SENSITIVITY ANALYSIS OF THE REACTOR SAFETY STUDY

by

W. Parkinson, N. Rasmussen, W. Hinkle

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ABSTRACT

The Reactor Safety Study (RSS) or Wash-1400 developed a methodology estimating the public risk from light water nuclear reactors. In order to give further insights into this study, a sensitivity analysis has been performed to determine the significant contributors to risk for both the PWR and BWR. The sensitivity to variation of the point values of the failure probabilities reported in the RSS was determined for the safety systems identified therein, as well as for many of the generic classes from which individual failures contributed to system failures. Increasing as well as decreasing point values were considered. An analysis of the sensitivity to increasing uncertainty in system failure probabilities was also performed. The sensitivity parameters chosen were release category probabilities, core melt probability, and the risk parameters of early fatalities, latent cancers and total property damage. The latter three are adequate for describing all public risks identified in the RSS. The results indicate reductions of public risk by less than a factor of two for factor reductions in system or generic failure probabilities as high as one hundred. There also appears to be more benefit in monitoring the most sensitive systems to verify adherence to RSS failure rates than to backfitting present reactors. The sensitivity analysis results do indicate, however, possible benefits in reducing human error rates.

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I. INTRODUCTION

The Reactor Safety Study (RSS), or WASH-1400, applied the methodology of fault trees and event trees to a complex nuclear safety systems analysis. This methodology had been developed and used successfully in the aerospace and defense industries and was extended to nuclear reactors. The purpose of the RSS was to estimate the public risk of nuclear power plants in the United States. The purpose of this study is to develop a methodology and obtain results which may be used to estimate the sensitivity of public risk to variations in the failure probability of different parts of a nuclear power plant safety system. This methodology and the results in the two reactors considered by the RSS may be useful in providing a basis for establishing priorities for reactor safety research, quality assurance, inspection, and regulation. This could result in more effective use of the public's dollars by enabling decision makers to assure the safety of nuclear power plants without causing economic distortions or hardships through inefficiency. Use of the RSS in the manner described above is consistent with recommendations made in a recent review of the RSS by the Lewis Committee. This methodology has already been used in selected situations.

The French have used system fault tree methodologies to assess design options for the AFWS* and PCS* for a PWR*. Similar uses have been employed in this country.[345] The NRC is also presently sponsoring studies to evaluate different containment designs. Many studies have been done, including

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*See Appendix A (List of Acronyms).

some at Sandia Labs[6], Battelle Columbus Labs[7], and General Atomic[8]. These studies have concerned sensitivity analyses of containment designs which would affect containment failure probabilities. The potential of each design for public risk reduction was evaluated. Further work is now moving in full swing towards a more sophisticated application of the RSS techniques to different containment designs.[9]

In carrying out this work the following questions were addressed:

1. What are the characteristics of the sensitivity of public risk to reductions or increases in input failure rates?

    The values of sensitivities calculated may be characterized by the magnitudes of the ratios of the new public risk to the base of public risk. The ratios of factor increase or reduction in sensitivities describe trends so that differential benefit analysis can be made.

2. What are the characteristics of the sensitivities to increases in the uncertainty of system failure rate probability distributions?

    The sensitivity of the failure rate probability distributions to increasing error spreads are characterized by larger median values and larger error spread for public risk probability distributions. Evaluations of the relation between point values and median values for different probability distributions can be made.

3. What are the major areas of potential public risk reduction?

    Given the sensitivity to reductions in failure probabilities, the most sensitive systems, individual component failure events, and generic classes of events may be identified. Combinations of these sensitivities and the breakdown of their essential elements can provide more detailed information on the potential for public risk reduction. Sensitivity to increases in failure probabilities and uncertainty can provide further information on reducing public risk.

4. What is the relationship between system and generic sensitivities and their individual component sensitivities?

The sensitivities of systems and generic classifications may be further characterized by the principal individual sensitivities which contribute to them. The information may be used to recommend specific actions for systems and to identify important subclasses for generic classifications.

5. What is the synergistic effect of combinations of failure rate changes?

By changing more than one failure probability simultaneously, sensitivities to combinations of components may be calculated and characterized. Those characteristics may be compared to the sensitivities of individual failure probabilities.

6. How do generic classes of failures affect risk as compared to system failures?

A comparison of the characteristics of generic classification sensitivities to system sensitivities may be made for factor reductions and increases as well as ratios of factors.

7. What parameters are best used to estimate the effect on overall public risk?

Sensitivities are calculated for four parameters. These include core melt probability and three of the risk parameters used in the RSS. A comparison of those four parameters may be carried out to determine which is the best estimator of public risk for the reactor in question.

8. What are the limitations of this study?

Limitations resulting from the assumptions made or the limitations of the input are outlined. Also, modifications are identified which are required in order to extend the methodology to other reactors not specifically addressed by this study.

This study will address these questions based on the following outline of the contents. First, a brief description of the methodology used is made in Section II. Elements of Questions 7 and 8 are addressed in that section. Next, Section III presents the analysis procedure and the results of the study. Questions 1 through 6 as well as Questions 7 and 8 are addressed in Section III. Finally, the conclusions and recommendations are presented in Section IV. Appendix A provides an alphabetical list of the acronyms used. Appendix B provides a users manual for the LWRSEN computer code. Appendix C contains the fault trees used for tree systems analysis. Appendix D describes the models of public risk and their use. Finally, Appendix E contains the computer code and the accuracies attained in the uncertainty analysis of failure probability distributions.

II. DESCRIPTION OF THE METHODOLOGY

A. Introduction and Overview

The objective of this study is to develop a calculational methodology to estimate sensitivity of public risk to light water reactor safety systems and components. This objective is accomplished by combining system failure rates and initiator rates to calculate the accident probabilities associated with a particular radioactive release category. Accordingly, a brief explanation of the major elements of the RSS is needed so that the reader may more easily understand the present study. For further details the Report itself should be referenced.

The RSS may be viewed as a breakdown of the calculation of public risk from nuclear reactors in the U.S. Five basic inputs contribute to that risk in this model. First, the probability of an accident-initiating event must be assessed. Then, the probability of system failure for systems which are needed to mitigate the effects of the particular initiating event is required. Given that those system failures which lead to core melt for a particular event tree are assessed, then the probabilities of containment failures are evaluated using models based on the containment conditions of the accident being examined. Assuming the containment fails, the consequence to the public will be determined by the associated radioactivity release. This consequence depends upon two other factors. They are the weather conditions and the population density and distribution. The present study assesses the effect on public risk from variation of the system failure probabilities. The containment failure probabilities are unchanged from the RSS. The weather and population information is included in the model for public risk since it is based on an average of distributions. The variation of initiator probabilities is considered, but only as a very minor aspect of the study.

B. Event Trees and Fault Trees

The combinations of initiator probability and system failure probability are described by accident event trees. Given a certain initiating event, the resulting states of the reactor may be reached by a tree of system functions which affect the outcome. Note that only event trees with potential core melt are considered since other events were assessed by the RSS to have little effect on public risk. The branches of the tree of system function states give the set of possible final states of the accident. The final state of the accident is assessed to determine which containment failure modes are possible. There are many combinations of system failures which lead to core melt, as determined by the event trees. An assessment must be made to determine which of these event trees contribute significantly to the probability. Table II-1 gives just such a list of event trees combined with containment event trees to determine accident types and their probabilities. Table II-2 is the key to the PWR event trees. Tables II-3 and II-4 are the BWR event trees and keys, respectively. These same event trees and resulting accident probabilities are used to determine public risk by the computer code LWRSEN. One limitation of this computer code concerning its application to diverse types of LWR's is whether or not the event trees are the same as the RSS reactors' event trees. A basic review of the RSS event tree reduction process in Appendix I of that report should be completed for reactors significantly different from those in the RSS. Even with similar reactors, a study would probably benefit from this type of review and re-evaluation.

The event trees use system failure probabilities as input. These probabilities are determined by a reliability (or actually, unreliability) analysis of the system. Fault trees are used for system unreliability determination in this analysis. They differ from event trees in that fault trees trace backward from possible failures which lead to the event (a system failure) while event trees trace forward to the state of the accident based on the success or failure of the systems necessary to mitigate accidents and their consequences. In this analysis, only the major parts of the RSS system fault tree analyses are retained, as documented in Appendix C. This allows concentration of effort on the most important contributors to risk. For reactors in which the systems are not exactly the same as those studied by the RSS, different fault trees must be input to the system failure analysis.

C. Risk Model and LWRSEN Computer Codes

The basic inputs to the system fault tree analysis are a number of different types of individual failures. These types of failures are further classified under generic categories. The category types were chosen based on the number of components of that type and the basic reliability classifications of failure modes. The PWR analysis contains more generic types than the BWR, due to the larger number of components and the higher level of detail necessary for the PWR sensitivity analysis. The fault tree reductions mentioned above are also made to fit the chosen generic categories. The generic categories for the PWR are human error, test and maintenance, electric power, control elements, valves, pumps, other hardware, and all hardware. (All hardware includes pumps, valves, and other hardware, but not control elements or electric power.) The generic categories chosen for the BWR are human error and test and maintenance combined, human error, test and maintenance, valves, pumps, and all hardware. (All hardware includes valves, pumps, and other hardware, including control elements on electric power contributions where applicable.) The BWR category which combines human error and test and maintenance was chosen since test and maintenance contributions many times involve human actions.

With the system failure probabilities, accident event trees, and containment failure probabilities, a determination of release category probabilities may be made. To translate these values to public risk, the RSS developed the Calculation of Reactor Accident Consequences (CRAC) code. This code computes the risk to the public by calculating various consequences of an accident given the magnitude of the radioactivity released, the release paths, the weather conditions, and the population distribution. The uncertainties in all of the above inputs cause the generation of probability distributions of various consequences which represent public risk. These probability distributions may be represented less accurately by point values, with consequences for each release category and a probability of occurrence for that category.

The point value models are employed in this study. For more detailed information on the risk models and the consequences used in the sensitivity study, see Appendix D. The two models used for the PWR and BWR are different in nature. The PWR model contains actual values for three risk parameters for each release category. The BWR model gives only their percentage contribution to risk. Consequently, translating the percentages to exact values of risk would involve integrating the BWR probability distributions for each of the three risk parameters and multiplying the result by that base value. A further approximation may be used by doing an approximate integration, given the tables of probabilities for particular consequences listed in Appendix D. Please refer to that appendix for further information and references for the risk models.

The computer code LWRSEN is written to reproduce the above methodology for point values of failure probabilities and consequences. The code contains three major routines for different types of sensitivity calculations. One routine calculates all the individual sensitivities and another calculates all the system and generic sensitivities. The third routine is a combination of both of the first two. Basically, the user first chooses the type of reactor, the consequence parameter, and the multiplicative factors for which the sensitivities are to be calculated. Then, the user decides whether to keep the RSS reduced fault trees or choose a different set of fault trees. The system failure probability equations are input for the method chosen. The code then calcualtes a base case of release category probabilities, consequence parameters, and core melt probability. The user inputs a set of attributes for each component and indicates which attributes are to be varied to determine their sensitivities. The new system failure probabilities are calculated, followed by the new values of public risk to compare to the base case values. A comparison is made to determine a sensitivity parameter and that parameter is output to the user. The code has the capability of varying any subset of contributing failure probabilities, either individually or in combinations.

D. Probabilistic Analysis

In addition to the point value approach, a probabilistic or random variable technique is also employed. The use of random variables to describe failure data results from variability from component to component and plant to plant, as well as from different operating conditions such as component environment. This idea of a population of conditions may be used to describe differing situations within one plant, among a system of plants, or among the entire sets of U.S. or world plants. This random variable approach results in a probability distribution that characterizes the component failure probability. Given these component probability distributions, a reliability analysis, in this case a fault tree analysis, will mathematically combine component distributions to form system failure probability distributions. In the same way, system failure probability distributions may be combined in an event tree analysis to give a probability distribution for the possibility of one type of accident, and the total probability of core melt and public risk parameters.

The RSS determined that all of these distributions could be adequately represented by a lognormal probability distribution. The lognormal distribution implies a normal distribution of the logarithm of failure rates, or data which, in general, vary by factors from lower bound to median, and from median to upper bound values. Information in Appendix III of the RSS documents failure data characteristics and should be referenced for more details concerning different types of data. This information indicates that the lognormal distribution adequately describes the general behavior associated with reliability analyses.

The exact characteristics of the lognormal distribution are given in Appendix E. In addition, Section III, "Presentation of Results", includes relationships between the characteristics of the distribution and a point value analysis. In general, the lognormal provides a conservative analysis for two reasons. First, the median value is always greater than the point value, giving conservative results for system failure probabilities. Second, the error factors, or ratios between the median and upper, and the median and lower bounds, are asymmetrical. In performing the calculations, the upper bound error factor, the larger of the two, is always chosen. Consequently, bounds and medians are conservatively overstated when compared to those values found from the symmetric distributions.

E. Accuracy and Limitations

The results that are to be presented in Section III correspond to the data and analyses presented in the RSS. Most importantly, the results are specific to the representative plants chosen for the RSS. In addition, the risk calculations are specific to a northeast river valley site and contain the approximations of the consequence code (CRAC) developed in the RSS. Nevertheless, the results are important since the general conclusions gleaned from a sensitivity study such as this should apply to almost all reactors of present design. In addition, the reactors chosen for the RSS were typical of many reactors in the U.S.

There are also differences between the RSS and this study. The technique of smoothing is dropped from the analysis of this study in order to provide clearer indications of sensitivity. Also, for all calculations except the variational or uncertainty analysis, point values rather than probabilistic distributions are used. Both of these effects will change the results obtained if they are included. For example, not including smoothing results in about twenty percent less early and latent deaths, and about fifty percent less property damage, than the RSS-reported results. The reduction is an even greater percentage when point values are used. In fact, point values have a tendency to underestimate unavailabilities, and consequently public risk. For this reason, as well as ease in understanding, all results are reported as being normalized by the point value calculations.

The amount of research work done since the RSS analysis was completed may also have an effect on the results. Specifically, in the case of LPIS check valve failure, some problems may occur. If the specific reactor being analyzed does not have this problem it will require an appropriate change in the input data. One may note that removal of smoothing also reduces the effect of this initiator, since it only contributes to release category two in the PWR in this analysis. Other improvements since the Study may be treated in the same fashion, noting that only actual probabilities will change whereas sensitivities will remain approximately the same.

The variational or uncertainty analysis indicates that, for RSS-reported uncertainties, the median value is close to the point value for system failure probability. In general, the ratio of median to point values for both reactor types is less than 1.8 and more then 1.4. Consequently, point values will have some varying relationship with the median values. However, the range of variation is small enough such that point values can adequately estimate sensitivity ratios. The confidence values of the point value vary between confidence limits of twenty-five and forty percent. For uncertainties on the order of three higher than that assessed in the RSS, the ratio of median to point values is closer to three. This is still small enough and consistent enough so that point value sensitivity analysis of reductions or increases by factors of 3, 10, 30, and 100 are useful.

These accuracies may not be sufficient for some applications and more specific Monte Carlo calculations may have to be made. The results of the following section will provide one with the tools to make an analysis consistent with the scope of this effort.

In the total analysis, the economics of the costs and benefits of safety work should, of course, be considered. These aspects of the problem are not treated in this study.

TABLE II-1

PWR DOMINANT ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

	RELEASE CATEGORIES						Core Melt	No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AG-α 9×10^{-11}	AB-γ 1×10^{-10} AB-δ 4×10^{-11} AHF-γ 2×10^{-11}	AD-α 2×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-β 1×10^{-11}	AD-β 4×10^{-9} AH-β 3×10^{-9}	AB-ε 1×10^{-9} AHF-ε 1×10^{-10} ADF-ε 2×10^{-10}	AD-ε 2×10^{-6} AH-ε 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
A Probabilities	3×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	1×10^{-6}	1×10^{-5}	1×10^{-4}
SMALL LOCA S_1	S_1B-α 3×10^{-11} S_1CD-α 7×10^{-11} S_1F-α 1×10^{-10} S_1G-α 3×10^{-10}	S_1B-γ 4×10^{-10} S_1B-δ 1×10^{-10} S_1HF-γ 6×10^{-11}	S_1D-α 3×10^{-8} S_1H-α 3×10^{-8} S_1F-δ 3×10^{-8} S_1G-δ 3×10^{-8}	S_1CD-β 1×10^{-11}	S_1H-β 3×10^{-9} S_1D-β 6×10^{-9}	S_1DF-ε 3×10^{-10} S_1B-ε 2×10^{-9} S_1HF-ε 4×10^{-10}	S_1D-ε 3×10^{-6} S_1H-ε 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-4}
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-8}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}
SMALL LOCA S_2	S_2B-α 1×10^{-10} S_2F-α 1×10^{-9} S_2CD-α 2×10^{-10} S_2G-α 9×10^{-10} S_2C-α 2×10^{-8}	S_2B-γ 1×10^{-9} S_2HF-γ 2×10^{-10} S_2B-δ 4×10^{-10}	S_2D-α 9×10^{-8} S_2H-α 6×10^{-8} S_2F-δ 1×10^{-7} S_2C-δ 2×10^{-6} S_2G-δ 9×10^{-8}	S_2DG-β 1×10^{-12}	S_2D-β 2×10^{-8} S_2H-β 1×10^{-8}	S_2B-ε 8×10^{-9} S_2CD-ε 2×10^{-8} S_2HF-ε 1×10^{-9}	S_2D-ε 9×10^{-6} S_2H-ε 6×10^{-6}		
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10^{-7}	3×10^{-7}	2×10^{-6}	2×10^{-5}		
REACTOR VESSEL RUPTURE - R	RC-α 2×10^{-12}	RC-γ 3×10^{-11} RF-δ 1×10^{-11} RC-δ 1×10^{-12}	R-α 1×10^{-9}				R-ε 1×10^{-7}		
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TMLB'-α 3×10^{-8}	TMLB'-γ 7×10^{-7} TMLB'-δ 2×10^{-6}	TML-α 6×10^{-8} TKQ-α 3×10^{-8} TKMQ-α 1×10^{-8}		TML-β 3×10^{-10} TKQ-β 2×10^{-10}	TMLB'-ε 6×10^{-7}	TML-ε 6×10^{-6} TKQ-ε 3×10^{-6} TKMQ-ε 1×10^{-6}		
T Probabilities	3×10^{-7}	3×10^{-6}	4×10^{-7}	7×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	2×10^{-5}	2×10^{-4}	4×10^{-4}	8×10^{-3}

Note: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability (See Section 4.1).

TABLE II-2

KEY TO PWR ACCIDENT SEQUENCE SYMBOLS

A - Intermediate to large LOCA.

B - Failure of electric power to ESFs.

B' - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.

C - Failure of the containment spray injection system.

D - Failure of the emergency core cooling injection system.

F - Failure of the containment spray recirculation system.

G - Failure of the containment heat removal system.

H - Failure of the emergency core cooling recirculation system.

K - Failure of the reactor protection system.

L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.

M - Failure of the secondary system steam relief valves and the power conversion system.

Q - Failure of the primary system safety relief valves to reclose after opening.

R - Massive rupture of the reactor vessel.

S_1 - A small LOCA with an equivalent diameter of about 2 to 6 inches.

S_2 - A small LOCA with an equivalent diameter of about 1/2 to 2 inches.

T - Transient event.

V - LPIS check valve failure.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment failure resulting from inadequate isolation of containment openings and penetrations.

γ - Containment failure due to hydrogen burning.

δ - Containment failure due to overpressure.

ε - Containment vessel melt-through.

TABLE II-3

BWR DOMINANT ACCIDENT SEQUENCES OF EACH EVENT TREE vs. RELEASE CATEGORY

	Core Melt				No Core Melt
	RELEASE CATEGORIES				
	1	2	3	4	5
LARGE LOCA DOMINANT ACCIDENT SEQUENCES (A)	AE-α $2x10^{-9}$ AJ-α $1x10^{-10}$ AHI-α $1x10^{-10}$ AI-α $1x10^{-10}$	AE-γ' $1x10^{-6}$ AE-δ $1x10^{-6}$ AJ-γ' $2x10^{-9}$ AI-γ' $2x10^{-9}$ AHI-γ' $2x10^{-9}$	AE-γ $1x10^{-7}$ AJ-γ $1x10^{-8}$ AI-γ $1x10^{-8}$ AHI-γ $1x10^{-8}$	AGJ-δ $6x10^{-11}$ AEG-δ $7x10^{-10}$ AGHI-δ $6x10^{-11}$	A $1x10^{-4}$
A Probabilities	$8x10^{-9}$	$6x10^{-6}$	$2x10^{-7}$	$2x10^{-8}$	$1x10^{-4}$
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_1)	S_1E-α $2x10^{-9}$ S_1J-α $3x10^{-10}$ S_1I-α $4x10^{-10}$ S_1HI-α $4x10^{-10}$	S_1E-γ' $4x10^{-6}$ S_1E-δ $1x10^{-6}$ S_1J-γ' $7x10^{-9}$ S_1I-γ' $7x10^{-9}$ S_1HI-γ' $6x10^{-9}$	SE-γ $1x10^{-7}$ S_1J-γ $3x10^{-8}$ S_1I-γ $4x10^{-8}$ S_1HI-γ $2x10^{-8}$ S_1C-γ $3x10^{-9}$	S_1GJ-δ $2x10^{-10}$ S_1GI-δ $2x10^{-10}$ S_1EI-ε $1x10^{-10}$ S_1GHI-δ $2x10^{-10}$	
S_1 Probabilities	$1x10^{-6}$	$9x10^{-6}$	$2x10^{-7}$	$2x10^{-6}$	
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_2)	S_2J-α $1x10^{-9}$ S_2I-α $1x10^{-9}$ S_2HI-α $1x10^{-9}$ S_2E-α $3x10^{-10}$	S_2E-γ' $1x10^{-6}$ S_2E-δ $4x10^{-9}$ S_2J-γ' $2x10^{-6}$ S_2I-γ' $2x10^{-6}$ S_2HI-γ' $2x10^{-6}$	S_2E-γ $4x10^{-6}$ S_2J-γ $8x10^{-8}$ S_2I-γ $9x10^{-8}$ S_2HI-γ $9x10^{-8}$ S_2C-γ $8x10^{-9}$	S_2[illegible]-δ $6x10^{-11}$ S_2GHI-δ $6x10^{-10}$ S_2EG-δ $3x10^{-10}$ S_2GJ-δ $6x10^{-10}$ S_2GI-δ $2x10^{-10}$	
S_2 Probabilities	$2x10^{-6}$	$1x10^{-7}$	$4x10^{-7}$	$4x10^{-6}$	
TRANSIENT DOMINANT ACCIDENT SEQUENCES (T)	TW-α $2x10^{-7}$ TC-α $1x10^{-7}$ TQUV-α $5x10^{-9}$	TW-γ' $3x10^{-6}$ TQUV-γ' $6x10^{-6}$	TW-γ $1x10^{-5}$ TC-γ $1x10^{-5}$ TQUV-γ $4x10^{-7}$		
T Probabilities	$1x10^{-6}$	$6x10^{-6}$	$2x10^{-5}$	$2x10^{-6}$	
PRESSURE VESSEL RUPTURE ACCIDENTS (R)		P.V. RUPT. $1x10^{-6}$ Oxidizing Atmosphere	P.V. RUPT. $1x10^{-7}$ Non-oxidizing Atmosphere		
R Probabilities	$2x10^{-9}$	$2x10^{-6}$	$1x10^{-7}$	$1x10^{-6}$	
SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORIES					
MEDIAN (50% VALUE)	$1x10^{-6}$	$6x10^{-6}$	$2x10^{-5}$	$2x10^{-6}$	$1x10^{-4}$
LOWER BOUND (5% VALUE)	$1x10^{-7}$	$1x10^{-6}$	$5x10^{-6}$	$5x10^{-7}$	$1x10^{-5}$
UPPER BOUND (95% VALUE)	$8x10^{-6}$	$1x10^{-5}$	$8x10^{-5}$	$1x10^{-5}$	$1x10^{-3}$

NOTE: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability (See Section 4.1).

TABLE II-4

KEY TO BWR ACCIDENT SEQUENCE SYMBOLS

A - Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.

B - Failure of electric power to ESFs.

C - Failure of the reactor protection system.

D - Failure of vapor suppression.

E - Failure of emergency core cooling injection.

F - Failure of emergency core cooling functionability.

G - Failure of containment isolation to limit leakage to less than 100 volume per cent per day.

H - Failure of core spray recirculation system.

I - Failure of low pressure recirculation system.

J - Failure of high pressure service water system.

M - Failure of safety/relief valves to open.

P - Failure of safety/relief valves to reclose after opening.

Q - Failure of normal feedwater system to provide core make-up water.

S_1 - Small pipe break with an equivalent diameter of about 2"-6".

S_2 - Small pipe break with an equivalent diameter of about 1/2"-2".

T - Transient event.

U - Failure of HPCI or RCIC to provide core make-up water.

V - Failure of low pressure ECCS to provide core make-up water.

W - Failure to remove residual core heat.

α - Containment failure due to steam explosion in vessel.

β - Containment failure due to steam explosion in containment.

γ - Containment failure due to overpressure - release through reactor building.

γ' - Containment failure due to overpressure - release direct to atmosphere.

δ - Containment isolation failure in drywell.

ϵ - Containment isolation failure in wetwell.

ζ - Containment leakage greater than 2400 volume per cent per day.

η - Reactor building isolation failure.

θ - Standby gas treatment system failure.

<u>TABLE II-5</u>

PROGRAM METHODOLOGY

Identify system failure rate functions and compile with program modules.

Choose reactor type, risk parameter, and four sensitivity factors.

Input component failure rates and containment failure probabilities.

Choose calculational routine.

a. Calculate individual failure rate sensitivites, including indicator probabilities.

b. Calculate system, generic, and combinations of failure rate sensitivities.

c. Calculate system, generic, or a group of components with a common attribute and break down their sensitivity by individual failure events.

d. Calculate the sensitivity of public risk to factor changes in failure rates.

Repeat for other categories of failure rates.

Order sensitivities and output sensitivity data.

III. PRESENTATION OF RESULTS

A. Computer Codes Employed

The methodology contained in the RSS for calculating release category probabilities, core melt probabilities, and approximate consequences using point values, was incorporated in the program LWRSEN. The users manual and the listing for that code are contained in Appendix B. The fault trees used to represent the system unavailabilities may be found in Appendix C. The contributing event trees may be found in Table III-1. The equations used to represent the fault and event trees may be found in the listing of the LWRSEN code. Only dominant failure modes were considered. Failure modes contributing less than one tenth of one percent to risk, even after a single system failure rate reduction by a factor of one hundred, were eliminated.

The uncertainty or variational analysis was done using the PLMODMC code. The code PLMOD and its sister codes (PLMODMC and PLMODT) are available through the NRC for calculating system unavailabilities. These codes also contain a fault tree reduction process. In addition to the uncertainty analysis, the code was used to check on the accuracy of the fault trees which were reduced by hand from those in Appendix II of the RSS. Results from the computer-calculated reductions compare favorably to those reduced by hand, using the same one tenth of one percent accuracy criterion. The PLMODMC code is an extension of the PLMOD code incorporating a Monte Carlo package for calculating complex fault or event trees with probabilistic lognormal distributions as inputs. The users manual for the PLMODMC code was recently documented[10]; however, the listing of the code is not yet available for public use except through the NRC. The exact characteristics of the Monte Carlo analysis are contained in Appendix E. For information on the accuracy of these results Appendix E should be consulted.

Given that the above two codes were written or made available, sensitivities to differing characteristics of the RSS can be studied. LWRSEN calculates point unavailabilities and was the main work horse of the present study. It was used for analysis of sensitivities to changing point values of component, system, and initiator probabilities. PLMODMC calculates median unavailabilities and error factors, and it was used in an auxiliary role to calculate sensitivity to changing error factors.

B. Sensitivity and Risk Parameters Used

Devising a set of parameters to analyze the resulting calculated release category probabilities is important in order to facilitate evaluation. A study performed at SAI used sensitivity indicators.[11] This analysis primarily gives ratios of top event probabilities. Where reductions are being performed, the sensitivity quoted is the base value divided by the new value, which was calculated from a perturbation of some failure probability by the designated factor. This gives a number greater than one, which is the factor by which the top event was reduced. For increases, the inverse is plotted to preserve parameter values at greater than one. It is, therefore, the factor by which the top event was increased from its base value. In addition, ratios of succeeding factors are given to illustrate a measure of the sensitivity from sequential perturbations of failure probabilities. These parameters give a measure of diminishing returns in the case of reductions, and increasing returns in the case of increases in failure probabilities.

When an important contributor is reduced to a level below other contributors, its sensitivity is at, or less than, the other, previously less dominant, contributors. Consequently, the sensitivity of the contributor is reduced. The diminishing return in the tables indicates, nevertheless, that the more sensitive systems sometimes provide more return, after significant previous reductions, than any reduction in a system less sensitive to changes of a factor of three. The increasing values give a measure of the rate of growth. This rate of growth should be compared with the change from one factor to another, which is approximately three. Finally, in the variational analysis, the ratio of error factors, or sometimes the ratio of upper bounds, are illustrated, along with the ratio of median failure probabilities.

The factors with which to perform the sensitivity analysis were chosen after personal conversations with the NRC staff.[12] It was decided to use factors of 3, 10, 30, and 100. Factors above one hundred seem impractical for reductions. The results of the study indicate that this also happens to be the limit of useful reductions. The same factors were chosen for increasing failure probabilities for similar reasons. In the case of increasing error factors, factors of 3 and 10 were used, as was 30 on occasion. Even factors of 30 had a tendency to give such large values for the new error factors that they seem unrealistic.

The values calculated for taking ratios are the following: core melt probability, the number of early deaths per year, the number of latent cancer fatalities per year, and the total property damage per year in millions of dollars. Core melt probability is the sum of the release category probabilities which lead to core melt. For further descriptions of these four parameters, see Appendix D. In addition to the ratios, the release category probabilities and the risk parameter values are given, except for the BWR, where only ratios of risk parameters and release category probabilities are given, due to the lack of a detailed model of the consequences by release category.

C. Results: PWR

The primary thrust of this work is the calculation of point value sensitivities. More accurate results would have to employ a probabilistic analysis such as those done in the uncertainty sensitivity calculations. However, the cost and complexity of a complete analysis was not justified by the extra accuracy attained. A sensitivity analysis of the RSS is of value only in showing directions or relative magnitudes. The probabilistic analysis was only completed for the first three release categories of each reactor type. Since these types of accidents account for almost all of the consequences, these results will represent quite effectively the risk to public safety. For more information regarding the point value and probabilistic studies, see Appendix E.

1. PWR Initiator Reductions

Table III-1 illustrates the basic contribution to each PWR release category by accident initiator, and the resulting dominant accident sequences for that initiator. First, we examine the initiating events which cause the dominant sequences. In the case of the BWR, virtually all of the sensitivity to any parameter is a result of the transient with on-site AC power. The transient without on-site AC power is less than two percent of the core melt sensitivity, and all LOCA and vessel rupture initiators contribute even less. In the case of the six initiators of the PWR, only the reactor vessel rupture contributes an insignificant amount. The large LOCA and the small LOCA with an equivalent diameter of two to six inches (indicated by an S1 in most tables) contributes very little. In fact, their sensitivity is less than many systems and even a few individual components. The primary contributors are the LPIS check valve failure, the transient, and the small LOCA with an equivalent break diameter of about one-half to two inches (indicated by an S2 in most tables). The sensitivities of the five significant PWR initiators are contained in Tables III-1 and III-2. Table III-1 illustrates the sensitivity of core melt probability and total property damage to changes in initiator probability. Table III-2 illustrates the sensitivity of early and latent deaths. The tables indicate that core melt probability is most sensitive to a reduction in the small LOCA (S2) probability.

The LPIS check valve failure is the dominant contributor to public risk. In those reactors where the likelihood of this event has been reduced, it will be much less significant. For example, had the LPIS check valve failure been reduced by yearly testing from its median value of 4.0×10^{-6}/year* to 6.8×10^{-7}/year* in a particular reactor, then the

*RSS Appendix V estimates.

sensitivity to a factor of 3 change, for a reactor with yearly testing, would be close to that given by the ratios of the factor reductions for factors of 30 and 10. (Since yearly testing gives about a factor of 10 reduction and a factor of 3 more would be a reduction of 30.) In this case the LPIS check valve failure would be the third largest contributor, thus making it lower in all categories except early deaths. For a failure probability corresponding to monthly testing, which is approximately one hundredth of the original RSS estimate, the event would become almost insignificant. It should be noted, however, that other specific sensitivities would no longer be entirely accurate. Only their relative magnitudes would stay the same. More information relative to this will be presented later under combinations of system reductions.

2. PWR Systems Reductions

a. System Failures

System failures are the next level in the event tree hierarchy. The PWR systems' contribution to public risk will be dependent on their contribution to the transient and small LOCA (S2) initiators, since the LPIS check valve event involves no other system failures. The systems which are indicated in the key to the PWR tables of sensitivities all contribute something to public risk, with the magnitude depending on their contributions to the important initiators.

The first results presented are the new release category probabilities which lead to a core melt for reductions in system failure probabilities. This information may be found in Table III-4. The ECCS (H and D) is a major contributor to the sensitivities in release categories 5, 6, and 7; however, they are minor contributors in the lower categories. The transient systems involving the auxiliary feedwater system (AFWS) and the power conversion system (PCS) contribute to all categories except 4 and 5. The containment spray injection system (CSIS) contributes heavily to categories 1 and 3.

The sensitivity results for the public risk parameters (early deaths per year, latent cancers per year, and total property damage in millions of dollars per year) are presented in Table III-5. The transient systems mentioned earlier contribute the most heavily to each risk parameter. The AFWS and PCS provide the most potential for reducing public risk in the PWR. The CSIS is the only other system which contributes to early deaths.

The sensitivity parameters, core melt ratios, and the three aforementioned risk parameter ratios are given in Tables III-6 through III-9. The ECCS contributes strongly to core melt, with forty and sixteen percent reductions for a factor of three reduction in failure probability for recirculation and injection modes, respectively. The forty percent reduction was the largest for any parameter or system for a factor of three. However, these systems contribute very little to risk, except total property damage. In short, core melt is not an accurate indicator of the consequences for the PWR. The transient systems AFWS and PCS offer thirty percent reductions for a factor of three in early deaths, with similar reductions in latent deaths and total property damage. In fact, all other systems offer less return than the transient systems, even when their unavailabilities are reduced by a factor of one hundred. The reactor protection contributes less than one percent to public risk reduction, even for factors as high as one hundred. The containment safety systems offer little potential for risk reduction, except for the CSIS and its sensitivity to latent cancer fatalities. The diminishing returns listed in the tables indicate little return for changes larger than factors of 3 for all but the most sensitive systems.

b. Systems Breakdown

The systems or functions listed in the tables may be further broken down into other reactor safety systems, subsystems, and individual failures. Tables III-10 and III-11 contain such a breakdown based on the most significant contributors to risk within a safety system or function. Table III-10 contains a breakdown of emergency core cooling and injection sensitivity for core melt probability as a risk parameter. Table III-11 illustrates the three most important contributors to latent deaths, namely, the AFWS, PCS, and CSIS. In the case of emergency core cooling, the high pressure recirculation and injection systems are much more sensitive to core melt than the low pressure systems. The high pressure cooling systems are the primary contributors to the small LOCA (S1 and S2) element of core melt probability. When the reductions of release category probabilities are translated to risk, however, the CSIS system has major potential for risk reduction. This system contributes heavily to release categories 1 and 3 and, therefore, latent deaths through the small LOCA (S2) event tree coupled with steam explosion and overpressure containment failures. The largest contributors within the CSIS are human errors from CLCS miscalibration and valves being left open. Hardware and test and maintenance contribute much less in the CSIS. The largest potential for risk reduction comes from the event trees involving the AFWS and PCS. In the particular case of the AFWS, one particular human error of three valves being left closed contributes more to risk than even the CSIS. Clearly, special attention should be placed on the procedures and environment of this valve's human action. The PCS subtree was not developed fully, but the main feedwater system has the most sensitivity. A more detailed analysis of the failure of this system should be done, with hopes of identifying the potential sensitivity.

c. Individual Failures

To take further advantage of the more detailed analysis of the PWR, a review of the most sensitive individual failures may be found for all four sensitivity parameters in Tables III-12 through III-15. The component numbers listed in the table refer to designations from the LWRSEN computer code. The individual failures' relation to the analysis may be found by reference to the fault trees in Appendix C, where component numbers are given. The equations used to calculate sensitivity may be found in Appendix B in the listing of computer codes. The components' algebraic relation to the rest of the analysis may be found from the component number also. It can be seen that very few components from systems other than H, D, L, M, and C have potential for reduction in total risk. Component 182, or the event B', is defined as the failure to recover electric power (off-site or on-site) within one to three hours after a transient with loss of off-site AC electric power. Increasing availability of electric power within this time window would provide much more reduction in public risk than the time window of one hour necessary for the ESF's to mitigate LOCA events. In addition, the independent failure of three or more control rods can be seen to have little impact on public risk from further reductions. It may be noted that while this analysis used statistical coupling techniques to determine a base value, the term was varied as a whole, so the resultant sensitivity is to a three rod failure rather than a one rod failure, and the resultant statistical translation is to three rods failing.

d. System Combinations

The results presented so far allow only for reductions of one system or component failure probability at a time. Since more than one dominant failure mode has been identified, a rational safety reduction policy would consider multiple reductions of failure probabilities. Table III-16 is the result of a simple analysis of multiple system reductions of core melt probability. Also illustrated for each case are the multiples of each individual sensitivity. Simultaneous reduction in failure probability of ECI, ECR, CSIS and AFWS have larger magnitudes than the multiples of individual reductions (case 1). It can also be seen that the returns for further reductions of the combination are larger than the multiples of individual reductions. This is caused by avoidance of "creating" other dominant failure modes as soon as one reduction in a major contributor is made. The cases illustrate that, for every case other than 11, this additional reduction is attained. In case 11, the combined systems both contribute primarily to the same event trees; consequently, a reduction of ten in both systems is equivalent to a reduction of one hundred in either.

In summary, there is not much potential for reduction of public risk from reduction of system failure probability in terms of magnitude. Magnitudes are generally much less than two. Also, diminishing returns indicate that little sensitivity to further reduction occurs after reductions of ten or more. The benefit for higher factor reductions can be increased with combinations of reductions. The magnitudes can also be increased up to about six if the four most important systems are reduced by a factor of one hundred each.

3. PWR Systems Increases

The magnitudes of the sensitivities change considerably when one considers increases to public risk from increases to system failure probability. The effects of increasing failure probability over those reported in the RSS for the parameters of release category probabilities and public risk are contained in Tables III-17 and III-18, respectively. The sensitivity by category and parameter for increases are similar to those for reductions. However, the magnitude of the sensitivity is greater. These magnitudes are illustrated for core melt probability and the public risk parameters in Tables III-19 through III-22. Since the relative potential among systems is the same for reductions as it is for increases, the valuable information in these tables may be found in the characteristics of the increasing return. The ratio between the consecutive factors 3, 10, 30, and 100 is approximately three. When the increasing return is near three, any increasing system unavailability is translated directly to public risk. This is the case for the functions M and L (systems PCS and AFWS). Latent deaths are very sensitive to the

CSIS, and total property damage is sensitive to a larger number of systems. The reactor protection system and other containment systems show very little sensitivity to increasing failures. The most sensitive increases by factors of 3 and 10 cause public risk increases of less than one half of the factor magnitudes, and most other cases show much less increase in public risk. Increasing public risk for sensitive systems nears almost the entire additional system increase at total factors such as 30 and 100, indicating they become dominant.

4. PWR Generic Failures

a. Generic Reductions

A systems analysis provides information on the specifics of nuclear reactor safety features. However, many times engineering advances are made in types of components, rather than one specific valve, pump, or subsystem. For this reason, a generic analysis was performed on the individual failure contributors which make up various systems. The generic analysis also indicates a credit obtainable in the reactor's safety as a whole if generic improvements are obtained.

The sensitivity of release category probabilities to the generic types mentioned earlier is shown in Table III-23. Human error shows potential reduction in all categories, the most significant ratio reduction being in categories 1 and 3. Electric power shows reductions in categories 1, 2 and 6. In category 2, only the dominating contribution of the LPIS check valve rupture remains after electric power is reduced by a factor of 100. Other failure types show little reduction, except for control, which contributes heavily to category 7, the largest contributor to core melt probability.

Table III-24 indicates reductions in risk parameters for reductions in generic component failure probability. Human error and electric power reductions cause reductions in all risk parameters, with electric power changes, based on its contribution to release category 1 from the transient event trees, causing greater sensitivity of early deaths. Total property damage shows the most sensitivity to other generic classifications. The actual sensitivities for core melt probability and public risk found in Tables III-25 to III-28 indicate that, while core melt is very sensitive to control, it is insensitive to all but total property damage because of the large reductions in release category 7, where only property damage is a significant consequence. Test and maintenance is less sensitive than control for the first factor of 3 reduction; however, it is more sensitive during subsequent reductions. This is because test and maintenance was more sensitive at all factors in category 1, which has higher property damage, while control only contributed for initial factors in category 7, which has low property damage consequences.

The magnitude of the initial reduction in generic failures is larger than that of the most sensitive single system. At the same time, the return for higher reductions is less than for generic components. This results from the fact that a failure in a system is usually the result of one of many possible failures of components from all generic classes. In addition, the failure probabilities of most components were of the same order of magnitude, so there were many contributors to system failure and, consequently, to public risk. Any reduction in one class could be initially reflected in overall sensitivity, however further reductions only left exposed other generic failures of similar magnitudes. In the case of systems, there was a larger variety of failure probability and therefore larger potential for further reduction. The larger magnitude reductions can also be explained. Generic failures contribute some to all events; while, in the case of public risk reductions for systems, there are two major contributing events, the transient and failure of AFWS and PCS, and the small LOCA (S2) and failure of the CSIS. Therefore, reduction of one fault tree leaves dominance of another. The larger number of contributors for generics leads to less further reduction, eventually negating the advantage of contributing something to all event trees. This is in contrast to the systems analysis, where contributions were limited by not affecting all trees, but the wider range of contributions leads to higher reductions at higher factors.

b. Generic Breakdown

A further breakdown of the more sensitive generic failures is contained in Tables III-29 and III-30. Human error contributions to the sensitivity occur primarily from valve operation errors. There are a number of errors with significant contribution to public risk, as measured by total property damage. The most significant hardware contribution results from the unavailability of diesels for electric power. The contribution of test and maintenance to early deaths is primarily that of turbine and safety valve maintenance resulting in unavailability of the AFWS.

c. Generic Increases

Generic failures also exhibit a markedly different sensivity to increases of failure rates. The initial increases are still slightly larger than those for systems; however, subsequent increases grow larger faster. The increases eventually exhibit an avalanching effect. This is due to the fact that a generic failure can contribute in many different parts of the system and also to the fact that there are second- or higher-order cut sets within the system failures that can cause exponential increases. These traits may be found in Tables III-31 through III-34.

5. PWR Uncertainty

The final analysis performed on the PWR was a variational or uncertainty analysis. The uncertainty analysis is effective in measuring a system's sensitivity to error propagation. Only increasing error factors were considered. The error factors, or ratios of the 95% confidence limits upper bound to the median value, were increased by factors of 3, and sometimes 10 and 30 as well. Given the fact that most systems have unreliabilities on the order of 10^{-4} or above, increases on the order of 30 may be unreasonably large; however, they can be used to indicate trends.

The first uncertainty analysis was performed on a system and its components. While the consideration of individual systems and their component levels was, in general, too complicated to be included in the time frame of this study, this analysis was performed as a check on the fault tree reduction process. The LPRS system contributes little to the uncertainty of release categories 2 and 3. Nevertheless, increasing the system error factors by almost seven had no effect on the release category distribution. It is doubtful that many individual components would have much effect on risk uncertainty; however, consideration from a generic viewpoint could have a profound effect on system uncertainty, considering the avalanching effect noted in increasing generic components in the point value analysis.

a. Initiators Uncertainty

The primary focus of the uncertainty study was to explore the dependences of major elements of the event tree analysis. Those elements include sensitive systems and the initiators. For the PWR, the initiators are of interest due to the more diverse nature of the PWR risk contributions. Table III-36 illustrates the sensitivities of all types of initiators as well as the combined effect of an increase in all initiators' uncertainty. Vessel rupture uncertainty increases have no effect on release category uncertainties, even when increased in error factors of 100. Release category 1 shows much less effect from uncertainty than categories 2 and 3. The dominance of the transient in category 1, and the low initial error factor associated with the transient, account for this insensitivity. Even when the transient error factor is increased by a factor of 3, it is still less than that of the small LOCA (S2), which is another major contributor to risk from category 1. Category 3 is very sensitive to LOCA uncertainties, exhibiting almost the full factor increase of 3 in the new upper bound. Category 2 is sensitive to the LPIS check valve rupture uncertainty. Tables III-37 through III-39 indicate new values of medians and error factors for increasing all error factors by factors of 3, 10, and 30. Comparing this to the initiator uncertainty results, one can see that most of the uncertainty from release category 3 is a result of the LOCA uncertainty. Release category 1 shows more sensitivity to increasing error factors, and it is presumed most of that comes from the systems uncertainty, since transient initiator uncertainty was significantly smaller. This is because the initiator transient contains higher order event tree "cut sets" and some multiplication of uncertainty results. This multiplication proves to be of little effect, however, since top event error factors increase little over the factor increase for all systems and initiators. Consequently, one can safely assume that increasing all uncertainties given in the RSS by a certain factor will increase public risk uncertainty approximately by the same amount. Increasing all error factors also affects the medians of release category probabilities. The medians show a more noticeable increase than the case of increasing all initiator factors. Therefore, it would appear that combinations of increasing uncertainty affect the ratio of the medians more than they affect the error factors.

b. Systems Uncertainty

Finally, an uncertainty analysis was performed on individual systems which contribute to release categories 1, 2, and 3. These results are presented in Table III-40. Systems L and M have a profound effect on release category 1, and system C has a slightly larger effect on release category 3. Increasing error factors in a system increases the top event uncertainty by about half the factor. Systems M and L also contribute more than a few percent to any top event uncertainty. An analysis for individual systems was also done for public risk of latent deaths.

This could be effectively approximated by the contributions from release categories 1, 2, and 3. Since L and M contributed at least some in all categories, they had the most effect on latent cancers. Increasing an individual system's uncertainty by 10 results in a maximum increase of about 2.5 in public risk uncertainty and 1.5 in median public risk. Only the AFWS, PCS, and CSIS appear to have noticeable sensitivity.

In summary, increasing all uncertainties will result in similar increase in top event uncertainty. Increasing an individual component sensitivity would have little effect in all but the most sensitive systems. Increasing system uncertainties also has little effect on public risk uncertainty, except for the AFWS, PCS, and CSIS, and those effects are muted compared to the system factor increases. Median values increase very little for uncertainty increases except when all factors are increased. Then a multiplicative effect is noticeable.

In general, PWR sensitivity is primarily concentrated in the AFWS, PCS, and CSFS for all types of analysis. In addition, generic classes of human error, electric power, and control are the most sensitive. Most of the sensitivity to public risk comes from deviations on the high side of RSS values, while reducing public risk has little relative potential unless a careful program considering combinations of effects is employed. The wide variety of failure modes discovered in the PWR has the effect of limiting sensitivity potential. This implies reasonable limitation of expectations for future safety reductions from the standpoint of engineered safety features.

D. Results: BWR

1. BWR System Reductions

The sensitivity analysis of the BWR considered only the transient event trees. The values of the probability ratios for release categories 1, 2, and 3 for failure probability reductions in the safety systems designed for transient events are found in Table III-42. Release category 4 was not included in the analysis, as it contributed very little to risk and would have involved a much larger volume of work for a small part of the sensitivity. Also, values for the risk parameters are not available since a model relating exact consequences to release category probabilities could not be obtained for a BWR. A model using percentages of the total risk was available, however, and this was used instead. Sensitivity can be obtained, however, since it is a dimensionless ratio of two risks; a base risk and a newly calculated risk. The actual values could then be obtained by dividing the sensitivities into the consequences of the base case, which could be obtained from another source. It should be noted that the sensitivity of W is overstated in the second release category since the LOCA event trees are not considered in this study. In this category the reactor protection system failure is not a contributor. When W is reduced by factors higher than 30, contributions from the LOCA event trees are about one-half the total contribution to category 2, leading to some loss of accuracy. In categories 1 and 3, there are two event trees of significantly larger magnitude than the LOCA trees, so the individual sensitivities are accurate.

Tables III-43 and III-44 document the system sensitivities for the BWR, with the core melt probability, total property damage, early deaths, and latent deaths as risk parameters. While core melt probability is actually the sum of release categories 1 through 4, the probability of category 4 is less than one-thousandth of the probability of category 3. Consequently, the sum of categories 1, 2, and 3 adequately represent core melt probability. Due to the significantly smaller number of dominant contributions to public risk, the magnitudes of the BWR sensitivities are larger than the PWR sensitivities. Systems A and C provide significant potential for risk reduction, even up to factors as high as one hundred. The systems Q, U, and V all provide much less sensitivity than the others, but more than most of the systems encountered in the PWR. The dominant LOCA-related systems not included in the study would have magnitudes of less than one-half of the sensitivity of the three systems. It can also be seen that the sum of categories 1, 2, and 3, and therefore core melt probability, are very good indicators of public risk. Core melt seems to slightly overestimate the risk associated with systems W, Q, U, and V, and slightly underestimate the risks of system C. The difference between the systems is caused by the presence of systems W, Q, and U in release category 2. Release category 2 contributes its heaviest percentage to core melt probability, or the sum of the three release categories. Then it contributes to the percentage of risk; first in early deaths, then in total property damage, and finally in latent cancers. This is illustrated in the tables by the relative sensitivities of each parameter with respect to core melt. For W, Q, U, and V, the ranking of the risk parameters in release category 2 is preserved. Since C, the reactor protection system, does not contribute to category 2 at all, the inverse ranking of parameters is found. Therefore, in the BWR, the sum of release categories 1, 2, and 3 are excellent measures of risk. Since this sum is virtually equivalent to core melt probability, any reactor similar to the BWR studied in the RSS can adequately describe public risk by the use of core melt probability. In the case of the PWR the sums of release categories 1, 2, and 3 are also reasonable measures of public risk. However, they bear no resemblance to core melt probability.

The system or safety functions indicated earlier were also broken down by subsystems. The sensitivity of core melt probability to subsystem failure probability reductions is given in Table III-45. In the two most sensitive safety functions, two subsystems are found to have the same sensitivity as the function. This is because, in each case, both of the subsystems are required to satisfy function success. Consequently, any reduction in one subsystem's failure rate has the same effect on the requirement for success. This is also true of systems M and L in a PWR transient event tree. Likewise, systems Q, U, and V in a BWR transient event tree have the same overall sensitivity. In a case where one of two or more subsystem failures is required, the sensitivity is divided. This division is approximately such that the multiples of the sensitivities equal the top event sensitivity. The multiples are roughly related to their relative order of magnitude. Therefore, the same general rules of combinations as were found in the PWR apply for the BWR, as would be expected.

2. BWR System Increases

The BWR was also analyzed for system failure probability increases. The results are presented in Tables III-46 and III-47. The characteristics of the results are very similar to those of the PWR systems analysis. However, in the BWR, one system, W, almost completely dominates any increase. For any factor increase of W beyond 3, the full value of the increase is felt by public risk. The system C is almost as sensitive, while Q, U, and V are insensitive, causing only factors of 4 increase in public risk for factors of 100 increase in system failure probability.

A breakdown of the risk increases to system failure rate increases may be found in Table III-48. Similar characteristics of the tables correspond to the breakdown of risk reductions; however, multiplying individual sensitivities to find combinations will not work. In fact, they seem to more closely approach addition of sensitivities. In particular, it can be seen that increasing the failure probability of three or more control rods failing independently is not very sensitive to factors of up to 30; however, increases of 100 in this probability could cause one order of magnitude increase in risk.

3. BWR Generic Reductions

As in the PWR, an analysis of generic failures in the BWR was performed. Tables III-49 through III-52 illustrate the results. Fewer categories are analyzed than in the PWR due to the previously mentioned lack of detail available for the systems involved in the transient event tree. Nevertheless, a significant amount of detail was given to adequately assess human errors, test and maintenance unavailabilities, and hardware, including the subsets valves and pumps. The additional category of human error and test and maintenance was added since many of the failures or increased unavailabilities could be procedure related. The results exhibit the general relationship of a combination of failures and individual failures, as documented in the PWR systems analysis. The orders of magnitude are all similar to the PWR, but larger. The large difference in all hardware between the BWR and PWR is due to differing definitions. Control and electric power were considered separately in the case of the PWR; but, in the BWR analysis, these categories were hidden within the hardware category due to a lack of further breakdown of some subsystems in the BWR. For similar reasons, the category of pumps is likely to be an inaccurate representation of its true sensitivity. One can also see that core melt probability is again a good measure of risk. In the same way as with systems, different generic categories contribute to different release categories by proportions resulting in an overestimation or an underestimation of sensitivity of public risk from using core melt probability as a sensitivity parameter. For human error and test and maintenance, the core melt parameter underestimates public risk. In the case of all hardware sensitivity, public risk is overestimated by core melt probability. The other risk parameters follow the same order, as indicated by the systems analysis. The other generic categories also show similar relationships between the risk parameters and core melt.

4. BWR Generic Increases

Examining the BWR for increases in generic component failure probabilities also indicates the avalanching effect observed in the PWR.

Table III-53 illustrates the effect of such increases on core melt probability. To obtain results in terms of public risk, a determination of the relationship of risk to core melt must be made for each category. The results indicate that test and maintenance contributions are more important for the BWR than for the PWR. Human error and other categories show slightly less sensitivity at higher factors than in the PWR. However, the orders of magnitude are very similar.

5. BWR System Uncertainty

An uncertainty analysis for the BWR was also performed. Tables III-54 and III-55 contain the results for release categories 1, 2, and 3. In comparison to the PWR, the BWR shows more sensitivity to error factor increases. The new release category error factors, for increases in all error factors, show an increase much larger than the factor by which all system and initiator error factors were increased. The ratio of the factor increase in release category error factors divided by the factor increase for each system, is about 1 for a factor increase of 3; about 2 for a factor increase of 10, and about 3 for a factor increase of 30. For the PWR the ratios are about 1, and little or no avalanching of error factors was observed. The median values listed in the tables for increasing all error factors show less increase than the analysis performed for the PWR.

The results of the analysis of increasing individual system error factors are presented in Table III-56. This table indicates that the system W is the most sensitive to increasing error factors. Almost the full increase in system error factors is felt by the release category error factor. System C shows less sensitivity in release categories 1 and 3. The systems Q, U, and V show little sensitivity to release category probability error factors. They all contribute through the same event tree, TQUV. However, their sensitivities are not exactly the same as those in the point value analysis. The ordering of the sensitivities implies that the systems with higher median values contribute more to increasing uncertainty. The system W still has by far the most profound effect on release category uncertainty. Where W is a more dominant contributor, as in category 2, the corresponding uncertainty increase is also greater. Given the results of the point value analysis and the effects of category 2 on risk, one can assume that the total uncertainty in public risk would be especially susceptible to system W.

The results for the BWR and PWR indicate that general risk sensitivity estimates for either LWR can, in some cases, be good estimates of the related sensitivity for the other reactor type. The primary difference is the slightly larger magnitude of reductions possible in the BWR due to the smaller number of contributors to public risk. Also, the BWR shows much more sensitivity to uncertainty that the PWR for the same reasons. In particular, increasing all error factors can cause even larger increases in release category probabilities in the BWR; while in the PWR, release category uncertainties generally follow increases in all error factors.

A general influence that is notable in this study concerns the differential safety gain, illustrated in the point value analysis by diminishing returns. This differential safety gain can be fitted to a power curve of the logarythm of the reduction factor. The coefficients of the power fit appear to be determined by the initial perturbation, that is, the differential safety gain from the base to a factor of three reduction in system failure probability. There seems to be a general consistency to both types of reactors and different combinations of release category probabilities (risk parameters), so further study of this property may have some value. This result reaffirms the extra safety gain noted from combinations of system reductions. In those cases a differing rate of diminishing return was noted. These results show that all sensitivities could probably be characterized with only the knowledge of an initial perturbation.

E. Summary

To summarize the major results of this analysis, the design of ECCS safety systems is excellent for mitigating the consequences of most pipe breaks. However, the smaller LOCA S2's engineered safety features should be more closely studied. In particular, reduction of the HPRS and HPIS failure rates by a factor of 3 offer up to 16 percent reductions in total property damage. In addition, the CSIS system is particularly sensitive to public risk because of its small LOCA (S2) event tree. The majority of the public risk sensitivity associated with the CSIS is human error. In particular, the miscalibration of the CLCS system could be a very important risk contribution. Additional action should be taken to reduce that failure probability and, more importantly, to assure that this particular event is actually of as low a probability as reported in the RSS. The other important contributors relate to human errors resulting in valves being left in incorrect positions. Clearly, the procedure relating to testing, maintenance, and operation of the CSIS system should be reviewed carefully and the operators and plant management should be informed of this system's importance to safety, particularly in small LOCA accident conditions. The sensitivity results indicate that reductions in CSIS failure rate of up to ten look promising and these procedures are a good place to start before considering system hardware design changes on future plants. The results of the variationsl or uncertainty analysis indicate that CSIS variations on the order of thirty can significantly affect release category 3 upper bounds. If qualitative limits are to be set for accident upper bounds, then the containment spray injection system should be monitored carefully, especially since it contributes to two of the three serious PWR accident types, including the most widely bounded of the three accidents.

The transient event tree is the most important for both reactors. In the PWR, the AFWS and PCS provide potential for significant reductions in safety. The sensitivity results show that almost fifty percent reductions in all public risk parameters would result for system failure rate reductions on the order of thirty. Since both the AFWS and PCS contribute primarily through the same event trees, a different strateg" should be considered for these systems. One could approach the AFWS and PCS as one system with subsystems, and try to reduce the overall system failure probability. The results of the combination analysis suggest that approach. There is also more sensitivity in this event tree, so reductions by factors of thirty may prove worthwhile. The major contribution to risk from this "system" is a human error. The chance that the AFWS will be inoperative because all three discharge pump valves are inadvertently left closed following a test, is assessed at 3.0×10^{-5}/year. Reducing this particular error by a factor of 10 would reduce AFWS failure probability by the same amount. Test and maintenance is the next most important factor, at about 3.0×10^{-6}/year. Consequently, reducing the procedure error rate by a factor of 30, and optimizing test and maintenance unavailability for another factor of 3 reduction, could lead to AFWS system failure rate reductions near thirty percent and public risk reductions possibly near fifty percent. By taking advantage of the PCS, more options could be invoked to develop a safety strategy, resulting in the aforementioned public risk reductions. For example, reducing the main feedwater failure probability for transient events by a factor of 3 could replace attempting to reduce test and maintenance contributions in the AFWS. In addition, reducing the probability of loss of on-site AC power after a loss of off-site AC power for between one and three hours, represented by component 182, would also contribute to reducing this very important event tree's contribution to public risk.

The results of the analysis of increase in failure probability illustrate the importance of monitoring failure probabilities in differing activities and vendors. In the particular case of the AFWS, increasing its failure probability by a factor of 10 would result in an increase in public risk by about a factor of 4. Given that the risk of AFWS failure is primarily dependent on a human error, with a wide uncertainty value, this procedure should be monitored closely by public regulators and utility employees. In the variational analysis, it was shown that the AFWS error factors must be maintained at as low a value as possible. Increases of this error factor to values near 30 would significantly affect uncertainty ranges of public risk.

To reduce public risk in the PWR by a more significant degree, some combination of simultaneous reductions would have to be done. Since combinations give higher returns for higher reductions, they can also make some system investments more worthwhile. A sensitivity study should be performed to optimize a set of system reductions. From the results of this study, factors of 100 or more in the function of the AFWS and PCS, similar factors in the CSIS, and factors lower by about 10 on the HPRS and HPIS could provide PWR risk reductions on the order of 5 or more from base values.

The BWR is dominated by transient systems. Any safety improvement policy should begin there, with a significant effort, before approachint contributions from LOCA-initiated events. Factor reductions of 100 would have to be achieved in the reactor protection system and in residual core heat removal systems before LOCA system reductions would show much benefit. There is a significant potential for reduction of public risk by reducing the failure probability of residual core heat removal. This risk is contributed by the RHR and PCS systems, which must both function. Consequently, reductions in each apply toward the total risk reduction. The RHR systems are the Low Pressure Coolant Injection and High Pressure Service Water systems. The LPCI is the much more sensitive of the two. Work on the PCS and the LPCI could result in public risk reductions of more than 2 for system failure reductions of less than 10. This BWR system definitely will show safety gains for many factor increases in reliability.

The reactor protection system also shows significant potential for public risk reduction. Since manual reserve shutdown must act in tandem with the RPS to perform this function, reductions in either can contribute to a significant reduction in the transient risk contribution. The probability of three independent rod failures indicates potential for reduction in the BWR (whereas it does not in the PWR) from its basic value of 1×10^{-4}/year. Given the studies already done, however, the most cost-effective reductions may prove to be elsewhere in the RPS, such as the manual reserve shutdown system.

By performing simultaneous reductions on the reactor protection and residual core heat removal functions, significantly greater reductions can be achieved. In the case of the PWR, many more systems contributed to public risk. This limits overall reductions to about factors of 5, unless one wishes to consider reductions in ten systems, instead of four or five. In the BWR, only two systems contribute to over ninety percent of the risk reduction potential. Conceivably, reducing RPS and EHR simultaneously could result in similar factor reductions in public risk.

The great potential of these systems for reductions also characterizes their behavior for increases in system failure probability. Any increase in RPW or RHR failure probability translates almost directly into increased public risk. The variational analysis also underscores this fact. A system of reactors that has higher uncertainty in the system W than that assessed in the RSS has almost equally higher uncertainty in public risk.

Reducing these uncertainties will also cause a reduced uncertainty in overall public risk.

This study shows that the potential for reduction of public risk in the LWR is not very high, unless one considers reducing failure rates for a combination of more than one system. In the PWR about five systems would have to be reduced, in addition to the LPIS check valve failure, in order to achieve substantial reductions. The BWR would only require two system reductions. The transient systems are the most important to analyze and offer the most potential for risk reduction. In particular, the power conversion system, PCS, plays a major role in both reactors. The reactor protection system, RPS, contributes heavily to the BWR and has a small benefit for the PWR. Both reactors show that human errors are the most important contributors to potential for public risk reduction. Reducing human error rates on the order of ten could halve public risk. In addition, human errors contribute to the uncertainty of some important system failure probabilities. The sensitivity tools developed by this study indicate the aforementioned results. They provide a basis for public decision-making in nuclear reactor safety.

Table III-1

Reduction in PWR Core Melt Probability and Total Property Damage for Reduction in Initiator Probability

Initiator	Factor Reduction in Parameter for Factor Reduction in Initiator				Ratios of Factors		
	3	10	30	100	10/3	30/10	100/30
				Core Melt Probability			
Small LOCA (S2)	1.785	2.460	2.758	2.880	1.379	1.121	1.044
Transient	1.117	1.164	1.178	1.184	1.043	1.012	1.004
LPIS ck valve	1.100	1.140	1.152	1.156	1.036	1.011	1.004
Large LOCA	1.031	1.042	1.046	1.047	1.011	1.003	1.001
Small LOCA (S1)	1.002	1.003	1.003	1.003	1.001	1.000	1.000
				Total Property Damage			
LPIS ck Valve	1.440	1.702	1.795	1.830	1.182	1.055	1.020
Transient	1.230	1.337	1.371	1.384	1.087	1.026	1.009
Small LOCA (S2)	1.197	1.286	1.313	1.323	1.074	1.001	1.008
Large LOCA	1.008	1.011	1.012	1.013	1.003	1.001	1.000
Small LOCA (S1)	1.003	1.004	1.004	1.004	1.001	1. 00	1.000

Table III-2

Reduction in PWR Early and Latent Deaths for Reduction in Initiator Probability

Initiator	Factor Reduction in Parameter for Factor Reduction in Initiator				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
			Early Deaths				
LPIS Check Valve	1.687	2.222	2.443	2.531	1.317	1.100	1.036
Transient	1.299	1.451	1.501	1.520	1.117	1.035	1.012
Small LOCA (S2)	1.029	1.040	1.043	1.044	1.010	1.003	1.001
Large LOCA	1.001	1.002	1.002	1.002	1.001	1.000	1.000
Small LOCA(S1)	1.001	1.001	1.001	1.001	1.000	1.000	1.000
			Latent Deaths				
LPIS check Valve	1.465	1.750	1.853	1.892	1.195	1.059	1.021
Transient	1.229	1.338	1.372	1.384	1.088	1.026	1.009
Small LOCA (S2)	1.184	1.266	1.292	1.301	1.069	1.020	1.007
Small LOCA (S1)	1.006	1.008	1.008	1.009	1.002	1.001	1.000
Large LOCA	1.002	1.003	1.003	1.003	1.001	1.000	1.000

TABLE III-3

Key to PWR Tables

M	Failure of the secondary system steam relief valves and the power conversion system.
L	Failure of the secondary system steam relief valves and the auxiliary feedwater system.
H	Failure of the emergency core cooling recirculation system.
D	Failure of the emergency core cooling injection system.
K	Failure of the reactor protection system.
Q	Failure of the primary system safety relief valves to reclose after opening.
C	Failure of the containment spray injection system.
F	Failure of the containment spray recirculation system.
G	Failure of the containment heat removal system.

Table III -4
New PWR Release Category Probabilities for Factor Reductions in System Failure Probability

System/Function Code(s)	factor	1	2	3	4	5	6	7
		Release Categories Leading to Core Melt						
base case (from point values WASH-1400 reduced trees)		3.9-8	6.2-6	2.6-6	1.3-11	6.7-8	5.4-7	2.0-5
H	3	3.9-8	6.2-6	2.4-6	1.3-11	4.5-8	5.4-7	1.2-5
	10	3.9-8	6.2-6	2.4-6	1.3-11	3.5-8	5.4-7	7.9-6
	100	3.9-8	6.2-6	2.4-6	1.3-11	3.4-8	5.4-7	7.7-6
D	3	3.9-8	6.2-6	2.5-6	4.2-12	5.6-8	5.3-7	1.6-5
	10	3.9-8	6.2-6	2.5-6	1.3-12	5.2-8	5.2-7	1.5-5
	30	3.9-8	6.2-6	2.5-6	4.2-13	5.1-8	5.3-7	1.4-5
	100	3.9-8	6.2-6	2.5-6	1.3-13	5.1-8	5.3-7	1.4-5
L,M	3	2.0-8	4.7-6	2.5-6	1.3-11	5.9-8	1.9-7	1.9-4
	10	1.4-8	4.2-6	2.5-6	1.3-11	5.7-8	6.6-8	1.9-4
	30	1.2-8	4.1-6	2.5-6	1.3-11	5.6-8	3.1-8	1.9-4
	100	1.1-8	4.0-6	2.5-6	1.3-11	5.6-8	1.6-8	1.9-4
C	3	3.2-8	6.2-6	1.1-6	4.7-12	6.7-8	5.3-7	2.0-5
	10	2.9-8	6.2-6	6.5-7	2.0-12	6.7-8	5.3-7	2.0-5
	30	2.9-8	6.2-6	5.1-7	1.2-12	6.7-8	5.3-7	2.0-5
	100	2.8-8	6.2-6	4.6-7	9.4-13	6.7-8	5.3-7	2.0-5
Q,K	3	3.9-8	6.2-6	2.5-6	1.3-11	6.2-8	5.4-7	2.0-4
	10	3.9-8	6.2-6	2.5-6	1.3-11	6.1-8	5.4-7	1.9-4
	30	3.9-8	6.2-6	2.5-6	1.3-11	6.0-8	5.4-7	1.9-4
	100	3.9-8	6.2-6	2.5-6	1.3-11	6.0-8	5.4-7	1.9-4
G,F	same as base case for all factors							

Table III-5

New PWR Public Risk Probabilities/Yr
for Factor Reductions of System Failure Probability

System/Function Code (s)	Factor	Early Deaths/Yr	Latent Cancers/Yr	Total Property Damage $\$10^6$/Yr
Base values (from WASH-1400 point values)		4.58-5	5.63-4	2.13-2
L,M	3	3.53-5	4.59-4	1.74-2
	10	3.16-5	4.23-4	1.74-2
	30	3.05-5	4.12-4	1.57-2
	100	3.02-5	4.09-4	1.55-2
C	3	4.46-5	4.85-4	1.99-2
	10	4.42-5	4.57-4	1.94-2
	30	4.41-5	4.49-4	1.93-2
	100	4.40-5	4.47-4	1.92-2
H	3	4.58-5	5.57-4	1.98-2
	10	4.58-5	5.55-4	1.93-2
	30	4.58-5	5.54-4	1.91-2
	100	4.58-5	5.54-4	1.90-2
G	3	4.58-5	5.59-4	2.12-2
	10	4.58-5	5.58-4	2.12-2
	30	4.57-5	5.58-4*	2.12-2
	100	4.57-5	5.57-4	2.12-2

*Apparent lack of change due to round off error.

Table III.5 (cont'd)

K,Q	3	4.58-5	5.61-4	2.12-2
	10	4.58-5	5.61-4	2.12-2
	30	4.57-5	5.61-4	2.12-2
	100	4.57-5	5.61-4	2.12-2
F	3	4.58-5	5.62-4	2.13-2
	10	4.58-5	5.62-4	2.13-2
	30	4.58-5	5.62-4	2.13-2
	100	4.58-5	5.62-4	2.13-2

TABLE III-6

Sensitivity of PWR Core Melt Probability to Reduction of System Failure Probabilities

System/ Function Code	3	10	30	100	10/3	30/10	100/30
H	1.401	1.629	1.709	1.739	1.163	1.049	1.018
D	1.155	1.222	1.242	1.250	1.058	1.016	1.006
L	1.097	1.136	1.147	1.151	1.036	1.010	1.003
M	1.097	1.136	1.147	1.151	1.036	1.010	1.003
C	1.051	1.070	1.075	1.077	1.018	1.005	1.002
Q	1.016	1.022	1.023	1.024	1.006	1.001	1.001
K	1.016	1.022	1.023	1.024	1.006	1.001	1.001
G	1.002	1.003	1.003	1.003	1.001	1.000	1.000
F	1.0005	1.0007	1.0007	1.0007	1.000	1.000	1.000

Table III-7

Reduction in PWR Public Risk of Early Death for Reductions of System Failure Probability

System/ Function Code(s)	Factor Reduction in Early Deaths Due to Reduction in System Failure Probability by a Factor of				Ratio of Factor Reductions to illustrate Diminishing Returns		
	3	10	30	100	10/3	30/10	100/30
L,M	1.299	1.450	1.500	1.519	1.116	1.034	1.013
C	1.027	1.037	1.040	1.041	1.010	1.003	1.001
G	1.001	1.002	1.002	1.002	1.001	1.000	1.000
H	1.001	1.001	1.001	1.001	1.000	1.000	1.000
D	1.000	1.001	1.001	1.001	1.001	1.000	1.000
K,Q,F	1.000	1.000	1.000	1.000	1.000	1.000	1.000

TABLE III-8

Reduction in PWR Public Risk of Latent Cancers/Yr for Reduction of System Failure Probability

System/ Function Code(s)	Factor Reduction in PWR of Public Risk of Latent Cancers/Yr due to a Reduction in System Failure Probability by a Factor of				Ratio of Factor Reductions to illustrate Diminishing Returns		
	3	10	30	100	10/3	30/10	100/30
L,M	1.266	1.332	1.365	1.378	1.052	1.025	1.010
C	1.161	1.231	1.252	1.260	1.060	1.017	1.006
H	1.010	1.014	1.015	1.016	1.004	1.001	1.001
G	1.006	1.009	1.009	1.010	1.003	1.000	1.001
D	1.005	1.007	1.008	1.008	1.002	1.001	1.000
K,Q	1.002	1.003	1.003	1.003	1.001	1.000	1.000
F	1.001	1.002	1.002	1.002	1.001	1.000	1.000

Table III-9

Reduction of PWR Total Property Damage for Reduction of System Failure Probability

System/ Function Code(s)	Factor Reduction in PWR Total Property Damage due to a Reduction of System Failure Probability by a Factor of				Ratio of Factor Reductions Illustrate Diminishing Returns		
	3	10	30	100	10/3	30/10	100/30
L,M	1.223	1.326	1.359	1.370	1.084	1.025	1.008
H	1.077	1.107	1.115	1.119	1.028	1.007	1.004
C	1.070	1.098	1.106	1.108	1.026	1.007	1.002
D	1.035	1.048	1.051	1.053	1.013	1.003	1.002
K,Q	1.005	1.006	1.007	1.007	1.001	1.001	1.000
G	1.003	1.004	1.004*	1.005	1.001	1.000*	1.001
F	1.001	1.001	1.001	1.001	1.000	1.000	1.000

*Apparent Lack of Change due to Round off Error.

TABLE III-10

System Sensitivity Breakdown for Reductions in Core Melt Probability

System/ Function Failure	Ratio of Factor Reductions 3/1	10/3	30/10	100/30
Emergency Core Cooling Recirculation (H)	1.401	1.163	1.049	1.015
High Pressure Recirculation System	1.363	1.145	1.043	1.015
157 Procedure Error (HRRS)	1.066	1.024	1.007	1.002
161 MOV to LPRS Pumps	1.066	1.024	1.007	1.002
160 MOV to Hot Legs	1.066	1.024	1.007	1.002
214 Control: sump lines	1.056	1.012	1.003	1.001
215 MOV Control	1.022	1.008	1.002	1.001
47 Section Damper	1.021	1.007	1.002	1.001
216 MOV Control	1.020	1.003	1.000	1.000
Low Pressure Recirculating System	1.020	1.007	1.002	1.001
157 Procedure Error (LPPS)	1.007	1.002	1.001	1.000
158 Procedure Error	1.007	1.002	1.001	1.000
Emergency Core Injection (D)	1.155	1.058	1.016	1.006
High Pressure Injection	1.143	1.053	1.015	1.005
107 Standby Pump	1.053	1.015	1.004	1.001
80 MOV Fails Open	1.033	1.006	1.001	1.000
227 Test and Maintenance	1.032	1.011	1.003	1.001
155 Valve Closed by Mistake	1.013	1.004	1.001	1.000
208 MOV Control	1.012	1.004	1.001	1.000
209 Detector Failure	1.010	1.004	1.001	1.000
211 MOV Control	1.010	1.001	1.000	1.000
Low Pressure Injection	1.008	1.003	1.001	1.000
156 Human Error	1.001	1.000	1.000	1.000
Accumulators	1.002	1.001	1.000	1.000

TABLE III-11

System Sensitivity Breakdown for Reductions in Latent Deaths

System Function Failure	Ratio of Factor Reductions 3/1	10/3	30/10	100/30
SSRV and AFWS(L)	1.266	1.052	1.025	1.010
143 Valves Left Closed	1.162	1.040	1.018	1.006
141 Valve Not Opened	1.024	1.006	1.002	1.001
221 Test and Maintenance (turb)	1.021	1.007	1.002	1.001
44 Diesels	1.021	1.007	1.002	1.001
222 Test and Mainten. (SOV)	1.008	1.003	1.001	1.000
201 Control Circuit	1.007	1.003	1.001	1.000
63 MSVH Valves	1.007	1.002	1.001	1.000
101 Turbine Pump	1.003	1.001	1.000	1.000
103 Pump Start	1.002	1.001	1.000	1.000
42 Header End Caps	1.002	1.001	1.000	1.000
67 Valve to Turbine	1.001	1.000	1.000	1.000
SSRV and PCS(M)	1.266	1.052	1.025	1.010
31 Main FW system	1.152	1.031	1.016	1.006
181 LOOS AC 1 hr	1.056	1.020	1.006	1.002
Containment Spray Injection	1.161	1.060	1.017	1.006
147 Miscal. CLCS	1.063	1.022	1.007	1.002
148 Valve Left Open	1.056	1.020	1.006	1.002
145 Valve Left Open	1.022	1.006	1.002	1.001
203 CLCS Control	1.001	1.004	1.001	1.000
224 Test & Maintenance	1.009	1.003	1.001	1.000
204 Pump Control	1.003	1.001	1.000	1.000
104 Pump Start	1.003	1.001	1.000	1.000

Table III-12
The Top 25 Individual Component Contributors to PWR Core Melt Probability

Component #	System	Generic Type	Factor Ratios for Reductions of Individual Component Failure Probability 3/1	10/3	30/10	100/30
157	LPRS	human	1.074	1.026	1.008	1.003
143	L:	human	1.072	1.026	1.007	1.003
31	M:FW	subsystem	1.068	1.024	1.007	1.003
182	B	electric	1.067	1.024	1.007	1.002
161	H:HPRS	human	1.066	1.024	1.007	1.002
160	H:HPRS	human	1.066	1.024	1.007	1.002
214	H:HPRS	control	1.056	1.012	1.003	1.001
107	D:HPIS	pump	1.053	1.015	1.004	1.001
80	D:HPIS	valve	1.033	1.006	1.001	1.000
227	D:HPIS	+ m	1.032	1.011	1.003	1.001
181	M	electric	1.026	1.009	1.003	1.001
215	H:HPRS	control	1.022	1.008	1.002	1.001
47	H:HPRS	hardware	1.021	1.007	1.002	1.001
147	C	human	1.021	1.007	1.002	1.001
216	H:HPRS	control	1.020	1.003	1.000	1.000
148	C	human	1.019	1.007	1.002	1.001
68	Q	valves	1.016	1.006	1.002	1.001
155	D:HPIS	human	1.013	1.004	1.001	1.000
208	D:HPIS	control	1.012	1.004	1.001	1.000
141	L:	human	1.011	1.003	1.001	1.000
209	D:HPIS	control	1.010	1.004	1.001	1.000
221	L:	+ m	1.010	1.004	1.001	1.000
44	L:	hardware	1.010	1.004	1.001	1.000
211	D:HPIS	control	1.010	1.001	1.000	1.000
45	K	hardware	1.008	1.003	1.001	1.000

TABLE III-13

The Top 20 Individual Component Contributors to Early Deaths

Component	System	Generic type	Factor Ratios for Reductions of Individual Component Failure Probability 3/1	10/3	30/10	100/30
182	B	electric	1.298	1.117	1.034	1.012
143	L	human	1.210	1.080	1.023	1.008
31	M:FW	subsystem	1.197	1.074	1.022	1.008
181	M	electric	1.070	1.025	1.007	1.003
141	L	human	1.030	1.008	1.002	1.001
221	L	+ m	1.026	1.009	1.003	1.001
44	L: diesels	hardware	1.026	1.009	1.003	1.001
147	C	human	1.011	1.004	1.001	1.000
148	C	human	1.010	1.004	1.001	1.000
222	L	+ m	1.009	1.003	1.001	1.000
201	L	control	1.009	1.003	1.001	1.000
63	L	valves	1.008	1.003	1.001	1.000
145	C	human	1.004	1.001	1.000	1.000
101	L: turbine	pump	1.003	1.001	1.000	1.000
103	L	pump	1.003	1.001	1.000	1.000
42	L	hardware	1.002	1.001	1.000	1.000
203	C: CLCS	control	1.002	1.001	1.000	1.000
224	C	+ m	1.002	1.001	1.000	1.000
67	L	valve	1.002	1.001	1.000	1.000
184	H	electric	1.001	1.001	1.000	1.000
45 3 or more rods fail			<.01%	<.01%	<.01%	<.01%

*apparent lack of change due to round off error.

TABLE III-14

The Top 20 Individual Component Contributors to Latent Cancers

Component	System	Generic type	Factor Ratios for Reductions of Individual Component Failure Probability			
			3/1	10/3	30/10	100/30
182	B	electric	1.221	1.084	1.025	1.009
143	L	human	1.162	1.040	1.018	1.006
31	M:FW	sub-system	1.152	1.031	1.016	1.006
147	C:CLCS	human	1.063	1.022	1.007	1.002
148	C	human	1.056	1.020	1.006	1.002
181	m	electric	1.056	1.020	1.006	1.002
141	L	human	1.024	1.006	1.002	1.001
145	C	human	1.022	1.006	1.002	1.001
221	L	+ m	1.021	1.007	1.002	1.000
44	L: diesels	hardware	1.021	1.007	1.002	1.001
203	C:CLCS	control	1.011	1.004	1.001	1.000
224	C	+ m	1.009	1.003	1.001	1.000
222	L	+ m	1.008	1.003	1.001	1.000
201	L	control	1.007	1.002	1.001	1.000
63	L	valve	1.007	1.002	1.001	1.000
184	H	electric	1.004	0.001	1.000	1.000
186	G	electric	1.004	1.001	1.000	1.000
101	L	pump	1.003	1.001	1.000	1.000
104	C	control	1.003	1.001	1.000	1.000
104	C	pump	1.003	1.001	1.000	1.000
45	3 or more rods fail		1.0011	1.0004	1.0001	<.01%

TABLE III-15

The Top 20 Individual Component Contributors to Total Property Damage

Component	System	Generic Type	Factor Ratios for Reductions of Individual Component Failure Probability 3/1	10/3	30/10	100/30
182	B	electric	1.212	1.080	1.023	1.008
143	L	human	1.197	1.074	1.022	1.008
31	M:FW	subsystem	1.160	1.059	1.017	1.006
181	M	electric	1.055	1.020	1.006	1.002
147	C	human	1.029	1.010	1.003	1.001
148	C	human	1.026	1.009	1.003	1.001
141	L	human	1.023	1.006	1.002	1.001
221	L	+ m	1.021	1.007	1.002	1.001
44	L: diesels	hardware	1.021	1.007	1.002	1.001
157	H: LPRS	human	1.017	1.006	1.002	1.001
161	H:HPRS	human	1.016	1.006	1.002	1.001
160	H:HPQS	human	1.016	1.006	1.002	1.001
214	H:HPRS	control	1.013	1.003	1.001	1.000
107	D:HPIS	pump	1.013	1.004	1.001	1.000
145	C	human	1.010	1.003	1.001	1.000
80	D:HPIS	valve	1.008	1.002	1.000	1.000
227	D:HPIS	+ m	1.008	1.003	1.001	1.000
222	L	control	1.007	1.002	1.001	1.000
63	L	valve	1.007	1.002	1.001	1.000
45 3 or more rods fail			1.0023	1.0008	1.0002	1.0001

Table III-16
Combinations for Perturbations
PWR Core Melt Probability

Case #	Combination	3	10	30	100	10/3	30/10	100/30
		Factor Reductions for Combination of Systems/Multiple of Single System Factor Reductions						
1	H,D,L and C	2.259	4.039	5.212	5.801	1.788	1.290	1.113
	H*D*L*C	1.860	2.419	2.620	2.697	1.301	1.803	1.029
2	H,D, and L	2.037	3.199	3.822	4.101	1.570	1.195	1.073
	H*D*L	1.775	2.262	2.435	2.503	1.274	1.076	1.028
3	H,L and C	1.733	2.332	2.587	2.690	1.346	1.109	1.040
	H*L*C	1.615	2.103	2.249	2.306	1.302	1.069	1.025
4	H and D	1.726	2.314	2.563	2.664	1.341	1.108	1.039
	H*D	1.618	1.991	2.123	2.173	1.231	1.066	1.024
5	A and L	1.599	2.024	2.190	2.254	1.266	1.082	1.029
	H*L	1.537	1.851	1.961	2.002	1.204	1.059	1.021
6	H and C	1.503	1.823	1.942	1.989	1.213	1.065	1.024
	H*C	1.472	1.743	1.838	1.873	1.184	1.055	1.019
7	H and K	1.432	1.689	1.778	1.813	1.179	1.053	1.020
	H*K	1.423	1.665	1.749	1.720	1.170	1.050	1.018
8	H and F	1.402	1.631	1.711	1.741	1.163	1.049	1.018
	H*F	1.401	1.630	1.710	1.740	1.163	1.049	1.018
9	D and C	1.224	1.327	1.360	1.372	1.084	1.025	1.009
	D*C	1.214	1.307	1.336	1.346	1.077	1.022	1.007
10	L and C	1.159	1.227	1.248	1.255	1.059	1.017	1.006
	L*C	1.153	1.215	1.234	1.241	1.054	1.016	1.006
11	L and M	1.134	1.152	1.153	1.154	1.016	1.001	1.001
	L*M	1.204	1.290	1.317	1.326	1.071	1.021	1.007
	L or M	0.097	1.136	1.147	1.152	1.036	1.010	1.004

TABLE III-17

New PWR Release Category Probabilities
for an Increase in System Failure Probability

System Function Code (s)	Factor	1	2	3	4	5	6	7
				Release Category Probabilities				
base case		3.9-8	6.2-6	2.6-6	1.3-11	6.7-8	5.4-7	2.0-5
B	3	3.9-8	6.2-6	2.9-6	1.3-11	1.3-7	5.4-7	4.5-5
	10	3.9-8	6.2-6	3.9-6	1.3-11	3.6-7	5.4-7	1.3-4
	30	3.9-8	6.2-6	7.1-6	1.3-11	1.0-6	5.5-7	3.8-4
	100	3.9-8	6.2-6	1.8-5	1.3-11	3.3-6	5.6-7	1-2-3
D	3	3.9-8	6.2-6	2.7-6	3.8-11	9.8-8	5.7-7	3.2-5
	10	3.9-8	6.2-6	3.3-6	1.3-10	2.1-7	6.6-7	7.2-5
	30	3.9-8	6.2-6	4.9-6	3.8-10	5-3-7	9.1-7	1.9-4
	100	3.9-8	6.2-6	1.0-5	1.3-9	1.6-6	1.8-6	6.1-4
L,M	3	9.5-8	1.1-5	2.7-6	1.3-11	8.9-8	1.6-6	2.2-5
	10	2.9-7	2.6-5	3.1-6	1.3-11	1.7-7	5.3-6	3.0-5
	30	8.4-7	7.1-5	4.2-6	1.3-11	3.9-7	1.6-5	5.1-5
	100	2.8-6	2.3-4	8.1-6	1.3-11	1.2-6	5.3-5	1.3-4
K,Q	3	3.9-8	6.2-6	2.6-6	1.3-11	8.0-8	5.4-7	2.1-5
	10	3.9-8	6.2-6	2.9-6	1.3-11	1.3-7	5.4-7	2.6-5
	30	3.9-8	6.2-6	3.6-6	1.3-11	2.6-7	5.4-7	3.9-5
	100	3.9-8	6.2-6	6.0-6	1.3-11	7.4-7	5.4-7	8.5-5
F	3	3.9-8	6.2-6	2.6-6	1.3-11	6.7-8	5.4-7	2.0-5
	10	4.0-8	6.2-6	2.7-6	1.3-11	6.7-8	5.4-7	2.0-5
	30	4.2-8	6.2-6	3.2-6	1.3-11	6.7-8	5.5-7	2.0-5
	100	4.9-8	6.2-6	4.6-6	1.3-11	6.7-8	5.7-7	2.0-5
G	3	4.0-8	6.2-6	2.7-6	1.4-11	6.7-8	5.4-7	2.0-5
	10	4.3-8	6.2-6	3.4-6	2.0-11	6.7-8	5.4-7	2.0-5
	30	5.2-8	6.2-6	5.4-6	3.6-11	6.7-8	5.4-7	2.0-5
	100	8.4-8	6.2-6	1.2-5	9.4-11	6.7-8	5.4-7	2.0-5
C	3	6.0-8	6.2-6	6.8-6	3.6-11	6.7-8	5.7-7	2.0-5
	10	1.3-7	6.2-6	2.2-5	1.2-10	6.7-8	6.6-7	2.0-5
	30	3.5-7	6.2-6	6.4-5	3.5-10	6.7-8	9.1-7	2.0-5
	100	1.1-6	6.2-6	2.1-4	1.2-9	6.7-8	1.8-6	2.0-5

TABLE III-18

New PWR Public Risk Parameters
for an Increase in System Failure Probability

System/ Function Code(s)	Factor	Early Deaths/yr	Latent Cancers/yr	Total Property Damage $10^6$$
base value		4.58-5	5.63-4	2.13-2
M,L	3	7.74-5	8.74-4	3.29-2
	10	1.88-4	1.97-3	7.37-2
	30	5.03-4	5.08-3	1.90-1
	100	1.61-3	1.60-2	5.97-1
C	3	4.94-5	7.97-4	2.55-2
	10	6.21-5	1.62-3	4.02-2
	30	9.82-5	3.96-3	8.23-2
	100	2.25-4	1.22-2	2.30-1
H	3	4.59-5	5.80-4	2.59-2
	10	4.64-5	6.41-4	4.18-2
	30	4.76-5	8.16-4	8.75-2
	100	5.20-5	1.43-3	2.47-1
G	3	4.60-5	5.74-4	2.15-2
	10	4.65-5	6.13-4	2.22-2
	30	4.81-5	7.20-4	2.41-2
	100	5.38-5	1.10-3	3.09-2
D	3	4.59-5	5.72-4	2.35-2
	10	4.61-5	6.03-4	3.10-2
	30	4.67-5	6.93-4	5.26-2
	100	4.90-5	1.01-3	1.28-1
K,Q	3	4.58-5	5.67-4	2.16-2
	10	4.59-5	5.80-4	2.26-2
	30	4.62-5	6.19-4	2.56-2
	100	4.72-5	7.54-4	3.59-2
F	3	4.59-5	5.65-4	2.13-2
	10	4.60-5	5.73-4	2.15-2
	30	4.63-5	5.97-4	2.19-2
	100	4.75-5	6.79-4	2.34-2

Table III-19

Increase in PWR Core Melt Probability for an Increase in System Failure Probability

System/ Function Code(s)	Factor Increase in Core Melt Probability For Factor Increase in System Failure Probability				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
H	1.86	4.86	13.4	43.5	2.61	2.76	3.25
D	1.40	2.82	6.86	21.0	2.01	2.43	3.06
L,M	1.27	2.20	4.86	14.2	1.73	2.21	2.92
C	1.15	1.65	3.11	8.19	1.43	1.88	2.63
K.Q	1.05	1.21	1.69	3.35	1.15	1.40	1.98
G	1.01	1.03	1.10	1.33	1.02	1.07	1.21
F	1.00	1.01	1.02	1.07	1.01	1.01	1.05

Table III-20

Increase in PWR Public Risk of Early Death for an Increase in System Failure Probability

System/Function Code(s)	Factor Increase in Early Death for Factor Increase in System Failure Probability				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
M,L	1.69	4.11	11.0	35.2	2.43	2.68	3.20
C	1.08	1.36	2.14	4.91	1.26	1.57	2.29
G	1.00	1.02	1.05	1.17	1.02	1.03	1.11
H	1.00	1.01	1.04	1.13	1.01	1.03	1.09
D	1.00	1.01	1.02	1.07	1.01	1.01	1.05
K,A,F	all less than 1.01						

Table III-21

Increase In PWR Public Risk of Latent Cancers/yr for an Increase in System Failure Probability

System/Function Code(s)	Factor Increase in Latent Cancers for Increase in System Failure Probability				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
M,L	1.55	3.49	9.03	28.4	2.35	2.59	3.15
C	1.42	2.88	7.04	21.6	2.03	2.44	3.07
G	1.02	1.09	1.28	1.95	1.07	1.17	1.52
H	1.03	1.14	1.45	2.54	1.11	1.27	1.25
D	1.02	1.07	1.23	1.79	1.05	1.15	1.46
K,Q	1.01	1.03	1.10	1.34	1.02	1.07	1.22
F	1.00	1.02	1.06	1.21	1.02	1.04	1.16

<u>Table III-22</u>

<u>Increase in PWR Total Property Damage 10^6\$/yr for an Increase in System Failure Probability</u>

System/Function Code(s)	Factor Increase in Total Property Damage for an Increase in System Failure Probability				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
D	1.10	1.46	2.47	6.01	1.33	1.69	2.43
M,L	1.55	3.46	8.92	28.0	2.23	2.58	3.14
C	1.20	1.89	3.87	10.8	1.58	2.05	2.79
H	1.21	1.96	4.11	11.6	1.62	2.10	2.82
K,Q	1.01	1.06	1.20	1.69	1.05	1.14	1.41
G	1.01	1.04	1.13	1.45	1.03	1.09	1.28
F	1.00	1.01	1.03	1.10	1.01	1.02	1.07

TABLE III-23

New Release Category Probabilities for a Reduction in Generic Component Failure Probabilities

Generic Failure Type	Factor	1	2	3	4	5	6	7
		Release Category Probabilities						
base value		3.9-8	6.2-6	2.6-6	1.3-11	6.7-8	5.4-7	2.0-5
Human Error	3	1.6-8	4.9-6	1.0-6	4.2-12	4.3-8	2.2-7	1.3-5
	10	7.8-9	4.5-6	5.5-7	1.8-12	3.5-8	1.2-7	1.0-5
	30	5.6-9	4.4-6	4.1-7	1.2-12	3.3-8	8.5-8	9.8-6
	100	4.9-9	4.3-6	3.6-7	1.0-12	3.2-8	7.5-8	9.5-6
Control	3	3.8-8	6.2-6	2.4-6	9.8-12	5.5-8	5.2-7	1.6-5
	10	3.7-8	6.1-6	2.3-6	9.1-12	5.2-8	5.2-7	1.5-5
	30	3.7-8	6.1-6	2.3-6	8.9-12	5.2-8	5.2-7	1.5-5
	100	3.7-8	6.1-6	2.3-6	8.8-12	5.1-8	5.2-7	1.5-5
Electric Power	3	1.8-8	4.6-6	2.5-6	1.2-11	6.5-8	1.6-7	2.0-5
	10	1.3-8	4.2-6	2.5-6	1.2-11	6.4-8	5.2-8	2.0-5
	30	1.2-8	4.1-6	2.5-6	1.2-11	6.4-8	2.6-8	2.0-5
	100	1.1-8	4.0-6	2.5-6	1.2-11	6.3-8	1.7-8	2.0-5
Test and Maintenance	3	3.6-8	6.0-6	2.4-6	1.0-11	6.1-8	4.9-7	1.9-5
	10	3.5-8	5.9-6	2.4-6	9.13-12	5.9-8	4.7-7	1.8-5
	30	3.4-8	5.9-6	2.4-6	9.1-12	5.9-8	4.6-7	1.8-5
	100	3.4-8	5.9-6	2.4-6	9.0-12	5.9-8	4.6-7	1.8-5
Pumps	3	3.8-8	6.2-6	2.5-6	9.7-12	6.2-8	5.3-7	1.8-5
	10	3.8-8	6.2-6	2.5-6	9.1-12	6.1-8	5.2-7	1.8-5
	30	3.8-8	6.2-6	2.5-6	8.9-12	6.1-8	8.2-7	1.8-5
	100	3.8-8	6.2-6	2.5-6	8.8-12	6.1-8	5.2-7	1.8-5
Values	3	3.8-8	6.2-6	2.5-6	9.6-12	5.8-8	5.2-7	1.8-5
	10	3.8-8	6.1-6	2.5-6	8.9-12	5.5-8	5.2-7	1.7-5
	30	3.8-8	6.1-6	2.5-6	8.7-12	5.4-8	5.1-7	1.7-5
	100	3.8-8	6.1-6	2.5-6	8.6-12	5.4-8	5.1-7	1.7-5

TABLE III-24

New PWR Risk Parameters for a Reduction in Generic Component Failure Probabilities

base value	Factor	Early Deaths/yr	Latent Cancers/yr	Total Property Damage 10^9 \$
Human Error	3	4.58-5	5.63-4	2.13-2
	10	3.18-5	3.31-4	1.33-2
	30	3.08-5	3.15-4	1.27-2
	100	3.05-5	3.09-4	1.25-2
Electric Power	3	3.42-5	4.48-4	1.71-2
	10	3.11-5	4.17-4	1.60-2
	30	3.04-5	4.09-4	1.57-2
	100	3.01-5	4.07-4	1.56-2
Test and Maintenance	3	4.41-5	5.41-4	2.04-2
	10	4.35-5	5.33-4	2.01-2
	30	4.34-5	5.30-4	2.00-2
	100	4.33-5	5.30-4	2.00-2
Control	3	4.53-5	5.48-3	2.03-2
	10	4.51-5	5.43-3	2.01-2
	30	4.51-5	5.42-3	2.00-2
	100	4.50-5	5.42-3	2.00-2
Pumps	3	4.55-5	5.57-4	2.09-2
	10	4.54-5	5.55-4	2.08-2
	30	4.54-5	5.54-4	2.07-2
	100	4.53-5	5.54-4	2.07-2
Valves	3	4.53-5	5.55-4	2.07-2
	10	4.51-5	5.53-4	2.06-2
	30	4.51-5	5.52-4	2.05-2
	100	4.51-5	5.52-4	2.05-2

Table III-25

Reduction in PWR Core Melt Probability
for a Reduction in Generic Component Failure Probabilities

Generic Failure Type	Factor Reduction in Core Melt Probability for Factor Reduction in Generic Failure Probabilities				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
Human Error	1.531	1.873	2.000	2.049	1.223	1.068	1.025
Control	1.161	1.199	1.208	1.210	1.033	1.008	1.002
Electric Power	1.085	1.111	1.118	1.120	1.024	1.006	1.002
Test and Maintenance	1.060	1.082	1.089	1.091	1.021	1.006	1.002
Pumps	1.061	1.077	1.081	1.082	1.015	1.004	1.001
Valves	1.078	1.102	1.108	1.110	1.022	1.005	1.002
All Hardware	1.194	1.259	1.277	1.283	1.054	1.014	1.005

Table III-26

Reduction in PWR Public Risk of Early Death
for a Reduction in Generic Component Failure Probabilities

Generic Failure Type	Factor Reduction in Early Death				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
Human Error	1.298	1.441	1.448	1.505	1.110	1.033	1.011
Electric Power	1.338	1.471	1.510	1.523	1.099	1.027	1.009
Test and Maintenance	1.038	1.052	1.056	1.058	1.013	1.004	1.003
Control	1.012	1.016	1.017	1.017	1.004	1.000	1.000
Pumps	1.007	1.010	1.010	1.011	1.003	1.003	1.001
Valves	1.012	1.015	1.017	1.017	1.003	1.002	1.000
All Hardware	1.047	1.063	1.068	1.070	1.015	1.005	1.002

Table III-27

Reduction in PWR Public Risk of Latent Cancers/Yr. for a Reduction in Generic Component Failure Probabilities

Generic Failure Type	Factor Reduction in Latent Cancers/Yr. 3	10	30	100	Ratio of Factors 10/3	30/10	100/30
Human Error	1.446	1.698	1.786	1.819	1.174	1.052	1.018
Electric Power	1.257	1.350	1.375	1.384	1.074	1.019	1.007
Test and Maintenance	1.041	1.056	1.061	1.062	1.014	1.005	1.001
Control	1.027	1.035	1.038	1.038	1.008	1.003	1.000
Pumps	1.011	1.014	1.015	1.016	1.003	1.001	1.001
Valves	1.014	1.018	1.020	1.020	1.004	1.002	1.000
All Hardware	1.050	1.067	1.071	1.073	1.016	1.004	1.002

Table III-28

Reduction in PWR Total Property Damage 10^6\$ for a Reduction in Generic Component Failure Probabilities

Generic Failure Type	Factor Reduction in Total Property Damage				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
Human Error	1.390	1.601	1.673	1.699	1.152	1.045	1.016
Control	1.049	1.060	1.063	1.064	1.010	1.003	1.001
Electric Power	1.244	1.330	1.355	1.363	1.069	1.019	1.006
Test and Maintenance	1.044	1.060	1.065	1.067	1.015	1.005	1.002
Pumps	1.021	1.026	1.028	1.028	1.005	1.002	1.000
Valves	1.027	1.036	1.038	1.039	1.009	1.002	1.001
All Hardware	1.080	1.105	1.112	1.115	1.023	1.006	1.003

TABLE III-29

Generic Sensitivity Breakdown for Reductions in Total Property Damage

Generic Failure	Ratio of Factor Reductions			
	3/1	10/3	30/100	100/30
Human Error	1.390	1.152	1.045	1.016
143 Valves Left Closed	1.197	1.074	1.022	1.008
147 Manual: CLCS	1.029	1.010	1.003	1.001
148 Valves Left Open	1.026	1.009	1.003	1.001
141 Valve not Opened	1.023	1.006	1.002	1.001
157 Procedure Error	1.017	1.006	1.002	1.001
161 MOV to Hot	1.016	1.006	1.002	1.001
160 MOV to LPRS Pumps	1.016	1.006	1.002	1.001
145 Valve Left Open	1.010	1.003	1.001	1.000
155 Valve Closed	1.003	1.001	1.000	1.000
158 Procedure Error	1.002	1.001	1.000	1.000
All Hardware	1.080	1.023	1.006	1.003
44 Diesels	1.021	1.007	1.002	1.001
107 Standby Pump	1.013	1.004	1.001	1.000
80 Valve Fails to Open	1.008	1.002	1.000	1.000
63 MSVH Valve	1.007	1.002	1.001	1.000
47 Suction Damper	1.005	1.002	1.001	1.000
68 Relief Valves Fail	1.005	1.002	1.001	1.000
101 Turbine Pump	1.003	1.001	1.000	1.000
45 3 or More Rods Fail	1.002	1 001	1.000	1.000
103 Pump Start	1.002	1.001	1.000	1.000
42 Header End Cap	1.002	1.001	1.000	1.000

TABLE III-30

Generic Sensitivity Breakdown for Reduction in Early Deaths

System Function Failure	Ratio of Factor Reductions 3/1	10/3	30/10	100/30
Test and Maintenance	1.038	1.013	1.004	1.002
221 L:Turbine	1.026	1.009	1.003	1.001
222 L:SOV-102	1.009	1.003	1.001	1.000
224 C:Spray Subsystem	1.002	1.001	1.000	1.000
227 Changing Pumps				
223 1 drain RPS	all contribute	<.01%		
230 F: inside Legs				
Control	1.012	1.004	1.001	1.000
201 Circuit Fails	1.009	1.003	1.001	1.000
203 CLCS	1.002	1.001	1.000	1.000
204 Pump	1.001	1.000	1.000	1.000

TABLE III-31

Increase in PWR Core Melt Probability for an Increase in Generic Component Failure Probability

Generic Failure Type	Factor Increase in Core Melt Probability for Factor Increase in Generic Failure Probabilities				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
Human Error	2.10	7.16	73.6	864	3.41	10.28	11.74
Control	1.95	11.6	94.0	1126	5.95	8.10	11.98
Electric Power	1.39	4.57	29.2	29.7	3.29	6.39	10.17
Test and Maintenance	1.17	1.76	3.46	9.44	1.50	1.97	2.73
Valves	1.35	4.18	25.9	261	3.10	6.20	10.08
Pumps	1.32	4.26	28.5	345	3.23	6.69	12.11

TABLE III-32

Increase in PWR Public Risk of Early Death for

an Increase in Generic Component Failure Probabilities

Generic Failure Type	Factor Increase in Early Death for Factor Increase in Generic Failure Probabilities				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
Human Error	1.80	7.11	73.1	2044	3.95	10.28	27.96
Control	1.05	1.40	3.84	29.8	1.33	2.74	7.76
Electric Power	2.29	13.1	97.5	1019	5.72	7.44	10.45
Test and Maintenance	1.11	1.50	2.60	6.47	1.35	1.73	2.49
Valves	1.04	1.25	2.47	13.9	1.20	1.98	5.63
Pumps	1.03	1.19	2.54	30.4	1.16	2.13	11.97

TABLE III-33

Increase in PWR Public Risk of Latent Cancers/yr

for an Increase in Generic Component Failure Probabilities

Generic Failure Type	Factor Increase in Latent Cancers for Factor Increase in Generic Failure Probabilities				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
Human Error	2.08	8.53	73.6	1766	4.10	8.63	23.99
Control	1.12	2.08	9.34	93.0	1.86	4.49	9.96
Electric Power	2.06	11.0	81.1	848	5.34	7.37	10.46
Test and Maintenance	1.12	1.53	2.60	6.47	1.37	1.70	2.49
Valves	1.05	1.34	3.15	20.9	1.28	2.35	6.63
Pumps	1.05	1.39	4.72	77.0	1.32	3.40	44.7

TABLE III-34

Increase in PWR Total Property Damage 10^6\$ for an

Increase in Generic Component Failure Probabilities

Generic Failure Type	Factor Increase in Total Property Damage for Factor Increase in Generic Failure Probability				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
Human Error	1.96	7.54	66.0	1674	3.85	8.75	25.36
Control	1.29	4.03	26.9	298	3.12	6.68	11.08
Electric Power	2.00	10.4	76.0	793	5.20	7.31	10.57
Test and Maintenance	1.13	1.16	2.84	7.29	1.03	2.49	2.57
Valves	1.12	1.97	8.24	74.7	1.76	4.18	9.07
Pumps	1.10	1.97	9.37	119	1.79	4.76	12.70

avalanching can help to set boundary criterion

Table III-35
PWR
LPRS System
Variational Analysis

Failure Type and Original Median and (Error Factor)	New Component Error Factor	Ratio of New to Old Median Failure Probability and Error Factor due to Increase in Generic Failure Error Factor for the Probability Distribution of:					
		LPRS Failure		Release Category 2		Release Category 3	
		median	EF	median	EF	median	EF
Common mode Operator Failures	10	1.25	2.50	1.000	1.000	1.000	1.000
$3x10^{-3}$(3)	30	1.58	6.48	1.000	1.000	1.000	1.000
Control Subsystem A,B	10	1.09	1.89	1.000	1.000	1.000	1.000
$3.2x10^{-3}$(3)	30	1.17	6.60	1.000	1.000	1.000	1.000
Valve Maintenance	10	1.02	1.00	1.000	1.000	1.000	1.000
$2.1x10^{-3}$(3)	30	1.07	1.05	1.000	1.000	1.000	1.000
Valve Subsystems A,B	10	1.01	1.00	1.000	1.000	1.000	1.000
$1.0x10^{-3}$(3)	30	1.05	1.04	1.000	1.000	1.000	1.000

*EF - error factor

TABLE III-36
Variational Analysis

PWR Initiators

New Release Category Median Probabilities,
New 95% Confidence Limit, and Ratio's of
New to Base Values for Increasing Initiator
Error Factor (EF)

	Release Category Probability Characteristics		
base case	1	2	3
point value	6.05-8	7.88-6	2.74-6
median value	1.10-7	1.30-5	4.08-6
95% limit	6.73-7	7.99-5	3.63-5
increasing LOCAs			
new EF = 30			
new median,(ratio to base)	1.26-7(1.15)	1.30-5(1.00)	5.33-6(1.31)
95% limit,(ratio to base)	9.63-7(1.43)	7.99-5(1.00)	9.33-5(2.57)
increase transient			
new EF = 6			
new median (ratio to base)	1.27-7(1.15)	1.51-5(1.16)	4.27-6(1.05)
95% limit (ratio to base)	1.04-6(1.55)	1.07-4(1.34)	3.65-5(1.01)
increase vessel rupture			
new EF = 30			
new median (ratio to base)	1.10-8(1.00)	1.30-5(1.00)	4.08-6(1.00)
95% limit (ratio to base)	6.73-7(1.00)	7.99-5(1.00)	3.63-5(1.00)
new EF = 100			
new median (ratio to base)	1.10-7(1.00)	1.31-5(1.01)	4.14-6(1.01)
95% limit (ratio to base)	6.73-7(1.00)	7.99-5(1.00)	3.63-5(1.00)

TABLE III-36

Variational Analysis
(cont'd)

	1	2	3
increase all initiators			
EF's x 3			
new median (ratio to base)	1.51-7(1.37)	1.73-5(1.33)	5.53-6(1.36)
95% limit (ratio to base)	1.29-6(1.92)	1.94-4(2.43)	9.46-5(2.61)
increasing LPIS ck. valve			
new EF = 30			
new median (ratio to base)		1.44-5(1.00)	
new 95% limit (ratio to base)		1.53-4(1.91)	

TABLE III-37
PWR
Variational Analysis
Increasing All Error Factors (EF)
Release Category 1

Release Category 1	Base case	Ratio of Median/point = 1.82	
Point Value	6.050-8	Confidence	
median	1.099-7	Level of point	
EF (95%)	6.1	value ~ 28%	
Increase all EF by 3		Ratios of:	
		median	error factor (95%)
median	2.257-7	2.05	
EF(95%)			3.26
confidence level of point	~ 20%		
ratio of med/ point	3.73		
Increase all EF by 10			
median	5.332.7	4.85	
EF (95%)	77.0		12.56
confidence level of point	~ 19%		
ratio of med/ pt	8.81		
Increase all EF by 30			
median	1.255-6	11.42	
EF (95%)	201.1		32.90
confidence level of point	~ 17%		
ratio of med/pt	20.7		

TABLE III-38

PWR
Variational Analysis
Increasing All Error Factors
Release Category 2

Release Category 2 base case

Ratio of median/point = 1.66
Confidence Level of Point Value ~ 30%

point value	7.880-6		
median	1.305-5		
EF (95%)	6.13		
		Ratios of:	
Increase all EF by 3		median	error factor(95%)
median	2.371-5	1.82	
EF(95%)	19.2		3.13
confidence level of point	~ 20%		
ratio of med/point	3.73		
Increase all EF by 10			
median	4.386	3.36	
EF (95%)	73.7		12.02
confidence level of point	~ 22%		
ratio of med/pt	5.57		
Increase all EF by 30			
median	7.07-5	5.83	
EF (95%)	152.8		24.93
confidence level of point	~22%		
ratio of med/pt	9.65		

TABLE III-39

PWR

Variational Analysis
Increasing All Error Factors (EF)
Release Category 3

Release Category 3	base case	ratio of median/point = 1.149 confidence level of point value ~ 37%	
point value	2.739-6		
median	4.083-6		
EF (95%)	8.90		
Increasing all EF by 3			
median	8.825-6	2.16	
EF (95%)	22.6		2.54
confidence (point)	24%		
ratio med/pt	3.23		
Increase all EF by 10			
median	2.789-5	6.83	
EF (95%)	69.5		7.81
confidence	~ 14%		
ratio med/pt	10.2		
Increase all EF by 30			
median	8.723-5	21.36	
EF (95%)	185.6		20.85
confidence	~ 10%		
ratio med/pt	31.9		

TABLE III-40

Variational Analysis
PWR Systems Uncertainty

New Release Category Median Values and Error Factors for Increasing System Error Factor by a Factor of 3 and 10

System Code	Factor	Release Categories 1 median	1 error factor	2 median	2 error factor	3 median	3 error factor
L	3	1.39-7	11.9	1.62-5	9.4	4.34-6	8.7
	10	1.82-7	28.0	2.13-5	19.6	4.74-6	10.0
M	3	1.33-7	11.2	1.61-5	8.9	4.32-6	8.7
	10	1.66-7	27.3	2.08-5	17.7	4.67-6	9.0
C	3	1.18-7	7.4	*	*	4.35-6	16.2
	10	1.28-7	11.6	*	*	4.51-6	43.6
F	3	1.12-7	6.2	*	*	4.36-6	9.4
	10	1.20-7	6.6	*	*	4.89-6	10.7
G	3	1.12-7	6.1	*	*	4.31-6	8.6
	10	1.17-7	6.1	*	*	4.71-6	10.3
K	3	*	*	*	*	4.20-6	8.7
	10	*	*	*	*	4.52-6	8.67

* no contribution to that release category

TABLE III-41

Variational Analysis

PWR Public Risk of Latent Deaths/yr
for an Increase in System Error Factors
by Factors of 3 and 10

System Function Code	Factor	lower bound	median	upper bound
	base case	base case = 1.112-3	confidence level = 41%	
L	3	3.286-4 (4.99)	1.639-3	1.083-2 (6.61)
	10	3.197-4 (6.79)	2.172-3	2.712-2 (12.5)
M	3	3.268-4 (4.95)	1.617-3	1.007-2 (6.23)
	10	3.251-4 (6.40)	2.082-3	2.398-2 (11.5)
C	3	3.224-4 (4.62)	1.491-3	7.648-3 (5.13)
	10	3.145-4 (5.46)	1.718-3	1.361-2 (7.92)
F	3	3.355-4 (4.10)	1.375-3	6.495-3 (4.72)
	10	3.440-4 (4.19)	1443-3	6.699-3 (4.64)*

* all others had insignificant effects on the median and error factor of the latent deaths probability distribution.

TABLE III-41A

Key to BWR Tables

W	Failure to remove residual core heat
C	Failure of the reactor protection system
Q	Failure of normal feedwater system to provide core make up water
U	Failure of HPCI or RCIC to provide core make up water
V	Failure of the low pressure ECCS to provide core make up water

TABLE III-42

New PWR Probabilities for Releases 1, 2 and 3

for Factor Reduction in Function/System Failure Probability

System/Function Code	Factor	Release Category Probabilities 1	2	3
base case		2.28-7	2.94-6	1.795-5
W	3	1.35-7	1.07-6	1.065-5
	10	1.02-7	4.20-7	8.05-6
	30	9.27-8	2.33-7	7.32-6
	100	8.94-8	1.68-7	6.96-6
C	3	1.74-7	2.94-6	1.365-5
	10	1.55-7	2.94-6	1.220-5
	30	1.50-7	2.-4-6	1.176-5
	100	1.48-7	2.94-6	1.161-5
Q,U,V	3	2.23-7	2.85-6	1.758-5
	10	2.22-7	2.81-6	1.745-5
	30	2.21-7	2.80-6	1.742-5
	100	2.21-7	2.80-6	1.741-5

TABLE III-43

Reduction in PWR Core Melt Probability and Total Property Damage for a Reduction in System Failure Probability

System/Function Code (s)	Factor Reduction due to a reduction in System Failure Probability				Ratio of Factors to Illustrate Diminishing Return		
	3	10	30	100	10/3	30/10	100/30
	Core Melt Probability						
W	1.790	2.475	2.779	2.904	1.383	1.123	1.045
C	1.255	1.328	1.418	1.432	1.098	1.029	1.010
Q,U,V	1.022	1.031	1.033	1.034	1.009	1.002	1.001
	Total Property Damage						
W	1.746	2.363	2.628	2.736	1.353	1.112	1.041
C	1.279	1.418	1.463	1.479	1.109	1.032	1.011
Q,U,V	1.022	1.029	1.032	1.033	1.007	1.003	1.001

TABLE III-44

Reduction in BWR Public Risk of Early and Latent Deaths for a Reduction in System Failure Probability

System/Functon Code(s)	Factor Reduction due to reduction in system failure probability				Ratio of Factors to Illustrate Diminishing Return		
	3	10	30	100	10/3	30/10	100/30
			Early Deaths				
W	1.762	2.403	2.682	2.796	1.364	1.116	1.043
C	1.270	1.378	1.446	1.462	1.085	1.049	1.081
Q,U,V	1.022	1.030	1.032	1.033	1.008	1.002	1.001
			Latent Deaths				
W	1.724	2.310	2.558	2.658	1.340	1.107	1.039
C	1.292	1.438	1.487	1.504	1.113	1.034	1.011
Q,U,V	1.021	1.029	1.031	1.032	1.008	1.003	1.001

TABLE III-45

BWR System Breakdown for Summary of Release Categories 1, 2 and 3 for a Reduction in Function/System Failure Probability

Function/System Failure is:	Factor Reduction in Release Categories due to reduction by a factor of			
	3	10	30	100
Remove Residual Core Heat	1.790	2.475	2.779	2.904
Power Conversion System	1.790	2.475	2.779	2.904
Residual Heat Removal	1.790	2.475	2.779	2.904
Low Pressure Coolant Injection	1.422	1.668	1.755	1.788
High Pressure Service Water	1.172	1.244	1.266	1.274
Reactor Protection	1.255	1.378	1.418	1.432
Manual Reserve Shutdown	1.255	1.378	1.418	1.432
Reactor Protection System	1.255	1.378	1.418	1.432
3 or more rods fail	1.082	1.114	1.123	1.127
Normal Feedwater System Makeup	1.022	1.031	1.033	1.034
with onsite AC	1.016	1.021	1.023	1.024
without onsite AC	1.006	1.009	1.009	1.010
HPCI or RCIC Makeup	1.022	1.031	1.033	1.034
HPCI	1.012	1.015	1.016	1.017
HPCI test and maintenance	1.010	1.013	1.014	0.015
RCIC	1.011	1.014	1.015	0.016
RCIC test and maintenance	1.011	1.014	1.016	1.016
Low Pressure ECCS Makeup	1.022	1.031	1.033	1.034
Manual Activation	1.012	1.017	1.018	1.018
Low Pressure ECCS System	1.010	1.013	1.014	1.015

TABLE III-46

Increase in BWR Public Risk of Early and Latent Deaths for an Increase in System Failure Probability

	Increase in Parameter for an Increase in System Failure Probability				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
			Early Deaths				
W	2.30	6.84	19.8	65.2	2.97	2.89	3.29
C	1.64	3.87	10.3	32.6	2.36	2.66	3.17
Q,U,V	1.06	1.29	1.94	4.19	1.22	1.50	2.16
			Latent Deaths				
W	2.26	6.67	19.3	63.4	2.95	2.89	3.28
C	1.68	4.05	10.8	34.5	2.41	2.67	3.19
Q,U,V	1.06	1.28	1.91	4.10	1.21	1.49	2.15

TABLE III-47

Increase in BWR and Total Property Damage for an Increase in System Failure Probability

	Increase in Parameter for an Increase in System Failure Probability				Ratio of Factors		
	3	10	30	100	10/3	30/10	100/30
	Core Melt Probability						
W	2.32	6.96	20.2	66.2	3.00	2.90	3.28
C	1.61	3.74	9.84	31.2	2.32	2.63	3.17
Q,U,V	1.07	1.30	1.95	4.26	1.21	1.50	2.18
	Total Property Damage						
W	2.28	6.77	19.6	64.4	2.97	2.90	3.29
C	1.66	3.95	10.5	33.4	2.38	2.66	3.18
Q,U,V	1.06	1.29	1.92	4.15	1.22	1.49	2.16

TABLE III-48

BWR System Breakdown for Summary of Release Categories
1,2 and 3 for Increase in Function/System Failure Probability

Function/ System Failure in:	Factor Reduction in Release Categories Due to a Failure Reduction by a Factor of			
	3	10	30	100
Remove Residual Core Heat	2.32	6.96	20.2	66.2
Power Conversion System	2.32	6.96	20.2	66.2
Residual Heat Removal	2.32	6.96	20.2	66.2
Low Pressure Coolant Injection	1.89	5.01	13.9	45.1
High Pressure Service Meter	1.49	3.81	8.95	29.8
Reactor Protection	1.61	3.74	9.84	31.2
Manual Reserve Shutdown	1.61	3.74	9.84	3.2
Reactor Protection System	1.61	3.74	9.84	31.2
3 or More Rods Fail	1.23	2.02	4.29	12.2
HPCI or RCIC Makeup	1.07	1.30	1.95	4.26
HPCI	1.04	1.15	1.48	2.60
HPCI Test and Maintenance	1.03	1.13	1.42	2.44
RCIC	1.03	1.13	1.44	2.47
RCIC Test and Maintenance	1.03	1.14	1.46	2.56
Low Pressure ECCS Makeup	1.07	1.30	1.95	4.26
Manual Activation ADS	1.04	1.16	1.53	2.80
Low Pressure ECCS System	1.03	1.13	1.42	2.45
Normal Feedwater System Makeup	1.07	1.30	1.95	4.26

TABLE III-49

Sensitivity of BWR Core Melt Probability to Reduction in Failure Probability of Various Generic Failure Types

Generic Failure Type	Factor Reduction in Core Melt Probability Due to Reduction in Failure Probability by a Factor of				Ratio of Factor Reductions to Illustrate Diminishing Returns		
	3	10	30	100	10/3	30/10	100/30
Human Error and Test & Maintenance	1.693	2.119	2.271	2.328	1.252	1.072	1.025
Human Error	1.491	1.749	1.835	1.866	1.173	1.049	1.017
All Hardware	1.446	1.711	1.806	1.841	1.183	1.056	1.019
Test & Maintenance	1.101	1.138	1.149	1.153	1.034	1.010	1.003
Valves	1.052	1.071	1.076	1.078	1.018	1.005	1.002
Pumps	1.003	1.008	1.009	1.009	1.005	1.001	1.000

TABLE III-50

Reduction in BWR Public Risk of Early Deaths for a Reduction in Generic Component Failure Probabilities

Generic Failure Type	Factor Reduction due to Reduction in Generic Component Probabilities				Ratio of Factors to Illustrate Diminishing Return		
	3	10	30	100	10/3	30/10	100/30
Human Error and Test and Maintenance	1.712	2.156	2.316	2.376	1.259	1.074	1.026
Human error	1.491	1.780	1.871	1.904	1.194	1.051	1.018
All Hardware	1.434	1.687	1.777	1.810	1.176	1.053	1.019
Test and Maintenance	1.099	1.136	1.146	1.150	1.034	1.009	1.003
Valves	1.051	1.069	1.075	1.077	1.017	1.006	1.002
Pumps	1.006	1.008	1.009	1.009	1.002	1.001	1.001

TABLE III-51

Reduction in BWR Public Risk of Latent Deaths for a Reduction in Generic Component Failure Probabilities

Generic Failure Type	Factor Reduction due to Reduction in Generic Component Probabilities				Ratio of Factors to Illustrate Diminishing Return		
	3	10	30	100	10/3	30/10	100/30
Human Error and Test Maintenance	1.739	2.209	2.380	2.445	1.270	1.077	1.027
Human Error	1.534	1.825	1.924	1.960	1.190	1.054	1.019
All Hardware	1.416	1.654	1.738	1.769	1.168	1.051	1.018
Test and Maintenance	1.097	1.132	1.143	1.146	1.032	1.010	1.003
Valves	1.049	1.067	1.072	1.074	1.017	1.005	1.002
Pumps	1.006	1.008	1.008	1.009	1.002	1.000	1.001

TABLE III-52

Reduction in BWR Total Property Damage for
a Reduction in Generic Component Failure Probabilities

Human Error and Test and Maintenance	Factor Reduction due to Reduction in Generic Component Probabilities				Ratio of Factors to Illustrate Diminishing Return		
	3	10	30	100	10/3	30/10	100/30
	1.723	2.178	2.343	2.405	1.264	1.076	1.026
Human Error	1.520	1.799	1.893	1.928	1.184	1.052	1.018
All Hardware	1.426	1.673	1.760	1.792	1.173	1.052	1.018
Test and Maintenance	1.098	1.134	1.145	1.148	1.033	1.010	1.003
Valves	1.050	1.068	1.074	1.076	1.017	1.006	1.002
Pumps	1.006	1.008	1.009	1.009	1.002	1.001	1.000

TABLE III-53

Increase in BWR Sum of Releases 1, 2, and 3 for an Increase in Generic Component Failure Probabilities

Generic Failure Type	Factor Increase due to Increase in Generic Component Probabilities				Ratio of Factors to Illustrate Increasing Return		
	3	10	30	100	10/3	30/10	100/30
Human Error and Test and Maintenance	2.85	16.7	119.	1211.	5.86	7.13	10.18
Human Error	2.40	12.2	82.0	814.	5.08	6.72	9.93
All Hardware	1.95	5.66	20.8	179.	2.90	3.67	8.61
Test and Maintenance	1.34	3.31	15.6	136	2.47	4.71	8.72
Valves	1.15	1.82	4.94	31.1	1.57	2.71	8.32
Pumps	1.02	1.11	1.61	7.83	1.09	1.45	4.86

Table III-54

Variational Analysis
BWR
Increasing All Error Factors

Release Category 1 and 3

Ratio of Probability of Release 3 to Probability of Release
1 = 7.93 - 1
Ratio of Median/point = 1.43
Confidence Level ~ 33%
of point value

Point Values	3.318-7		
Median	4.760-7		
EF (95%)	5.67	Median	Ratios of EF
All EF increased by 3			
median 8.768-7		1.84	
EF (95%) 20.37			3.59
by 10			
median 1.853-6		3.89	
EF (95%) 111.5			19.66
by 30			
median 3.541-6		7.44	
EF (95%) 484.9			85.52

Table III-55

BWR
Variational Analysis
Increasing All Error Factors (EF)
Release Category 2

Release Category 2		base case	ratio of median/point = 1.42
Point Value		3.996-6	confidence
Median		5.670-6	level of point value ∿ 39%
EF(95%)		7.769	
New Values		median ratios	of error factor (95%)
Increase all EF by 3			
median	9.630-6	1.70	
EF(95%)	32.74		4.21
Increase all EF by 10			
median	2.115-5	3.73	
EF (95%)	220.4		28.37
Increase all EF by 30			
median	3.221-5	5.68	
EF(95%)	673.5		86.69

TABLE III-56

Variational Analysis

BWR System Uncertainty

System Code	Factor	Release Category Characteristics 1 median	1 upper bound	2 median	2 upper bound
base case		4.76-7	2.70-6	5.67-6	4.41-5
W	3	5.59-7	6.30-6	6.82-6	1.16-4
		1.17	2.33	1.20	2.63
	10	6.71-7	2.02-5	8.51-6	3.97-4
		1.41	7.48	1.50	9.00
	30	7.92-7	6.15-5	1.08-5	1.21-3
		1.66	22.8	1.90	27.4
C	3	5.46-7	3.66-6	*	*
		1.15	1.36		
		Event C Does Not contribute			
	10	6.16-7	7.37-6	*	*
		1.29	2.73		
	30	6.69-7	1.79-5	*	*
		1.41	6.63		
Q	3	5.21-7	3.38-6	6.39-6	6.20-5
		1.09	1.25	1.13	1.41
	10	5.73-7	4.98-6	6.92-6	9.04-5
		1.20	1.84	1.22	2.05
U	3	5.09-7	3.10-6	6.08-6	551-5
		1.07	1.15	1.07	1.25
	10	5.43-7	4.35-6	6.51-6	8.07-5
		1.14	1.61	1.15	1.83
V	3	5.09-7	3.02-6	6.12-6	5.33-5
		1.07	1.12	1.08	1.21
	10	5.55-7	3.94-6	6.53-6	7.18-5
		1.17	1.46	1.15	1.63

*Does not contribute

IV. Conclusions and Recommendations

The computer code LWRSEN is developed to provide a calculational method for determining the sensitivity of the RSS results to changes in the input data. With this code the sensitivity of public risk to point value failure rates may be explored. The computer code PLMODMC, which was previously developed under NRC contract research, is used to calculate sensitivity to failure probability distribution uncertainties and to establish relations between point value and probabilistic approaches to sensitivity calculations. These codes provide the sensitivity analysis tools to help in decision making in research, quality assurance, inspection, and regulation. In addition to the development of the methodology for calculating sensitivity, this study performs a sensitivity analysis of RSS values for system and individual component failure probabilities. The results of that analysis from the basis for addressing the questions in the introduction.

1. What are the characteristics of sensitivity of public risk to reductions or increases in input failure rates?

The magnitude of risk reduction for reductions of both system and generic failure rates are generally less than, or about equal to, two, even for large reductions of up to factors of one hundred in failure rates. Another general characteristic of the results are the diminishing returns found at high reductions. In general, only reductions on the order of ten seem practical, with only a few major contributors deserving reductions as high as thirty.

These results imply that there is no easy way to further reduce the risks associated with the two nuclear power plants analyzed in the RSS.

They also imply that there are no dominant failure modes in LWR's,* but a combination of failure modes of roughly the same order of magnitude. There are, however, four or five systems or functions, and two or three generic categories, which are dominant contributors. Reducing these to the level of the other failure modes probably results in an overall system design for which nothing short of a reduction of all failure modes, or some significant change in initiators or containment failure modes, could result in a significant further reduction in public risk.

The magnitudes associated with increasing failure probabilities are much higher than found in the reduction analysis. This is to be expected, because while reducing certain failure rates eventually results in other failure rates dominating, increasing certain failure rates results in one mode becoming more and more dominant. Not surprisingly, the dominant system increases are the same as the systems which dominate the reductions. By examining the increases, systems which are very sensitive can be identified so that utilities, vendors and the NRC can be sure values like those in the RSS are actually achieved. This should be of value to quality assurance managers and inspectors responsible for public safety.

2. What are the characteristics of the sensitivities to increases in only the uncertainty of system failure rate probability distributions?

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*The LPIS check valve failure rates have been reduced for most, resulting in approximately fifty percent less risk.

In the case of the BWR, increasing the error spread of the function (removal of residual core heat) by a factor of ten results in an increase of the median value by about fifty percent, and a ninefold increase in the upper bound. Therefore, a system of reactors such as those in the United States, or some small subset such as a multireactor utility, could have higher risk just from a lack of efficient quality assurance from reactor to reactor. This fifty percent increase is equal to or greater than most reductions. When one considers the effect of the significantly higher upper bound on public risk calculations, as well as public confidence, the importance of monitoring safety system reliability with careful quality assurance programs is evident. Uncertainty analysis on the system level can provide information on which systems the quality control must be more strictly maintained. An analysis of uncertainties of generic classes is not performed, but, given the results of the point value generic analysis and the systems uncertainty analysis, generic category uncertainties could also be very important. The uncertainty analysis also gives an indication of propagation of system errors.

3. What are the major areas of potential public risk reduction?

For reducing public risk by improving the reliability of present systems, the generic class of human error events offers the most potential. Human error is more sensitive than any one engineered safety feature and would also appear to be the least costly to remedy. Work in this area would also have the added benefit of reducing concern over human initiated events, in addition to possibly increasing the ability of human intervention to mitigate accident consequences. The systems which contribute the most to public risk, as measured by early and latent deaths and total property damage, are those designed to mitigate the consequences of transient events. In the case of the PWR, the only contribution to risk from containment safety systems is the core spray injection system; otherwise, transient and ECCS systems dominate the sensitivity to risk. In the BWR, only transient events are considered, since they comprise almost one hundred percent of the risk.

4. What is the relationship between system and generic sensitivities and their individual component failure event sensitivities?

The results of the breakdown and combination analysis indicate that an approximate relationship between individual sensitivities and sensitivities to combinations of individual failures exists for low sensitivities with ratios near 1.0. It also reaffirms a strategy of reducing dominant failure modes to the levels of other contributors as the most effective means of reducing public risk.

5. What is the synergistic effect of combining failure rate reductions?

By reducing more than one important failure rate at a time, the total sensitivity is larger than appropriate combinations of the sensitivity to single failure rates. The values of further reductions are also larger. For failure rates which are not very important (those with sensitivities only slightly greater than one), the approximate magnitude of a combination of failure rates can be estimated by the multiples of the individual sensitivities.

6. How do generic classes of failure affect risk as compared to system failures?

The differences between systems and generic sensitivity characteristics are best illustrated by the results of the sensitivity to factor increases in failure rates. While system effects tend to be additive, since a system cannot fail twice in a failure event, generic effects can be multiplicative. One or more human errors can combine, through and-gates in system fault trees, to produce risk increases greater than the initial factor by which all the failure probabilities of generic type are increased. For example, increasing human error by a factor of one hundred results in an increase in latent cancers of almost two thousand over the base case results. The avalanching effect should be carefully monitored for each generic type. Different generic types start to "take off" at different factor increases. These take off points should provide upper bounds for allowable variation of failure rates from plant to plant or utility to utility. In the case of human error, this take off point is between factors of ten and thirty. To be more conservative, lower points may be chosen to insure that large increases in public risk do not occur for particular nuclear power plants. In addition, sensitivities to smaller factor reductions are larger for generic classes than for any systems. The system sensitivities tend to increase slightly faster at higher factor reductions, however.

7. What parameters are best used to estimate effects on overall public risk?

This study shows that core melt probability is an effective measure of public risk for the BWR. In the case of the PWR, however, the larger probability of low consequence core melt accidents, such as release category 7, make core melt a misleading indicator. The most important release categories are numbers 1, 2, and 3.

8. What are the limitations of this study?

Any limitations from inaccuracies in the risk models are addressed with other limitations in the methodology in Section II. In the summary of Section III, modifications of the results, which are necessary to account for differences between a particular reactor and those addressed in the RSS, are outlined.

The limitations include the approximation used in WASH-1400 and the limitations of the code developed here. It should also be noted that this analysis does not include sensitivities to containment failure modes, and uses only the reduced fault trees of WASH-1400. However, because this analysis deals only with ratios of changes, many of these approximations may cancel.

It should be noted that the above conclusions are based on the analysis of the specific BWR and PWR analyzed in WASH-1400. Although it is suspected that these conclusions may apply much more generally, no conclusion about plants of a different design should be reached without an analysis of that specific design. However, this study demonstrates that, given the fault trees and event trees for any plant, the methods presented here can be directly applied to provide a sensitivity analysis of that plant.

Finally, the results of this study indicate the following recommendations.

1. Further research on the relation between point value and probabilistic techniques should be made, so that these

results and similar ones can be used to estimate probabilistic results without the expensive calculations inherent to such techniques.

2. The characteristic of the differential safety gains, represented by succeeding factor reductions, indicate generalities that should be studied further. The values of differential safety gains seem to be dependent only on the magnitude of small changes from the base value.

3. Some method of combining these results with other studies which provide for parametric description of risk probability distributions, should be attempted, so that an overall model, which will minimize the need for time consuming, expensive, detailed calculations, can be established.

4. Further study should be done to assess the application of this work to the licensing process. The feasibility of developing a more rational decision-making process, dependent on a specific methodology, may eventually be developed.

This study provides a calculation framework for analyzing the sensitivity of risk to the input variable of the WASH-1400 analysis. It can be used to provide a better understanding of how further risk reduction can be obtained most efficiently. Such methods should be a useful tool for industry and government.

Appendix A

Acronyms

A list of acronyms, and the phrases they stand for, is provided in order to aid the reader. The acronyms used are the same as those used in the Reactor Safety Study.

ACC	Accumulators
ADS	Automatic Depressurization System
AFWS	Auxilliary Feedwater System
CHRS	Containment Heat Removal System
CLCS	Consequence Limiting Control System
CSIS	Containment Spray Injection System
CSRS	Containment Spray Recirculation System
ECRS	Emergency Core Cooling Recirculation System
ECI	Emergency Core Injection
EF	Error Factor
HPCI	High Pressure Coolant Injection
HPIS	High Pressure Injection System
HPRS	High Pressure Recirculation System
HPSW	High Pressure Service Water
LPCI	Low Pressure Coolant Injection
LPECCS	Low Pressure Emergency Core Cooling System
LPIS	Low Pressure Injection System
LPRS	Low Pressure Recirculation System
PCS	Power Conversion System

Acronyms (Continued)

RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPS	Reactor Protection System
RSS	Reactor Safety Study
SSRV	Secondary Steam Relief Valves
SICS	Safety Injection Control System
T & M	Test and Maintenance

Appendix B

LWRSEN Computer Code User's Manual

1. Introduction

The LWRSEN computer code was written to calculate the values of public risk for light water nuclear reactors using the methodology developed in the RSS. The code calcualtes the sensitivity to changes in the basic inputs to the public risk calculation. These sensitivities are then output under certain formats depending on the type of sensitivity calculations performed. Basically, the three main subroutines, COMP, COMBIN, and ATTR, calculate respectively: (1) individual sensitivities; (2) system, generic, or combinations of sensitivities; and (3) combinations of sensitivities and breakdown by individual components.

The code begins with the dominant event trees and uses system failure rates to calculate release category probabilities. The system failure rates are calculated from user-supplied subroutines. This flexibility allows the user the advantage of using the code for his specific reactor. In addition, the user may choose to develop the complexity of the system fault trees to his own desired level of completeness. For example, one could input a system failure equation of one hundred elements for one system while, at the same time, giving only a point value without a tree for another system. Later in this manual a key is given to program statements for which changes may be necessary to accommodate more than 250 and 130 components for the PWR and BWR, respectively. Two models for calculating public risk are also given; one for each reactor. For many of the reactors located in the United States, only minor modifications to the model would be required in order to use it for an analysis similar to that reported. The model may easily be replaced by other more specific or advanced models. If the reactors are identical to those chosen in the RSS, then this program could be used without modification.

2. Input Preparation and Use

To prepare the code for use the first step would be to compare the reactor under consideration with the two reactors analyzed in the RSS. The system fault trees for those reactors are illustrated in Appendix C. The equations for those systems, as well as comments on the contributions to risk, may be found in the listing of the system subroutines at the end of this section. However, the user may choose to develop his own fault trees. This is recommended since it would be just as easy for a user to familiarize himself with his own reactor's fault trees as to study and become familiar with the fault trees for the reactors used in the RSS.

The routines COMBIN and ATTR employ the subroutine FACTOR to vary system and/or generic failure probabilities. The capabilities of FACTOR should be considered when one develops his own system fault trees. FACTOR allows for thirty modules to be named for each reactor type. The array PCHNG controls the system subroutines by allowing for sensitivity of modules which may comprise an entire function, a system within a function, or some user-chosen subsystem or module of components. The subroutine FACTOR, upon receiving input in array AA of a number less than thirty, activates a flag which will cause that module to be reduced by the factor VERIBY. Additionally, FACTOR allows for structured data in the array of individual components: PCMPNT (250) for the PWR, and CMPNT (130) for the BWR. Upon inputting to FACTOR a component number which is a multiple of ten, the subroutines will vary the nine components between the next multiple of ten. This allows for generic classifications of data. The base case data for the RSS was constructed this way. In that analysis components in the category of human errors were placed in elements 140 through 179 of the PCMPUT array. Therefore, to change human errors one would input to FACTOR through AA component numbers 140, 150, 160, and 170. Then all of the human error components could be varied at once. The structure is available for as much classification as the user wishes.

The routine ATTR allows for additional classification of data without predetermined structures. One may input an attribute for each component in the array ATTR (10, 250). The number of attributes for each array is given by the variable NATTR (10). There are places for ten such cases to be run by the routine ATTR. A component with no attribute is signified by inputting a zero. The ATTR routine then copies the component numbers of the same attribute into the array AA for input to FACTOR. Up to fifty components may be combined in this way for calculations. Attributes may include all types of designations. For example, location within the plant could be considered an attribute; thus, common mode effects dependent upon fire or earthquake can be approximated by varying all components in the same general environment for an earthquake or fire initiator. While this approach may not adequately describe common mode effects, it could perhaps be used to set intuitive bounds.

Given the available data structure, one should attempt to visualize unusual modules when constructing system subroutines. Taking advantage of the ability of FACTOR to change structures of data when designing the system fault tree equations can lead to simple characterization of data. Finally, the use of attributes in the ATTR routine provides an additional classification and combination ability for unusual categories of components for which the structural data approach is inapplicable or for which the classification was unnoticed at the time of the fault tree identification and construction process.

For the situation where user-supplied system fault tree routines are used, the number of components for either the PWR (250) or the BWR (130) may be too small. In this case the arrays may be expanded to allow for more complex trees. Tables B-1 and B-2 contain a list of all program statements which must be changed to facilitate this expansion. This method was used to avoid excessive waste of memory by the code. Table B-2 contains the location of the risk models so that they can be easily changed to fit a specific reactor or permit substitution of a different model.

Once the user decides whether to use the supplied system subroutines, component unavailabilities, and risk models, and follows the process described above for replacing supplied routines, the next step is to compile the code and store the compiled version for easy access. Table B-3 provides the job control language (JCL) for the IBM 370 virtual machine. The code was written to run on this machine. A standard FORTRAN IV was used as the programming language as referenced by Reference 1. Any differences among "standard" FORTRAN IV's should be corrected after identification by the compiler.

The next process is the description of the input flow for calculations. All input cards and their associated variables and formats may be found in Table B-4. The card numbers correspond to requirements for data. Cards 1 through 14 are required for all cases. Cards designated by a prefix B or P correspond to input cards for either reactor type. The choice of cards here is dependent on which reactor was chosen by the user with the variable REACTOR. The cards with the prefix C alone are control cards and indicate whether calculations are to be performed by each main routine. C1 through C3 apply to the routines COMP, COMBIN, and ATTR, respectively. The prefix CA refers to the necessary cards for the COMP routine. The prefixes CB and CC correspond to COMBIN and ATTR.

The first set of cards is the basic input to the program. The variable COMJOB is available for ten cards of input comments to provide job title, etc. The variable REACTOR chooses reactor type; the variables PRNTCOM and PARAM provide for the risk parameter choice; and the array VERIBY stores the four factors by which failure probabilities are to be reduced. (Note that factors less than one may be input, which in effect calls for increases by the factor's inverse.) The reactor-dependent values are input next. These arrays and variable lists accept the component unavailabilities and containment event tree probabilities for the reactor type chosen.

If one wishes to calculate and sort by magnitude the individual component sensitivities, the routine COMP is activated by setting NCASE equal to 0 (or inserting a blank card). The variable N determines how many sensitivities are to be ordered and printed out. The arrays DESIG and COM provide literal values for comments to describe each individual component. The output generated by this section of the code will be covered later.

If the user wishes to calculate sensitivities using the structured data or the modules developed in the system subroutines, then he should use the subroutine COMBIN. Card C2, which follows C1 if COMP is not used, or the last card in the CA series if COMP is used, controls activation of this routine. A number not equal to zero is interpreted as the number of cases to be run. Following that, a comment card and three more cards are input for each case. The three cards contain inputs to the array MCASE. These values are, in turn, transferred to the array AA for use by the FACTOR subroutine.

Finally, if the user wishes to calculate sensitivities to a group of components with the same attributes and then break them down by their individual sensitivities in order of increasing magnitude, the card C3, for the subroutine ATTR, is set equal to the number of cases of group sensitivities. Then a comment card for the array HCOM is read in, followed by a card with the number of attributes for this case (NATTR), followed by the attribute of each component number on seven cards for the array ATTR. These attributes are later searched to find the components to be varied as a whole and then individually to illustrate sensitivity breakdowns. A sample input listing and program output is given.

3. <u>Output</u>

The output of the code LWRSEN is dependent upon which routines are chosen by the user. A title page and a listing of base values and input values are printed by the routine DEBUGO for every run. A reduced version of this output is presented in Table B-5. The contents of the variable COMJOB which provides the user with job-specific comments is printed. Then the base values of the release categories are printed. The input values of component unavailabilities in the fault tree and the containment failure probabilities are listed next. Finally, a list of initiator probabilities and a list of system failure probabilities are given. This information is adequate to debug the inputs including the system fault tree equations.

The routine COMP outputs the values for sensitivities for individual components. The sensitivities are calculated for each factor of VERIBY. The sensitivities are printed out for the N largest sensitivities for each factor of VERIBY. Finally, the ratio of successive factors of VERIBY are printed out in the order of the sensitivity of the initial perturbations. The values of the arrays DESIG and COM are printed with their sensitivity values. These provide space for literal designations, such as generic classifications, as well as comment space for further identification of the individual failure. The component number and order number from the sensitivity sorting are also given. An example of the output from COMP is given in Tables B-6 and B-7.

If one wishes to calculate the sensitivity to some combination of components, modules of components, or generic classes of components, then the routine COMBIN should be used. The type of sensitivity parameter and the sensitivity of the combination are printed out. The sensitivity to core melt probability is given, followed by the new release category probabilities. The base values of each release category and the sensitivities of each category are printed below the new values. The COMBIN routine has the additional advantage of printing out additional sensitivity information. It can be used for important individual failure probabilities simply by inputting the component number. Following the category sensitivities, the factor VERIBY is printed along with the component numbers input to the FACTOR subroutine. The variable HCOM for that case is printed to provide 80 characters of case comment for the user. The above format is executed for each of the four factors of VERIBY and for each of NCASE cases. An example of the output of COMBIN is given in Table B-8.

The last major piece of output comes from the other major routine, ATTR. The ATTR routine's output is very similar to both the COMBIN and COMP routines. The last comment, sensitivity parameter, and sensitivity of the attribute considered are printed. Output similar to COMBIN is also printed giving the rest of the sensitivity information. Finally, the breakdown of the total sensitivity by the components which contribute is printed out in an information format similar to that of the COMP routine. An example of the output of the ATTR routine is given in Table B-9. It is an output page from the sample case.

4. Program Structure

The program is structured so that it can be changed so as to meet the needs of specific reactors, yet at the same time does not contain an extraordinary amount of generality which results in time and memory inefficiencies. In order to achieve this result the user must become involved in the actual construction of the final program. For this reason, an attempt was made to write a computer code which would be easily understandable. Consequently, the program was written so as to be very structured. That is, many subroutines were constructed to perform specific functions. Levels of programming were also developed to more clearly identify and separate important functions. In the listing of the program every routine is allotted one page. This helps the user by exposing him to only one level or function of the program at once. Communication between the subroutines is by argument, but system, component, and other important values are left in common blocks for access by most routines. An example of the simplicity of construction is the COMBIN routine. In order to calculate the sensitivity information the methodology of the RSS must be used. In LWRSEN, sensitivity to public risk is calculated the following way:

```
CALL FACTOR (PCMPNT, AA, PCHNG, VERI)
CALL SYSTEM
CALL RLESE7 (RLEASE)
CALL OUTPUT (RLEASE, BASE, III, VERIBY)
CALL RISK (RLEASE, BASE CM, SNESUM, PARAM, RISC, REACTOR)
```

These subroutines are all on the same level, as indicated by Table B-11, except for the OUTPUT subroutine. These subroutines were all called d·ring the execution of the routine COMBIN. The subroutine FACTOR varies a component, a group of components, or a module of components by the factor VERI. SYSTEM calls all the lower level system subroutines which in turn calculate all the new system failure probabilities. RLESE7 operates on the array RLEASE which contains the release category probabilities. The subroutine OUTPUT performs output functions, but it initially calls the subroutine RISK. The subroutine RISK takes the release category probabilities and translates them into public risk and sensitivities. The same general structure is maintained in the program, with specific levels only calling lower levels. In this way algebraic equations, confusing GO TO statements, and DO loops are limited to operating on very few basic concepts at once. The programmer may then deal with the programming details on a smaller subroutine level. This also allows the user to make easy modifications. The way the program works is controlled by its subroutines, so if one wishes to expand the capabilities of the program, as well as changing the characteristics of the system subroutines, the user may change only small modules without fear of destroying the basic methodology of the program.

In the following three subsections there are brief descriptions of each subroutine and its important characteristics, flow diagrams for further aid, and a sample input. Finally, a listing with comments is provided, together with the input, output, and control processes discussed in this manual. The examples of each in the tables are adequate and a set of diagnostic tools has been provided for the user to easily calculate sensitivity to public risk for all different varieties of design changes or extreme situations.

5. Descriptions of Subroutines

This section defines the scope of each subroutine and the method by which it completes its purpose.

MAIN The main program has three purposes. It reads the input and stores it in common access for the subroutines. Secondly, the program calculates the base-case values for each release category and risk parameter. It then outputs those values, comments, and important input values. Finally, it controls program flow by calling the main subroutines, COMP, COMBINE, and ATTR, which are subprograms calculating individual sensitivities, system and generic sensitivities, and breakdown of sensitivities, respectively.

COMP This subprogram calculates the sensitivities of each individual component to the chosen risk parameter: core melt probability, release category probabilities, early deaths, latent cancers, and total property damage. The differential values and N-factor sensitivity ratios are also calculated. The sensitivities are then sorted in order of greatest sensitivity by SORT. Finally, the program outputs these values for the top N-chosen sensitivities through the output subroutine OUTTOP.

COMBIN This subprogram calculates sensitivities to systems, generic categories, and arbitrary combinations. It also calculates sensitivities for each parameter and values of VERIBY. The program calls the subroutine OUTPUT to print all the release category probabilities and risk parameters, and their component sensitivities. The code will input up to fifty cases of up to fifty systems, generic classes, or individual components for each case.

ATTR This subprogram uses the results of the COMP routine and must be run in tandem with it, using any value of N. A calculation of some combination is made and output through the subroutine OUTATT. Then the sensitivity of every element, sorted by its contribution to the combination, is output by the routine OUTTTT. The sensitivity parameter chosen bv the user is used for the sensitivity breakdown.

FACTOR This subroutine takes the elements of the system fault trees and varies some combination of them as determined by the array AA and the factor VERI. The array AA contains component identification numbers, system identification numbers, or generic identification numbers.

SORT This subroutine sorts the input array in order of highest value. A parameter is input to indicate the number of passes and, hence, the top number of sorted values. The array INDEX serves as a pointer for the sorted values such that the input array remains unchanged. The method of sorting is a bubble sort. This method starts from the bottom of an array and bubbles up the higher values by comparisons.

SYSTEM This subroutine serves as an intermediate step in the control process. It calls all the lower-level system subroutines such that all values for the systems may be accessed for further calculations.

RLESE7 This subroutine calculates release category probabilities from the values calculated by the system subroutines activated by SYSTEM in a previous call. The release category functions are dependent on reactor type, as represented by REACTOR.

RISK This subroutine takes release category probabilities and calculates the values of various risk parameters. It also then calculates the sensitivities to all of the above, plus core melt probability. It also sets the value of RISK, the parameter chosen for sensitivity comparisons, depending on the user-chosen value of PARAM.

DEBUGO This subroutine serves as the output routine for the MAIN program. This output comprises the title page of the program output, as well as an input check. Values inputted to the system fault trees and containment event trees are reproduced along with base values of the release category probabilities, system failure probabilities, and risk parameters.

OUTTOP This subroutine serves as the output routine for the routine COMP. The top N sensitivities are output with component number, generic category, and a comment. In its final call from COMP it prints out the values of differential sensitivity ratios.

OUTPUT This subroutine serves as the output routine for the routine COMBIN. It outputs the comment code of the sensitivity parameter and its value. It also prints out core melt sensitivity, release category values and their sensitivities, the case comment, and value of the factor VERIBY.

OUTATT This subroutine serves as one of two output routines for the routine ATTR. It outputs the heading of the breakdown analysis and values for the sensitivity parameters and release categories for the combination.

OUTTTT This subroutine complements OUTATT by printing out the sensitivities of the components making up the combination being examined by ATTR. It also prints out a measure of the contribution of that component to the total combination's sensitivity. Information about the component's routine is also printed.

Failure Subroutines The following subroutines require algebraic equations or point values for systems necessary for the calculation of release category probabilities. The system variable(s) found are listed after each routine by their familiar codes from the RSS. For the PWR: LFAILP, L; MBPFAL, M, B^1; QFAILP, Q; KFAILP, K; CFAILP, C; DFAILP, D; DI, DZ; HFAILP, H, HS; FFAILP, F; GFAILP, G; BFAILP, B. For the BWR: WFAIL, W; CFAIL, C; QUVFAL, Q, U, V.

TABLE B-1

The following table contains the changes necessary to increase the number of components for the system analysis to more than 250 for the PWR and 130 for the BWR. The statement numbers correspond to those from the subroutines listed.

SORT

2	2 uses	of 250
4	1	250
9	1	250

FACTOR

3	1 use of 250

WFAIL, QUVFAL, and all other system subroutines

2	1 use of 250	1 use of 130

RLESE7, SYSTEM, OUTPUT

2	1 use of 250	1 use of 130

OUTTIT

4	1 use of 250	
5	2	250

TABLE B-1 (CONT.)

MAIN				
2	1 use of	250	1 use of	130
4	2	250		
6	1	250		
9	1	250		
10	1	250		
25	1	250		
26	1	250		
28			1	130
29			1	130
44	1	250		
49			1	130
67	1	250		
COMP				
2	1 use of	250		
3	1	250		
4	1	250		
6	1	250		
22	1	250		
OUTTOP				
2	2 uses of	250		
4	2	250		

TABLE B-1(CONT.)

DEBUGO		
2	1 use of 250	1 use of 130
8	1 250	
12		1 130
ATTR		
2	1 use of 250	1 use of 130
3	1 250	
4	1 250	
5	1 250	
12	1 250	
COMBIN		
2	1 250	
3	1 250	
14	1 250	
18		1 use of 130

TABLE B-2

Job Control Language (JCL) for IBM 370

When compiling and running

```
System Headings
//S1 EXEC FORG60
// C. SYSIN DD  , DCB = BLKSIZE = 2000

System Subroutines
program modules
//G. SYSIN DD*
data cards
/*
/*EOJ
```

When running previously compiled modules and system subroutines

```
//S1 EXEC PGM = LWRSEN
//G. SYSIN DD*
            data cards
/*
/* EOJ
```

Table B-3

INPUT VARIABLES IN ORDER OF READ STATEMENTS

Card #	Variable Format Code	Comment
1 through 10	COMJOB (10,80)	ten cards of 80 character comments for title page and job characteristics
11	REACTOR F3.2	identifies reactor type 1.0 = PWR 2.0 = BWR
12	PRNTCM (30) 30A1	comment describing sensitivity parameter
13	PARAM	identifies risk parameter for sensitivity 0 = core melt 1→7 = Release category # 8 = early deaths 9 = latent cancers 10 = total property damage
14	VERIBY (4) 4(F8.4)	sensitivity factors range 9999. to .001

The next two inputs are reactor dependent

		for	PWR REACTR = 1.0
P1 through P25	PCMPNT(I)		individual component unavailabilities and initiators
	25(10(E7.1, 1X)		
P26	ALPHA		steam explosion for hot release
	ALPHA1		steam explosion for cold release
	BETA		isolation failure
	GAMMA		hydrogen burning

Table B-3 (cont.)

Card #	Variable Format Code	Comment
	DELTA	overpressurization
	DELTA	overpressurization from transient event
	EPSILN	melt through
	EPSLNT	melt through from transient
	EPSBHF	melt through given LOCA and systems B, H, or F
	9(2x, F6.4)	
	or for	BWR REACTR = 2.0
B1 through B13	CMPNT (I)	individual component unavailabilities and initiator
	13(10(E 7.1, 1x)	
B14	ALPHAB	steam explosion in the vessel
	GAMMAB	overpressure release through reactor building
	GAMAPB	overpressure release direct to atmosphere

The next inputs relate to program control

C1	NCASE	= 0 do COMP routine
		≠ 0 go to COMBIN
	I3	
		if NCASE was equal to 0
CA1	N	# of sensitivity values to be sorted
	I3	

TABLE B-3 (CONT.)

Card #	Variable Format Code	Comment
CA2 through CA(n+1) where n = # of non zero components	DESIG (250, 30)	designation (generic classification) for each non zero component
	COM (250, 50)	comment for each individual component that is non zero
C2	NCASE	≠ 0 do COMBIN routine for NCASE cases up to 50 cases
		= 0 go to ATTR
	I3	
		if NCASE was ≠ 0
CB1 through CB (a) where a=# of inputs	HCOM (50, 80)	Comment for each case up to fifty inputs for sensitivity combination for use by FACTOR
CB (a+1) through CB(4C)	3(20(I3, 1x)1)	subroutine in array AA
		$\leq$ 30 indicates system sensitivity
		= multiple of ten, the next 9 components for generic applications
		= any other #, that component
C3	NCASE	= 0 go to STOP
		≠ do ATTR routine for NCASE cases
	I3	up to 10 cases
		if NCASE was ≠ 0
CC1 through CCa	HCOM (10, 80)	comment for each case
	80A1	
where a = NCASE		
CCa + 1 through CC 2a	NATTR (10) I3	# of attributes for this case

TABLE B-3 (CONT.)

Card #	Variable Format Code	Comment
(C 2a+1 through CC 8a)	(6(40I21), 10I21)	Parameter for further sub-grouping reactor components that are not covered by system or generic groupings.

TABLE B-4

BASE CASE OUTPUT FOR LWRSEN COMPUTER CODE

THE SENSITIVITY PARAMETER FOR THIS RUN IS CORE MELT

THE SENSITIVITY FACTORS FOR THIS RUN ARE 3 10 30 100

RELEASE CATEGORY PROBABILITIES

1	2	3	4	5	6	7
.30E-07	.62E-05	.59E-06	.25E-10	.54E-07	.53E-06	.28E-04

COMPONENT UNAVAILABILITIES IN GROUPS OF TEN

.10E-03	.40E-05	.10E+02	.30E-03	.90E-03	-.0	-.0	-.0	-.0	-.0
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
.10E-01	.83E-03	.33E-02	.61E-02	.88E-02	.99E-04	-.0	-.0	-.0	-.0
-.0	.10E-06	.36E-07	.10E-01	.17E-04	.12E-03	.10E-02	.10E-04	.26E-03	-.0
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
.10E-03	.10E-03	.36E-05	.10E-03	-.0	-.0	.10E-02	.10E-01	.20E-03	.40E-04
.12E-05	.12E-05	.10E-03	.10E-03	.10E-03	.30E-03	.10E-03	.10E-03	.10E-03	.20E-01
.30E-02	.10E-02	.10E-02	.10E-03	.10E-03	.10E-04	.10E-03	.40E-02	.10E-03	.10E-03
.10E-02	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
.10E-02	.24E-03	.10E-02	.10E-02	.10E-02	.20E-03	.24E-01	.72E-02	.72E-03	.10E-02
.10E-02	.72E-03	.24E-01	.10E-02	-.0	-.0	-.0	-.0	-.0	-.0
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
.30E-02	-.0	.30E-04	.30E-04	.10E-01	.10E-02	.10E-02	.90E-03	.30E-03	.10E-02
.10E-03	.30E-03	.10E-02	.10E-03	.30E-03	.10E-04	.30E-02	.30E-02	.10E-02	.30E-02
.30E-02	.10E-02	.10E-02	.10E-02	.10E-02	.10E-04	-.0	-.0	-.0	-.0
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
.29E+00	.50E+00	.34E-03	.11E-02	.41E-04	.37E-01	-.0	-.0	-.0	-.0
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
.37E-02	.97E-03	.46E-02	.10E-02	.52E-02	.18E-02	.45E-04	.60E-02	.50E-03	.26E-03
.18E-01	.10E-02	.10E-03	.32E-01	.23E-01	.18E-01	.39E-01	.16E-02	.10E-02	.18E-02
.79E-02	.58E-02	.61E-02	.41E-02	.17E-03	.43E-04	.57E-01	.21E-02	.65E-02	.59E-02
-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0	-.0
.19E-02	.50E-03	.13E-02	-.0	-.0	-.0	-.0	-.0	-.0	-.0

CONTAINMENT FAILURE PROBABILITIES

ALPHA	ALPHA1	BETA	GAMMA	DELTA	DELTA1	EPSILN	EPSLNT	EPSDHF
.010	.005	.002	.240	.995	.560	.990	.190	.800

INITIATORS

A	S1	S2	R	V	T
.10E-03	.30E-03	.90E-03	-.0	.40E-05	.10E+02

SYSTEMS

B	BP	C	D	D1	D2	F
-.0	.50E+00	.24E-02	.41E-02	.69E-02	.64E-02	.25E-03
G	H	HS	K	L	M	Q
.62E-04	.84E-02	.79E-02	.34E-04	.40E-04	.14E-01	.10E-01

THE SENSITIVITY FACTORS ARE: 3.0000 10.0000 30.0000 100.0000

TABLE B-5

OUTPUT FOR COMP ROUTINE

THE TOP 40 MOST SENSITIVE COMPONENTS ARE:

THE SENSITIVITY PARAMETER IS CORE MELT SENSITIVITY

SENSITIVITY		COMPONENT #	DESIGNATION	COMMENTS
1	1.0787	143	HUMAN ERROR	VALVE N O (L)
2	1.0741	31	SUBSYSTEM	MAIN FW SHUTDOWN
3	1.0739	182	ELECTRIC POWER	LOOS AC 1 HR/3 HR (M)
4	1.0721	161	HUMAN ERROR	MOV N O (D) HPRS
5	1.0721	160	HUMAN ERROR	
6	1.0609	214	CONTROL	JUMP LINE CONTROL
7	1.0360	80	VALVES	VALVE F O HPIS
8	1.0347	227	TEST N MAINT.	
9	1.0288	181	ELECTRIC POWER	LOOS AC 1 HR (M)
10	1.0237	215	CONTROL	
11	1.0229	147	HARDWARE	JUCTION DAMPER
12	1.0229	216	CONTROL	MOV CONTROL (H) HPIS
13	1.0214	148	HUMAN ERROR	
14	1.0204	68	VALVES	RELIEF SAFETY V F O
15	1.0175	165	HUMAN ERROR	

TABLE B-6
OUTPUT FOR COMP ROUTINE

SENSITIVITY PROBABILITY	RATIO	COMPONENT #	DESIGNATION	COMMENTS
3/1	1.0787	143	HUMAN ERROR	VALVE N O (L)
10/3	1.0283			
30/10	1.0082			
100/30	1.0023			
3/1	1.0743	31	SUBSYSTEM	MAIN FW SHUTDOWN
10/3	1.0267			
30/10	1.0077			
100/30	1.0027			
3/1	1.0739	182	ELECTRIC POWER	LOOS AC 1 HR/3 HR (M)
10/3	1.0721			
30/10	1.0076			
100/30	1.0027			
3/1	1.0721	161	HUMAN ERROR	MOV N O (D) HPRS
10/3	1.0259			
30/10	1.0075			
100/30	1.0026			
3/1	1.0721	160	HUMAN ERROR	
10/3	1.0259			
30/10	1.0075			
100/30	1.0026			

TABLE B-7
OUTPUT FOR COMBIN ROUTINE

FAILURE PROBABILITY BY RELEASE CATEGORY PROBABILITY AND OVERALL SENSITIVITY

****

THE SENSITIVIETY PARAMETER IS CORE MELT SENSITIVITY

CORE MELT SENSITIVITY IS 2.26181

EDETH = 1.004 LDETH = 1.047 CSTS = 1.315

NEW VALVES	0.460E-04	0.588E-03	0.280E-01
BASE VALVES	0.458E-04	0.563E-03	0.213E-01

1	2	3	4	5	6	7
0.39E-07	0.62E-05	0.30E-05	0.38E-05	0.16E-06	0.57E-06	0.56E-04
0.39E-07	0.62E-05	0.26E-05	0.13E-10	0.67E-07	0.54E-06	0.20E-04
0.10+01	0.10E+00	0.84E+00	0.33E+00	0.41E+00	0.95E+00	0.35E+00

VERIBY = 0.01 COMPONENT # 6 5 0 0 0

SYSTEMS H AND D

TABLE B-8

OUTPUT FOR ATTR ROUTINE

FAILURE PROBABILITY BY RELEASE CATEGORY PROBABILITY AND OVERALL SENSITIVITY

****

SYSTEMS H AND D

THE SENSITIVITY PARAMETER IS   CORE MET SENSITIVITY

THE SENSITIVITY IS   2.26181

CORE MELT SENSITIVITY IS 2.26181

RELEASE CATEGORY PROBABILITIES

1	2	3	4	5	6	7
0.39E-07	0.62E-05	0.30E-05	0.38E-10	0.16E-06	0.57E-06	0.56E-04
0.39E-07	0.62E-05	0.26E-05	0.13E-10	0.67E-07	0.54E-06	0.20E-04
0.10E+10	0.10E+01	0.84E+00	0.33E+00	0.41E+00	0.95E+00	0.35E+00

VERIBY = 0.01   COMPONENT #'s   6 5 0 0 0

****

Table B-9

INPUT AND PROGRAM FLOW	ROUTINES	SUBROUTINES	SYSTEM SUBROUTINES	OUTPUT ROUTINES
MAIN				DEBUGO
	COMP			OUTTOP
	COMBIN			OUTPUT
	ATTR			OUTTTT OUTATT
		SORT		
		FACTOR		
		SYSTEM		
		RLESE 7		
		RISK		
			LFAILP	
			MBFALP	
			QFAILP	
			CFAILP	
			DFAILP	
			HFAILP	
			FFAILP	
			GFAILP	
			BFAILP	
			CFAIL	
			QUVFAIL	
			WFAIL	

TABLE B-10

```
C MAIN PROGRAM
C PARAMETERS NECESSARY FOR EVENT TREE / FAULT TREE CALCULATIONS
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,RICOLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      COMMON/BBGG/PRNTCM(30),PARAM
      COMMON/OUTP/AA(50),HCOM(50,80)
      COMMON/OUTT/DESIG(250,30),COM(250,50)
      COMMON/RSKT/EDETHB,LDETHB,CSTSB,        EDETH,LDETH,CSTS
      COMMON/SENS/SENSY(250,7)
      REAL LDETHB,LDETHS,LDETH
      INTEGER MCASE(50,50),AA,PCHNG
      DIMENSION RLEASE(7),PFACT(250)
      DIMENSION COMJOB(10,80),NATTR(10),AATTR(10,250)
      REAL K,L,M
      READ (5, 21) ((COMJOB(I, J), I=1,10), J=1,80)
   21 FORMAT(10(80A1/))
      WRITE (6, 22) ((COMJOB(I, J), I=1,10), J=1,80)
   22 FORMAT (1H1, 19X, "SENSITIVITY STUDY",  / , 27X, "BASE CASE", / ,
     A 10(10X,80A1/))
C PROGRAM HEADING AND JOB SPECIFIC COMPONENTS
      READ (5, 23) REACTR
   23 FORMAT (F3.2)
C READ IN REACTOR TYPE  1.0 PWR, 2.0 BWR
      READ(5,1010) (PRNTCM(I),I=1,30)
 1010 FORMAT (30A1)
      READ(5,30) PARAM
   30 FORMAT(I2)
      READ(5,99) (VERIBY(I),I=1,4)
   99 FORMAT(4(F8.4))
      IF (REACTR .EQ. 2.0) GOTO 24
      READ(5,1) (PCMPNT(I),I=1,250),ALPHA,ALPHA1,BETA,GAMMA,DELTA,
     A DELTAT,EPSILN,EPSLNT,EPSBHF
    1 FORMAT(25(10(E7.1,1X)/),9(2X,F6.4))
      GOTO 25
   24 READ(5,201) (CMPNT(I),I=1,130), ALPHAB, GAMMAB, GAMAPB
  201 FORMAT(13(10(E7.1,1X)/), 3(2X,F6.4))
   25 CONTINUE
      CALL SYSTEM
      CALL RLESE7(RLEASE)
      DO 102 I = 1, 7
      BASE(I) = RLEASE(I)
  102 CONTINUE
      EDETHB=(RLEASE(1)-RICOLD)*3.+RICOLD*91.+RLEASE(2)*7.+RLEASE(3)*0.4
      LDETHB=(RLEASE(1)-RICOLD)*114.+RICOLD*120.+RLEASE(2)*67.
     A +RLEASE(3)*55.+RLEASE(4)*18.+RLEASE(5)*6.+RLEASE(6)
      CSTSB=(RLEASE(1)-RICOLD)*2270.+RICOLD*2050.+RLEASE(2)*2440.
     A +RLEASE(3)*987.+RLEASE(4)*335.+RLEASE(5)*201.+RLEASE(6)*173.
     B +RLEASE(7)*171.
C CALCULATE AND STORE BASE CASE
```

TABLE B-10

```
      CALL DEBUGO
C PRINT BASE RELEASE CATEGORIES AND BASE CASE VARIABLES
      READ(5,2) NCASE
      IF(NCASE.NE.0) GO TO 10
C MODE OF OPERATION -  COMP OR COMBIN ROUTINE
      READ (5, 2) N
      IF (REACTR .NE. 2.0) GOTO 7
      DO 100 I=1,250
      IF(PCMPNT(I).EQ.0.) GO TO 100
      READ(5,4) (DESIG(I,J),J=1,30),(COM(I,J),J=1,50)
  100 CONTINUE
    4 FORMAT (30A1, 50A1)
      GO TO 101
    7 DO 101 I = 1,  130
      IF (CMPNT(I) .EQ. 0) GOTO 101
      READ (5, 4) (DESIG(I,J), J=1,30), (COM(I,J), J=1,50)
  101 CONTINUE
C READ DATA FOR COMP ROUTINE
      CALL COMP
C CALCULATE SENSITIVITIES TO INDIVIDUAL CHANGES
      READ (5, 2) NCASE
      IF (NCASE .NE. 0) GOTO 10
      GOTO 6
C MODE OF OPERATION -  COMBIN OR ATTR
   10 DO 12 II = 1, NCASE
   12 READ (5, 3) (HCOM(II,J), J=1,80), (MCASE(II,JJ), JJ=1,50)
    3 FORMAT (80A1, /, 3(20(I3, 1X), /))
C READ IN DATA TO COMBIN ROUTINE
      CALL COMBIN(MCASE,HCOM)
C CALCULATE SENSITIVITIES TO COMBINED CHANGES
    6 READ (5, 2) NCASE
      IF (NCASE .NE. 0) GOTO 9
      GOTO 5
C MODE OF OPERATION -  ATTR OR STOP
    9 DO 104 II = 1, NCASE
      READ (5,  3) (HCOM(II,J), J=1,80)
      READ (5, 2) NATTR(II)
      READ(5,26) (AATTR(II,I),I=1,250)
  104 CONTINUE
C READ IN DATA TO ATTR ROUTINE
      CALL ATTR(AATTR, NATTR, HCOM, NCASE)
   26 FORMAT (6(40I2, /), 10I2)
    2 FORMAT(I3)
    5 CONTINUE
      STOP
      END.


      SUBROUTINE COMP
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
```

TABLE B-10

```
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      COMMON/SENS/SENSY(250,7)
      DIMENSION            ASENSY(250),INDEX(250),RLEASE(7),SENSUM(7)
      SNSITV=0.0
      DO 1 I=30,250
      IF(PCMPNT(I).EQ.0.) GO TO 1
      DO 2 LL=1,4
      VERI=VERIBY(LL)
      PCMPNT(I)=PCMPNT(I)/VERI
      CALL SYSTEM
      CALL RLESE7(RLEASE)
      CALL RISK(RLEASE,BASE,CM,SENSUM,PARAM,RISC,REACTR)
      SENSY(I,LL)=RISC
      PCMPNT(I)=PCMPNT(I)*VERI
    2 CONTINUE
      SENSY(I,5)=SENSY(I,2)/SENSY(I,1)
      SENSY(I,6)=SENSY(I,3)/SENSY(I,2)
      SENSY(I,7)=SENSY(I,4)/SENSY(I,3)
    1 CONTINUE
      DO 4 K=1,4
      DO 3 I=1,250
      ASENSY(I)=SENSY(I,LL)
    3 CONTINUE
      CALL SORT(ASENSY,N,INDEX)
      CALL OUTTOP(INDEX,LL,SENSY,N,VERIBY)
    4 CONTINUE
      CALL OUTTOP(INDEX,LL,SENSY,N,VERIBY)
      RETURN
      END


      SUBROUTINE COMBIN(MCASE,HCOM)
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      DIMENSION PFACT(250),MCASE(50,50),HCOM(50,50)
      INTEGER AA(50)
      DO 5 III=1,NCASE
      DO 11 I=1,50
      AA(I)=MCASE(III,I)
      IF(AA(I).EQ.0) GO TO 12
   11 CONTINUE
   12 CONTINUE
      DO 9 LL=1,4
      VERI=VERIBY(LL)
      IF(REACTR.EQ.2.0) GO TO 6
      DO 8 I=1,250
```

TABLE B-10

```
  8 PCMPNT(I)=PFACT(I)
    CALL FACTOR(PCMPNT,AA,PCHNG,VERI)
    GO TO 14
  6 DO 13 I=1,130
 13 CMPNT(I)=PFACT(I)
    CALL FACTOR(CMPNT,AA,PCHNG,VERI)
 14 CALL SYSTEM
    CALL RLESE7(RLEASE)
    CALL OUTPUT(RLEASE,BASE,III,VERI)
  9 CONTINUE
    DO 900 I=1,30
    PCHNG(I)=0
900 CONTINUE
  5 CONTINUE
    RETURN
    END


    SUBROUTINE ATTR(AATTR, NATTR, HCOM, NCASE)
    COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
   A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
   Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
   XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
    COMMON/SENS/SENSY(250,7)
    DIMENSION AASEN(50), INDEX(250)
    DIMENSION AATTR(10,250), NATTR(10), HCOM(10,80)
    INTEGER AA(50)
    REAL K,L,M
C
    DO 1 II= 1, NCASE
    NAT=NATTR(II)
    DO 2 J=1,NAT
    KJ = 1
    DO 3 IJ = 1, 250
    IF (AATTR(II,IJ).NE. I) GOTO 3
    AA(KJ) = IJ
    KJ = KJ + 1
  3 CONTINUE
    DO 4 LL = 1, 4
    VERI = VERIBY(LL)
    CALL FACTOR(PCMPNT, AA, PCHNG, VERI)
    CALL SYSTEM
    CALL RLESE7(RLEASE)
    CALL OUTATT(RLEASE, BASE, VERI, AA, HCOM, II)
    DO 5 JI = 1, 50
    IF (AA(JI) .EQ. 0) GOTO 6
    LJK = AA(JI)
    AASEN(JI) = SENSY(LL, LJK)
  5 CONTINUE
  6 CONTINUE
```

```
      JI = JI - 1
      CALL SORT(AASEN, JI, INDEX)
      CALL OUTTTT(AA, AASEN, INDEX)
    4 CONTINUE
    2 CONTINUE
    1 CONTINUE
      RETURN
      END


      SUBROUTINE DEBUGO
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      COMMON/RSKT/EDETHB,LDETHB,CSTSB,        EDETH,LDETH,CSTS
      REAL LDETHB,LDETHS,LDETH
      REAL K,L,M
      WRITE(6,1) (BASE(I),I=1,7)
      IF(REACTR.EQ.2.0) GO TO 10
      WRITE(6,2) (PCMPNT(I),I=1,250)
      WRITE(6,3) ALPHA,ALPHA1,BETA,GAMMA,DELTA,DELTAT,EPSILN,EPSLNT,
     AEPSBHF
      WRITE(6,5) A,S1,S2,R,V,T,B,BP,C,D,D1,D2,F,G,H,HS,K,L,M,Q
      WRITE(6,7) (VERIBY(I),I=1,4)
      GO TO 11
   10 WRITE(6,2) (CMPNT(I), I=1,130)
      WRITE(6,4) ALPHAB,GAMMAB,GAMAPB
      WRITE(6,6) T,C,Q,U,V,W
      WRITE(6,7) (VERIBY(I),I=1,4)
   11 CONTINUE
    1 FORMAT(20X,"RELEASE CATEGORY PROBABILITIES"//21X,"1",13X,"2",13X,
     A"3",13X,"4",13X,"5",13X,"6",13X,"7"//15X,7(3X,E8.2,3X)//)
    2 FORMAT(20X,"COMPONENT UNAVAILABILITIES IN GROUPS OF TEN"/
     A 25(20X,10(E8.2,2X)/)//)
    3 FORMAT(20X,"CONTAINMENT FAILURE PROBABILITIES"//22X," ALPHA",4X,
     1 "ALPHA1",4X," BETA ",4X," GAMMA",4X," DELTA",4X,"DELTAT",4X,
     2 "EPSILN",4X,"EPSLNT",4X,"EPSBHF"/22X,9(F5.3,5X))
    4 FORMAT(20X,"CONTAINMENT FAILURE PROBABILITIES"//
     A 30X,"ALPHAB",4X,"GAMMAB",4X,"GAMAPB"/30X,3(F5.3,5X))
    5 FORMAT(10X,"INITIATORS          A          S1          S2",
     1 "        R          V          T"/ 24X,6(E8.2,2X)//
     2 13X,"SYSTEMS",9X,"B",8X,"BP",9X,"C",9X,"D",8X,"D1",8X,"D2",9X,"F"
     3 / 24X,7(E8.2,2X)/ 29X,"G",9X,"H",8X,"HS",9X,"K",9X,"L",9X,"M",
     4 9X,"Q"/ 24X,7(E8.2,2X)//)
    6 FORMAT(20X,"INITIATOR T=",E8.2/ 23X,"SYSTEMS    C=",E8.2,3X,"Q=",
     1 E8.2,3X,"U=",E8.2,3X,"V=",E8.2,3X,"W=",E8.2)
    7 FORMAT(20X,"THE SENSITIVITY FACTORS ARE - ",4(F8.4,2X))
      RETURN
      END
```

TABLE B-10

```
      SUBROUTINE RLESE7(RLEASE)
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      COMMON/RSKT/EDETHB,LDETHB,CSTSB,        EDETH,LDETH,CSTS
      REAL LDETHB,LDETHS,LDETH
      REAL K,L,M
      DIMENSION RLEASE(7)
      IF (REACTR.EQ.2.0) GO TO 2
      A=PCMPNT(1)
      S1=PCMPNT(4)
      S2=PCMPNT(5)
      T=PCMPNT(3)
      V=PCMPNT(2)
      B=PCMPNT(39)
      R=PCMPNT(6)
      DO 7 J=1,7
      DO 8 I=1,6
      ESCAPE(I,J)=0.0
    8 CONTINUE
    7 CONTINUE
      R1COLD=ALPHA*((A+S1+S2)*(F+G))+ALPHA1*S2*C
      ESCAPE(1,1) =ALPHA*A *(B+C*D)+ALPHA1*A *(F+G)
      ESCAPE(2,1) =ALPHA*S1*(B+C*D1)+ALPHA1* S1*(F+G)
      ESCAPE(3,1) =ALPHA*S2*(B+C*D2)+ALPHA1* S2*(F+G)
      ESCAPE(4,1)=ALPHA1*R*C
      ESCAPE(6,1)=ALPHA*T*M*L*BP
      ESCAPE(1,2)=A *((B+H*F)*GAMMA+B*DELTA)
      ESCAPE(2,2)=S1*((B+H*F)*GAMMA+B*DELTA)
      ESCAPE(3,2)=S2*((B+H*F)*GAMMA+B*DELTA)
      ESCAPE(4,2)=R*((C+F)*DELTA+C*GAMMA)
      ESCAPE(5,2)=V
      ESCAPE(6,2)=T*M*L*BP*(GAMMA+DELTAT)
      ESCAPE(1,3)=A*((D+H)*ALPHA1+(F+G)*DELTA)
      ESCAPE(2,3)=S1*((D+HS)*ALPHA1+(F+G)*DELTA)
      ESCAPE(3,3)=S2*((D+HS)*ALPHA1+(F+G)*DELTA)
      ESCAPE(4,3)=R*ALPHA1
      ESCAPE(6,3)=T*ALPHA*(M*L+K*Q+K*M*Q)
      ESCAPE(1,4)=A*C*D*BETA
      ESCAPE(2,4)=S1*C*D*BETA
      ESCAPE(3,4)=S2*C*D*BETA
      ESCAPE(1,5)=A*BETA*(D+H)
      ESCAPE(2,5)=S1*BETA*(D+HS)
      ESCAPE(3,5)=S2*BETA*(D+HS)
      ESCAPE(6,5)=T*BETA*(M*L+K*Q)
      ESCAPE(1,6)=A*(EPSILN*D*F+EPSBHF*(B+H*F))
      ESCAPE(2,6)=S1*(EPSILN*D*F+EPSBHF*(B+HS*F))
      ESCAPE(3,6)=S2*(EPSILN*D*F+EPSBHF*(B+HS*F))
      ESCAPE(6,6)=T*M*L*BP*EPSLNT
      ESCAPE(1,7)=A*EPSILN*(D+H)
      ESCAPE(2,7)=S1*EPSILN*(D+HS)
      ESCAPE(3,7)=S2*EPSILN*(D+HS`
      ESCAPE(4,7)=R*EPSILN
      ESCAPE(6,7)=T*EPSLNT*(M*L+K*Q+K*Q*M)
      DO 4 J=1,7
       ESCSUM=0.0
      DO 5 I=1,6
      ESCSUM=ESCAPE(I,J)+ESCSUM
    5 CONTINUE
      RLEASE(J)=ESCSUM
    4 CONTINUE
      GO TO 1
    2 T=CMPNT(16)
      RLEASE(1)=ALPHAB*T*(W+C+QUV)
      RLEASE(2)=GAMAPB*T*(W+QUV)
      RLEASE(3)=GAMMAB*T*(W+C+QUV)
1     CONTINUE
      RETURN
      END
```

TABLE B-10

```
      SUBROUTINE OUTTOP(INDEX,L,SENSY,N,VERIBY)
      COMMON/OUTT/DESIG(250,30),COM(250,50)
      COMMON/BBGG/PRNTCM(30),PARAM
      DIMENSION SENSY(250,7),INDEX(250)
      DIMENSION VERIBY(4)
      IF(L.EQ.5) GO TO 80
      WRITE(6,1) N
      WRITE(6,1010) (PRNTCM(J),J=1,30)
 1010 FORMAT(20X,"THE SENSITIVITY PARAMETER IS ",30A1/)
      WRITE(6,2)
      DO 10 I=1,N
      J=INDEX(I)
   10 WRITE(6,3) I,SENSY(J,L),J,(DESIG(J,K),K=1,15),(COM(J,K),K=1,30)
      GO TO 70
    1  FORMAT(1H1,19X,"THE TOP ",I2,"MOST SENSITIVE COMPONENTS ARE - "/)
    2 FORMAT(10X,"SENSITIVITY",10X,"COMPONENT #",15X,"DESIGNATION",25X,"
     1COMMENTS"//)
    3 FORMAT(9X,I2,2X,F7.4,15X,I3,12X,30A1,8X,50A1//)
    4 FORMAT(20X,54(1H*)//20X,"L= ",I2//)
    6 FORMAT(8X,I3,1H/,I2,3X,F7.4,15X,I3,12X,15A1,8X,30A1)
    7 FORMAT(8X,I3,1H/,I2,3X,F7.4)
    8 FORMAT(7X,"PROBABILITY RATIO"/)
   80 CONTINUE
      WRITE(6,2)
      WRITE(6,8)
      DO 40 I=1,N
      MM=2
      J=INDEX(I)
      IGO=1
      ID=VERIBY(1)
      WRITE(6,9)
      WRITE(6,6) ID,IGO,SENSY(J,1),J,(DESIG(J,K),K=1,15),
     A  (COM(J,K),K=1,30)
      DO 20 M=5,7
      IDM=IGO
      IGO=ID
      ID=VERIBY(MM)
      MM=MM+1
      WRITE(6,7) ID,IGO,SENSY(J,M)
   20 CONTINUE
   40 CONTINUE
   70 CONTINUE
    9 FORMAT(25X,"    ")
      RETURN
      END

      SUBROUTINE SORT(ASENSY,N,INDEX)
      DIMENSION ASENSY(250),INDEX(250)
      INTEGER TOP,BOT
      DO 1 I=1,250
    1 INDEX(I)=I
      DO 20 I=1,N
           TOP=INDEX(I)
           L=I+1
           DO 10 K=L,250
                BOT=INDEX(K)
               IF(ASENSY(BOT).LE.ASENSY(TOP)) GO TO 10
               INDEX(K)=TOP
                TOP=BOT
   10      CONTINUE
           INDEX(I)=TOP
   20 CONTINUE
      RETURN
      END
```

TABLE 8-10

```
      SUBROUTINE OUTPUT(RLEASE,BASE,III,VERIBY)
      COMMON/OUTP/AA(50),HCOM(50,80)
      COMMON/BBGG/PRNTCM(30),PARAM
      COMMON/RSKT/EDETHB,LDETHB,CSTSB,        EDETH,LDETH,CSTS
      REAL LDETHB,LDETHS,LDETH
      DIMENSION RLEASE(7),BASE(7),SENSUM(7)
      CALL RISK(RLEASE,BASE,CM,SENSUM,PARAM,RISC,REACTR)
      WRITE(6,1)
 1010 FORMAT(20X,"THE SENSITIVITY PARAMETER IS ",30A1/
     A 20X," THE SENSITIVITY IS ",E8.2)
      WRITE(6,1010) (PRNTCM(J),J=1,30), RISC
      WRITE(6,6) CM
      WRITE(6,5)
      WRITE(6,4) (RLEASE(I),I=1,7),(BASE(I),I=1,7),(SENSUM(I),I=1,7)
      WRITE(6,2) VERIBY,(AA(I),I=1,50)
      WRITE(6,3) (HCOM(III,J),J=1,80)
    1 FORMAT(1H1,19X,63HFAILURE PROBABILITY BY RELEASE CATEGORY AND OVER
     1ALL SENSITIVITY//20X,63(1H*)///)
    2 FORMAT(30X,7HVERIBY=,F8.4,5X,13HCOMPONENT #"S/
     A 40X,2(25(1X,I3)/))
    3 FORMAT(1H0,24X,80A1)
    4 FORMAT(10X," NEW ",7(2X,E9.3,3X)//10X," BASE",7(2X,E9.3,3X)//
     A 10X,"RATIO",7(2X,E9.3,3X)//)
    5 FORMAT (50X,"RELEASE CATEGORY PROBABILITIES",
     A        21X,"1",13X,"2",13X,"3",13X,"4",13X,"5",13X,"6",13X,"7"//)
    6 FORMAT(20X,"CORE MELT SENSITIVITY IS ",F9.5)
    7 FORMAT(20X/15X,98(1H*))
      RETURN
      END


      SUBROUTINE OUTTTT(AA,AASEN,INDEX)
      COMMON/RSKT/EDETHB,LDETHB,CSTSB,        EDETH,LDETH,CSTS
      COMMON/OUTT/DESIG(250,30),COM(250,50)
      REAL LDETHB,LDETHS,LDETH
      DIMENSION AASEN(50), INDEX(250)
      INTEGER AA(50)
      WRITE(6,1)
      WRITE(6,2)
      DO 10 I=1,JI
      J=INDEX(I)
   10 WRITE(6,3) I,AASEN(J),AA(J),(DESIG(J,K),K=1,30),(COM(J,K),K=1,50)
      WRITE(6,7)
    1 FORMAT(20X,"BREAKDOWN BY MOST SENSITIVE COMPONENTS")
    2 FORMAT(10X,"SENSITIVITY",10X,"COMPONENT #",15X,"DESIGNATION",25X,"
     1COMMENTS"//)
    3 FORMAT(9X,I2,2X,F7.4,15X,I3,12X,30A1,8X,50A1//)
    7 FORMAT(20X/15X,98(1H*))
      RETURN
      END
      SUBROUTINE FACTOR(CMPNT,A,CHNG,VERIBY)
      INTEGER A(50)
      DIMENSION CMPNT(250),CHNG(30)
      DO 3 I=1,50
      IF(A(I).EQ.0) GO TO 4
      IF(A(I).LT.30) GO TO 1
      IF(MOD(A(I),10).NE.0) GO TO 2
      JJ=A(I)+1
      JJJ=A(I)+10
      DO 10 J=JJ,JJJ
   10 CMPNT(J)=CMPNT(J)/VERIBY
      GO TO 3
    1 CHNG(A(I))=1
      GO TO 3
    2 CMPNT(A(I))=CMPNT(A(I))/VERIBY
    3 CONTINUE
    4 CONTINUE
      RETURN
      END
```

TABLE B-10

```
      SUBROUTINE OUTATT(RLEASE,BASE,VERI,AA,HCOM,II)
      COMMON/RSKT/EDETHB,LDETHB,CSTSB,        EDETH,LDETH,CSTS
      COMMON/BBCG/PRNTCM(30),PARAM
      REAL LDETHB,LDETHS,LDETH
      DIMENSION AA(50),HCOM(50,80)
      DIMENSION RLEASE(7),SENSUM(7),BASE(7)
      INTEGER AA
      CALL RISK(RLEASE,BASE,CM,SENSUM,PARAM,RISC,REACTR)
      WRITE(6,1) (HCOM(II,J),J=1,80),(PRNTCM(J),J=1,30),RISC
      WRITE(6,6) CM
      WRITE(6,5)
      WRITE(6,4) (RLEASE(I),I=1,7),(BASE(I),I=1,7),(SENSUM(I),I=1,7)
      WRITE(6,2) VERIBY,(AA(I),I=1,50)
      WRITE(6,7)
    1 FORMAT(1H1,19X,63HFAILURE PROBABILITY BY RELEASE CATEGORY AND OVER
     1ALL SENSITIVITY//20X,63(1H*)//25X,80A1//
     2        20X,"THE SENSITIVITY PARAMETER IS ",30A1/
     A 20X," THE SENSITIVITY IS ",E8.2)
    2 FORMAT(30X,7HVERIBY=,F8.4,5X,13HCOMPONENT #"S/
     A 40X,2(25(1X,I3)/))
    4 FORMAT(10X." NEW ",7(2X,E9.3,3X)//10X," BASE",7(2X,E9.3,3X)//
     A 10X,"RATIO",7(2X,E9.3,3X)//)
    5 FORMAT (50X,"RELEASE CATEGORY PROBABILITIES",
     A        21X,"1",13X,"2",13X,"3",13X,"4",13X,"5",13X,"6",13X,"7"//)
    6 FORMAT(20X,"CORE MELT SENSITIVITY IS ",F9.5)
    7 FORMAT(20X/15X,98(1H*))
      RETURN
      END


      SUBROUTINE SYSTEM
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      REAL K,L,M
      IF(REACTR.EQ.2.0) GO TO 10
    1 CALL LFAILP
    2 CALL MBPFAL
    3 CALL QFAILP
    4 CALL CFAILP
    5 CALL DFAILP
    6 CALL HFAILP
    7 CALL FFAILP
    8 CALL GFAILP
    9 CALL KFAILP
      GO TO 11
   10 CALL WFAIL
      CALL QUVFAL
      CALL CFAIL
   11 CONTINUE
      RETURN
      END

      SUBROUTINE QFAILP
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
C           BASED ON PWR OPERATING EXPERIENCE ESTIMATED
C Q=1.0*E-2  (10)
      Q=PCMPNT(68)
      IF(PCHNG(3).EQ.1) Q=Q/VERI
      RETURN
      END
```

TABLE 5-10

```
      SUBROUTINE RISK(RLEASE,BASE,CM,SENSUM,PARAM,RISC,REACTR)
      COMMON/RSKT/EDETHB,LDETHB,CSTSB,R1COLD,EDETH,LDETH,CSTS
      REAL LDETHB,LDETHS,LDETH
      DIMENSION RLEASE(7),BASE(7),SENSUM(7)
      DO 10 I=1,7
      IF(BASE(I).EQ.0.) GO TO 10
      SENSUM(I)=BASE(I)/RLEASE(I)
   10 CONTINUE
      BSE=0.
      RLSE=0.
      DO 2 I=1,7
      RLSE=RLEASE(I)+RLSE
      BSE=BASE(I)+BSE
    2 CONTINUE
      CM=BSE/RLSE
      IF(REACTR.EQ.2.0) GO TO 100
      EDETHS=(RLEASE(1)-R1COLD)*8.+R1COLD*91.+RLEASE(2)*7.+RLEASE(3)*0.4
      LDETHS=(RLEASE(1)-R1COLD)*114.+R1COLD*120.+RLEASE(2)*67.
     A +RLEASE(3)*55.+RLEASE(4)*18.+RLEASE(5)*6.+RLEASE(6)
      CSTSS=(RLEASE(1)-R1COLD)*2270.+R1COLD*2050.+RLEASE(2)*2440.
     A +RLEASE(3)*987.+RLEASE(4)*335.+RLEASE(5)*201.+RLEASE(6)*173.
     B +RLEASE(7)*171.
      EDETH=EDETHB/EDETHS
      LDETH=LDETHB/LDETHS
      CSTS=CSTSB/CSTSS
      GO TO 200
  100 EDETH=0.0
      LDETH=0.0
      CSTS=0.0
  200 CONTINUE
      IF(PARAM.EQ.0) RISC=CM
      IF(PARAM.EQ.0) GO TO 5
      IF(PARAM.GT.7) GO TO 6
      RISC=RLEASE(PARAM)
      GO TO 5
    6 IF(PARAM.EQ.8)  RISC=EDETH
      IF(PARAM.EQ.9) RISC=LDETH
      IF(PARAM.EQ.10) RISC=CSTS
    5 CONTINUE
      RETURN
      END

      SUBROUTINE WFAIL
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      REAL LPCI,HARD,HERR,HHARD,HPSW
     1HR,PCS
C       W  REMOVAL OF DECAY HEAT
C       RESIDUAL HEAT REMOVAL AND POWER CONVERSION SYSTEM
C       RHR  LPCI OR HPSW
C       LPCI  HARDWARE OR OPERATOR VALVE FAILURE OR PLUGGED VALVE
      HARD=CMPNT(82)
      HERR=CMPNT(54)
      VALV=CMPNT(95)
      LPCI=HARD+HERR+VALV
C       HPSW  COMMON MODE, T + M, HARDWARE
C       CM  OPERATOR FAILURE TO START W/IN 25 HRS
      CM=CMPNT(56)
      TNM=CMPNT(35)
C       HARDWARE  VALVE RUPTURE OR VALVE AND OPERATOR OF 2 LEGS
C       TWO LEGS  OPERATOR FAILURE W/ WALKAROUND OR MAINT. OR VALVE FAILURE
C                 OR HARDWARE FAULTS (SQUARED)
      TLEGS=(0.22*CMPNT(55)/3.0+CMPNT(96)+CMPNT(83))**2.0
      VALOP=CMPNT(97)*(CMPNT(55)+CMPNT(84))
C       SUM W/ VALVE RUPTURE
      HHARD=TLEGS+VALOP+CMPNT(98)
      HPSW=CM+TNM+HHARD
```

TABLE B-10

```
C     PWR SUBROUTINES
      SUBROUTINE LFAILP
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,O,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,RICOLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      REAL L
C L   SECONDARY STEAM RELIEF AND AUXILLIARY FEEDWATER SYSTEM
C L=3.7*10**(-5)    (3)    FOR ALL EVENTS NOT INCLUDING LOOS
C L=1.5*E-4    (3)    FOR ALL LOOS EVENTS
C           FEEDWATER    3 LOOPS   2 ELECTRIC PUMP   1 TURBINE PUMP
C       SMALL PIPE BREAK (0 TO 8 HOURS) OR TRANSIENT WITH LOSS OF MAIN FEEDWATER
C           BUT NOT LOOS
C QS=5.1*E-7    RUPTURE MAIN HEADERS     PLUGGED VENTS FROM CONDENSATE TANK
C QD=6.5*E-7    RUPTURE IN AFSW WITHIN MSVH AND FAILURE OF TURBINE LOOP
C                      FAILURE CHECK VALVES BOTH HEADERS
C QT=8.7*E-7    COMBINATIONS OF INDEPENDENT FAULTS   ALL 3 LOOPS
C QTNM=3.2*E-6  FAILURE IN TWO LOOPS WHILE THIRD IN MAINTENANCE OR TEST
C QCM=3.0*E-5   DISCHARGE VALVES ALL THREE PUMPS LEFT CLOSED FOLLOWING TESTS
C QTNMLS=1.4*E-4  3 DIESELS FAIL AND TEST OR MAINTENANCE ON TURBINE LOOP
      QS=4.*PCMPNT(42)+3.*PCMPNT(43)
      QTURB=2.*PCMPNT(141)+PCMPNT(101)+PCMPNT(61)+PCMPNT(62)
      QD=(18.*PCMPNT(63)+0.1*PCMPNT(1))*(QTURB+PCMPNT(67))+(2.*PCMPNT(64
     1))**2.0
      QA=PCMPNT(201)+PCMPNT(102)+PCMPNT(103)+2.*PCMPNT(141)+PCMPNT(61)
     A+PCMPNT(62)
      QT=QA**2.0*QTURB
      QTNM=PCMPNT(221)*(QA*QA)+PCMPNT(221)*2.*QTURB*QA+(PCMPNT(221)
     A+PCMPNT(222))*(18.*PCMPNT(63)+0.1*PCMPNT(1))
      QCM=PCMPNT(143)
C   TAKING LOOS ONTO ACCOUNT
      QCMLS=PCMPNT(44)*(QTURB+PCMPNT(67))
      QTNMLS=PCMPNT(44)*(PCMPNT(221)+PCMPNT(222))
      L=QS+QD+QT+QTNM+0.02*(QTNMLS+QCMLS)+QCM
      IF(PCHNG(1).EQ.1) L=L/VERI
      RETURN
      END
      SUBROUTINE FFAILP
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,O,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,RICOLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
C F     CONTAINMENT SPRAY RECIRCULATION SYSTEM    CSRS      RECIRCULATION OF
C                    CONTAINMENT SUMP WATER THRU HEAT EXCHANGERS OF CONTAINMENT
C                       HEAT REMOVAL SYSTEM         4 TRAINS- 3500 GPM PUMP,HX,
C                        AND SPRAY HEADER       2 PUMPS INSIDE CONTAINMENT
C                         SUCCESS -  PUMPING BY 2 OF 4 TRAINS FIRST 24 HOURS
C                              1 OF 4 AFTER THAT
C QD = 2.6*E(-6)      2 LEGS FAIL -  POWER TRAIN AND 1 OF OTHER 2 LEGS (MECHANICAL)
C QT = 2.6*E(-6)      COMPONENT FAILURES   3 OF 4 LEGS
C QTNM=4.3*E(-5)      REDUCED REDUNDANCY
C QCM= 2.8*E(-5)
      QEP=PCMPNT(185)+PCMPNT(184)+PCMPNT(219)
      QCH=PCMPNT(220)+2.*PCMPNT(90)+PCMPNT(111)+PCMPNT(112)
      QEH=PCMPNT(111)+PCMPNT(241)
      QCX=PCMPNT(162)
      QEX=PCMPNT(113)
      QCM=PCMPNT(114)
      QEM=PCMPNT(166)
      QP=QEP*(QCH+QEH+QCX+QEX)+(QEH+QEX)*(QCH*QCH+2.*QCH*QCX+QCX*QCM)+(Q
     ACH+QCX)*(QEH*QEH+2.*QEH*QEX+QEX*QEM)+(QCH+QCX)*2.*(QEH+QEX)*(PCMPN
     BT(229)+PCMPNT(230))+PCMPNT(230)*(QCH*QCH+2.*QCH*QCX+QCX*QCM)+PCMPN
     CT(229)*(QEH*QEH+2.*QEH*QEX+QEX*QEM)+2.*QEP*(PCMPNT(229)+PCMPNT(230
     D))
      F=2.*QP
      IF(PCHNG(7).EQ.1) F=F/VERI
      RETURN
      END
```

TABLE B-10

```
      SUBROUTINE CFAILP
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
C C   CONTAINMENT SPRAY INJECTION SYSTEM
C C=2.4E-3
C         DELIVERS BORATED COLD WATER THRU SPRAY HEADS TO CONTAINMENT FROM
C           REFUELING WATER STORAGE TANK(RWST) FOR 1ST 1/2 HR AFTER LARGE LOCA
C         REDUCES CONTAINMENT PRESSURE
C         FAILURE=FAILURE TO DELIVER SPRAY FLUID EQUIVALENT TO FULL DELIVERY FROM
C           1 OF 2 PUMPS
C QD=3.2E-4     INDEPENDANT SPRAY SYSTEM FAILURES
C QTNM=1.5E-4   REDUCED REDUNDANCY DUE TO T+M
C QCM=1.9E-3     COUPLED HUMAN ERRORS IN CALIBRATING CONSEQUENCE LIMITING CONTROL
C                   SYSTEM (CLCS) AND DURINGMONTHLY FLOW TEST OF CSIS SUBSYSTEMS
      QS=PCMPNT(145)+PCMPNT(146)+PCMPNT(203)+PCMPNT(204)+PCMPNT(104)
      QD=QS*QS
      QTNM=2.*PCMPNT(224)*QS
      QCM=PCMPNT(147)+PCMPNT(148)
      C=QD+QTNM+QCM
      IF(PCHNG(4).EQ.1) C=C/VERI
      RETURN
      END
      SUBROUTINE MBFFAL
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      REAL M
C M   SECONDARY STEAM RELIEF AND POWER CONVERSION SYSTEM
C         PORTIONS OF POWER CONVERSION SYSTEM THAT PROVIDE FOR MAIN FEEDWATER
C         DELIVERY TO STEAM GENERATORS
C M=10**(-1)
C B"  FAILURE TO RECOVER EITHER ON OR OFF SITE POWER WITHIN 1 TO 3 HOURS
C     FOLLOWING LOOS TRANSIENT
C B"=5*10**(-1)
      M=PCMPNT(31)+.02*PCMPNT(181)
      BP=PCMPNT(182)
      IF(PCHNG(2).EQ.1) M=M/VERI
      RETURN
      END

      SUBROUTINE KFAILP
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,R1COLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
      REAL K
C K   REACTOR PROTECTION SYSTEM     3.6*E-5    (1)
C QD=5.4E-6      SEVERAL TRIP CIRCUIT BREAKER FAULTS AND WIRE FAULTS ON EACH BRAN
C                    BRANCH OF TRIP BRAEKER SYSTEM
C QTNM=1.2E-5  RESULT FROM DECREASED REDUNDANCY DURING T+M OF BREAKER.
C QROD=1.7E-5  POSSIBILITY OF 3 OR MORE RODS INDEPENDANTLY FAIL TO ENTER CORE
      QD=(PCMPNT(202)+PCMPNT(243))**2.0
      QROD=PCMPNT(45)
      QTNM=2.*PCMPNT(223)*PCMPNT(202)
      K=QD+QROD+QTNM
      IF(PCHNG(9).EQ.1) K=K/VERI
      RETURN
      END
```

TABLE B-10

```
C     BWR SUBROUTINES
      SUBROUTINE CFAIL
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ESCAPE(6,7),REACTR,RICOLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB,GAMAPB,CMPNT(130)
C       REACTOR SHUTDOWN=C
C       REACTOR PROTECTION SYSTEM AND MANUAL RESERVE SHUTDOWN
      REAL MVC
C       RPS ROD FAILS TO INSETT, HUMAN SWITCH ERROR
      SWITCH=CMPNT(51)
      TNM=CMPNT(34)
      ROD=CMPNT(61)
      ROD2=CMPNT(65)
      RPS=300.0*ROD+2.8*ROD2+TNM+2.0*SWITCH
C       MANUAL VALVE CLOSING
      MVC=CMPNT(52)
C       AND GATE
      C=RPS*MVC
      IF(PCHNG(1).EQ.1) C=C/VERI
      RETURN
      END




      SUBROUTINE QUVFAL
      COMMON/LWR/PCMPNT(250),A,BP,C,D,F,G,H,K,L,M,Q,V,T,ALPHA,BETA,GAMMA
     A,DELTA,EPSILN,S1,S2,N,PCHNG(30),EPSLNT,EPSBHF,DELTAT,ALPHA1
     Z,B,R,D1,D2,HS,BASE(7),VERIBY(4),ZSCAPE(6,7),REACTR,RICOLD,VERI,
     XU,QUV,W,ALPHAB,GAMMAB GAMAPB,CMPNT(130)
C       LOSS OF FEEDWATER  Q
C       HPCI OF RCIC FOR MAKEUP WATER  U
C       LOW PRESSURE ECCS FOR MAKE UP WATER
      REAL LPECCS,MANADS,HTEST,HFAIL,HARD
C       Q  2 DISTINCT VALUES DEPENDANT ON OFF SITE POWER AVAILABILITY
      Q=CMPNT(13)+.02*CMPNT(121)
C       U  HPCI OR RCIC = FAIL-FAIL, FAIL-TEST, TEST-FAIL
      HTEST=CMPNT(31)
      HFAIL=3.0*CMPNT(91)+CMPNT(62)+3.0*CMPNT(93)+CMPNT(94)+2.0*CMPNT(11
     A1)+2.0*CMPNT(112)+CMPNT(81)
      RTEST=CMPNT(32)
      RFAIL=HFAIL-CMPNT(111)-CMPNT(112)
      U=RFAIL*HFAIL+RFAIL*HTEST+HFAIL*RTEST
C       V  LOW PRESSURE ECCS OR OPERATOR FAILURE TO ACTUATE ADS
C       LPECCS = TEST OR HARDWARE
      TNM=CMPNT(33)
      HARD=CMPNT(113)+CMPNT(92)
      LPECCS=TNM+HARD
C       OPERATOR FAILURE TO ACTUATE ADS
      MANADS=CMPNT(53)
      V=LPECCS+MANADS
      QUV=Q*U*V
      IF(PCHNG(2).EQ.1) QUV=QUV/VERI
      RETURN
      END
```

Appendix C

Reduced Fault Trees

This appendix documents the fault trees which are used as input to the LWRSEN computer code. These fault trees are developed from the fault trees used in the RSS. The exact trees can be found in Appendices II and V of the RSS. The exact trees are much more complex and involve a much larger number of individual inputs.

In order to make the analysis more tractable, the trees are reduced such that insignificant contributions are eliminated. The criterion for determining whether to include a cut set or not is that the cut set should not contribute more than one-tenth of one percent to the top event failure probability. In addition, since one goal of the study is to explore the sensitivity of different generic classifications, numbers of smaller components are combined to basically fit these categories where applicable. The generic categories chosen for the PWR are human error, test and maintenance, control, electric power, pumps, valves, and other hardware. The number of generic categories chosen for the BWR are fewer because the transient analysis in the RSS is less detailed. For the BWR those categories are human error, test and maintenance, pumps, valves, and all hardware. The reductions are completed for the five most important systems or functions in the BWR and the six most important systems or functions in the PWR. Other systems in the PWR are less detailed and consequently not documented by a reduced fault tree. This information is documented in Table C-1. The systems contained in the study, as well as the subsystems which are the fundamental elements of those systems, are listed and defined, along with their common abbreviations, in Appendix A. The reduced fault trees of the eleven systems considered in detail and a key are contained in Table C-2 and Figures C-1 through C-11.

Table C-1

Reduced Fault Trees and Systems Considered in This Study

BWR

Systems/Functions Considered in Some Detail

W	Remove Residual Core Heat
C	Reactor Protection System
U	HPCI or RCIC
V	Low Pressure ECCS

Systems/Functions Considered in Less Detail

Q	Normal Feedwater System

PWR

Systems/Functions Considered in Detail

L	Secondary Steam Relief and Auxiliary Feedwater System
K	Reactor Protection System
C	Containment Spray Injection System
D	Emergency Core Cooling Injection System LPIS, HPIS, ACC
H	Emergency Core Cooling Recirculation System LPRS HPRS
F	Containment Spray Recirculation System
G	Containment Heat Removal System

Systems/Functions Considered in Less Detail

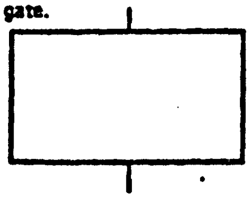
M	Secondary Steam Relief and Power Conversion System
B	Loss of Electric Power
B'	Recovery of off site power 1 - 3 hrs. following
Q	Reactor Coolant System Relief and Safety Valves Fail to Close
V	LPIS Check Valve
R	Reactor Vessel Rupture

TABLE C-2

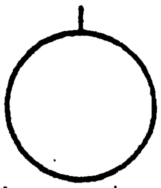
FAULT TREE SYMBOLISM

EVENT REPRESENTATIONS

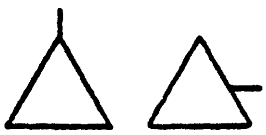
The rectangle identifies an event that results from the combination of fault events through the input logic gate.



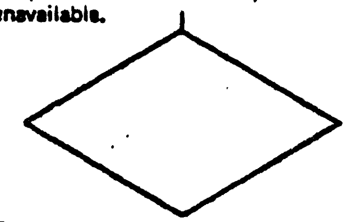
The circle describes a basic fault event that requires no further development. Frequency and mode of failure of items so identified are derived from empirical data.



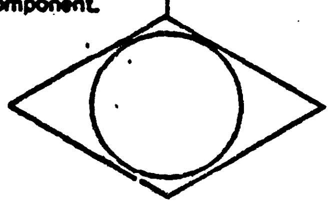
The triangles are used as transfer symbols. A line from the apex of the triangle indicates a transfer in and a line from the side or bottom denotes a transfer out.



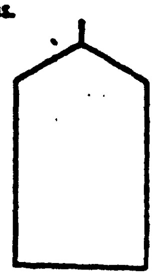
The diamond describes a fault event that is considered basic in a given fault tree. The possible causes of the event are not developed further because the event is of insufficient consequence or the necessary information is unavailable.



The circle within a diamond indicates a subtree exists, but that subtree was evaluated separately and the quantitative results inserted as though a component.

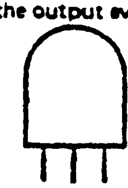


The house is used as a switch to include or eliminate parts of the fault tree as those parts may or may not apply to certain situations.

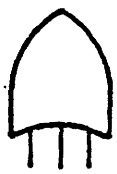


LOGIC OPERATIONS

AND gate describes the logical operation whereby the coexistance of all input events is required to produce the output event.



OR gate defines the situation whereby the output event will exist if one or more of the input events exists.



or

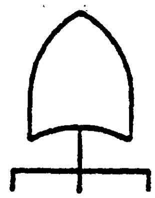
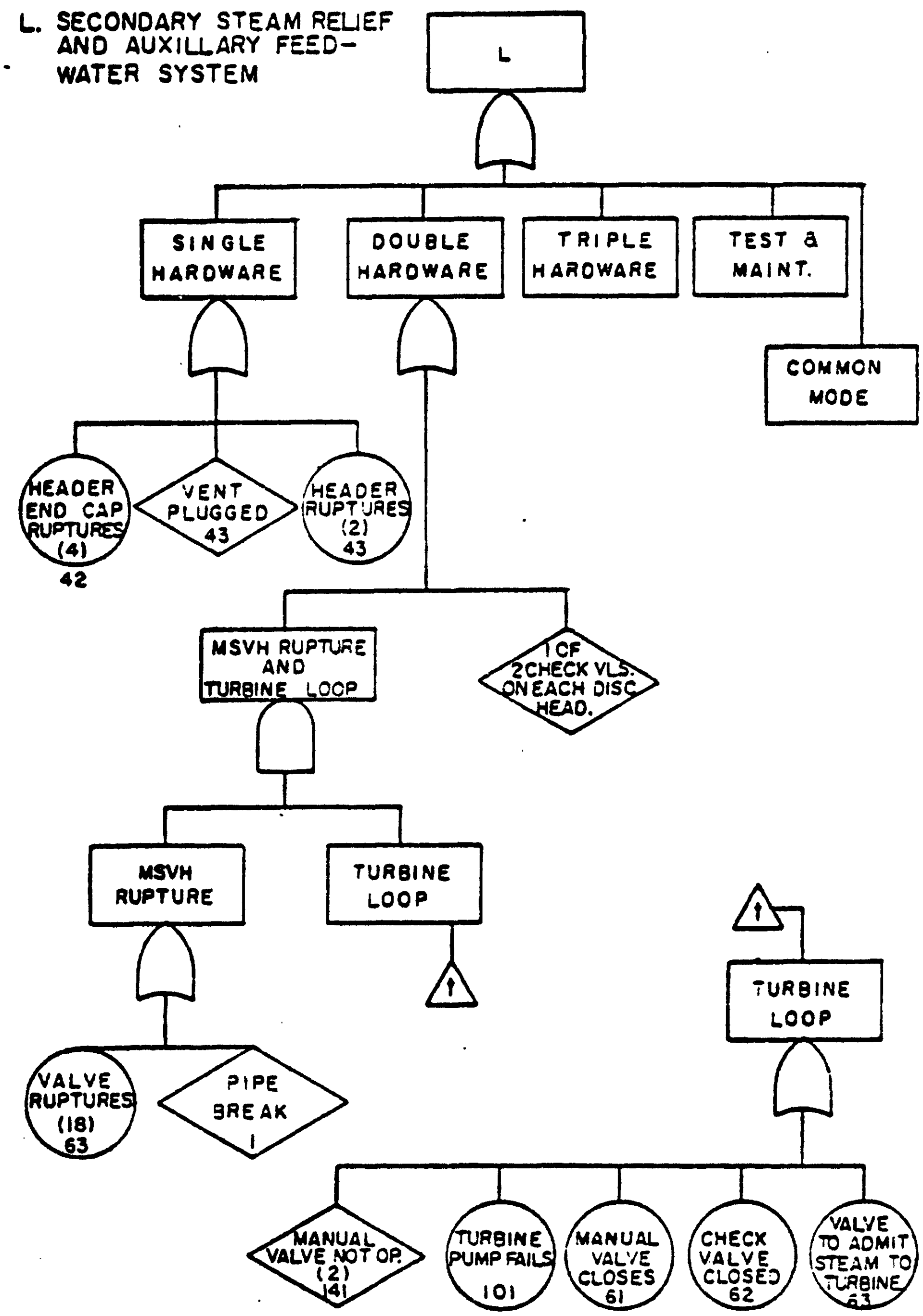
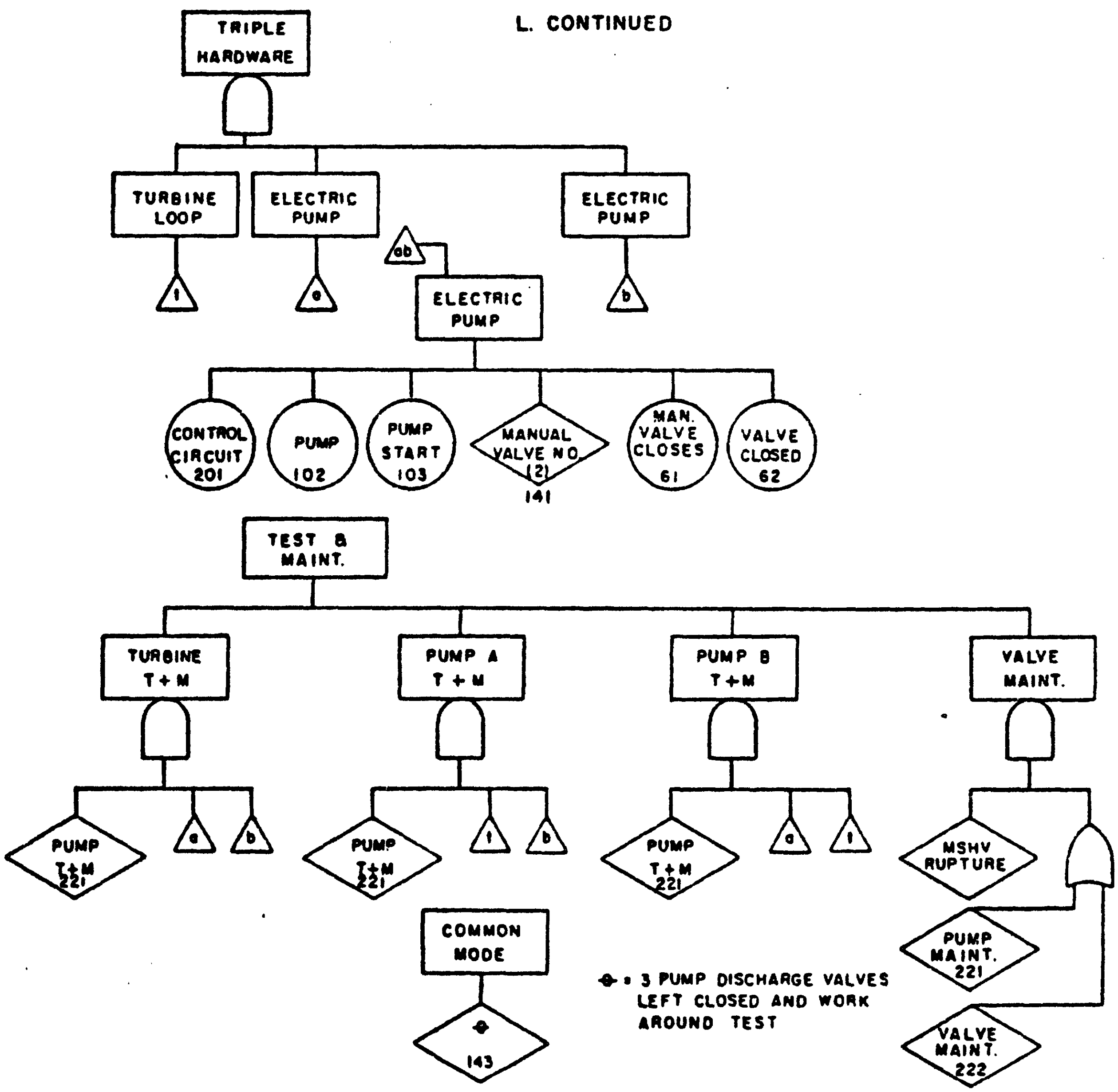


FIGURE C-1



L. SECONDARY STEAM RELIEF AND AUXILLARY FEED-WATER SYSTEM
L
SINGLE HARDWARE
DOUBLE HARDWARE
TRIPLE HARDWARE
TEST & MAINT.
COMMON MODE
HEADER END CAP RUPTURES (4) 42
VENT PLUGGED 43
HEADER RUPTURES (2) 43
MSVH RUPTURE AND TURBINE LOOP
1 OF 2 CHECK VLS. ON EACH DISC HEAD.
MSVH RUPTURE
TURBINE LOOP
VALVE RUPTURES (18) 63
PIPE BREAK 1
TURBINE LOOP
MANUAL VALVE NOT OP. (2) 141
TURBINE PUMP FAILS 101
MANUAL VALVE CLOSES 61
CHECK VALVE CLOSED 62
VALVE TO ADMIT STEAM TO TURBINE 63


TRIPLE HARDWARE
L. CONTINUED
TURBINE LOOP
ELECTRIC PUMP
ELECTRIC PUMP
ab
ELECTRIC PUMP
CONTROL CIRCUIT 201
PUMP 102
PUMP START 103
MANUAL VALVE NO. (2) 141
MAN. VALVE CLOSES 61
VALVE CLOSED 62
TEST & MAINT.
TURBINE T+M
PUMP A T+M
PUMP B T+M
VALVE MAINT.
PUMP T+M 221
PUMP T+M 221
PUMP T+M 221
MSHV RUPTURE
PUMP MAINT. 221
VALVE MAINT. 222
COMMON MODE
143
= 3 PUMP DISCHARGE VALVES LEFT CLOSED AND WORK AROUND TEST

L. CONTINUED

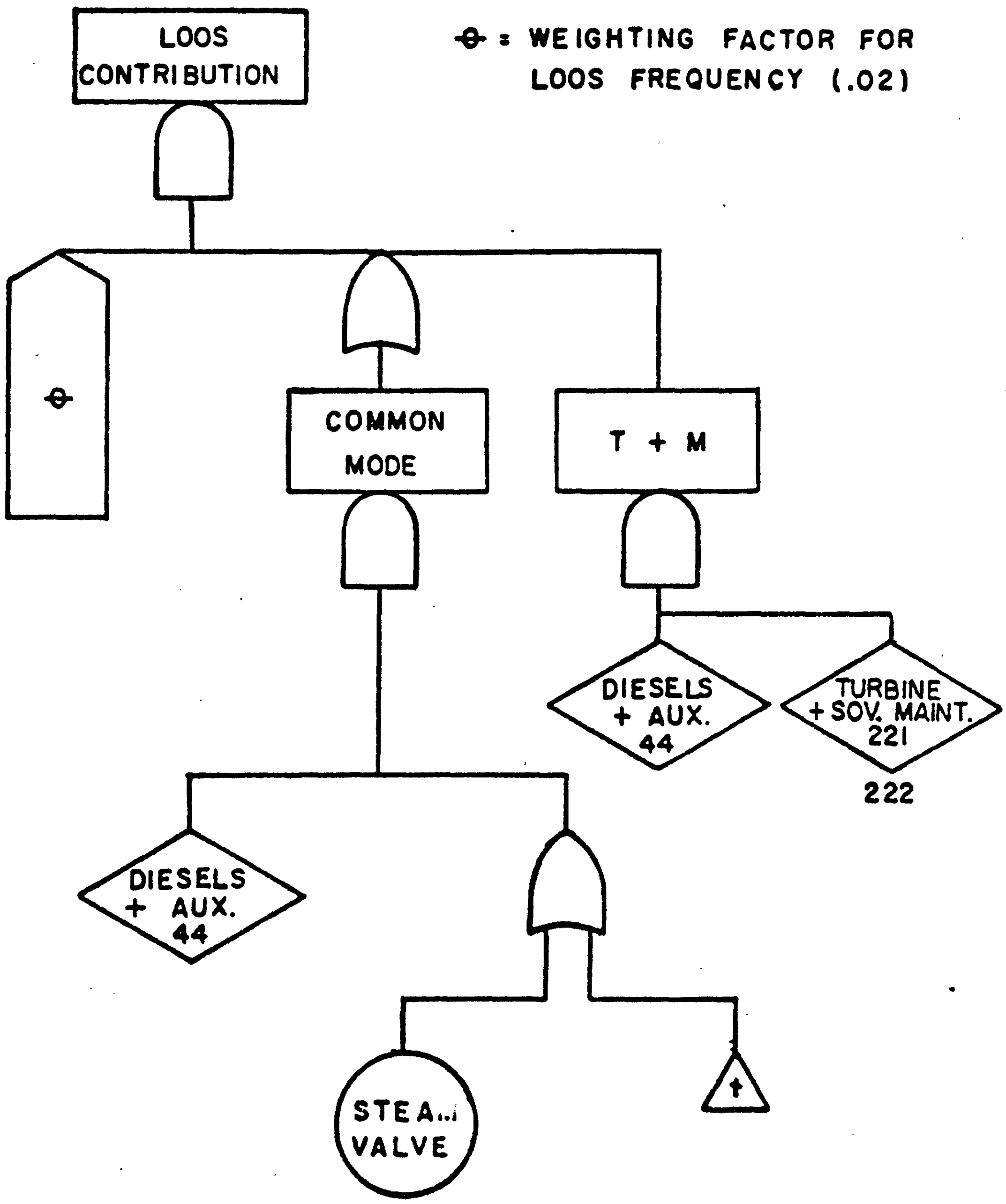
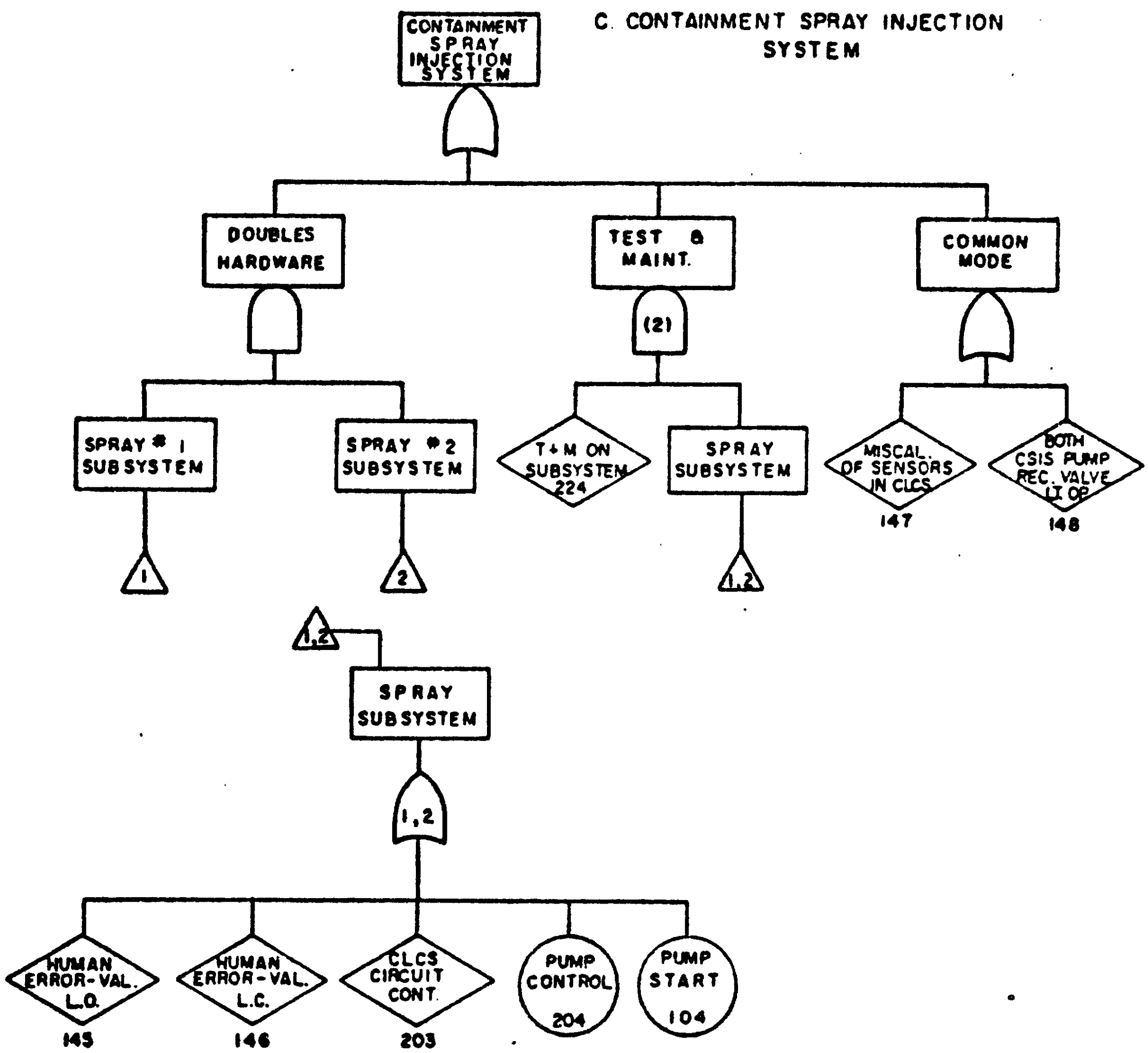

LOOS
CONTRIBUTION
-θ- = WEIGHTING FACTOR FOR
LOOS FREQUENCY (.02)
θ
COMMON
MODE
T + M
DIESELS
+ AUX.
44
TURBINE
+ SOV. MAINT.
221
222
DIESELS
+ AUX.
44
STEAM
VALVE

FIGURE C-2



C. CONTAINMENT SPRAY INJECTION SYSTEM
CONTAINMENT SPRAY INJECTION SYSTEM
DOUBLES HARDWARE
TEST & MAINT.
(2)
COMMON MODE
SPRAY # 1 SUBSYSTEM
SPRAY # 2 SUBSYSTEM
T+M ON SUBSYSTEM 224
SPRAY SUBSYSTEM
MISCAL. OF SENSORS IN CLCS
147
BOTH CSIS PUMP REC. VALVE LT OP
148
1
2
1,2
1,2
SPRAY SUBSYSTEM
1,2
HUMAN ERROR-VAL. L.O.
145
HUMAN ERROR-VAL. L.C.
146
CLCS CIRCUIT CONT.
203
PUMP CONTROL
204
PUMP START
104

FIGURE C-3

G. CONTAINMENT HEAT REMOVAL SYSTEM

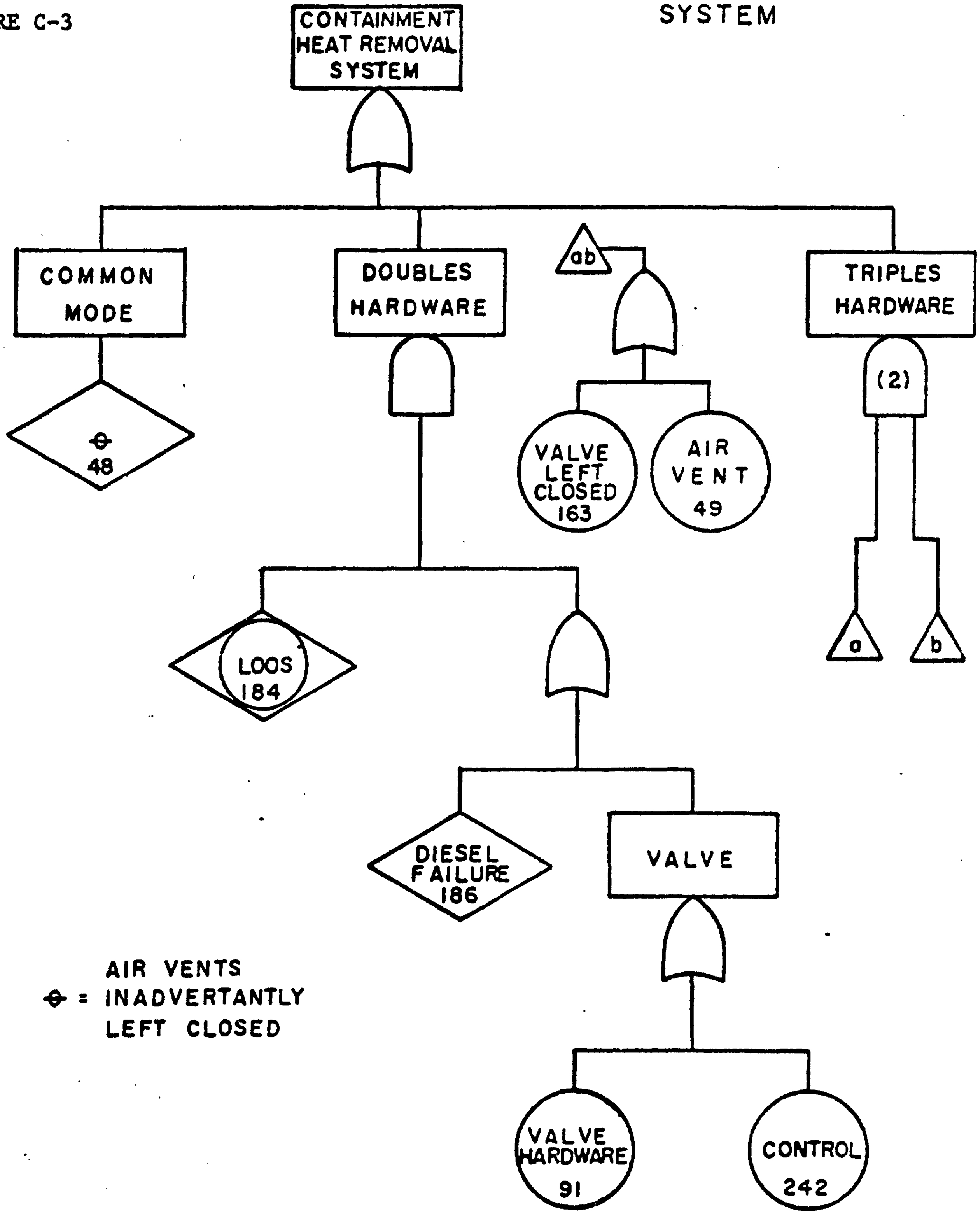


FIGURE C-4

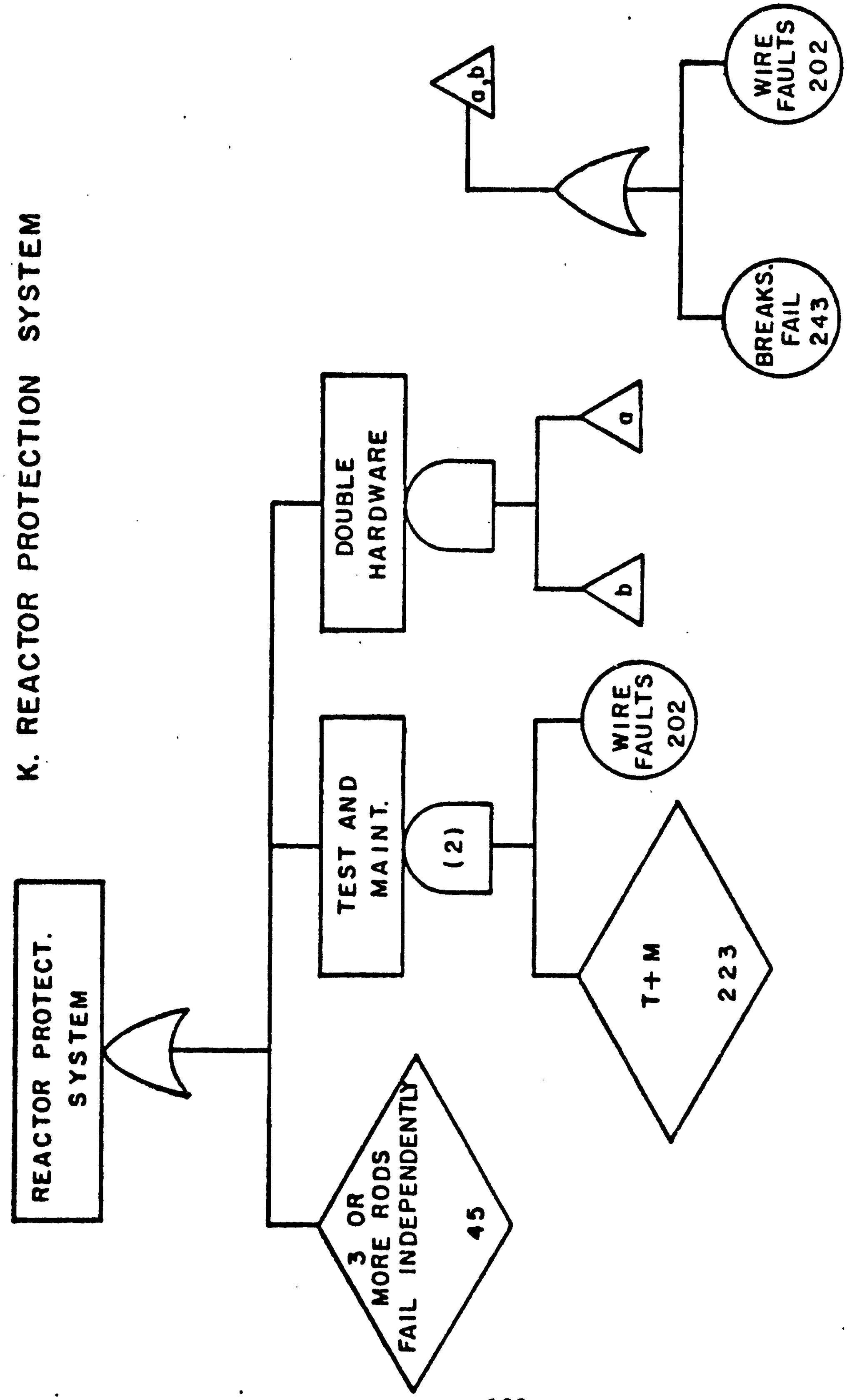
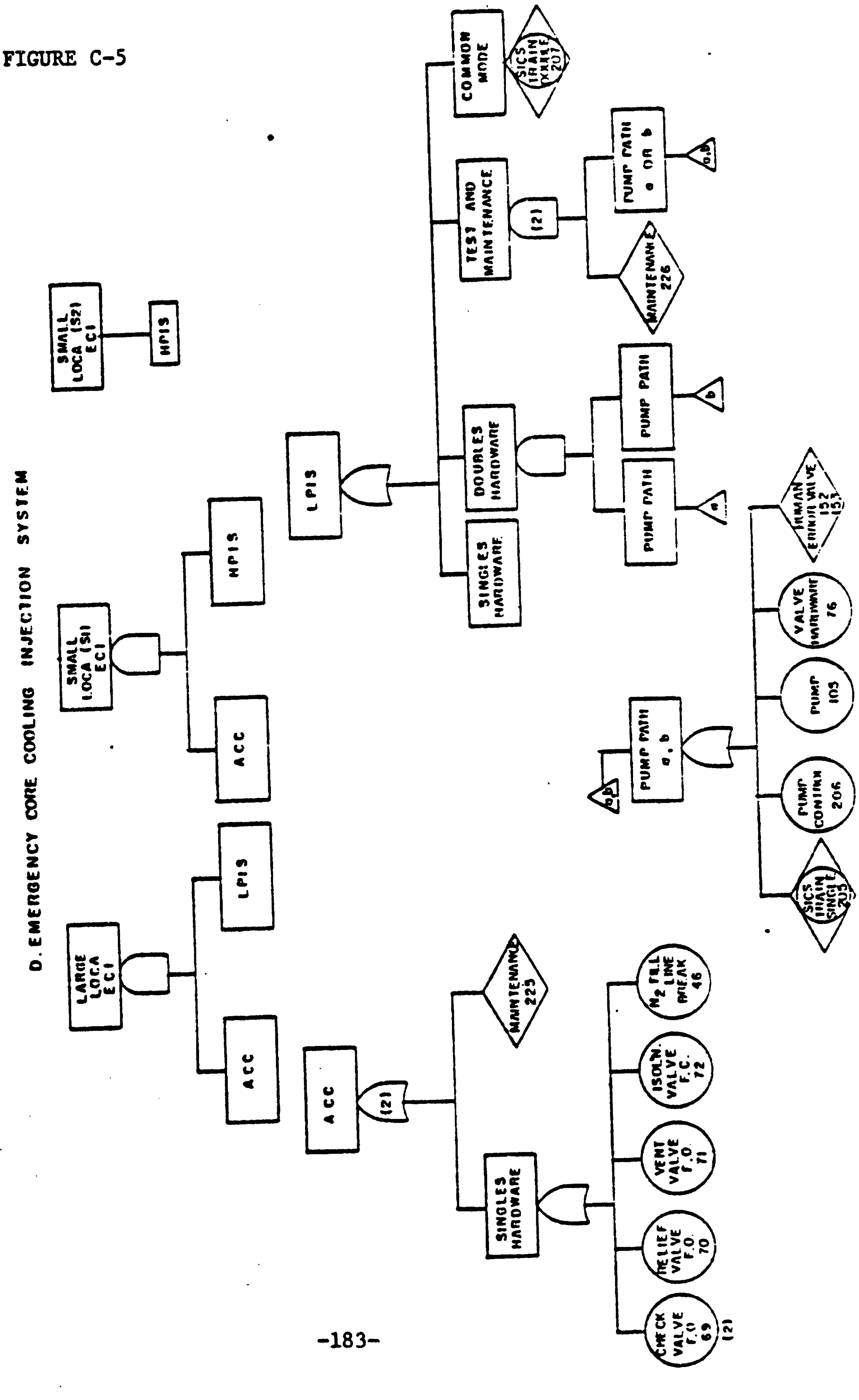
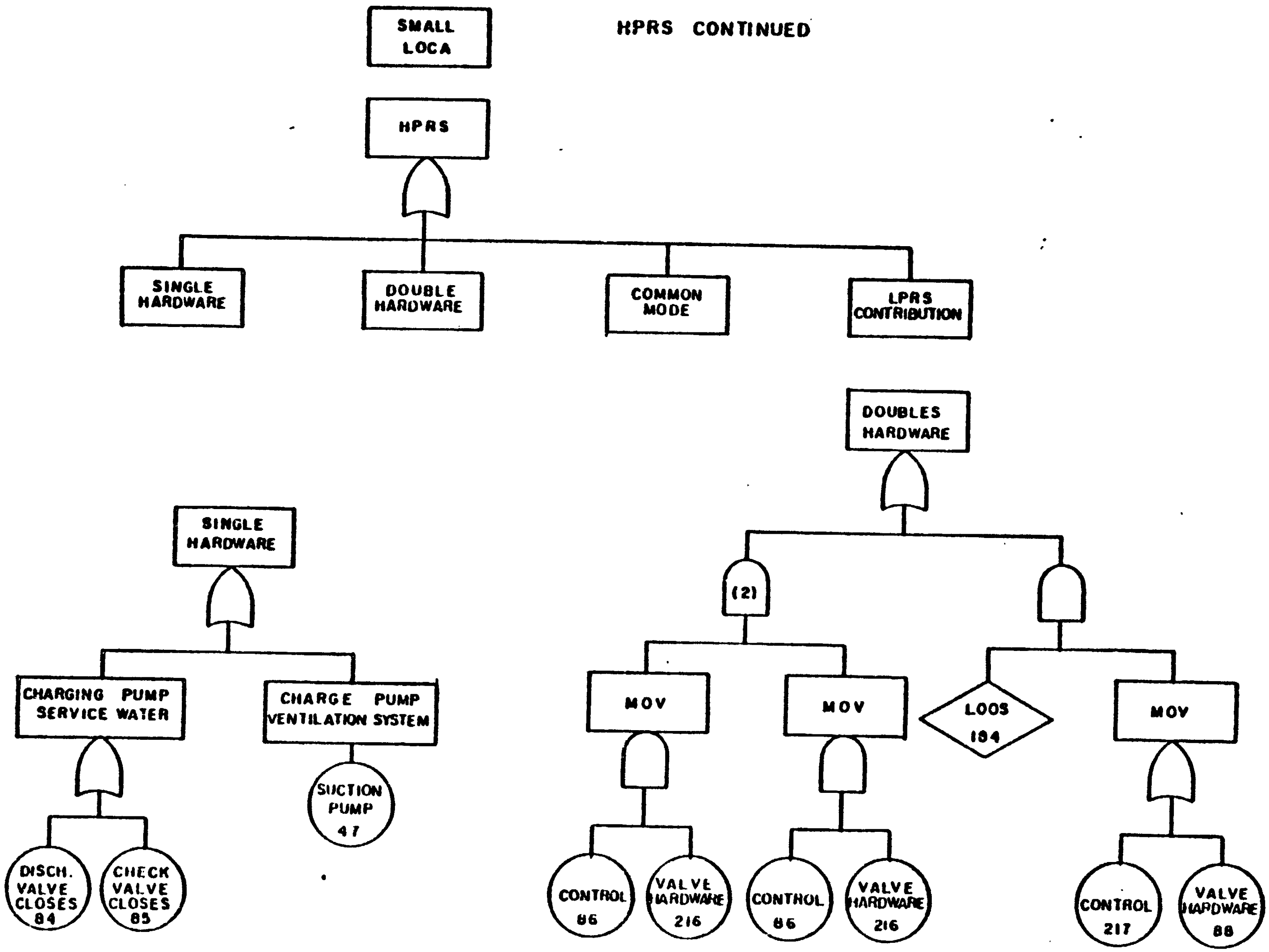

K. REACTOR PROTECTION SYSTEM
REACTOR PROTECT. SYSTEM
3 OR MORE RODS FAIL INDEPENDENTLY
45
TEST AND MAINT.
(2)
T+M
223
WIRE FAULTS 202
DOUBLE HARDWARE
a
b
a,b
BREAKS. FAIL 243
WIRE FAULTS 202

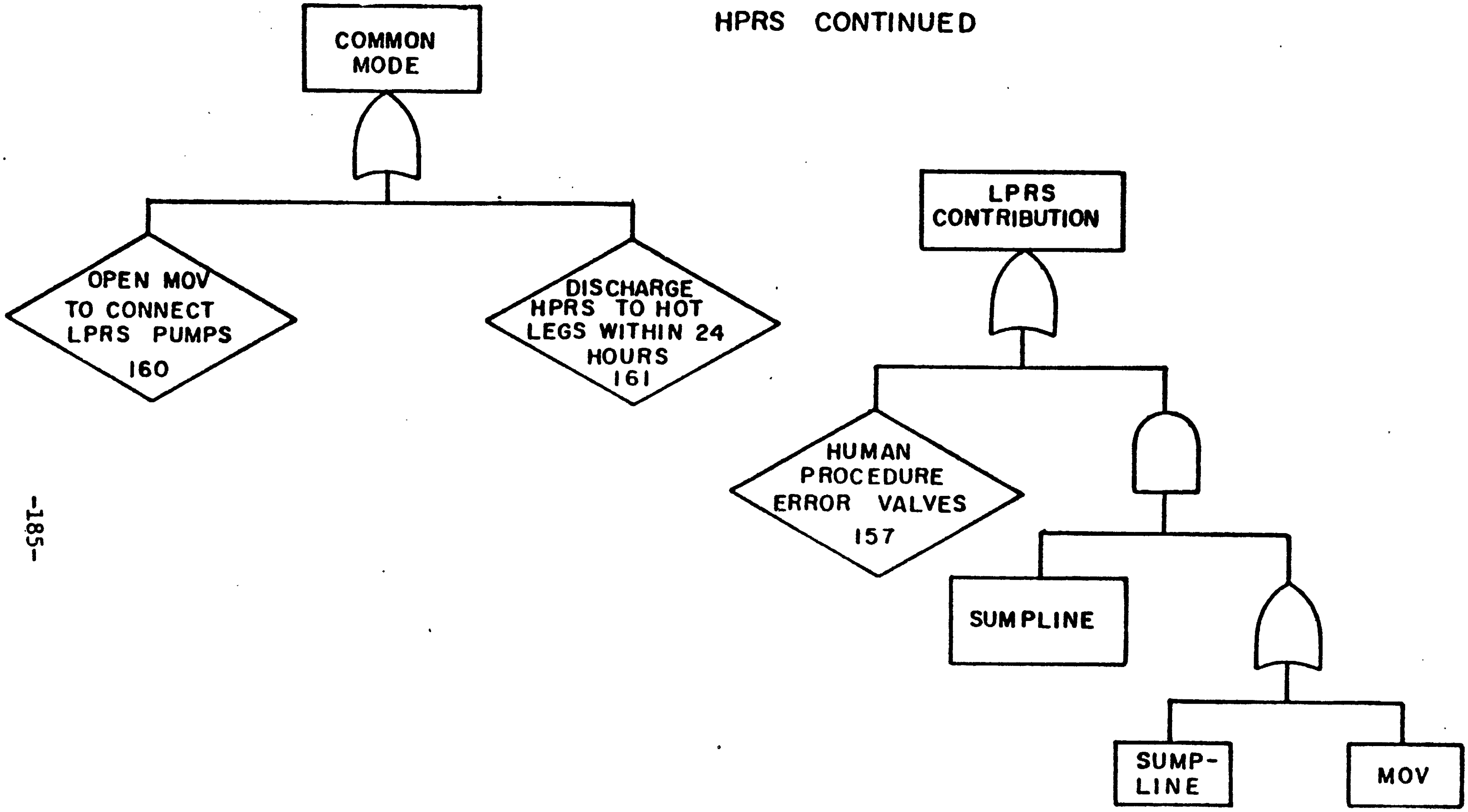
FIGURE C-5


D. EMERGENCY CORE COOLING INJECTION SYSTEM
SMALL LOCA (S2) ECI
HPIS
SMALL LOCA (S1) ECI
HPIS
ACC
LARGE LOCA ECI
LPIS
ACC
LPIS
SINGLES HARDWARE
DOUBLES HARDWARE
TEST AND MAINTENANCE
COMMON MODE
(2)
PUMP PATH
PUMP PATH
MAINTENANCE 226
PUMP PATH a OR b
ACC
(2)
MAINTENANCE 225
SINGLES HARDWARE
PUMP PATH a, b
PUMP 105
VALVE HARDWARE 76
CHECK VALVE F.O. 69 (2)
VENT VALVE F.O. 71
ISOLN. VALVE F.C. 72

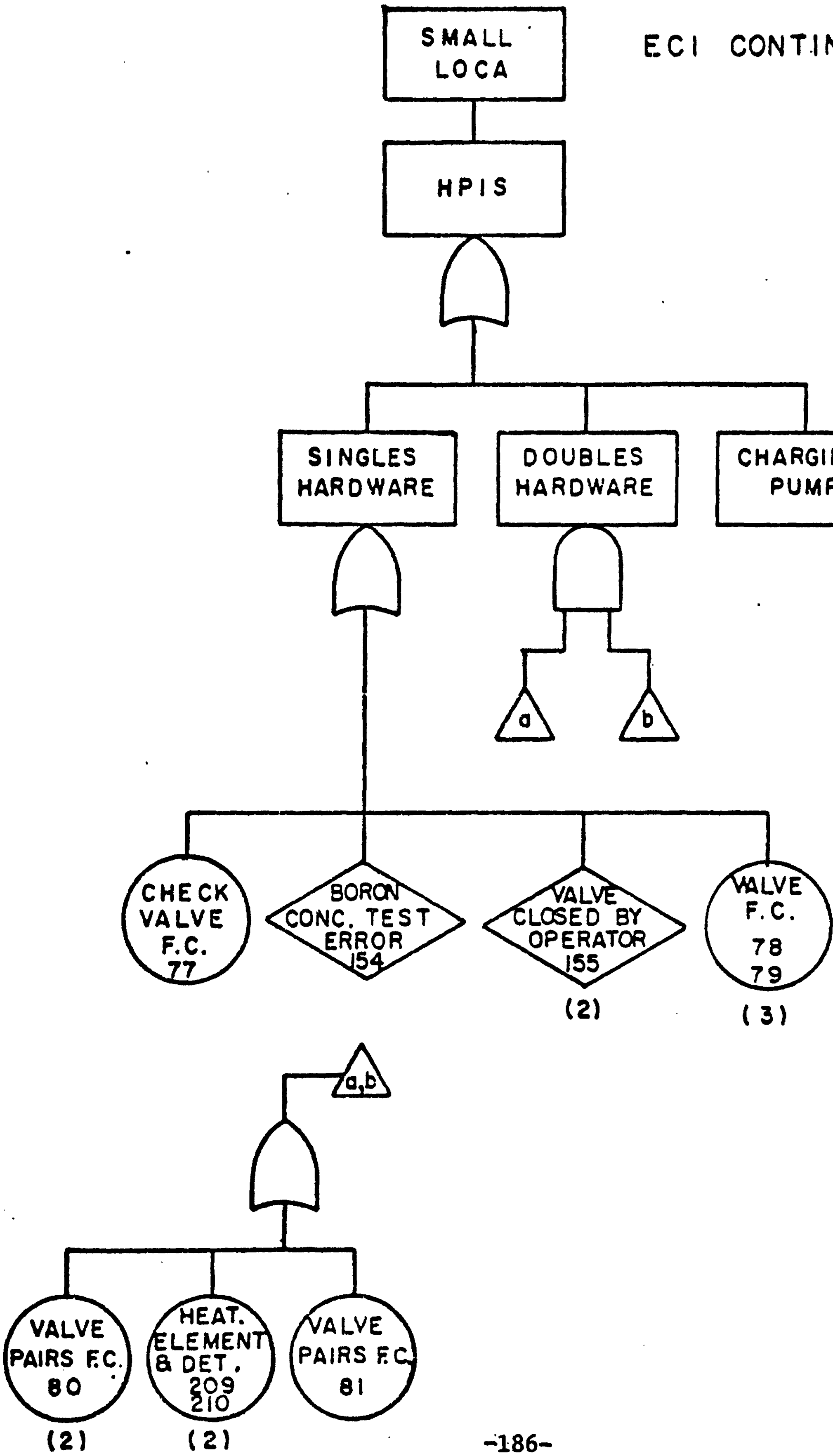
HPRS CONTINUED



HPRS CONTINUED



ECI CONTINUED



ECI CONTINUED

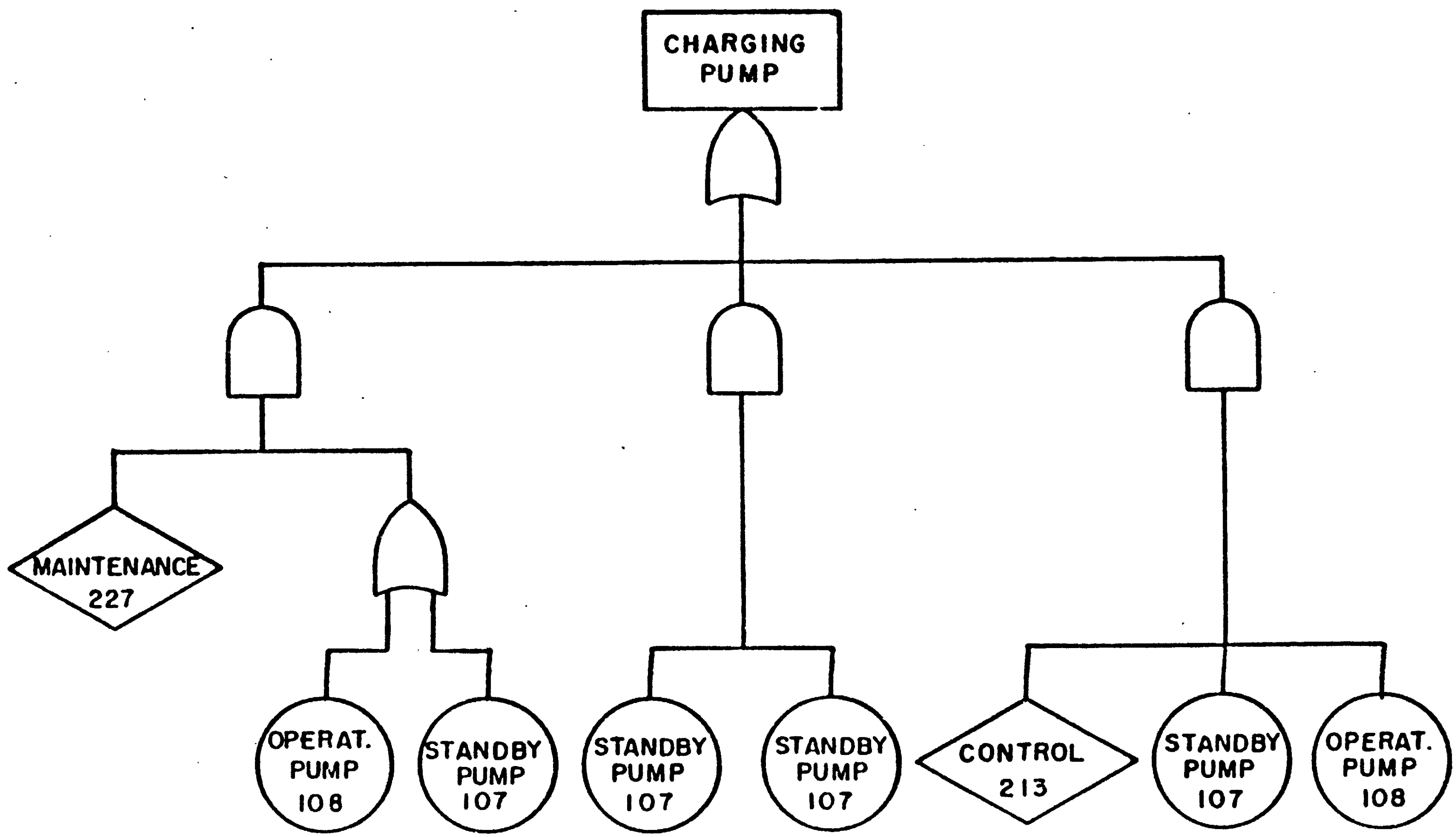


FIGURE C-6

H. FAILURE EMERGENCY CORE COOLING RECIRCULATION SYSTEM

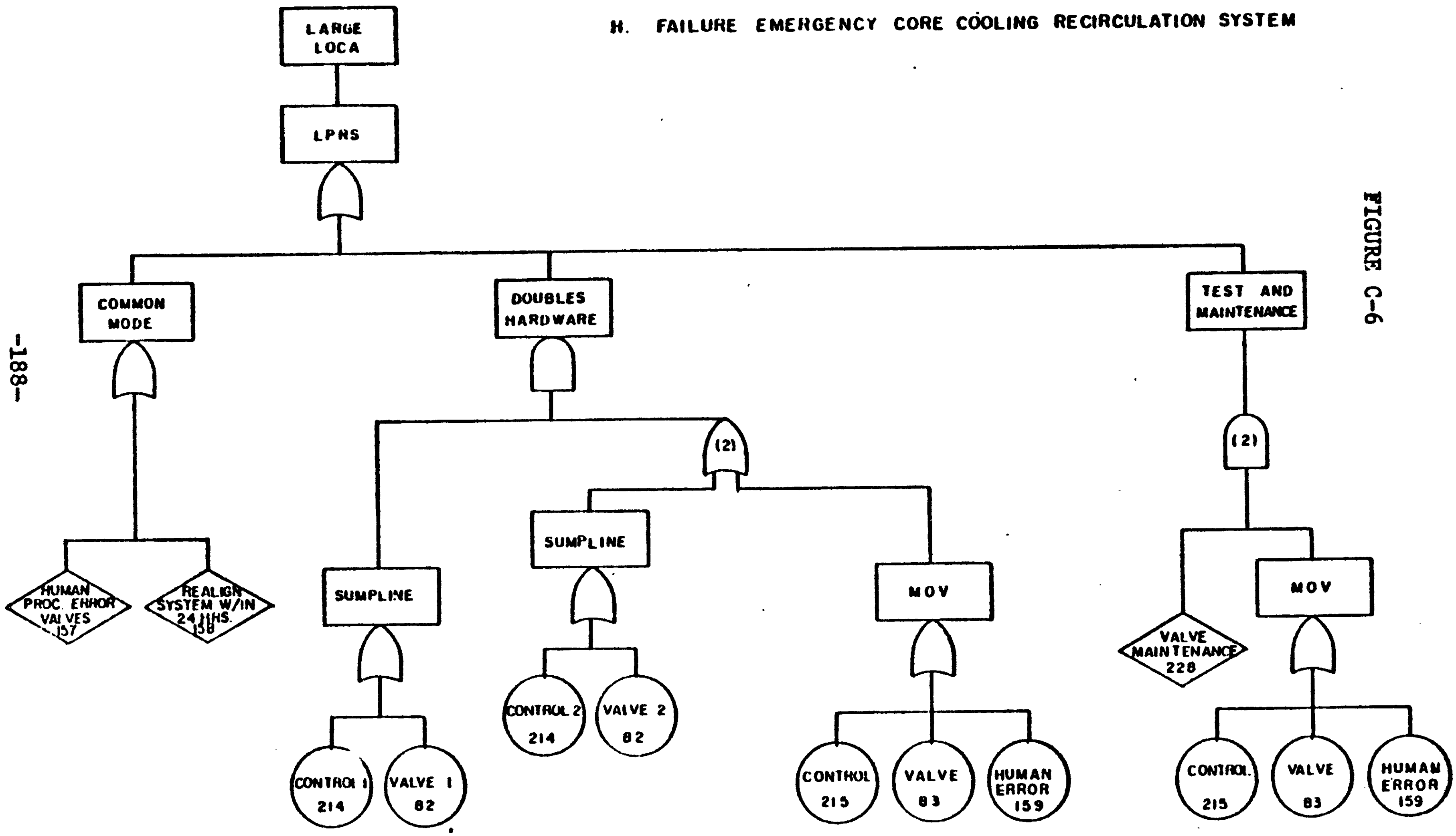


FIGURE C-7

W. FAILURE TO REMOVE RESIDUAL CORE HEAT

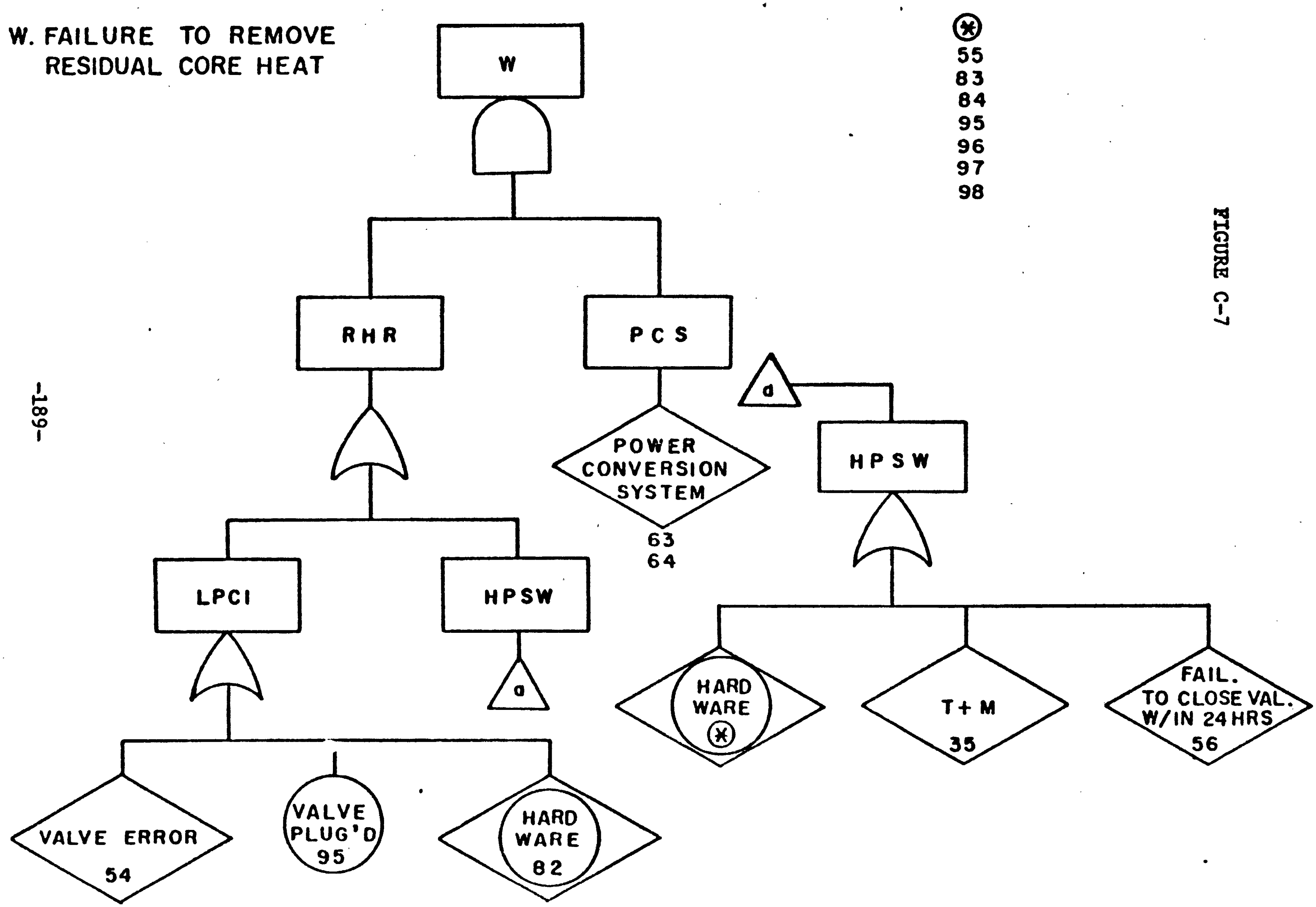


FIGURE C-8

Q. REACTOR PROTECTION SYSTEM

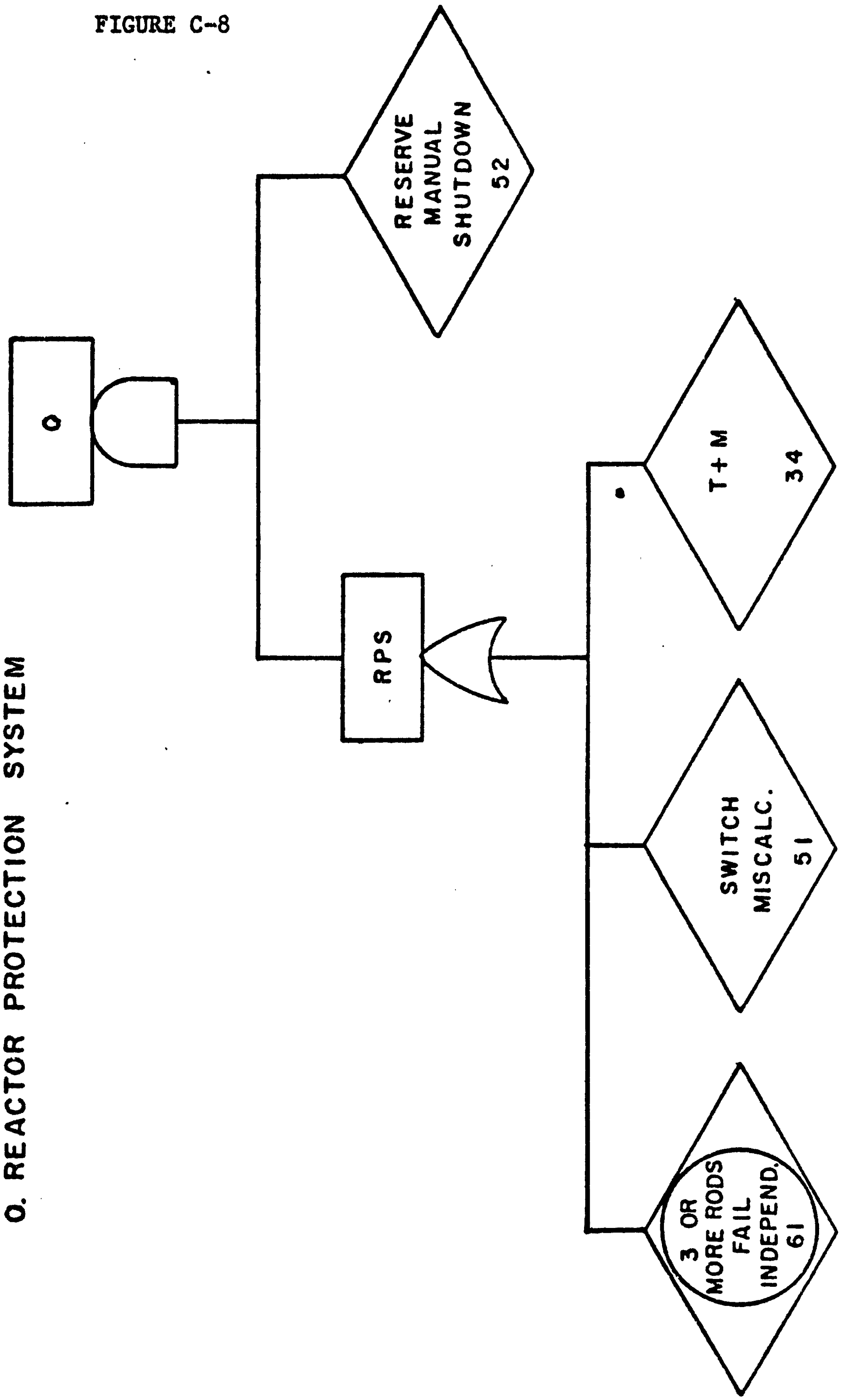


FIGURE C-9

V. FAILURE OF LOW PRESSURE ECCS TO PROVIDE CORE MAKEUP WATER

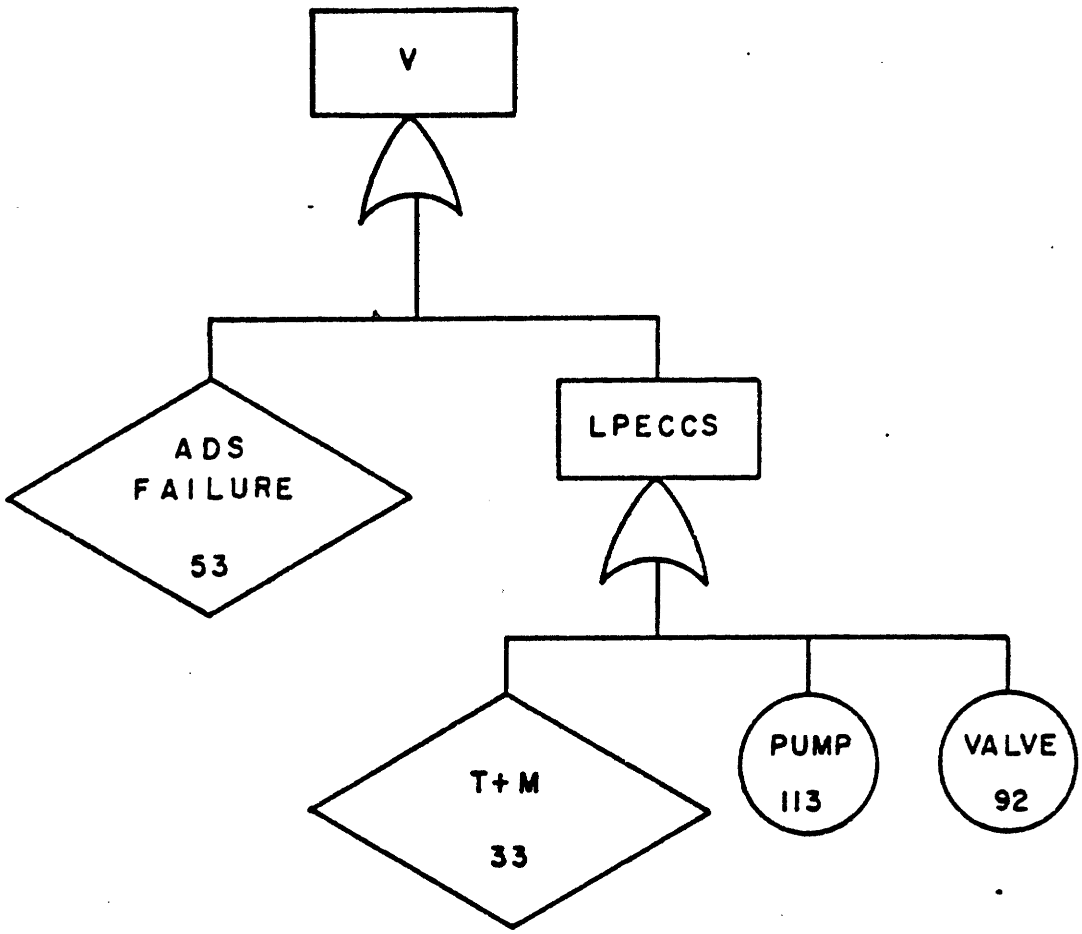


FIGURE C-10

Q. FAILURE OF NORMAL FEEDWATER SYSTEM TO PROVIDE CORE MAKEUP WATER

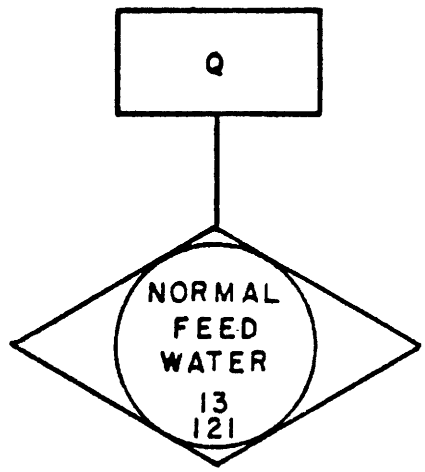
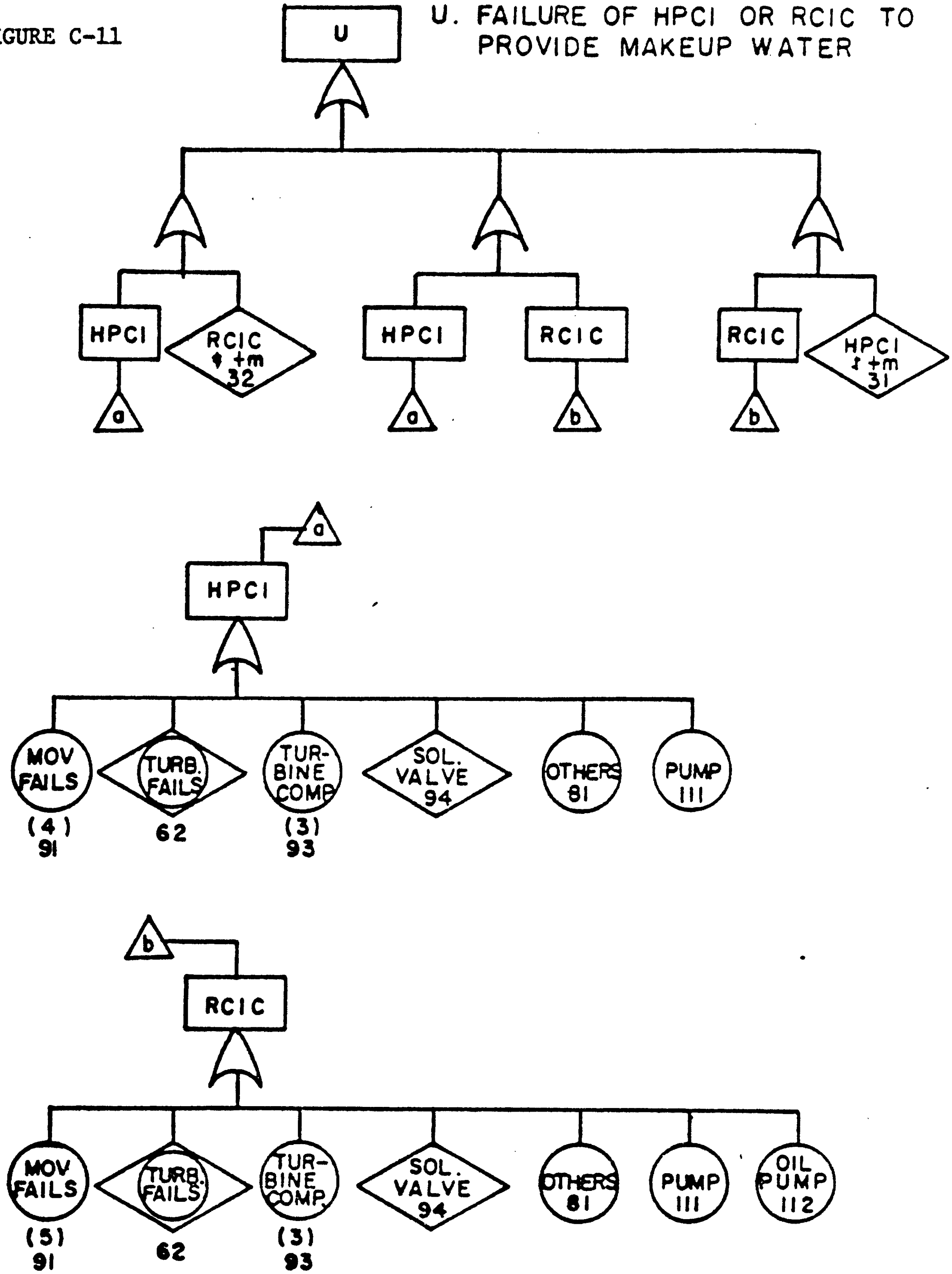


FIGURE C-11


U
U. FAILURE OF HPCI OR RCIC TO PROVIDE MAKEUP WATER
HPCI
RCIC
+m
32
a
HPCI
a
RCIC
b
RCIC
b
HPCI
+m
31
a
HPCI
MOV FAILS
(4)
91
TURB. FAILS
62
TUR-BINE COMP
(3)
93
SOL. VALVE
94
OTHERS
81
PUMP
111
b
RCIC
MOV FAILS
(5)
91
TURB. FAILS
62
TUR-BINE COMP.
(3)
93
SOL. VALVE
94
OTHERS
81
PUMP
111
OIL PUMP
112

Appendix D

Risk Parameters

The reactor safety study generally showed that the highest consequence core melt accidents tend to have the lowest probability of occurrence. Core melt probability is the sum of the first seven release categories, in the PWR, and the first four categories, in the BWR. Since public risk is determined by the product of consequence and probability of occurrence, and the consequences are not the same for each category, core melt probability is not a completely adequate measure of risk. To circumvent this problem, the RSS used the CRAC code to determine the consequences for each accident type. In order to reduce time and money spent, an effort was made to find a simpler method to relate release category probability to risk. A few studies have been done[13] to develop an average set of consequences for each accident, given the complexity of widely varying sites for nuclear power plants. The original scope of the report was to provide for a countrywide average; however, the most complete results published concerned the consequences for a northeast river valley composite site. The description for such a site can be found in the RSS[14]. A functional relationship (for both BWR and PWR reactor types) for three consequences from the RSS was performed by Sandia Labs[15]. The results may be found in Table D-1a & D-1b. One can see that release category one is divided into two separate accident types. The cold release occurs when the containment fails due to overpressure before a steam explosion occurs. This accident results from sequences involving a large or small LOCA and failure of any of the following systems: the containment spray injection system (C), the containment spray recirculation system (F), and the containment heat removal system (G). The hot releases result from transient event sequences, as well as LOCA's involving failure of ECCS systems, injection or recirculation modes. In this case the containment can fail by steam explosion. A more detailed analysis of containment failure modes can be found in the RSS. A synopsis of that analysis from the RSS for both reactor types can be found in Tables D-2 and D-3.

The Sandia study considered only early and latent fatalities and property damage. The RSS reported a more complete list of consequences, namely, early fatalities, early illnesses, thyroid nodules, latent cancer fatalities, genetic effects, relocation and decontamination area, and total property damage. The complementary cumulative distribution functions reported in the RSS can be found in Figures D-1 through D-7. By examining these figures, it can be seen that the early fatalities distribution is similar to the early illness curve. In the same manner, latent cancer fatalities are similar to genetic effects and thyroid nodules, and total property damage is similar to relocation and decontamination area. The similarities are in the shape of the distribution function, as well as the relative probabilities and variation in magnitude of consequences. Given that risk is the product of probability and consequence, the total risk to the public is the integral under the complementary cumulative distribution function. The result of that integration is approximately equal to the product of the median probabilities for each release category and the consequences listed in Table D-1. The form of the distribution function is a result of the uncertainty contained in the release category calculation and the variations in weather and population for a composite site for the entire U. S. While the information regarding the form of the distribution function is lost by not performing consequence calculations with the CRAC code, sufficient evidence for a sensitivity study can be found from the integrated values. An exact calculation would be wasteful and unproductive, given the uncertainties inherent in using only point values in release category calculations, as well as the uncertainties reported in the RSS itself.

In performing a sensitivity study it would be convenient to have a single parameter to represent public risk, in order to simplify both the analysis and the presentation of the results. However, combining the three risk values calculated for the three consequences - early deaths, latent deaths, and costs - can be accomplished only by applying a monetary value to life. In order to avoid prejudicing the results of this, all three parameters are reported, where it is convenient. At the same time, core melt probability is reported, since it satisfies the requirement of a single parameter and it is useful to regulatory agencies. By examining the magnitude of the probabilities for each release category, it can be seen that release category seven in the PWR and release category three in the BWR will contribute most to changes in core melt probabilitv. However, the consequences of those categories are small compared to the others, particularly in the case of the PWR. For this reason the sensitivities reported using the core melt parameter must be kept in perspective when one is considering reduction in public risk.

This study reports results using all of the four parameters previously discussed. Due to the problems mentioned earlier in connection with combining these parameters to represent risk, a reasonable methodology for evaluating public safety considerations would be to consider larger sensitivities from any one of the four parameters. Specific safety analyses must make some assessment of the relative value of each of the parameters, in order to adequately calculate the benefits to the public from any reduction in accident consequences. It should be noted that individual release category probabilities may also be considered as sensitivity parameters, especially since they contain more specific information as to accident types.

An approximate example of how to use the results can be shown by the use of the sensitivity tables on early deaths, latent cancer fatalities, and total property damage. For the core spray injection system C, reductions of approximately three, sixteen, and seven percent are attained in early deaths, latent cancers, and total property damage, respectively. Social scientists and medical personnel could provide some value for an early death and an early illness. There are roughly one hundred times more early illnesses than early fatalities and their treatment must be accounted for in the early death parameter. The latent cancer parameter must be translated into latent cancer fatality costs, the cost of treating about ten times that many cases of thyroid nodules, and the cost of roughly one tenth as many genetic effects per year. Finally, the total property damage parameter must also account for public aversion and the costs of the relocation and contamination area. Considering only the reduction in property damage, a credit of sixty thousand dollars could be attained over a forty year plant life for a factor reduction in CBIS failure probability of three. This is clearly not in the range of a worthwhile backfit investment;

however, considering engineer's salaries, it is certainly worth considering future research, if the system proves promising, for reductions of any amount near three or more. A significant cost benefit from the other factors, especially those resulting from the latent cancer parameter, indicates that there are probably benefit to cost ratios greater than one for many possible changes for future plants. Given that the total property damages amount to about ten thousand dollars per percent reduction over a single plant life, many individual components could prove promising for further research and design work.

In summary, the four sensitivity parameters reported in this study are core melt probability, early deaths, latent deaths, and total property damage. These consequence parameters are representative of the integrals of the complementary cumulative distribution functions reported in the RSS and found in Figures D-1 through D-7. Their relative use is dependent on the concerns of the user of this study and are beyond the scope of this work. Nevertheless, they can be considered adequate to provide insights into reactor safety.

TABLE D-1A

Expected Consequences per Release
Northeast River Valley Composite Site

PWR

Category	Early Fatalities	Latent Cancer Fatalities	Property Damage (10^6 \$)
1a (cold)	91	120	2050
1b (hot)	8	114	2270
2	7	67	2440
3	0.4	55	987
4	0	18	335
5	0	6	201
6	0	1	173
7	0	~0	171
8	0	~0	1.
9	0	~0	0

BWR

Category	Early Fatalities	Latent Cancer Fatalities	Property Damage (10^6 \$)
1	7	154	2450
2	~1	100	2970
3	0	51	789
4	0	3	29
5	0	~0	~0

TABLE D-1B

RISK CONTRIBUTIONS FROM CORE-MELT ACCIDENTS

	RELEASE CATEGORIES	APPROX. RISK CONTRIBUTION %		
		EARLY DEATHS	LATENT DEATHS	COSTS
CORE MELT	PWR 1	35	9	6
	PWR 2	41	46	53
	PWR 3	24	39	21
		(~100)	(~94)	(~80)
	PWR 4			
	PWR 5			
	PWR 6			
	PWR 7			
	PWR 8			
	PWR 9			
CORE MELT	BWR 1	93	9	7
	BWR 2	7	32	46
	BWR 3	---	58	46
		(~100)	(~100)	(~100)
	BWR 4			
	BWR 5			

TABLE D-2

PWR 1

This release category can be characterized by a core meltdown followed by a steam explosion on contact of molten fuel with the residual water in the reactor vessel. The containment spray and heat removal systems are also assumed to have failed and, therefore, the containment could be at a pressure above ambient at the time of the steam explosion. It is 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 10 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. The total release would contain approximately 70% of the iodines and 40% of the alkali metals present in the core at the time of release.[1] 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 category. This category also includes certain potential accident sequences that would involve the occurrence of core melting and a steam explosion after containment rupture due to overpressure. In these sequences, the rate of energy release would be lower, although still relatively high.

PWR 2

This category is associated with the failure of core-cooling systems and core melting concurrent with the failure of containment spray and heat-removal systems. Failure of the containment barrier would occur through overpressure, causing a substantial fraction of the containment atmosphere to be released in a puff over a period of about 30 minutes. Due to the sweeping action of gases generated during containment vessel meltthrough, the release of radioactive material would continue at a relatively low rate thereafter. The total release would contain approximately 70% of the iodines and 50% of the alkali metals present in the core at the time of release. As in PWR release category 1, the high temperature and pressure within containment at the time of containment failure would result in a relatively high release rate of sensible energy from the containment.

PWR 3

This category involves an overpressure failure of the containment due to failure of containment heat removal. Containment failure would occur prior to the commencement of core melting. Core melting then would cause radioactive materials to be released through a ruptured containment barrier. Approximately 20% of the iodines and 20% of th alkali metals present in the core at the time of release would be released to the atmosphere. Most of the release would occur over a period of about 1.5 hours. The release of radioactive material from containment would be caused by the sweeping action of gases generated by the reaction of the molten fuel with concrete. Since these gases would be initially heated by contact with the melt, the rate of sensible energy release to the atmosphere would be moderately high.

PWR 4

This category involves failure of the core-cooling system and the containment spray injection system after a loss-of-coolant accident, together with a concurrent failure of the containment system to properly isolate. This would result in the release of 9% of the iodines and 4% of the alkali metals present in the core at the time of release. Most of the release would occur continuously over a period of 2 to 3 hours. Because the containment recirculation spray and heat-removal systems would operate to remove heat from the containment atmosphere during core melting, a relatively low rate of release of sensible energy would be associated with this category.

PWR 5

This category involves failure of the core cooling systems and is similar to PWR release category 4, except that the containment spray injection system would operate to further reduce the quantity of airborne radioactive material and to initially suppress containment temperature and pressure. The containment barrier would have a large leakage rate due to a concurrent failure of the containment system to properly isolate, and most of the radioactive material would be released continuously over a period of several hours. Approximately 3% of the iodines and 0.9% of the alkali metals present in the core would be released. Because of the operation of the containment heat-removal systems, the energy release rate would be low.

PWR 6

This category involves a core meltdown due to failure in the core cooling systems. The containment sprays would not operate, but the containment barrier would retain its integrity until the molten core proceeded to melt through the concrete containment base mat. The radioactive materials would be released into the ground, with some leakage to the atmosphere occurring upward through the ground. Direct leakage to the atmosphere would also occur at a low rate prior to containment-vessel meltthrough. Most of the release would occur continuously over a period of about 10 hours. The release would include approximately 0.08% of the iodines and alkali metals present in the core at the time of release. Because leakage from containment to the atmosphere would be low and gases escaping through the ground would be cooled by contact with the soil, the energy release rate would be very low.

PWR 7

This category is similar to PWR release category 6, except that containment sprays would operate to reduce the containment temperature and pressure as well as the amount of airborne radioactivity. The release would involve 0.002% of the iodines and 0.001% of the alkali metals present in the core at the time of release. Most of the release would occur over a period of 10 hours. As in PWR release category 6, the energy release rate would be very low.

PWR 8

This category approximates a PWR design basis accident (large pipe break), except that the containment would fail to isolate properly on demand. The other engineered safeguards are assumed to function properly. The core would not melt. The release would involve approximately 0.01% of the iodines and 0.05% of the alkali metals. Most of the release would occur in the 0.5-hour period during which containment pressure would be above ambient. Because containment sprays would operate and core melting would not occur, the energy release rate would also be low.

PWR 9

This category approximates a PWR design basis accident (large pipe break), in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into the containment. The core would not melt. It is assumed that the minimum required engineered safeguards would function satisfactorily to remove heat from the core and containment. The release would occur over the 0.5-hour period during which the containment pressure would be above ambient. Approximately 0.00001% of the iodines and 0.00006% of the alkali metals would be released. As in PWR release category 8, the energy release rate would be very low.

BWR 1

This release category is representative of a core meltdown followed by a steam explosion in the reactor vessel. The latter would cause the release of a substantial quantity of radioactive material to the atmosphere. The total release would contain approximately 40% of the iodines and alkali metals present in the core at the time of containment failure. Most of the release would occur over a 1/2 hour period. Because of the energy generated in the steam explosion, this category would be characterized by a relatively high rate of energy release to the atmosphere. This category also includes certain sequences that involve overpressure failure of the containment prior to the occurrence of core melting and a steam explosion. In these sequences, the rate of energy release would be somewhat smaller than for those discussed above, although it would still be relatively high.

BWR 2

This release category is representative of a core meltdown resulting from a transient event in which decay-heat-removal systems are assumed to fail. Containment overpressure failure would result, and core melting would follow. Most of the release would occur over a period of about 3 hours. The containment failure would be such that radioactivity would be released directly to the atmosphere without significant retention of fission products. This category involves a relatively high rate of energy release due to the sweeping action of the gases generated by the molten mass. Approximately 90% of the iodines and 50% of the alkali metals present in the core would be released to the atmosphere.

BWR 3

This release category represents a core meltdown caused by a transient event accompa[nied] by a failure to scram or failure to remove decay heat. Containment failure would occur either before core melt or as a result of gases generated during the interaction of the molten fuel with concrete after reactor-vessel meltthrough. Some fission-product retention would occur either in the suppression pool or the reactor building prior to release to the atmosphere. Most of the release would occur over a period of about 3 hours and would involve 10% of the iodines and 10% of the alkali metals. For those sequences in which the containment would fail due to overpressure after core melt, the rate of energy release to the atmosphere would be relatively high. For those sequences in which overpressure failure would occur before core melt, the energy release rate would be somewhat smaller, although still moderately high.

BWR 4

This release category is representative of a core meltdown with enough containment leakage to the reactor building to prevent containment failure by overpressure. The quantity of radioactivity released to the atmosphere would be significantly reduced normal ventilation paths in the reactor building and potential mitigation by the secondary containment filter systems. Condensation in the containment and the actio of the standby gas treatment system on the releases would also lead to a low rate of energy release. The radioactive material would be released from the reactor building or the stack at an elevated level. Most of the release would occur over a 2-hour period and would involve approximately 0.08% of the iodines and 0.5% of the alkali metals.

BWR 5

This category approximates a BWR design basis accident (large pipe break) in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into containment. The core would not melt, and containme leakage would be small. It is assumed that the minimum required engineered safeguards would function satisfactorily. The release would be filtered and pass throug the elevated stack. It would occur over a period of about 5 hours while the containment is pressurized above ambient and would involve approximately 6×10^{-9}% of the iodines and 4×10^{-7}% of the alkali metals. Since core melt would not occur and containment heat-removal systems would operate, the release to the atmosphere would involve a negligibly small amount of thermal energy.

TABLE D-3

RELEASE CATEGORY ACCIDENT CHARACTERISTICS

RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	9×10^{-7}	2.5	0.5	1.0	25	520[d]	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	2.0	1.5	25	130	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	3×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

(a) A discussion of the isotopes used in the study is found in Appendix VI.* Background on the isotope groups and release mechanisms is found in Appendix VII.*

(b) Includes Mo, Rh, Tc, Co.

(c) Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

(d) A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI.*

TABLE D-4

CONSEQUENCES FOR VARIOUS PROBABILITIES

CONSEQUENCES OF REACTOR ACCIDENTS FOR VARIOUS PROBABILITIES FOR ONE REACTOR

Chance per Reactor-Year	Consequences				
	Early Fatalities	Early Illness	Total Property Damage $\$10^9$	Decontamination Area Square Miles	Relocation Area Square Miles
One in 20,000[a]	<1.0	<1.0	<0.1	<0.1	<0.1
One in 1,000,000	<1.0	300	0.9	2000	130
One in 10,000,000	110	3000	3	3200	250
One in 100,000,000	900	14,000	8	-	290
One in 1,000,000,000	3300	45,000	14	-	-

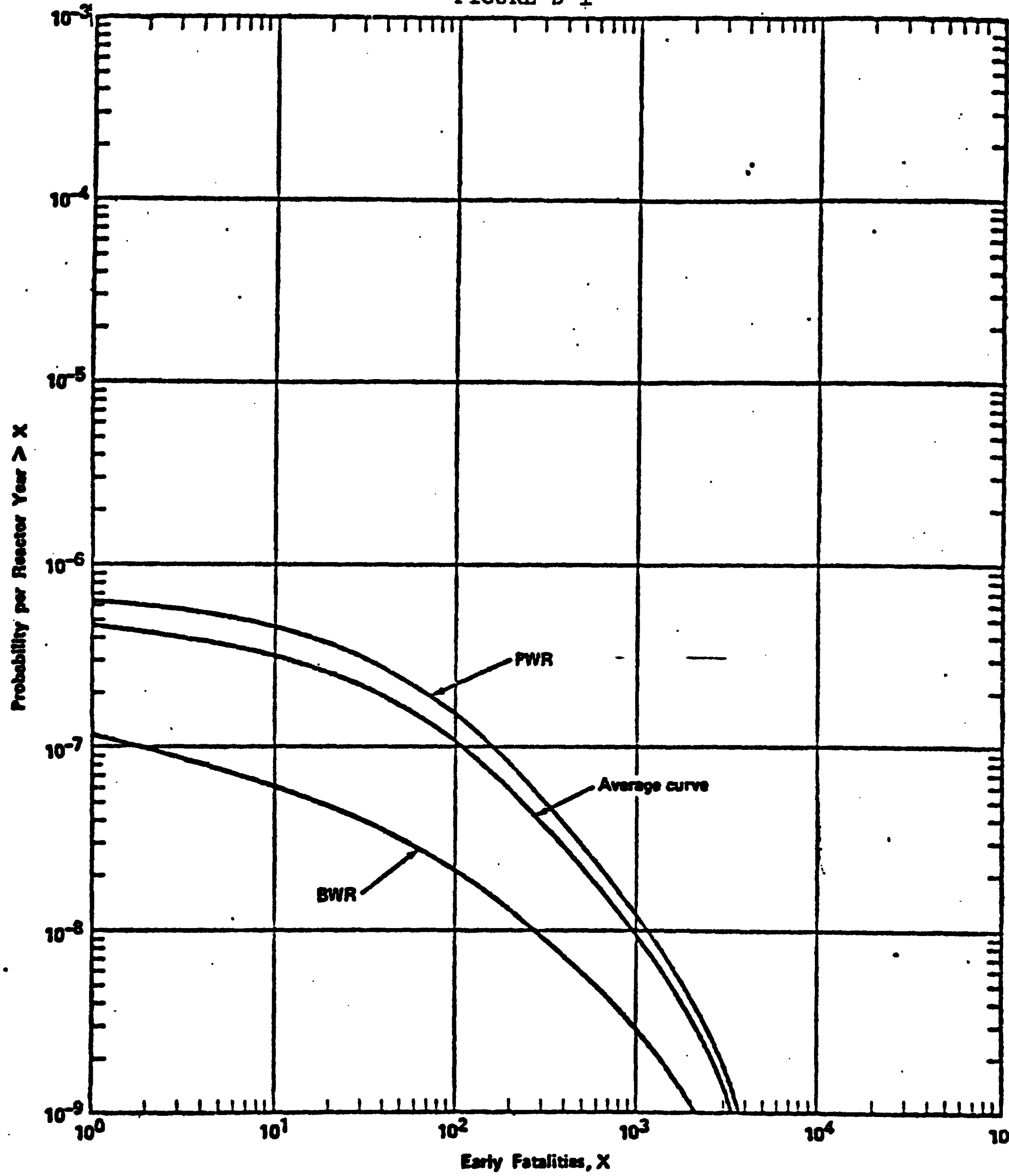
(a) This is the predicted chance of core melt per reactor year.

CONSEQUENCES OF REACTOR ACCIDENTS FOR VARIOUS PROBABILITIES FOR ONE REACTOR

Chance Per Reactor-Year	Consequences		
	Latent Cancer[b] Fatalities (per year)	Thyroid Nodules[b] (per year)	Genetic Effects[c] (per year)
One in 20,000[a]	<1.0	<1.0	<1.0
One in 1,000,000	170	1400	25
One in 10,000,000	460	3500	60
One in 100,000,000	860	6000	110
One in 1,000,000,000	1500	8000	170
Normal Incidence	17,000	8000	8000

(a) This is the predicted chance of core melt per reactor year.
(b) This rate would occur approximately in the 10 to 40 year period following a potential accident.
(c) This rate would apply to the first generation born after a potential accident. Subsequent generations would experie··e effects at a lower rate.

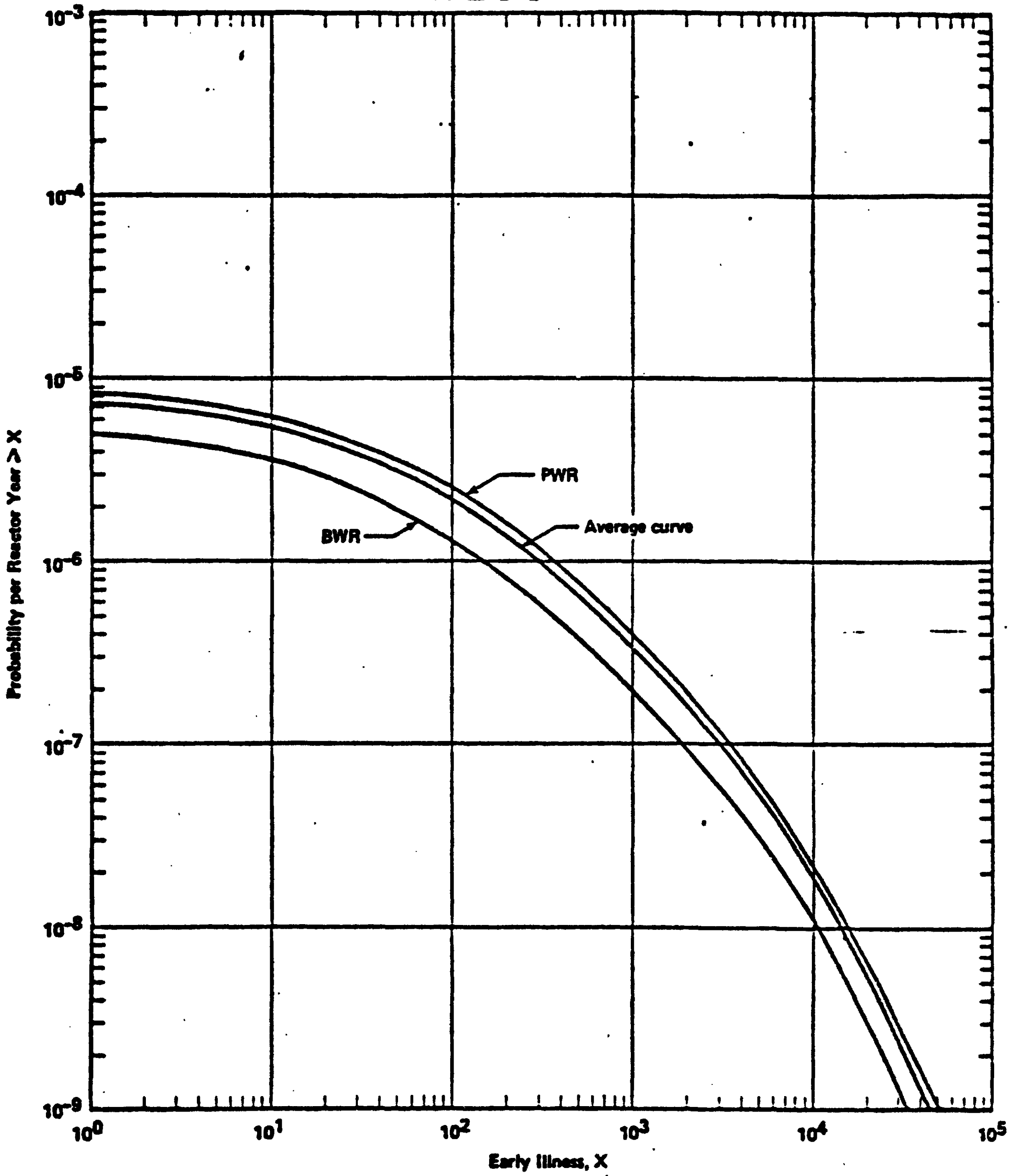
FIGURE D-1



Probability Distribution for Early Fatalities per Reactor Year

Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

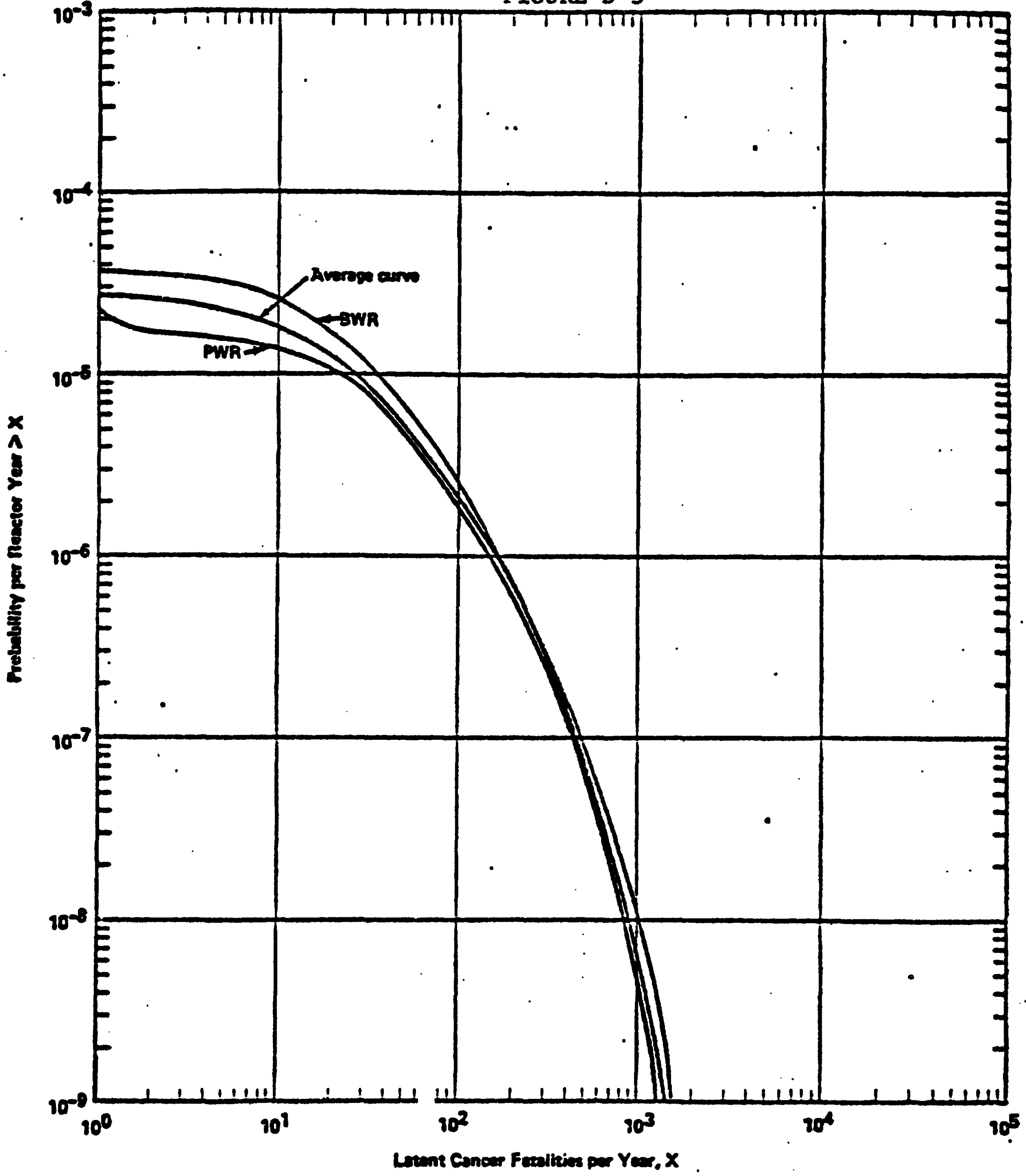
FIGURE D-2



Probability Distribution for Early Illness per Reactor Year

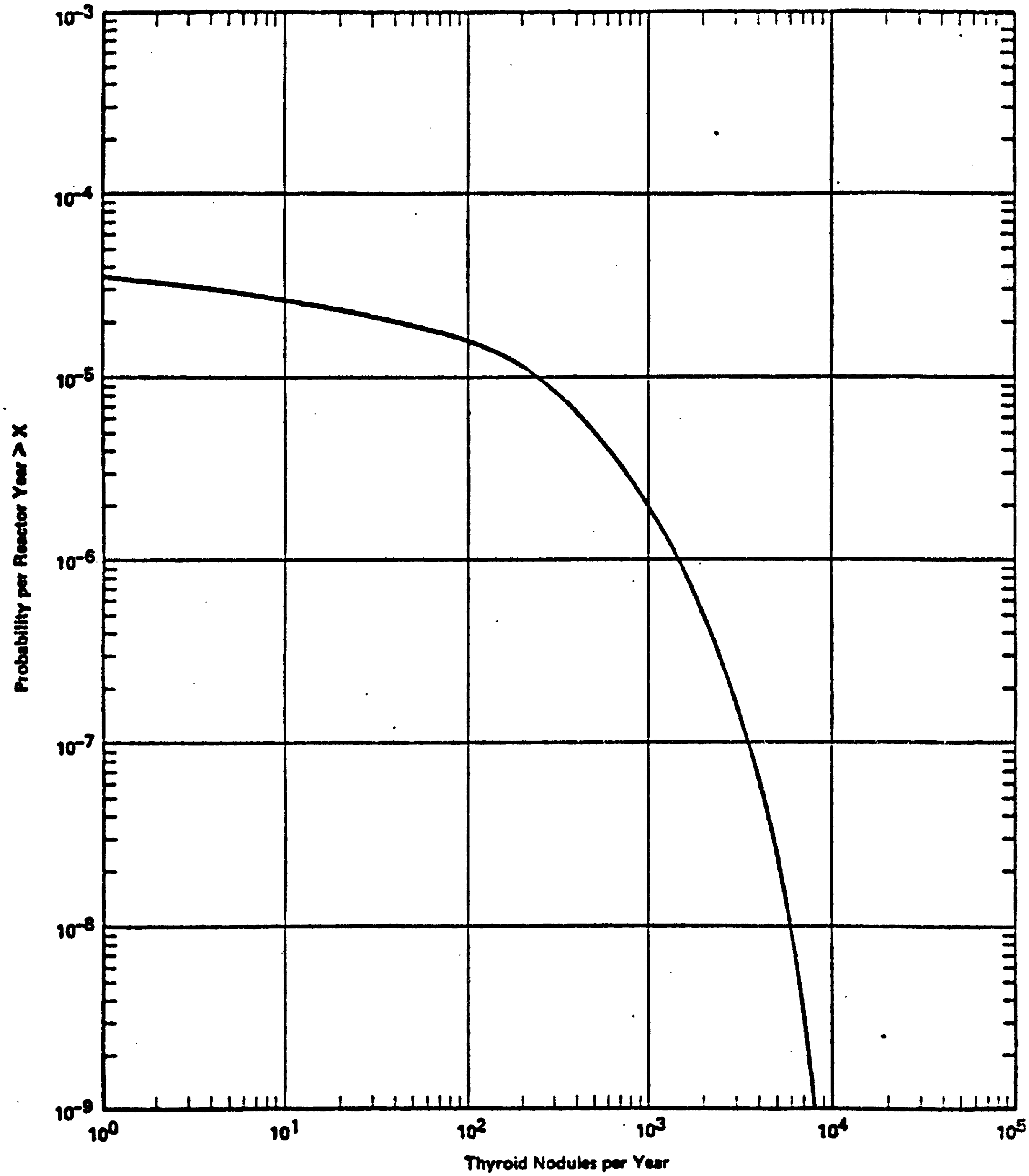
Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

FIGURE D-3



Probability Distribution for Latent Cancer Fatality Incidence per Reactor Year

Note: Approximate uncertainties are estimated to be represented by factors of 1/6 and 3 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.



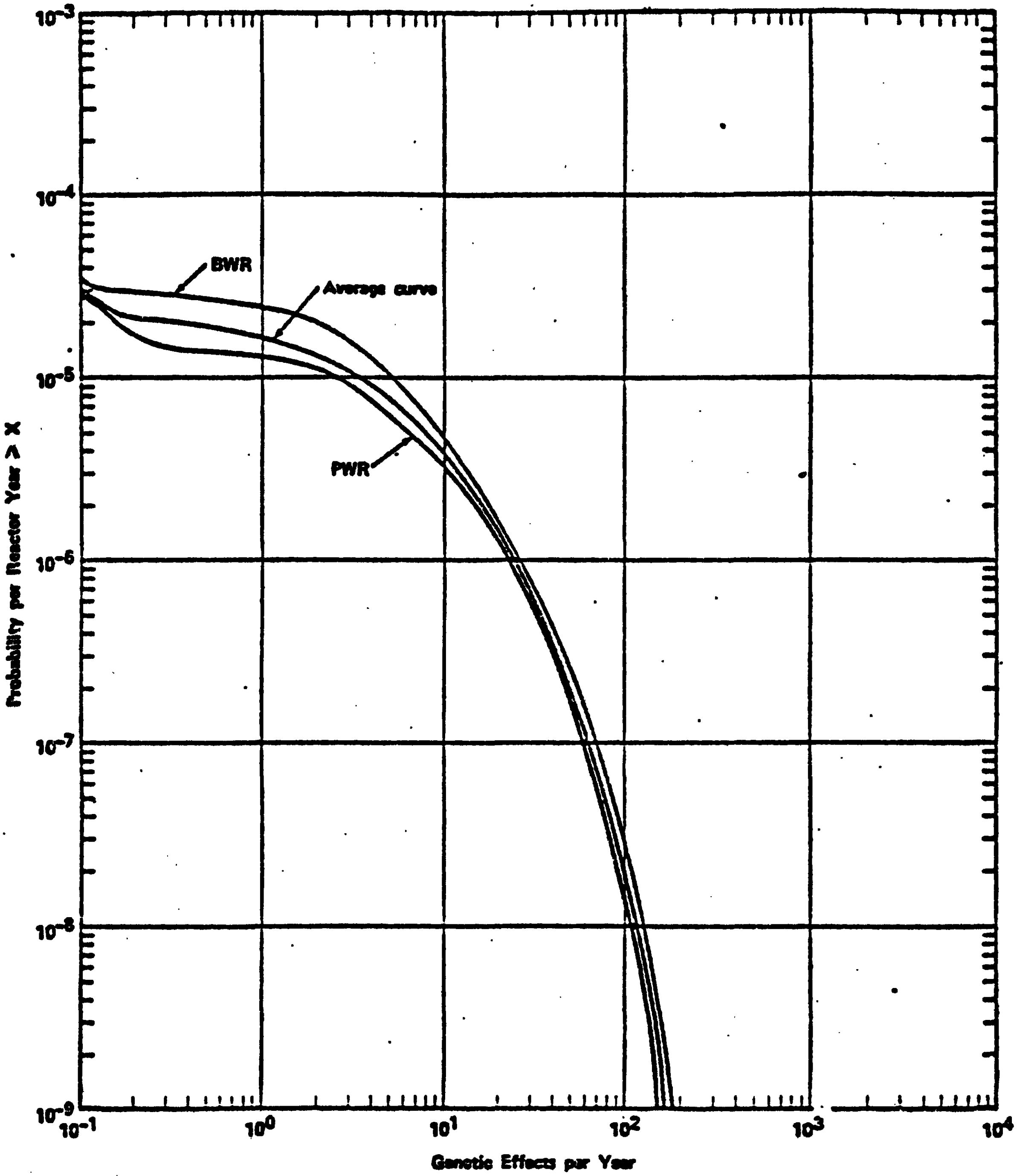
Probability Distribution for Thyroid Nodule Incidence per Reactor Year

Notes: 1. Approximate uncertainties are estimated to be represented by factors of 1/3 and 3 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

2. PWR and BWR are nearly identical.

FIGURE D-4

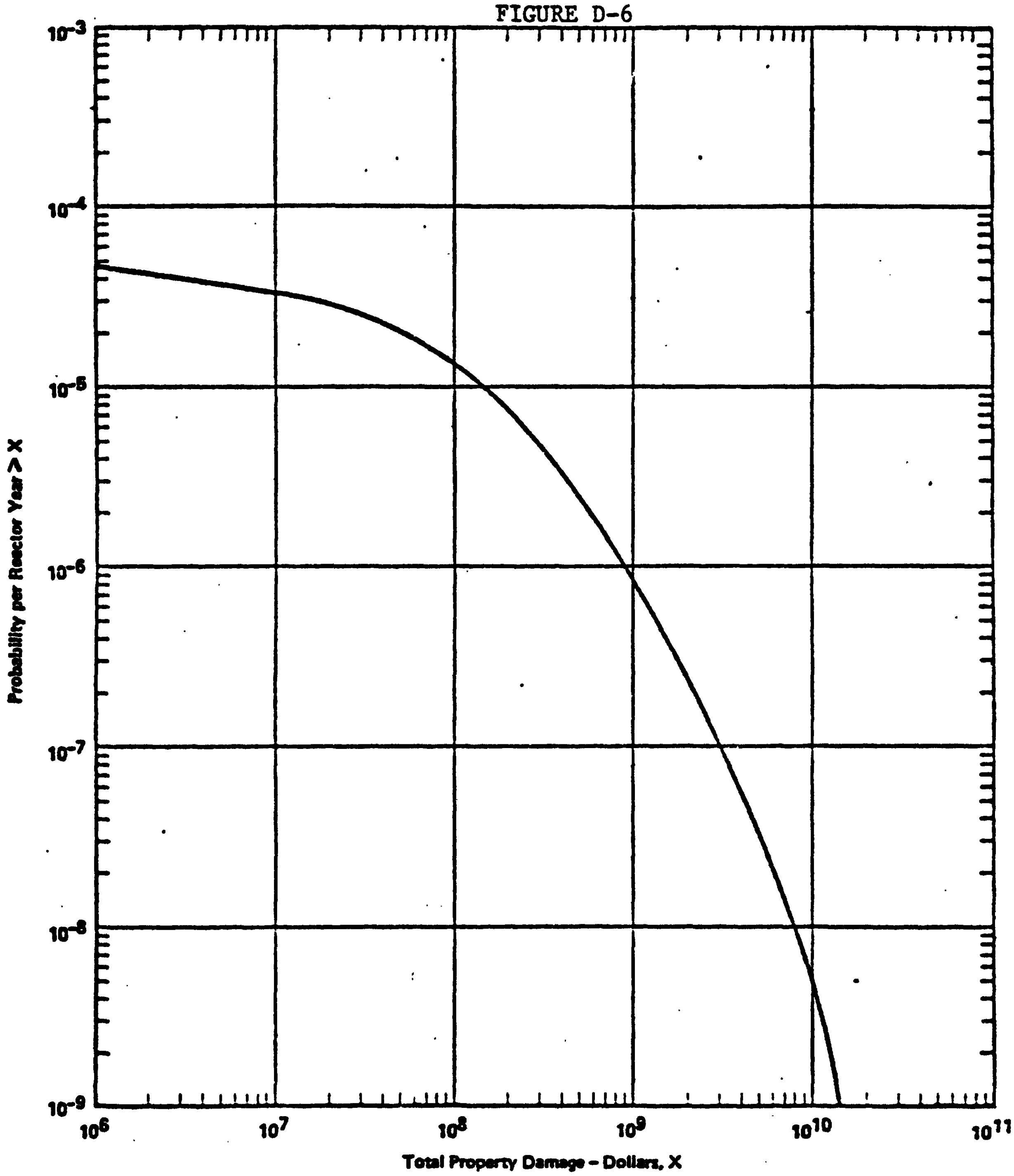
FIGURE D-5



Probability Distribution for Incidence of Genetic Effects per Reactor Year

Note: Approximate uncertainties are estimated to be represented by factors of 1/3 and 6 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

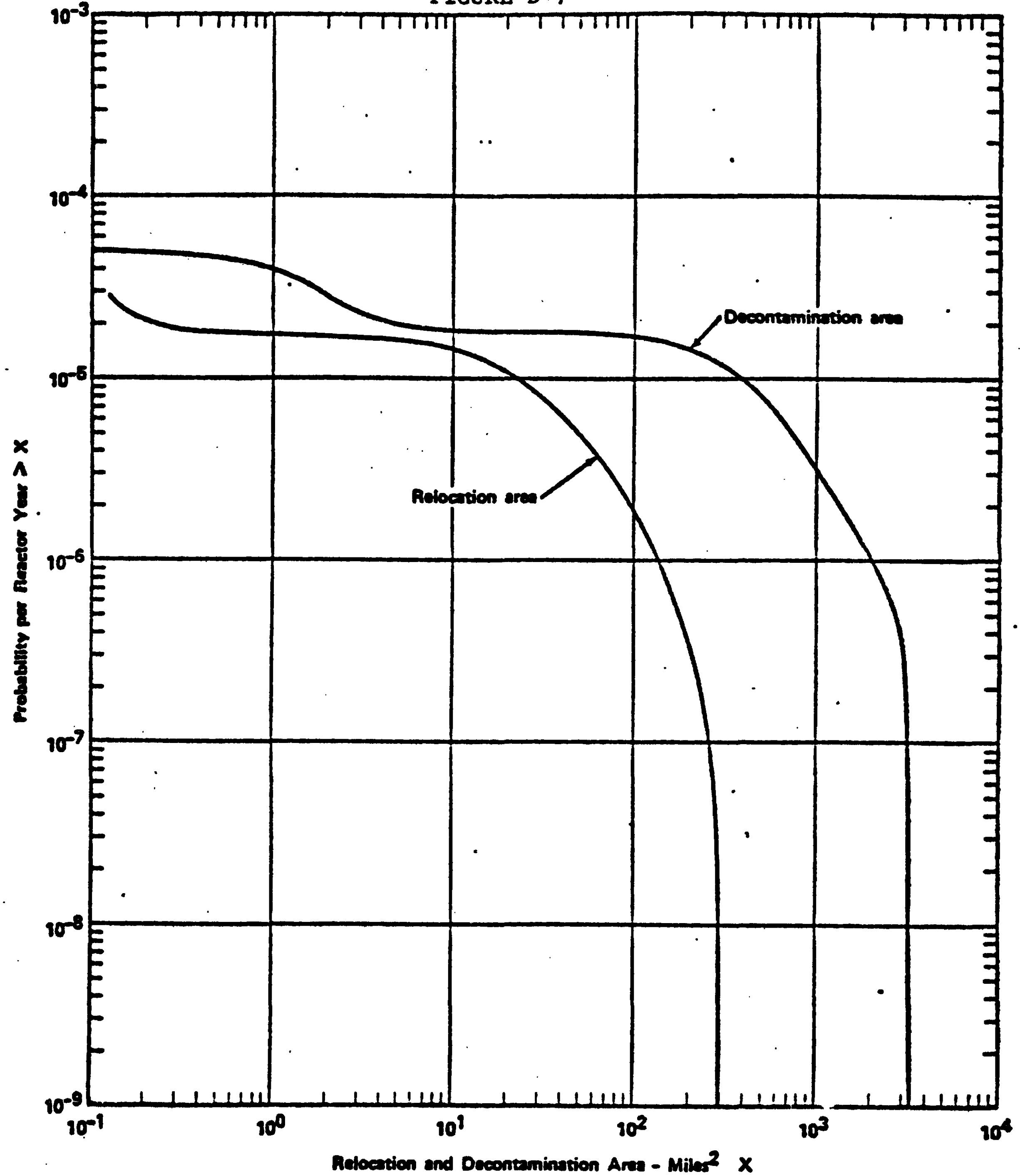
FIGURE D-6



Probability Distribution for Property Damage per Reactor Year

Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

FIGURE D-7



Probability Distribution for Relocation and Decontamination Area per Reactor Year

Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

APPENDIX E

Uncertainty Analysis

Finding release category probability distributions from system failure probability distributions cannot usually be done using closed form mathematical expressions with the characteristics of a distribution. For this reason Monte Carlo methods are employed for calculations of this type. A Monte Carlo calculation involves random sampling of the input distributions to generate point values, followed by a calculation of a point value for the top event, in this case a release category probability. The calculation is repeated many times and the results are stored to construct a histogram which will accurately represent the actual distribution.

In this analysis the code PLMODMC is used for Monte Carlo calculations of release category probabilities and, in one case, latent cancer fatility probabilities. The PLMODMC code uses a fast PL-1 random number generator, as well as routines developed for other Monte Carlo analysis codes, such as SAMPLE, which was used in the RSS. An example of the output of the code may be found in Table E-1. A listing of the input is given, followed by a point value for the top event. Then the m·dian, the 5% and 95% confidence limit error factors, and the histogram resulting from the calculation are shown. Values characterizing the accuracy of the Monte Carlo analysis, the minimum probability, and the maximum error are also given for each confidence level in the histogram.

The PLMODMC cose uses only the lognormal probability distribution. The characteristics of that distribution are as follows:

The probability density function (PDF) is

$$f(t) = \frac{1}{\sqrt{2\pi\sigma}} \exp - \left[\frac{(\ln t-\mu)^2}{2\sigma^2}\right] \mathcal{E} > 0$$

Mode (the most probable value) $= t_m = e^{\mu - \sigma^2}$

Median: $\mathcal{E}_{0.5} = e^{\mu}$ or $\mathcal{E}_{0.5} = \sqrt{X_U \cdot X_L}$

where X_U, X_L are upper and lower bounds, respectively.

Mean: $\bar{\mathcal{E}} = e^{\mu+\sigma^2/2}$

Variance: $V = e^{2\mu+\sigma^2}\left[e^{\sigma^2} - 1\right]$

By providing the code with median values and the error factor (which is the factor by which the upper and lower bounds differ from the median), the code will calculate the necessary parameters to describe the PDF for that input's distribution. Together with the Boolean equation for the top event probability, the code will generate an approximation of the top event PDF, after many calculations are performed.

The value of any Monte Carlo analysis is determined by its accuracy. The process of sampling for estimating distributions is well studied. Methods for approximating the accuracy of a sampling process can be found in Ref. 20. The results found there indicate that

$$pr(|X(P)-P \leq \mathcal{E}) = \text{erf}\left(\frac{t}{\sqrt{2}}\right) + R$$

where

$$X(P) = m/N$$

N is the number of trials and m is the number of successes

$$t = \varepsilon\sqrt{\frac{N}{pq}}$$

p is the probability of success from the binomial distribution

q is the probability of failure = 1-p

R is the error associated with the probability measure and is given by

$$|R| \leq \frac{e^{-t^2/2}}{\sqrt{2\pi Npq}} + \frac{0.2+0.25|p-q|}{Npq} + e^{-3/2\sqrt{Npq}}$$

erf(t) is the error function.

The symbol $pr(|X(P)-P \leq \varepsilon)$ represents the probability that the confidence limit of P lies between the confidence interval $P \pm \varepsilon$. This value is independent of the distribution and only dependent on the confidence level. Given a large sample size N these reduce to:

$$pr(|t_{est}-t_{exact}| \leq \frac{1.36}{\sqrt{N}}) = 0.95$$

where t_{est} is the estimated distribution and t_{exact} is the exact distribution. In this analysis N is always equal to two thousand. This translates to an accuracy such that one can be 95 percent sure that the estimated distribution differs from the exact distribution by not more than a .03 confidence interval.

The inaccuracies inherent in a sample size of two thousand indicate that, in some cases presented in the results, actual changes in medians, upper and lower bounds, and error factors can be related to sampling error rather than changes in the actual distributions. This may particularly be the case where the ratios of the median values change by more than the ratios of the upper bounds or 95% confidence limit error factors . Considering that the ratio of two factors of about the same magnitude and error gives a possible error of approximately twice the individual factor errors, this may easily explain some cases. Given that the most important results involve large changes in medians and upper bounds, these accuracies pose little problem.

The base cases used for the PWR analysis are contained in Table E-1 through Table E-3. They represent release categories 1 through 3. The equations used to calculate the top event probability for the release categories are represented cryptically in Table E-4. Table E-5 defines the system that each number represents. The base cases of the BWR are presented in Table E-6 and Table E-7. Table E-6 represents release category 1 and 3, since they are merely multiples of each other. Table E-7 represents release category 2. The equations for each of the three categories are found in Table E-8, and the associated system definitions in Table E-9.

TABLE E-1

BASE CASE FOR PWR RELEASE CATEGORY 1

CALCULATION OF THE MEDIAN POINT VALUE
NUM FREE EVENT INPUTS = 22
NUM REPLICATED EVENT INPUTS = 0

FREE INPUT	MEDIAN VALUE		SPEAD
1	1.000000	E-04	10
2	3.000000	E-04	10
3	1.000000	E-03	10
4	1.000000	E-07	10
5	4.000000	E-06	10
6	1.000000	E+01	2
7	1.000000	E-02	10
8	3.700000	E-05	8
9	3.600000	E-05	4
10	1.000000	E-02	10
11	2.400000	E-03	4
12	1.000000	E-04	9
13	9.500000	E-04	2
14	4.700000	E-03	2
15	8.600000	E-03	3
16	8.300000	E-03	2
17	1.300000	E-02	3
18	8.500000	E-05	4
19	1.000000	E-05	10
20	2.000000	E-01	2
21	2.000000	E-01	3
22	1.500000	E-04	3

TOP EVENT MEDIAN PROBABILITY = 2.740954E-06

TOTAL NUM OF TRIALS = 2000

MEAN PROB = 2.297413E-07 STANDARD DEVIATION = 5.234100E-07 MEDIAN PROB = 1.098858E-07

ERROR FACTOR (5%) = 4.48197 ERROR FACTOR (95%) = 6.12725

CONFIDENCE LEVEL	PROBABILITY	MAX ERROR	MIN PROBABILITY
0.50	1.317781 E-08	5.461876 E-02	9.438580 E-01
1.00	1.501728 E-08	3.091324 E-02	9.444679 E-01
2.50	1.989360 E-08	2.949033 E-02	8.184315 E-01
5.00	2.479389 E-08	2.865501 E-02	6.664432 E-01
10.00	3.367395 E-08	2.474575 E-02	5.191976 E-01
15.00	4.156787 E-08	2.200509 E-02	4.468271 E-01
20.00	4.938266 E-08	2.016928 E-02	4.036806 E-01
30.00	6.658644 E-08	1.799588 E-02	5.564188 E-01
40.00	8.548526 E-08	1.692864 E-02	3.349944 E-01
50.00	1.098858 E-07	1.654343 E-02	3.287856 E-01
60.00	1.377680 E-07	1.692864 E-02	3.349944 E-01
70.00	1.800181 E-07	1.799588 E-02	3.564188 E-01
80.00	2.537698 E-07	2.016928 E-02	4.036806 E-01
85.00	3.081599 E-07	2.200509 E-02	4.468271 E-01
90.00	4.296943 E-07	2.474574 E-02	5.191976 E-01
95.00	6.784293 E-07	2.865501 E-02	6.664432 E-01
97.50	1.171866 E-06	2.949033 E-02	8.184315 E-01
99.00	1.783976 E-06	3.091334 E-02	9.444769 E-01
99.50	1.836953 E-06	5.461876 E-02	9.438580 E-01

TABLE E-2

BASE CASE FOR PWR RELEASE CATEGORY 2

CALCULATION OF THE MEDIAN POINT VALUE
NUM FREE EVENT INPUTS = 22
NUM REPLICATED EVENT INPUTS = 0

FREE INPUT	MEDIAN VALUE	SPEAD
1	1.000000 E-04	10
2	3.000000 E-04	10
3	1.000000 E-03	10
4	1.000000 E-07	10
5	4.000000 E-06	10
6	1.000000 E+01	2
7	1.000000 E-02	10
8	3.700000 E-05	8
9	3.600000 E-05	4
10	1.000000 E-02	10
11	2.400000 E-03	4
12	1.000000 E-04	9
13	9.500000 E-04	2
14	4.700000 E-03	2
15	8.600000 E-03	3
16	8.300000 E-03	2
17	1.300000 E-02	3
18	8.500000 E-05	4
19	1.000000 E-05	10
20	2.000000 E-01	2
21	2.000000 E-01	3
22	1.500000 E-04	3

TOP EVENT MEDIAN PROBABILITY = 6.050860E-08

TOTAL NUM OF TRIALS = 2000

MEAN PROB = 2.47568E-05 STANDARD DEVIATION = 4.688788E-05 MEDIAN PROB = 1.30465E-05

ERROR FACTOR (5%) = 4.46874 ERROR FACTOR (95%) = 6.12571

CONFIDENCE LEVEL	PROBABILITY	MAX ERROR	MIN PROBABILITY
0.50	1.486187 E-06	5.461876 E-02	9.438580 E-01
1.00	1.899938 E-06	3.091324 E-02	9.444679 E-01
2.50	2.380315 E-06	2.949033 E-02	8.184315 E-01
5.00	2.919511 E-06	2.865501 E-02	6.664432 E-01
10.00	3.851987 E-06	2.474575 E-02	5.191976 E-01
15.00	4.901623 E-06	2.200509 E-02	4.468271 E-01
20.00	6.013522 E-06	2.016928 E-02	4.036806 E-01
30.00	7.977299 E-06	1.799588 E-02	5.564188 E-01
40.00	1.051710 E-05	1.692864 E-02	3.349944 E-01
50.00	1.304655 E-05	1.654343 E-02	3.287856 E-01
60.00	1.675435 E-05	1.692864 E-02	3.349944 E-01
70.00	2.215365 E-05	1.799588 E-02	3.564188 E-01
80.00	3.041438 E-05	2.016928 E-02	4.036806 E-01
85.00	3.689158 E-05	2.200509 E-02	4.468271 E-01
90.00	4.986575 E-05	2.474574 E-02	5.191976 E-01
95.00	7.991919 E-05	2.865501 E-02	6.664432 E-01
97.50	1.168213 E-04	2.949033 E-02	8.184315 E-01
99.00	1.799959 E-04	3.091334 E-02	9.444769 E-01
99.50	2.652712 E-04	5.461876 E-02	9.438580 E-01

TABLE E-3

BASE CASE FOR PWR RELEASE CATEGORY 3

CALCULATION OF THE MEDIAN POINT VALUE

NUM FREE EVENT INPUTS = 22

NUM REPLICATED EVENT INPUTS = 0

FREE INPUT	MEDIAN VALUE	SPEAD
1	1.000000 E-04	10
2	3.000000 E-04	10
3	1.000000 E-03	10
4	1.000000 E-07	10
5	4.000000 E-06	10
6	1.000000 E+01	2
7	1.000000 E-02	10
8	3.700000 E-05	8
9	3.600000 E-05	4
10	1.000000 E-02	10
11	2.400000 E-03	4
12	1.000000 E-04	9
13	9.500000 E-04	2
14	4.700000 E-03	2
15	8.600000 E-03	3
16	8.300000 E-03	2
17	1.300000 E-02	3
18	8.500000 E-05	4
19	1.000000 E-05	10
20	2.000000 E-01	2
21	2.000000 E-01	3
22	1.500000 E-04	3

TOP EVENT MEDIAN PROBABILITY = 7.880222E-06

TOTAL NUM OF TRIALS = 2000

MEAN PROB = 9.988090E-06 STANDARD DEVIATION = 3.074563E-05 MEDIAN PROB = 4.0826E-06

ERROR FACTOR (5%) = 6.24986 ERROR FACTOR (95%) = 8.89584

CONFIDENCE LEVEL	PROBABILITY	MAX ERROR	MIN PROBABILITY
0.50	2.821983 E-07	5.461876 E-02	9.438580 E-01
1.00	3.455407 E-07	3.091324 E-02	9.444679 E-01
2.50	4.663371 E-07	2.949033 E-02	8.184315 E-01
5.00	6.542655 E-07	2.865501 E-02	6.664432 E-01
10.00	9.478299 E-07	2.474575 E-02	5.191976 E-01
15.00	1.280559 E-06	2.200509 E-02	4.468271 E-01
20.00	1.545484 E-06	2.016928 E-02	4.036806 E-01
30.00	2.147491 E-06	1.799588 E-02	5.564188 E-01
40.00	2.952665 E-06	1.692864 E-02	3.349944 E-01
50.00	4.082656 E-06	1.654343 E-02	3.287856 E-01
60.00	5.597488 E-06	1.692864 E-02	3.349944 E-01
70.00	7.520596 E-06	1.799588 E-02	3.564188 E-01
80.00	1.154035 E-06	2.016928 E-02	4.036806 E-01
85.00	1.558878 E-05	2.200509 E-02	4.468271 E-01
90.00	2.172468 E-05	2.474574 E-02	5.191976 E-01
95.00	3.631886 E-05	2.865501 E-02	6.664432 E-01
97.50	5.808714 E-05	2.949033 E-02	8.184315 E-01
99.00	8.113575 E-05	3.091334 E-02	9.444769 E-01
99.50	1.163627 E-04	5.461876 E-02	9.438580 E-01

TABLE E-4

Equations for calculation of PWR release category point values from system failure rates and event trees

The numbers in these equations correspond to those in Table E-5.

PWR release category 1

$$C_1*((\ 1 + 2 + 3) * (12 + 18) + 3 * 11 + 4 * 11$$

$$+ (6 * 7 * 8 + 20 * 21 * 22 * C_2))$$

PWR release category 2

$$5 + 4 * (11 * C_4 + 12 * C_5) +$$
$$(6 * 7 * 8 + 20 + 21 + 22) * C_3$$

PWR release category 3

$$C_6 + ((1 + 2 + 3) * 12 + 18) + 3 * 11)$$

$$+ C_7 * (\ 4 + 1 * (\ 13 + 14) + 2 * (13 + 15) + 3 * 15$$

$$+ 1 * 16 + (\ 2 + 3\ ) * 17 + 6 * (\ 7 + 8 + 9 * 10 *$$

$$(1.0 + 7))$$

TABLE E-5

Key to the equations in Table E-4

Also the values of median and error factor for the system failure rate probability distributions (base cases)

# from equation in Table E-4	RSS Acronym	Median Value	Error Factor
1	A	1×10^{-4}	10
2	S1	3×10^{-4}	10
3	S2	9×10^{-4}	10
4	R	1×10^{-7}	10
5	V	4×10^{-6}	10
6	T	10	2
7	M	1×10^{-2}	10
8	L	3.7×10^{-5}	8
9	K	3.6×10^{-5}	4
10	Q	1×10^{-2}	10
11	C	2.4×10^{-3}	4
12	F	1×10^{-4}	9
13	ACC	9.5×10^{-4}	2
14	LPIS	4.7×10^{-3}	2
15	HPIS	8.6×10^{-3}	3
16	LPRS	9.0×10^{-3}	3
17	HPRS	1.3×10^{-2}	3

TABLE E-5 (CONT.)

# from equation in Table E-4	RSS Acronym	Median Value	Error Factor
18	G	8.5×10^{-5}	4
19	B	1×10^{-5}	4
20	TLOOS	.2	2
21	MLOOS	2×10^{-1}	3
22	LLOOS	1.5×10^{-4}	3
C_1	ALPHA 1	.005	
C_2	(ALPHA/ALPHA1)*B[1]	1.0	
C_3	(GAMMA + DELTAT)*B[1]	0.4	
C_4	GAMMA + DELTA	1.235	
C_5	DELTA	.995	
C_6	DELTA	.995	
C_7	ALPHA	.01	

TABLE E-6

BASE CASE FOR BWR RELEASE CATEGORY 1 and 3

NUM FREE EVENT INPUTS = 9

NUM REPLICATED EVENT INPUTS = 0

FREE INPUT	MEDIAN VALUE	SPREAD
1	1.000000 E+01	2
2	1.600000 E-06	10
3	1.300000 E-06	4
4	1.000000 E-02	10
5	7.800000 E-03	4
6	3.000000 E-03	3
7	2.000000 E-01	2
8	4.600000 E-06	4
9	2.000000 E-01	3

REP INPUT MEDIAN VALUE SPREAD

TOP EVENT MEDIAN PROBABILITY = 5.659865E-06

THE MONTE_CARLO SIMULATION STARTS NOW

TOTAL NUM OF TRIALS = 2000

MEAN PROB = 8.335788E-07 STANDARD DEVIATION = 1.302139E-06 MEDIAN PROB = 4.740165E-07

ERROR FACTOR (5%) = 3.97215 ERROR FACTOR (95%) = 5.66739

CONFIDENCE LEVEL	PROBABILITY	MAX ERROR	MIN PROBABILITY
0.50	5.614859 E-08	5.461876 E-02	9.438580 E-01
1.00	6.886103 E-08	3.091324 E-02	9.444679 E-01
2.50	9.429004 E-08	2.949033 E-02	8.184315 E-01
5.00	1.199185 E-07	2.865501 E-02	6.664432 E-01
10.00	1.594424 E-07	2.474575 E-02	5.191976 E-01
15.00	1.948193 E-07	2.200509 E-02	4.468271 E-01
20.00	2.229978 E-07	2.016928 E-02	4.036806 E-01
30.00	3.053918 E-07	1.799588 E-02	5.564188 E-01
40.00	3.844935 E-07	1.692864 E-02	3.349944 E-01
50.00	4.760165 E-07	1.654343 E-02	3.287856 E-01
60.00	6.166870 E-07	1.692864 E-02	3.349944 E-01
70.00	7.730288 E-07	1.799588 E-02	3.564188 E-01
80.00	1.072692 E-06	2.016928 E-02	4.036806 E-01
85.00	1.334735 E-06	2.200509 E-02	4.468271 E-01
90.00	1.755578 E-06	2.474574 E-02	5.191976 E-01
95.00	2.697770 E-06	2.865501 E-02	6.664432 E-01
97.50	3.647563 E-06	2.949033 E-02	8.184315 E-01
99.00	5.653026 E-06	3.091334 E-02	9.444769 E-01
99.50	7.932585 E-06	5.461876 E-02	9.438580 E-01

TABLE E-7

BASE CASE FOR BWR RELEASE CATEGORY 2

NUM FREE EVENT INPUTS = 9

NUM REPLICATED EVENT INPUTS = 0

FREE INPUT	MEDIAN VALUE	SPREAD
1	1.000000 E+01	2
2	1.600000 E-06	10
3	1.300000 E-06	4
4	1.000000 E-02	10
5	7.800000 E-03	4
6	3.000000 E-03	3
7	2.000000 E-01	2
8	4.600000 E-06	4
9	2.000000 E-01	3

REP INPUT MEDIAN VALUE SPREAD

TOP EVENT MEDIAN PROBABILITY = 3.317997E-07

THE MONTE_CARLO SIMULATION STARTS NOW

TOTAL NUM OF TRIALS = 2000

MEAN PROB = 1.2197958E-05 STANDARD DEVIATION = 2.488999E-05 MEDIAN PROB = 5.65984E-06

ERROR FACTOR (5%) = 5.81489 ERROR FACTOR (95%) = 7.76926

CONFIDENCE LEVEL	PROBABILITY	MAX ERROR	MIN PROBABILITY
0.50	1.7369888 E-07	5.461876 E-02	9.438580 E-01
1.00	4.934289 E-07	3.091324 E-02	9.444679 E-01
2.50	7.934289 E-07	2.949033 E-02	8.184315 E-01
5.00	9.759711 E-07	2.865501 E-02	6.664432 E-01
10.00	1.179500 E-06	2.474575 E-02	5.191976 E-01
15.00	1.988588 E-06	2.200509 E-02	4.468271 E-01
20.00	2.635019 E-06	2.016928 E-02	4.036806 E-01
30.00	3.077576 E-06	1.799588 E-02	5.564188 E-01
40.00	4.182261 E-06	1.692864 E-02	3.349944 E-01
50.00	5.669845 E-06	1.654343 E-02	3.287856 E-01
60.00	7.448787 E-06	1.692864 E-02	3.349944 E-01
70.00	1.028484 E-05	1.799588 E-02	3.564188 E-01
80.00	1.532857 E-05	2.016928 E-02	4.036806 E-01
85.00	1.945048 E-05	2.200509 E-02	4.468271 E-01
90.00	2.795192 E-05	2.474574 E-02	5.191976 E-01
95.00	4.485048 E-05	2.865501 E-02	6.664432 E-01
97.50	1.032510 E-04	2.949033 E-02	8.184315 E-01
99.00	1.554879 E-04	3.091334 E-02	9.444769 E-01
99.50		5.461876 E-02	9.438580 E-01

TABLE E-8

Equations for calculation of BWR release category point values from system failure rates and event trees

The numbers in the equations correspond to those in Table E-9

BWR release category 1 and category 3

$$(1 * (2 + 3 + 4 * 5 * 6)+(7 * 8 * 9))* C_{1,3}$$

BWR release category 2

$$(1 * (2 + 4 * 5 * 6)+(7 * 8 * 9))*C_2$$

TABLE E-9

Key to the equations in Table E-8

Also the values of median and error factors

For the system failure rate probability distributions

(base cases)

#'s from Equations in Table E-4	RSS Acronyms	Median Value	Error Factor
1	T	10	2
2	W	1.6×10^{-6}	10
3	C	1.3×10^{-6}	4
4	Q	1×10^{-2}	10
5	U	7.8×10^{-3}	4
6	V	3×10^{-3}	3
7	TLOOS	2×10^{-1}	2
8	WLOOS	4.6×10^{-6}	4
9	QLOOS	2×10^{-1}	3
C_1	ALPHA	.01	
C_2	GAMMAP	.198	
C_3	GAMMA	.972	

References

1. Lewis, H.W., chairman, et.al., "Risk Assessment Review Group report to the U.S. Nuclear Regulatory Commission NUREG/CR-0400. Spetember 1978.
2. Portal, R. and A. Villemeur. "A Probabilisitic Approach to Safety Assessment and Design Optumization of Two PWR Safety Related Systems." 1977.
3. "Human Factors Review of Nuclear Power Plant Control Room Design," Summary Report, EPRI-NP-309-SY, 1976.
4. "Potential for the Reduction of Core Melt Probability," E.P. Epler, 1977.
5. "The Effect of Human Error on the Availability of Periodically Inspected Redundant Systems," PB-261-840, 1977.
6. Carlson, David D., Jack W. Hickman, "A Value Impact Assessment of Alternate Containment Concepts," Sandia Laboratories. NUREG/CR-0615. June 1978.
7. Cybulskis, et.al., "Effect of Containment Venting on the Risk of LWR Melt Down Accidents." NUREG/CR-0138. May 1978.
8. Barsell, A.W. And Orvis, D.D., "Containment Design Options for the HIGR: An Application of Probability Risk Assessment." 6A-A14970. May 1978.
9. Personal Conversation. Marc Cunningham. NRC. Octorber 1978.
10. Modarres, Mohammad and Lothar Wolf. "PL-MODT Two Codes for for Reliabilty and Availability Analysis of Complex Technical Systems Using the Fault Tree Modularization Technique." Department of Nuclear Engineering, MIT Report, MITNE-222, November 1978.
11. Kelly, J.E., Leverenz, F.L., McCormick, N.J., and Erdmann, R.C. (SAI) "Sensitivity Assessments in Reactor Safety Analysis." Nuclear Technology, Vol. 32, February 1977.
12. Personal Conversation with William Vesely, Jr., Nuclear Regulatory Commission, Probabilistic Analysis Staff, June 1978.
13. Carlson, op.cit.,
14. Cybulskis, P., Wooten, R.O., and Denning, R.S. "Effect of Containment Venting on the Risk of LWR Meltdown Accidents." Battelle Columbus Laboratories, NUREG/CR-0138, May 1978.
15. Reactor Safety Study, Appendix VI, Calculation of Reactor Accident Consequences, Section 5 and 10. NUREG-75/014, October 1975.

16. Carlson, _op. cit.,_ p. 18.

17. Reactor Safety Study, Main Report, October 1975.

18. Reactor Safety Study, Appendix VI, October 1975.

19. Calculations of Reactor Accident Consequences (CRAC) Computer Code User's Manual, Draft, 1977.

20. David K. Lloyd and Myron Lipow, _Reliability, Management, Methods, and Mathematics,_ 2nd. Edition (1977.)

21. Modarres, Mohammad and Lothar Wolf, "PL-MODT and PL-MODMC, Two Codes for Reliability and Availability Analysis of Complex Technical Systems Using the Fault Tree Modularization Techniques," MITNE-222, November 1978.

22. "Mathematical Methods of Probabilistic Safety Analysis," G.E. Apostolalcic, UCLA-ENG-7464, September 1974.

23. "Reliability Technology," A.E. Green and A.J. Bouane, Wiley-Interscience, London, 1972.