6-6) were classified according to the RSS release categories.

All the sequences were then expanded to include both system success and system failure for each function in each sequence. The following example gives the expansion of the large LOCA sequences in release category 1, which have containment failure modes α = 0.01. Table 6-3 shows the RSS nomenclature for these sequences, together with the expanded versions, which includes both system successes and system failures for each function.

Table 6-3

EXPANSION OF LARGE LOCA SEQUENCES
(α = .01) IN RSS RELEASE CATEGORY 1

WASH-1400 SEQUENCE	EXPANDED SEQUENCE
AE-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,E\,\overline{G}\ -\alpha$
AJ-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,\overline{G}\,\overline{H}\,\overline{I}\,J\ -\alpha$
AHI-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,\overline{G}\,H\,I\ -\alpha$
AI-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,\overline{G}\,\overline{H}\,I\ -\alpha$
ADF-α	$A\,\overline{B}\,\overline{C}\,D\,\overline{E}\,F\ -\alpha$
AHJ-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,\overline{G}\,H\,\overline{I}\,J\ -\alpha$
AGJ-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,G\,\overline{H}\,\overline{I}\,J\ -\alpha$
AGI-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,G\,\overline{H}\,I\ -\alpha$
AGHJ-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,G\,H\,\overline{I}\,J\ -\alpha$
AGHI-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,\overline{E}\,\overline{F}\,G\,H\,I\ -\alpha$
AEG-α	$A\,\overline{B}\,\overline{C}\,\overline{D}\,E\,G\ -\alpha$
ADJ-α	$A\,\overline{B}\,\overline{C}\,D\,\overline{E}\,\overline{F}\,\overline{H}\,\overline{I}\,J\ -\alpha$
ADI-α	$A\,\overline{B}\,\overline{C}\,D\,\overline{E}\,\overline{F}\,\overline{H}\,I\ -\alpha$
ADHI-α	$A\,\overline{B}\,\overline{C}\,D\,\overline{E}\,\overline{F}\,H\,I\ -\alpha$
ADE-α	$A\,\overline{B}\,\overline{C}\,D\,E\ -\alpha$

These expanded sequences were input to the SETS computer code and reduced to their simplest Boolean equivalents (prime implicants). The results for the sequences listed in Table 6-3 are shown in Table 6-4.

Table 6-4

REDUCED LARGE LOCA SEQUENCES

$A \overline{B} \overline{C} E - \alpha$

$A \overline{B} \overline{C} \overline{F} J - \alpha$

$A \overline{B} \overline{C} \overline{F} I - \alpha$

$A \overline{B} \overline{C} D J - \alpha$

$A \overline{B} \overline{C} D I - \alpha$

$A \overline{B} \overline{C} D F - \alpha$

This expansion/reduction procedure was followed for all event trees and accident sequences. The reduced sequences will be formulated as inputs to WAM-CUT for the numerical evaluation required for sensitivity analysis.

SENSITIVITY STUDY FOR POTENTIAL VAPOR EXPLOSION

A limited sensitivity study was undertaken to evaluate the contribution to societal risk from the probability of a vapor explosion associated with a postulated core melt. The data generated by this study contributed to a paper on fuel-coolant interactions (6-7).

This study was conducted with the PWR model developed for the ESF sensitivity study discussed previously. A preliminary model was used for the BWR calculations.

For this study the RSS probability of 0.01 of a vapor explosion leading to containment failure (6-4) was varied from 10^{-3} to unity. The results for the PWR case are illustrated in terms of man-rem per reactor-year and dollars per reactor-year in Figures 6-2 and 6-3 respectively. Results for the BWR case are shown in Figures 6-4 and 6-5.

The greatest relative change to total risk occurred at high probabilities (greater than 10^{-1}). For PWRs the results indicate that the probability of a vapor explosion must be greater than 0.1 in order to have an effect on societal risk. For BWRs the change in risk is somewhat less. It should be noted, however, that the BWR calculations were based on a preliminary, untested modeling effort.

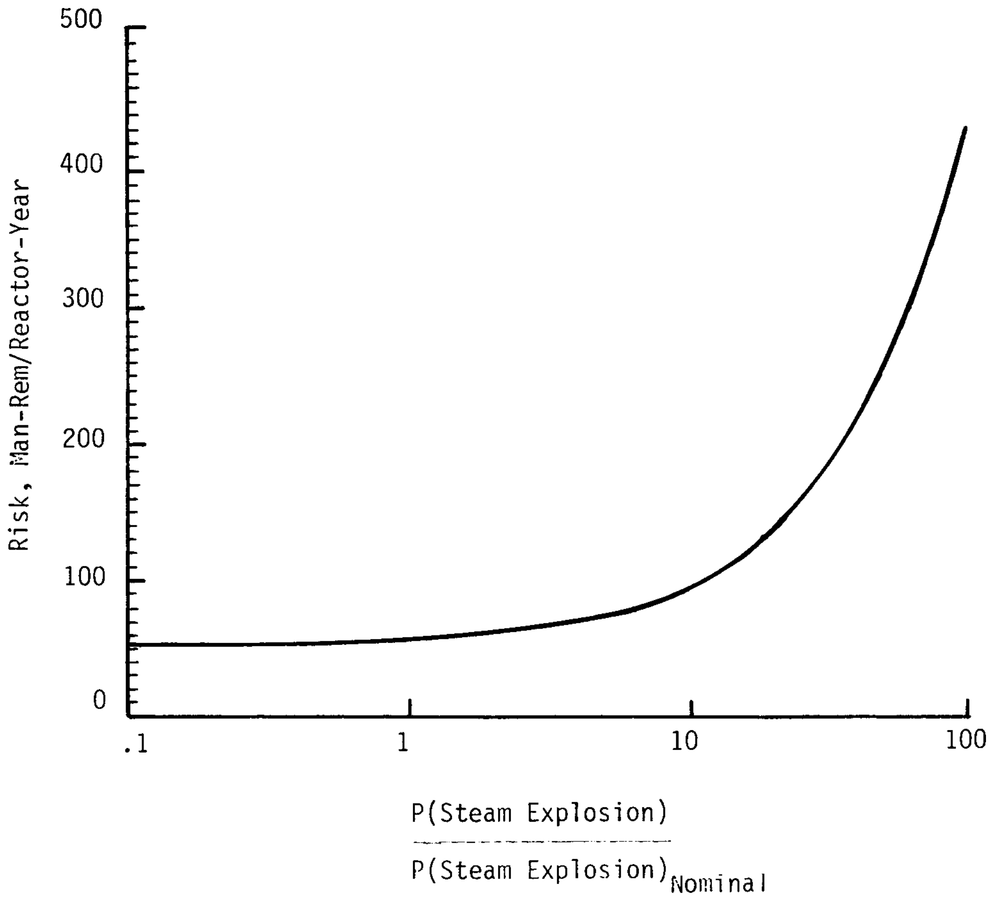


Figure 6-2. Sensitivity of PWR Risk in Man-Rem/Reactor-Year to the Probability of Steam Explosion

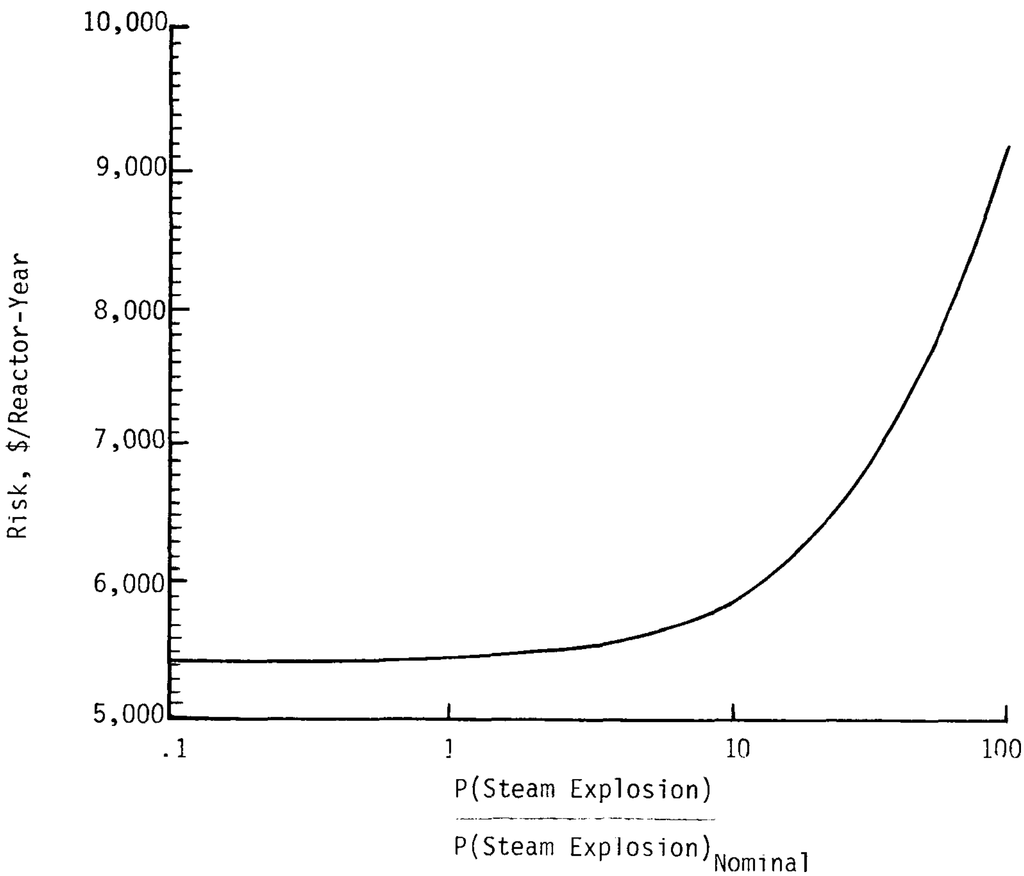


Figure 6-3. Sensitivity of PWR Risk in $/Reactor-Year to Changes in the Probability of Steam Explosion

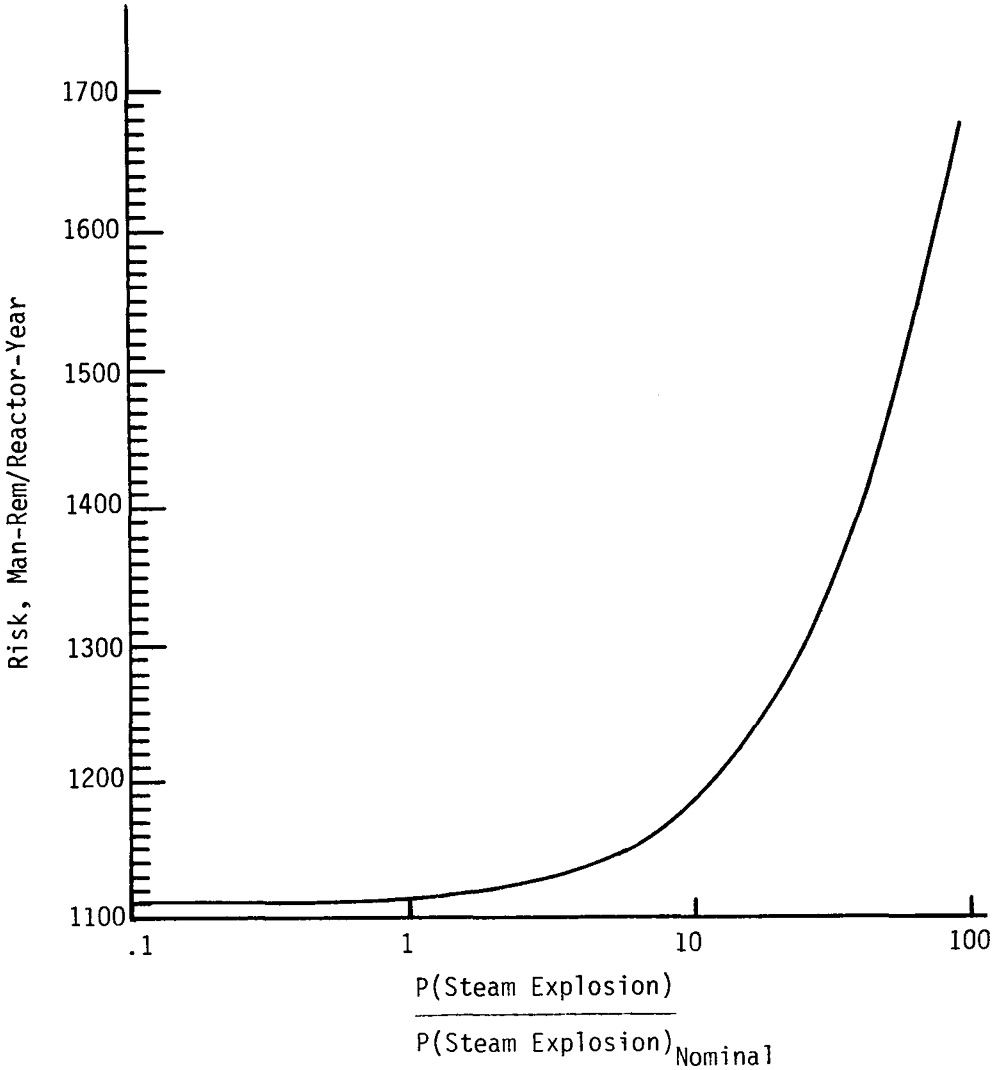


Figure 6-4. Sensitivity of BWR Risk in Man-Rem/Reactor-Year To the Probability of Steam Explosion

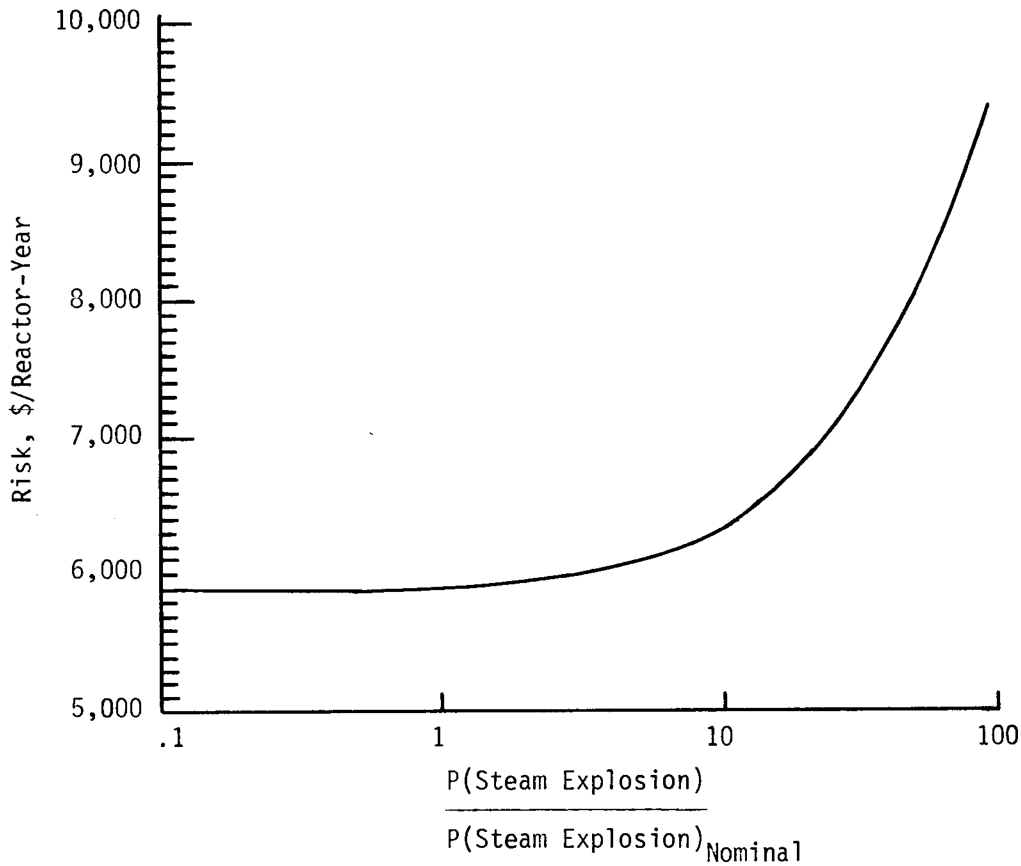


Figure 6-5. Sensitivity of BWR Risk in $/Reactor-Year to the Probablity of Steam Explosion

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Section 7

CONSEQUENCE CODE DEVELOPMENT

Consequence analysis efforts during the past year have focused on the continued development and expansion of computer codes. In particular, a containment behavior code package, merging the features of several separate programs, is being created to provide input required for the CORRAL code (7-1). The new code package (INCOR) will predict time-dependent containment conditions following a loss-of-coolant accident (LOCA). Another new code has been assembled to calculate radiation doses to internal organs. This internal radiation dose calculation code (INRAD) incorporates the features of an improved lung model and a gastrointestinal (GI) tract model. Finally, in conjunction with the LWR sensitivity study (reported in Section 6), the capabilities of the Calculations of Reactor Accident Consequences (CRAC) code have been extended to generate intermediate health effects for radiation doses to a distance of 50 miles from the plant site, and a supplementary program, CRAC-FINAL, was developed to stage CRAC results read from a tape.

INCOR CONTAINMENT BEHAVIOR CODE PACKAGE

The objective of this effort has been to develop a general purpose computer code to predict reactor response during severe accident conditions. These predictions will provide time-dependent post-LOCA containment conditions, which are required for input to the CORRAL code. This addition to CORRAL's capabilities will make it possible to analyze more complex accident sequences than those accomplished for the Reactor Safety Study (RSS).

Input requirements for the CORRAL code include containment thermodynamics (pressure, temperature, and vapor composition), intercompartmental flow rates, and release rates to the atmosphere, all as a function of time. For the RSS, containment data input to CORRAL were originally provided by time-consuming hand calculations using simplified energy balance, heat transfer, and fluid flow equations. The INCOR code package will provide an improved computational tool for generation of containment data for CORRAL.

CONTEMPT-LT (7-2) was chosen to form the basis of the INCOR containment behavior code (7-3). CONTEMPT is a systems analysis program that uses a numerical method to analyze transient containment behavior. It will provide a coupling between the core melt behavior models that describe system conditions and the model that describes radionuclide transport within containment.

The models in the BOIL code, developed for the RSS (7-4), have been chosen to describe phenomena pertaining to the early phases of a core melt. BOIL calculates core heatup for a LOCA without ECC flow such that water boils out of the pressure vessel, uncovering the core. The approach used is to divide the core into small volumes, or nodes. The code calculates the heat produced in each node and performs heat balances between the fuel and coolant nodes. Heat generated from metal-water reactors, as well as fission product decay heat, is considered. It then calculates the water-steam mixture level in the core and estimates the steam boiloff rate. Local meltdown is assumed when the temperature of a node exceeds the melting point of uranium dioxide.

The INTER code (7-5), created by Sandia Laboratories, provides information on the thermal and chemical interactions between the core melt and the concrete that forms the base of the containment. INTER uses empirical heat-transfer coefficients, derived from melt-concrete tests, in a simplified mechanistic approach to calculate temperature profiles, predict ablation rates, and estimate gas production with chemical reaction in the metallic portion of the melt system.

Development of the INCOR code package will require merging these three codes, CONTEMPT, BOIL, and INTER. Separate models will be generated to fill gaps in a core meltdown sequence not included in these three codes. Because CONTEMPT is already a rather large code, modifications to its basic structure will be kept to a minimum. The models describing core meltdown behavior will be treated as subroutines that can be added without major changes to the CONTEMPT programming sequence.

Results of a detailed review of CONTEMPT, BOIL, and INTER indicate that the merge can best be accomplished by working with mass and energy from the primary compartment. These mass and energy sources are caused by decay heat, metal-water reactions, and other chemical reactions, which are represented by tabular input routines. BOIL will provide the mass and energy source data for reactions taking place during the core meltdown process. INTER will provide similar data for the coremelt-concrete interaction phase after the pressure vessel bottom head fails. Input of these data to the CONTEMPT program will replace the table lookup that is presented required.

A new model, PVMELT, will be developed to fill in the gap between the above two phases. The same programming approach will be taken, so that PVMELT will provide mass and energy source data derived from decay heat and metal-water interactions during pressure vessel meltthrough. This new model will be developed after the three present codes have been merged and the integrated INCOR code is operational.

The relationship between the codes comprising INCOR, with reference to the mass and energy transfer from the primary system compartment to the containment is shown in Figure 7-1.

INTERNAL RADIATION DOSE CALCULATION CODE (INRAD)

Consequence analysis of hypothetical radiological releases has been performed in the EPRI program with calculations of radiation dose to internal organs. These calculations have been based on published tables of dose conversion factors, in terms of rem per microcurie, from various sources.

The Calculation of Reactor Accident Consequence (CRAC) code (7-6) uses data based on the new lung model proposed by the International Commission on Radiological Protection (ICRP) Task Group on Lung Dynamics (7-7,7-8). However, the dose conversion factors using this new Task Group Lung Model (TGLM) are not as readily available in the literature as those using the initial ICRP2 lung model (7-9).

As shown in Figure 7-2, the ICRP2 lung model uses a single lung compartment, with the translocation rates shown in Table 7-1. The new TGLM replaces this single compartment with four distinct compartments and detailed translocation rates from each compartment. The new lung model (TGLM) consists of two basic schemes, briefly summarized below:

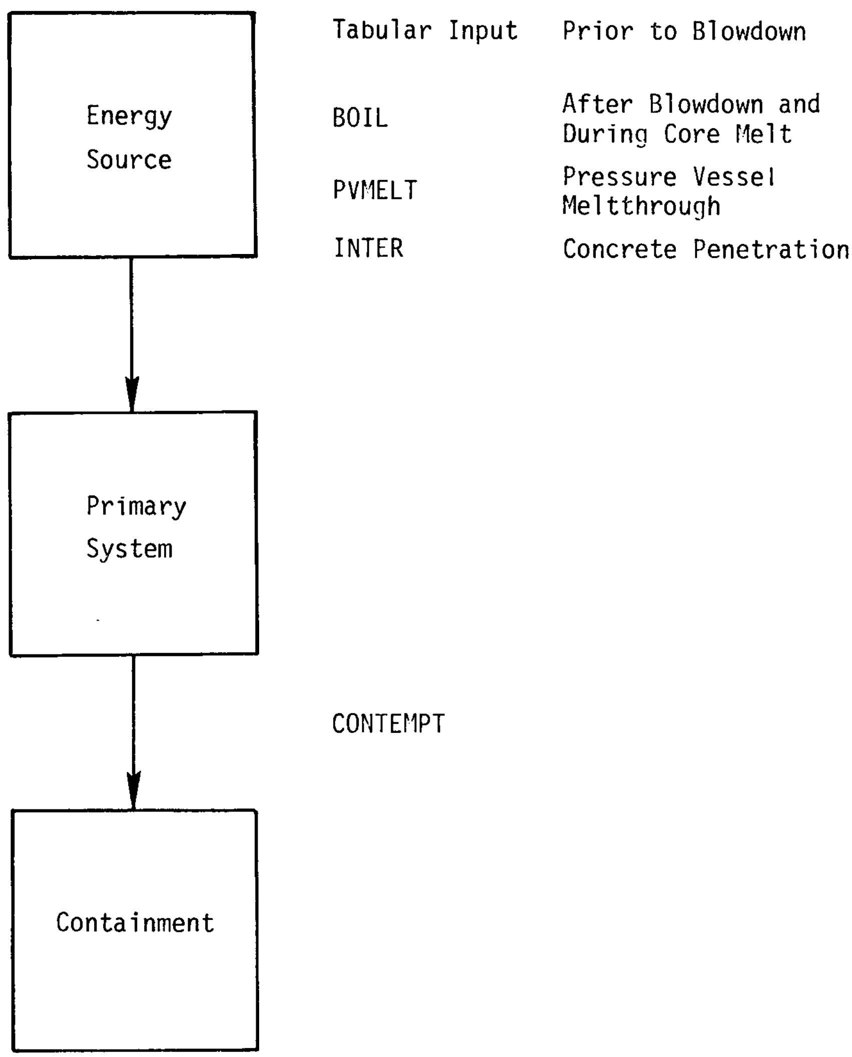

Figure 7-1. INCOR Code Configuration

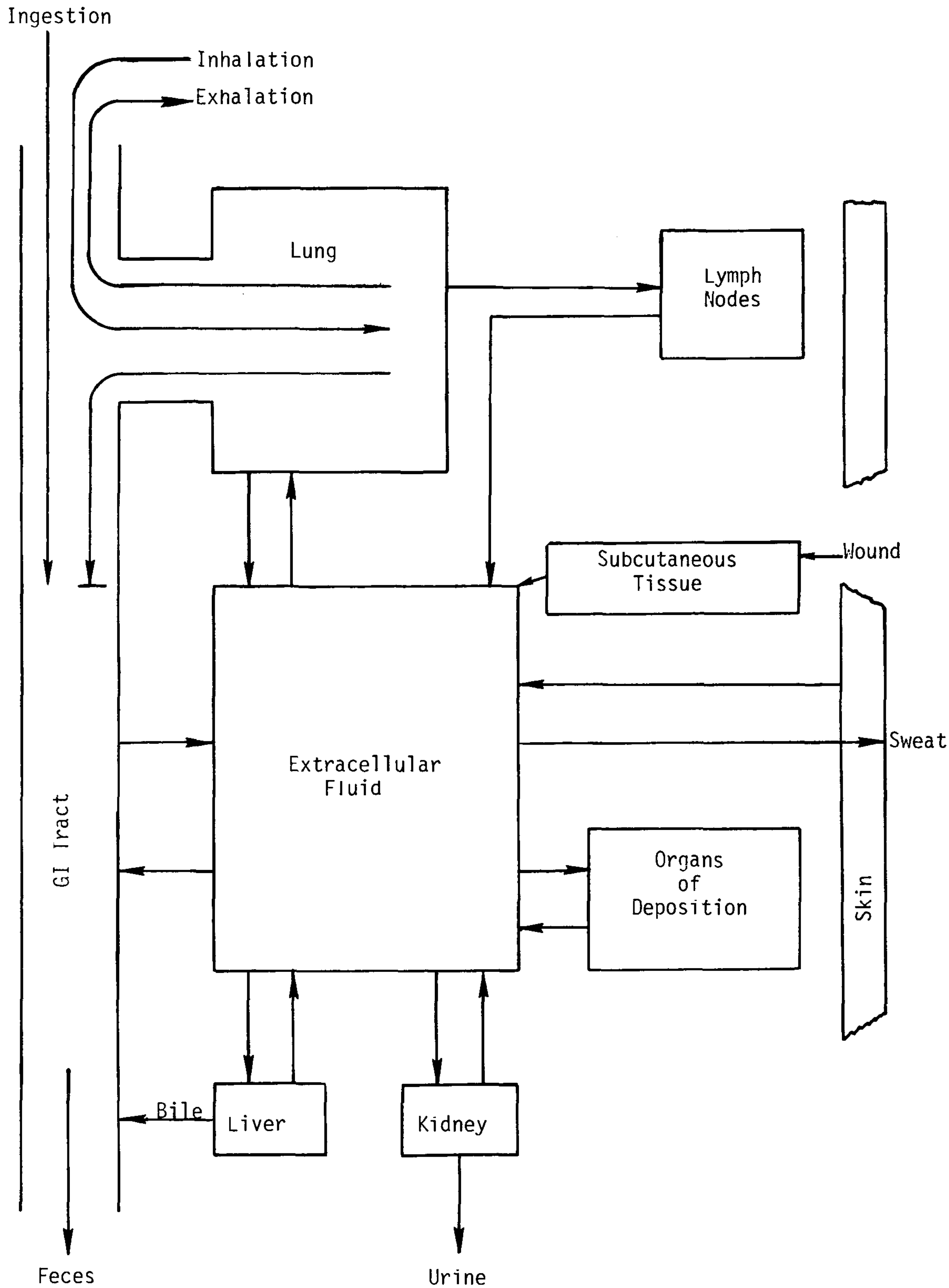

Figure 7-2. The ICRP2 Lung Model

Table 7-1

ICRP2 LUNG MODEL DEPOSITION AND TRANSLOCATION RATES

	Readily Soluble Compounds (%)	Other Compounds (%)
Exhaled	25	25
Deposited in upper respiratory passages and subsequently swallowed	50	50
Deposited in the lungs (lower respiratory passages)	25 (this is taken up into the body)	25*

*Of this, half is eliminated from the lungs and swallowed in the first 24 hours, making a total of 62-1/2% swallowed. The remaining 12-1/2% is retained in the lungs with a half-life of 120 days, it being assumed that this portion is taken up into the body fluids.

1) A dust deposition scheme utilizes dust sampling data and describes dust deposited in terms of three major regions of the respiratory tract:

   - Nasopharyngeal (NP)
   - Tracheobronchial (TB)
   - Pulmnary (P)

   Figures 7-3, 7-4, and 7-5 show the deposition variation with particle size. Table 7-2, representing Figure 7-3, lists the normal tidal volume rates.

2) A dust clearance scheme quantitatively treats dust deposited in each respiratory compartment according to pathways. Table 7-3 is a classification of retention tendency, and Table 7-4 presents clearance rates. These inputs may be modified from time to time, when appropriate physiological data become available.

Figure 7-6 is a schematic diagram of all dust deposition sites and clearance pathways. The translocation processes between the blood and the organ of interest, as well as excretion pathways, are indicated.

Dose conversion factors derived from the new TGLM are more realistic and therefore preferable for use in calculations of radiation doses to internal organs. However, the data available with the CRAC model are limited to the 54 radionuclides investigated in the Reactor Safety Study. Furthermore, these data are applicable only for acute inhalation exposure, not for chronic inhalation exposure.

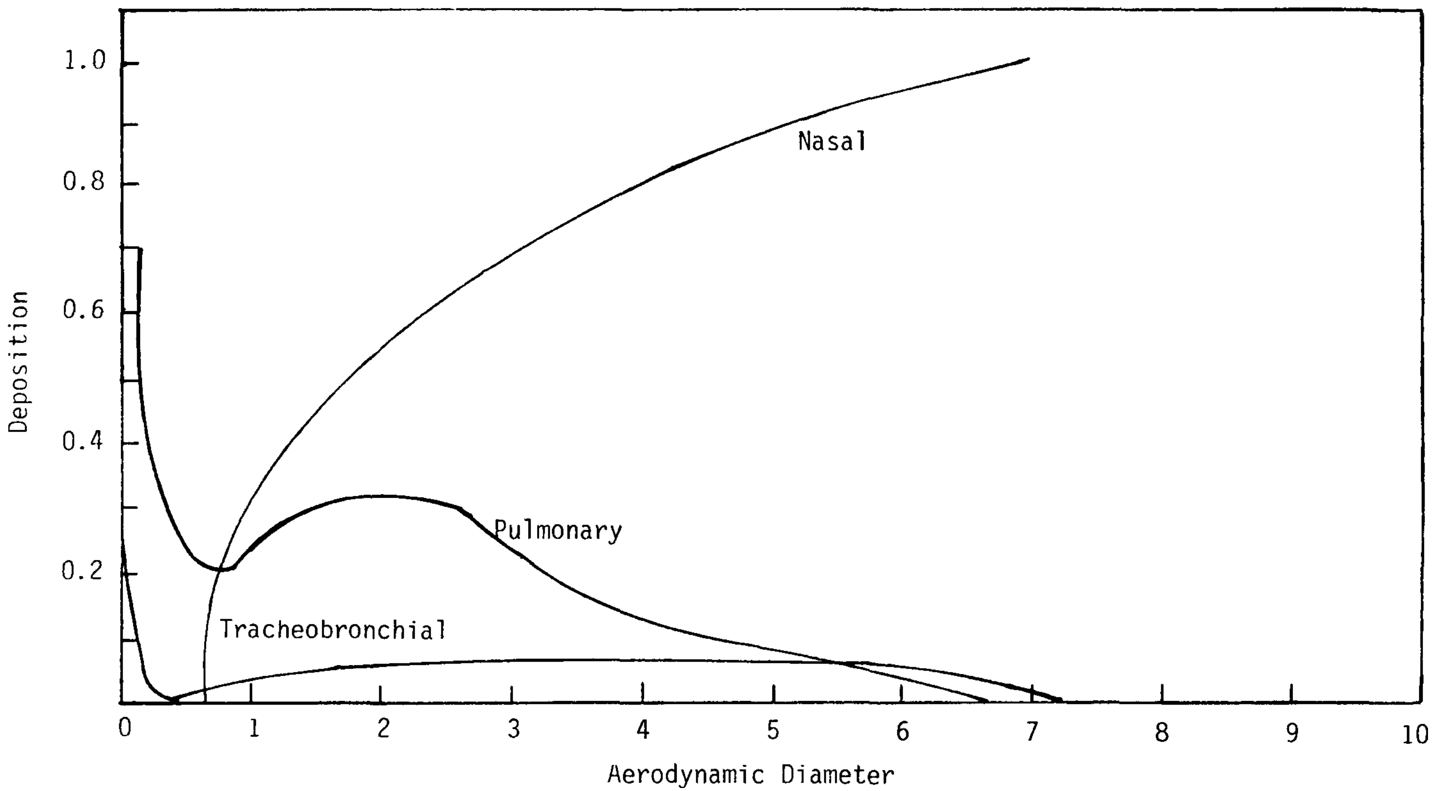

Figure 7-3. TGLM Deposition as a Function of Particle Size for 15 Respirations/Minute, 1450 cm^3 Tidal Volume

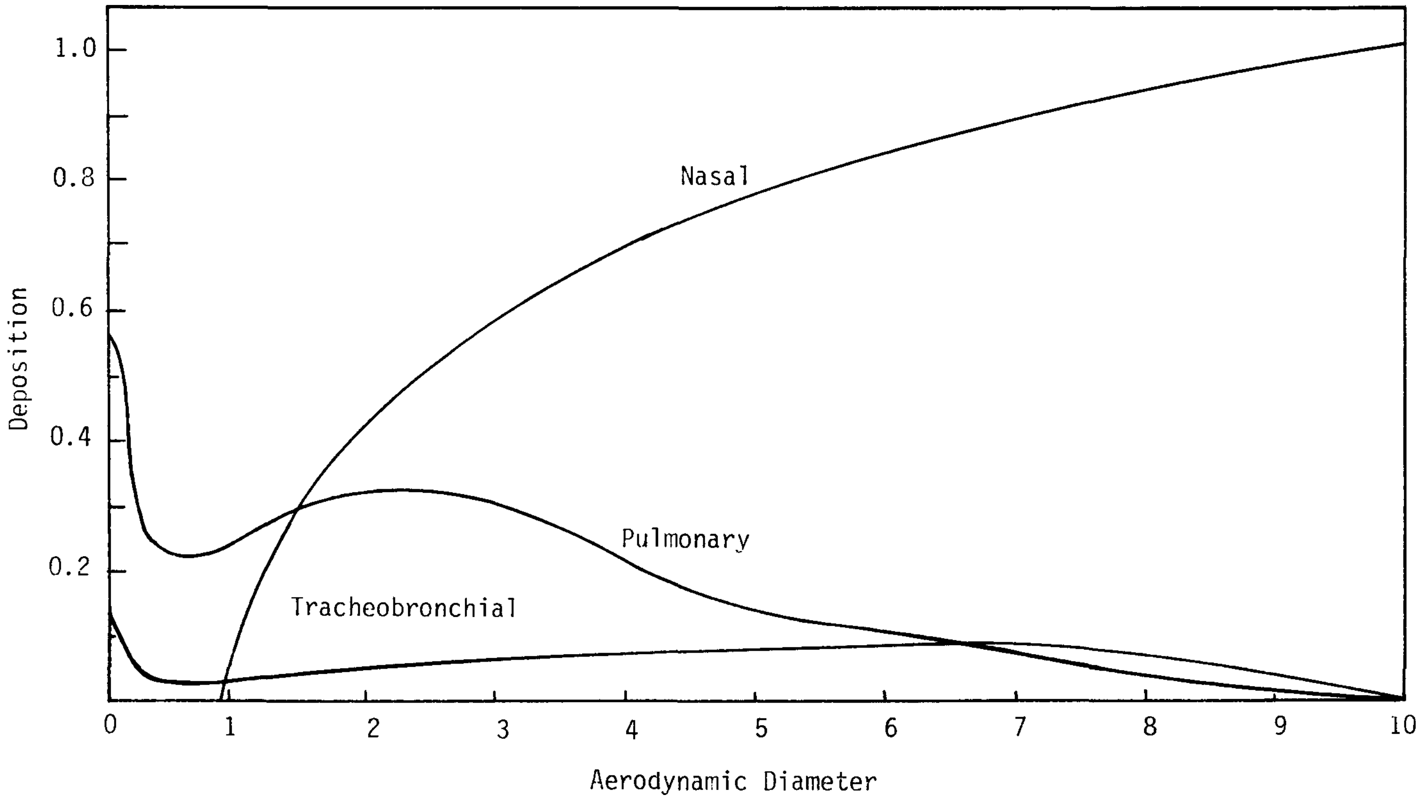

Figure 7-4. TGLM Deposition as a Function of Particle Size for 15 Respirations/Minute, 750 cm^3 Tidal Volume

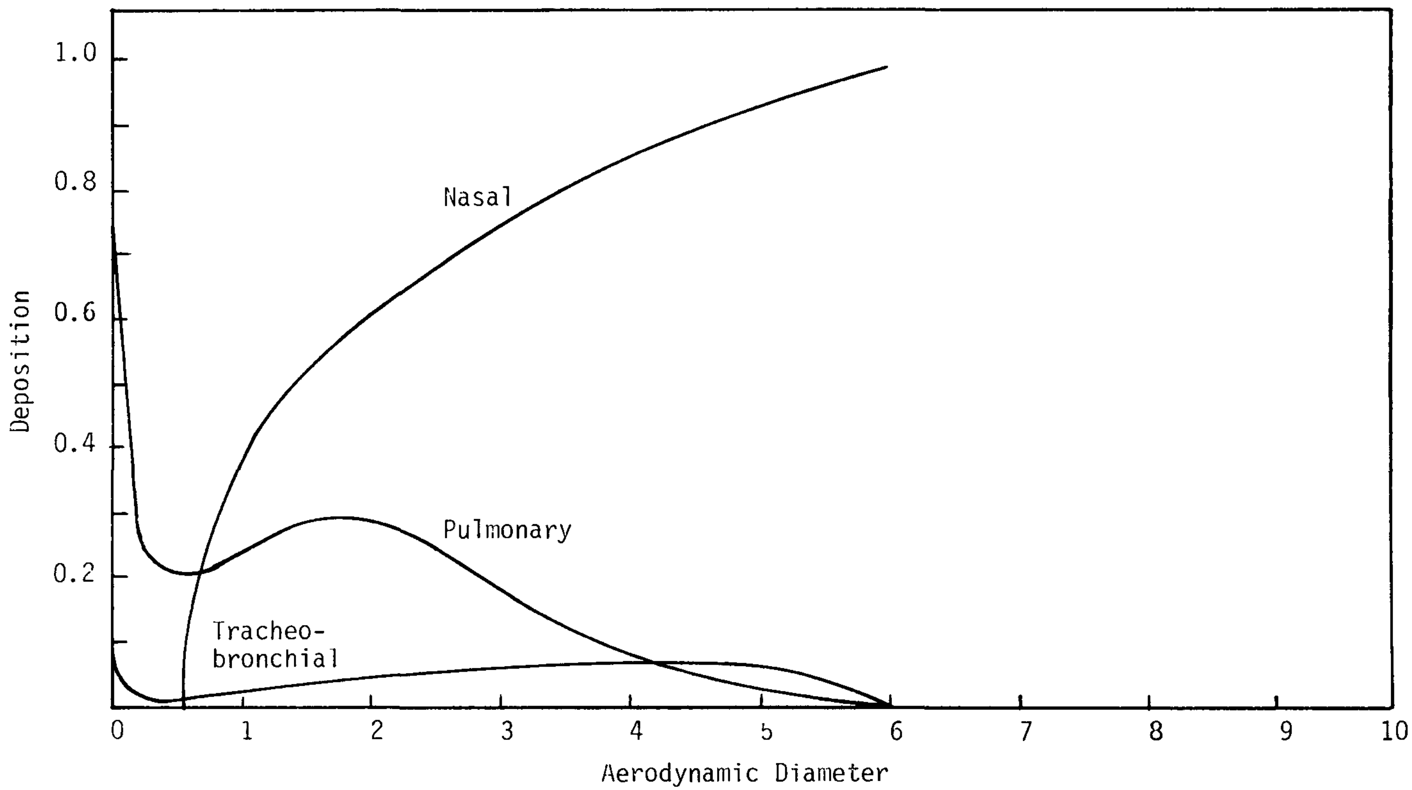

Figure 7-5. TGLM Deposition as a Function of Particle Size for 15 Respirations/Minute, 2150 cm^3 Tidal Volume

Table 7-2

DEPOSITION OF DUST PARTICLES IN THE RESPIRATORY TRACT AS A FUNCTION OF PARTICLE DIAMETER

(Tidal Volume = 1450 ml; Respiratory Rate = 15 per Minute)

Mass Median Aero Dynamic Diameter (μm)	D_3 N-P Region	Deposition Fraction D_4 T-B Region[a]	D_5 P Region
0.01			0.75
0.05			0.60
0.10	0.01	0.08	0.50
0.20	0.05	0.08	0.45
0.50	0.15	0.08	0.30
1.0	0.30	0.08	0.25
2.0	0.50	0.08	0.20
5.0	0.75	0.08	0.15
10.0	0.90	0.08	0.095

[a]Deposition in the Tracheobronchial compartment can be approximated by considering it constant for particles between 0.1 and 10.0 microns. (6-7)

Table 7-3

PULMONARY CLEARANCE CLASSIFICATION OF INORGANIC COMPOUNDS

CLASS Y -- Avid retention: cleared slowly (years)

Carbides -- actinides, lanthanides, Zr, Y, Mn
Sulfides -- none
Sulfates -- none
Carbonates -- none
Phosphates -- none
Oxides and hydroxides -- lanthanides, actinides Groups 8 (V and VI), 1b, 2b (IV and V), 3b except Sc^{3+}, and 6b
Halides -- lanthanide fluorides
Nitrates -- none

CLASS W -- Moderate retention: intermediate clearance rates (weeks)

Carbides -- cations of all Class W hydroxides except those listed as Class Y carbides
Sulfides -- Groups 2a (V + VI), 4a (IV-VI), 5a (IV-VI), 1b, 2b and 6b (V + VI)
Sulfates -- Groups 2a (IV-VII), and 5a (IV-VI)
Carbonates -- lanthanides, Bi^{3+} and Group 2a (IV-VII)
Phosphates -- Zn^{2+}, Sn^{3+}, Mg^{2+}, Fe^{3+}, Bi^{3+} and lanthanides
Oxides and hydroxides -- Groups 2a (II-VII), 3a (III-VI), 4a (III-VI), 5a (IV-VI), 6a (IV-VI), 8, 2b (VI), 4b, 5b and 7b Sc^{3+}
Halides -- lanthanides (except fluorides), Groups 2a, 3a (III-VI), 4a (IV-VI), 5a (IV-VI), 8, 1b, 2b, 3b (IV-V), 4b, 5b, 6b, and 7b
Nitrates -- all cations whose hydroxides are Class Y and W

CLASS D -- Mineral retention: rapid clearance (days)

Carbides -- see hydroxides
Sulfides -- all except Class W
Sulfates -- all except Class W
Carbonates -- all except Class W
Phosphates -- all except Class W
Oxides and Hydroxides -- Groups 1, 3a (II), 4a (II), 5a (II,III), 6a (III)
Halides -- Groups 1a and 7a
Nitrates -- all except Class W
Noble Gases -- Group 0

NOTE: Where reference is made from one chemical form to another, it implies that an *in vivo* conversion occurs, e.g., hydrolysis reaction.

Table 7-4

CONSTANTS FOR USE WITH TGLM CLEARANCE MODEL

		CLEARANCE CLASSES					
		(D)		(W)		(Y)	
Region	Pathway[a]	f[b]	λ (sec^{-1})	f	λ (sec^{-1})	f	λ (sec^{-1})
N-P	(a)	0.5	8.023 E-04	0.1	8.023 E-04	0.01	8.023 E-04
	(b)	0.5	8.023 E-04	0.9	8.914 E-06	0.99	2.006 E-05
T-B	(c)	0.95	8.023 E-04	0.5	8.023 E-04	0.01	8.023 E-04
	(d)	0.05	4.011 E-05	0.5	4.011 E-05	0.99	4.011 E-05
P	(e)	0.8	1.605 E-05	0.15	1.605 E-07	0.05	1.605 E-08
	(f)	-	-	0.4	8.023 E-06	0.4	8.023 E-06
	(g)	-	-	0.4	1.605 E-07	0.4	1.605 E-08
	(h)	0.2	1.605 E-05	0.05	1.605 E-07	0.15	1.605 E-08
L	(i)	1.0	1.605 E-05	1.0	1.605 E-07	0.9	8.023 E-09

[a]Pathways are shown in Figure 6-6

[b]f represents the faction that follows the specific pathway listed

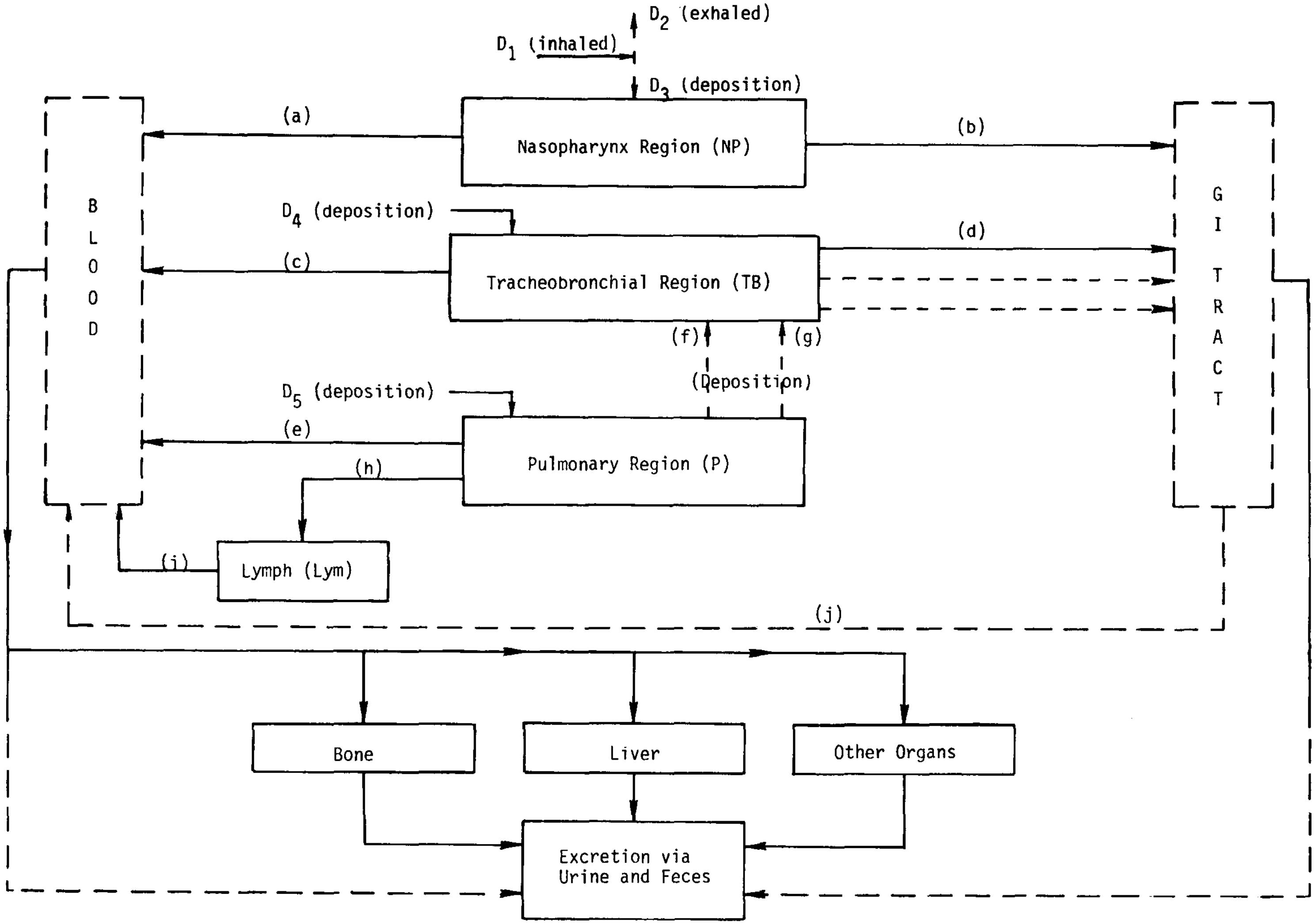

Figure 7-6. Schematic Diagram of TGLM Dust Deposition Site and Clearance Pathways

DACRIN (7-10) is an existing code which uses the TGLM to calculate inhalation dose conversion factors for both acute and chronic exposure for the lungs. However, DACRIN does not include a model for the GI tract, which is needed to obtain complete dose conversion factors. Therefore, another code would be required in order to calculate the total ingestion dose conversion factors required for CRAC.

To simplify generating the various input requirements for consequence calculations, a single code has been designed to calculate doses to internal organs for both acute and chronic exposure, and for both inhalation and ingestion pathways. This code, INRAD, uses extensive tables of rem per microcurie-days published by Snyder et al. (7-11).

INRAD was derived from CONVOLX (7-12), which uses a convolution integration numerical scheme. The code was enhanced to calculate the time-integrated inventory of the deposited radionuclide and its daughters for different time periods and for various combinations of internal exposure modes. As many as six radionuclides, including the parent nuclide, may be input for each decay chain. It was also made to run several times faster, with more flexibility, than the original CONVOLX. Both the TGLM and a GI tract model (7-13) have been incorporated into the INRAD programming. Internal doses are calculated using data for 223 radionuclides of the average dose equivalent in various target organs of an adult, per unit accumulated activity in a source organ (7-11). For acute exposure, these doses to the target organs are calculated in terms of rem per microcurie inhaled or ingested. For chronic exposure the target organ doses are calculated in terms of rem per microcurie per day inhaled or ingested. The time interval of chronic exposure, as well as the time integration periods for the dose commitment, can be specified. The program input parameters are listed in Tables 7-5 and 7-6.

Organ retention models summarize the transfer rate from the blood and the elimination rate to the outside for each element considered. The retention in organs of the activity entering the blood is generally in the form

$$R(t) = Ae^{-\lambda t} \tag{7-1}$$

Table 7-5

INRAD PROGRAM INPUT FORMATS AND VARIABLE DESCRIPTIONS

CARD SET	FORMAT	VARIABLE	DESCRIPTION
1	(4I5)	NI	The number of isotopes to be used for this run, maximum of 6 are allowed
		NP	The number of time periods, maximum of 9
		IGI	If = 0 inhalation occurs; if .NE.0, no inhalation
		ISOT	If .NE.0 call ISOT7 subroutine (uses Tape 10, and tape 69)
2	(8E10.4)	TPEMOD(I), (I=1, NP)	The time integration periods (years), maximum of nine time periods
3	(3E10.4)	PSS	Particle size (microns)
		TCHRON	Time period for chronic inhalation (years); if = 0 no chronic inhalation occurs
		TCRING	Time Period for chronic ingestion (years); if = 0 no ingestion occurs
4	(8E10.4)	F1 (I) (I=2,NI)	Fraction of each isotope transferred to the blood from the small intestine (SI)
5	(16I5)	ISTY(I) (I=2,NI)	Nuclide clearance class (1 to 3)
6	(8E10.4)	RLAMB(I) (I=2,NI)	Nuclide half-life (years)
7	6(A6,1X,A2,1X)	ISONAM(I), (NOMT(I), (I=1,NI)	Pairs of nuclide name and clearance class (D,W,Y) for each nuclide (left adjusted)
8	(2I5)	NO1	Not presently used
		NO2	Number of blood-to-organ and organ-to-outside transfer rate indexes to be read (ISR and IRC)
9	(16I5)	ISR (I) (I=1, NO2)	Compartment index for source organ - see Table 7-6 containing the organs for the index value (1 to 35)
10	(16I5)	IRC (I) (I=1, NO2)	Compartment index for receiver organ - see ISR

Table 7-5 (continued)

CARD SET	FORMAT	VARIABLE	DESCRIPTION
11	(8E10.4)	FS[IRC(I),ISR(I)] (I=1, NO2) (k=2, NI)	Transfer rates (second^{-1}) (blood-to-organ and organ-to-outside) using ISR,IRC as index variables pointing to the organs being transferred to: there is one set of data for each nuclide (2 through NI)
12	6(AH,1X,A2,3X)	AMAS(I),ANI(I) (I=2, NI)	Atomic mass and atomic weight of each nuclide (right adjusted with period on AMASS); no period on AWT

Table 7-6

ORGAN NUMBERING SCHEME FOR INRAD INDEX

INDEX	ORGAN
1	N-P Lung
2	N-P Lung
3	T-B Lung
4	T-B Lung
5	Pul Lung
6	Pul Lung
7	Pul Lung
8	Pul Lung
9	Lymph
10	Stomach
11	Small Intestine (SI)
12	Upper Large Intestine (ULI)
13	Lower Large Intestine (LLI)
14	Blood
15	Bone
16	Bone
17	Liver
18	Liver
19	Kidney
20	Kidney
21	Testes
22	Ovaries
23	Total Body
24	Total Body
25	Total Body
26	Muscle
27	Spleen
28	Pancreas
29	Thyroid
30	Thyroid
31	Thyroid
32	Other
33	Soft Tissue
34	Soft Tissue
35	Outside

where is the excretion rate from the organ, A is the fraction that reaches the organ from the blood, and t is time after deposition. If there is more than one retention rate, i.e.,

$$R(t) = Ae^{-\lambda 1^{t}} + Be^{-\lambda 2^{\ t}} + \ldots \tag{7-2}$$

the organ may be subdivided into more than one compartment to account for the difference in transfer rates. The excretion rate and fractional deposition to the organ from the blood determine the transfer rates (FS) needed as input variables.

Several test cases will be run in the near future in order to verify the INRAD calculations.

POPULATION DOSE CALCULATION MODIFICATIONS TO CRAC CODE

The consequence models in the CRAC code (7-6) calculate the potential health effects (e.g., latent cancer fatalities, early fatalities) after the release of radioactive materials from containment. This calculation accounts for the effectiveness of the radiation doses in causing cancer for specific organs summed over all spatial intervals to a distance of 500 miles from the reactor plant site for any postulated accident.

The objective of the PWR sensitivity study (reported in Section 6) was to assess the impact of safety system functions by comparing the benefits derived from these plant functions in terms of the dollar value assigned by 10CFR50, Appendix I. This required calculation of the annual whole body and thyroid doses to the population within 50 miles of the plant site for all PWR composite sites and for accident categories 1 through 7.

For this reason the CRAC code was modified to provide an intermediate result for the 50-mile population, instead of the total 500-mile demography. In addition, the code was modified to provide the total organ doses to the thyroid, instead of thyroid cancers or nodules.

In order to calculate thyroid doses to the population without applying dose effectiveness factors (normally done for thyroid cancer incidence), results options were added to CRAC to allow calculation of a linear extrapolation to zero doses to the thyroid, bypassing the computation of thyroid doses normally used to compute cancer effects. The results are then similar to those for whole-body man-rem calculations.

The mean and the Complementary Cumulative Distribution Function (CCDF) for these two specific results were then calculated using the accident probabilities and site probability distribution for a PWR plant.

The capability to store the results of the individual runs for each site and accident category on magnetic tape was also implemented into the present CRAC code. Site and accident category were assigned a probability of unity when stored on the tape so that when results were read from the tape, they could be assigned any desired probability. This strategy necessitated an ability to manipulate the tape results, for which a separate code, CRAC-FINAL, was created.

CRAC-FINAL reads the tape, staging each site and accident category individually as needed. Input requirements have been minimized by storing related information about each data file on the tape.

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Section 8

FUEL CYCLE RISK ASSESSMENT*

SUMMARY AND BACKGROUND

EPRI has been conducting a study of the radiological risks of the fuel cycle supporting the production of electric power by commercial nuclear reactors. The primary purpose of this work is to complete the estimated radiological risk of nuclear electric power generation by addressing the risk of the supporting fuel cycle. The routine risk from mining and milling and the accident risks from reprocessing, mixed-oxide (MOX) fuel fabrication, transportation of recovered material, and waste disposal have been investigated and reported in five draft documents. This set of fuel cycle steps was selected as representing the dominating radiological risks of a fuel cycle involving recovery of plutonium and uranium. The draft reports were modified in response to extensive peer review, and the results are presented in a separate status report (8-1). This section is a summary of that status report.

Specifically, the results of the EPRI fuel cycle risk assessment are that the supporting fuel cycle contributes about 1% of the risk of generating nuclear electric power. Thus, the results of the Reactor Safety Study (WASH-1400) (8-2) reasonably approximate the full risk of nuclear power. For perspective, the radiological risk from a very large nuclear power industry (685 plants), projected for the future, would provide less than 1/200ths of the exposure that the public would receive from the radioactivity of the earth and that coming from the sky.

The work reported here ranks the fuel cycle steps in decreasing order of risk as: 1) mining and milling, 2) transportation, 3) mixed-oxide fuel fabrication, 4) reprocessing, 5) waste disposal preclosure, and 6) waste disposal post-closure. This ordering is approximately the reverse of that perceived by the

*This section is derived from a paper prepared for the Fourth International Conference of the Systems Safety Society, San Francisco, July 1979.

public, as reflected in the news media. The highest ranking radiological accident risk contributor, transportation, has an overall risk of about 1/100th that of being run over by a waste-carrying truck.

The conclusions of the status report are presented pictorially in Figure 8-1. The volume of the cubes is proportional to the radiological health effects risk of the fuel cycle. The large block of nuclear power plant risk rests on the extremely large plateau of natural background. The block for risk of long-term waste disposal, too small to plot, would be approximately 0.05 micrometers on a side.

These data are presented more conventionally in Table 8-1. Included for comparison is the projected U.S. population dose due to natural background per unit electric power (GWe). Since this background radiation or more has existed for all time, and since the whole nuclear fuel cycle is smaller than background by more than a factor of one thousand, radiological accidents due to the production of electricity by nuclear energy should have a minor effect on the environment.

Figure 8-2 presents the latent cancer fatality risk per gigawatt-electric year in graphic form. The ordinate is the probability of observing x or more latent effects as a function of x plotted on the abscissa. Also included are straight lines representing the risk (probability times consequence) envelope of the accidents analyzed in this study and a risk envelope of power plant accidents (taken from WASH-1400). The risk envelopes are lines of constant product of probability and consequences. If the ordinate and abscissa scales are the same, the lines are at an angle of 45^{0} to the abscissa. The perpendicular distance between the curves is the difference in risk between the fuel cycle and power plant risks.

APPROACH

The methods used in this study were based on procedures and practices used in WASH-1400 in order to determine as accurately as possible the risk of the supporting fuel cycle relative to that of the nuclear power plant. Significantly different procedures would have required a reanalysis of the nuclear power plant, a task considerably outside the committed resources. WASH-1400 practices have been modified to reflect numerous reviews and criticisms of

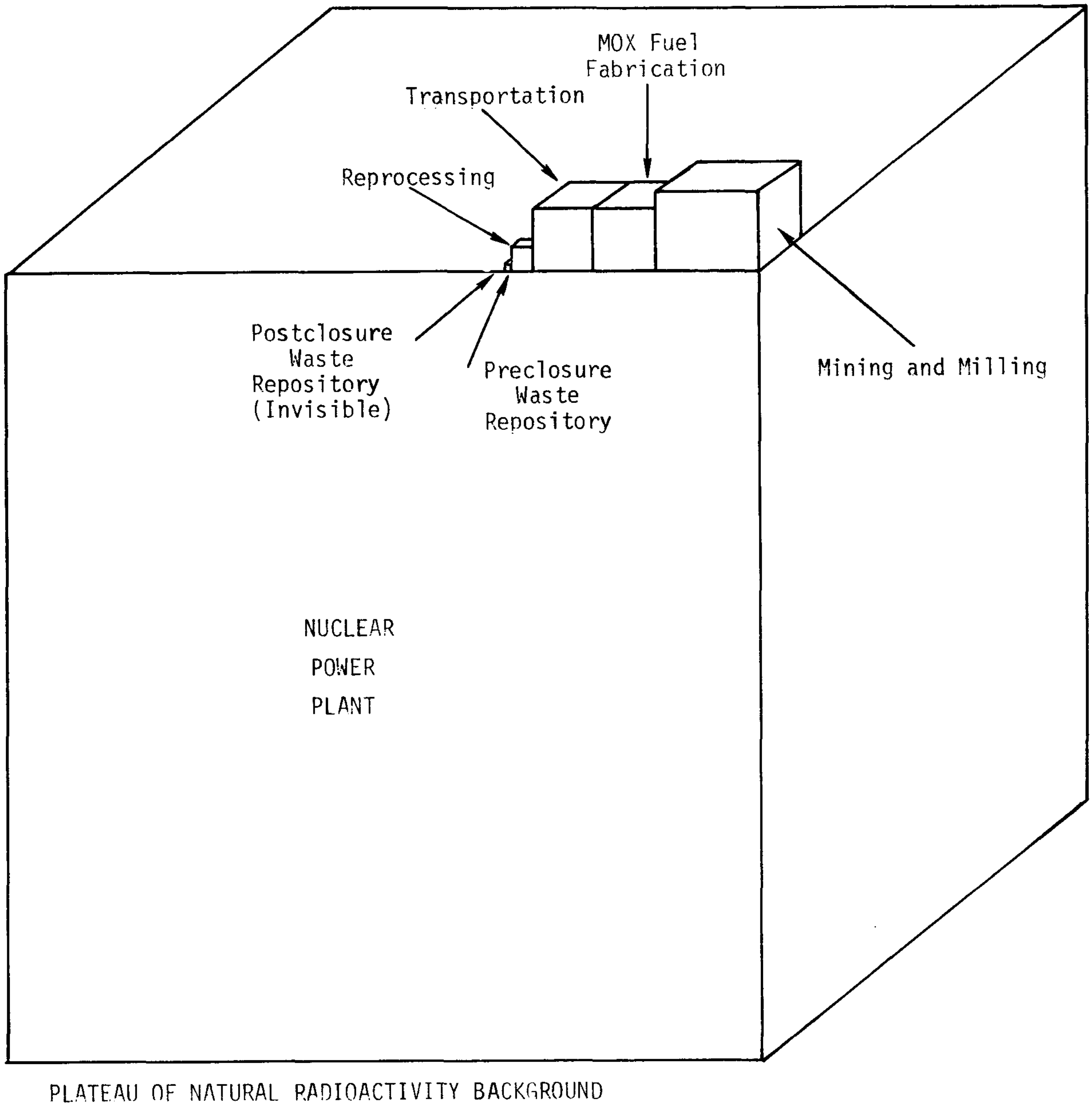

Figure 8-1. The Blocks of Radiological Risk Resting on the Extremely Large Plateau of Natural Background.

Table 8-1

SUMMARY OF THE NUCLEAR FUEL CYCLE RADIOLOGICAL RISKS (CONSEQUENCES TIMES PROBABILITIES) INVOLVED IN THE PRODUCTION OF ONE GIGAWATT-PER-YEAR

Cycle Step	Dose (Whole Body Person-Rem)	Health Effects (Latent Cancer Fatalities
Nuclear Power Plant	257[a]	0.02[b]
Mining and Milling		
Accident	not addressed	not addressed
Routine	0.2	2×10^{-5}[c]
Reprocessing	2×10^{-4}	3×10^{-8}
Mixed Oxide Fuel Fabrication	4×10^{-2}	3×10^{-6}
Transportation	3×10^{-2}	3×10^{-6}
Waste Repository		
Preclosure	4×10^{-5}	2×10^{-10}
Long-Term (10^6 years)	5×10^{-11}	5×10^{-15}[e]
Natural Background	7×10^{4}[d]	

(a) Number estimated from WASH-1400 (Final) based on genetic effects using Tables XI 4-1 and VI 9-11.
(b) From WASH-1400 (Final), Table XI 4-1.
(c) Based on 100 cancer deaths per million person-rem
(d) 3×10^8 people x 150 mrem/685 GWe in 2005.
(e) Based on 30-year individual dose rate integrated over 10^6 years and a population of 10^6.

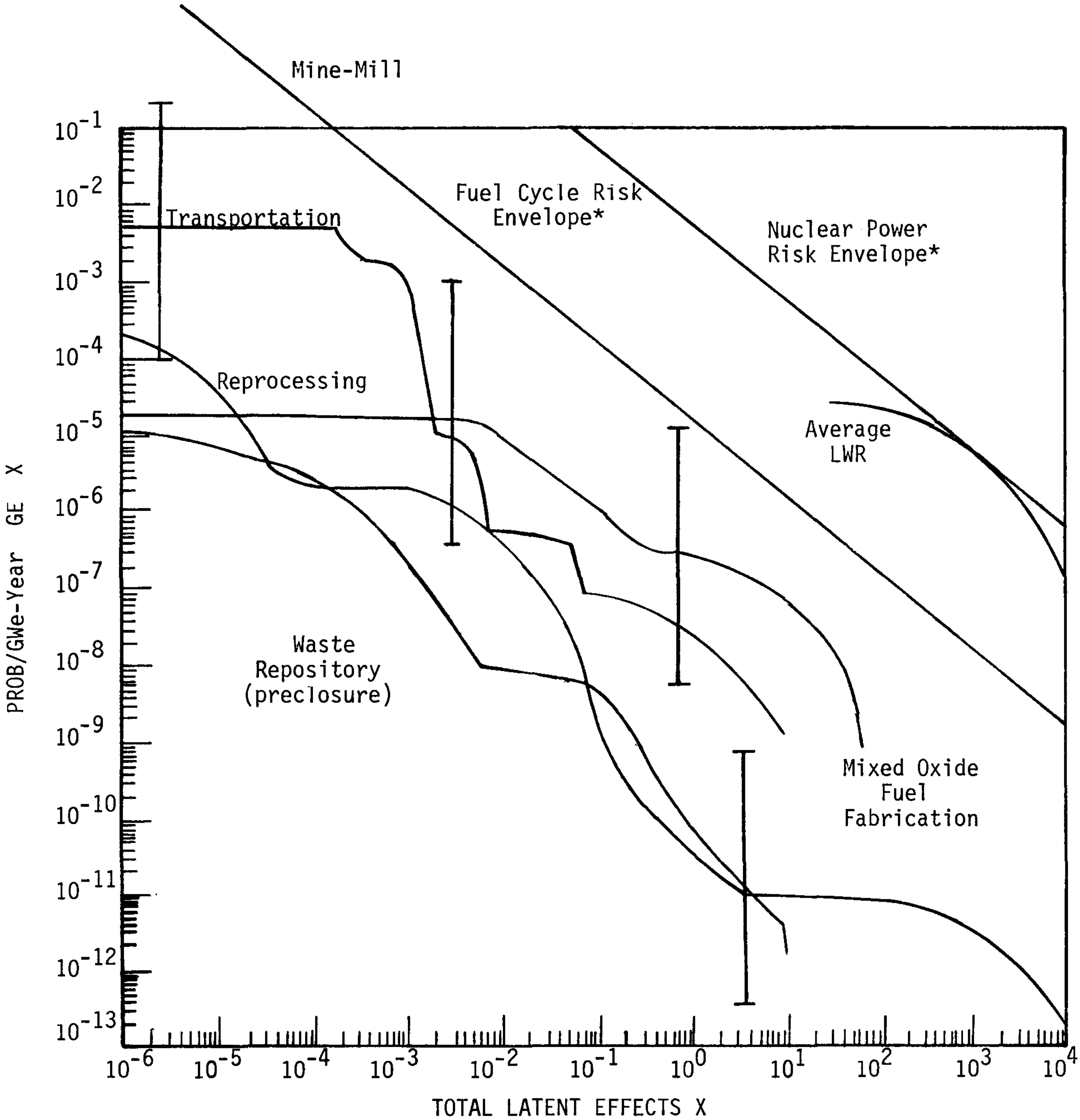

Figure 8-2. Health Effects Risk of the Fuel Cycle Compared With Nuclear Power Plants

that study. A particular effort has been made to indicate excessively encompassing error bounds in order to mislead no one regarding the precision of the estimates.

The general method used in the analysis of radiological accidents for a fuel cycle step or facility was to prepare a condensed engineering description oriented toward safety rather than production features. From this description a preliminary hazards analysis (PHA) was prepared. The emphasis in a PHA is on completeness rather than credibility. Working from the PHA, the higher risk accidents were selected, and the multiple barrier failures that must occur before the public is affected were diagrammed in the form of fault trees. These fault trees were quantified using failure rate data that were as appropriate as could be found.

Release source terms were calculated from the material mobility and from the forces available to disperse the material. This, too, is subject to error that has not been specifically included in the results. The quantity of radioactive material released outside the plant would depend on the performance of ventilation filters and the size of particles being dispersed. Experimentally measured dispersions for both wet and dry processes were used in this study. While two dispersion categories do not precisely characterize fuel cycle plant aerosols, they do provide better characterization than the usual assumption of a single most penetrating particle size. Consideration is also given to the fact that high-efficiency particulate (HEPA) filters may fail and release collected material. Doses and health effects were calculated with the Calculation of Reactor Accident Consequences (CRAC) code, which was appropriately modified for isotopic compositions other than those found in a reactor, and for continuous as well as puff releases.

An area of conservatism in the analysis was in the use of 40,000 MWd/T for fuel burnup and 90 to 150 days for cooling. In addition, some conservative release fractions were used in the reprocessing treatment. One area of future work will be refining values for both probabilities and consequences and more detailed treatment of error bounds to more accurately reflect the uncertainties. The general individual accident error factor estimated in the status report is 50, but much larger error factors are given for extremely unlikely events. An aspect which may be regarded as misleading is the quoting of results per year while some risks persist into the distant future. This was done for consistency with WASH-1400. A time-integrated treatment would require the calculation of plant lifetime risks, including decommissioning, and must also remain for future work.

THE FUEL CYCLE

The nuclear fuel cycle considered in this study is that which makes the best use of natural resources, namely the recycle of recoverable fissile and fertile material to the power plants. This fuel cycle is described in Figure 8-3. The ratios of 1 GWe light water reactor power plants to fuel cycle facilities are:

- Mining and milling:9
- Reprocessing:53
- Mixed-oxide fuel fabrication:15
- Waste repository:3800

The amount of transportation to link this dispersed industry, in terms of thousand shipment miles per GWe-year, is:

- Waste from fuel fabrication - 2.06
- Waste from power plants - 27.0
- Spent fuel by rail - 4.3
- Spent fuel by truck - 10.6
- Plutonium powder by truck - 0.19
- High-level waste by rail - 0.7
- Cladding hulls by rail or truck - 7.4
- Reprocessing waste - 15.0

Mining and Milling

The term, "mining and milling," refers to the process of removing uranium-bearing ore from the earth and crushing, grinding, and chemically separating the values into U_3O_8, commonly known as yellow cake.

It is not surprising that no radiological accidents could be found or postulated, considering the disperse nature of the ore and the low concentration of radioactive material after concentration in the mill. However, there is

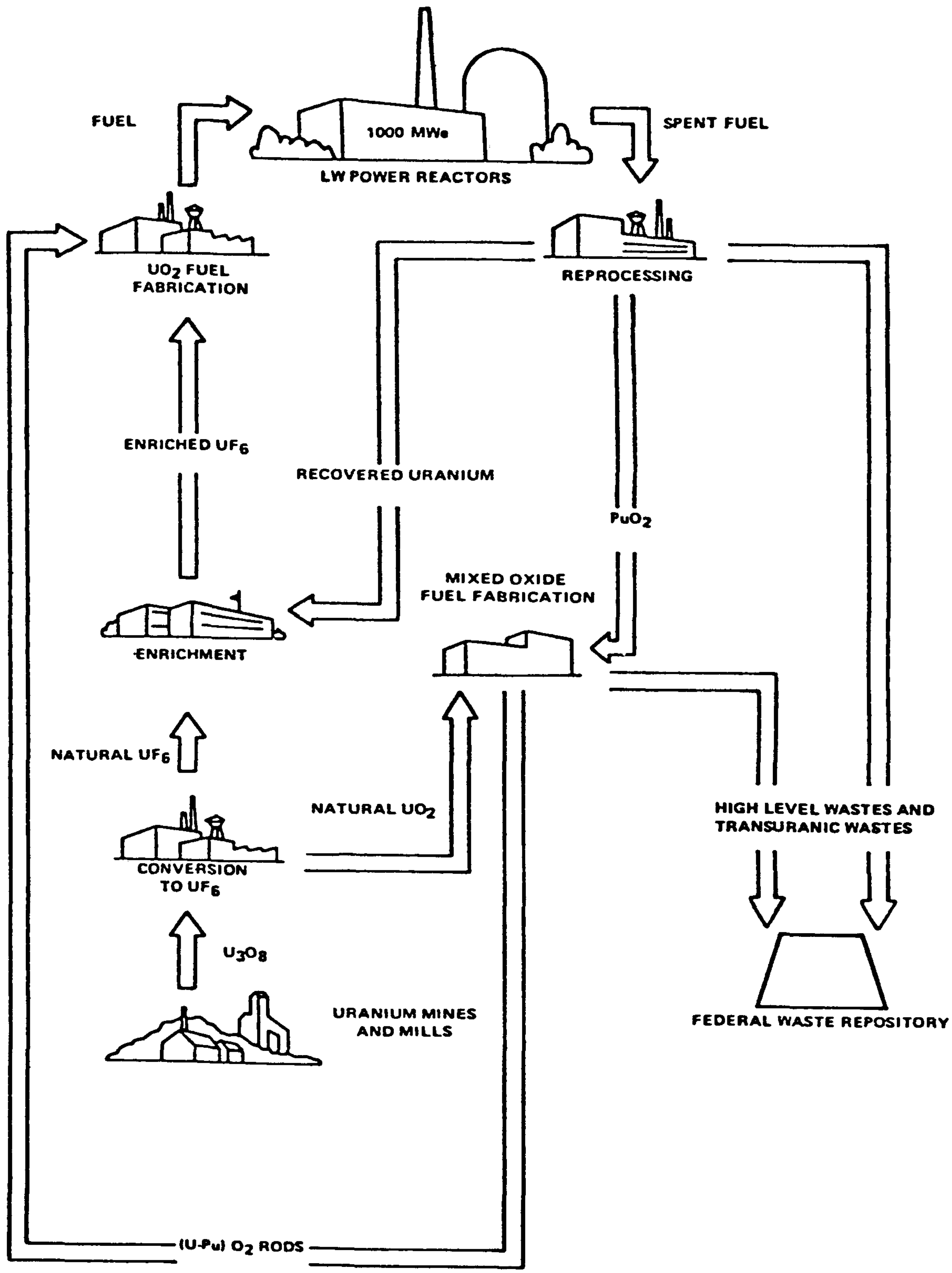

Figure 8-3. Light Water Reactor Fuel Cycle - Uranium and Plutonium Recycle

continuous evolution of radon from the mine and from the mill tailings. Because it is anticipated as part of the process, this continuous evolution is called a routine release. It represents the major radiological impact of the fuel cycle. These routine risks are presented in Table 8-1.

Reprocessing

After nuclear fuel has been depleted in fissionable material and "poisoned" by neutron-absorbing fission products, it is chemically reprocessed to separate the uranium and plutonium from the wastes. There is considerable accident potential at a reprocessing plant; however, the massive structure required for radiation shielding, the filtration system, and the design in anticipation of problems result in a facility with less public accident radiological risk than MOX fuel fabrication or transportation.

Following the use of a PHA, eight accidents were selected for fault tree and consequence analysis:

1) Loss of fuel storage pool
2) Ion-exchange bed fire and explosion
3) Criticality in a process cell
4) Hydrogen explosion in a "high acid feed" (HAF) tank
5) Fire in low-level waste
6) Fuel assembly drop
7) Explosion in the high-level waste calciner
8) Fracture of a krypton storage cylinder

These were considered under a variety of accident conditions, including none, one, and two final HEPA filter failures. The latent cancer effects per GWe-year for each accident category are presented in Table 8-2. It should be noted that some of these accidents are failures of systems designed to mitigate routine risk, thus transforming a routine risk into an accident risk.

Mixed Oxide (MOX) Fuel Fabrication

The MOX fabrication plant accepts plutonium from the reprocessing plant and combines it with natural uranium to provide fresh reactor fuel. Except for wet scrap recovery and laboratory procedures, this is a dry process conducted

Table 8-2

LATENT CANCER EFFECTS FOR REPROCESSING

Accidents	Latent Cancers/GWe-Year
1) Loss of the Fuel Storage Pool Water	2×10^{-10}
2) Ion-Exchange Resin Fire	9×10^{-11}
3) Criticality	2×10^{-8}
4) Hydrogen Explosion in a HAF Tank	5×10^{-13}
5) Fuel Assembly Drop	4×10^{-11}
6) Fire in Low-Level Waste	1×10^{-9}
7) Explosion in the High-Level Waste Calciner	1×10^{-8}
8) Fracture of a Krypton Storage Cylinder	4×10^{-9}

in massive structures made up of multiple barriers with triple HEPA filtration of air before it is exhausted. The radiological accident potential is from plutonium when it is in the form of a fine powder.

Using the procedures previously defined, eight accidents were selected for fault trees and detailed consequence analysis:

1) Earthquake greater than design basis
2) Aircraft crash into the plant
3) Hydrogen explosion in the reduction-oxidation-reduction (ROR) reactor
4) Hydrogen explosion in the sintering furnace
5) Ion-exchange resin fire
6) Dissolver explosion in wet scrap recovery
7) Loaded final filter failure
8) Criticality accident

The latent cancer effects per GWe-year for each accident category are given in Table 8-3.

Table 8-3

LATENT CANCER EFFECTS FOR MIXED-OXIDE FUEL FABRICATION

Accidents	Latent Cancers/GWe-Year
1) Earthquake > DBE	3×10^{-6}
2) Aircraft Crash Into Plant	6×10^{-8}
3) Hydrogen Explosion in ROR Reactor	9×10^{-17}
4) Hydrogen Explosion in the Sintering Furnace	2×10^{-15}
5) Ion-Exchange Resin Fire	4×10^{-18}
6) Dissolver Explosion in Wet Scrap	3×10^{-13}
7) Loaded Filter Failure	4×10^{-10}
8) Criticality	5×10^{-8}

Transportation

Transportation links the geographically dispersed industry. Seven transportation steps were analyzed:

1) Spent fuel by rail
2) Spent fuel by truck
3) Plutonium powder by truck
4) Solidified high-level waste by rail
5) Cladding hulls, by rail and by truck
6) Transuranic (TRU) wastes by truck
7) Nontransuranic contaminated wastes by truck

Preliminary hazards analyses were used for the selection of significant accidents. The fault trees were constructed by defining accident categories that were related to barrier failures and hence to release fractions. Consequences were calculated using data on release fractions, and doses and health effects were calculated using CRAC with average U.S. demography and meteorology. The results in terms of latent cancers per GWe-year are presented in Table 8-4 and are compared to the national fatality rate for a nonradiological truck accident. It should be noted that plutonium or plutonium-contaminated material provides the highest risk, although this risk is not very large.

Table 8-4

LATENT CANCER EFFECTS FOR TRANSPORTATION

Accidents	Latent Cancers/GWe-Year
1) Spent Fuel by Rail	2×10^{-9}
2) Spent Fuel by Truck	1×10^{-9}
3) Plutonium Powder by Truck	1×10^{-7}
4) Solidified Waste by Rail	7×10^{-8}
5) Cladding Hulls by Rail and Truck	6×10^{-10}
6) Transuranic Wastes by Truck	5×10^{-7}
7) Nontransuranic Wastes by Truck	2×10^{-6}

Fatality Rate for Nonradiological Truck Accident	1×10^{-4}*

*Source: Accident Facts, National Safety Council, 1976

Waste Disposal

The fuel cycle ends at the waste repository. Accidents may naturally be distinguished by the preclosure and postclosure phases. Using methods previously outlined, the preclosure accidents selected for fault tree and consequence analysis are:

1) Fuel truck crash into high-level waste receiving
2) Fuel truck crash into clad waste receiving
3) Fuel truck crash into TRU waste receiving
4) Air crash into receiving area
5) Elevator drop
6) TRU pallet drop
7) Final filter failure

The risks caused by these accident categories are shown in terms of latent cancers per GWe-year in Table 8-5.

The postclosure risks are adapted from NUREG-0279 (8-3), which uses Markov chains for the probability modeling and an extended one-dimensional diffusion model for nuclide migration. The results are 1×10^{-14} latent cancer per GWe-year averaged over 10^6 years for one million people.

Table 8-5
LATENT CANCER EFFECTS FOR WASTE DISPOSAL

Accidents	Latent Cancers/GWe-Year
Preclosure Failure	
Fuel Truck Crash Into:	
1) High-Level Waste Receiving	9×10^{-10}
2) Clad Waste Receiving	4×10^{-12}
3) TRU Waste Receiving	5×10^{-11}
4) Air Crash Into Receiving Area	2×10^{-10}
5) Elevator Drop	4×10^{-16}
6) TRU Pallet Drop	2×10^{-10}
7) Final Filter Failure	1×10^{-9}
Postclosure Failure	1×10^{-14}

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Section 9

SEISMIC HAZARD ANALYSIS

During the past year seismic research efforts have centered on three areas. The first area was the publication of past work. Over the past few years progress has been made in finding regularities in frequency-magnitude (f-M) relations for earthquakes when considering relatively large regions of the earth as compared to world total data. This effort involved a complex statistical analysis of available data bases, particularly that maintained by the National Geophysical and Solar-Terrestrial Data Center (NGSDC) (9-1). Preliminary results have been presented in a past EPRI report (9-2). Since that time two papers on the final outcome of this phase of seismic hazard analysis have been submitted for publication.

The success of establishing f-M relations to large areas has prompted a second study applying this method to smaller size regions. Thus, during the past year a significant effort has been made to show that a universal f-M shape is applicable to regions whose areas are comparable to those of the tectonic zones normally used for the estimation of seismic hazard.

The third effort, also initiated during the past year, is a separate investigation of strong-motion data from large seismic events. The results of the frequency-magnitude and strong-motion studies will eventually be combined to provide a probability density of acceleration at a given location.

FREQUENCY-MAGNITUDE (f-M) ANALYSIS PAPERS

The statistical analysis of the NGSDC data base (9-1) has resulted in two papers submitted for publication during the past year.

The first paper, entitled "Frequency-Magnitude-Time Relationships in the NGSDC Earthquake Data Base" (9-3), is slated for publication in December 1979. This paper discusses temporal completeness problems in the NGSDC data base of event recording and assigned event magnitude. The paper suggests techniques that could serve to circumvent most inaccuracies. In particular, the choice of event magnitude is important in terms of minimizing distortion in frequency-magnitude (f-M) distributions. A new working magnitude, M_3, is suggested as useful, where

$$M_3 = \begin{cases} M_s, \text{ if it is present} \\ \text{Max}(m_b, M_L, M_u),\ M_b < 6,\ M_s \text{ is not present} \\ \text{Max}(2.0m_b - 5.4),\ M_L, M_u), m_b \geq 6,\ M_s \text{ is not present} \end{cases}$$

The use of M_3 reduces distortion of f-M distributions derived from the NGSDC data base as compared to previous magnitude definitions used.

The second paper (9-4) records the observation of a universal shape regularity in earthquake f-M distributions. Frequency-magnitude distributions were obtained from sets of seismic events originating within widely separated regions of the earth. The regions have geological diversity and areas greater than 6×10^5 km^2. To within estimated error, the shape of total world data agrees with similar plots of data subsets taken from these eight separate regions of the earth.

APPLICATION OF f-M RELATIONS TO TECTONIC ZONES

The application of the universal f-M shape to the estimation of earthquake hazard is the principal goal of these seismic analysis efforts. The utility of this concept will be significantly enhanced if it can be shown that a universal f-M shape will hold for regions of less than continental size. To date this effort has taken two directions. The first has involved determining that earthquake data from independently defined (9-5) seismic source areas (a sum over several tectonic zones) are in agreement with the world f-M shape. The other direction of this study involves comparing geologic estimates of earthquake activity in an independently defined tectonic zone with those obtained from applying the universal curve to statistical seismic histories for that zone. The following examples are representative of these two types of investigation, which are currently under way.

Comparison of Regional Tectonic Zone Data with World f-M Shape

In order to produce a meaningful comparison with the shape of the world f-M curve, historical earthquake data must be gathered for regions of sufficient size to obtain statistically significant indications. Conversely, as mentioned previously, these regions must be smaller than continental size in order to prove applicability to seismic hazard analysis. In the initial f-M relation study reported in references 9-3 and 9-4, one region, the California/Nevada region, had an area of 0.6 x 10^6 km^2 (significantly less than the others). The rather complete earthquake data base for the California/Nevada region is supplemented by its relatively high frequency of earthquake occurrence. Figure 9-1 depicts the complete agreement of the shape of the f-M curve for the California/Nevada region with the world f-M curve.

The regions shown in Figure 9-2 encompass parts of Washington, Oregon, Nevada, Utah, and Colorado, as well as all of Idaho. The area of this region is 0.9 x 10^6 km^2. This region is geologically quite different from the California/ Nevada region in that it is representative of an intracontinental region. Results of a statistical analysis of the earthquake f-M curve for this region are shown in Figure 9-3. Agreement with the world f-M shape is excellent, indicating the world curve can be used in areas as small as 10^6 km^2.

Comparison of Geologic Estimates of Seismic Activity with Estimates Obtained Using Historical Data and the World f-M Curve

In order to apply the concept of a universal f-M curve to earthquake hazard analysis, the successful use of the universal curve is required in areas at least as small as tectonic zones. A number of studies of the tectonic zoning of the United States have been made (9-5,9-6). In this study the zoning of Algermissen and Perkins (9-5) has been chosen as representative.

In attempting to demonstrate the applicability of the universal f-M curve to areas as small as tectonic zones (typically 10^4 - 10^5 km^2), one is forced to confront the problem of limited statistics for instrumentally determined earthquakes. This situation makes a direct comparison of f-M shapes indeterminate. Therefore, a decision was made to first obtain a geological estimate of earthquake activity in a few of the well-defined tectonic zones of Algermissen and Perkins (9-5). Then another estimate in the same zones is

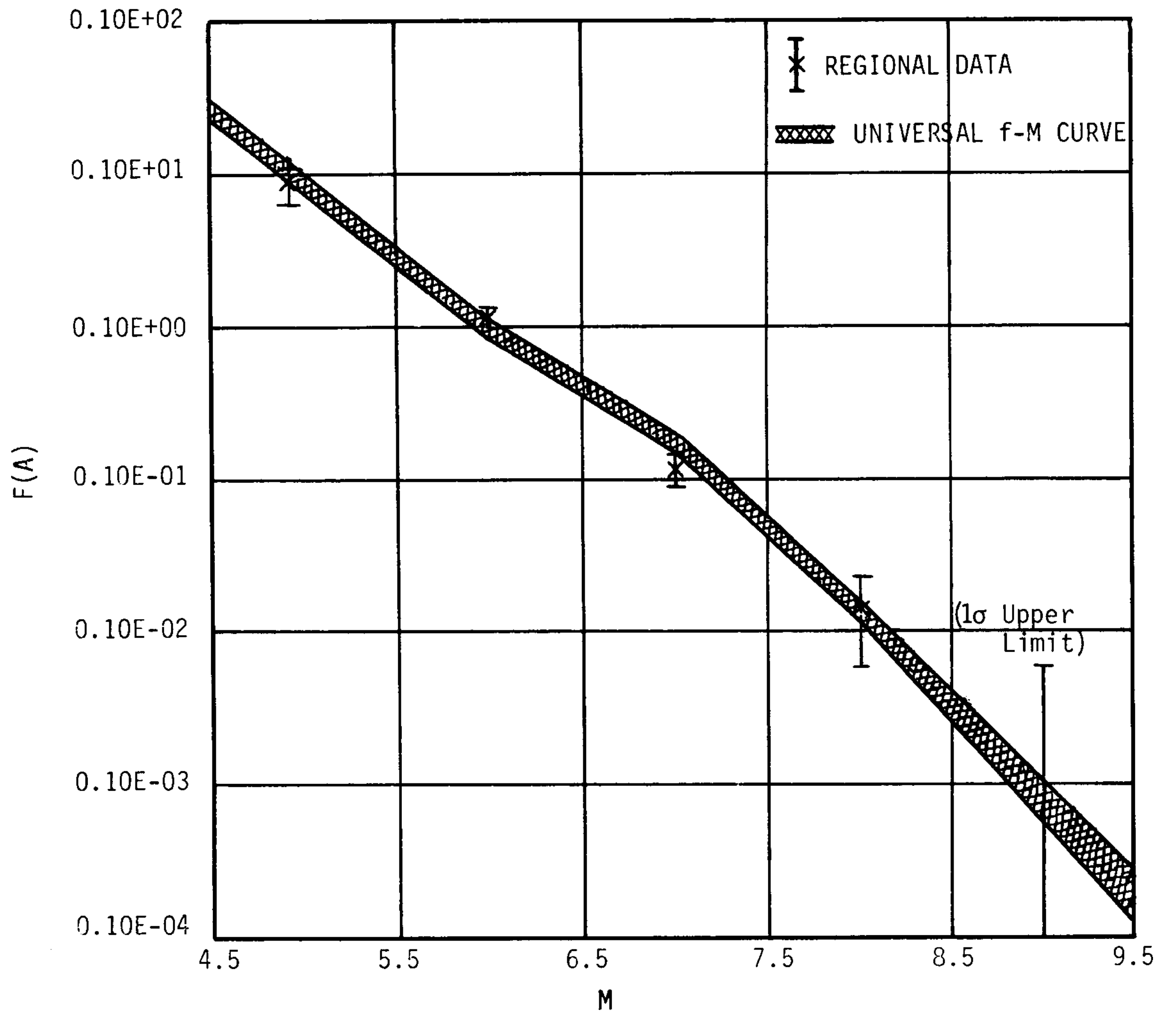

Figure 9-1. Comparison of the f-M Curve for California/Nevada Region with World f-M Curve

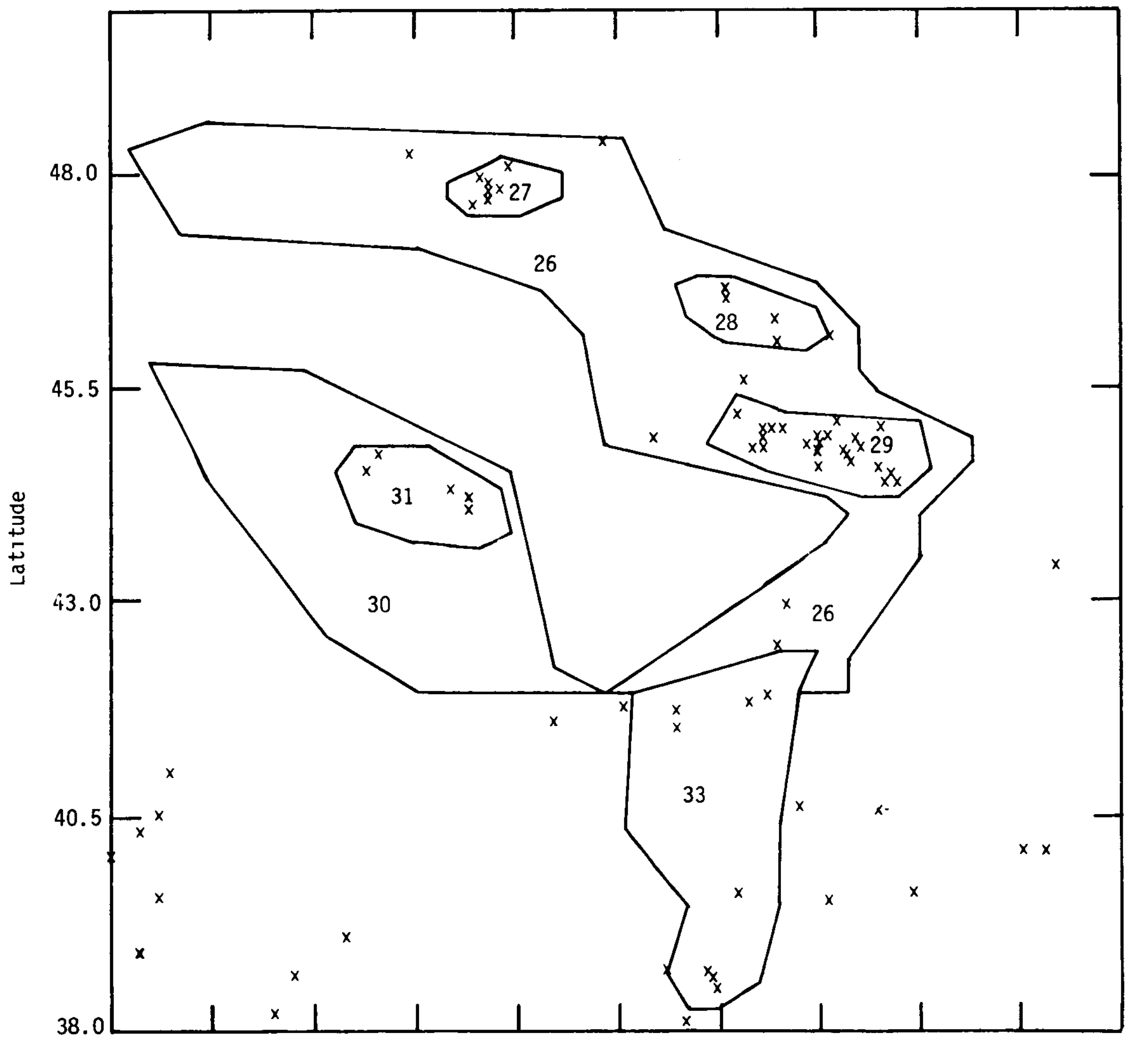

Figure 9-2. Intracontinental Region Centering on Idaho

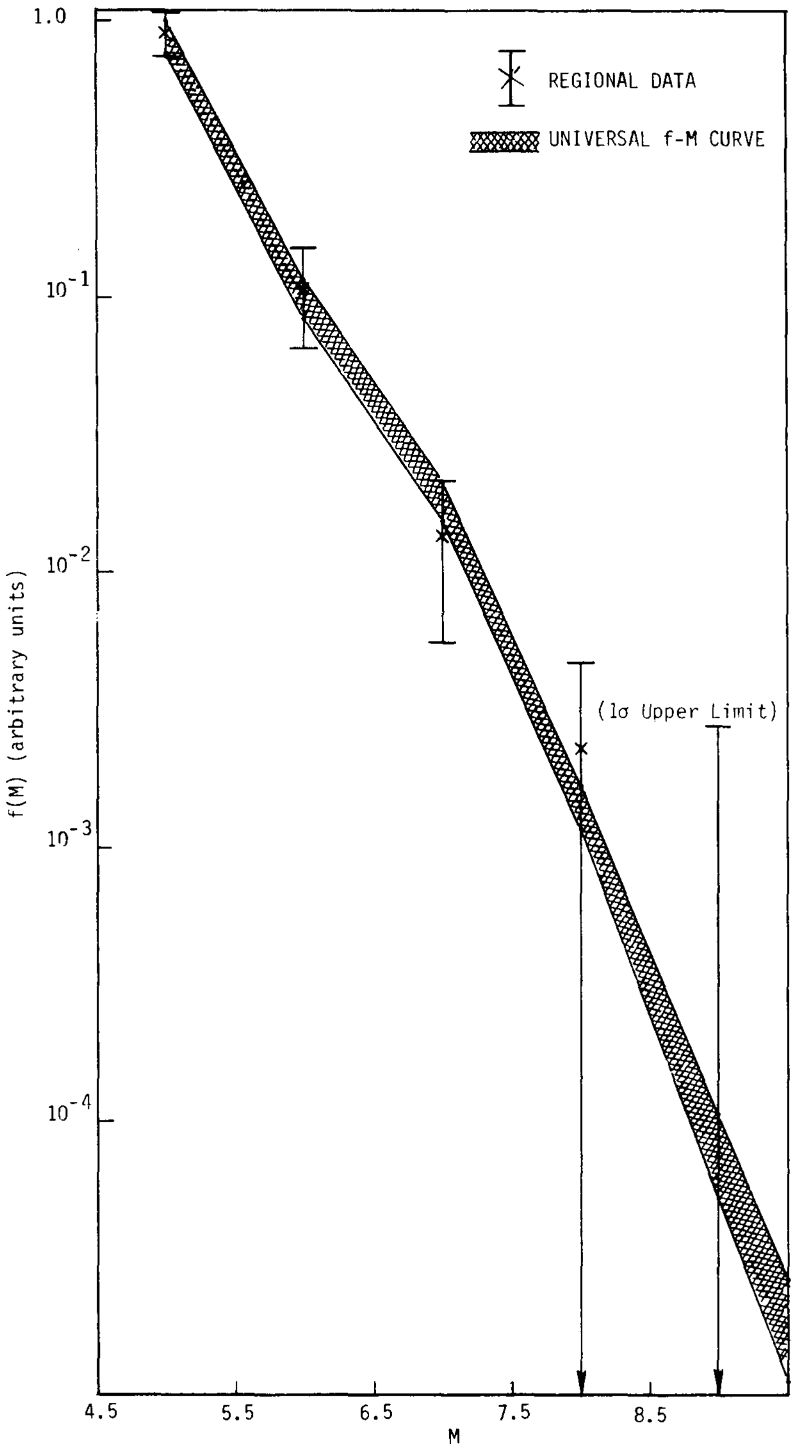

Figure 9-3. Statistical Analysis of f-M Curve for Intracontinental Region Centering on Idaho. Comparison of world f-M curve and regional data. The agreement of shape is within error.

obtained by fitting the universal f-M curve to the frequency of instrumentally determined earthquakes. It should be noted that, in such an application of the universal f-M curve, the lower magnitude ($M \gtrsim 4.5$), instrumentally measured earthquakes are heavily weighted such that the frequency of occurrence of the larger events ($M \gtrsim 7$) is essentially being derived from the frequency of the lower-magnitude events. The comparison is then made between the universal f-M curve estimate of the return period for events with $M \gtrsim 7$ and the geologically determined return period for events of this magnitude range for the particular tectonic zone being considered. The universal f-M curve method emphasizes short times (~50 years) and lower event magnitudes. The geologic method emphasizes long times ($\sim 2 \times 10^3$ years) and large (but uncertain) event magnitudes.

One particularly well-studied tectonic zone is the San Andreas Fault zone in California. The precise zone boundaries used here have been defined by Algermissen and Perkins (9-5) as their zone 2, shown in outline in Figure 9-4. This figure also shows the location and magnitude range of earthquakes in the San Andreas zone, as obtained from the 1976 NGSDC data tape (9-1). The data extend back in time to 1900; however, as pointed out in one of our recent publications (9-3), the events have an important time bias. This bias, a lack of recording of the smaller events, becomes progressively more severe as earlier times are considered. After correcting for this time bias, as described in Reference 9-4, a fit to the world f-M shape was accomplished. Return periods for earthquakes in the San Andreas tectonic zone were obtained as a function of magnitude, as shown in Figure 9-5. Note the uncertainty in return period, which is generated by an uncertainty of $\pm$ 0.25 in M. This completes the second step of the statistical-geologic comparison outlined in the beginning of this section.

In order to obtain a geological estimate of return period for portions of the San Andreas tectonic zone, the recently reported work of Sieh (9-7) is used. Sieh conducted fault offset studies along the central reach (Wallace Creek) and along the southern reach (Pallet Creek) of the San Andreas Fault. At Wallace Creek he found a mean return period for events with $M \gtrsim 7.5$ of 255 $\pm$ 30 years by examining the offset of the active channel of the creek. At Pallet Creek, using trenching techniques, he identified nine events with $M \gtrsim 7.5$ going back 1500 years. The dates of each of the events were determined with carbon dating and had errors of typically $\pm$ 50 years. The average return

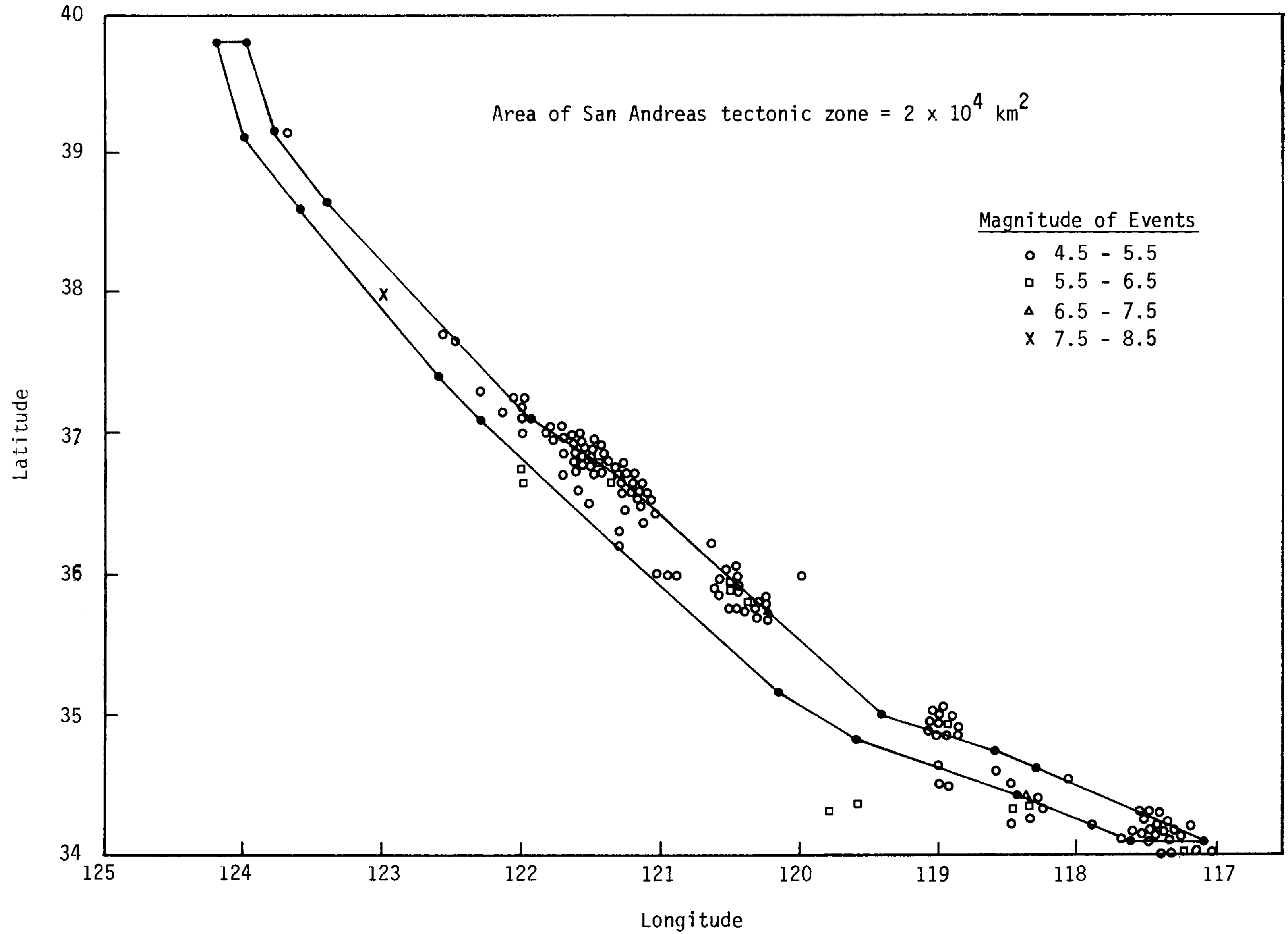

Figure 9-4. NGSDC Data Tape (9-1) Estimate of Seismicity in Algermissen and Perkins (9-5) Region 2 (San Andreas Fault Zone). Events of various magnitudes are shown approximately at their epicentral location. The lines indicate the boundary of the tectonic zone.

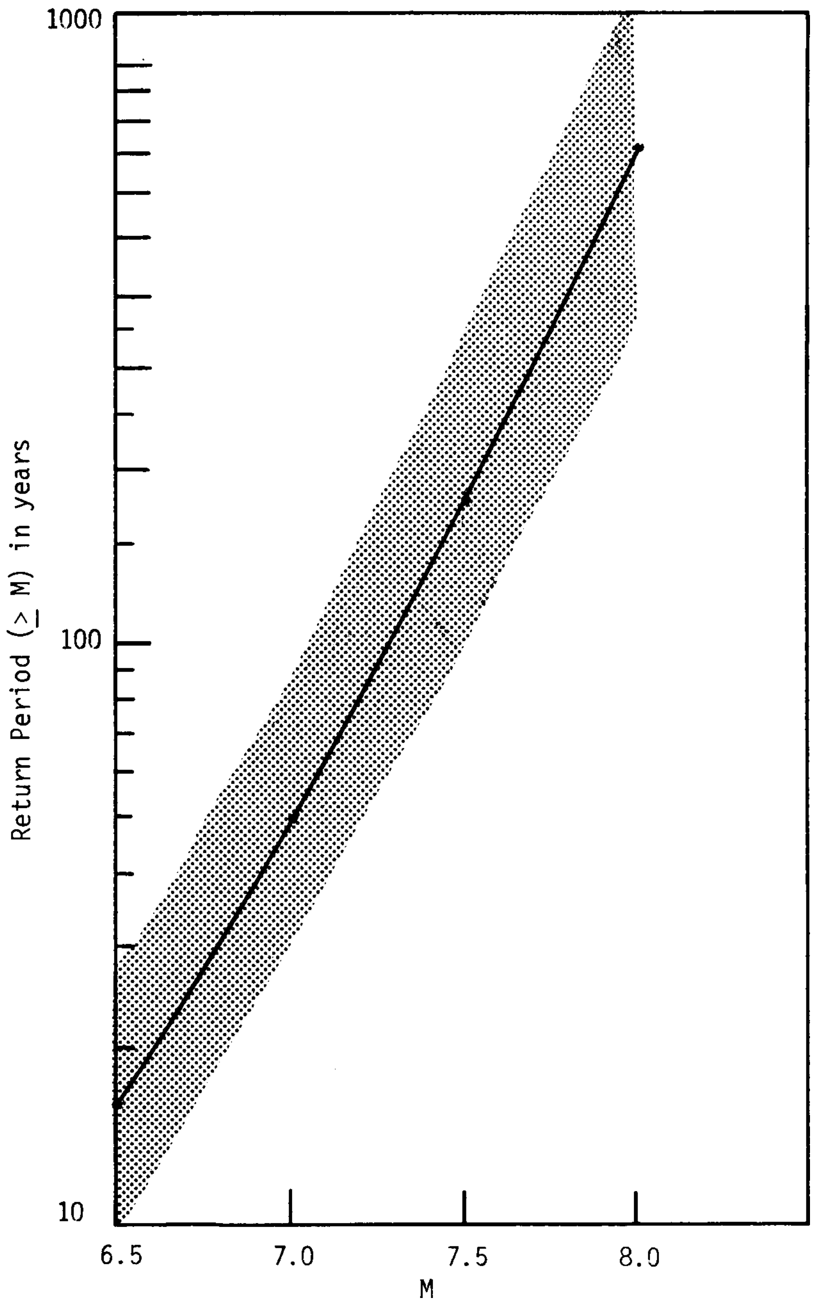

Figure 9-5. Return period for earthquakes with magnitude greater than or equal to M as a function of M. This curve is calculated from application of the universal f-M curve to earthquake data obtained from the NGSDC data tape (9-1). The shaded area represents an uncertainty in return period generated by an uncertainty of magitude of ± 0.25.

period obtained at Pallet Creek from the nine observed events was 153 years. It should be noted that there may be a major source of error in the above described approach. The geologically determined magnitude must be compared to a magnitude that is defined instrumentally. The uncertainties that could result from this procedure can be illustrated by an examination of the 1940 earthquake in Imperial Valley, California (9-7). The present estimated instrumental magnitude for this event is 6.4; however, previously published magnitudes have varied from 6.7 to 7.1. The confusion arises from the vast surficial effect of the earthquake compared to its recorded instrumental magnitude. Geologists would most likely have assigned this event a magnitude of more than 7.25 on the basis of length of faulting, size of displacement, and other physical effects. It is thus essential that an error margin of at least ± 0.5 be assigned to magnitudes when assessing geologically determined return periods.

The question remains of how universal f-M curve and geological results should be compared. By using the Pallet Creek data, an upper limit of 153 years on the return period over the tectonic zone for events with $M \gtrsim 7.5$ (± 0.5) is obtained. This geologically determined upper limit corresponds to a statistically determined return period of 50 to 720 years for M of 7 to 8 for the entire tectonic zone (using Figure 9-5). How to estimate a geologically determined return period for the entire tectonic zone from the above data is not presently clear. However, given the possible errors in magnitude determination, it appears that no conflict exists between these geologically and statistically determined return periods. Work is continuing to further refine this approach and to allow tests of universal f-M curve application to yet smaller geographical areas.

STRONG-MOTION STUDIES

Progress in strong-motion studies developed in three major areas during the past year:

- Mathematical modeling of surface waves
- Inspection and data reduction of the NRC strong-motion data base (9-9)
- Development of a semi-empirical model based on the above two sources for predicting the probability of strong motion at a field site point

Mathematical Modeling

The objective of this first phase of strong-motion records study has been to develop a mathematical representation of earthquake wave propagation. The relationships among physical parameters in such an idealized case can be a guide to a judicious analysis of the strong-motion data base.

A simplified model was constructed in order to render the mathematical techniques and results tractable. The model is that of an isotropic scaler surface wave initially represented as a Gaussian displacement at the earthquake epicenter. Solution of the wave equation using the Hankel transform method (9-10) has led to the acceleration spectrum

$$\tilde{a}\,(\Omega,\rho) \sim E^{n}\sigma^{-(1+nm)}\Omega^{3}e^{-\Omega^{2}/2}J_{0}\,(\Omega\rho) \qquad (9\text{-}1)$$

where

E = energy released

σ = fault dimension

$\Omega = \frac{w\sigma}{c}$ = dimensionless frequency

$\rho = r/$ = dimensionless epicentral distance

A plot of a versus Ω and ρ is shown in Figure 9-6. The Fourier spectrum was then transformed into the time domain and the peak acceleration noted as the wave passed over each epicentral distance. A relationship linking peak acceleration to the distance from the site of energy release then emerged as

$$\log\,(a_{peak}) = -1/2\log(r) + C \qquad (9\text{-}2)$$

or

$$a_{peak} \sim \frac{1}{r^{1/2}} \qquad (9\text{-}3)$$

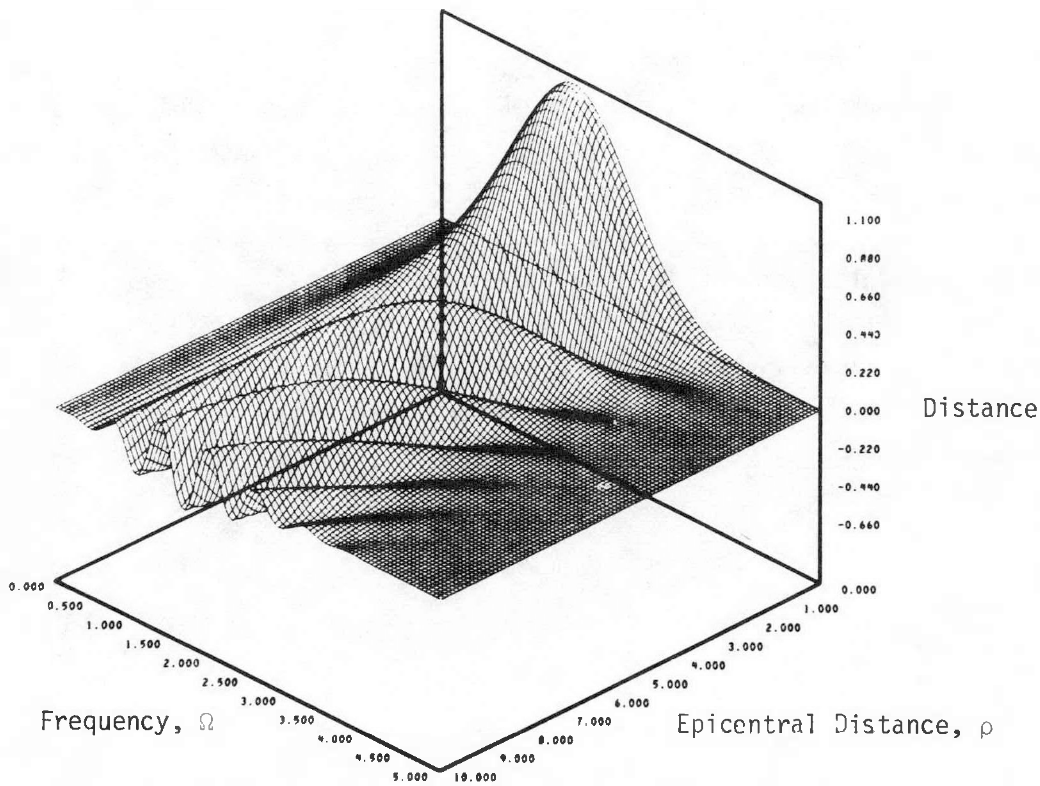

Figure 9-6. Acceleration Spectrum. Axes are in Normalized Units

Inspection of the NRC Data Tape

The NRC data tape (9-9), containing about 1300 strong-motion records, was examined to extract attenuation behavior data. Fourteen earthquakes from 1937 to 1972 in Southern California were examined. Magnitudes of these events ranged from 3.2 to 7.7, and records consisted of as few as six reports in one case to as many as 99. In each case the decline of peak acceleration with epicentral distance closely followed the simple power law

$$\frac{a}{a_0} = \frac{1}{1 + \left(\frac{r}{r_0}\right)^{-\alpha}} \qquad (9\text{-}4)$$

or

$$\log a \sim -\alpha \log r \qquad (9\text{-}5)$$

The value of α (the exponential attenuation coefficient) was determined using a least squares linear regression technique. The value of α was found not to be a universal constant; instead it varied outside of statistics from earthquake to earthquake. The value of α was shown not to correlate with event magnitude, focal depth, time, or average station geology. It was noticed, however, that α was strongly correlated with event location.

Predictive Calculations

Plotting all 14 values of α on a map revealed lines of constant attenuation (isoattenuation contours) that are very nearly concentric rings centered around the Los Angeles Basin area. The value of α is associated only with the distance, r, from a central point (33.7N, 118.3W) as

$$\alpha(r) = 6.70 \times 10^{-3} \; r + 0.83 \tag{9-6}$$

where r is in kilometers. A least squares linear regression analysis was employed to obtain these coefficients and demonstrated a correlation coefficient of 88%.

The dependence of α in the Los Angeles Basin area then inspired development of a computer code that propagates an earthquake of given magnitude, focal depth, and location to any field point. This code, written and run on a microcomputer, allows an operator to compute ground shaking at a desired location due to a single earthquake, or to a stochastic array of energy sources whose magnitudes, depths, and locations are made to follow empirical distributions. It has been shown that, given a single event, an array of recording stations can report values of peak acceleration in good agreement with observed data. It appears that "scatter" in actual reported data is due more to the variation in α for equal epicentral distance than to instrumental or random error.

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Section 10

RAPID RESPONSE TO THREE MILE ISLAND

The event at the Three Mile Island (TMI) nuclear power plant's Unit Two occurred on March 28, 1979. In the following days EPRI received a request to supply assistance in probabilistic analysis to Burns and Roe, the architect/engineering firm that designed the plant. Two members of the EPRI probabilistic analysis group spent one week assisting the Burns and Roe task force evaluating the reliability and safety of existing and alternative systems being used at TMI.

There was concern at the site about reliability of electric power to the plant. The NRC had expressed similar concern during the TMI2 licensing process, with the result that some changes were made concerning potential hookup with TMI1.

Task 27, assigned to the reliability section of the Burns and Roe task force, was described as follows:

> Evaluate the reliability of off-site electric power and other critical systems, including Decay Heat Removal (DHR) and a recently designed temporary system alignment to use the secondary side of the steam generators (filled with water) for emergency feedwater to cool the plant down further.

A separate request was also made for a rapid-response comparison of designs of nuclear steam supply systems (NSSS) of the three U.S. vendors of pressurized water reactors (PWR).

EVALUATION OF RELIABILITY OF OFF-SITE ELECTRIC POWER

The off-site electric power connection at TMI2 consists of three lines of 230 kV and and two lines of 500 kV. Two of the 230 kV lines are strung on the same poles, but the third line is on separate poles. Dedicating one of these lines to TMI2 did not offer a significant increase in the reliability of off-site power.

In the history of commercial nuclear power plant operation in the United States, there have been only three incidents involving simultaneous loss of off-site and on-site power. All three incidences occurred as a result of adverse weather conditions. It was concluded that the local central Pennsylvania weather season was not conducive to a similar occurrence.

EVALUATION OF DECAY HEAT REMOVAL (DHR) SYSTEM

In order to determine the weak points in the Decay Heat Removal and Decay Heat Closed Cooling Water systems, the fault tree diagrams presented in Figures 10-1 through 10-12 were constructed.

It was found that the positions of certain valves could reduce system reliability significantly. In particular, the manual isolation valves could have been left out of position following maintenance. The positions of these valves are apparently not indicated in the control room.

In addition, certain motor-operated valves have to change position in order to activate decay heat removal. It was recommended that these valves be put in the DHR-mode position to minimize the total number of valve position changes required for activation without sacrificing isolation from the reactor core system.

The DHR system was not intended for use at the time of this investigation. However, a loss of off-site power prior to installation of back-up modifications could have resulted in slow depressurization due to loss of pressurizer heaters. In such an event there would apparently have been no other alternative to depressurizing and activating DHR. It was therefore necessary to make preparations to utilize DHR in an emergency. A verification of valve positions, as indicated in Table 10-1, would result in a greatly enhanced probability of successful system operation in such an emergency.

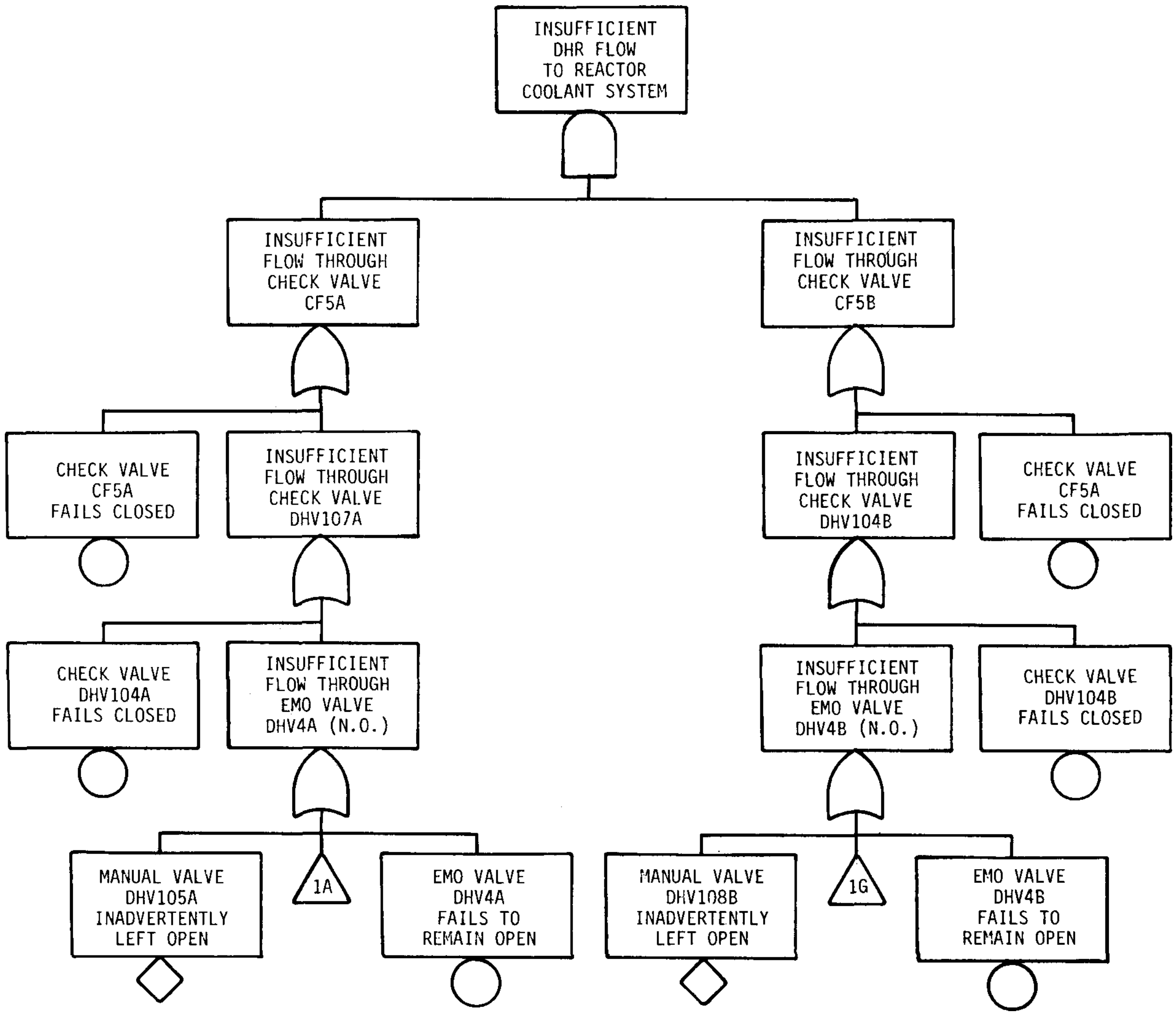

Figure 10-1. Fault Tree Analysis of TMI2 Decay Heat Removal (DHR) System

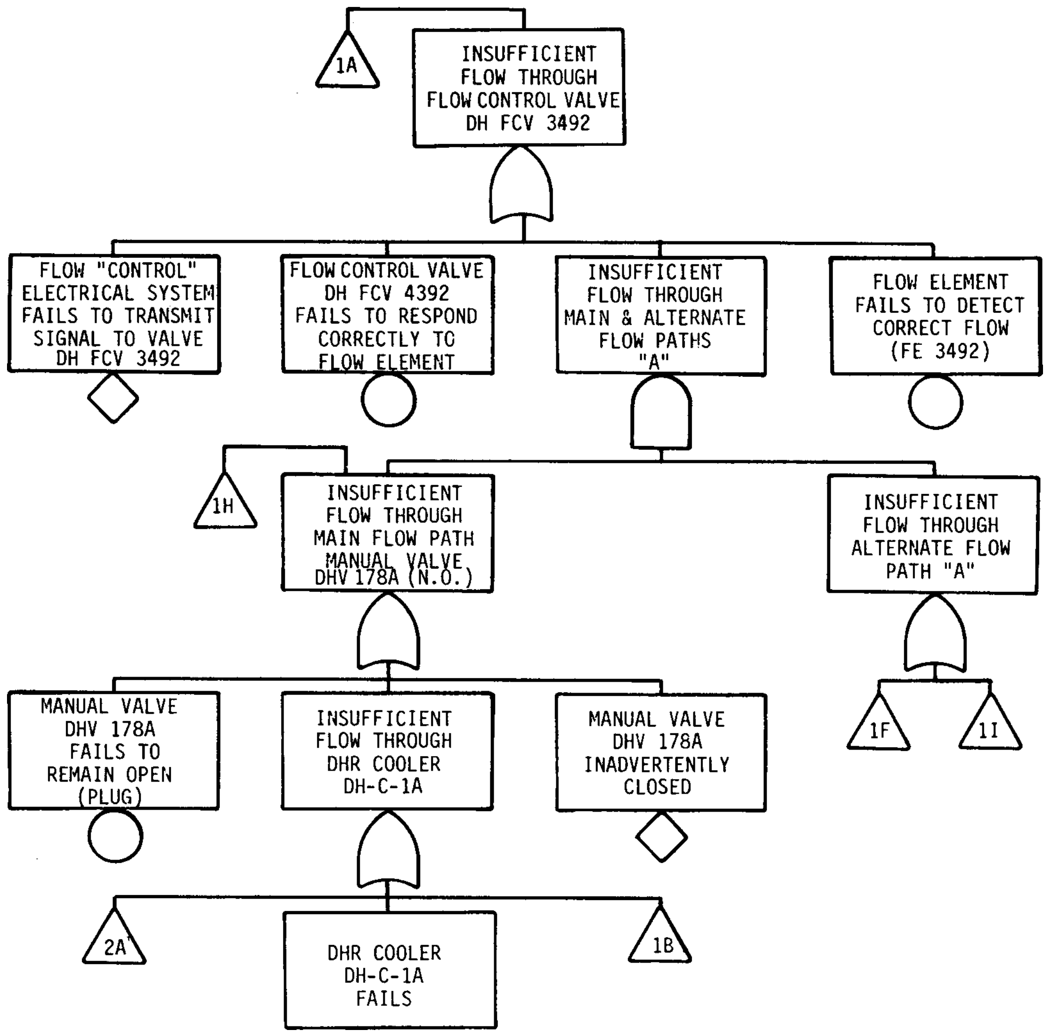

Figure 10-2. Fault Tree Analysis of TMI2 DHR System

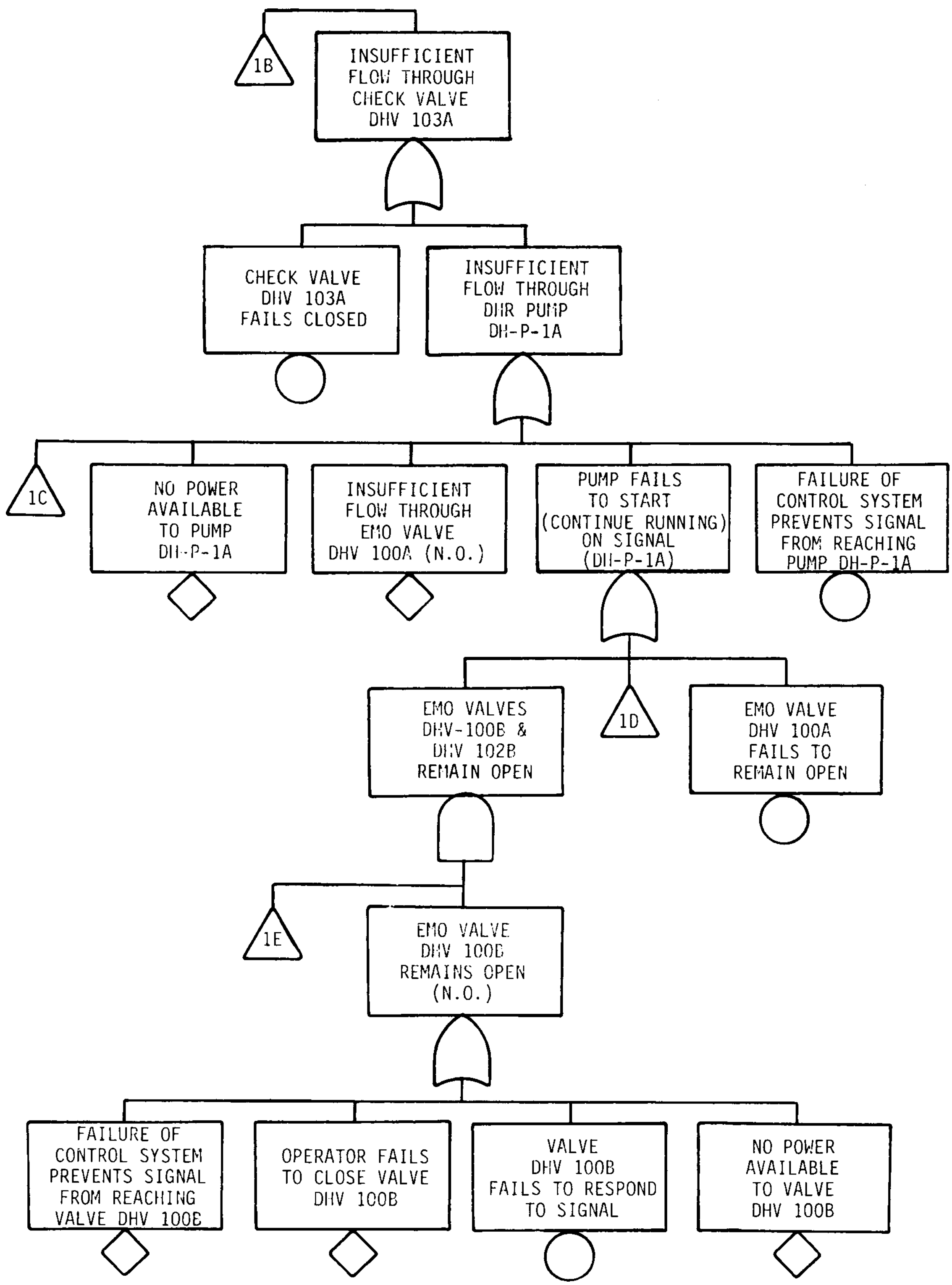

Figure 10-3. Fault Tree Analysis of TMI2 DHR System

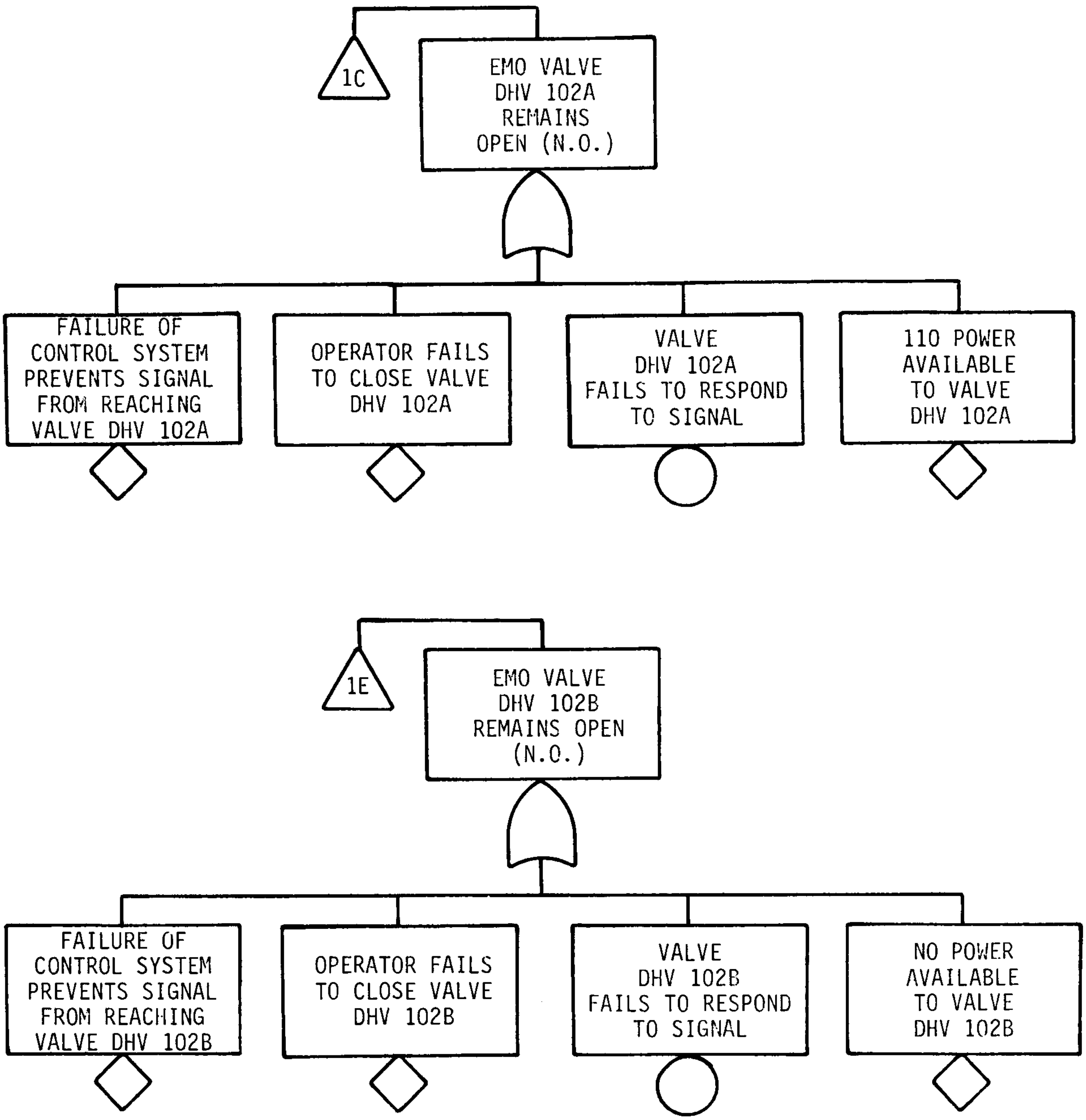

Figure 10-4. Fault Tree Analysis of TMI2 DHR System

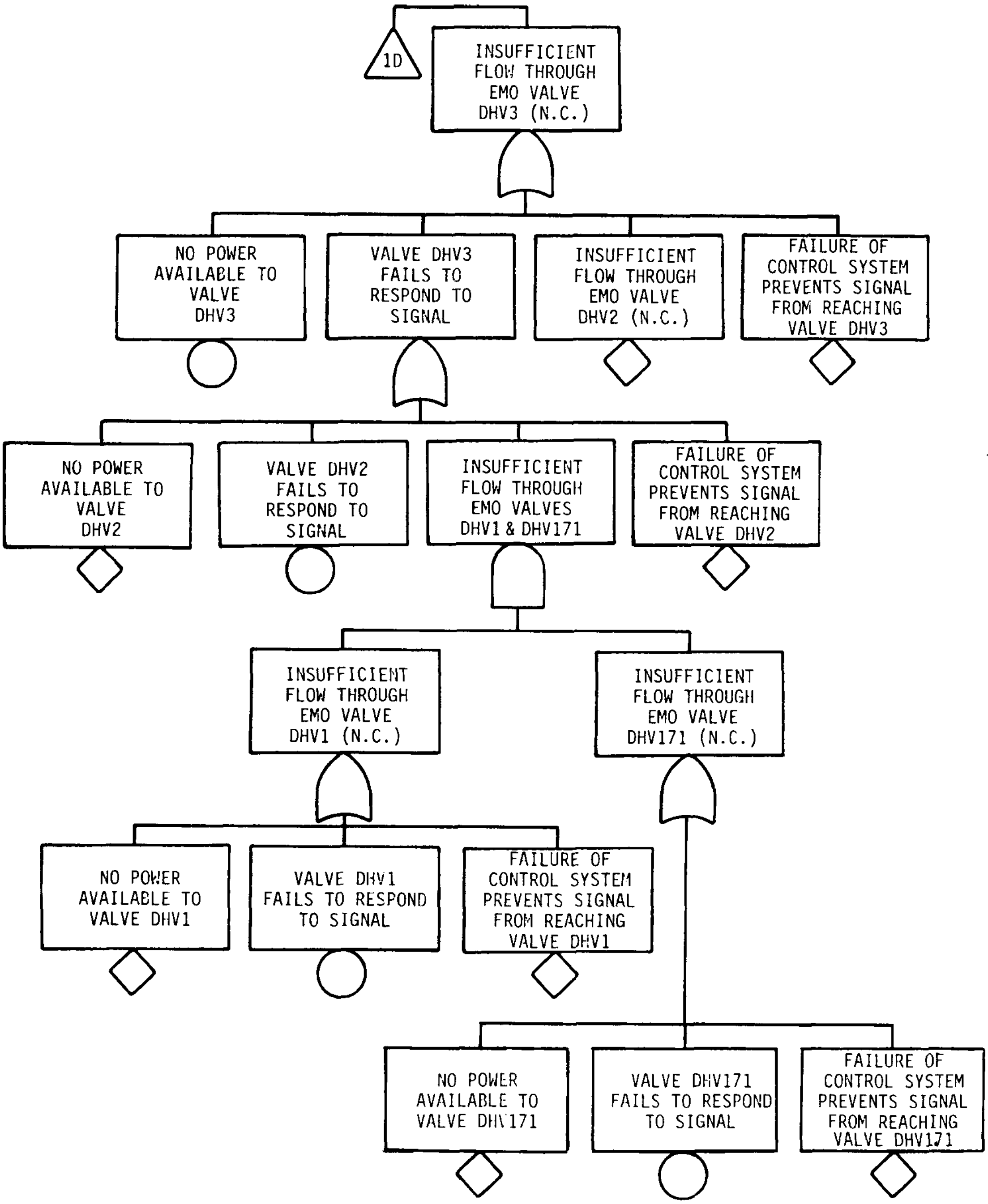

Figure 10-5. Fault Tree Analysis of TMI2 DHR System

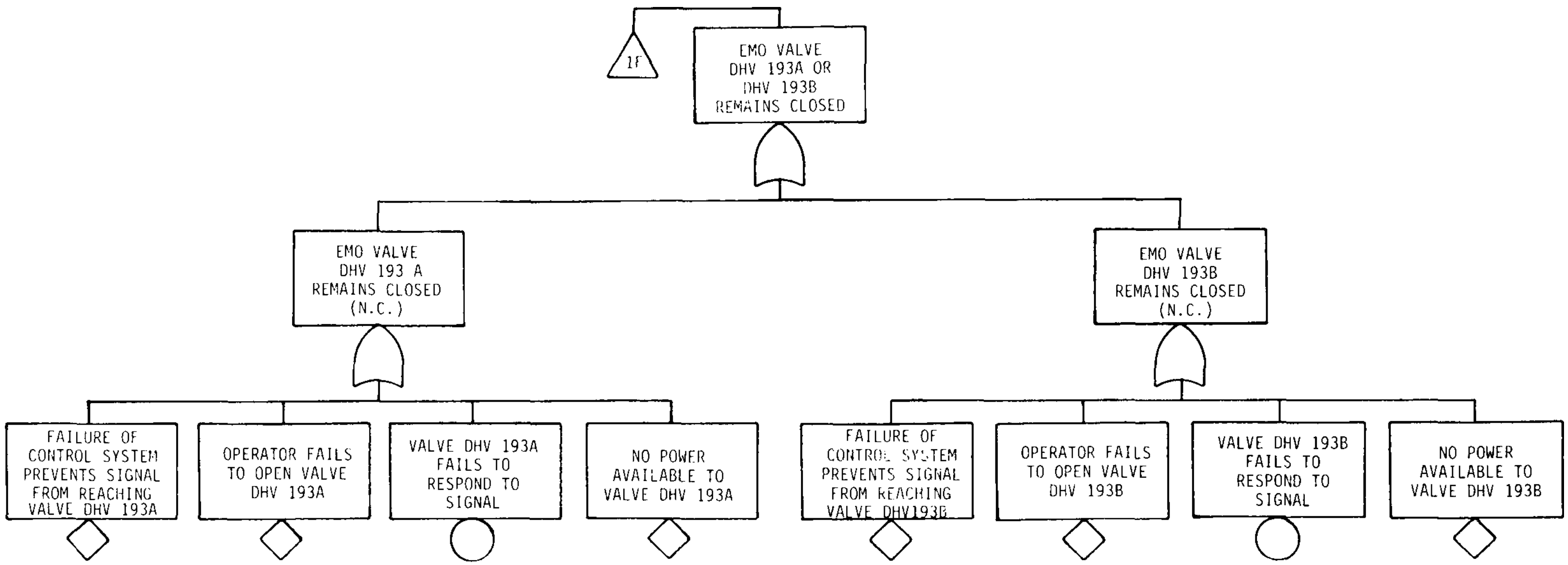

Figure 10-6. Fault Tree Analysis of TMI2 DHR System

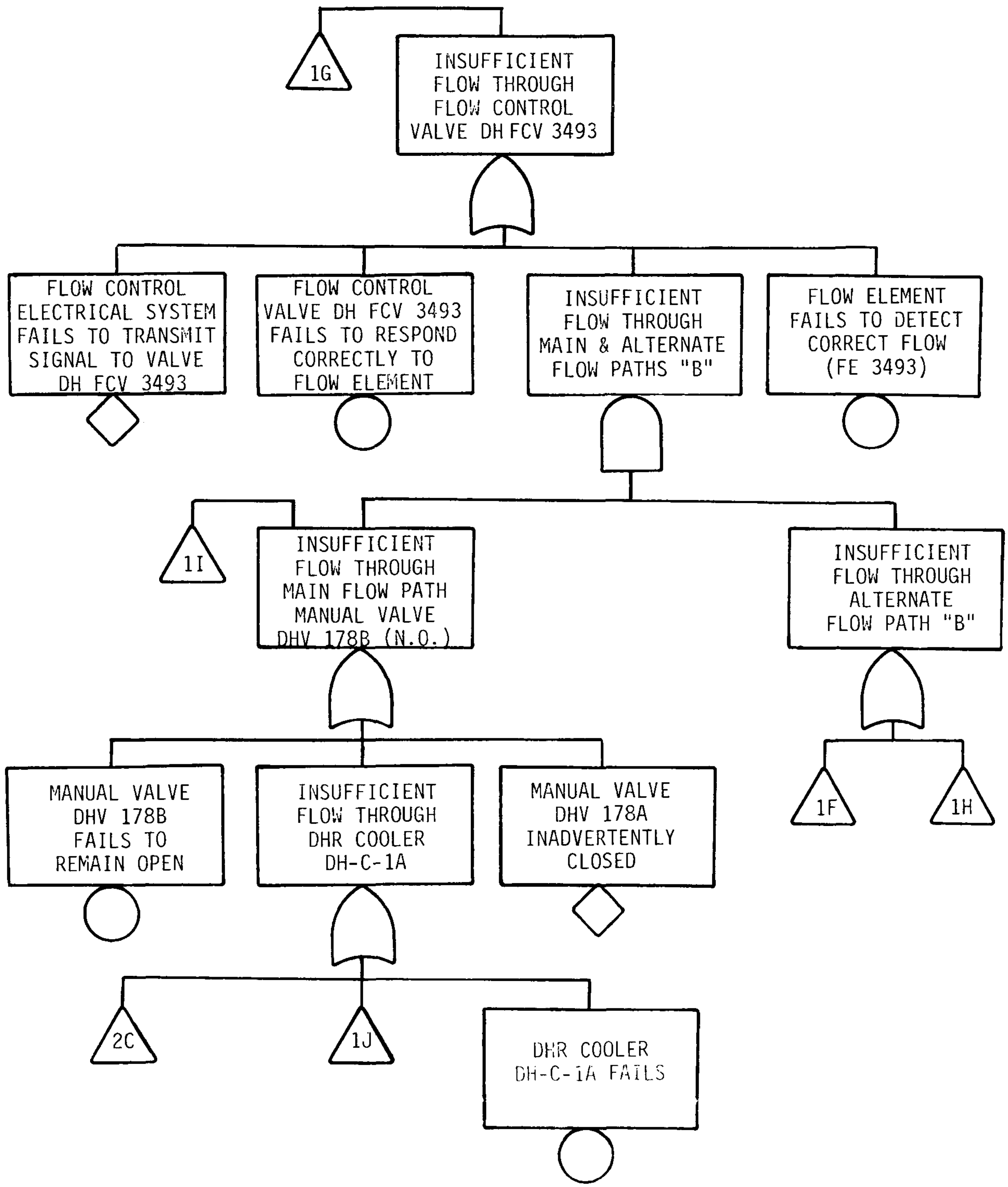

Figure 10-7. Fault Tree Analysis of TMI2 DHR System

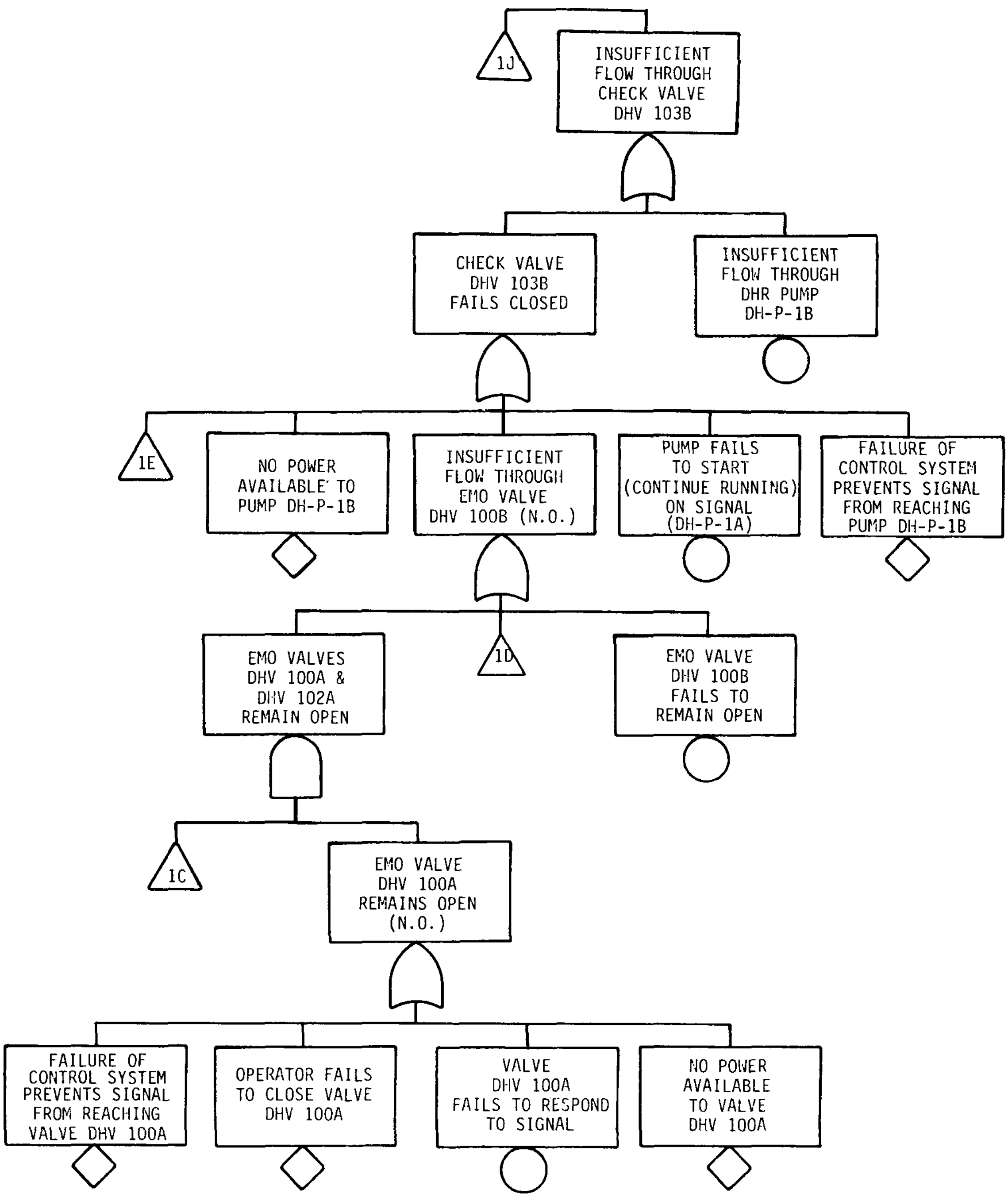

Figure 10-8. Fault Tree Analysis of TMI2 DHR System

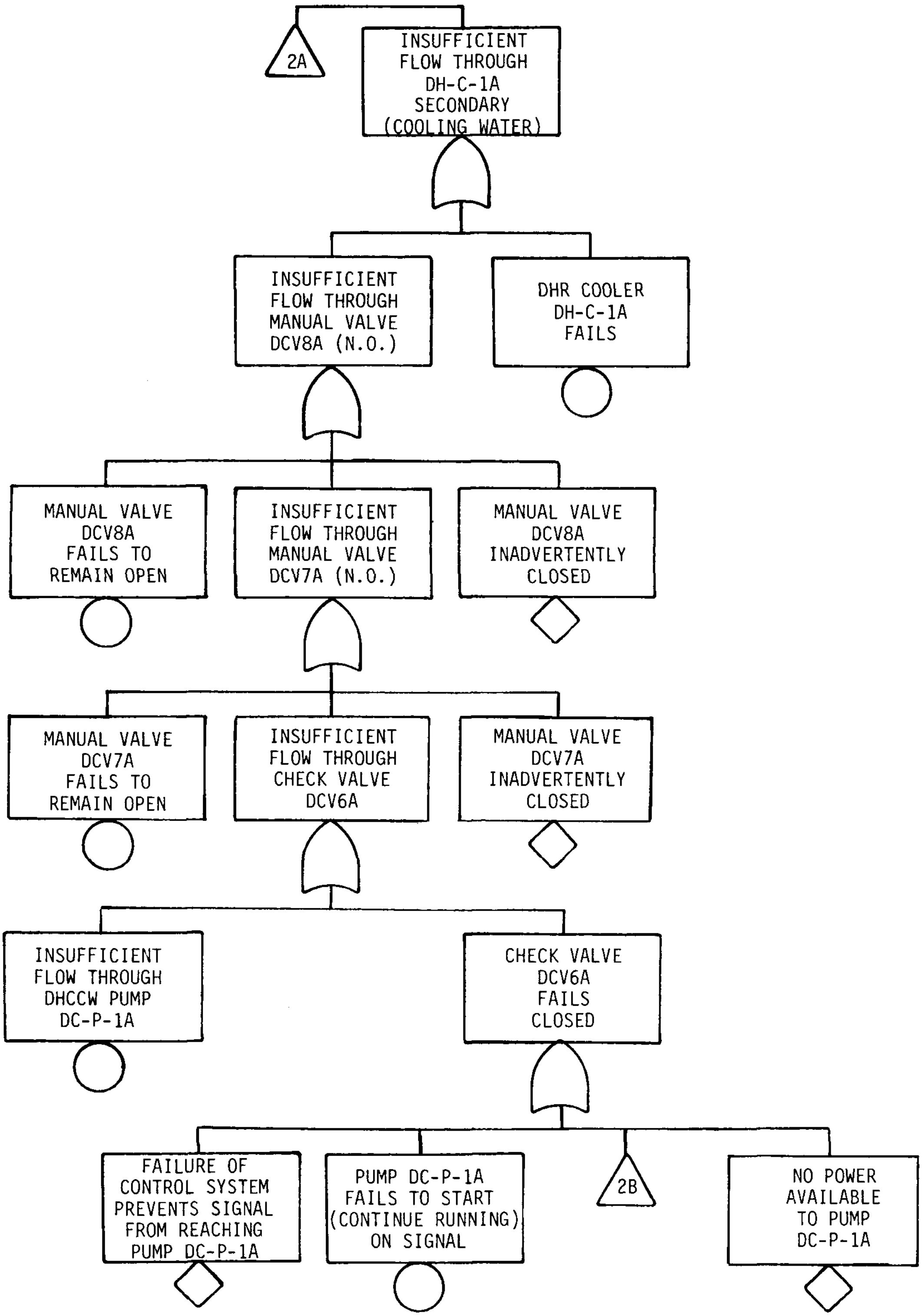

Figure 10-9. Fault Tree Analysis of TMI2 DHR System

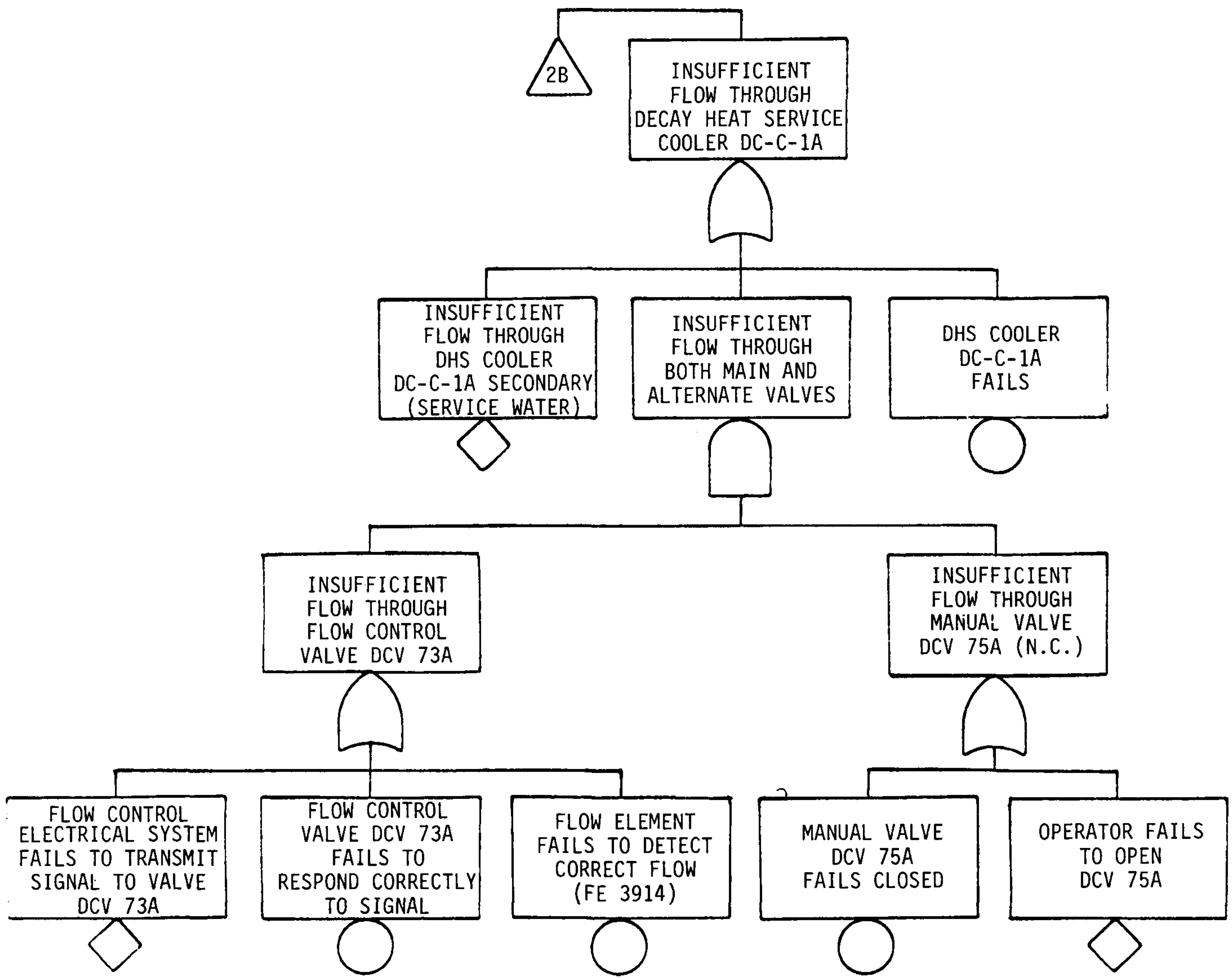

Figure 10-10. Fault Tree Analysis of TMI2 DHR System

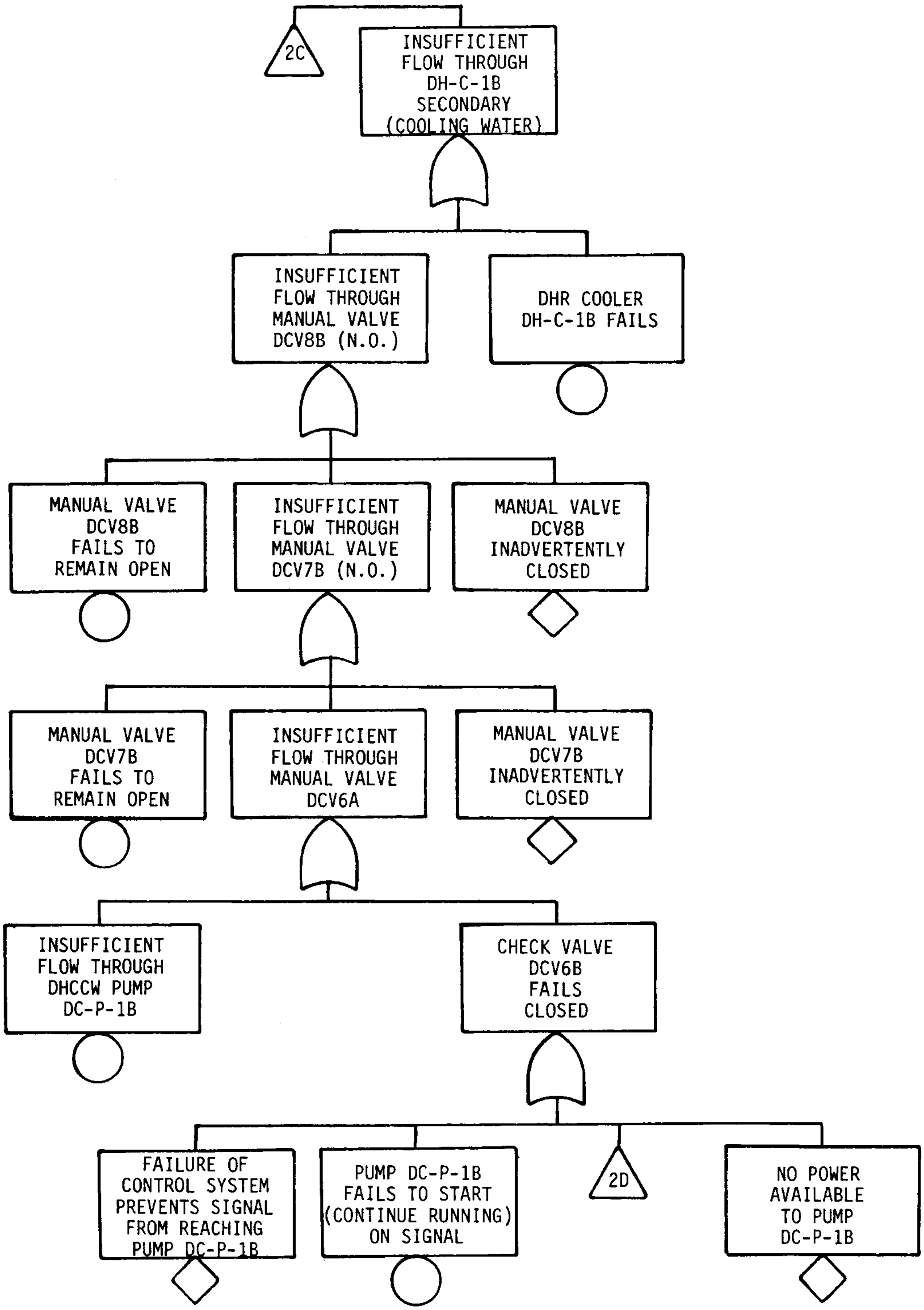

Figure 10-11. Fault Tree Analysis of TMI2 DHR System

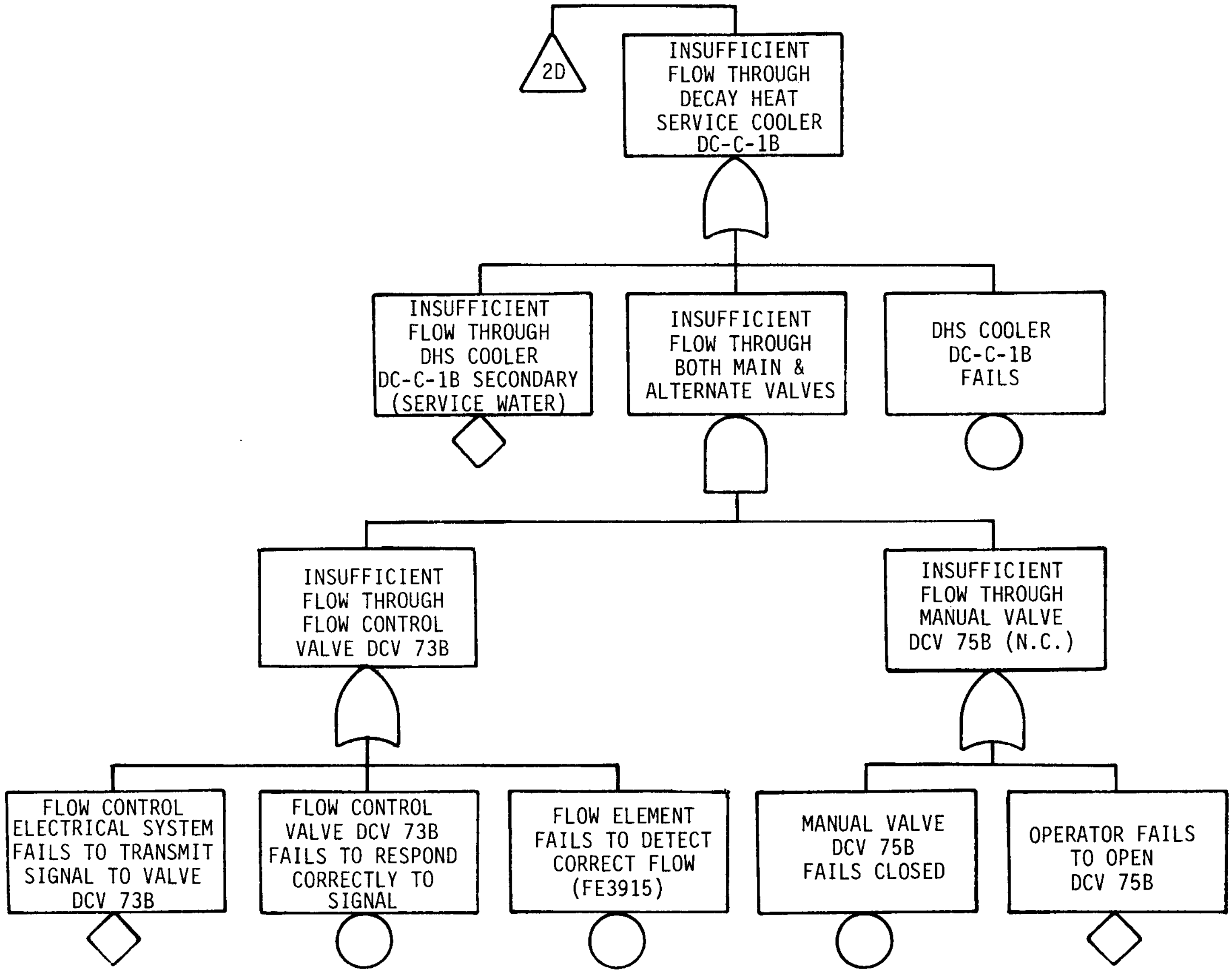

Figure 10-12. Fault Tree Analysis of TMI2 DHR System

Table 10-1

DECAY HEAT REMOVAL VALVES

VALVE #	DWG	ACTUATOR	DESIRED POSITION	COMMENTS
DH-V-4A	2026	Electric Motor	Open	
DH-V-108A	2026	Manual	Closed	
DH-V-178A	2026	Manual	Open	
DH-V-100A	2026	Electric Motor	Open	
DH-V-102A	2026	Electric Motor	Closed	
DH-V-3	2026	Electric Motor	Open*	*Verify DH-V-1, DH-V-2, and DH-V-171 Closed Before opening DH-V-3 (All Electric Motor Operated)
DH-V-4B	2026	Electric Motor	Open	
DH-V-108B	2026	Manual	Closed	
DH-V-178B	2026	Manual	Open	
DH-V-100B	2026	Electric Motor	Open	
DH-V-102B	2026	Electric Motor	Closed	
DH-V-193A	2026	Electric Motor	Open	
DH-V-193B	2026	Electric Motor	Open	
DC-V-8A	2035	Manual	Open	
DC-V-7A	2035	Manual	Open	
DC-V-8B	2035	Manual	Open	
DC-V-7B	2035	Manual	Open	
DR-V-3	2033	Electric Motor	Open	
DR-V-197	2033	Electric Motor	Open	
DR-V-40A	2033	Electric Motor	Open	
DR-V-40B	2033	Electric Motor	Open	Allows Flow Through Decay Heat Service Coolers
DR-V-123A	2033	Electric Motor	Open	
DR-V-123B	2033	Electric Motor	Open	

EVALUATION OF PROPOSED TEMPORARY EMERGENCY FEEDWATER SYSTEM

At the time of this analysis, a temporary system alignment had recently been designed to provide emergency feedwater via the secondary side of the steam generators (filled with water) to further cool down the plant. This proposed system was subjected to fault tree analysis. Although no apparent problems were found with the proposed system alignment, it did not offer the redundancy and reliability inherent in a modified design, to include a new temporary crossover and diesel-driven pumps, defined on April 3, 1979. Figure 10-13 is a simplified schematic drawing of this later design, which was analyzed with the fault trees shown in Figures 10-14 through 10-25.

This system eliminated all potential single failures existing in the previously proposed temporary system. It offered a double redundancy from steam generator A back to the main feedwater pump FW-P-1A and the emergency feedwater pump EF-P-2A. The addition of the diesel-driven emergency feedwater pump provided a mechanical triple redundancy at the pumps. Furthermore, it provided backup for the emergency power, which in turn backed up off-site power.

The modification also offered double redundancy in the pump suction subsystem. A new heat exchanger incorporated in the modification provided an ultimate heat sink for the secondary water system, enabling conservation of secondary cooling water.

It was recommended that a check valve be added on the main feedwater line side of the new throttling valve. The reason for this recommendation was based on the fact that the length of new pipe connecting the emergency feedwater to the main feedwater was not accurately known. The check valve would in any case prevent loss of main feedwater due to an emergency feedwater failure.

It was further recommended that the normally open valves EF-V5A and EF-V5B (on the B leg) be closed. This would avoid failures in the A system that could result from failures in the B system.

The addition of the two new diesel-driven pumps would bring the total number of pumps available for the temporary system to six. In light of the fact that the likelihood of off-site power failure was exceedingly low, it was concluded

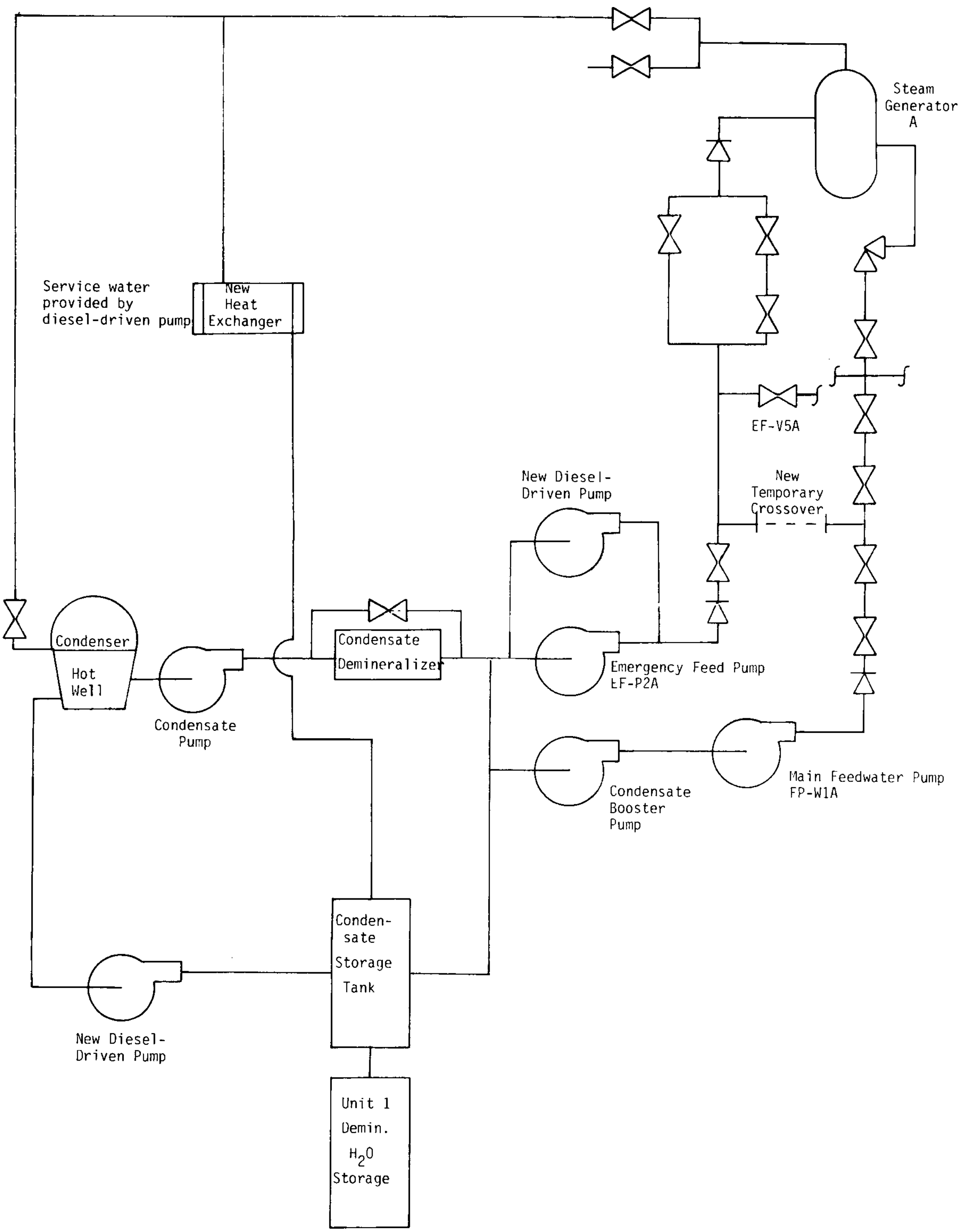

Figure 10-13. Revised A System Steam Generator Loop

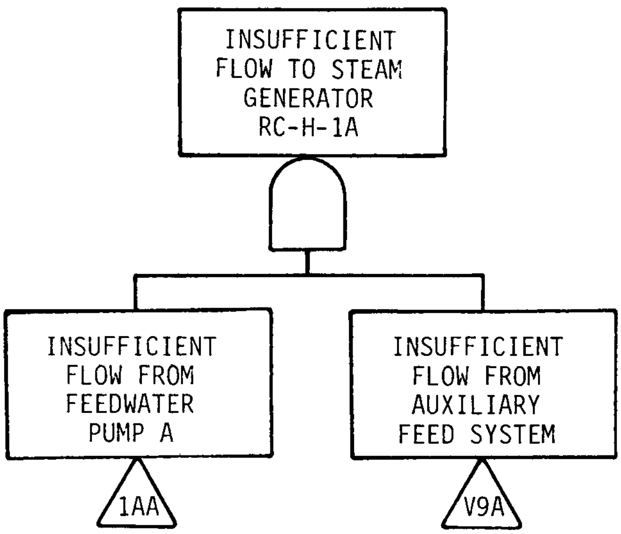

Figure 10-14. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

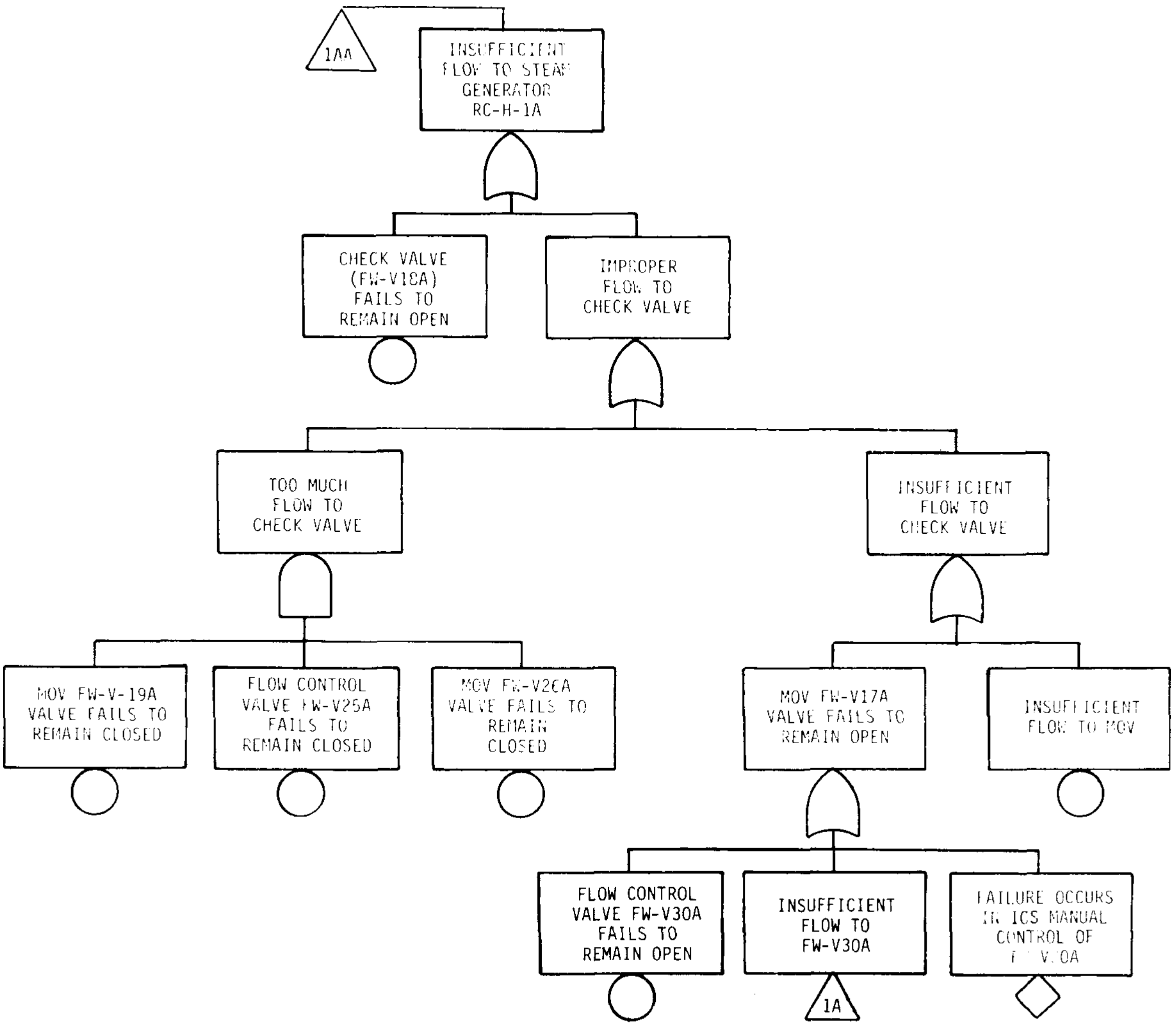

Figure 10-15. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

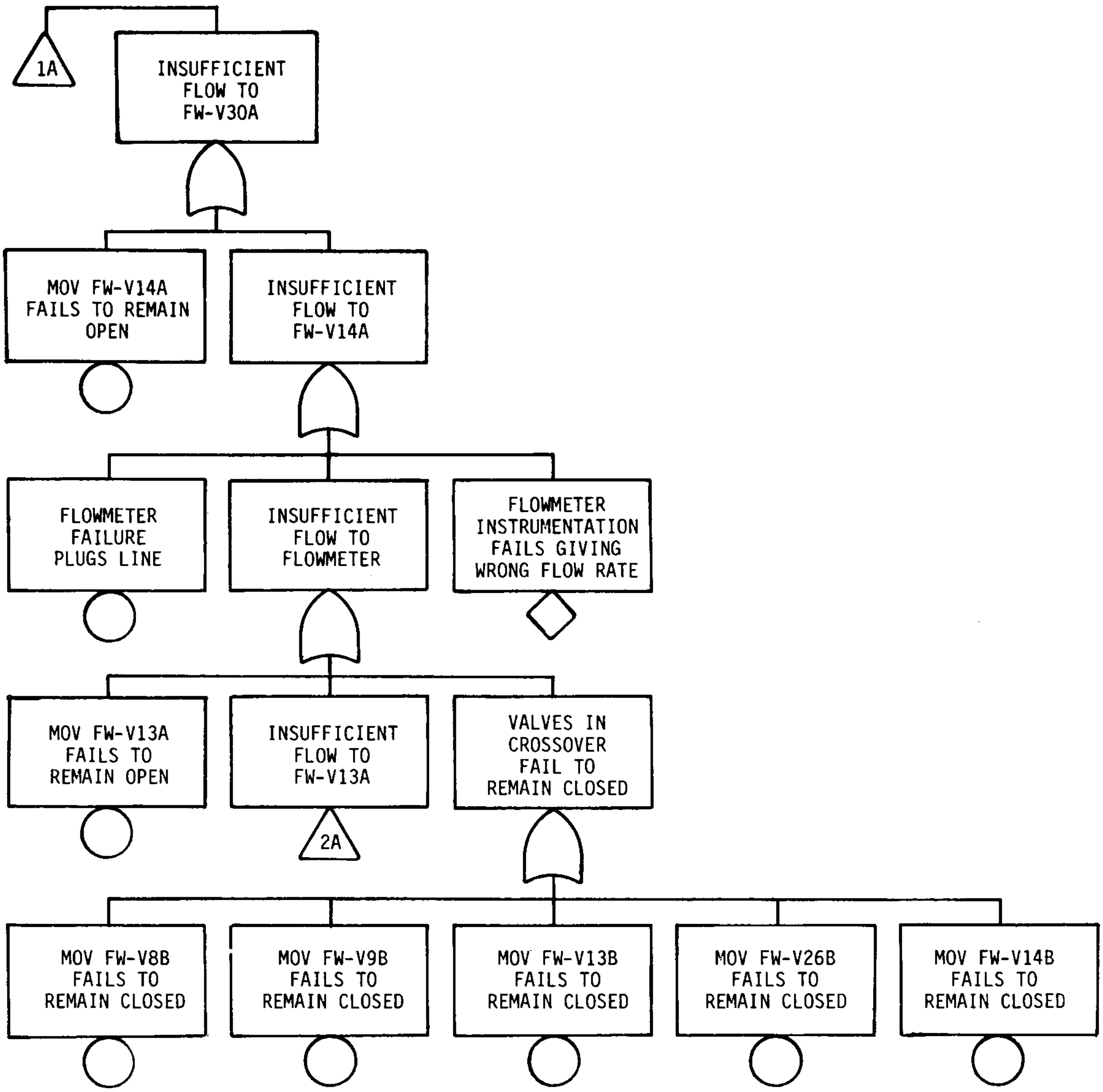

Figure 10-16. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

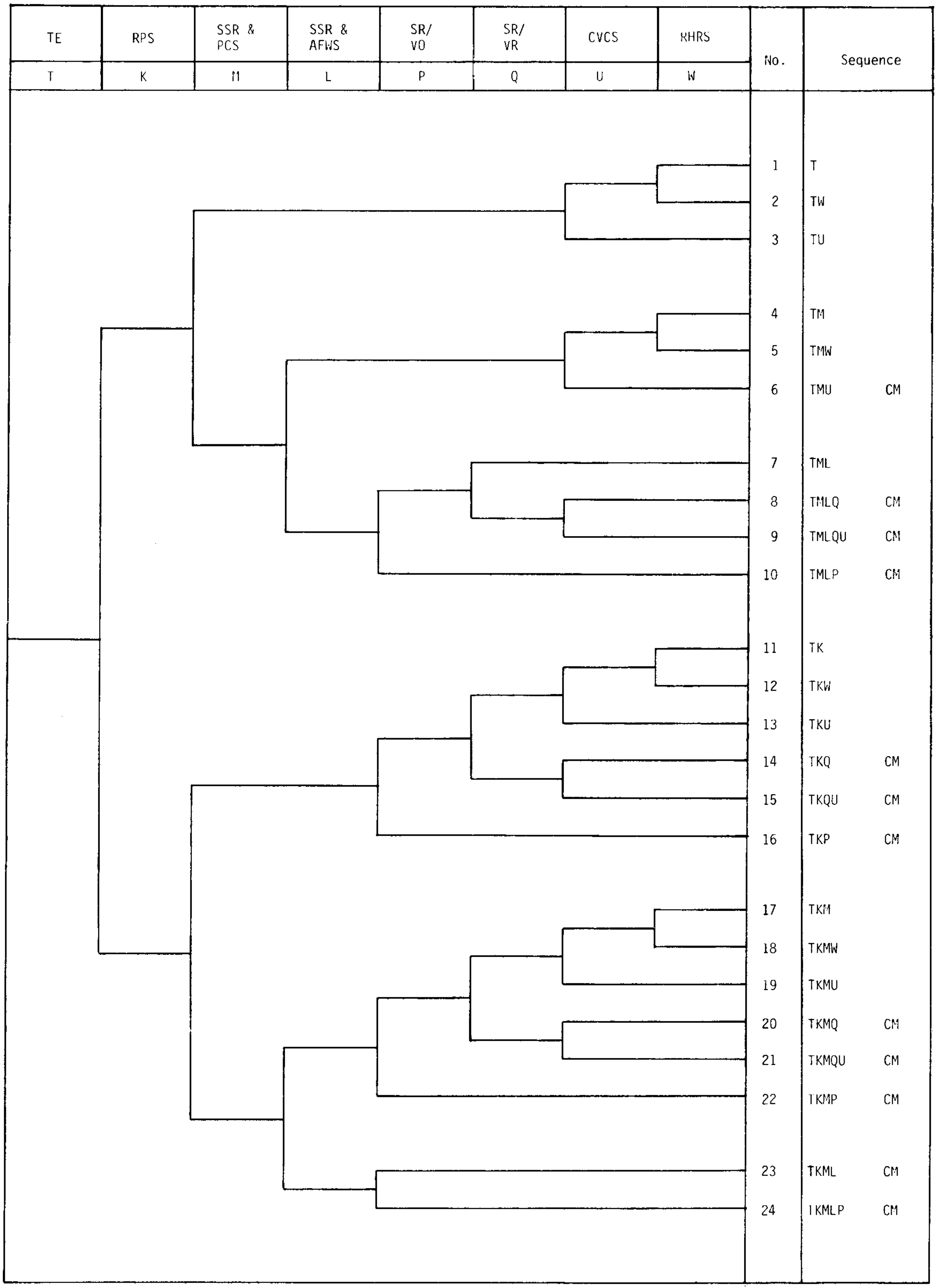

Figure 10-17. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

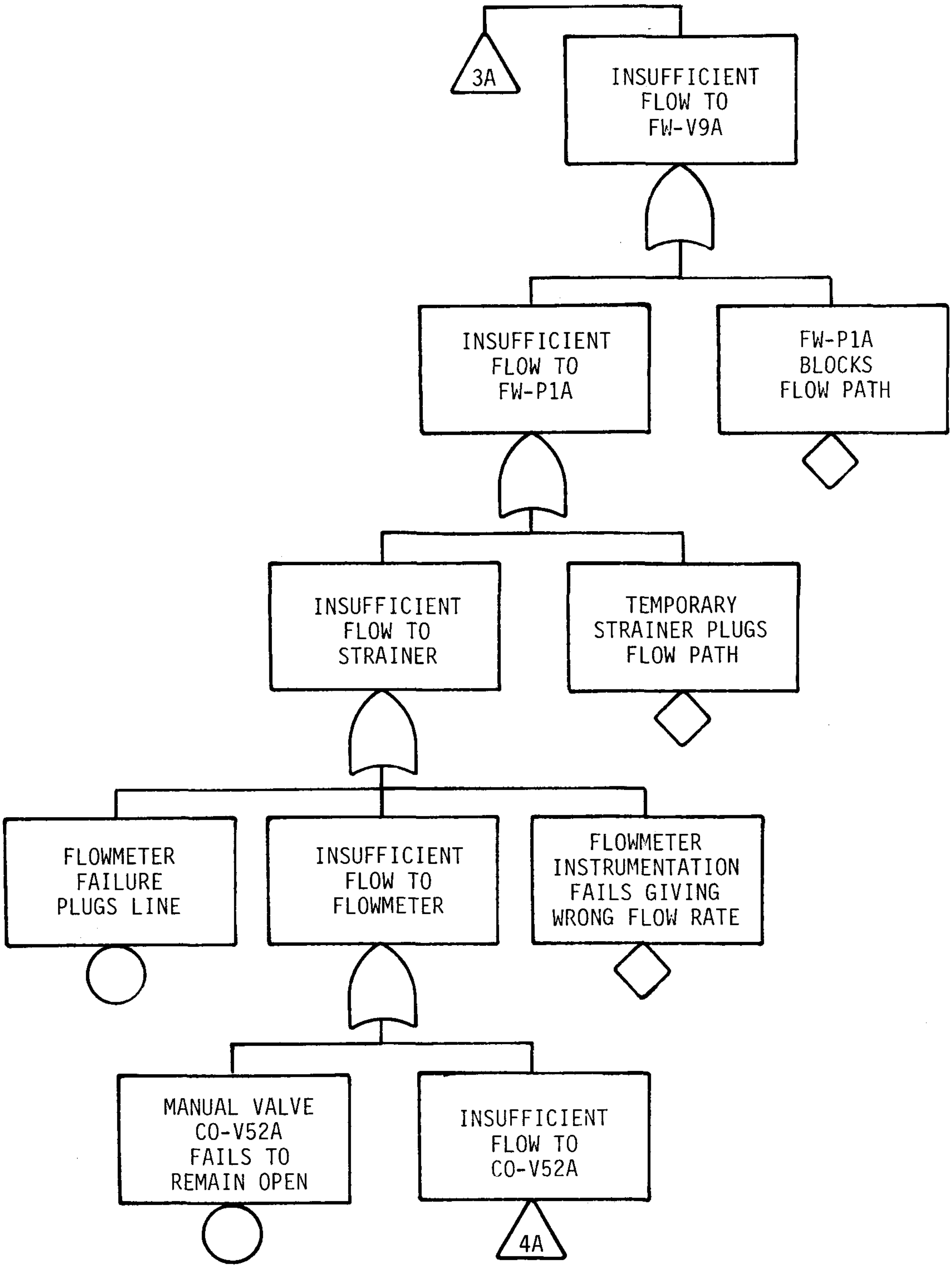

Figure 10-18. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

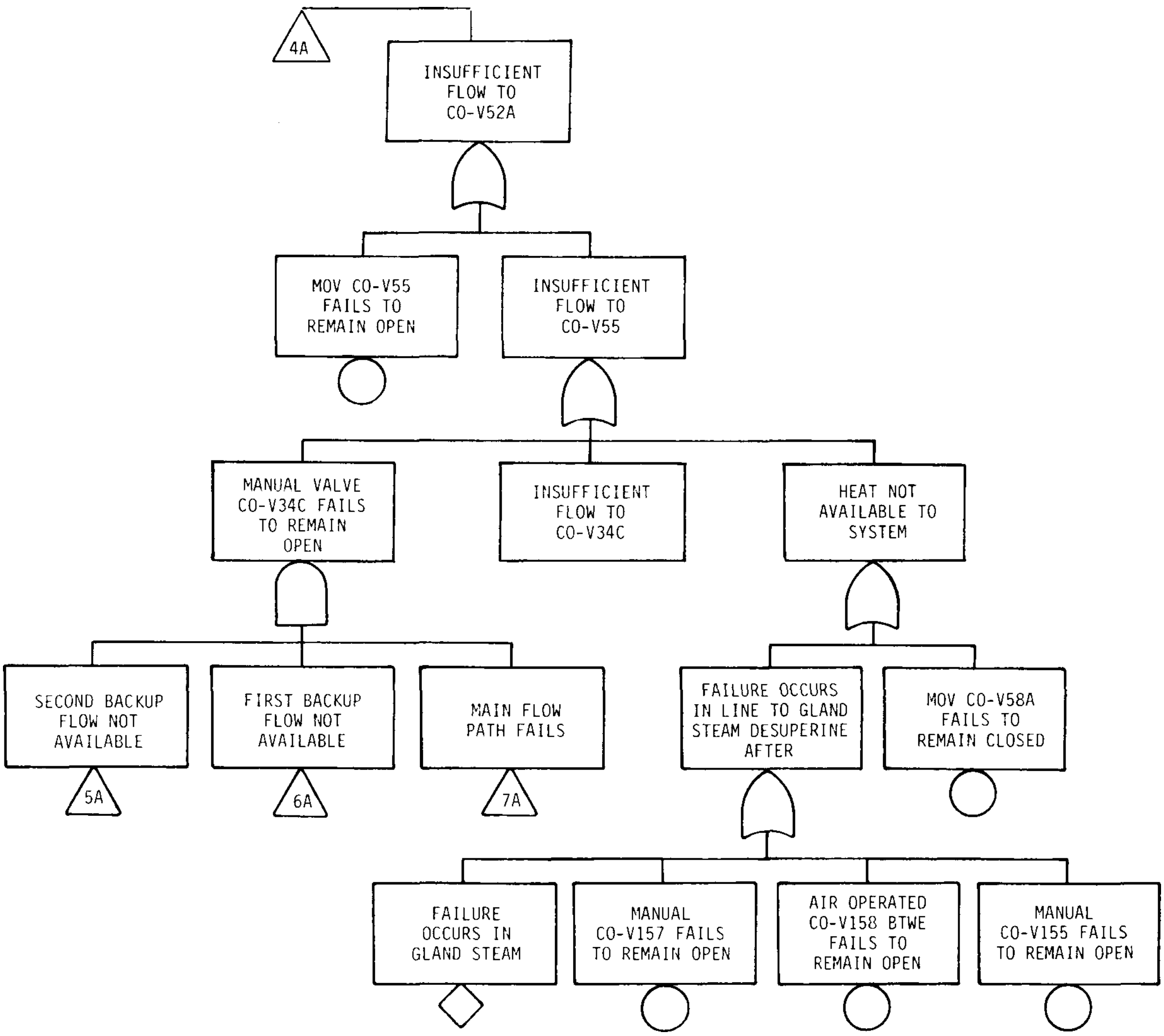

Figure 10-19. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

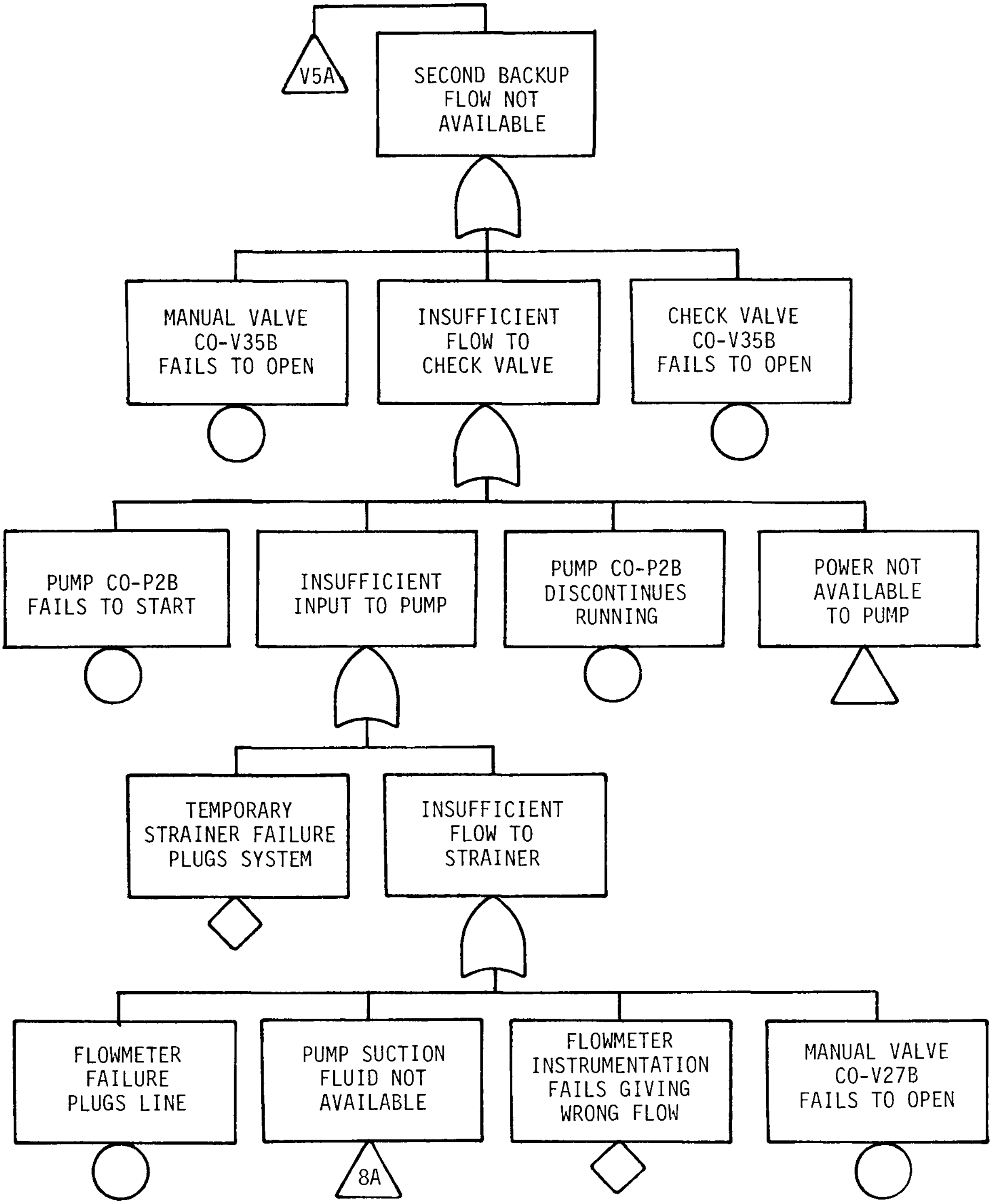

Figure 10-20. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

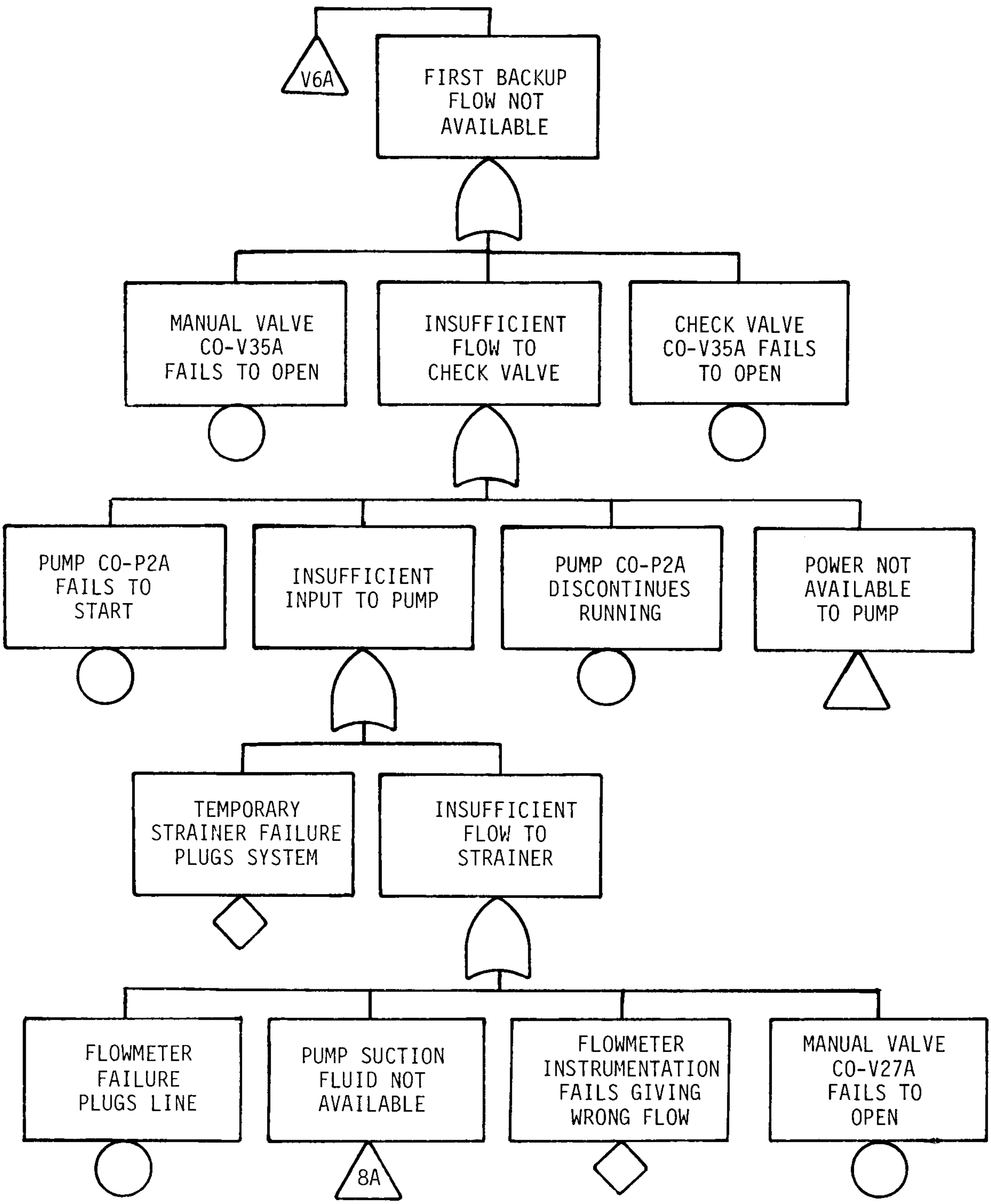

Figure 10-21. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

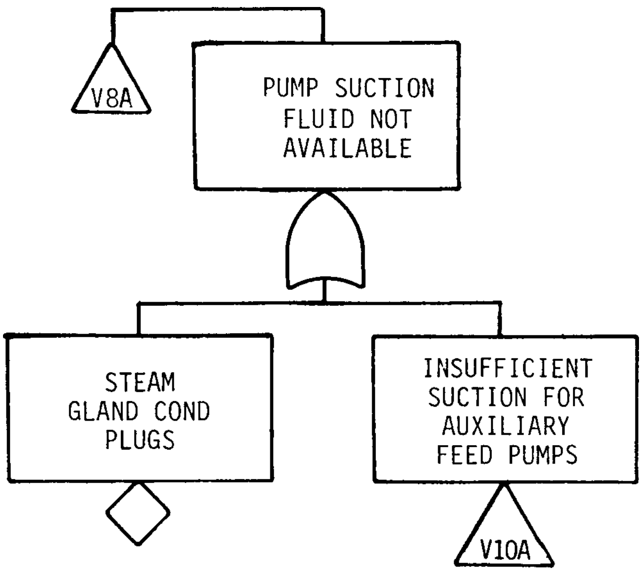

Figure 10-22. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

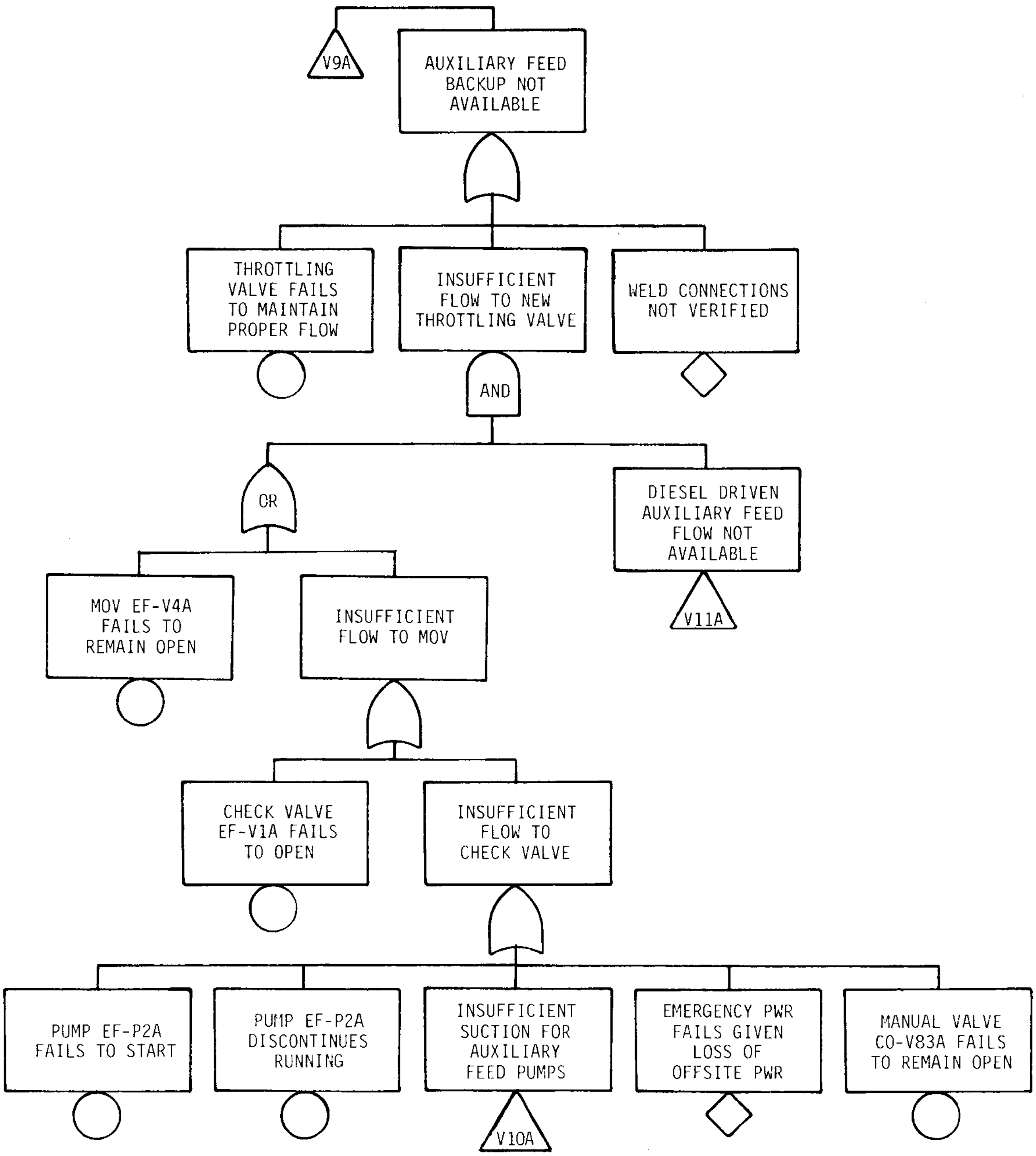

Figure 10-23. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

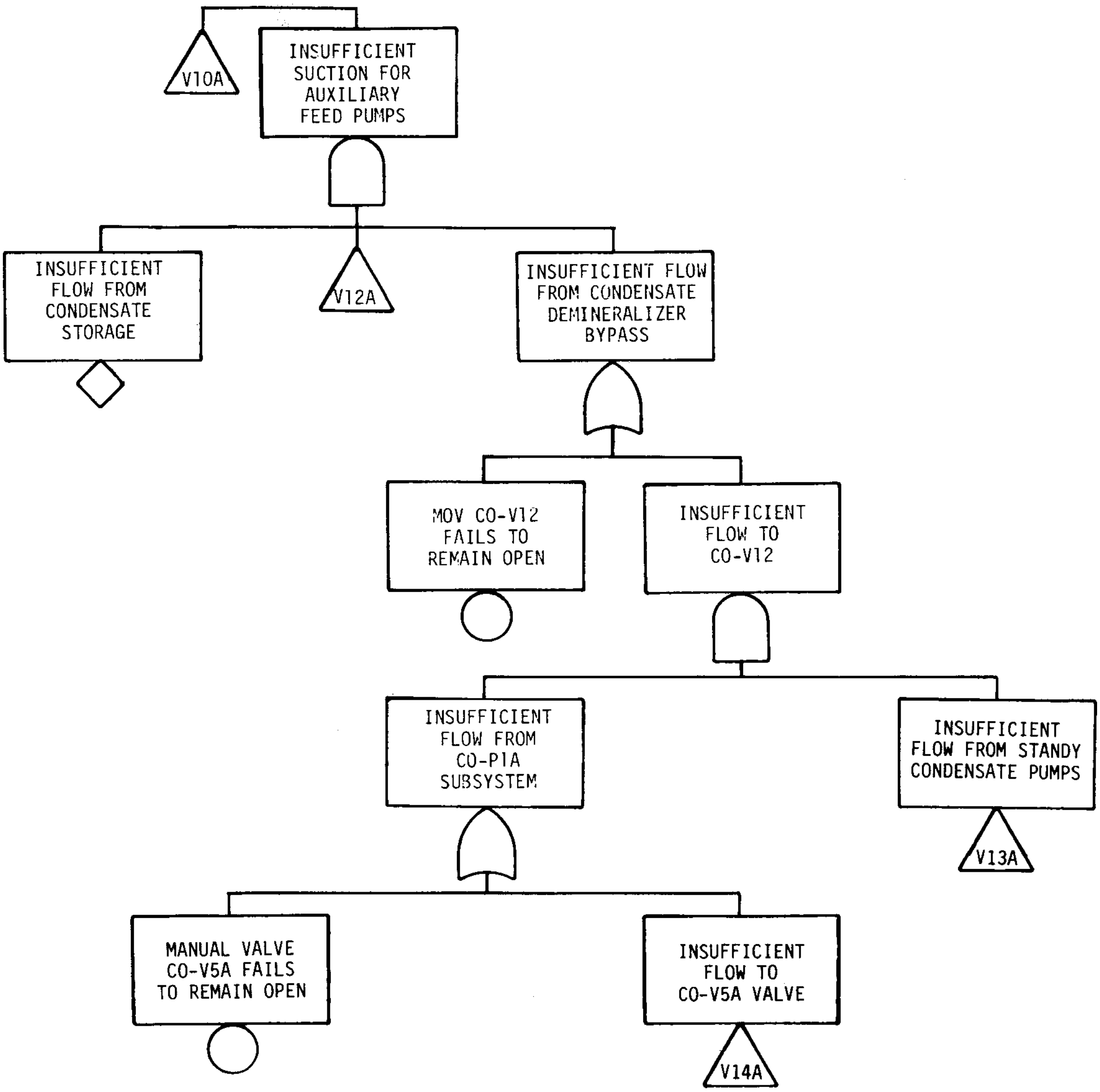

Figure 10-24. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

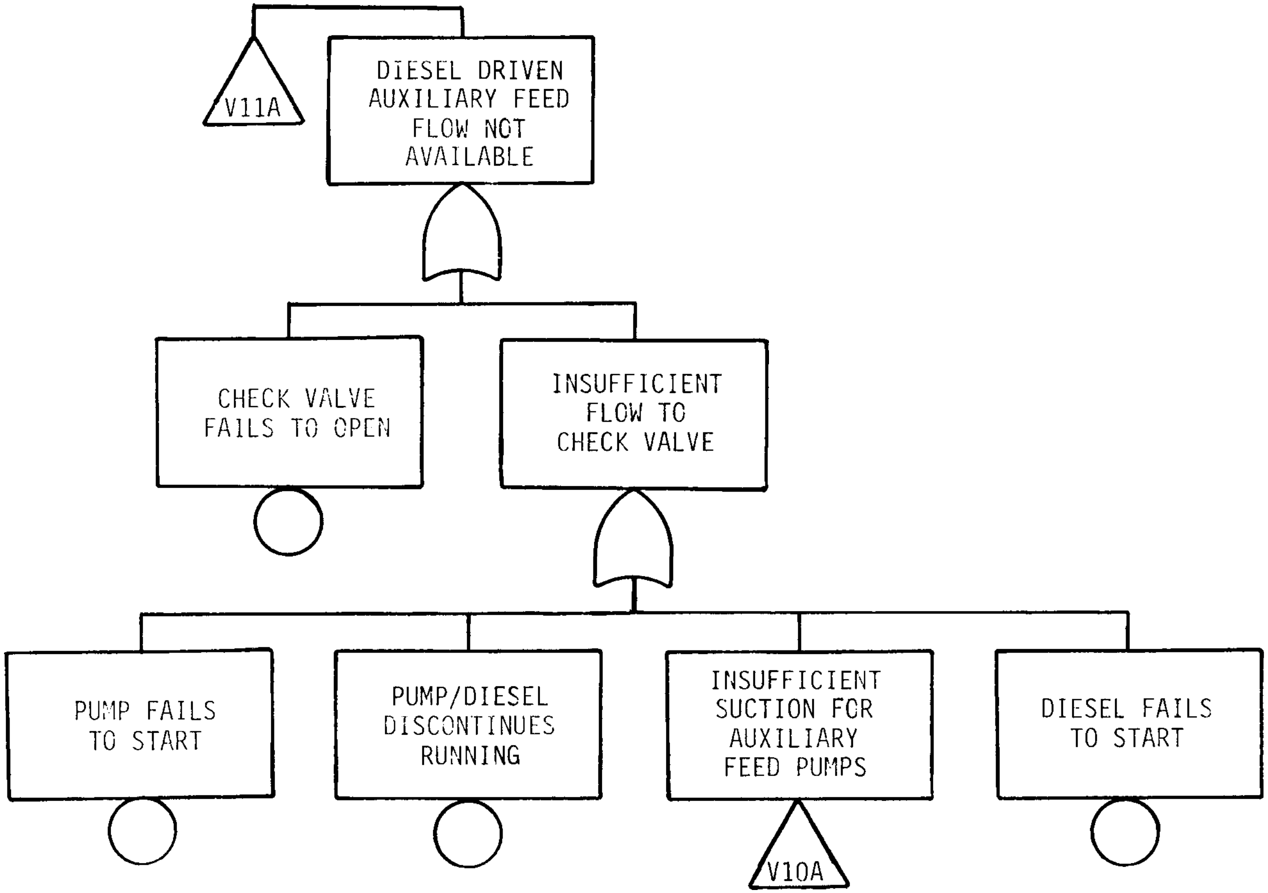

Figure 10-25. Fault Tree Analysis of Proposed Temporary Emergency Feedwater System

that the two new pumps would not offer much of an increase in system reliability. Conversely, the installation and checkout procedures required to add new components and piping crossovers presented the possibility of installation error. The proposed connection on the main feedwater line was a flange that could not be isolated, indicating a potential interruption in existing flow path in order to complete the installation. Thus the possible installation error was deemed much more likely than any additional reliability to be offered by the modification.

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEMS (NSSS)

In the days following the Three Mile Island incident, a request was made for a rapid-response comparison of NSSS designs. There are three U.S. PWR vendors which have produced such designs: Combustion Engineering (CE), Westinghouse, and Babcock & Wilcox (B&W). One CE design and two designs each for B&W and Westinghouse were compared.

A search was made through available Final Safety Analysis Reports (FSARs), from which a set of values for important parameters was collected and tabulated. Four basic data categories were researched:

- Core parameters
- Reactor coolant system (RCS) volumes
- Steam generator characteristics
- RCS overpressure protection characteristics

The results of the comparison are presented in Tables 10-2 and 10-3.

Table 10-2

COMPARISON OF NSSS PARAMETERS

PARAMETER NAME	CE (CESSAR)	WESTINGHOUSE (ZION)	WESTINGHOUSE (DIABLO CANYON)	B & W (BELLEFONTE)	B & W (TMI)
CORE PARAMETERS					
Total core power (MW_t)	3800.	3250.	3338.	3600.	2772.
Design systems flow (10^6 lb/m)	164.	135.	132.9	150.5	137.8
Design core flow (10^6 lb/hr)	157.4				129.5
Core flow heat transfer (ft^2)	60.8	51.5	51.1	56.6	49.2
Inlet Temperature (^oF)	565.	530.2	545.	572.	557.
Outlet Temeprature (^oF)	621.	594.2	609.	630.	607.7
Core coolant velocity (ft/sec)	16.6		15.4	16.2	16.5
Reactor coolant pump power (MW_t)	17.			16.	16.
Operating Pressure (psig)	2250.	2235.	2235.	2195.	2155.
RCS VOLUMES					
Vessel (ft^3)	5741.7	4945.	4945.	4791.	4010.
Steam generator (ft^3/no.)	2158/2	1080/4	1080/4	2094/2	2017/2
RCS pumps (ft^3/no.)	98.5/4	56/4	56/4	183/4	98/4
Hot leg (ft^3/no.)	129.5/2			749.5/2	469/2
Cold leg (ft^3/no.)	224.5/2	4 loops & surge line 1545.	4 loops & surge line 1545.	290/2	237.5/4
Surge line (ft^3)	35.			39.	20.
Pressurizer, water (ft^3)	900.	1080.	1080.	1200.	800.
Pressurizer, steam (ft^3)	900.	720.	720.	1050.	700.
TOTAL VOLUME	12,095	11,890	11,890	13,029	11,144

Table 10-3
COMPARISON OF NSSS PARAMETERS

PARAMETER NAME	CE	WESTINGHOUSE		B&W	
	(CESSAR)	(ZION)	(DIABLO CANYON)	(BELLEFONTE)	(TMI)
STEAM GENERATOR					
SECONDARY SIDE					
Steam pressure (psi)	1070.	720.	805.	1060.	910.
Steam temperature (^{o}F)	552.9	506.3	519.	602.	570.
Steam flow (10^6 lb/hr)	8.59	3.5	3.62	7.75	6.12
Feedwater temperature	450	428.6	432.1	465.	420.
Water volume		1838.	1983.	3345.	3412.
Steam volume		4030.	3775.		
Superheat				50.	35.
PRIMARY SIDE					
Pressure	2250.	2235.	2235.	2195.	2155.
Coolant volume	2158.	1080.	1080.	2094.	2017.
Coolant flow	82.0	33.5	33.5	75.25	63.95
RCS OVERPRESSURE PROTECTION					
Operating pressure	2250.	2235.	2235.	2195.	2155.
High pressure trip	2400.	2385.	2385.	2340.	2355.
Relief valve opens		2335.	2335.	2295.	2255.
Relief valve closes				2245.	2205.
Safety valve opens				2500.	2450.
Relief valve flow (10^3 lb/hr)		210.	210.		112.
Safety valve flow (10^3 lb/hr)	386.3	420.	420.		690.