PROBABILISTIC RISK ASSESSMENT
LIMERICK GENERATING STATION
PHILADELPHIA ELECTRIC COMPANY

VOLUME I

March 1981

Docket Nos. 50-352
50-353

EXECUTIVE SUMMARY

Background

At the request of the Nuclear Regulatory Commission, Philadelphia Electric Company, its contractor, the General Electric Company, and prime subcontractor, Science Applications, Inc., have performed a probabilistic risk assessment (PRA) of the Limerick Generating Station. The purpose of the analysis was to assess the risk of Limerick, specifically with regard to its location near a high population density area. These risks were evaluated to determine if they represent a disproportionately high segment of the total societal risk from postulated nuclear reactor accidents. The NRC requested that the Limerick analysis employ the methodology of the Reactor Safety Study (WASH-1400), with modifications to account for both the design differences and the site-specific differences between Limerick and the WASH-1400 reference plant and site. In addition, the NRC requested that recognition of the various criticisms of the WASH-1400 analysis approach and conduct (including those of the Lewis Committee) be considered in the Limerick Study.

The Limerick PRA

The Reactor Safety Study (WASH-1400) published in 1975 was the first thorough application of probabilistic methods to analyzing nuclear power plant risk. The Limerick analysis employs the basic approach and techniques utilized in the Reactor Safety Study, i.e., the use of fault trees and event tree logic models to quantify the probability of accident sequences. However, in response to NRC direction, the Limerick analysis accounted for a revised list of accident initiators based on the Limerick plant design and a more detailed analytical modeling of event sequences following each accident initiator. Plant-specific and site-specific data were also included in the analysis of the Limerick Mark II containment and in the meteorology and demography inputs to the evaluation of accident consequences. Criticisms of the WASH-1400 methodology (specifically those in the Risk Assessment Review Group Report to the NRC (NUREG/CR-0400)) were addressed in the Limerick PRA through a more detailed evaluation of accident sequences and sequence probabilities. Furthermore, there has been an attempt to apply the learning and experience gained over the six years since WASH-1400 to update and improve the risk assessment methodology. The most recent available data from operating experience was employed. Additionally, updated consequence evaluation analytical techniques were utilized. The Limerick analysis includes consideration and characterization of uncertainties in the results.

Results

The figure of merit employed in the Reactor Safety Study to quantify nuclear reactor risk is a graphical representation of the complementary cumulative distribution function (CCDF). The CCDF relates the expected frequency (number of accident events per year) to the consequence (e.g. number of early or latent fatalities).

Figure 1 presents the calculated CCDF for the Limerick site-specific analysis for early fatalities. Also shown in Figure 1 is the calculated CCDF for the composite site for the WASH-1400 BWR. For comparison, the CCDF curves estimated for total man-caused risk (excluding automobile crashes) and total natural risk to the population around the Limerick site are also plotted in Figure 1. In addition, the CCDF curve for airplane crashes causing early fatalities to humans on the ground is separated out from the total man-caused risk and presented. This component has been specifically included as an example of the risk to which the population is subjected without a conscious decision.

Figure 2 presents the calculated CCDF for both the Limerick site-specific case and the WASH-1400 BWR composite site case for latent fatalities due to radionuclide releases following low probability accident sequences.

Conclusions

A comparison of the CCDF curves generated in this study for the Limerick site with those presented in the reactor safety study leads to the following conclusions:

> The Limerick site-specific best estimate CCDF curves are slightly below the WASH-1400 curves for both early fatalities and latent fatalities for all calculated consequences.
>
> The Limerick CCDF for early fatalities is several orders of magnitude below the CCDF due to all natural and manmade risks as evaluated in WASH-1400.

Based upon this analysis, the Limerick Generating Station is not expected to represent a disproportionately high segment of the total societal risk from reactor accidents.

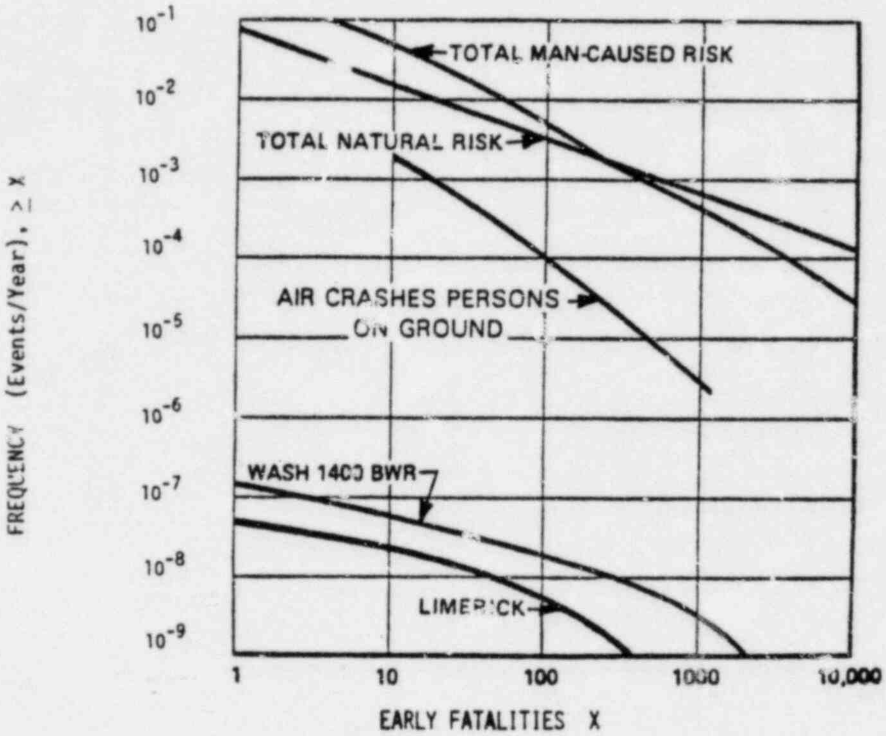

Figure 1 Limerick Risk Assessment
Limerick/WASH-1400 Risk Comparison

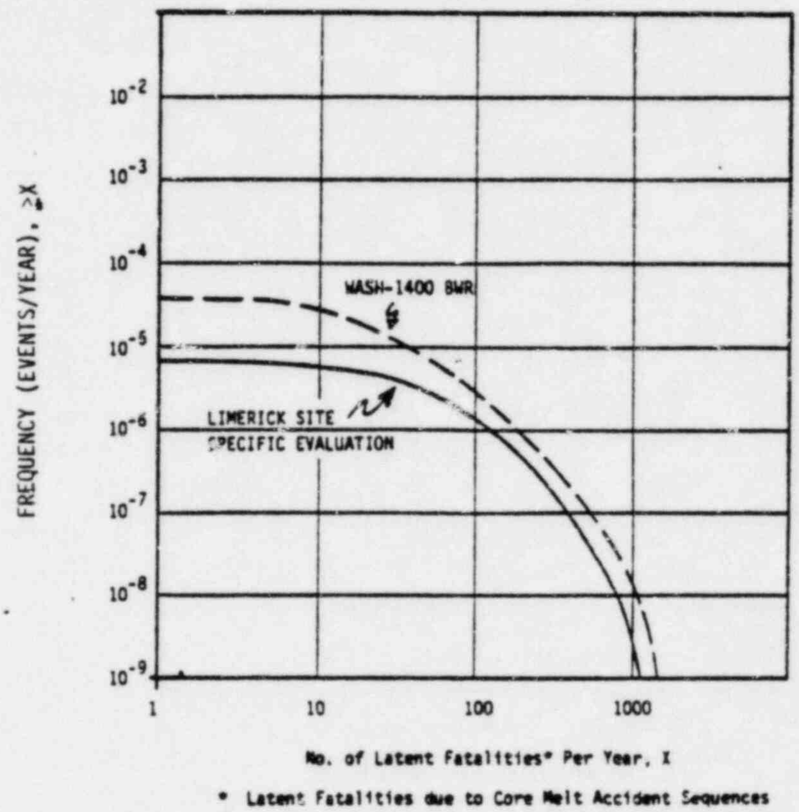

Figure 2 Comparison of the CCDF for Latent Fatalities for the Population Surrounding the LGS: WASH-1400 Composite Site BWR and the Limerick Site Specific Evaluation

LIMERICK GENERATING STATION PROBABILISTIC RISK ASSESSMENT

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Section 1

INTRODUCTION

The Limerick Generating Station (LGS) is a dual unit nuclear power plant with a General Electric boiling water reactor (BWR/4) and a Mark II containment. The plant is being constructed and will be operated by Philadelphia Electric Company. To confirm that the Limerick Generating Station does not represent a disproportionately high segment of societal risk from reactor accidents, a Probabilistic Risk Assessment (PRA) has been performed. Methods similar to those used in the Reactor Safety Study (WASH-1400) are used in the Limerick study. The following evaluations are performed as part of the Limerick PRA:

- Quantitative evaluation of accident initiators and the possible course of accidents following the initiators
- In-containment analysis of radioactive source terms possible following the accident sequences
- Containment analysis to assess mechanisms and potential locations for containment failure
- Offsite consequence analysis to determine the risk of plant operation to the general public.

1.1 BACKGROUND

A risk assessment of the Limerick Generating Station was requested by the Nuclear Regulatory Commission (NRC), in May, 1980 (1-1). In response to this request, the Philadelphia Electric Company (PECo) engaged the services of the General Electric Company (GE), the manufacturer of the nuclear steam supply system (NSSS). GE and prime subcontractor Science Applications, Inc. (SAI) performed the analysis. PECo also retained the services of NUS Corporation as a consultant, and requested that the Bechtel Power Corporation assist by performing an independent analysis of the containment and providing information on the balance of plant.

This report contains an updated Reactor Safety Study (WASH-1400) type of analysis, and also responds to the major criticisms of the methods used in that analysis; incorporates new information and modeling techniques; and makes use of experimental data wherever applicable.

In a manner similar to WASH-1400, the Limerick PRA is performed on a realistic basis, rather than a licensing or design basis. Equipment capability, success criteria, and event sequences are modeled realistically to determine, as accurately as possible, the expected course of events and conditions. Conservatisms are included only where deemed necessary due to uncertainties in input data, or the knowledge of physical effects or phenomena, and simplifications to bound the length of the study.

1.2 METHODOLOGY AND REPORT STRUCTURE

1.2.1 General Outline of Risk Analysis

In analyzing nuclear reactor risk, both the probability (likelihood) of an accident occurring, and the effects (consequences) of this occurrence to a population group must be assessed. The Reactor Safety Study (WASH-1400) presented their results as the probability (or frequency)* of exceeding a given level of consequences. Figure 1.1 outlines in simplified form, the tasks necessary for a probabilistic risk analysis.

*Probability and frequency are related as follows: Frequency is related to a sample of a population; whereas probability is the theoretical equivalent extended to the entire population.

The three major tasks are:

Task I -- Determination of the probability of radioactive release

Task II -- Determination of the magnitude of the radioactive releases for each unique accident sequence including the radioactive species and release time

Task III -- Determination of the consequences of a radioactive release to the environment or the public.

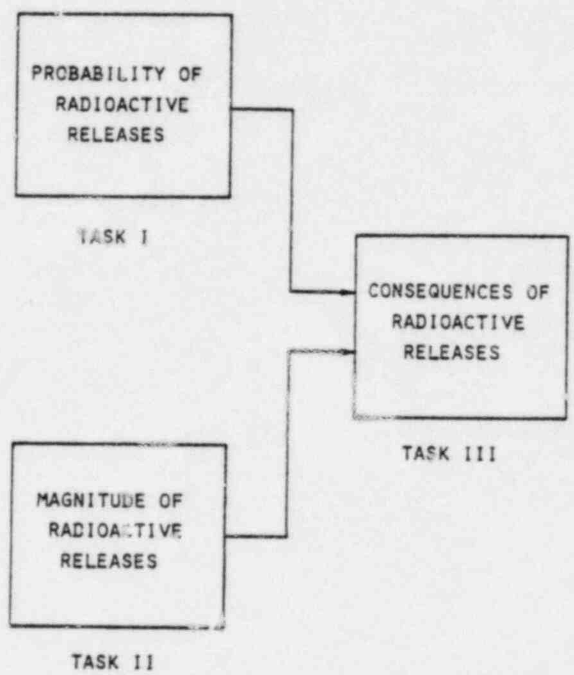

Figure 1.1 Major Tasks of a Probabilistic Risk Analysis of a Nuclear Reactor

After the probability and consequences of each accident sequence are determined for each accident or initiating event, the individual results are summed to obtain the overall probability versus consequences.

1.2.2 Methodology Outline and Report Structure

The application of probabilistic safety analysis to the evaluation of reactor systems provides a method of estimating the likelihood of all failures which can interfere with the ability of the safety systems to maintain a desired function such as providing adequate core cooling. The methodology used in the Limerick study is a fault tree analysis, similar to the approach used in the Reactor Safety Study (WASH-1400). Fault trees are a set of logic diagrams describing the potential equipment failure modes which could disable a system or group of systems. These logic diagrams are evaluated numerically by available computer programs.

Figure 1.1, is a general outline of PRA and is broken down into more detailed tasks in Figure 1.2. The analysis begins with the selection of initiators (a). Because of the "defense-in-depth", concept utilized in nuclear plant design, it is highly probable that a given accident sequence will be terminated before it can affect the public. Because of the complex multiple barriers to radioactive release that exist in nuclear units, diagrams are needed to show each accident sequence and the impact of the sequence. These diagrams are called event trees (b). The course of each sequence depends on the probabili-

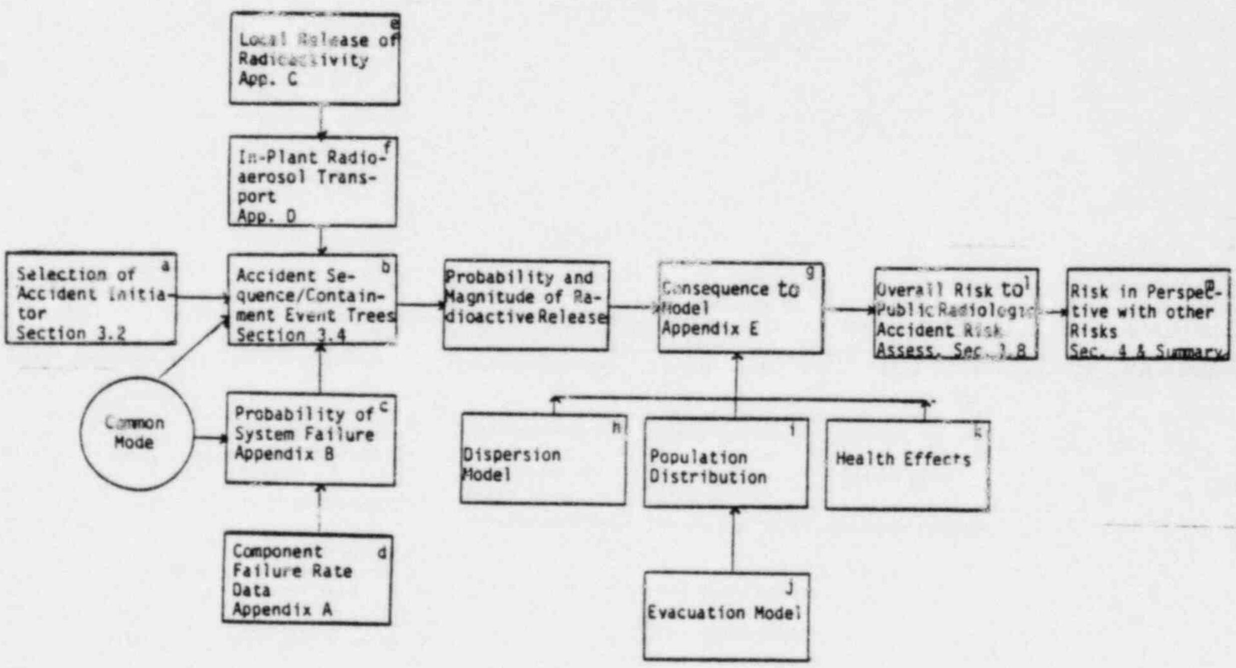

Figure 1.2 Simplified Task Diagram for the LGS Probabilistic Risk Assessment . Figure also indicates the sections of this report containing a discussion of the various analyses.

ties at each decision point in the event tree. To determine the probabilities at the decision points (nodes) of the event trees, fault trees (c) are prepared to model the system reliabilities. Component failure rates are obtained from the data base (d), and are inputs to the fault tree evaluations.

Accident consequences are evaluated by calculating the mass of radioactive material that can be released through each of the confining barriers (e, f) for specified accident sequences. Finally, the risk of a particular accident sequence is calculated for the plant site. This is done utilizing the consequence model (g) which is composed of a dispersion model (h) that calculates the radioactivity incident on the population distribution (i) modified for the mitigating effects of evacuation (j), and then calculating the health effects (k). These results are combined to produce an overall risk calculation by summing the risks of the individual accident sequences (l). The numerical results are then compared with other and more familiar risks (m).

Event Tree/Fault Tree Analysis (ET/FTA--boxes a, b, c, and d of Figure 1.2) is a formalized deductive analysis technique that provides a systematic approach to investigating the possible modes of occurrence of a defined system state, or undesired event. The fault tree model of a plant or system has been used as a logical method of displaying and quantifying component and system interrelationships. One of the principal benefits of ET/FTA is its ability to assess the reliability of redundant systems with multiple failures or system unavailabilities.

ET/FTA consists of two major steps: (1) the construction of the event trees and fault trees, and (2) their evaluation. The reliability of a combination of systems used to meet a desired goal is determined through the following steps:

- Understanding of the system interactions which will prevent the occurrence of an undesired consequence

- Defining the system functions and success criteria
- Reducing the system description (including controls and interfaces) to fault tree format
- Incorporating operating procedures into a fault tree model
- Combining test and maintenance procedures and technical specifications (limiting conditions for operation, surveillance requirements) into the analysis
- Quantification of the fault tree events
- Incorporating the probabilities into the event trees.

The event trees used in this study provide the basic tool for displaying the various accident sequences considered, and for relating the probabilities of radioactive release on a consistent basis. Probabilities for the events shown on the event tree are estimated by a system fault tree analysis to identify components or human interactions that may contribute to failures of systems and functions, and to quantify the probability of these system failures under accident conditions.

The event trees provide a framework for linking together the results of the fault tree analyses. Functional failure probabilities for a system are derived from evaluation of the applicable failure modes of the system, given the initiator and accident sequence. The same functional failure may have different probabilities for each type of initiating event or accident sequence. Determination of a failure probability to a system requires precise success criteria, and consideration of the conditions under which the system is called upon to perform.

After the probability of a radioactive release has been estimated, the next step (Task II from Figure 1.1) is to determine the magnitude of radioactive releases from the various sequences obtained in Task I. WASH-1400 succeeded in decoupling, to some degree, the release category calculations from the accident sequence quantification; thus allowing both tasks to proceed nearly in parallel (with some minimal exchange of information). A series of physical models represented in computer codes, as shown in Figure 1.3, were used for the evaluation of radioactive release categories. Basically, estimating the magnitude of a radioactive release is deterministic, in the sense that models are used to follow various accident sequences. Initial inputs required include the radioactive source term; core, reactor plant, and containment design; operational data; and information from the defined sequences of Task I, with regard to the characteristics of the off-normal condition which may lead to core damage. The Task II analysis proceeds by modeling those phenomena which lead to radioactive releases from fuel. The magnitude of the gaseous, vapor, and aerosol releases are calculated as they escape from the reactor system, and then from the containment. The result of Task II is a histogram of release frequency versus release magnitude, as shown in Figure 1.4, for each accident sequence.

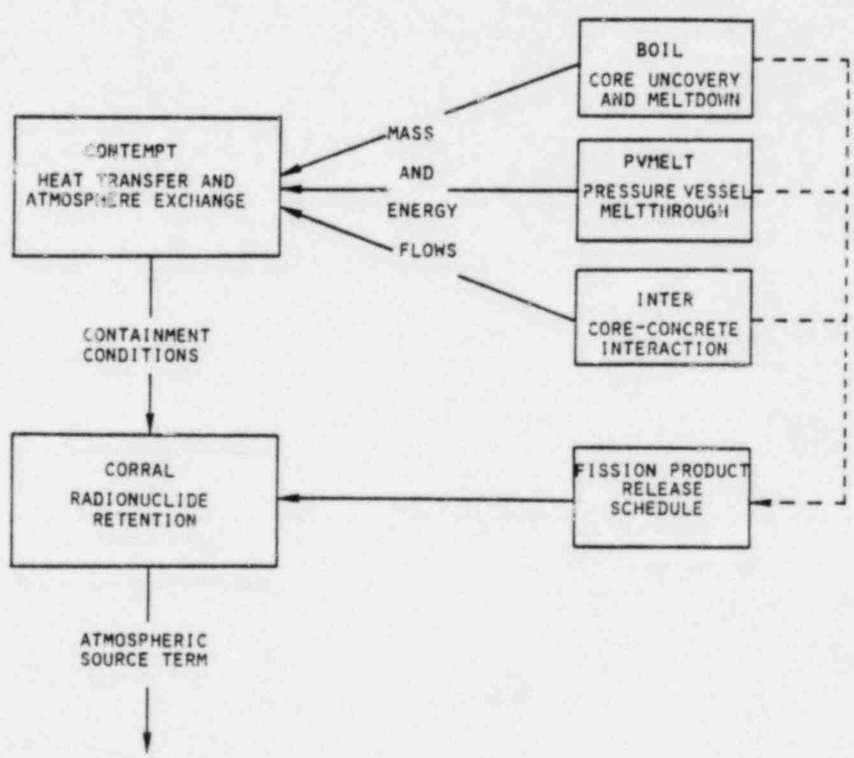

Figure 1.3 Code Networks for the Analysis of Release Categories and Their Magnitudes (Task II from Figure 1.1 or Boxes e and f from Figure 1.2).

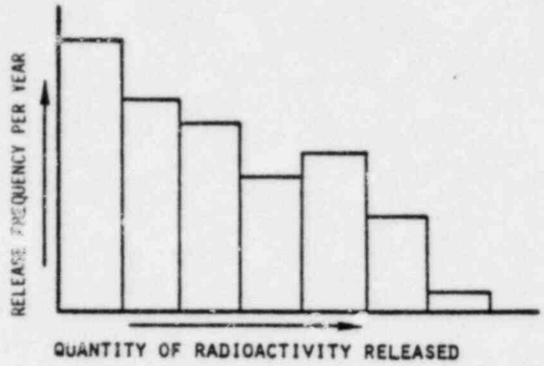

Figure 1.4 Histogram of Release Frequency Versus Release Magnitude

Radioactive releases are calculated for each of the unique accident sequences. The accident sequences define accident time histories, system involvement in accident sequences, and magnitude of radioactive releases.

The key parameters necessary to identify the accident sequence types to be used in the consequence evaluation are the following:

- Time of radioactive release
- Duration of the release
- Elevation of the release
- Energy release from containment
- Fraction of core inventory in release
- Time available for protection or evacuation of the public.

The calculation of the radioactivity released from the containment barrier for any key accident sequence requires input data obtained from various sources. These input data consist of the physical description of the containment and the building surrounding it, the physical phenomena occurring, and the amounts of radioactive materials released from the containment.

In order to determine a Limerick-unique set of release fractions, the following factors were evaluated as they apply to the Limerick BWR/4 Mark II system:

- Scrubbing by the pressure suppression pool
- Removal by filters
- Plateout on structural surfaces
- Resuspension
- Other deposition processes
- Radioactive decay.

The release fractions are influenced by the plant physical arrangement, containment type and size, primary system characteristics, and mitigating safety systems. These plant design characteristics are then coupled with actual reactor power, history, and the factors listed previously (time of release, duration of release, etc.) to perform the consequence calculation (i.e., the effects of the radioactive release on the environment). Figure 1.5 shows a schematic of the factors that are included in the consequence analysis of Task III. Key factors in Task III include the population exposed to the radionuclides, the shielding which they experience, and the meteorological conditions following the release.

Two categories of health effects are evaluated as a measure of consequence to the public, they are:

1. Early fatalities

2. Latent cancer fatalities.

Radiation transport and health effects from a containment failure and a given radioactive release category were calculated with the aid of the Calculation of Reactor Accidents Code (CRAC) (see Appendix E). The calculations were performed using an improved version of CRAC for site-specific risk calculations. These improvements include an improved chronic-health effects model which conserves radioactive inventory (feature not in the original CRAC or in most subsequent CRAC versions), and improved output routines providing for better analysis of results, including sensitivity studies where applicable.

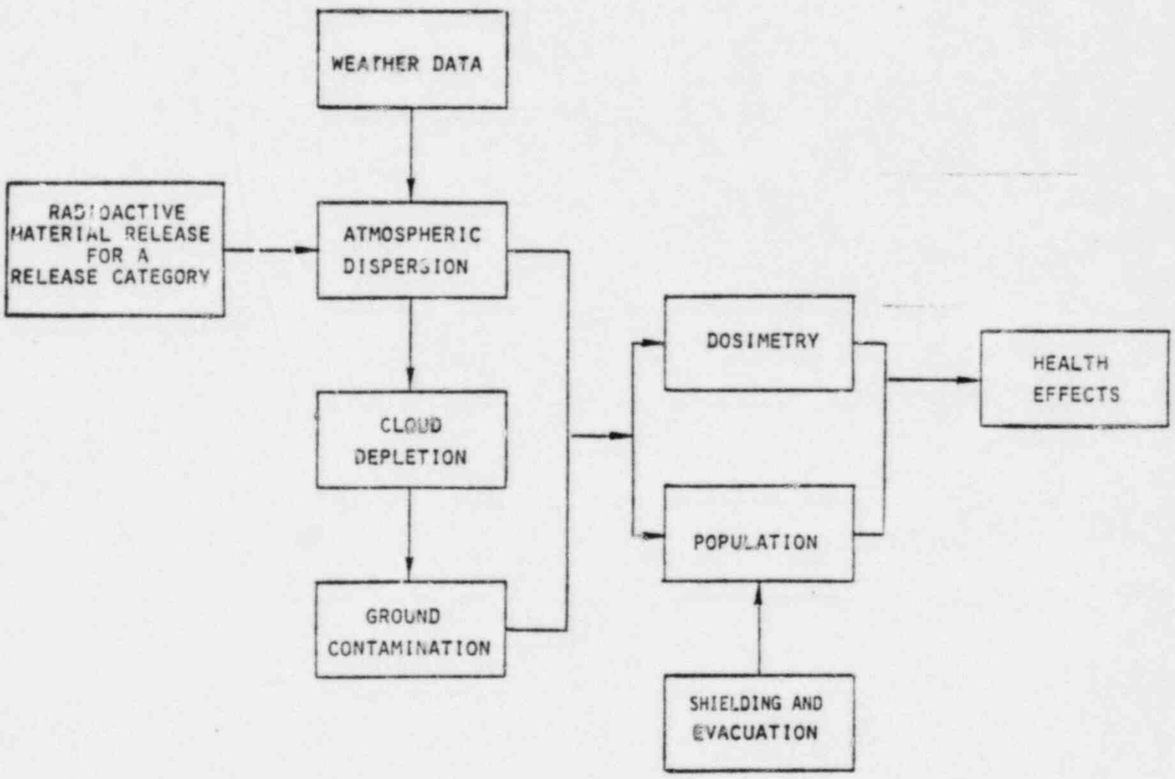

Figure 1.5 Schematic Diagram of Consequence Model

A site review was a major part of the analysis This consisted of a review of important weather conditions to determine important correlations between wind direction throughout a weather sequence. The population, either sheltered or evacuated, along specific evacuation routes was identified. A review of topological features was also made in conjunction with this review.

Finally, information from all three tasks shown in Figure 1.1 was assembled to present an evaluated risk of the Limerick plant in comparison with the original WASH-1400 BWR results. These comparisons are presented in Section 4.

1.3 RELATIONSHIP OF THIS STUDY TO THE REACTOR SAFETY STUDY

1.3.1 Adaptation of Reactor Safety Study Methodology

The Reactor Safety Study (RSS) (1-2) was a thorough application of probabilistic methods to analysis of nuclear power plant risk. The study that is presented here is a risk assessment of Limerick 1, a BWR/4, having essentially the same thermal power rating as the WASH-1400 BWR, but utilizing a later containment design, the Mark II. (Design characteristics of Limerick are given in Section 2.3.)

The RSS methodology has been adopted for the Limerick risk assessment. However, there are a number of changes required to implement the methodology for Limerick. These changes include:

- A revised list of accident initiators
- A new more detailed set of event trees to model the sequence of events following each initiator
- A new plant-specific set of fault tree logic models for Limerick
- A containment analysis specific to the Mark II containment

- An improved version of CRAC
- Site-specific meteorology and demography
- Changes in the RSS methodology to account for criticisms (primarily in the Lewis Committee report).

1.3.2 Overview of the Comments on the Reactor Safety Study

The RSS was issued in draft form and was widely reviewed and issued in final form as reference 1-2. Appendix XI of the RSS presents comments on the draft report, some of which were addressed in the final report, and some of which are applicable to the final report. Only a few critiques have been published of the final report (1-3, 1-4), and of these, the Lewis Committee's comments (1-4) are the more recent. Some of the major WASH-1400 comments by the Lewis Committee that are significant to the Limerick analysis are paraphrased below:

1. Despite its shortcomings, WASH-1400 provides at this time the most complete single picture of accident probabilities associated with nuclear reactors. The fault tree/event tree approach coupled with an adequate data base is the best available with which to quantify these probabilities.

2. We are unable to determine whether the absolute probabilities of accident sequences in WASH-1400 are high or low, but we believe that the error bounds on those estimates are, in general, greatly understated. This is true in part because of an inability to quantify common cause failures, and in part because of some questionable methodological and statistical procedures.

3. It should be noted that the dispersion model for radioactive material developed in WASH-1400 for reactor sites as a class cannot be applied to individual sites without significant refinement and sensitivity tests.

4. The biological effects models should be updated and improved in the light of new information.

5. After having studied the peer comments about some important classes of initiating events, we are unconvinced of the correctness of the WASH-1400 conclusion that they contribute negligibly to the overall risk. Examples include fires, earthquakes, and human accident initiation.

6. It is conceptually impossible to be complete in a mathematical sense in the construction of event trees and fault trees; what matters is the approach to completeness and the ability to demonstrate with reasonable assurance that only small contributions are omitted. This inherent limitation means that any calculation using this methodology is always subject to revision and to doubt as to its completeness.

7. The statistical analysis in WASH-1400 leaves much to be desired. It suffers from a spectrum of problems, ranging from lack of data on which to base input distributions to the invention and use of wrong statistical methods. Even when the analysis is done correctly, it is often presented in so murky a way as to be very hard to decipher.

8. For a report of this magnitude, confidence in the correctness of the results can only come from a systematic and deep peer review process. The peer review process of WASH-1400 was defective in many ways and the review was inadequate.

9. Lack of scrutability is a major failing of the report, impairing both its usefulness and the quality of possible peer review.

This report responds to finding (1) by adopting the same fault tree/event tree structure as that used in RSS. It does use a different organization structure, however, in response to item (9), by using Section 3.0 as a point of coalescence for the results of the separate analyses. In analogy to computer programming, it is a "calling" routine that calls in results from the detailed analyses that are contained in the appendices. The event trees contained in Section 3.0 provide the organizing skeleton on which to bring results from the appendices together. This allows the Limerick PRA to be read at several different levels: as a summary accepting all the results without details; as a discussion of methodology, accepting results from appendices; and in entirety, in which results in the summary are developed in Section 3.0 to one level of detail which, in turn, are fully developed in the appendices. This is meant to facilitate the review process and the traceability of results through the report.

The Limerick PRA addresses item (2) of the Lewis Committee criticisms cited above by a reevaluation of the component failure probabilities, using the most applicable data or reasonable extrapolations to the accident environment, with individual error factor estimates on the quantities. Error bands are also estimated for human error rates, for release fractions, and for parameters in the consequence calculations. The characterization of uncertainties in the final risk curve qualitatively assesses the error bounds attributable to component failure rate uncertainties, core melt phenomenology uncertainty, and ex-plant dose uncertainties.

Item (3) is addressed through the use of site-specific data and sensitivity analyses appropriate to the Limerick site. The Lewis Committee's items (4) and (5) may warrant consideration which is outside of the scope of this project. The analysis presented here for these items is consistent with the Reactor Safety Study. Specifically, fire, flood, sabotage, and earthquake are excluded from this study as initiating events.

It is recognized that completeness cannot be assured in a mathematically rigorous sense (item 6); however, in Section 3.2, this study does address the problem in a systematic manner.

The Lewis Committee's criticisms of the statistical analysis (item 7), while technically correct, do not reflect the problems associated with working with data derived from small sample sets. This report avoids subjective estimates of common-mode coupling factors. Appendices F and G provide further discussion of the statistical methods used in this report.

While it is not possible to have a complete peer review during the report preparation (item 8), the present work was prepared under continuous internal review by GE, SAI, PECo, and NUS (under separate contract with PECo).

1.4 UNCERTAINTIES IN THE ANALYSIS

The Limerick risk assessment provides a realistic estimate of the risk associated with the operation of Limerick, and characterizes the potential uncertainties in the evaluation. The principal contributors to the uncertainty in this risk assessment are generally in areas where experimental or operating experience data is lacking. These areas are then subject to modeling uncertainties and quantification uncertainties.

Characterization of the uncertainties is presented in Section 3.8. The principal areas of uncertainty are:

- The health effects models
- Release fractions and dispersion calculation
- Events possibly requiring further investigation
- Completeness
- Common-mode failure treatment
- Data base
- Core melt phenomenology
- Steam explosions and their effects.

1.5 GROUNDRULES AND ANALYTICAL BASES

This section summarizes the groundrules and analytical bases for probabilistic risk assessment of the Limerick Generating Station:

- Accident initiators
- End point of the analysis
- Methodology
- Plant configuration

- Systems included
- Operator action
- Nomenclature
- Source analyses
- Systematic failure causes
- Containment integrity
- Core melt phenomenology
- Meteorological data
- Ex-plant consequence model
- Population data
- Health effects model
- Evacuation model
- Radioactive decontamination factors
- Component failure rate data
- Maintenance and test data
- Success criteria
- Uncertainties
- Fault tree modeling.

Accident Initiators: The initiating events considered in this evaluation are discussed in Appendix A.1. They include LOCAs, anticipated transients, and transients which are unlikely but postulated to occur with a low probability. Initiating events which have been excluded from this risk assessment include seismic events; fires; other external events such as tornadoes, hurricanes, floods; and sabotage. The initial condition of the reactor at the time of accident initiation is taken to be State F, shown in Table 1.1

Table 1.1

DEFINITION OF BWR OPERATING STATES

CONDITIONS	STATES					
	A	B	C	D	E	F
Reactor vessel head off	X	X				
Reactor vessel head on			X	X	X	X
Shutdown*	X		X		X	
Not shutdown		X		X		X
Pressure $\leq$ 850 psig	X**	X**	X	X		
Pressure > 850 psig					X	X

*K_{eff} sufficiently less than 1.0 so that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentrations.

**Because the reactor vessel head is off in States A and B, pressure is atmospheric pressure.

End-Point of the Analysis: Each accident sequence identified in the analysis is terminated in either of two ways:

1. Acceptable: When the reactor reaches the condition of hot*, stable shutdown. The reactor is in hot shutdown when the mode switch is in shutdown, the reactor is subcritical, presure in the reactor is stabilized, temperatures in the fuel and reactor are within all limits, containment and suppression pool cooling are being maintained, and vessel level is controlled. (This is generally about 20 hours after the occurrence of the initiating event for most accident sequences evaluated.)

*The analysis was not carried to cold shutdown due to the potentially long time period needed to reach cold shutdown, and because of the routine nature of the transition from hot to cold shutdown.

2. Unacceptable: When above conditions are not met and some radioactive release may occur. For these cases the off-site consequences are calculated.

Methodology: The methodology used in the risk assessment of Limerick is an updated set of techniques originally used in the WASH-1400 Reactor Safety Study. Specifically, event tree and fault tree methods are used to logically display the plant system interactions for the identified accident sequences. Quantitative and qualitative evaluations are carried out using the WAM series of computer codes which were developed for EPRI (see Appendix K). Radioactive release fractions for the accident sequences are developed using the INCOR code series, also developed for EPRI, and include updated versions of the physical models and codes used in WASH-1400. Calculation of consequences to the public is carried out using the updated version of CRAC.

Plant Configuration: Limerick was considered a completed design. The system evaluation has been performed using design drawings from GE and Bechtel for Limerick Unit 1 only, and considers no cross ties, benefits, or other effects between the two units.

Systems Included: All important systems are included regardless of safety classification (e.g., condensate and feedwater).

Operator Action: Those planned or unplanned manipulations which are required by procedure, or which are possible remedies to a failed system, are depicted and evaluated. Operaton actions which defeat system performance or which aggravate the achievement or maintenance of stable hot shutdown are generally not evaluated.

Nomenclature: All system and component references are consistent with the Limerick terminology.

Source Analyses: Source data for this analysis include those thermal/nuclear transient analyses which have been performed in support of system design and safety analysis for the events and conditions depicted on the reliability fault trees.

Systematic Failure Causes: In a manner similar to WASH-1400, sabotage, seismic event, or fire have not been postulated in coincidence with other initiators. The results of this analysis are valid to the extent that the systems have been successfully designed as independent and redundant. Current NRC requirements on seismic design, equipment separation, environmental qualification, and security are assumed adequate to limit the probability of core melt scenarios due to common-mode cause to below that of other sequences identified in this analysis.

Containment Integrity: The BWR analysis carried out in WASH-1400 assumed that for conditions leading to core melt, the containment would eventually fail. The mechanisms of failure, however, differed greatly, and this affected the radioactive release fractions and ex-plant consequences. The containment characteristics used in the Limerick risk assessment are that:

1. Containment has an ultimate capability of 140 psig (more than two and one-half times design pressure) similar to the WASH-1400 assumption for concrete structures. Appendix J contains a structural analysis of the Limerick containment capability under extreme plant conditions.

2. Containment pressure relief is used to prevent containment overpressure failure, in accordance with emergency procedure guidelines developed as a result of post-TMI analyses. Possible failure of overpressure relief is included in the analysis.

3. Containment sprays would effectively reduce containment pressure in certain accident sequences; however, their effect was not evaluated in this analysis, and their use is not required for successful termination.

4. Significant containment overpressure failures near the top of the containment (designated γ) result in a direct release to the atmosphere, since calculations show the secondary containment (Reactor Enclosure) is not capable of withstanding the surge.

5. Significant containment overpressure failures near the base of containment (designated γ') will travel directly out through ground level blowout panels in the Reactor Enclosure.

6. The Limerick containment is to be inerted, and therefore hydrogen burning and/or explosion probability is considered possible only during periods when the containment is deinerted.

7. Containment isolation or leakage at high internal pressures is evaluated and included as a potential containment failure mode.

8. The standby gas treatment system (SGTS) is given credit for reducing radioactivity releases for cases of small containment leakage. For cases of gross containment failure no credit is given for SGTS operation.

9. Liquid pathways are not considered. Calculations with the INTER code (Reference 1-5) preclude more than a four-foot penetration of concrete. The Mark II containment has a concrete diaphragm and water source directly below the reactor.

Core Melt Phenomenology: The modeling of the in-containment radioactive source term for input into CRAC for postulated core melt sequences is based on the following:

1. Zirconium water reaction is calculated during core uncovering and melting, however, it is not included during the interaction of the molten core with the bottom head.

2. Control rod drive penetrations in the bottom head RPV are not explicitly modeled, but are incorporated by qualitative analysis (see Appendix C and J).

3. The molten core is assumed to be distributed over a large segment of the diaphragm floor for corium/concrete interaction calculations.

4. Molten core is treated as a stratified layer of metallic and oxide components during its interaction with concrete.

5. Concrete interaction modeling is based on small scale laboratory tests.

6. At the time of calculated 80% core melt, the core grid is assumed to fail, as in WASH-1400.

7. Aerosol agglomeration and settling processes are treated conservatively.

8. Water pool scrubbing factors are used to quantify the suppression pool retention of radionuclides.

Meteorological Data: The following items form the basis for the meteorological data used for the Limerick site-specific analysis:

- Data taken from one (1) meteorological tower; data from other towers used to confirm or fill in missing data.
- The evaluated risk is based upon five years of meteorological data.

Ex-Plant Consequence Model: Bases for the consequence modeling include:

1. CRAC is updated hourly for wind speed, stability class, and precipitation.

2. A "puff" release is used for the CRAC analysis. (Puff release is a term used in describing the assumption that the radionuclide release from containment is assumed to all occur over a very short time following containment failure. A 30 minute puff release is felt to be more realistic than the 3 minute release used in WASH-1400 which was selected due to modeling limitations.)

3. Gross topological features are accounted for.

4. The treatment of evacuation includes the use of an average effective rate of travel for the duration of the accident evacuation period. The treatment and effective speed utilized were consistent with WASH-1400.

5. Cloud deposition due to terrain roughness is calculated using a terrain characteristic of the Limerick site.

6. Shielding factors for the population are calculated in the following manner:

- Ground dose and cloudshine are calculated for buildings characteristic of Pennsylvania per the WASH-1400 survey.

- The inhalation dose is calculated so as to provide a more realistic estimate of the inhalation dose than the "no shielding" case assumed in WASH-1400.

7. Site-specific meteorology and population are used in the LGS evaluation.

Population Data: The population data for the Limerick site-specific analysis is based upon the following:

- Data compiled by PECo out to a 50-mile radius for 1970
- Additional data compiled by the Center for Planning and Research for 1970 out to a 500-mile radius.

Health Effects Model: The Limerick PRA employed the same health effects models as used in WASH-1400.

Evacuation Model: The LGS PRA is carried out using a model similar to that used in WASH-1400 for evacuation of the population from the vicinity of the plant. The principal features of the model include:

- Rate of departure of the population
- Radius of evacuation
- Shielding factors of the population during evacuation
- Fraction remaining within the vicinity.

Radioactive Decontamination Factors (DF): The DFs used in the Limerick PRA are the result of an updated evaluation of the physical processes involved in deposition of radioactive isotopes inside containment. This evaluation is based upon data and analysis from GE, EPRI, SAI, and other industry sources.

The decontamination factors are determined as follows:

- Saturated suppression pool DF from current literature sources not available to WASH-1400
- Reactor system DFs estimated from standard sources and from consultation with experts in the field (ultimately resulted in using values similar to WASH-1400).
- Plateout and settling DFs for the primary containment are modeled for each accident sequence using CONTEMPT and CORRAL phenomenology.
- Reactor enclosure plateout and settling DFs treated similarly to primary containment
- Filters on standby gas treatment system are treated using technical specification values.

Component Failure Rate Data: The quantification of event tree/fault tree models is based upon the best available data base applicable to each specific component or system. However, the availability of data appropriate for such quantification is limited on both a generic basis* and a plant-specific basis. The sources of data chosen for the Limerick risk assessment, in order of priority, are the following:

- Plant- or component-specific (e.g., loss of offsite power and Target Rock safety/relief valves)
- NRC (pumps, valves, diesels, and human errors)
- General Electric
- WASH-1400.

In some instances it was useful to combine several data sources, plant data, or plant-specific data. The use of Bayesian statistics provides one of several methods of combining existing generic data with new data.

*Even NRC's latest data reports make only a small contribution to the data base needed to characterize specific pump application, from specific manufacturers, and of specific models.

However, this technique does not improve the data; it only provides a formal method for arriving at a mean value for quantification. Combining of various data sources is discussed in greater depth in Appendix A.2.

Variabilities which are not generally incorporated into the component level failure probability quantification are the following:

- Variability among components, e.g., motor operated valves (size, application, qualification requirements, environment)
- Variability among manufacturers (considered in some cases)
- Variability among component models (considered in some cases)
- Variability in method of installation
- Variability in component age.

<u>Maintenance and Test Data</u>: Maintenance outages of safety systems may contribute a large portion of the unavailability of a system. These outages can be either forced or scheduled outages. There are technical specification requirements (Limiting Conditions of Operation-LCO) which limit the number of safety systems which can be concurrently unavailable. Two sources of data were used in the Limerick assessment:

- A recent GE survey of BWRs and Peach Bottom plant-specific data (see Appendix A)
- WASH-1400.

Generally, Peach Bottom LCOs were used, since Limerick technical specifications do not exist at this time. Safety system test frequencies are important in determining the failure probability of a component given a demand. Since no test frequencies exist for Limerick at this time, the test frequencies used for the Limerick risk assessment are those given in the proposed Susquehanna Technical Specifications. The technical specification values from Peach Bottom and the test frequencies from Susquehanna are used as reasonable expectations of future Limerick specifications.

Success Criteria: The systems, as discussed in this report, have three primary functions* related to accident mitigation:

- Short-term coolant injection, or reflood, which is required following a LOCA or transient
- Coolant recirculation which is required for long-term core cooling following a LOCA
- Containment cooling which is assumed to be required for both transients and LOCA conditions.

The success criteria are defined here for the event initiators considered in the LGS PRA.

Table 1.2 summarizes the success criteria for the systems available to provide mitigation following LOCA and transient event initiators. For LOCA and transient events, the success criteria are defined in terms of the minimum number of systems required to prevent excessive fuel clad temperature, and to remove the decay heat. The bases for establishing the success criteria for the Limerick PRA are the best estimate predictions of Emergency Core Cooling System performance. The success criteria given in Table 1.2 were generated from analyses which incorporated best estimate decay heat, modeled with the mean value of the 1978 American Nuclear Society (ANS) decay heat standard, but with modeling of core heatup to account for steam cooling effects following postulated core uncovering.

*The additional required function of primary system pressure relief was not modelled, since it is a passive function (spring-loaded safety valves) with very low probability of failure.

Table 1.2

SUMMARY OF SUCCESS CRITERIA FOR THE MITIGATING SYSTEMS TABULATED AS A FUNCTION OF ACCIDENT INITIATORS

ACCIDENT INITIATOR	SUCCESS CRITERIA*	
	Coolant Injection	Containment Heat Removal
Large LOCA: Steam Break $\geq 0.08ft^2$ Liquid Break $\geq 0.1ft^2$	1 of 4 LPCI Pumps OR 1 of 2 Core Spray Subsystems (2 pumps)	1 RHR
Medium LOCA: Steam Break 0.016 to $0.08ft^2$	HPCI OR 1 of 4 LPCI Pumps OR 1 of 2 CS Subsystems } and ADS*	1 RHR OR COR
Small LOCA: Steam Break $< 0.016ft^2$ Liquid Break $< .004ft^2$	HPCI OR RCIC OR 1 Feedwater Pump OR 1 of 2 CS Subsystems OR 1 of 4 LPCI Pumps OR 1 Condensate Pump } and ADS*	Normal Heat Removal OR 1 RHR OR COR
Transient	Same as Small LOCA	Same as Small LOCA
IORV	Same as Small LOCA	Same as Small LOCA
Transient + SORV	Same as Small LOCA	Same as Small LOCA

*ADS requires operation of only two safety/relief valves for adequate depressurization.

The success criteria for LOCA events are dependent upon both the size and location of the break. To account for these effects, two break types (liquid and steam) are distinguished, and the entire spectrum of break sizes is divided into three catego·ies. The Large LOCA covers the upper end of the spectrum, where the break is sufficiently large that rapid reactor vessel depressurization, due to loss of inventory through the break, enables initiation of the low pressure systems without ADS actuation, and generally without manual depressurization. At some locations, breaks toward the lower end of the size range may require operation of one or two safety/relief valves.

The Medium LOCA is defined as the range of break sizes where the break flow is greater than the capacity of the RCIC system, but is not great enough to cause timely initiation of the low pressure systems without the help of ADS* or manual depressurization. HPCI will maintan adequate core cooling automatically for 1-1/2 to 2 hours before low steam pressure trip after which time operator action may be required. If low pressure systems are unavailable, feedwater may also be used, if available, but is not shown since the MSIVs may close due to possible loss of instrument air. Use of COR for decay heat removal would be accomplished with the reactor in the alternate shutdown cooling mode, as defined in the Emergency Procedure Guidelines to minimize suppression pool loading.

The Small LOCA covers the lower portion of the break-size spectrum, where the break flow would be equal to or less than the capacity of the RCIC. Operator action might be required if RCIC trips on high reactor water level or low reactor pressure. One feedwater pump** would also be adequate for core cooling if MSIVs were open or bypassed. For Small LOCAs, the MSIVs may remain open and the main condenser could be used for normal heat removal. When using COR, the reactor should be in the alternate shutdown cooling mode.

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*Only two safety/relief valves are required for adequate depressurization.

**A condensate pump is also adequate for cases where electric power and suction water are available.

For transient events, including IORV and SORV, ECCS system response is very similar to the response to a small LOCA, except that manual depressurization, by opening two safety/relief valves, would be required to initiate low pressure ECCS systems.

Table 1.3 summarizes system capability for a transient event with failure to scram(ATWS). For these events, as in the LOCA and transient events, the criteria are defined in terms of the minimum number of systems required to prevent excessive fuel clad temperature and remove decay heat. (The table shows failed systems or functions.)

In Table 1.3, for all conditions shown as "acceptable" (A), suppression pool bulk temperature is maintained below 220^{o}F. For those cases requiring the use of containment overpressure relief (COR), suppression pool temperature may rise above 220^{o}F. The unlikely case of one standby liquid control pump and both RHRs failed may result in suppression pool condensation conditions that have not been proven to be acceptable. The effect of this uncertainty on risk is discussed in Section 3.8 (Table 3.8.2).

The success criteria, as shown in Table 1.3 are based on the following:

- An automatic standby liquid control (SLC) system
- Failure of alternate rod insertion (ARI)
- Turbine trip ATWS initiator event has 25% steam flow bypass available
- No inadvertent ADS initiation.

For the degraded condition of feedwater and HPCI failed, the operator may be required to inhibit ADS. For the condition where HPCI continues to run, operator action within 10 minutes to prevent overfilling the vessel would provide successful shutdown.

Table 1.3

SUMMARY OF LGS CAPABILITY FOR ATWS MITIGATION
(Alternate 3A Modifications)

Transient Initiator	Failed Systems or Functions									
	1 SLC PUMP	1 SLC + FW + RCIC	1 SLC + 1 RHR	1 SLC + 2 RHR	FW + RCIC	FW + HPCI	HPCI LEVEL 8 TRIP	FW RUNBACK	MSIV LEVEL 1 TRIP	RPT TRIP
TURBINE TRIP	A	A	A	A	A	A	N	A	A	N
MSIV CLOSURE	A	A	A	COR	A	A	N	A	A	N
LOSS OF OFFSITE POWER	A	A	A	COR	A	A	N	A	A	A
INADVERTENT OPEN RELIEF VALVE	A	A	A	COR	A	N	N	A	A	A

A:acceptable
N:not acceptable
COR: Containment Overpressure Relief

Uncertainties: The major objective of this risk assessment is to develop a best estimate complementary cumulative distribution function (CCDF) for early and latent fatalities for the Limerick plant. In addition, the following characterization of uncertainties is performed:

1. Uncertainties for selected dominant sequence probabilities are generated, using a Monte Carlo simulation of the system models, and the individual component uncertainty distribution.

2. Subjective characterization of CCDF uncertainty, including the uncertainties in:

   - Sequence evaluations
   - In-core radioactive release processes
   - Ex-plant consequence calculations.

Guidelines Used in Evaluating System Reliability with a Fault Tree Model: The Limerick risk assessment uses the following guidelines in modeling the BWR systems:

1. The scope of the fault tree model extends down to the component level (e.g., pumps, valves, sensors). Generic component fault trees are used to identify component failure modes; however, subcomponent parts are not treated separately.

2. The fault tree model logic has been developed allowing each component to operate as designed, or to be in a failed state. Partial component operation is considered a failure. This is also the basis for the failure rate data used in this analysis.

3. Systematic failure mechanisms are characterized as follows:

   - Seismic loadings are not coincident with any accident initiators
   - Transient initiators and subsequent system failures do not cause fires

- Component design meets the requirements needed for proper operation

- Sabotage is not considered

- Operator training, supervision, and plant maintenance are adequate for proper operation of a nuclear power plant.

4. The maintenance contributions in the fault tree model are mutually exclusive among certain systems, as provided by the Limiting Conditions of Operation. An example of this would be that HPCI is not allowed in maintenance if RCIC is unavailable. For these cases, a "NOT" gate is used to represent this mutual exclusivity when two or more such systems are combined together for redundant operation. The derivation of each of the maintenance contributions to each system fault tree is included in Appendix A.4.

5. Disablement of individual systems due to pipe rupture coincident with other accident initiators is included for completeness. It is viewed as a much less likely failure event than other possible active failures, and therefore, its quantification does not affect the calculated reliability. Pipe ruptures are included as possible at the inlet, pump connection, and discharge for each applicable system or subsystem. In each case, the pipe is treated as a "section", where the probability of failure is ascertained from sources which estimate probabilities of failure per section (e.g., 1000 ft. sections).

6. Valve and pump ruptures are not explicitly included in the fault tree model, because of their low failure rate relative to other valve and pump failure modes.

7. In modeling the instrumentation and control systems, the following considerations were taken into account:

    - Most systems depend on instrumentation to monitor parameters and control system/component operation.

    - Instrumentation for a system is not localized, but begins in one compartment, takes power from another, is routed through a number of others, and is displayed in still another.

    - Location-dependent common-mode failures are not considered.

8. Common-mode miscalibration of similar sensors is incorporated into the model (see Appendix A).

9. Manual Operation -- Several guidelines are used to define the operator action assumptions used in the model:

   Detailed analysis of the adequacy of core cooling under extreme conditions indicates that positive manual operations can be delayed for more than 30 minutes (in most cases, 2 to 4 hours). This is based upon the adequacy of core cooling even if the effective reactor water level is below the top of the active fuel. In the analysis involving evaluation of adequate core cooling and core uncovering, human intervention to establish core coolant injection is not considered to be necessary for at least 30 minutes.

   The event tree/fault tree analysis has been performed using the human-error rates documented in Appendix A. These error rates have been applied to obvious actions which the operator can perform during an accident sequence. In addition, those maintenance recovery actions which may be in error and which would adversely affect the system operator have been included in the component failure rates (see the generic component fault trees). Operator action to restore failed or tripped systems has been included in the case of the power conversion system (PCS) and the diesels.

10. The bases for fault tree quantification are:

    - The best estimate for a given probability is associated with the mean value of the data. The failure rates used in the study are representative of the equilibrium portion of the plant life.

    - The entire analysis is based on the use of realistic assumptions, data, and success criteria, and is intended to model, insofar as possible, actual events and actions as they would be expected to occur.

11. The failure of display of information to the operator is treated as a random independent failure or set of failures and is not dependent on the accident sequence.

1.6 LIST OF ACRONYMS

Table 1.4 lists the acronyms and abbreviations used in the Limerick Probabilistic Risk Assessment.

TABLE 1.4

ACRONYMS

AC	Alternating Current
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission (predecessor of the NRC)
ANSI	American National Standards Institute
APED	Atomic Power Equipment Department (GE)
APRM	Average Power Range Monitor
ATWS	Anticipated Transient (s) Without Scram
BWR	Boiling Water Reactor
CCDF	Complementary Cumulative Distribution Function
CHF	Critical Heat Flux
COR	Containment Overpressure Relief
CORRAL	Containment of Radionuclides Released after LOCA
CRAC	Calculation of Reactor Accident Consequences
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
CT	Cooling Tower
D	Demand
DBA	Design Basis Accident
DC	Direct Current
DF	Decontamination Factor
DG	Diesel Generator
ECCS	Emergency Core Cooling Systems
EHC	Electro Hydraulic Control
EPS	Electric Power Safeguard
ESF	Engineered Safety Feature
ESW	Emergency Service Water
ETA	Event Tree Analysis
FC	Fails Closed
FO	Fails Open
FSAR	Final Safety Analysis Report

TABLE 1.4 (continued)

FTA	Fault Tree Analysis
FT/ETA	FTA and ETA
FW	Feedwater
GE	General Electric Company
HCU	Hydraulic Control Unit
HEPA	High Efficiency Particulate Air/Absolute - referring to Filters
HP	High Pressure
HPCI	High Pressure Coolant Injection
H&V	Heating and Ventilating
HVAC	Heating Ventilating and Air Conditioning
HX	Heat Exchanger
IAC	Interim Acceptance Criteria (AEC)
IBV	Inboard Isolation Valve
I&C	Instrumentation and Control
IEEE	Institute of Electrical and Electronic Engineers
INCOR	Inputs to CORRAL
IORV	Inadvertent Open Relief Valve
IRM	Intermediate Range Monitor
LC	Locks Closed
LCO	Limiting Condition for Operation
LCS	Leakage Control System
LDS	Leak Detection System
LGS	Limerick Generating Station
LO	Locks Open
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LP	Low Pressure
LPCI	Low Pressure Core Injection (a mode of RHR)
LPCS	Low Pressure Core Spray (or Core Spray)
LPRM	Local Power Range Monitor
MCC	Motor Control Center

TABLE 1.4 (continued)

MMH	Monorail Mounted Hoist
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
NC	Normally Closed
NED	Nuclear Energy Division (GE)
NLC	Normally Locked Closed
NLO	Normally Locked Open
NO	Normally Open
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUS	NUS Corporation
OBV	Outboard Isolation Valve
PCS	Power Conversion System
PECo	Philadelphia Electric Company
P&ID	Process and Instrumentation Drawing
PRA	Probabilistic Risk Assessment
PRM	Power Range Monitor
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal-Service Water
RPS	Reactor Protection System
RPT	Reactor Pump Trip
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Clean-Up
SAI	Science Applications, Inc.
SAR	Safety Analysis Report
SDV	Scram Discharge Volume
SF	Shielding Factor

TABLE 1.4 (continued)

SFSP	Spent Fuel Storage Pool
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Injector
SLC	Standby Liquid Control
SP	Suppression Pool
SPASM	System Probabilistic Analysis by Sampling Methods
SRM	Source Range Monitor
S/RV	Safety/Relief Valve
SSE	Safe Shutdown Earthquake
SW	Service Water
TCV	Turbine Control Valve
TG	Turbine Generator
TIP	Traversing In-Core Probe
UHS	Ultimate Heat Sink

REFERENCES

1-1 Letter from D. G. Eisenhut (USNRC) to E. G. Bauer, Jr. (Philadelphia Electric Co.), "Risk Evaluation - Limerick Generating Station, Units 1 and 2", May 6, 1980.

1-2 Reactor Safety Study, "An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants", USNRC report WASH-1400, October 1975.

1-3 H. W. Kendall, Study Director, "The Risks of Nuclear Power Reactors", Union of Concerned Scientists, Cambridge, MA, August 1977.

1-4 H. W. Lewis, Chairman, "Risk Assessment Review Group Report of the U. S. Nuclear Regulatory Commission", NUREG/CR-0400, September 1978.

SECTION 2

LIMERICK GENERATING STATION

This section provides a general description of the Limerick site and generating units. An overview is presented in Section 2.1. Section 2.2 emphasizes the engineered safety features of the plant and Section 2.3 provides some of the pertinent Limerick design characteristics.

2.1 GENERAL DESCRIPTION

2.1.1 Background

The Limerick Generating Station (LGS), owned and operated by the Philadelphia Electric Company, is located on the east bank of the Schuylkill River: 4 miles downriver from Pottstown, 35 miles upriver from Philadelphia, and 49 miles above the confluence of the Schuylkill with the Delaware River. The site contains 587 acres -- 415 acres in Montgomery County and 172 in Chester County (see Figure 2.1.1).

The rural site consists of open field with considerable wood growth which enhances the atmospheric dispersion of effluent. The countryside is traversed by numerous valleys containing small streams that empty into the Schuylkill River. Two parallel streams, Possom Hollow Run and Brooke Evans Creek, traverse the site running southwest.

The site is in the Triassic Lowland section of the Piedmont Physiographic Province with the underlying rocks including Precambrian and Lower Paleozoic crystalline types. The remote possibility that seismic activity could occur in the site area was considered, and, therefore, a design basis earthquake has been hypothesized equivalent to the 1871 Wilmington, Delaware earthquake (intensity VII) near the site. Such an event

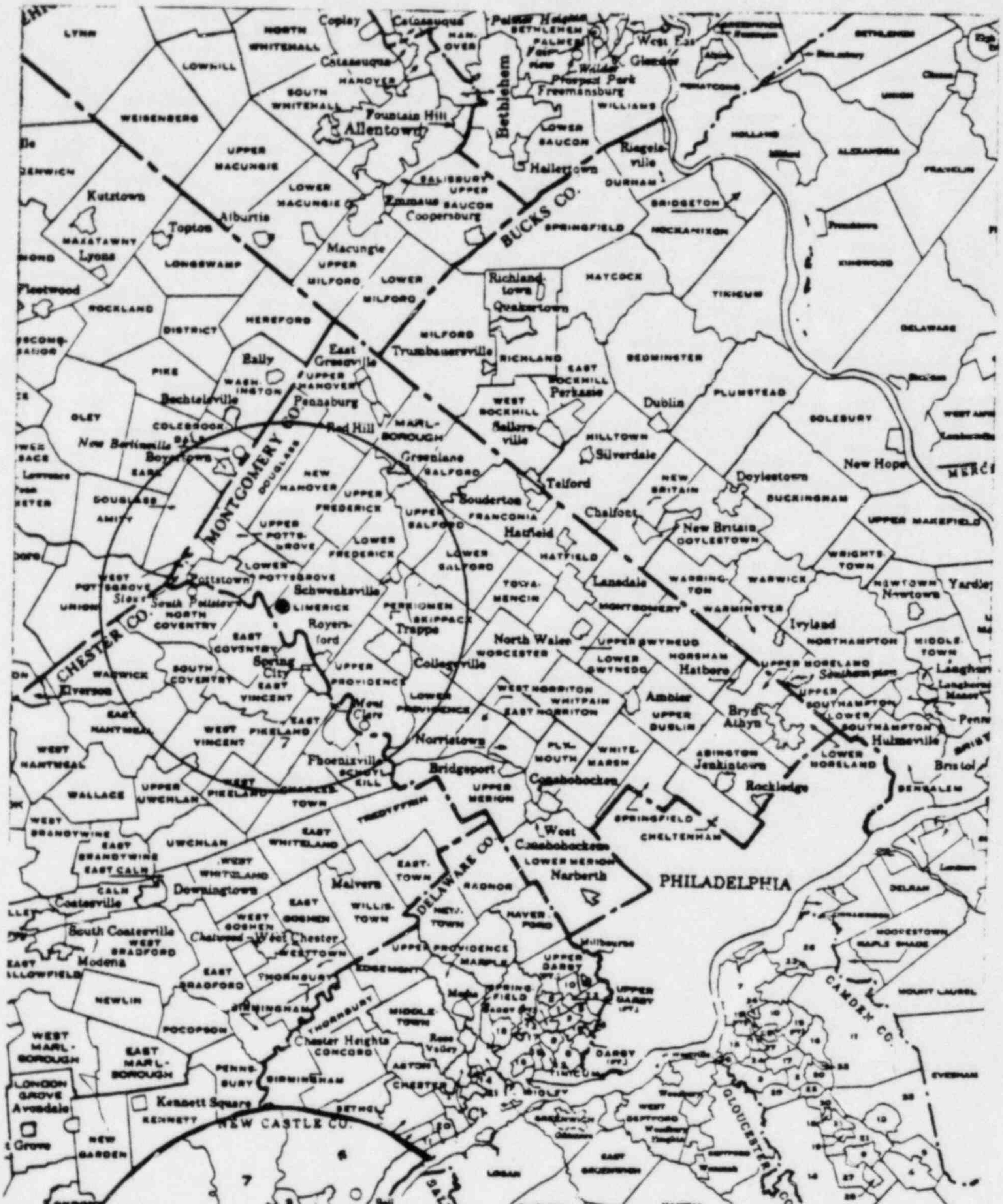

Figure 2.1.1 Limerick Site and Vicinity

is highly improbable and this hypothesis is very conservative. However, the plant does have the capability for safe shutdown if subjected to a peak horizontal ground acceleration of 0.15g. Seismic Category I structures are founded on the competent siltstone and sandstone bedrock of the Brunswick Lithofacies. Design spectra consistent with the safe shutdown earthquake are also used for the dynamic analysis of the Seismic Category I structures and equipment.

In the site area, Triassic-age siltstone, sandstone, and shale occur at shallow depths beneath a thin, relatively impermeable cover of residual soils. Most groundwater in the area is found in joints, fractures, and other secondary openings in the rock at relatively shallow depths, except in the vicinity of pumping wells. Because of the limited quantities of available groundwater which accounts for about 3% of the total industrial and commercial use in the region, surface water is the primary source of supply in the region. Although several water wells are located in the general site area, the geologic,hydrologic, and topographic conditions are such that the operation of a nuclear power plant at the proposed site has an extremely remote chance of adversely affecting the wells.

The flows of the Schuylkill differ considerably at different points along the river. This is mainly due to the varying topographic, climatologic, and geohydrologic conditions along the river. The extreme and average daily flows as recorded at Pottstown gauging station, (located about 5.5 miles upstream from the Limerick Generating Station site) are:

Flow (cfs)		Date
Minimum	87	August 13, 1980 (instantaneous)
Average	1,793	October 1926 - September 1969
Maximum	95,900	June 1972 (instantaneous)

Probable maximum flood (PMF) flow peak and height at the plant site have been estimated to be 500,000 cfs and 174 feet elevation, respectively (ground elevation of the plant is 217 feet). The effects of this maximum flood have been adequately handled in the plant design to provide safe shutdown of the reactor.

The river is primarily used to supply municipal and industrial water, although it is also used for recreational fishing and boating.

The Limerick site generally has a humid-continental climate. The region hosts continental air masses in winter, and alternating continental and maritime tropical air masses in summer. The site is near the track of most eastwardly-moving low pressure systems which are brought from the interior of the U. S. by the prevailing westerlies. Annual average wind speeds in the region are between 9 and 10 mph, and temperatures rarely exceed $100^{o}F$ or drop below $0^{o}F$. The region receives a moderate amount of precipitation, which is well distributed over the year.

The Limerick Generating Station site has a number of advantages associated with it which reduce the risk to the public relative to some other nuclear power plants. The location advantages of Limerick arise from its inland and relatively remote, rural setting, that is, an area which does not present unusual, external hazards to safe plant operation.

The site location can significantly affect the risk incurred by the public if adverse effects from outside the site cause a deterioration of safe plant conditions, which could in turn lead to the release of radionuclides to the environment. The proper choice of a site minimizes the potential for adverse effects on the plant either by improving plant reliability or minimizing man-made hazards, by providing:

1. A location central to sufficient power sources to provide a high level of assurance that off-site power sources will be available for safe shutdown. Limerick has at least five offsite power sources available.

2. A location remote from take-off and landing path-routes of aircraft to make airplane crashes affecting the plant a low probability. The Limerick site meets this criteria.

3. A location on a sparsely travelled inland waterway which, coupled with the Limerick ultimate heat sink (UHS) design*, minimizes the possibility of fouling the ultimate heat sink with oil or chemical spills.

In addition, natural disaster** demand frequencies for the LGS are at least as low as for northeast utility sites for:

- Seismic activity
- Hurricanes
- Tsunamis
- Flooding.

Meteorological data, collected for five years on the Limerick site, were used in the analysis.

The LGS consists of two boiling water reactor (BWR) generating units. Each is designed to operate at a rated core thermal power of 3293 MWt (100% steam flow) with a corresponding gross electrical output of 1092 MWe. Since approximately 37 MWe are used for auxiliary power, the net electrical output is about 1055 MWe. The multi-stage steam-driven turbine, which exhausts to the main condenser, provides the motive force for the electrical generator.

Condenser cooling is provided by water circulated through natural draft cooling towers.

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*The Limerick ultimate heat sink (UHS) is a spray pond. River water intake can be shut off if required to maintain UHS integrity and cleanliness.

**Not evaluated in the LGS risk assessment.

The Containment System limits the release of radioactive materials to the environs in the unlikely event that there would be a breach of the reactor system with radioactivity released inside containment. The design consists of a dual barrier: the primary containment and the secondary containment. The primary containment, which is a steel-lined reinforced concrete structure of the over-and-under configuration, employs the pressure suppression features of the BWR/Mark II containment concept. The secondary containment is the concrete reactor enclosure, which encloses the reactor, the primary containment, and fuel storage areas.

The LGS Preliminary Safety Analysis Report was submitted on February 26, 1970 (AEC Dockets 50-352 and 50-353). The Construction Permits, CPPR-106 and CPPR-107, were issued on June 19, 1974. Environmental impact is discussed in the Applicant's Environmental Report -- Construction Permit Stage (revised), dated May 1972. The Atomic Energy Commission (now Nuclear Regulatory Commission) issued the LGS Draft Environmental Statement in December 1972. A revised draft was issued in August 1973, and the LGS Final Environmental Statement was issued in November 1973.

2.1.2 Overview of the Nuclear Electric Power Plant

The nuclear system includes a single-cycle, forced circulation, General Electric boiling water reactor (BWR) producing steam for direct use in the steam turbine. Figure 2.1.2 is a simplified schematic of the BWR reactor system.

2.1.2.1 Reactor Vessel and Internals

The reactor vessel (see Figure 2.1.3) contains the core and supporting structure; the steam separators and dryers; the jet pumps; the control rod guide tubes; distribution lines for the feedwater, core spray, and standby liquid control; the incore instrumentation; and other components. The main connections to the vessel include the steam lines, the coolant recirculation lines, the feedwater lines, the control rod drive (CRD) and nuclear instrumentation housings, and the ECCS lines.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure is 1020 psia in the steam space above the separators. The vessel is fabricated of carbon steel and is clad internally with stainless steel (except for the top head which is not clad).

The reactor core is cooled by demineralized feedwater that enters the lower portion of the core and is heated as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through four main steam lines. Each steam line is provided with two isolation valves in series, one on each side of the primary containment barrier.

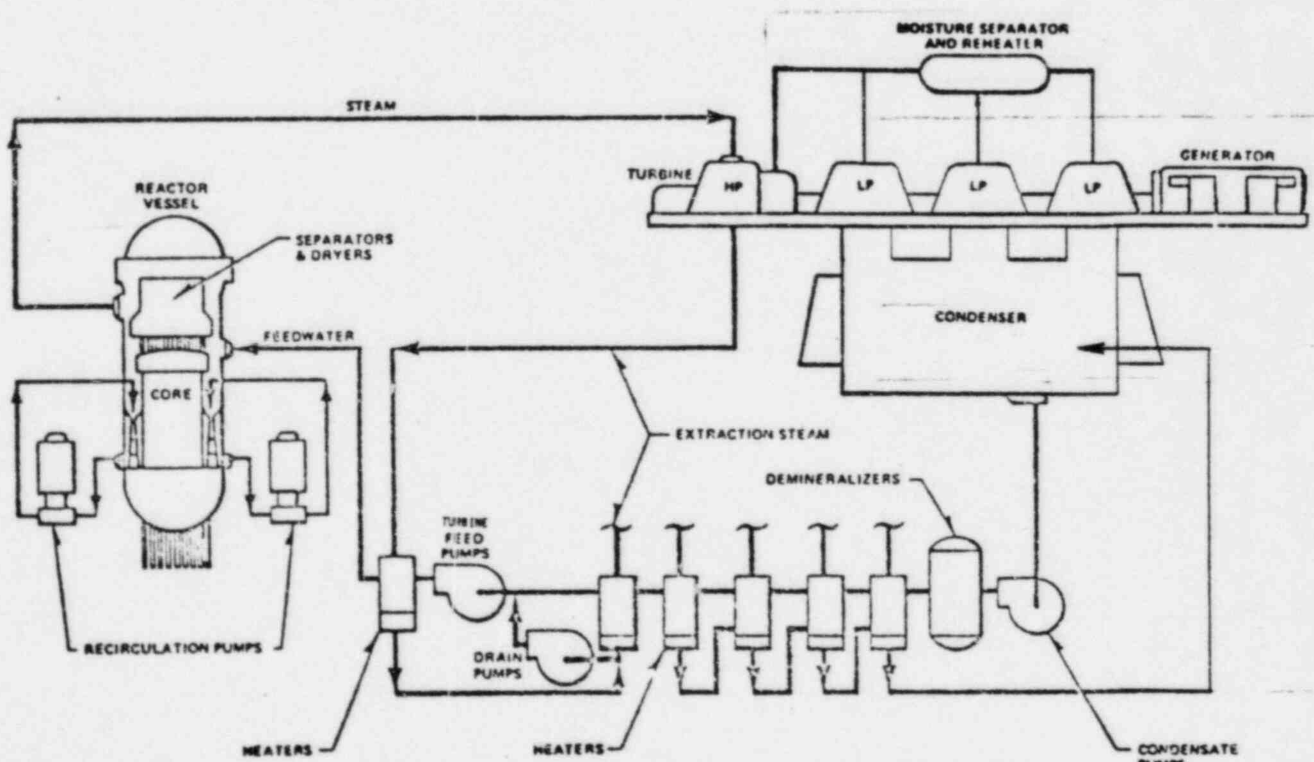


Figure 2.1.2 Simplified Schematic of BWR Direct-Cycle System

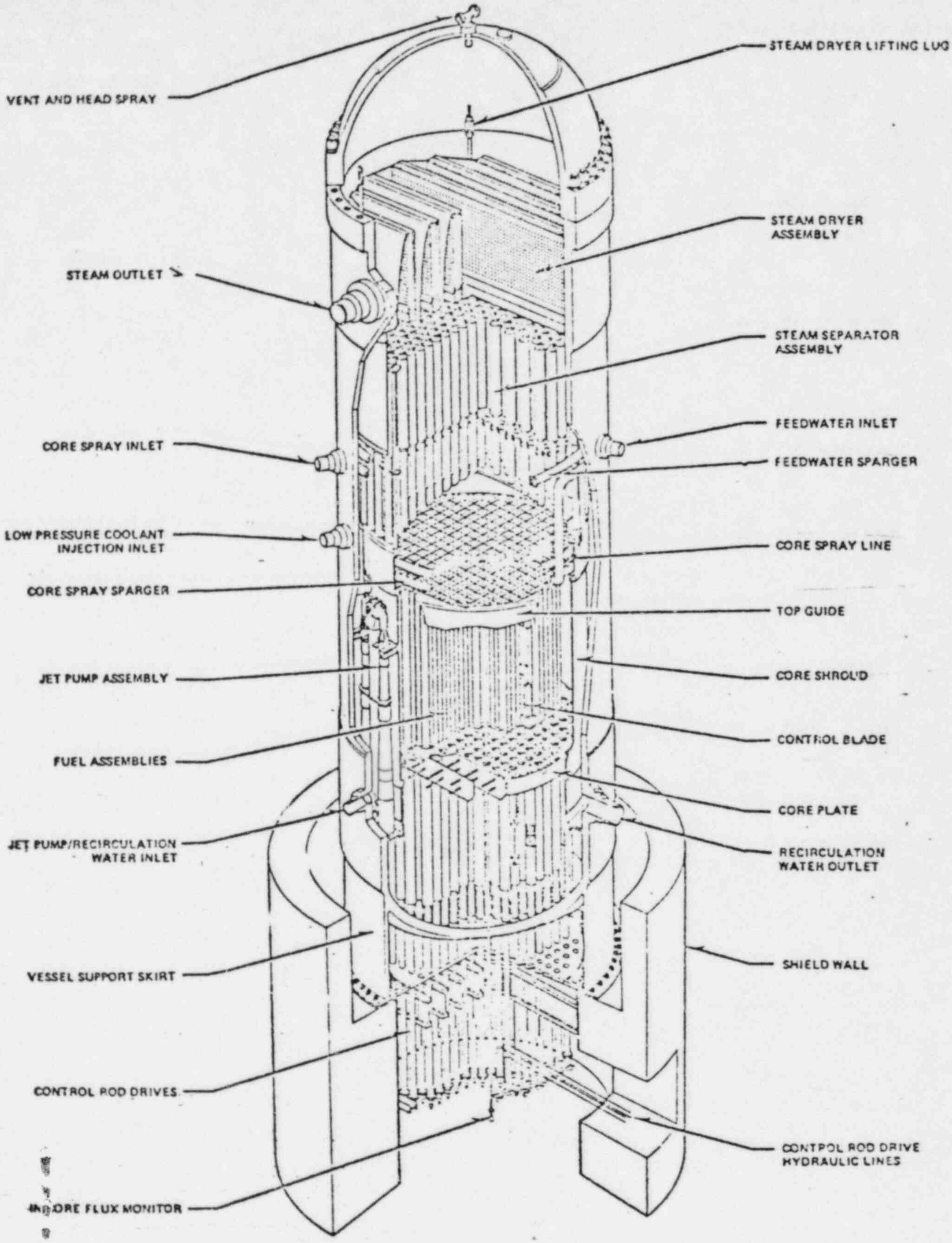


Figure 2.1.3. Reactor Assembly

2.1.2.2 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps, which provide a continuous internal circulation path for the major portion of the core coolant flow. Each loop has one motor-driven recirculation pump. Recirculation pump speed can be varied to allow some control of reactor power level through the effects of coolant flow rate on moderator void content.

2.1.2.3 Residual Heat Removal System

The residual heat removal (RHR) system consists of pumps, heat exchangers, and piping that fulfill the following functions:

1. Removal of decay and sensible heat during and after plant shutdown

2. Injection of water into the reactor vessel following a LOCA, to reflood the core independent of other core cooling systems

3. Removal of heat from the primary containment following a LOCA to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the suppression pool water (containment cooling) and by spraying the drywell and suppression pool air spaces (containment spray) with suppression pool water.

2.1.2.4 Reactor Water Cleanup System

A reactor water cleanup (RWCU) system is provided to clean up the reactor cooling water, to reduce the amounts of activated corrosion products in the water, and to remove excess reactor coolant from the nuclear system under controlled conditions.

2.1.2.5 Nuclear Leak Detection System

The nuclear leak detection and monitoring system consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

- Main steam lines
- RWCU system
- RHR system
- RCIC system
- Feedwater system
- ECCS systems
- Miscellaneous systems.

Small leaks generally are detected by monitoring the temperature, radiation levels, and drain sump fill-up and pump-out rates. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

2.2 ENGINEERED SAFETY FEATURES

The Limerick station is designed, fabricated, erected, and operated in such a manner as to confine any release of radioactivity to the limits and guidelines prescribed in applicable government regulations. Safety related systems are designed to permit safe plant shutdown and accommodate postulated accidents without endangering the public health and safety.

The previous section briefly described the barriers to the release of radioactivity. This section discusses the engineered safety features of Limerick which are available to prevent or mitigate accidents. The order of these topics is: reactor, containment, shutdown system, emergency cooling, ultimate heat sink, and electrical power.

2.2.1 Reactor

The BWR possesses the inherent safety feature that a power increase without corresponding cooling increase or, conversely, a cooling decrease at constant power is immediately accompanied by an increase in void formation with concomitant loss of moderation and power decrease. This is demonstrated by the ability of a BWR to control reactivity by coolant flow while the control rods remain fixed.

BWRs also exhibit the following desirable features:

1. Low Core Power Density - The power density in Limerick is relatively low (48.7 kW/ltr).

2. Jet Pump Recirculation - The use of jet pumps increases the in-vessel water flow rate, thereby reducing the size of the recirculation lines and recirculation pump power.

3. Fuel Coolability - Each fuel bundle is enclosed by a fuel channel. This channel would tend to confine damaged fuel from a reactor accident and prevent it from blocking the passage of the control rods. Should the water flowing through a fuel assembly become blocked, the fuel element can be cooled by heat conduction through the fuel channel.

4. Core Shroud - The core shroud enhances natural circulation by channeling cooling water down around the sides of the core for bottom entry. Thus, steam blockage of cooling water channels cannot occur.

5. Energy Absorber - While an in-vessel steam explosion is not considered possible, should such an explosion occur, the steam dryers above the core and the control mechanisms below the core will have the effect of dissipating the explosive energy. Thus, reactor vessel rupture (as postulated in the Reactor Safety Study) is very unlikely.

2.2.2 Primary Containment

2.2.2.1 Vapor Suppression

The BWR Mark II is a vapor-suppression containment (see Figure 2.2.1) which causes a reactor fluid release -- such as from relief valves

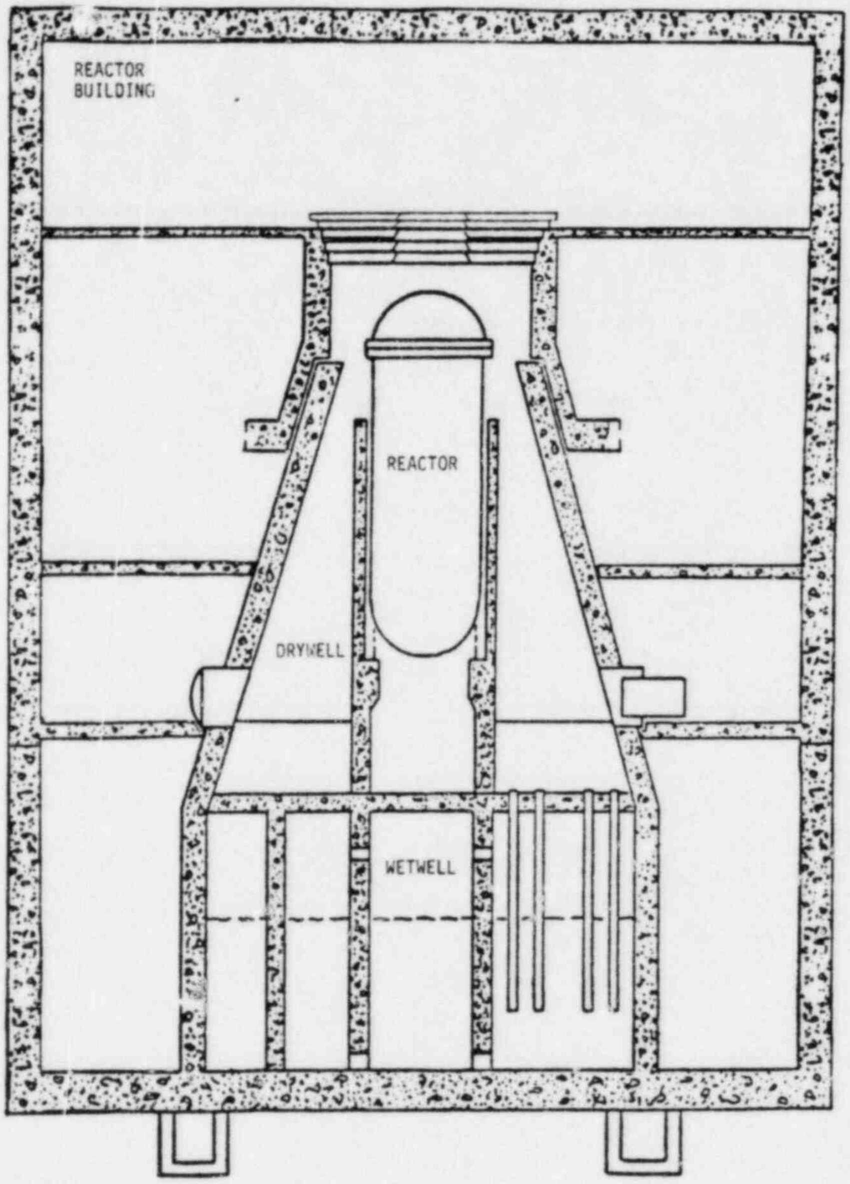


Figure 2.2.1 Mark II Containment

or a failure of the reactor coolant boundary -- to bubble through the pressure suppression pool (about 20 feet deep). The use of a vapor-suppression containment provides a double benefit:

- It reduces the size requirements for the containment.
- It acts as a filter or scrubber in the unlikely event of a release of radioactivity (iodine or particulates) from the core.

It was shown as a result of the TMI accident that the iodine release (which tends to cause the major accident consequences) was greatly reduced over what was previously anticipated due to scrubbing.

2.2.2.2 Containment Capability and Response

The primary containment has been designed to withstand the internal pressures resulting from the design basis accidents. The design pressure of the Mark II LGS containment is 55 psig.

The Mark II containment design incorporates reinforced concrete exterior walls (6 feet thick) with a steel liner. The reactor vessel is supported by a concrete pedestal which rises from the concrete basemat. The containment drywell is separated from the wetwell by a 3-1/2-foot thick concrete diaphragm floor.

There are some postulated low probability accidents, beyond the design basis, which may lead to pressures above the design pressure. These accident sequences are discussed in Section 3. An analysis of the ultimate capability of the LGS Mark II containment concluded that pressures up to 140 psig can be withstood (see Appendix J).

2.2.2.3 Inerted Containment

During most of the time that the unit is in operation, the primary containment is maintained inerted with nitrogen such that the oxygen concentration is less than four percent. Thus, should an accident occur that results in the formation of hydrogen from zirconium-water reactions, the resulting mixture would not be explosive or even combustible (oxygen is not released in this reaction or in core-concrete reactions, both of which tend to be oxygen absorbing not producing).

2.2.3 Shutdown System

Each Limerick unit uses 185 control rods, each with independent control rod drives and hydraulic control units. Each rod may be rapidly inserted by accumulator pressure or by the reactor pressure with a force many times that of gravity.

A separate and diverse shutdown mechanism is provided by the Standby Liquid Control system -- a redundant system for injecting sodium pentaborate into the reactor coolant for neutron absorption and reactor shutdown.

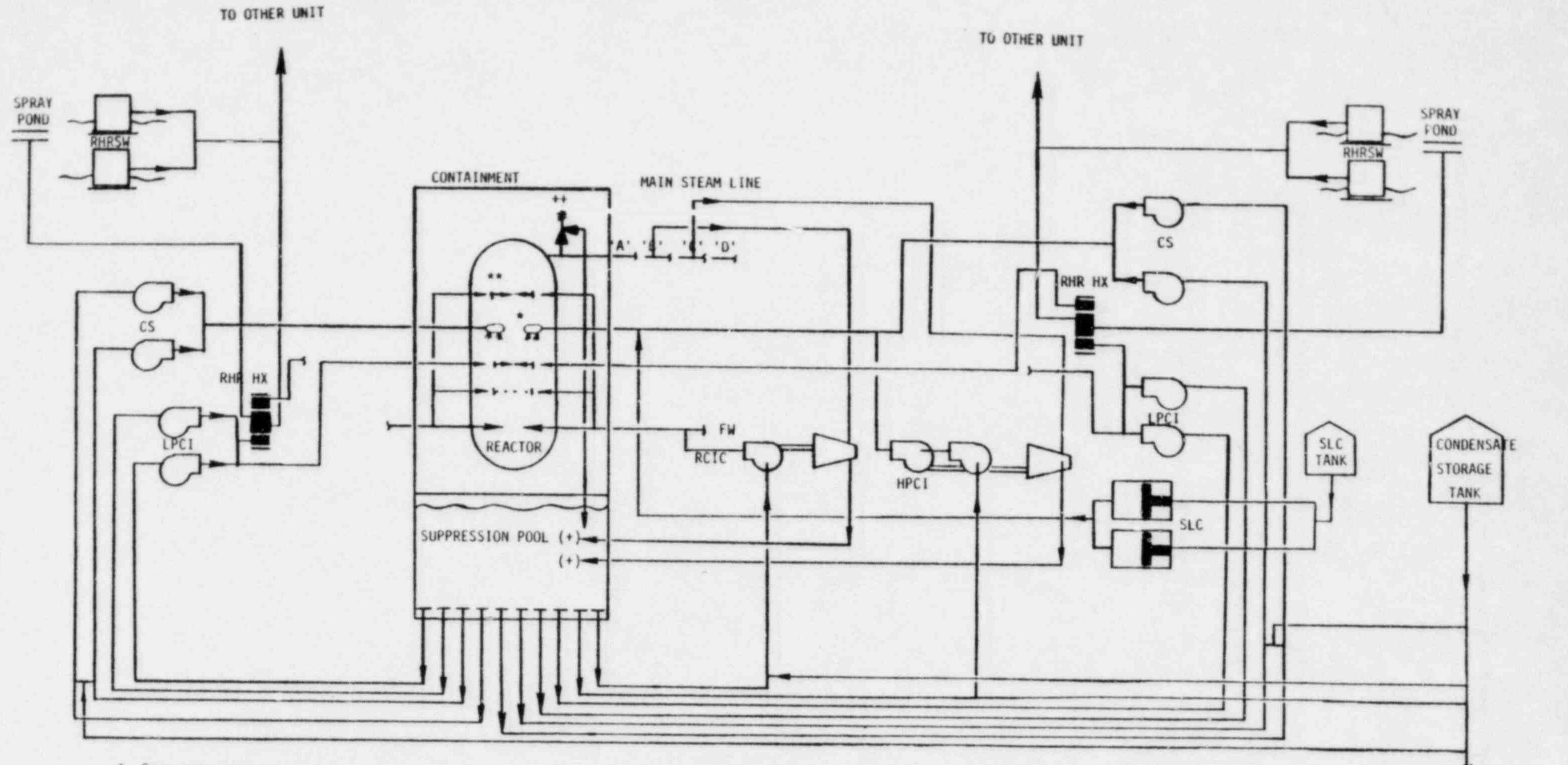
2.2.4 Emergency Core Cooling Systems (See Figure 2.2.2)

The emergency cooling systems are discussed in Appendix B Their principal features, as used in the Limerick PRA, include:

- Coolant inventory makeup water
- Heat removal capability

The High Pressure Coolant Injection (HPCI) system, the Reactor Core Isolation Cooling (RCIC) system*, the Low Pressure Coolant Injection

*This system is not an Engineered Safety Feature, but is important in regards to coolant inventory makeup during an accident condition.



* Core spray sparger
** Feedwater sparger
++ Safety/Relief valve path (reactor to suppression pool)
\+ HPCI/RCIC turbine exhaust

Figure 2.2.2 Schematic of the LGS (BWR/4) Systems Used to Provide:

- Coolant Injection (HPCI, RCIC, LPCI, FW, CS)
- Containment Heat Removal (RHR)
- Poison Injection (SLC)

(LPCI) system (part of the Residual Heat Removal system), and the Core Spray (CS) system all provide inventory makeup for a variety of accident conditions. Decay heat removal is provided by the Residual Heat Removal system (RHR).

The systems can be divided into high and low pressure systems. The HPCI and RCIC maintain RPV water inventory following small breaks or transients which do not depressurize the reactor vessel (i.e., RPV at high pressure). The remaining systems can only operate as coolant injection systems at lower RPV pressure. If the HPCI and RCIC are inadequate for coolant inventory makeup, and if feedwater is unavailable, then the RPV must be depressurized in order to allow the low pressure systems to supply makeup to the reactor. The Automatic Depressurization System (ADS) reduces reactor pressure so that the LPCI or CS systems can actuate.

Containment heat removal is provided by the RHR system (RHR pumps and the RHR service water pumps) which removes heat from the suppression pool, and rejects it to the spray pond.

2.2.5 Ultimate Heat Sink

The ultimate heat sink for the Limerick Station is a spray pond. Pumped makeup water is provided from the Schuylkill River, but is not required for at least 30 days following an accident.

2.2.6 Electrical Power

2.2.6.1 General System Description

The electrical power systems of the Limerick Generating Station (LGS) are designed to generate and transmit electric power into the Pennsylvania-New Jersey-Maryland (PJM) power network.

The two independent offsite electric power source connections to LGS are designed to provide reliable power sources for plant auxiliary loads and the engineered safeguard loads such that any single failure can affect only one power supply and cannot propagate to the alternate source. A third independent offsite source, available as a potential source for emergency use, can be connected to supply the engineered safeguard loads in the event of the loss of one of the connected offsite power sources.

The onsite ac electric power system consists of Class 1E and non-Class 1E power systems. The two offsite power systems provide the preferred ac electric power to all Class 1E loads. One source is the 220-13 kV startup transformer in the 220 kV substation. The second source is from a 13kV tertiary winding of the 220-500 kV bus-tie auto-transformer in the 500 kV substation. In the event of total loss of offsite power sources, eight onsite independent diesel-generators (four diesel-generators per unit) provide the standby power for all engineered safeguard loads.

The non-Class 1E ac loads are normally supplied through the unit auxiliary transformer from the main generator. However, during plant startup, shutdown, and post-shutdown, power is supplied from the offsite power sources through the 220-13 kV startup transformer and the 220-500 kV bus-tie auto-transformer.

Onsite Class 1E and non-Class 1E dc systems supply all dc power requirements of the plant.

2.2.6.2 Utility Power Grid and Offsite Power Systems

The LGS generator is connected by a separate isophase bus to its main step-up transformer bank. The LGS main step-up transformer bank, with three single-phase power transformers, steps up the 22 kV generator voltage to 220 kV. The 220 kV and 500 kV substations each utilize a breaker and one-half scheme arranged in an interior main bus hopover design. Each substation has three elements initially and is arranged for future expansions to four or more elements. The substations are approximately 2150 feet apart and are interconnected by a 500-200 kV bus tie transformer and transmission line. The 500 kV substation feeds two substations on the Philadelphia Electric Company system, Whitpain and Peach Bottom, which are part of the Keystone 500 kV grid. Both 500 kV substations and the 220 kV Plymouth Meeting and North Wales substations are tied into the PJM Interconnection.

The 33 kV third offsite source to LGS is made available from the Cromby-Moser 33 kV tie line. The Moser substation receives bulk power from the Cromby Generating Station and is tied to a 33 kV distribution system.

Plant startup power which is the preferred power for the engineered safeguard systems is provided from two independent offsite power sources. The power for the engineered safeguard systems can also be provided from the third independent offsite source. The three sources are as follows:

- 220-13 kV transformer connected to the 220 kV substation
- A 13 kV tertiary winding on the 500-220 kV bus tie autotransformer
- 33/13.2-4.16 kV transformer for connections to the 33 kV Cromby-Moser tieline.

The Perkiomen Pumping Station receives power from two 33 kV transmission circuits to supply power to the makeup water pumps and their auxiliaries.

The transmission system, including the 220 kV line to LGS main transformer and the two offsite power lines to the startup sources, is to be operational before LGS fuel load.

2.2.6.3 Onsite Power Systems

The onsite power system for each unit is divided into two major categories:

1. Class 1E Power System: The Class 1E power system supplies all Class 1E loads and other loads that are needed for safe and orderly shutdown and maintaining the plant in a safe shutdown condition.

   The Class 1E power system for each unit consists of four independent channels, A, B, C, and D, which provide power to four divisions of Class 1E loads.

2. Non-Class 1E Power System: The non-Class 1E onsite power system supplies electric power to nonsafety-related plant auxiliary loads. The non-Class 1E auxiliary system distributes power at 13.2 kV, 2.3 kV, 440V, and 208/120V voltage levels. These distribution levels are grouped into two symmetrical bus systems emanating from the 13.2 kV level.

2.3 LIMERICK PLANT DESIGN CHARACTERISTICS

Tables 2.3.1 through 2.3.7 list the major design characteristics of the Limerick Generating Station. Table 2.3.8 summarizes the more significant safety related features of the LGS.

Table 2.3.1

LIMERICK NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

DESIGN CHARACTERISTIC	VALUE
A. THERMAL AND HYDRAULIC DESIGN (See FSAR Section 4.4)	
1. Rated power, MWt	3293
2. Design power, MWt (ECCS design basis)	3458
3. Steam flow rate, lb/hr	14.156×10^6
4. Core coolant flow rate, lb/hr	100×10^6
5. Feedwater flow rate, lb/hr	14.117×10^6
6. Feedwater temperature, °F	420
7. System pressure, nominal in steam dome, psia	1020
8. Average power density, kW/liter	48.7
9. Maximum linear heat generation rate, kW/ft	13.4
10. Average linear heat generation rate, kW/ft	5.34
11. Maximum heat flux, Btu/hr-ft^2	361,600
12. Average heat flux, Btu/hr-ft^2	143,700
13. Maximum UO_2 temperature, °F	3435
14. Average volumetric fuel temperature, °F	2130
15. Average fuel rod surface temperature, °F	566
16. Minimum critical power ratio (MCPR)	1.29
17. Coolant enthalpy at core inlet, Btu/lb	526.1
18. Core maximum exit voids within assemblies, %	77.2
19. Core average exit quality, % steam	14.1
20. Design power peaking factor	
a. Maximum relative assembly power	1.4
b. Local peaking factor	1.15
c. Axial peaking factor	1.4
d. Total peaking factor	2.25
B. NUCLEAR DESIGN (FIRST CORE) (See FSAR Section 4.3)	
1. Water/UO_2 volume ratio (Cold, BOC)	2.74
2. Reactivity with strongest control rod out, k_{eff}	<0.99
3. Dynamic void coefficient (EOC)	
a. At core average voids, %	39.7
b. At rated output, k/%	-7.48
4. Fuel temperature droppler coefficient (BOC) at rated output (1/k) (dk/dt) (1/°C)	-1.35×10^{-5}
5. Initial average U-235 enrichment wt. %	1.88
6. Initial cycle discharge exposure, MWd/short ton	9600
C. CORE MECHANICAL DESIGN (See FSAR Sections 4.2 and 4.6)	
1. Fuel assembly	
a. Number of fuel assemblies	764
b. Fuel rod array	8x8
c. Overall length, in.	176
d. Weight of UO_2 per assembly, lb (pellet type)	456
e. Weight of fuel assembly, lb	679

Table 2.3.1
(Continued)

DESIGN CHARACTERISTIC	VALUE
2. Fuel rods	
a. Number per fuel assembly	62
b. Outside diameter, in.	0.483
c. Cladding thickness, in.	0.100
d. Diametral gap, pellet to cladding, in.	0.009
e. Length of gas plenum, in.	9.48
f. Cladding material	Zircaloy-2
g. Cladding process	Free standing loaded tubes
3. Fuel pellets	
a. Material	UO_2
b. Density, % of theoretical	94
c. Diameter, in.	0.410
d. Length, in.	0.410
4. Fuel channel	
a. Overall length, in.	166.9
b. Thickness, in.	0.100
c. Cross-section dimensions, in.	5.48x5.48
d. Material	Zircaloy-4
5. Core assembly	
a. Fuel weight as UO_2, lb	348,939
b. Core diameter (equivalent), in.	187.1
c. Core height (active fuel), in.	150
6. Reactor control system	
a. Method of variation of reactor power	Movable control rods and variable forced coolant flow
b. Number of movable control rods	185
c. Shape of movable control rods	Cruciform
d. Pitch of movable control rods, in.	12.0
e. Control material in movable rods	B_4C granules compacted in SS tubes
f. Type of control rod drives	Bottom entry locking piston
g. Type of temporary reactivity control for initial core	Burnable poison; gadoliniaurania fuel rods
7. Incore neutron instrumentation	
a. Total number of LPRM detectors	172
b. Number of incore LPRM penetrations	43
c. Number of LPRM detectors per penetration	4
d. Number of SRM penetrations	4
e. Number of IRM penetrations	8
f. Total nuclear instrument penetrations	55

Table 2.3.1
(Continued)

DESIGN CHARACTERISTIC	VALUE
g. Range (and number) of detectors:	
• Source range monitor	Shutdown through criticality (4)
• Intermediate range monitor	Prior to criticality to low power (8)
• Power range monitors	1% to 125 power
- Local power range monitor	172
- Average power range monitor	6
h. Number and type of incore neutron sources	7; Sb-Be
D. REACTOR VESSEL DESIGN (See FSAR Section 5.3)	
1. Material	Low alloy steel/stainless clad
2. Design pressure, psia	1250
3. Design temperature, °F	575
4. Inside diameter, ft-in.	20-11
5. Inside height, ft-in.	72-11
6. Minimum base metal thickness (cylindrical section), in.	6.187
7. Minimum cladding thickness, in.	1/8
E. REACTOR COOLANT RECIRCULATION DESIGN (See FSAR Section 5.4)	
1. Number of recirculation loops	2
2. Design pressure	
a. Inlet leg, psig	1250
b. Outlet leg, psig	1500
3. Design temperature, °F	575
4. Pipe diameter, in.	28
5. Pipe material, ANSI	316
6. Recirculation pump flow rate, gpm	45,200
7. Number of jet pumps in reactor	20
F. MAIN STEAM LINES (See FSAR Section 5.4)	
1. Number of steam lines	4
2. Design pressure, psig	1250
3. Design temperature, °F	575
4. Pipe diameter, in.	26
5. Pipe material	Carbon steel

Table 2.3.2

LGS ENGINEERED SAFETY FEATURES AND AUXILIARY SYSTEMS DESIGN CHARACTERISTICS

DESIGN CHARACTERISTIC	VALUE
A. EMERGENCY CORE COOLING SYSTEMS (Systems sized on design power) (See FSAR Section 6.3)	
1. Core spray system	
Number of loops	2
Flow rate, gpm, per loop (two pumps per loop)	6350 at 105 psid
2. High pressure coolant injection system	
Number of loops	1
Flow rate, gpm	5600
3. Automatic depressurization system	
Number of relief valves	5
4. Low pressure coolant injection	
Number of loops	4
Number of pumps	4
Flow rate, gpm/pump	10,000 at 20 psid
B. AUXILIARY SYSTEMS (See FSAR Sections 5.4 and 9.2)	
1. Residual heat removal system	
a. Reactor shutdown cooling mode:	
Number of pumps	2
Flow rate, gpm/pump	10,000
Duty, Btu/hr/heat exchanger	41.6×10^6
Number of heat exchangers	2
b. Primary containment cooling mode:	
Flow rate, gpm/heat exchanger	10,000
2. Service water systems	
Flow rate, gpm/heat exchanger	12,000
Number of pumps	3
3. Reactor core isolation cooling system	
Flow rate, gpm	625 at 1120 psid
4. Fuel pool cooling and cleanup system	
Capacity, Btu/hr	11.25×10^6

Table 2.3.3

LGS POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

DESIGN CHARACTERISTIC	VALUE
A. TURBINE-GENERATOR (See FSAR Section 10.2)	
Design power, MWe (gross)	1138
Generator speed, rpm	1800
Design steam flow, lb/hr	14.85×10^6
Inlet pressure, psig	950
B. STEAM BYPASS SYSTEM (See FSAR Section 10.4.4)	
Capacity, % design steam flow	25
C. MAIN CONDENSER (See FSAR Section 10.4.1)	
Heat removal capacity, Btu/hr	7800×10^6
D. CIRCULATING WATER SYSTEM (See FSAR Section 10.4.5)	
Number of pumps	4
Flow rate, gpm/pump	113,000
E. CONDENSATE AND FEEDWATER SYSTEMS (See FSAR Section 10.4.7)	
Design flow rate, lb/hr	14.885×10^6
Number of condensate pumps	3
Number of condensate booster pumps	-
Number of feedwater pumps	3
Condensate pump drive	AC power
Booster pump drive	-
Feedwater pump drive	Turbine

Table 2.3.4

LGS CONTAINMENT DESIGN CHARACTERISTICS

DESIGN CHARACTERISTICS	LIMERICK
A. PRIMARY CONTAINMENT (See FSAR Section 3.8)	
Type	Pressure Suppression
Construction	Concrete with steel liner
Drywell	Frustum of cone upper portion
Pressure-suppression chamber	Cylindrical lower portion
Pressure-suppression chamber internal design pressure, psig	55
Pressure-suppression chamber external design pressure, psi	5
Drywell internal design pressure, psig	55
Drywell external design pressure, psi	5
Drywell free volume, ft^3	248,700
Pressure-suppression chamber free air volume, ft^3	149,425 (high water) 161,350 (low water)
Pressure-suppression pool water volume, ft^3	130,825 (max) 118,655 (min)
Submergence of vent pipe below pressure pool surface, ft.	12' 3" (High water) 10 (Low water)
Design temperature of drywell, °F	340
Design temperature of pressure-suppression chamber, °F	220
Downcomer vent pressure loss factor	2.5
Break area/total vent area	0.0194
Calculated maximum pressure after blowdown to drywell, psig	47.6
Pressure-suppression chamber, psig	28.2
Initial pressure-suppression pool temperature rise, °F	43
Leakage rate, % free volume/day	0.5
B. SECONDARY CONTAINMENT (See FSAR Section 3.8)	
Type	Controlled leakage, roof level release
Construction	
Lower levels	Reinforced concrete
Upper levels	Reinforced con-concrete super-structure and siding
Roof	Reinforced concrete
Internal design pressure, psig below atmospheric	0.25
Design in leakage rate, % free volume/day at 0.25 in. H_2O	100

Table 2.3.5

LGS STRUCTURAL DESIGN CHARACTERISTICS

DESIGN CHARACTERISTICS	LIMERICK
A. SEISMIC DESIGN (See FSAR Section 3.7)	
Operating basis earthquake	
- horizontal g	0.075
- vertical g	0.05
Safe shutdown earthquake	
- horizontal g	0.15
- vertical g	0.10
B. WIND DESIGN (See FSAR Section 3.3)	
Maximum sustained - mph	80
C. TORNADO DESIGN (See FSAR Section 3.3)	
Translational - mph	60
Tangential - mph	300

Table 2.3.6

LGS RADIOACTIVE WASTE MANAGEMENT SYSTEMS DESIGN CHARACTERISTICS

DESIGN CHARACTERISTICS	LIMERICK
A. GASEOUS RADWASTE (See FSAR Section 11.3)	
Design bases, noble gases, μc/sec	100,000 at 30 min.
Process treatment	Recombiner Charcoal Delay
Design condenser in leakage, cfm	75
Release point-height above ground, ft.	197
B. LIQUID RADWASTE (See FSAR Section 11.2)	
Treatment of:	
1. Floor drains	F, D, R
2. Equipment drains	F, D, R
3. Chemical drains	E, D concentrates to solid radwaste, distilate R
4. Laundry drains	F, C

Legend:

D = demineralized
F = filtered
E = evaporator/concentrator
R = recycled i.e., returned to concentrate storage
O = discharged

Table 2.3.7

LGS ELECTRICAL POWER SYSTEMS DESIGN CHARACTERISTICS

DESIGN CHARACTERISTICS	LIMERICK
A. TRANSMISSION SYSTEM (See FSAR Section 8.2)	
Outgoing lines, number - rating	3-500 kV 2-230 kV
B. NORMAL AUXILIARY AC POWER (See FSAR Sections 8.2 and 8.3)	
Incoming lines, number - rating	3-500 kV 2-230 kV
Auxiliary transformers (No.)	2
Startup transformers (No.)	2
Safeguard transformers (No.)	2
C. STANDBY AC POWER SUPPLY (See FSAR Section 8.3)	
Number of diesel-generators	4/unit
Number of 4160V shutdown buses	4/unit
Number of 480V shutdown buses	4 (ESF)
D. DC POWER SUPPLY (See FSAR Section 8.3)	
Number of 125V or 250V batteries	4-125V 4-125V/250V 2-250V
Number of 125V buses	24
Number of 250V buses	12

Table 2.3.8
LIMERICK SAFETY RELATED DESIGN FEATURES

MK II Reinforced Concrete Steel-lined Containment
Large Standby Gas Treatment System
Containment Overpressure Relief
High Quality and Large Number of Safety/Relief Valves
AISI 316 Reactor Piping
Highly Reliable Shutdown System (ATWS Alternate 3A)
Spray Pond for Emergency Cooling Water
No NPSH Requirement for Emergency Pumps
Four Dedicated Emergency Diesel Generators
Highly Reliable Offsite Power (Five Sources)

Section 3

ACCIDENT ANALYSES: STATE OF THE ART LIMERICK CALCULATION

In this section, each of the principal aspects of the analysis are summarized together with the results. These include the following:

1. Probabilistic Analysis
   - Sources and Mobility of Radioactive Material (Section 3.1)
   - Accident Initiators (Section 3.2)
   - Radioactive Release Sequence Classes (Section 3.3)
   - Accident Sequence Event Trees (Section 3.4)
   - Quantification of Event Tree Sequences (Section 3.5)
2. In-plant Radioactive Release Fractions (Section 3.6)
3. Offsite Consequence (Section 3.7)
4. Uncertainties (Section 3.8).

Additional details can be found in the appendices.

The analyses presented in this section are based upon the event tree/fault tree methodology used in WASH-1400. However, some details of the methodology implementation are different. The differences are discussed in detail in Section 4, they are generally due to the use of:

- A plant-specific model for the LGS plant and site
- Updated methodology, data, and core melt phenomenology.

3.1 SOURCES AND MOBILITY OF RADIOACTIVE MATERIAL

The PRA methodology applied to the analysis of LGS treats all aspects of those accident scenarios associated with the postulated release of radioactivity into the environment. The scenarios are developed first by identifying the sources and mobility of radioactivity inside the plant, then identifying the processes by which significant amounts of this radioactivity can be released to the environment outside the containment building.

The sources of radioactivity at the Limerick Generating Station include the following:

- Reactor core
- Spent fuel storage pool
- Liquid radwaste storage tanks
- Off gas treatment system
- Spent fuel shipping casks
- Miscellaneous contaminated equipment and radiography sources.

Table 3.1, which is taken from WASH-1400, gives an approximate measure of the potential radioactive sources associated with a typical 1000 MWe LWR.

The fresh uranium dioxide pellets that serve as the fuel are only slightly radioactive, but the fission process occurring during reactor operation produces large amounts of radioactive nuclides (fission products) in the fuel. Also, the structural materials and coolant become irradiated during power operation and therefore have some induced radioactivity. This induced radioactivity is immobile and represents only a minute fraction of the total radioactivity. Therefore, it is not important in assessing the overall risk to the public.

Table 3.1

TYPICAL RADIOACTIVITY INVENTORY FOR A
1000 MWe NUCLEAR POWER REACTOR

Location	Total Inventory (Curies)			Fraction of Core Inventory		
	Fuel	Gap	Total	Fuel	Gap	Total
Core (a)	[illegible].0 x 10^9	1.4 x 10^8	8.1 x 10^9	9.8 x 10^{-1}	1.8 x 10^{-2}	1
Spent Fuel Storage Pool (Max.) (b)	1.3 x 10^9	1.3 x 10^7	1.3 x 10^9	1.6 x 10^{-1}	1.6 x 10^{-3}	1.6 x 10^{-1}
Spent Fuel Storage Pool (Avg.) (c)	3.6 x 10^8	3.8 x 10^6	3.6 x 10^8	4.5 x 10^{-2}	4.8 x 10^{-4}	4.5 x 10^{-2}
Shipping Cask (d)	2.2 x 10^7	3.1 x 10^5	2.2 x 10^7	2.7 x 10^{-3}	3.8 x 10^{-5}	2.7 x 10^{-3}
Refueling (e)	2.2 x 10^7	2 x 10^5	2.2 x 10^7	2.7 x 10^{-3}	2.5 x 10^{-5}	2.7 x 10^{-3}
Liquid Waste Storage Tank	-	-	9.5 x 10^1	-	-	1.2 x 10^{-8}

(a) Core inventory based on activity 1/2 hour after shutdown.

(b) Inventory of 2/3 core loading; 1/3 core with three day decay and 1/3 core with 150 day decay.

(c) Inventory of 1/2 core loading; 1/6 core with 150 day decay and 1/3 core with 60 day decay.

(d) Inventory for one fuel assembly with three day decay.

(e) Inventory for one fuel assembly with three day decay.

The transfer of spent fuel assemblies from the reactor core, where essentially all the radioactivity in LGS is initially created, results in radioactive fission products being located in other parts of the plant, such as the spent fuel storage pool (SFSP). Taking into consideration the length of power operation and radioactive decay, the reactor core contains by far the largest inventory of radioactive material among all the sources in LGS.

Accidents which involve release of radioactivity from the refueling process, shipping casks, or the liquid radwaste would not result in public consequences nearly as serious as accidents involving melting of the fuel in the reactor core or in the SFSP. The melting of the fuel in the SFSP is assumed (as in WASH-1400) to be of sufficiently low probability as to not require detailed examination in the study.

Brookhaven National Laboratory (BNL) has determined, in a study for the Nuclear Regulatory Commission (NRC), that the highest risk to the public from operation of a nuclear power plant is presented by postulated accidents involving the reactor core (3-1). The other smaller sources of radioactivity at a nuclear power plant are found on a generic basis by BNL to make a negligible contribution to the public risk.

The Limerick Probabilistic Risk Assessment has focused on the potential for the release of radioactivity to the public by accidents involving the reactor core, and taking place during power operation.

3.2 ACCIDENT INITIATORS

Event trees provide a logical method for developing and displaying the sequences which may occur during postulated accidents. One of the most important aspects of this technique is that it ensures that all of the key accident initiators are identified. Items considered when identifying accident initiators include the following:

- Previous risk analyses (e.g., WASH-1400, CRBRP, IREP)
- Plant unique features leading to specific initiators
- Operating experience
- Licensing basis accidents.

See Appendix A for further discussion of accident initiators. The initiating events can be separated into three general groups:

1. Transients and manual shutdowns

2. Loss of Coolant Accidents (LOCAs)

3. Anticipated Transients Without Scram (ATWS).

Table 3.2.1 summarizes the transient frequencies used in the Limerick analysis. Evaluation of BWR operating experience data indicates that there are approximately 6.2 transients per year which result in a scram shutdown. As shown in the table, different transients are discriminated by type into five distinct groups:

1. Isolation events (MSIV Closure)

2. Turbine trips

3. Loss of offsite power

4. Inadvertent open relief valves (IORV)

5. Loss of feedwater.

Table 3.2.2 summarizes the frequency of pipe failures used in the Limerick analysis to characterize the LOCA accident initiators as a function of pipe size. The details of the development of these values are given in Appendix A. The pipe failures are not to be construed as relating to the licensing basis LOCA sizes, but rather are classified according to the volume rate of water makeup which would be required to successfully mitigate the accidents; that is, the numerical value of the pipe failure frequencies are derived to represent any pipe failure which demands various levels of ECCS equipment to operate (divided among small, medium, and large LOCAs). The available data by pipe size are representative of, and were used for the three LOCA categories analyzed in the LGS PRA (see Table 1.2).

The frequency of ATWS initiation was taken from NUREG-0460 to be 3×10^{-5} per demand. It is felt that this value is conservative since a recent GE analysis indicates a substantially smaller number may be appropriate.

Table 3.2.1

SUMMARY OF THE FREQUENCY OF TRANSIENT INITIATORS AND THE CATEGORIES INTO WHICH THEY HAVE BEEN CONSOLIDATED

TRANSIENT	FREQUENCY (Per Reactor Year)
MSIV Closure	1.08
Closure of all MSIVs	1.00
Turbine Trip Without Bypass	0.01
Loss of Condenser	0.067
Turbine Trip	3.98
Partial Closure of MSIVs	0.20
Turbine Trip with Bypass	1.33
Startup of Idle Recirculation Loop	0.25
Pressure Regulator Failure	0.67
Inadvertent Opening of Bypass	0.00
Rod Withdrawal	0.10
Disturbance of Feedwater	0.68
Electric Load Rejection	0.75
Loss of Offsite Power	.38
Inadvertent Open Relief Valve	.06
Loss of Feedwater	.70
TOTAL	6.2
MANUAL SHUTDOWNS	3.2

Table 3.2.2

EVALUATED FREQUENCY OF PIPE FAILURE IN A BWR BASED UPON OPERATING EXPERIENCE DATA

PIPE SIZE	FREQUENCY (Per Reactor Year)
Large Pipe $\geq 4"$ Diam.	4.0×10^{-4}
Medium Pipe $< 4"$ Diam. $\geq 1"$ Diam.	2.0×10^{-3}
Small Pipe $< 1"$ Diam.	1.0×10^{-2}

3.3 RADIOACTIVE RELEASE SEQUENCE CLASSES

This section and subsequent sections (3.4 and 3.5) emphasize one of the principal differences between the WASH-1400 and Limerick analyses. This difference lies in the method of calculating the consequences associated with a given accident sequence. There are two driving forces in determining this treatment:

1. The time, manpower, and cost of the consequence evaluation for each accident sequence
2. The necessity of determining the dominant sequences first and then calculating the consequences of those sequences. This makes the entire process a set of serial procedures and extends the time before an answer can be produced. Decoupling of the dominant sequence determination from the consequence calculation is therefore desirable.

For WASH-1400 the method chosen to deal with the above two items for BWRs was to divide the consequences of all accident sequences into categories determined almost solely by virtue of the containment failure mode. Therefore there were five radioactive release categories which required characterization by radionuclide release fractions, time of release, and energy of release. Nearly all accident sequences which are postulated to lead to core melt were treated equivalently in terms of consequences. Specifically, this means that ATWS sequences, LOCAs, and transients with loss of decay heat removal were all treated similarly in terms of their effect on the containment and the associated radionuclide release to the environment. In order to ensure that the effects of lumping all sequences together did not underestimate the risk, the artifice referred to as "smoothing" was used in WASH-1400 to place a certain fraction of the probability associated with each category in adjacent categories.

The Limerick analysis has taken advantage of the many efforts over the past five years in accident sequence evaluation and consequence calculation to more precisely associate accident sequence type with the potential consequence it yields.

Because a number of accident sequences are similar in both their impact on containment and their potential for release of radioactive material, these similar accident sequences have been combined into classes representative of the types of accident sequences leading to core melt.

Table 3.3.1 describes the type of event sequences which are described by the four generic accident classes used in the Limerick analysis.

Table 3.3.1

GENERIC ACCIDENT SEQUENCE CLASSES

Generic Accident Sequence Designator	Physical Basis for Classification	System Level Contributing Event Sequence
Class I (C1)	Relatively fast core melt; containment intact at core melt and at low pressure	Transients involving loss of inventory makeup, small LOCA events involving loss of inventory makeup
Class II (C2)	Relatively slow core melt due to lower decay heat power; containment failed prior to core melt	Transients or LOCAs involving loss of heat removal, inadvertent SRV opening accidents with inadequate heat removal capability
Class III (C3)	Relatively fast core melt; containment intact at core melt, but at high internal pressure	Transients involving loss of scram function and inability to provide coolant makeup, large LOCAs with insufficient coolant makeup transient with loss of heat removal and long term loss of inventory makeup
Class IV (C4)	Relatively fast core melt; containment fails prior to core melt due to over-pressure	Transients involving loss of scram function and loss of containment heat removal or all reactivity control, but which have coolant makeup capability

The containment effects associated with one particular accident sequence for each class are calculated and used to characterize the response for that class. Event sequences are then grouped by similarity and are treated as having the same impact on containment, and the same radioactive release terms.

Since the Limerick analysis is performed with much greater definition in accident sequence consequence evaluation there is little or no justification for "smoothing" in the Limerick analysis. Instead, each coupled set of accident class and containment failure mode is calculated explicitly with CRAC (see Section 3.7) rather than by force fitting accident sequences with different times to core melt, different release fractions, and different containment failure times into the same category. Section 3.5.5 summarizes the probabilities associated with each class for each containment failure mode.

3.4 EVENT TREES USED IN THE PROBABILISTIC ANALYSIS

Event trees are used to present those accident sequences which may result from a specific initiating event. The philosophy used in the LGS analysis is to develop and quantify separate event trees for those initiating events which would have a strong effect on the systems available for accident mitigation. Using this guideline, event trees are developed for the accident initiators discussed in Section 3.2 and Appendix A. The event trees include:

1. Transient Event Trees (Section 3.4.1)
   - Turbine Trip
   - Manual Shutdowns
   - MSIV Closure/Loss of Feedwater
   - Loss of Offsite Power
   - Inadvertent Open Relief Valve
2. Loss of Coolant Event Trees (Section 3.4.2)
   - Large LOCA
   - Medium LOCA
   - Small LOCA

3. Event Trees for Low Probability Events (Section 3.4.3)

   - ATWS Event Trees

   - Interfacing LOCA

4. A "bridge" event tree which displays the key scenarios for sequences where containment overpressure may occur prior to any core damage (Section 3.4.4).

5. The containment event tree applicable in the unlikely event that a severe disruption of the core occurs (Section 3.4.5).

Differences in potential consequences to the public are differentiated by the success or failure of the functions identified in the event trees. Items which affect the functions but do not change the level of consequences are addressed in the fault tree models. In other words, the fault tree models of each system contain all the identified modes of system unavailability due to hardware failure, human interactions, and test and maintenance. These are discussed more completely in Appendix A. System interdependencies are treated through the Boolean combination of the fault tree models using the WAM* series of computer codes (this represents an improvement over WASH-1400 where these dependencies were calculated by hand using reduced fault trees).

The event trees are used to trace the sequence paths which may be encountered following a specific accident initiator and leading to the following conditions:

- The most likely case of a stable, hot shutdown condition, or

- The unlikely case of a postulated degraded core condition, disruption, or melt.

*WAM code series developed for, and available from, EPRI (see Appendix K).

The event trees constructed for the Limerick Generating Station (LGS) risk assessment lead to several hundred accident sequences to be analyzed. It would be very time-consuming and expensive to perform an in-containment phenomenological analysis and an ex-plant consequence analysis for each of the accident sequences. However, because a number of the accident sequences which may lead to degraded core conditions are similar in their impact on containment and their potential for release of radioactive material, these similar accident sequences are grouped into classes.

The consequences of each class are then treated separately by containment failure mode and release fraction. This method leads to a much larger number of consequence groups to analyze than the five used in the WASH-1400 BWR analysis; yet, it maintains the number of calculations which need to be performed at a manageable level. This technique provides a greater specificity in accident sequence definition than used in WASH-1400.

Table 3.3.1 (see preceeding section) gives a description of each of the accident sequence classes as well as their physical containment phenomenology. For each of the event sequence classes, a detailed in-plant consequence evaluation has been performed.

3.4.1 Event Tree Analysis of Transient Events and Manual Shutdowns

The following two types of initiators are discussed in this subsection:

- Anticipated Transients
- Manual Shutdowns.

The WASH-1400 analysis has treated all transients as producing similar plant effects. The WASH-1400 assessed value of the anticipated transient initiator was ten per reactor year, which included seven scrams and three manual shutdowns per reactor year. Each of these was treated as an equivalent demand on the plant systems required to respond.

WASH-1400 has previously identified a major fraction of the risk associated with BWR operation to be associated with demands on safety systems following a transient initiator. The fact that groups of transients interact differently with the mitigating systems caused the Limerick analysis to be based on several classes of anticipated transients. These classes can be quantified using the available operating experience data (3-2, 3-3, 3-4).

The Limerick analysis considers separately those sequences initiated by a manual shutdown of the reactor. This approach is more realistic than the WASH-1400 approach which incorporated manual shutdowns into the single anticipated transient category. Because manual shutdowns are slow, controlled reductions in power, the demands on heat removal systems are generally anticipated, and adequate preparations can be made to have both the normal heat removal system and the safety-related systems available. Therefore, the Limerick analysis makes a more realistic assessment of plant reliability under this special class of initiators.

The techniques used in the event tree analysis are to define the possible paths of an accident through the success and failure states of principal plant functions. The functions in turn are defined in terms of systems and the systems in terms of components (see Appendix B). Each accident sequence event tree associated with a given initiator produces a set of sequences which are quantified. These quantified sequences are then placed into one of the following groups:

1. The core is determined to be safely shutdown, there is no release of radioactivity to the environment, and stable heat removal has been established. This is the highest probability path in all of the accident sequence event trees.

2. A sequence is determined to be of a given class (I through IV) which may lead to the release of radioactivity.

3. A sequence does not produce a release of radioactive material but does have an impact on containment. The sequence is transferred to the bridge tree for processing.

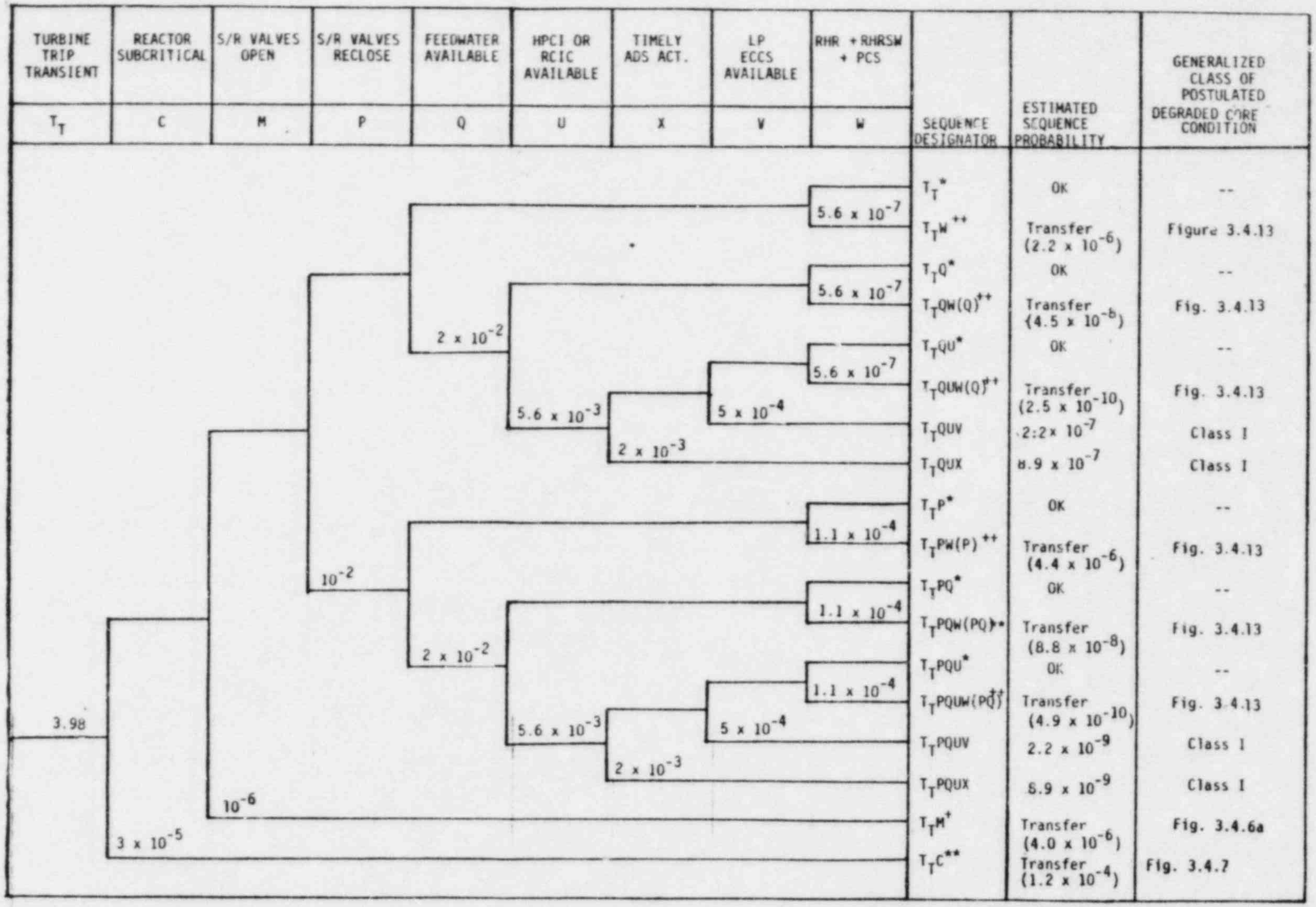
4. A sequence resulting from one accident initiator strongly resembles the initiator of another event.

The class of accident initiators, known as anticipated transients, demand control rod insertion and proper operation of the plant heat removal systems to ensure a safe shutdown and cooling of the reactor core. These transients include those listed in Appendix A.1 and have been grouped together into initiators which have similar characteristics with respect to:

- Systems available for accident mitigation
- Initial conditions
- Pressure, temperature, and power effects.

Figures 3.4.1 through 3.4.5 are the event trees describing the sequence of events or functions which affect the course of the following anticipated transient and manual shutdown accident initiators:

- Turbine Trip (T)
- Manual Shutdowns (M)
- MSIV Closure/Loss of Feedwater (F)
- Loss of Offsite Power (E)
- Inadvertent Open Relief Valve (I).



*Not core melt sequence
**ATWS initiators are treated in a separate event tree
+Transfer to large LOCA event tree
++ Transfer to bridge tree

NOTE: This figure includes manual shutdowns for the purpose of calculating the demands on long-term containment heat removal only

Figure 3.4.1 Turbine Trip Transient Event Tree

3.4.1.1 T_T -- Turbine Trip Transient

The turbine trip transient involves the least challenge to reactor shutdown systems. T_T does not, of itself, prevent the safe shutdown of the plant using either the normal heat removal systems (i.e., condenser) or the safety related systems. In the development of this sequence, those turbine trip events for which there is a failure of the turbine bypass valves to open are transferred directly to the MSIV closure event tree. Figure 3.4.1 is the turbine-trip-initiated event tree. Each of the functional events listed across the top of the event tree are discussed below.

C -- Reactor Subcritical. Failure to bring the reactor subcritical is an accident sequence leading to a transfer to the ATWS event trees (Figure 3.4.7, for turbine trip). The sequences assessed in Figure 3.4.1 are those in which control rods are successfully inserted.

M -- Safety/Relief Valves Open. This column represents the opening of the safety relief valves to limit reactor coolant pressure to 110 percent of the reactor-coolant pressure-boundary design pressure. Failure of a sufficient number of valves to open may lead to excessive pressure and a potential LOCA condition. For the most severe transient, i.e., turbine trip from high power without turbine bypass, eight of the fourteen valves are required to open to be successful.

P -- Safety/Relief Valves Reclose. The safety/relief valves that open as a result of a transient must reclose to prevent discharge of an excessive quantity of reactor coolant and heat to the suppression pool. The impact on plant safety arises from the additional heat load on the RHR system due to the stuck open relief valve(s). The additional heat load creates a demand for additional heat removal from the suppression pool via the RHR system.

Q -- Feedwater System. The feedwater system is used to maintain an adequate coolant inventory in the reactor vessel. Replenishment of the reactor inventory must be initiated within 30 minutes after the initiation of the reactor trip signal. The feedwater system is the normal method of maintaining reactor coolant inventory and therefore is expected to be available following a turbine trip. Since the feedwater pumps are turbine driven they will be available as long as there is steam available to their turbines. They can be regulated for reduced flow. Failure of this normal plant system to provide sufficient coolant makeup to the reactor would require that either the other high pressure injection systems provide makeup or that the reactor be depressurized and the low pressure systems provide the required makeup inventory.

U -- High Pressure Injection System. In addition to feedwater there are other sources of high pressure injection water to maintain reactor coolant inventory, these include:

- High Pressure Coolant Injection (HPCI)
- Reactor Core Isolation Cooling (RCIC)
- Control Rod Drive Injection (CRDI)

The CRDI is a relatively low capacity system which was not given credit in the LGS analysis, however, it does provide an additional source of injection capability which can be utilized under certain conditions. The HPCI and RCIC pumps provide the principal sources of water for maintaining core coolant inventory if feedwater is not available. Successful operation of either of these high pressure systems is considered adequate to maintain water level in the reactor following a turbine trip if initiation occurs within 30 minutes.

Lack of success of the high pressure systems may result from either:

- A failure of both high pressure systems, <u>or</u>
- A perceived failure of the systems by the operator who then initiates ADS and depressurizes the plant.

Both of these cases are considered in the fault tree development of the high pressure system availability.

<u>X -- Timely ADS Actuation</u>. In the unlikely event that there is an insufficient supply of coolant from high pressure sources, it would then be necessary for automatic or manual initiation of ADS to reduce reactor pressure below 289 psia to allow the low pressure injection systems to maintain reactor inventory. The event tree function, X, is used to illustrate that the operator must manually initiate ADS in a timely fashion. Failure to actuate ADS can be due to operator error, failure of the low pressure pumps* to start, or hardware failures within the ADS itself. The potential scenarios which could occur and lead to unacceptable core conditions include:

1. Timely ADS initiation does not occur, the core water inventory is depleted, and core melting is 50 to 90 percent complete before ADS is initiated. A cold slug of low pressure water is then injected into the vessel increasing the probability of a reactor vessel steam explosion.

2. Alternatively, the manual blowdown which is performed by opening individual valves may result in lowering the reactor pressure, causing increased voids and lower power within the core, while at the same time not reaching the setpoint to allow the low pressure injection system valves to open. Eventually the water inventory will be depleted and core melting may begin if complete depressurization does not occur.

3. The operator violates the emergency procedure guidelines and waits too long to initiate depressurization. The suppression pool temperature rises above acceptable limits due to loss of RHR, and then, per the emergency procedure guidelines, ADS is not permitted.

*ADS logic requires that one LPCI and one CS pump from the same division be operating otherwise ADS is inhibited.

The probability that the operator will correctly act to depressurize the reactor when required within thirty minutes is assigned a high probability of success based on the following:

1. Operator training emphasizes that the low pressure systems are the final source of water to maintain adequate core cooling, and that they are only useable when the reactor can be brought to low pressure.

2. The operator has abundant indication of the potential problems which may exist in the core:

   - No HPCI/RCIC/FW flow to the reactor

   - Water level in the reactor dropping on several indicators

3. Minimal other distractions which may divert his attention.

Alternate methods of depressurization are available if required:

1. Depressurize through the safety relief valves. The SRVs require an air supply from outside containment which is isolated on low reactor level, this path may require breaking isolation to provide successful depressurization. A keylocked bypass switch is provided.

2. Depressurize through the main condenser. This requires that the MSIVs be opened (a violation of containment integrity if a reactor low level signal is present). It also requires that offsite power and the condenser be available to provide a viable path.

3. Depressurize through the HPCI and/or RCIC turbines. This implies that while these units may not be able to pump to the reactor vessel, they are not isolated from the steam supply.

While these alternate features are known to be viable methods of depressurization, they involve creative operator actions under potentially stressful conditions. Such actions are difficult to quantify. These methods of depressurization are given low probability for providing successful depressurization.

Operation of three out of five ADS valves is used* to depressurize the vessel so that the low pressure systems can be used. The calculated random independent mechanical failure of three of five ADS valves is assessed to be low; however, common-mode failures which could disable three ADS valves dominate the probability of ADS failure as determined from the minimum cutsets of the ADS fault tree. The mechanical failure of ADS appears in the event tree under the event "V". There is an approximately equal contribution to the "V" event from the ADS and low pressure system unavailabilities.

<u>V -- Low Pressure ECCS</u>. This event represents failure of low pressure injection pumps to operate. The low-pressure pumps included in this evaluation are:

- RHR/LPCI (4 pumps)
- Core Spray (4 pumps)
- Condensate (4 pumps) (not an ECCS system).

<u>W -- Residual Heat Removal (RHR), RCIC Steam Condensing, Power Conversion System (PCS)</u>. For success:

1. The RHR system must provide a complete flow path from and to the reactor coolant system through at lease one RHR heat exchanger. In addition, the RHRSW system must provide cooling water to the corresponding RHR heat exchanger from the spray pond or the cooling tower basin.

2. For the PCS to successfully transfer fission product decay heat to the environment requires all of the following:

   - One complete condensate-feedwater piping system is operable and able to deliver water from the condenser hotwell to the reactor vessel. This requires that the condensate and feedwater pumps in the piping system be operable, or that the condensate pump be operable and that the operator reduces reactor pressure to below 540 psia by using the relief valves.

*Only 2 valves are really needed as shown on Table 1.2.

- The main steam line isolation valves in one of the four main steam lines must remain open (or be reopened if they closed as a result of the initiating transient). Further, the turbine bypass valve must open. If condenser vacuum falls below seven inches of Hg, the low vacuum interlocks on the bypass valves must be overridden.

- At least one of the main condenser circulating water pumps must be operable and delivering cooling water to the main condenser.

3. Heat removal via the RCIC steam condensing mode is viewed as an additional design feature which allows the operator flexibility in maintaining a safe reactor condition in the face of unusual plant occurrences. The RCIC steam condensing mode utilizes the HPCI steam lines, the RCIC turbine and pump, the RHR heat exchangers, and the RHR service water to transfer reactor decay heat to the ultimate heat sink. Because of the large number of systems required, and the dependence of this alternative on RHR systems, the RCIC steam condensing mode provides only a small improvement in the overall calculated probability of successful containment heat removal.

Heat rejection to the environment must be initiated using method (1), (2), or (3) within about 20-30 hours* after the initiating transient in order to be successful. As in WASH-1400, the Limerick analysis has treated failures of all types which disable portions of the system to be non-recoverable.

The failure of the containment heat removal (W) function following a transient initiator (T) in WASH-1400 was assumed to eventually lead to core melt. Similar logic is used in the Limerick analysis to define conditions under which an acceptable core condition cannot be assured. This assumption of loss of decay heat removal leading to a degraded core condition results from the following assumptions:

*More time available with containment overpressure relief.

1. With heat being produced, and no PCS, the reactor will continue to blowdown to the suppression pool through the safety/relief valves.

2. Without decay heat removal from containment, the suppression pool will eventually heat up, and steam will be generated in the wetwell. Pressure in containment will continue to increase.

3. Once above the containment design pressure, the containment may leak sufficiently to stabilize the pressure rise. However, there are cases evaluated probabilistically for which insufficient leakage occurs and the pressure rises toward the failure pressure.

4. The failure of the containment would lead to two potential phenomena which would compromise the ability to maintain the core covered:

   - The suppression pool may have a substantial portion of the inventory flash to steam leading to:

     - possible cavitation causing damage to the pump during the postulated containment blowdown phase

     - potential piping or valve damage due to the large steam generation rate during blowdown of containment

     - venting of steam into the reactor building, adversely affecting the switch gear, motor control centers, or instrumentation of the high pressure injection systems

   - The failure of the containment may also lead to failure of the coolant injection piping supplying water to the vessel from the hotwell, the condensate storage tank, and the suppression pool due to deflections of the reactor enclosure.

Controlled Containment Overpressure Relief (COR) to avoid containment failure significantly reduces the probability of TW leading directly to a degraded core condition. The reduction in probability of core degradation is on the order of a factor of 100 and is included in the bridge tree (Figure 3.4.13).

The detailed method of assessing containment heat removal success probabilities is through a comprehensive event tree structure reflecting the phased mission aspects of the containment heat removal function. This method provides a more formalized treatment of qualitatively and quantitatively assessing the success probability of the containment heat removal function (see Section 3.4.1.4).

W(P) -- W Given Event P Fails. This function is similar to W except there is a greater demand (less time available) for decay heat removal operation. Since a stuck open relief valve(s) (P) dumps heat directly to the suppression pool, the RHR system may be required for this event. The LGS analysis assumes that for one SORV the operator is capable of successfully removing sufficient heat via the PCS (if available) so that the RHR is not required. For multiple SORVs the RHR is assumed to be required. Also for other transient initiators, the single stuck open relief valve is treated in the LGS analysis as aggravating an accident sequence which may exist, but does not affect the success criteria for any of the identified plant functions. The failure of multiple relief valves to reclose is viewed as a slightly different problem since it leads to a demand on the RHR system to prevent containment overpressure from occurring. Therefore, when "P" fails (i.e., multiple SORVs) in the event tree, one RHR is required to operate.

W(Q) -- W Given Event Q Fails. Similar to W except that the PCS is not available for decay heat removal since the condensed steam cannot be removed from the condenser hotwell.

W(PQ) -- W Given Both Events P and Q have Failed. Similar to W(P) in that RHR system operability is required.

T_T: Qualitative Results

For turbine trip initiators with scram (ATWS events treated separately), the sequences which are identified to present the highest frequency of postulated degraded core conditions are those sequences involving:

- Failure to supply coolant inventory makeup to the reactor due to loss of feedwater, high pressure systems, and low pressure systems (T_TQUV and T_TQUX).
- Failure to adequately remove decay heat from the containment (T_TW Mode 1*).

3.4.1.2 T_M -- Manual Shutdown

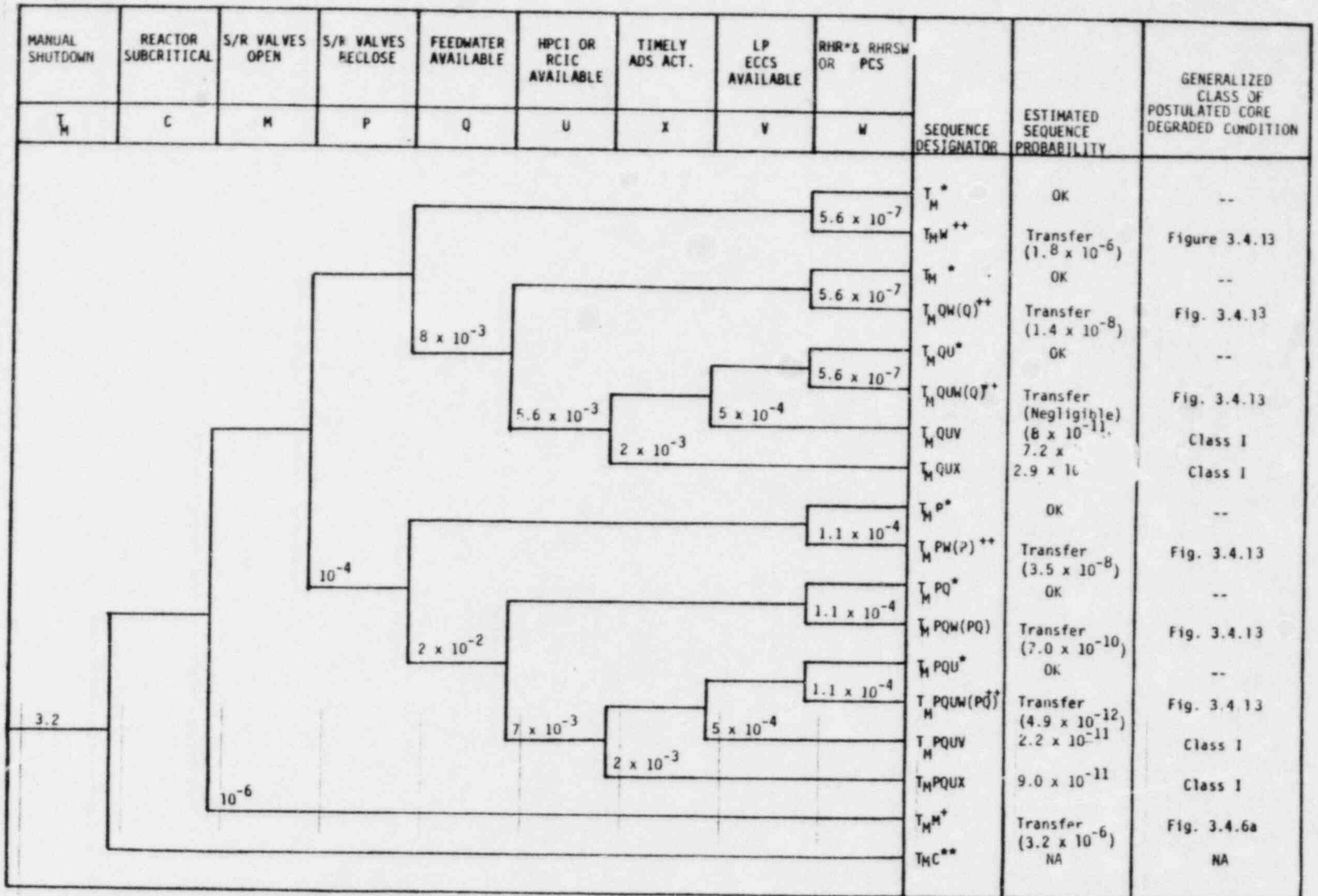
One type of challenge to the reactor systems which is included as a special category is the case of a demand associated with a controlled manual shutdown of the reactor plant. Figure 3.4.2 is the event tree used to characterize this situation. Since manual shutdowns occur with a relatively high frequency (see Appendix A.1), it is important to adequately characterize the system response required during these challenges.

T_M -- Functions in Event Tree

The discussion in Section 3.4.1.1 on turbine trip events applies to the manual shutdown case, with the following exceptions:

1. ATWS is not a problem for manual shutdowns due to the longer time available to react. Those small fraction of events which are of a nature requiring immediate shutdown are represented by turbine trip events.
2. The frequency of loss of feedwater from high power during a manual shutdown is lower than for the turbine trip transient. Therefore, the probability of TQUV sequences (loss of core coolant injection) is lower in the manual shutdown case than in the turbine trip transient case. Even if feedwater is tripped during the power rundown, it is possible to restore the feedwater capability with a high probability.
3. The options available to remove decay heat from the reactor are more reliable during a slow, controlled shutdown than during a transient demand. Specifically, the PCS is available during the shutdown, therefore the probability of successful heat removal through the PCS is high. There is some possi-

*Mode 1 is the highest probability sequence for TW sequences and is discussed in Section 3.4.



*Not core melt sequence
**ATWS is judged not to be risk contributor for manual shutdowns.
+Transfer to large LOCA event tree
++Transfer to bridge tree

Figure 3.4.2 Manual Shutdown Event Tree

bility of a commonality between the requirements for shutdown and the unavailability of RHR, that is, the plant technical specifications require plant shutdown if one RHR loop is unavailable for more than seven days. In such cases, the reason for the plant shutdown also eliminates one of the methods of containment heat removal. This situation can be expected to occur (based upon historical operating experience) with a frequency of 8.7×10^{-3}/reactor year.* Using this initiator frequency and the availability of the remaining single loop of the RHR, the estimated frequency of reactor shutdown without RHR is approximately 1.5×10^{-4}/reactor year. This compares with the frequency for other accident initiators of $\sim6\times10^{-4}$/reactor year. The numerical contribution of this type of sequence is small, but this is a situation which requires specific operating procedures and careful operator action in using the PCS during the controlled shutdown.

3.4.1.3 T_E -- MSIV Closure/Loss of Feedwater/Loss of Condenser Transient (See Figure 3.4.3)

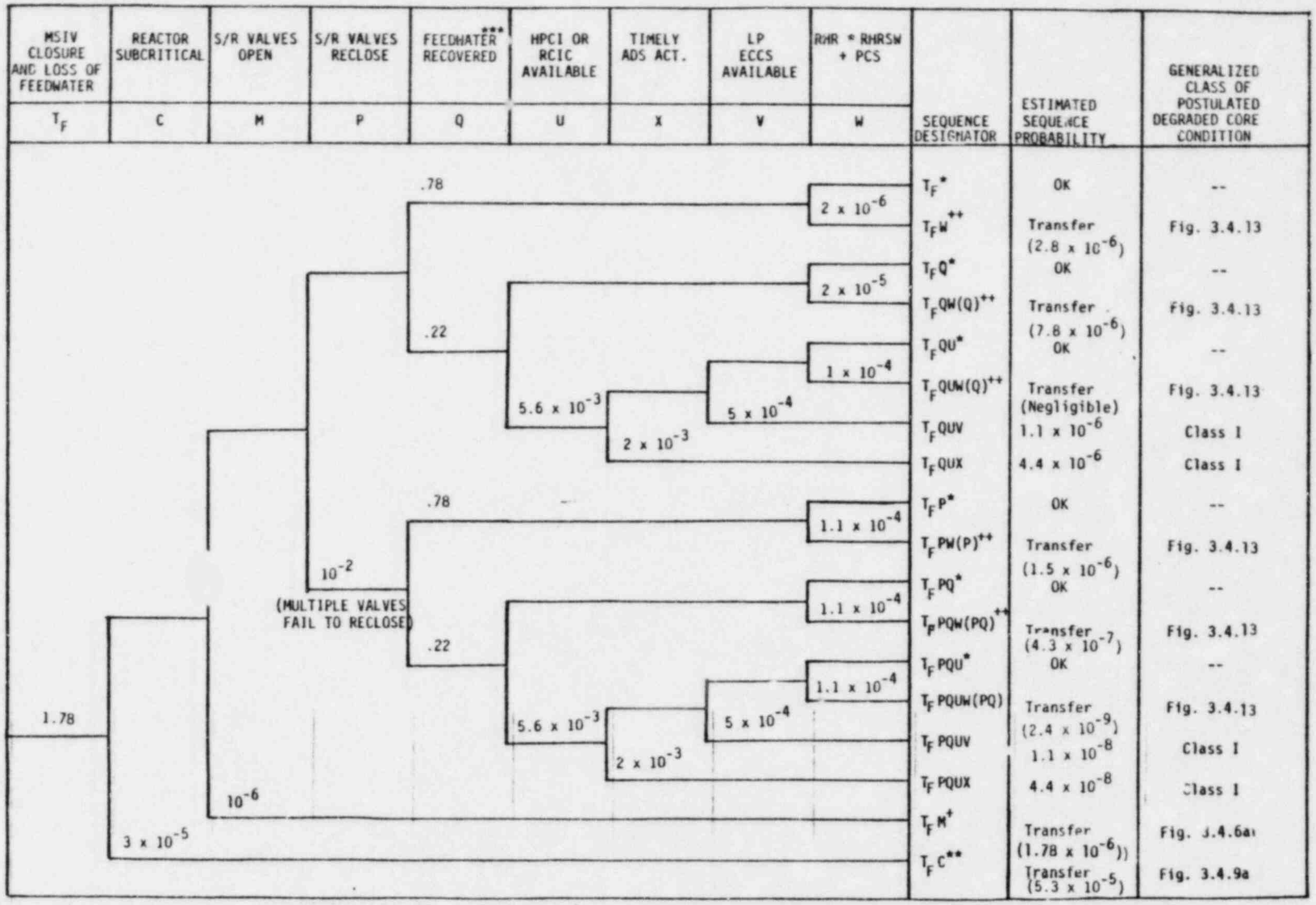
MSIV closure, loss of feedwater, and loss of condenser are similar transients, which may occur together, and are treated together in this analysis. These transients** present a more significant challenge to the reactor coolant makeup system than the turbine trip transient. This initiating event is defined such that feedwater may not be available for at least 30 minutes. The MSIVs are assumed to close, thereby requiring successful operation of the other coolant makeup systems if feedwater cannot be restored.

C -- Reactor Subcritical. Failure to bring the reactor subcritical is treated in the ATWS event trees to follow (see Section 3.4.3).

M -- Safety/Relief Valves Open. See Section 3.4.1.1 for definition of event M.

P -- Safety/Relief Valves Reclose. See Section 3.4.1.1 for definition of event P.

*That is, the probability of a single RHR loop being out of service and leading to a reactor shutdown during one year is 8.7×10^{-3}.
**These transients are treated together because of their assessed common effect of eventually leading to at least a temporary loss of feedwater injection and power conversion heat sink. In some cases, feedwater can be restored by either reopening or bypassing the MSIVs.



*Not core melt sequence
**ATWS initiators are treated in a separate event tree
+Transfer to large LOCA event tree
++Transfer to bridge tree

***Feedwater is found to be recoverable in 90% of the loss of feedwater accidents and 70% of the MSIV closure incidents

Figure 3.4.3 MSIV Closure/Loss of Feedwater/Loss of Main Condenser Transient Event Tree

Q -- Feedwater Available. See Section 3.4.1.1 for definition of event Q.

U -- High Pressure Injection Systems. See Section 3.4.1.1 for definition of event U.

X -- Timely ADS Actuation. This event is similar to the events defined in Section 3.4.1.1.

V -- LPECCS Available. See Section 3.4.1.1 for definition of event V.

W -- RHR and RHRSW or PCS. This event is similar to the event defined in Section 3.4.1.1 with the additional requirement that if the PCS is to be used to remove heat, it must be restored to operation.

In the calculation of long term containment heat removal for this transient, there is a long time available to restore the PCS. It is important to account for the ability to restore the PCS within the 20 to 30 hours available prior to exceeding containment design pressure. Based upon operating experience data for recovery of the PCS, at least 50% of the initiator events which cannot be recovered within 30 minutes can be recovered in 5 hours. Based upon the mean repair times from WASH-1400 of approximately 20 hours, there is sufficient time available to the operator to repair approximately another 50% of the remaining incidents. Therefore, a conditional failure probability of 0.25 is used to quantify the failure probability of PCS for 30 hours, given that it has become unavailable at the transient initiation.

For those cases where feedwater can be recovered within 30 minutes, it is assumed that the long term reliability of the PCS may still be degraded from that anticipated during a turbine trip. The PCS reliability is assigned 0.98 for such cases.

W(P) -- W Given that Event P Fails. This event is similar to W with the exception that the time available for residual heat removal using the RHR is decreased due to heat load from an open S/R valve. Multiple SORVs are assumed to lead to a demand that RHR operate successfully. This is the same as the turbine trip evaluation procedure.

There is one additional method of heat removal - the steam condensing mode of the RCIC. This feature was not evaluated in the analysis, since the specific operating procedures have not yet been written.

3.4.1.4 T_E -- Loss of Offsite Power Transient (See Figure 3.4.4)

Because of the pervasive nature of loss of offsite power, WASH-1400 treated this transient initiator separately from other transients, which were treated as a group. Electric power is required for long-term operation of most systems. If offsite power is unavailable, these systems require power from one of the emergency diesels for successful system operation. The systems whose reliability is adversely affected by the loss of offsite power include:

1. High Pressure Injection Systems

   - Feedwater is unavailable since the condensate pumps and feedwater controller are electric, and are normally powered from the normal AC power bus.

   - HPCI and RCIC although steam-turbine-driven, must rely on station batteries for starting and control. Battery life is 4 hours, without offsite power.

2. Low Pressure Coolant Systems

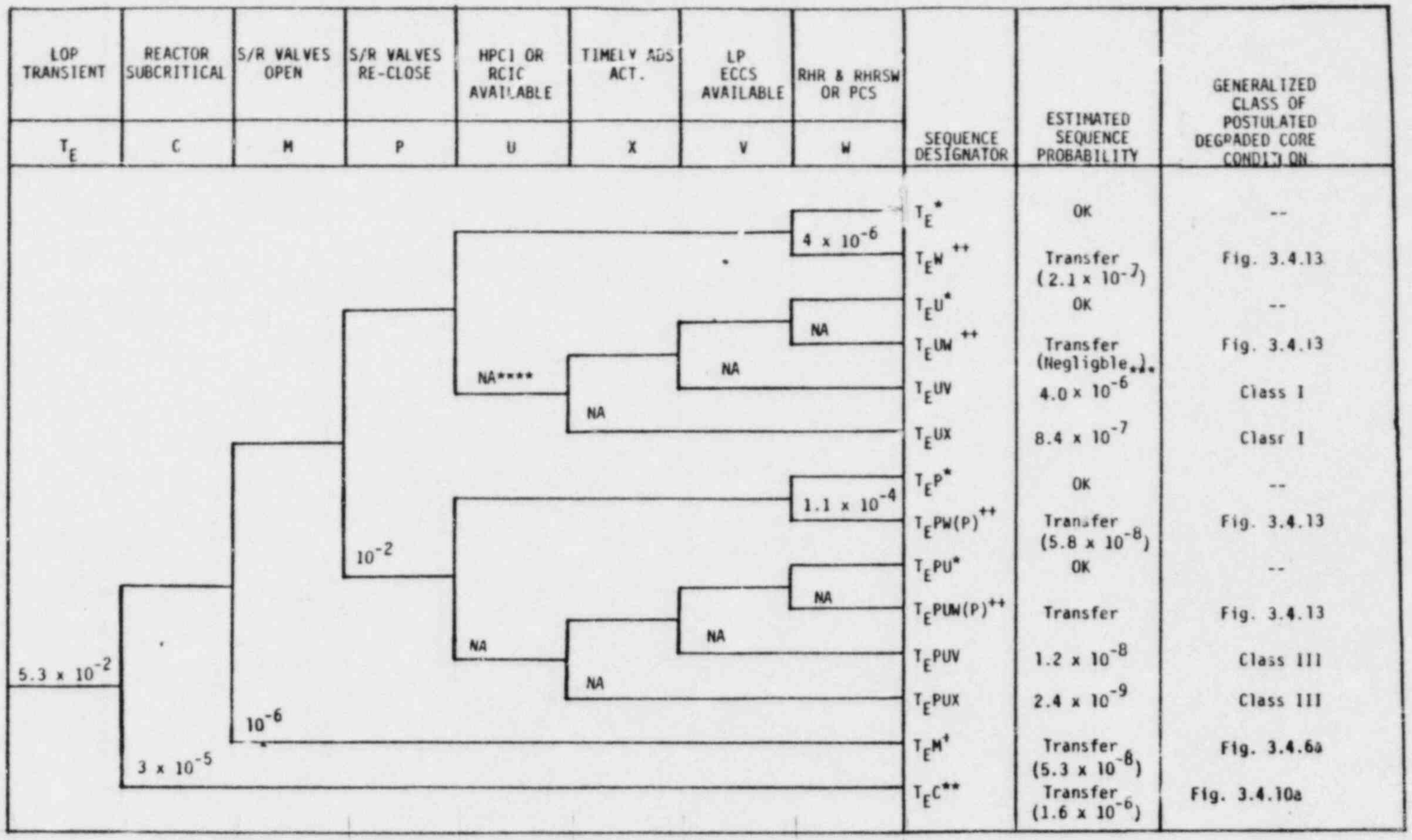
   - All low pressure pumps require 4160 v AC power which is available from the emergency buses supplied by the diesel generators.

   - Portions of the RHR and CS systems draw power from the same electrical bus.

3. Containment Heat Removal

   - The Power Conversion System is unavailable in the absence of offsite power due to the loss of electric-driven pumps.

   - The RHR pumps and valves depend upon the emergency AC power sources.



*Not core melt sequence
**ATWS initiators are treated in a separate event tree
+Transfer to large LOCA tree
++Transfer to bridge tree

***Due to common-mode failure of all electric power for two hours (loss of diesels = 1.8×10^{-3}) plus a 5% chance that both RCIC and HPCI will not work due to high room temperatures
****See Table 3.4.1

Figure 3.4.4a Loss of Offsite Power Transient Event Tree

- In this analysis Limerick is treated as a one-unit plant, with two RHR service water pumps, each supplied by one of the four Limerick 1 diesels. When Unit 2 enters service, RHR service water for both units will be powered from the same bus.

Some functions are not affected by the loss of offsite power, these include:

- X -- ADS
- M -- Safety Valves Open
- P -- Safety Valves Reclose.

The transient event tree for loss of offsite power describes the interaction of systems and their response for various time periods ranging from 2 to 6 hours following a loss of offsite power. System AC power requirements are time dependent, so failure rates vary with time.

The following is a summary of the events in the loss of offsite power Event Tree, Figure 3.4.4. In addition, two of the functions are discussed in more detail to indicate the nature of the time variance of failure probabilities. The two functions assessed using the time phased event trees are:

- Coolant Injection, Figure 3.4.4
- Containment Heat Removal, Figure 3.4.4.

The principal events for the loss of offsite power sequence are the following:

<u>C -- Reactor Subcritical</u>. Failure to bring the reactor subcritical is treated in ATWS event trees to follow (see Section 3.4.3.1). Subcriticality is assumed to be successful in this event tree.

M -- Safety/Relief Valves Open. See Section 3.4.1.1 for definition of event M.

P -- Safety/Relief Valves Reclose. See Section 3.4.1.1 for definition of event P.

U -- High Pressure Coolant Injection Systems Available. This event is similar to the event appearing in Section 3.4.1.1, in that HPCI and RCIC can supply inventory makeup to the reactor. Since offsite AC power is not available, feedwater will not be available. HPCI and RCIC are designed to be able to start and run initially without any AC dependence. Therefore, the emergency 4160V AC power bus is not required during the initial stages of a loss of offsite power transient.

However, room cooling is necessary for the RCIC and HPCI rooms. This requirement can be satisfied by either use of the ESW system or through a natural circulation ventilation path. The former requires AC power; therefore the emergency power buses supplying the RCIC or HPCI ESW pumps need to be restored using either offsite power or the emergency diesels. The latter option, natural circulation ventilation, requires action by the operator to recognize the problems and to bypass several normal plant features. The operator has two hours to perform the following in order to initiate natural circulation ventilation in the HPCI and RCIC rooms:

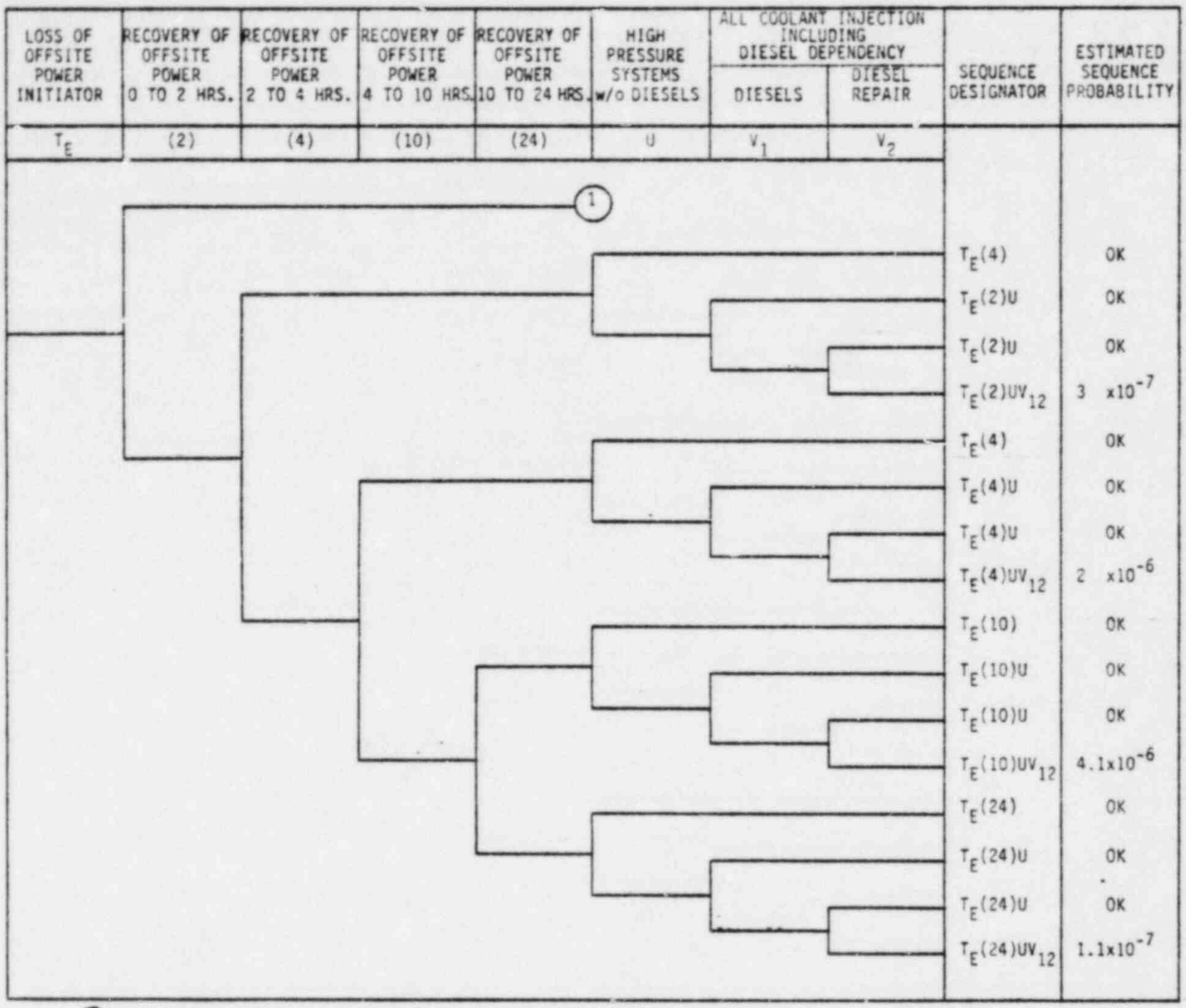
- Identify that a security problem or fire is not in progress.
- Open the security/fire doors between the HPCI room and/or RCIC room and the stairwell.
- Open the door from the stairwell to the refueling floor.

These actions will establish sufficient ventilation to ensure continued HPCI/RCIC operation for the LGS configuration.

In addition to the need for adequate room cooling, another need arises after four hours without AC power. The station batteries will be expended after four hours without recharging. Therefore, after four hours, HPCI and RCIC are no longer reliable sources of coolant injection, due to the loss of all automatic control and operator indication. Despite this loss of all power, there is a possible mode of local, manual RCIC operation which could be used in this case. However, because of the adverse environment, a very low probability of success is given to this path.

In order to better define the contributors to potential degraded core conditions resulting from a loss of offsite power initiator, a time-phased event tree is presented in Figure 3.4.4b. The terminology used in the time phased event tree is slightly different than used in other event trees in order to emphasize the dependency of the coolant injection function on AC power. The time periods of most interest for the loss of offsite power initiator are:

TIME	QUALITATIVE DESCRIPTION
0 - 2 hours	The high pressure injection systems, HPCI and RCIC, can be operated using the on-site DC power sources. No offsite or emergency AC power sources are required during this time.
2 - 4 hours	An AC power dependence may arise if the HPCI or RCIC rooms cannot be adequately cooled.
>4 hours	DC battery sources are anticipated to last for 4 hours. Past this time offsite power or emergency power must be restored to assure coolant injection.



① This branch transfers to MSIV closure initiators since it effectively is an MSIV Closure when offsite AC power is restored.

Figure 3.4.4b Loss of Offsite Power Transient Event Tree (Time-Phased Coolant Injection)

As seen in the time-phased event tree and Table 3.4.1, the timer periods of highest probability of inadequate coolant injection are the periods 2 - 4 hours and 4 - 10 hours.

Table 3.4.1

QUANTITATIVE EVALUATION OF THE TIME PHASES OF THE LOSS OF OFFSITE POWER ACCIDENT SEQUENCE

PHASE	TIME PHASE OF ACCIDENT SEQUENCE	ACCIDENT INITIATOR T_E	FAILURE TO RECOVER OFFSITE POWER*†	HIGH PRESSURE SYSTEMS U	LOW PRESSURE SYSTEMS $V^†$	COMMON-MODE DIESEL GENERATOR FAILURE PROBABILITY	FAILURE OF DIESEL GENERATOR REPAIR	TOTAL FREQUENCY (per reactor year)
I	0 - 2 hours	5.3×10^{-2}	.66	8×10^{-3} ††	†	1.08×10^{-3}	1.0	3×10^{-7}
II	2 - 4 hours	5.3×10^{-2}	.35	.15*	†	1.08×10^{-3}	.66	2.0×10^{-6}
III	4 - 10 hours	5.3×10^{-2}	.158	1.0**	†	1.08×10^{-3}	.47	4.2×10^{-6}
IV	10 - 72 hours	5.3×10^{-2}	.01	1.0**	†	1.08×10^{-3}	.2	1.1×10^{-7}

*Probability of requiring ventilation of HPCI and RCIC rooms coupled with the probability of the operators establishing a natural circulation ventilation path for these rooms.

**Conditional probability of successful operation of RCIC using manual control with no power (DC or AC) for times greater than 4 hours.

†Because of the redundancy of the available low pressure pumps the dominant contributor to the loss of the low pressure systems during a loss of offsite power is the common-mode failure of all the emergency diesels.

††No AC power required for HPCI/RCIC operation during the initial 2 hours following the loss of offsite power.

*†Probability of recovery of offsite power is derived from the data analysis performed in Appendix A.

X -- Timely ADS Actuation. This is similar to the event appearing in Section 3.4.1.1, with an increase in failure probability due to potential reluctance of operators to depend on the diesel-powered low-pressure system pumps, or the inability of some portion of the diesels to start and run on full load and therefore prevent some low pressure pumps from starting, thus inhibiting ADS.

V -- LPECCS. Similar to the event appearing in Section 3.4.1.1 with AC power dependency.

W -- RHR and RHRSW or PCS or RCIC Steam Condensing Mode. The RHR and RHRSW systems have a dependency on the diesel generators when offsite power is unavailable. The PCS is unavailable when offsite power is lost. The reasons for dividing the sequences in a time phased diagram for loss of offsite power are the following:

1. Short term loss of offsite power (<4 hours) is not a rare event; however, loss of containment heat removal has the potential to become a serious problem only after about 20 hours. Loss of offsite power for less than 4 hours coupled with complete loss of containment heat removal for more than 20 hours is considered to be a low probability event for the LGS configuration.

2. Loss of offsite power for periods in the range of 15 hours is of some concern, because the PCS may not be recoverable in sufficient time to be of use in containment heat removal. The PCS is given a low probability of success for these cases.

3. Loss of offsite power for periods greater than 15 hours has the following effects:

   - The PCS is treated as totally unavailable

   - The RHR system is the only available system to perform active containment heat removal.

The net result of this breakdown of postulated sequences is that the dominant sequence leading to possible containment overpressure as a result of the failure to remove heat from containment is loss of offsite power for a period greater than 24 hours. The frequency of loss of offsite power for greater than 24 hours is estimated to be 1/500 years (see Appendix A).

W(P) -- W Given that Event P Occurs. This event is similar to W except that the time available for RHR initiation is decreased due to increased heat load from the open S/R valve.

There are a number of reasons why the calculated level of risk associated with the loss of offsite power initiator is different for Limerick than that evaluated in WASH-1400:

1. The initiator frequency associated with the Pennsylvania-New Jersey-Maryland Interconnection is lower than that used in WASH-1400.

2. The HPCI and RCIC systems require pump room cooling if there is a loss of offsite power for greater than 2 hours, or battery charging for long-term loss of offsite power. (Neither of these appear to have been included in the WASH-1400 model.)

3. The anticipated maintenance unavailability on diesel generators may be significantly different than that assumed in WASH-1400.

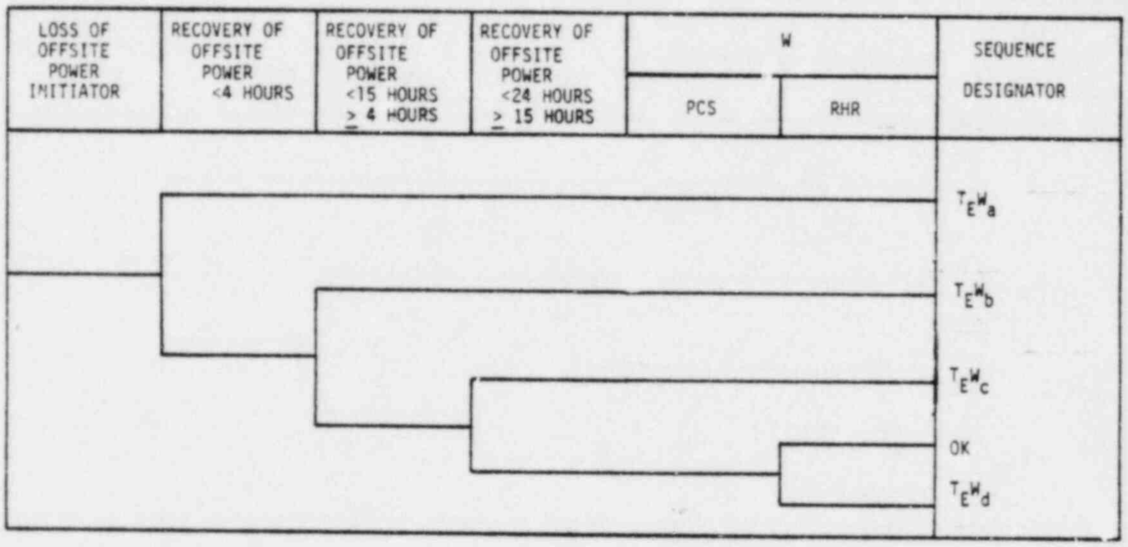
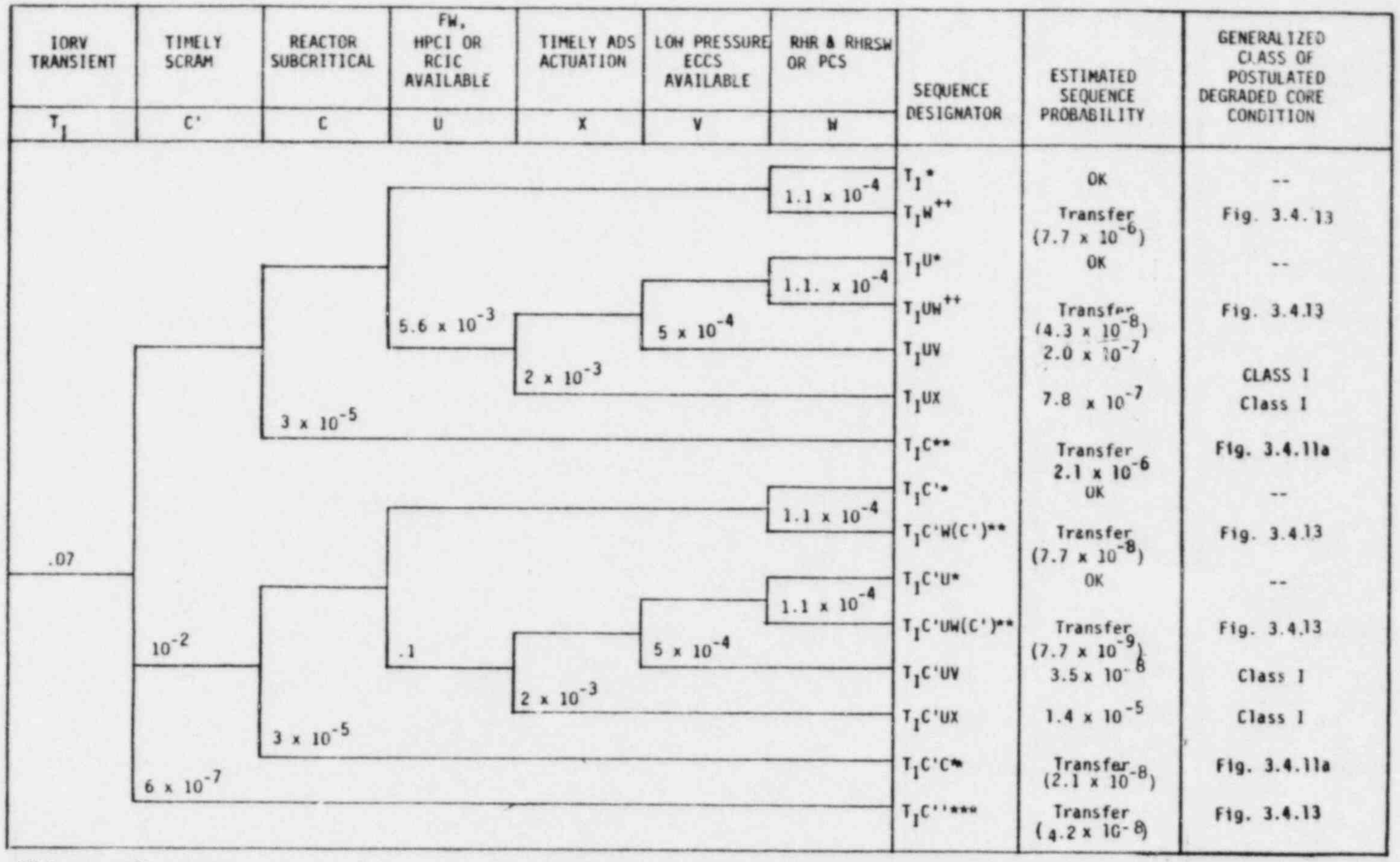


Figure 3.4.4c Time Phased Event Tree for Calculating Containment Heat Removal Capability Following a Loss of Offsite Power

3.4.1.5 Inadvertent Open S/R Valve Transient (See Figure 3.4.5)

Examination of the WASH-1400 analysis, and a review of new operating data, has revealed an accident initiator previously considered unimportant may result in a group of accident sequences which contribute to calculated risk. This initiator is the Inadvertent Opening of Safety Relief Valves (IORV) during full power operation.



*Not core melt sequence

**ATWS initiators are treated in a separate event tree

***Manual scram too late to prevent a challenge to the containment similar to a Class IV event.

++Transfer to bridge tree

Figure 3.4.5 Inadvertent Open Safety Relief Valve Transient Event Tree

A Licensee Event Report (LER) data search has shown that the frequency of occurrence for IORV events in BWRs is greater than the frequency for an SORV occurring during a transient. About half of the BWR IORV events occurred at greater than 80% power levels, and half of those valves remained open until the reactor pressure was below 200 psi.

The IORV event tree includes aspects of both the small LOCA event trees and transient trees. The IORV initially acts as a small LOCA, with respect to the makeup systems, but the safeguards which react to high drywell pressure (as may occur during a small LOCA) are not activated, so the operator must manually scram the reactor and start the makeup systems. Once the reactor is shut down, the IORV event tree is similar to the turbine trip transient event tree. However, since the reactor has been at full power, and has been releasing steam into the suppression pool for the time prior to scram, the suppression pool temperature may have increased significantly. Operating experience data indicate that the MSIVs will close during this event, causing all decay heat to enter the suppression pool. This decreases the time allowed for initiation of RHR to preclude suppression pool failure, loss of makeup, and eventual fuel damage or core melt. The decrease in time available for RHR initiation along with the manual scram requirements, are the factors which increase the probability that RHR will be unsuccessful.

The principal events for the IORV sequences are:

$\underline{T_I}$ -- Initiator. This event consists of a safety/relief valve opening inadvertently during >80% power operation. This event differs from other transient events primarily due to the extra heat load placed on the RHR system by the blowdown to the suppression pool.

<u>C' -- Timely Scram</u>. There are no "trip" signals generated by the reactor protection system during the IORV event sequence. The operator

will be alerted to an IORV condition by observing the SRV position indicators. Failure of C' implies failure of the operator to scram the reactor prior to the suppression pool reaching a temperature requiring both RHR heat exchangers to be operational.

C". Failure to manually scram before the suppression pool reaches a temperature which will eventually raise containment pressure and temperature beyond the capacity of the RHR.

C -- Reactor Subcritical. This event consists of a successful manual scram and is analyzed by the ATWS event tree in Section 3.4.3.1.

U - HPCI or RCIC. This event is similar to the event appearing in Section 3.4.1.1.

X. This event is similar to the event appearing in Section 3.4.1.1 with some additional considerations due to the high temperature in the suppression pool.

V -- LP ECCS Available. This event is the same as the event in Section 3.4.1.1.

W. This event is similar to the event appearing in Section 3.4.1.1 with the exception that the heat removal requirements are somewhat greater for the IORV initiator, i.e., the suppression pool temperature at scram is assumed to be 110^{o}F. In addition, the MSIVs must be reopened to activate the PCS.

W(C'). This event is similar to W except that both RHR loops must be operative or the PCS must be recovered to prevent containment failure.

3.4.2 Event Tree Analysis-LOCA Event Trees

The LOCA event trees used for the Limerick analysis are only slightly different than those used in WASH-1400. The Limerick event trees more realistically model the actions of the coolant injection systems than those used in WASH-1400. Three LOCA event trees are used in the Limerick analysis: one depicting LOCAs which depressurize the reactor (large LOCAs); and two which deal with medium and small LOCAs which do not cause the reactor to depressurize (see Figure 3.4.6a, b, and c, respectively).

The large LOCA tree is similar to the one used in WASH-1400. It contains the same systems and structure as the WASH-1400 event tree with the exception of the electric power (B), vapor suppression (D), containment leakage (G), and core cooling (F) functions. Electric power was eliminated from the LGS LOCA event tree because a more proper treatment of electric power and its interactions with systems was made by entering electric power into the individual system fault trees at the component level. In addition, containment leakage and vapor suppression were also eliminated from the LGS LOCA event trees, since they did not explicitly affect the LOCA sequence at Limerick. Instead, they are included in the containment event tree (see Section 3.4.5). At Limerick, the low pressure pumps are designed to be able to pump saturated water from the suppression pool with no back pressure requirement in the containment. The presence of containment leakage does not adversely affect their performance. Emergency core cooling functionability*, has also been removed from the event tree, since there was no idendified physical basis for this event.

The medium LOCA and small LOCA event trees (see Figures 3.4.6a and c) for Limerick also differ from the WASH-1400 small LOCA event trees. Electric power (B), leakage vapor suppression (D), and containment leak-

*WASH-1400 included a probability that the core would be disrupted at the time of emergency core cooling initiation and could not subsequently be properly cooled.

age (G), were eliminated using the same reasoning given for the large LOCA. Since the plants' reaction to a small LOCA is similar to a transient, the small LOCA event tree resembles a transient event tree.

3.4.2.1 Definition of Events in the Large LOCA Event Tree (See Figure 3.4.6a)

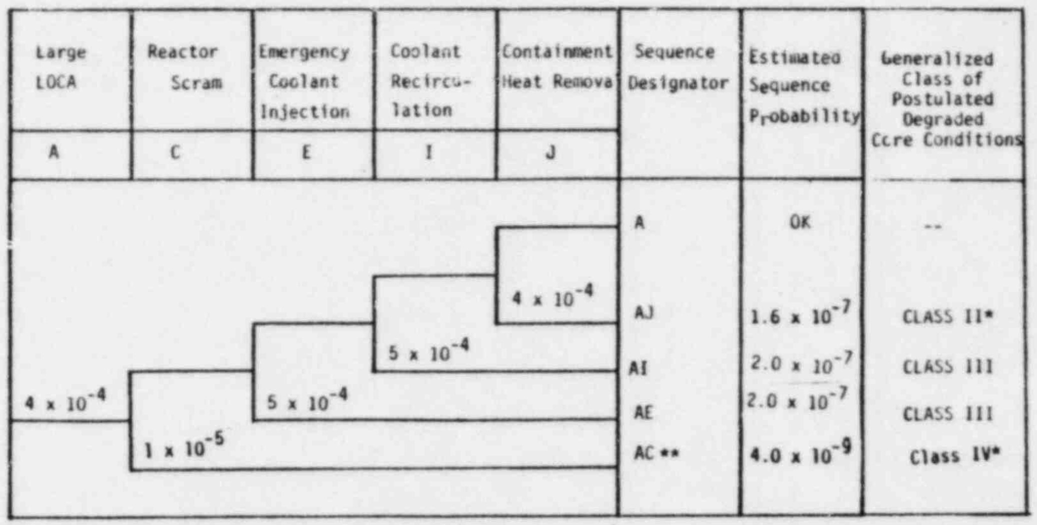
Event A - Large LOCA: Large LOCA in the context of the Limerick Probabilistic Risk Assessment is any break in the reactor system piping which could lead to the loss of sufficient coolant resulting in the following:

- Relatively rapid depressurization
- Demand for coolant injection by either LPCS or LPCI
- Demand for long term recirculation
- Demand for containment heat removal.

The probability of a large LOCA used in this analysis is much larger than the probability of the design basis LOCA which is a "worst case" double ended shear of the reactor coolant pipe (see Appendix A for further discussion). For this analysis, the large LOCA is assumed to occur in two possible ways: (1) a break of a recirculation line (water break); and (2) break in lines above the top of the core (steam break).

Event C - Reactor Scram: This event is defined as insertion of the control rods.

Event E - Emergency Coolant Injection: Two systems provide the capability for emergency coolant injection, subsequent to a large LOCA. Each system provides coolant to the reactor via separate processes. The Core Spray system (CS) contains two loops, each with two pumps, which deliver coolant to spray spargers directly above the core.



*Transfer to Bridge Tree inappropriate in this case

**Independence assumed between the control rod insertion system and the LOCA blowdown forces; in addition only mechanical failures in the control rod system affect this sequence. ARI will reduce the probability of electrical failures in the RPS to a negligible value.

Figure 3.4.6a Limerick Large LOCA Event Tree

The second system, the Low Pressure Coolant Injection (LPCI) system, is an operating mode of the RHR system. This system consists of four pumps which automatically inject directly into the reactor vessel.

In order to simplify the LOCA event tree, all combinations of CS and LPCI failures resulting in failure of E were combined in a functional level fault tree. Appendix B summarizes the functional level fault tree for coolant injection. This fault tree reflects the success criteria established in Section 1.5.

Event I - Coolant Recirculation: This event involves the long term recirculation of the water to the core from the suppression pool. This function can be accomplished with either LPCI or LPCS. The success criteria and calculated probability are similar to that for short-term coolant injection.

Event J - Containment Heat Removal: In order to preserve primary containment integrity following a LOCA, the RHR system must be initiated within 25 hours as determined by INCOR calculations (see Appendix C). Residual heat removal has to be maintained for approximately six months. Within the six month period, provisions can be made for transferring the fuel to the spent fuel storage pool, or alternate methods of core cooling can be provided if required.

Because of the potential for fission products inside the primary system and containment following a large LOCA, neither the PCS nor the COR are assumed available to perform the containment heat removal function. Therefore, the redundant RHR system is required to remove decay heat from containment. The large LOCA event tree (Figure 3.4.5a) displays this sequence as AJ, where J is composed of only RHR.

3.4.2.2 Definition of Events in the Medium LOCA Event Tree (see Figure 3.4.6b)

The medium and small LOCA events differ only in the availability of the High Pressure Injection Systems for successful mitigation.

Event S_1 - Medium LOCA: This event is a LOCA which does not depressurize the reactor. The medium LOCA event is defined as a break of between .004 and .1 ft^2 for a liquid line, and between 0.016 and 0.08 ft^2 for a steam break. Larger breaks will depressurize the reactor without HPCI or ADS assistance and are classified as large LOCAs. Since the reactor may be isolated subsequent to a medium LOCA, feedwater may be unavailable for coolant injection.

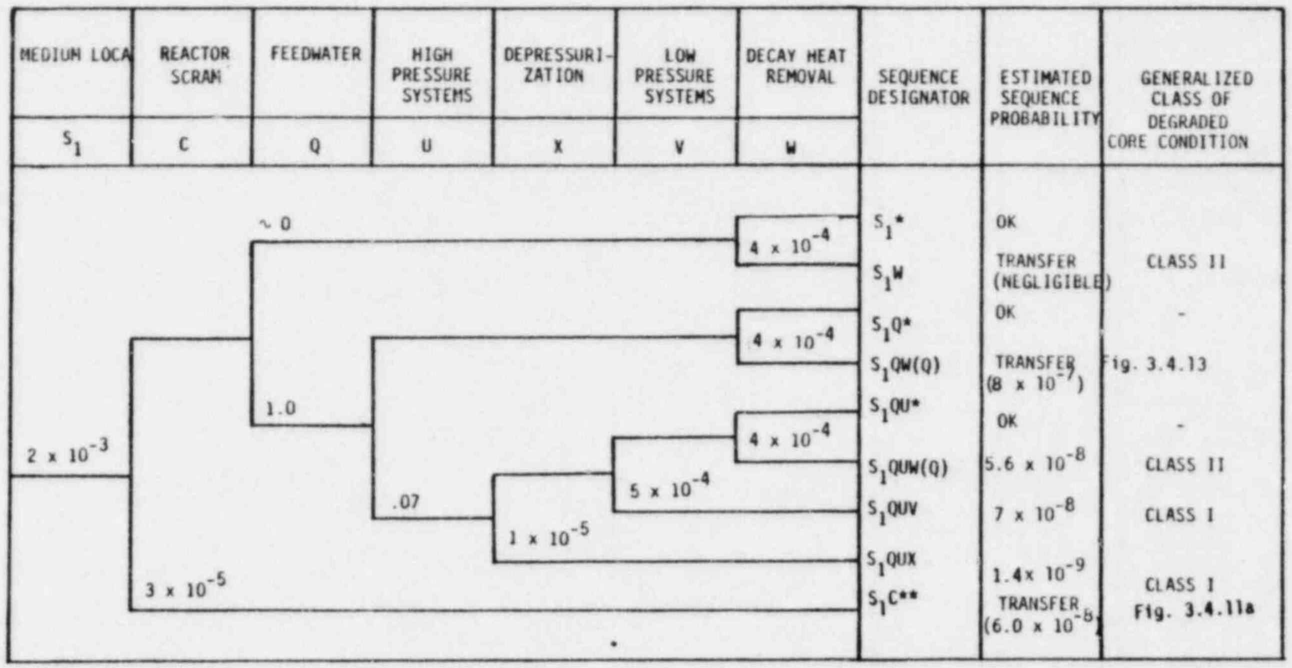
Event C - Reactor Scram: This event is defined as insertion of the control rods.

Event U - High Pressure Systems: A functional level fault tree depicting the failure of U for a medium LOCA is simply a failure of HPCI.

Event X - Depressurization: This event consists of either automatic or manual depressurization of the reactor to allow low pressure systems to operate. Failure of this system involves the ADS system failure to manually or automatically actuate, or failure of the low pressure systems to start, thus inhibiting ADS.

Event V - Low Pressure System: This event is the same as Event V appearing in the transient event trees (Section 3.4.1.1).

Event W: This event contains both coolant recirculation and heat removal from the containment. Success requires either recovery of the PCS or availability of the RHR service water system and 1 RHR heat exchanger, combined with an injection path to the core.



*Not a Core Melt Sequence

**Treated in ATWS Trees

Figure 3.4.6b Limerick Medium LOCA Event Tree (S_1)

For medium LOCAs, if RHR is unavailable to remove containment heat, COR can be used, as long as the core remains covered; however for cases where HPCI fails and ADS is required, COR is assumed to not be useable. Since only HPCI is available as the high pressure injection source, its failure coupled with the medium LOCA initiator leads to a direct demand on the RHR system, without the possibility of using the PCS or COR. This sequence is the highest Class II probability sequence from the medium LOCA event tree. The next most likely sequences leading to containment overpressure are those for which HPCI is available, but RHR and COR fails.

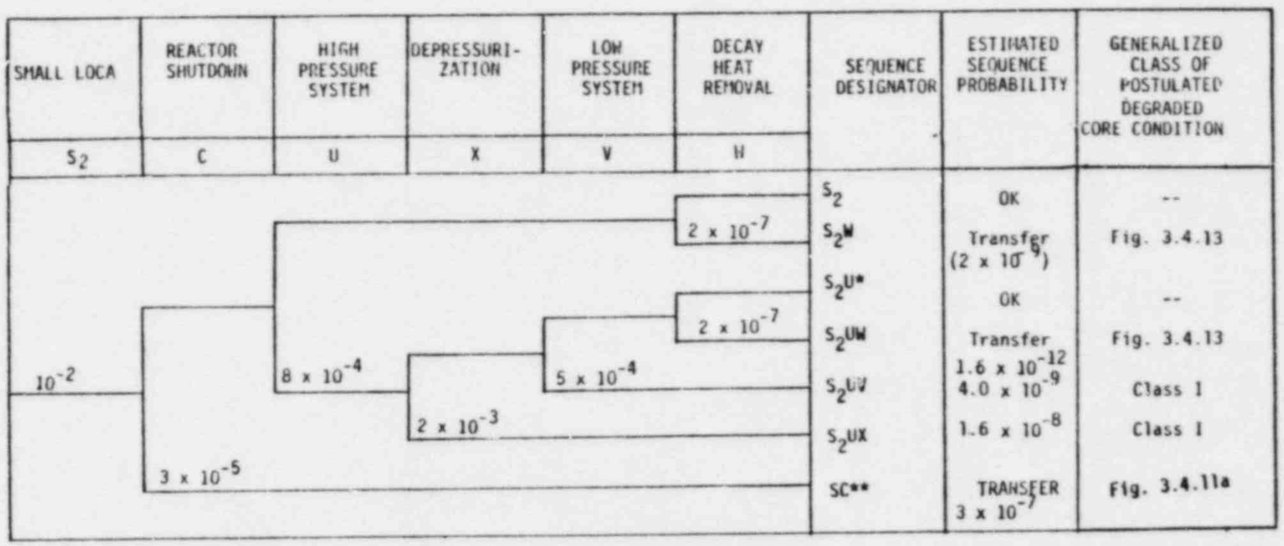
3.4.2.3 Definition of Events in Small LOCA Tree (See Figure 3.4.6c)

Pipe breaks of less than 0.004 ft^2 (liquid) are included in the small LOCA category. The small LOCA tree is exactly the same as the medium LOCA tree appearing in Figure 3.4.5b, with the exception of the requirements for the high pressure systems.

Event U - High Pressure Systems: A functional fault tree depicting failure of the high pressure systems subsequent to a small LOCA is given in Appendix B. It is the same as the high pressure system requirements for an S_2 LOCA given in WASH-1400 Appendix I.

3.4.3 Event Trees for ATWS and Other Low Probability Events

There a number of events which have been postulated as possible at nuclear power plants that, because of their low probability, are referred to as unanticipated events. The Limerick Probabilistic Risk Assessment has included consideration of three of these identified rare event sequences because of potentially high consequences.



*Not a core melt sequence

**Treated in ATWS trees

Figure 3.4.6c. Limerick Small LOCA Event Tree (S_2)

1. Anticipated Transient Without Scram (ATWS) (Section 3.4.3.1)

2. Reactor pressure vessel failure (Section 3.4.3.2)

3. Interfacing LOCA (Section 3.4.3.3).

3.4.3.1 Anticipated Transient Without Scram (ATWS)

The ATWS event can be divided into two distinct portions for discussion and analysis. These are:

1. Prevention: This includes those system features designed to assure that the control rods will be inserted when required.

2. Mitigation: This includes the systems or features designed to provide a diverse method of reactor shutdown if the control rods cannot be inserted.

Limerick will have an ATWS prevention/mitigation system at least as good as the Alternate 3A modification identified by the NRC staff in NUREG-0460. This system includes the following features:

- Additional safety grade level sensors in the scram discharge volume*
- Alternate rod insertion (ARI) circuitry and solenoid valves (discussed further in Appendix B)
- Recirculation pump trip (RPT)
- Automatic standby liquid control (SLC) system to inject boron solution (discussed further in Appendix B)
- Feedwater runback.

These improvements are incorporated into the ATWS event trees and the systems level fault trees which describe each function.

---

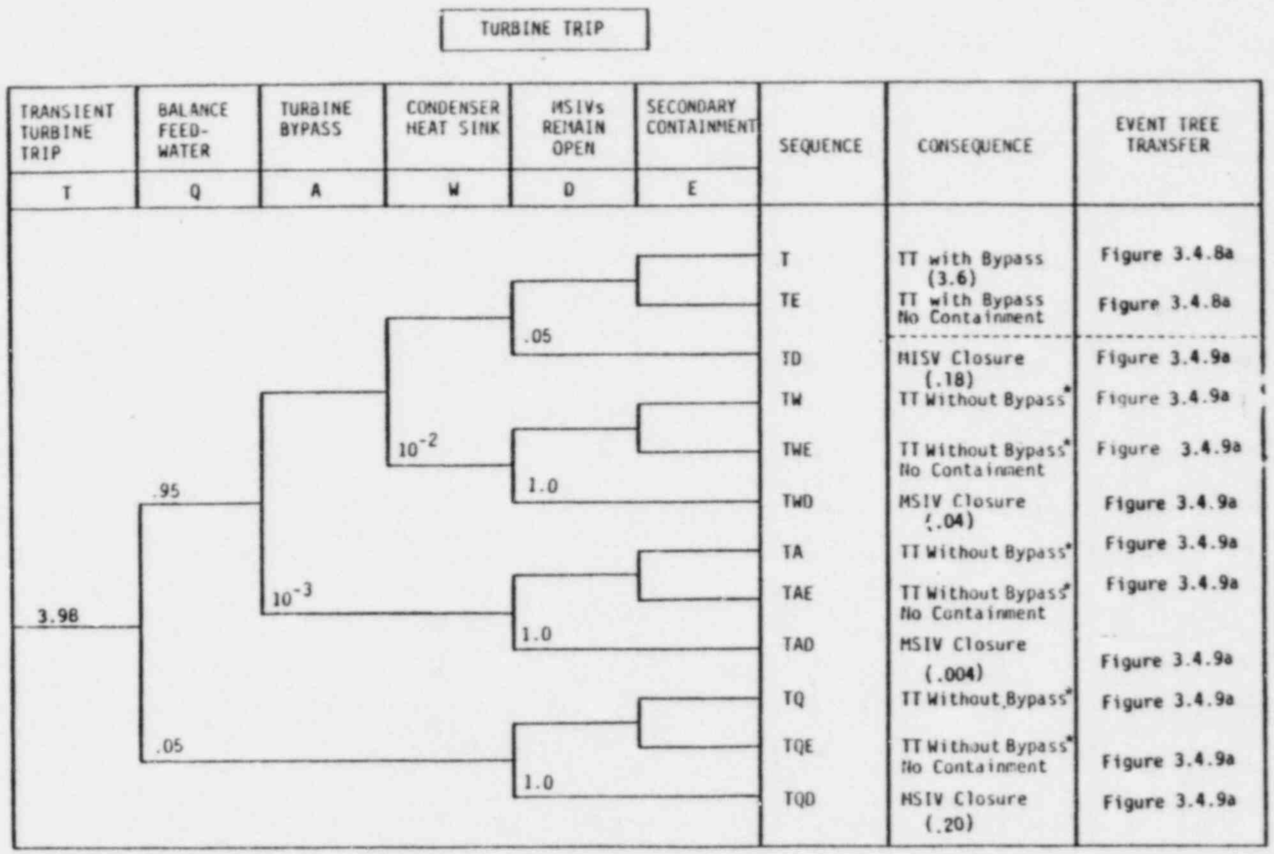
*These improvements are the result of the design to eliminate the postulated single point mechanical failure of the control rod drive mechanisms identified in WASH-1400, EPRI, and General Electric investigations.

The event trees which have been developed to describe the postulated ATWS sequences are given in Figures 3.4.7 through 3.4.11. As noted in Section 3.4.1, each of the anticipated transients has an event tree branch corresponding to the transient initiator and the failure of the control rods to insert (i.e., an ATWS). Since each of the identified transient initiators has a different interaction with the mitigating systems available, the ATWS event trees are constructed to parallel the event trees developed in Section 3.4.1 for the anticipated transients. Tables 3.4.2a and 3.4.2b explain the ATWS event tree notes and define the applicable event tree system functions.

3.4.3.1.1 Turbine Trip ATWS Initiator

The majority of transient initiators result from turbine trip or lead to turbine trip. Figure 3.4.7 is the event tree used to identify the potential outcomes of a turbine trip event. The two cases used in this analysis are turbine trips which proceed as planned with the condenser available as a heat sink, and those turbine trips with the condenser unavailable due to the closure of the turbine bypass valve, loss of feedwater, or loss of heat sink. In the first case, the following occurs: the turbine trips, the bypass opens, the condenser remains available, and the feedwater is properly controlled to maintain adequate flow from the condenser hotwell to the reactor. The postulated effects of ATWS on the turbine trip are summarized in Figure 3.4.8 for cases with the bypass available; Figure 3.4.8a and 3.4.8b for ATWS prevention and mitigation, respectively. Implicit in the construction of the event tree for turbine trip with bypass is the fact that feedwater is available to supply coolant injection to the reactor, as shown in Figure 3.4.7. Cases where feedwater is unavailable following a turbine trip are treated in the MSIV closure event tree (Figure 3.4.9).

In addition to the turbine trip with bypass available, there is a second turbine trip case, for which the bypass fails. For those cases, the feedwater system could still be available for coolant injection at reduced flows if condensate makeup is transferred from the CST to the hotwell. In



*All Turbine Trips for which bypass to the condenser is not functional, are considered to be equivalent to MSIV Closure Events.

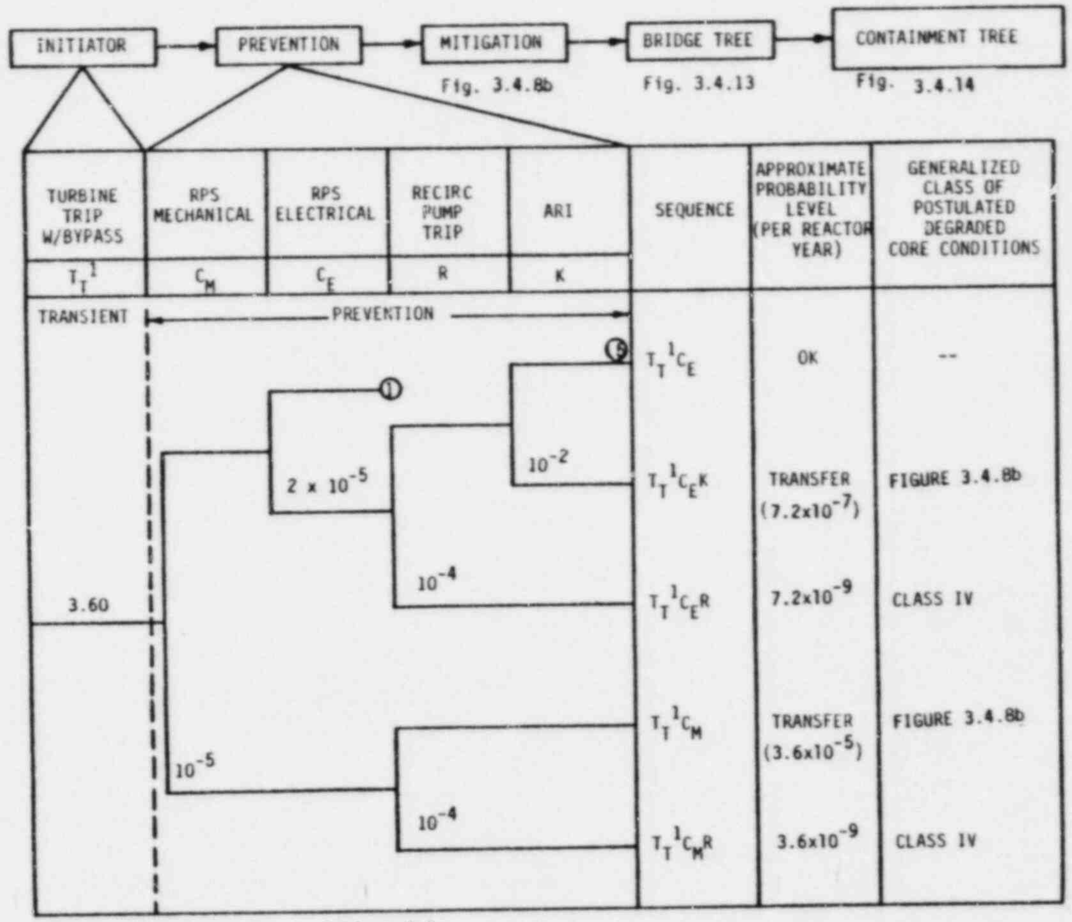
Figure 3.4.7 Event Tree Diagram of Accident Sequences Following a Turbine Trip Initiator.

NOTE: This event tree is evaluated assuming that a turbine trip followed by a failure to scram is in progress. The use of the tree is to discriminate between events leading to isolation and those for which the condenser remains availab.e the analysis all failures of the bypass valves are assumed to lead to loss of heat sinks, condenser, and feedwater. For the purposes of this analysis such a situation resembles an MSIV closure event. Therefore, turbine bypass failures and loss of feedwater cases are treated as MSIV closures. These sequences are classified as MSIV closures, because they result in effectively eliminating both the coolant injection function and decay heat removal function of the power conversion system, are then incorporated into the initiator for the event tree developed for MSIV closure (Figure 3.4.9).

The principal events for the ATWS turbine trip sequence are:

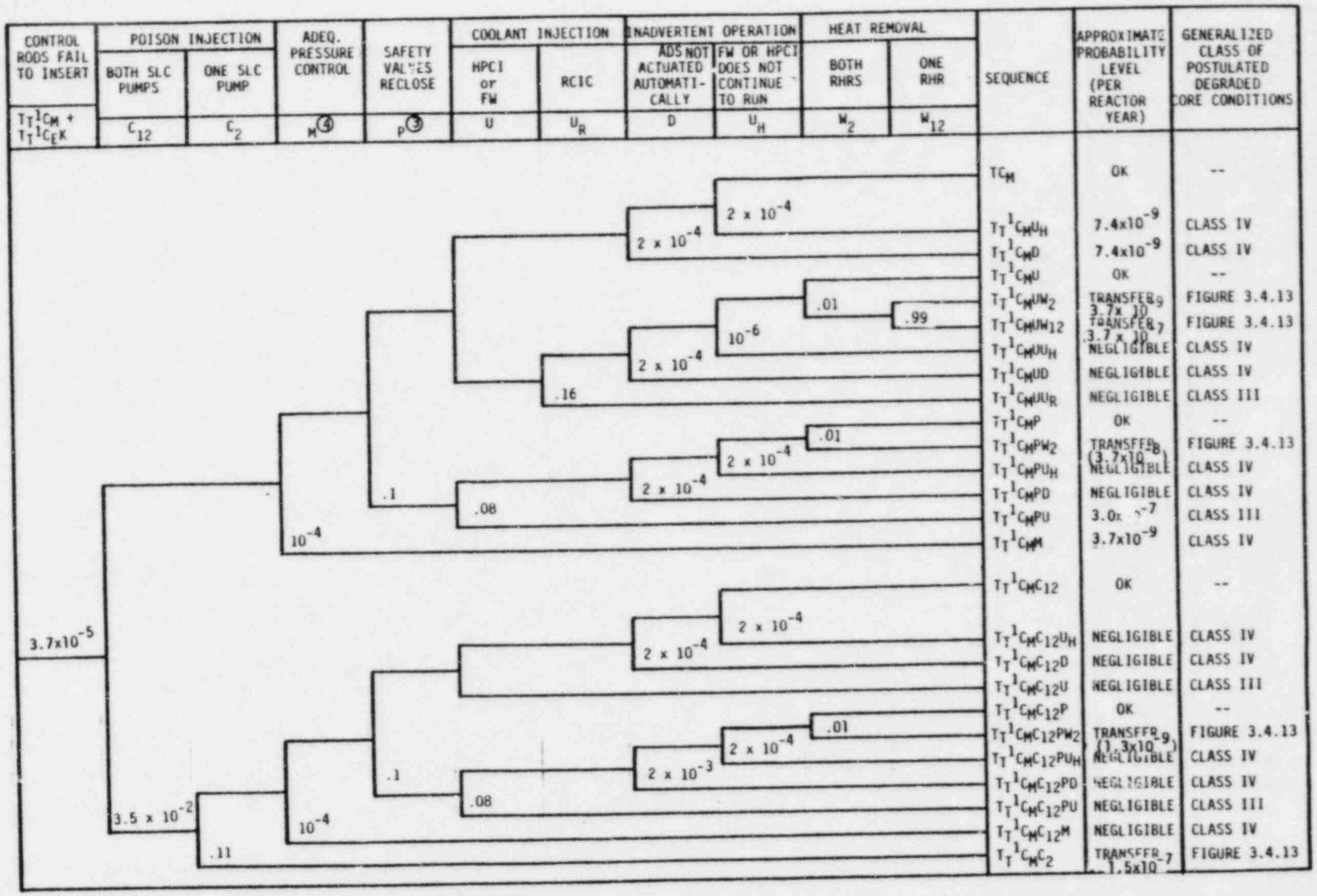
> C_M -- The mechanical redundancy of the control rod drive mechanisms makes the common-mode failure of multiple adjacent control rods unlikely.
>
> C_E -- The electrical diversity in sensors, logic, and scram solenoids help to reduce the potential for common-mode failures leading to failure of multiple rods to insert.
>
> R -- Recirculation pump trip (RPT) is implemented to reduce the effective power level of the core from 100% to approximately 30% with the control rods out.
>
> K -- Alternate Rod Insertion (ARI) incorporates a number of changes includding additional sensors, additional logic, and additional solenoid valves on each mechanism to provide added assurance that the postulated electrical failures will not prevent control rod insertion.

Beyond the design capability to prevent ATWS (which is the preferred method of treating any ATWS case), there is also a combination of systems which can effectively mitigate the consequences of a postulated ATWS. The functions required for ATWS mitigation during the turbine trip event are those identified in Figure 3.4.8 and discussed below:



All Notes are explained in Table 3.4.2a.

Figure 3.4.8a Event Tree Diagram of Postulated ATWS Accident Sequences Following A Turbine Trip Initiator



All Notes are in Table 3.4.2a.

Figure 3.4.8b Event Tree Diagram of Postulated ATWS Accident Sequences Following A Turbine Trip Initiator

C_2 -- Poison Injection: For Limerick, Philadelphia Electric has committed to the installation of an automated boron injection system corresponding to the NRC staff modification designated Alternate 3A. The system has been evaluated in a reliability model assuming that its characteristics are the minimum required by the Alternate 3A NRC guidelines. The initiation of the poison injection is diverse from that for control rod insertion and similar to that for RPT and ARI.

The loss of poison injection (C_2) capability in addition to the inability to insert control rods (C_M) may not necessarily lead directly to a degraded core condition. However, for the purposes of this analysis, this event sequence is treated as eventually leading to an MSIV closure (see below) and therefore having consequences similar to the ATWS MSIV closure sequence discussed in the next section. Note that the RHR system is not discussed in this sequence, since scram failure and failure of poison injection lead to containment overpressure directly (COR* is not used in these cases). If the overpressure failure occurs prior to initiation of a core melt, this sequence is placed in Class IV. The Bridge Tree, Figure 3.4.10 provides the sequences considered.

M -- Adequate Pressure Controls: The large number of safety/relief valves at Limerick provide a high level of confidence that there will be a sufficient relief valve capacity to avoid excessive pressures inside the reactor system following an ATWS. Given the ATWS initiator, failure of M implies that a LOCA would probably result.

*COR (containment overpressure relief) is not used for sequences where fuel damage or high radiation may exist inside the containment.

The LOCA is assumed to result in replenishing the core with water via the low pressure systems. This cold unborated water is assumed to cause recriticality leading to containment failure. Following containment failure, it is assumed that the coolant injection fails leading to eventual core melt.

P -- All Safety Valves Reclose: While the large number of relief valves provides high reliability of maintaining pressure within the reactor system capability, the probability of leakage, or stuck open/inadvertent open relief valves must be considered.

U -- Coolant Injection: Reactor coolant is assumed to be injected via the feedwater system and therefore HPCI and RCIC are not, in general, required. There are cases where SORVs may result in forcing the closure of the MSIVs. In such cases, HPCI or RCIC may be required to maintain adequate coolant injection. As noted in the success criteria (Section 1.5), RCIC is not considered adequate for those cases where only one SLC pump is available. The event tree is constructed to reflect this requirement.

D -- ADS Actuation (Inadvertent Operation): During the course of an ATWS event, the drywell pressure may rise and the reactor water level may drop. Due to calibration errors or instrument drift, sufficient automatic signals may exist to initiate ADS. The operator is required to inhibit ADS until the automatic initiation signals clear. The consequences of such an incident are similar to those evaluated for Class IV, since the containment loads associated with blowdown at high suppression pool temperature may lead to containment failure.

U_H -- FW or HPCI Continue to Run (Inadvertent Operation): Following poison injection and reactor shutdown, it is required that feedwater and HPCI be shut off to avoid poison removal/dilution from the reactor vessel. The shutoff of the high pressure systems can be accomplished either by the automatic high level trips or manual action.

W -- Heat Removal: The ability to remove decay heat from containment is a function* which helps to avoid the release of radioactive material to the environment. The systems available to fulfill this function are:

- The normal heat removal path through the condenser
- The RHR heat exchangers, and
- Containment Overpressure Relief (COR).

For the turbine trip with bypass transient sequences, the heat sink used for nearly all cases is the main condenser. For those cases where multiple relief valves fail open, the RHR is required to operate successfully. Other sequences which result in isolation of the reactor from the main condenser are treated in the MSIV closure event tree.

The ATWS sequences which are subject to the most uncertainty with respect to the plant response are the turbine trip initiated sequences. In lieu of detailed analysis or extensive operating experience for these situations, several assumptions have been made to allow the estimation of accident sequence likelihood. The two turbine trip ATWS sequences with the largest impact on calculated probability of risk to the public are the following:

---

*NUREG-0460, Volume IV provides a discussion which considers it acceptable to have a failure of decay heat removal from containment. The NRC staff evaluation of ATWS Alternates 3A and 4A in NUREG-0460, Volume IV, stated that a failure mode from WASH-1400 which was taken to lead directly to core melt is not assumed to lead to core melt in the ATWS evaluation. This sequence is the TW sequence, that is, the ATWS or transient coupled with the failure to remove decay heat using the main condenser or the RHR. WASH-1400 assessed this accident sequence as one of the principal sources of risk associated with a BWR, however, the recent NRC evaluation in NUREG-0460 indicates a different approach is acceptable. The new approach recognizes that containment failure will not necessarily compromise the ability to inject cool water to the reactor vessel or the ability to maintain adequate liquid poison. Therefore, loss of containment by overpressure is not a direct core melt but requires other failures to occur to preclude adequate coolant inventory or liquid poison inventory in the core before a core melt will occur. On the other hand, the NRC Probabilistic Analysis Staff (PAS) in rebaselining BWRs has placed containment overpressure failures in a much more prominent position as major risk contributors. The evaluation of Limerick capability given an ATWS event does consider the failure of heat removal capability from the containment to lead to unacceptable core conditions (similar to the PAS rebaselined assumptions).

1. Sequence:

Turbine Trip, Failure of Control Rods to Insert, Failure of SLC ($T_TC_MC_2$)

Anticipated Result:

Steam generated at 25 to 30% power is being dumped to the condenser; feedwater is supplying makeup to the reactor; it is assumed that probabilistically this condition cannot persist for a long period of time. Specifically, it is assumed that 90%* of the time the MSIVs will isolate due to high radiation associated with incipient fuel failures. The MSIV closure leads to the loss of the power conversion system. COR is not used due to the potential for radiation in the containment with no SLC. Therefore, the transfer to the bridge tree is made with these limitations. The result is a sequence which may lead to containment failure prior to core melt initiation, a Class IV event.

2. Sequence:

Turbine Trip, Failure of Control Rods to Insert, Failure of RPT (T'_TC_MR)

Anticipated Result:

RPT and FW runback are assumed to be tripped from the same set of logic and sensors. Therefore RPT failure is assumed to also result in failure of FW runback. The failure of RPT following a turbine trip initiated ATWS may then result in the production of a relatively high power and steam flow. The turbine bypass system is sized for approximately 25% power. The excess power generation will be dumped directly to the suppression pool. The success criteria of Section 1.5 state this situation to be unacceptable. The COR may not be adequate to handle the excess steam flow and containment failure may occur prior to core melt.

3.4.3.1.2 MSIV Closure ATWS Initiator

The MSIV closure transient initiators, discussed further in Appendix A, include loss of condenser and loss of feedwater. Also in-

*The remaining 10% of the time, sufficient time is available to manually drive the control rods in, and there is no significant degradation of core integrity.

cluded are those turbine trip initiated sequences (Figure 3.4.7) for which turbine bypass or feedwater are not functional. At Limerick, the closure of all main steam isolation valves leads to:

- The loss of main feedwater*
- The loss of normal heat removal via the main condenser.

Therefore, closure of MSIVs accompanied by an ATWS would leave HPCI and, under some conditions, RCIC to maintain adequate core coolant inventory, and the RHR to remove heat from containment.

Figure 3.4.9 is the event tree for MSIV closure-initiated ATWS accident sequences. The functions are nearly identical to the turbine trip case (Section 3.4.3.1.1) with the exception that coolant injection and heat removal are added explicitly. The event tree is again shown in two parts, Figure 3.4.9a for evaluation of ATWS prevention functions, and Figure 3.4.9b for evaluation of ATWS mitigation functions.

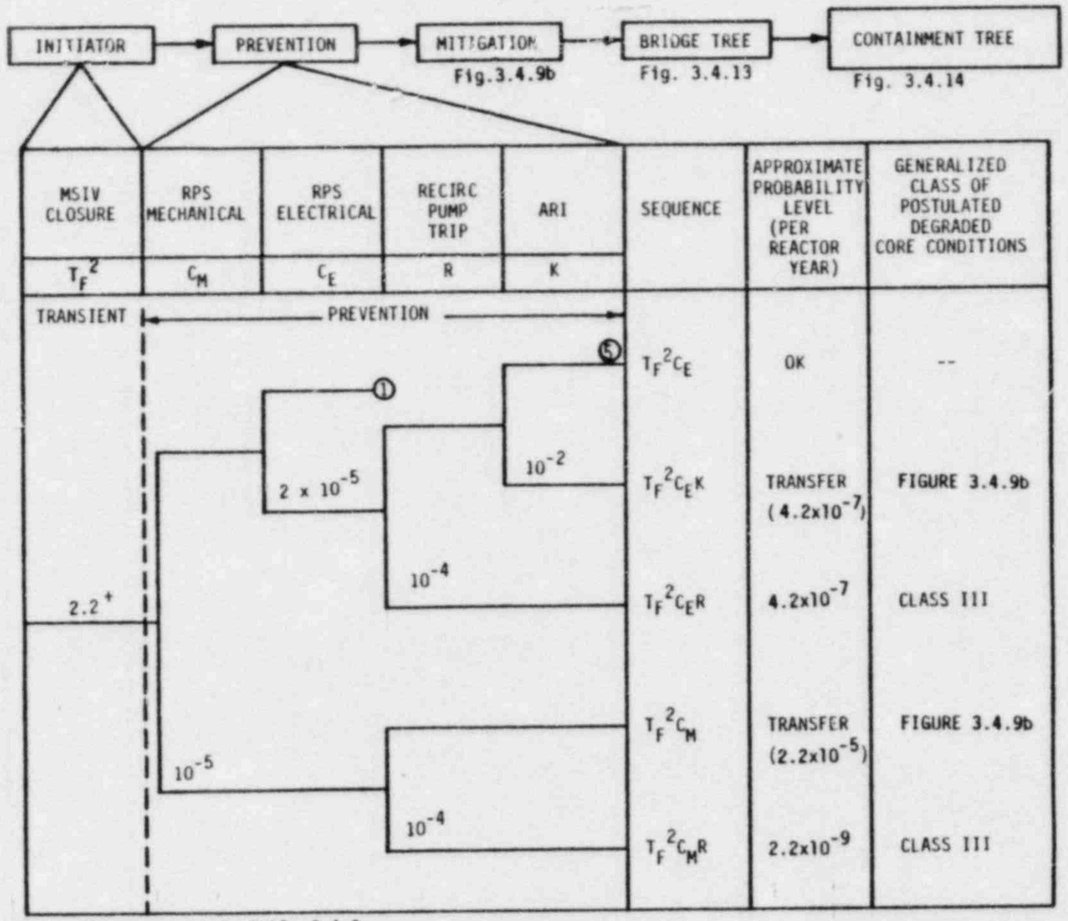
Discussion of MSIV Closure Event Tree Functions

U -- Coolant Injection: The success criteria for postulated MSIV closure ATWS events are given in Section 1.5. Those systems available to supply sufficient coolant injection to the reactor following an MSIV closure ATWS are:

1. HPCI when at least one 43 gpm SLC pump is available for boron injection. Shutdown is calculated to occur within 16 minutes for two SLC pumps operational and 30 minutes with one pump operational. For the one SLC pump case, it may be necessary to provide containment overpressure relief (COR).

2. RCIC when both SLC pumps are available for boron injection.

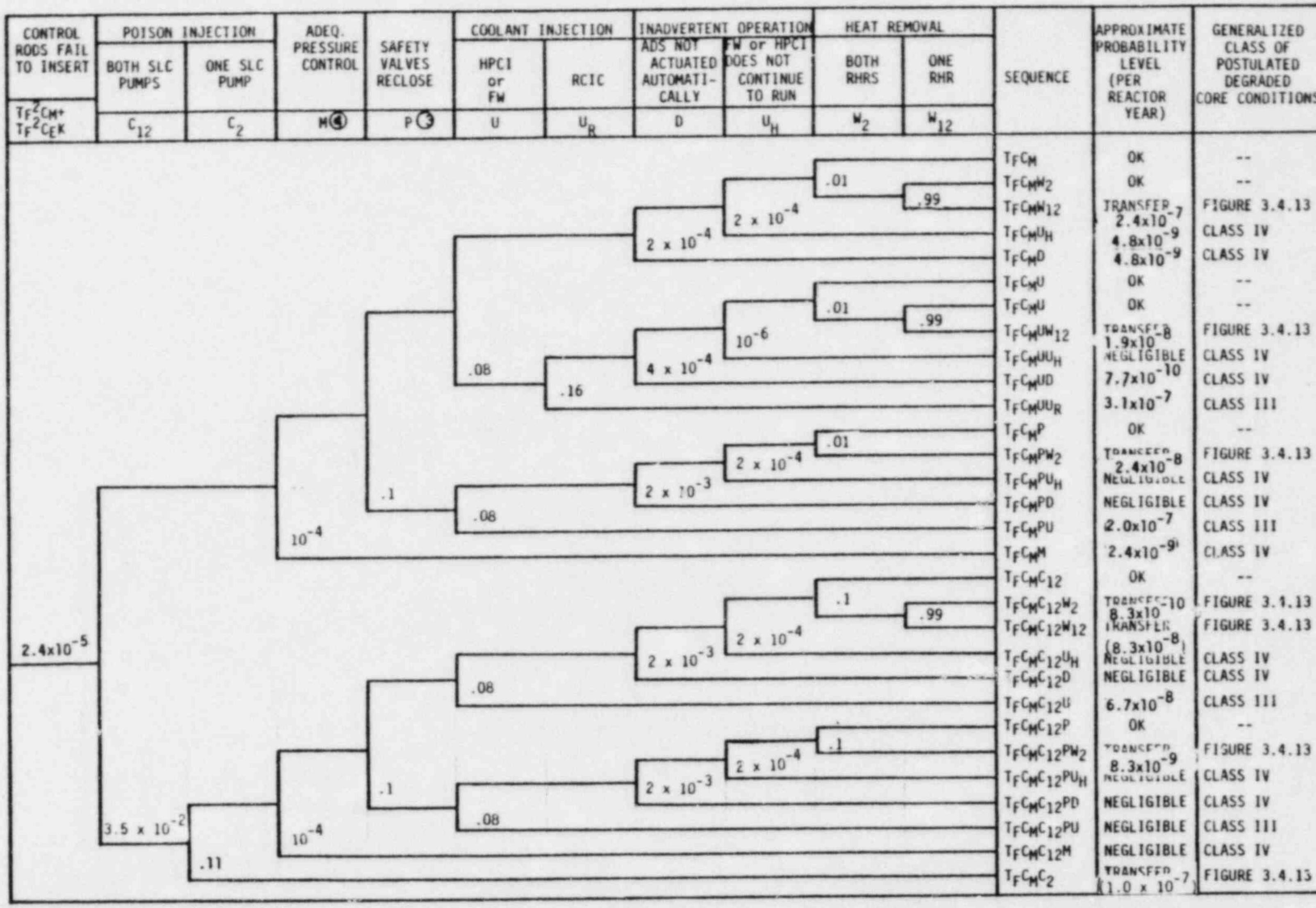
*Feedwater is supplied by turbine driven feedwater pumps. While there does exist a bypass line around the MSIVs, this path may not be available, with any high degree of assurance, following an ATWS, because of the short reaction time available.



All notes are in Table 3.4.2a.

+This initiator is the sum of initiators considered in Figure 3.4.3 and those transferred from Figure 3.4.7.

Figure 3.4.9a Event Tree Diagram of Postulated ATWS Accident Sequences Following an MSIV Closure Initiator



All notes are in Table 3.4.2a.

Figure 3.4.9b Event tree Diagram of Postulated ATWS Accident Sequences Following An MSIV Closure Initiator

The other sources of high pressure water have not been explicitly used in the reliability evaluation of the coolant injection function since analysis does not predict that adequate coolant injection will occur from any combination of these sources.

W -- Heat Removal. The probability of having adequate heat removal available is a function of the RHR reliability under the various conditions set in the specific accident sequences. For example, multiple stuck open relief valves (P) leads to a requirement for both RHR subsystems to operate.

Because of the relatively short time available for the initiation of RHR following a postulated ATWS, the calculated probability of RHR failure is dominated by human error in failing to align and initiate the RHR.

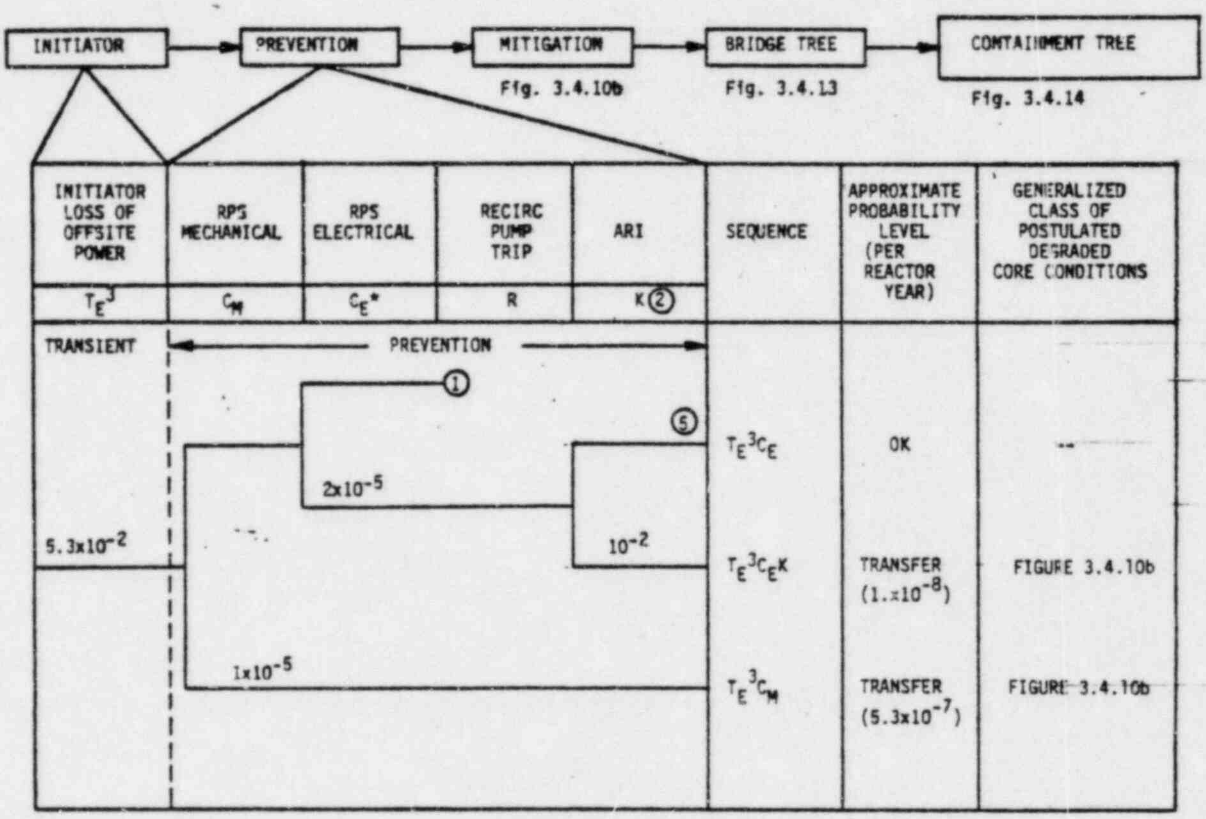
Discussion of ATWS MSIV Closure Sequences

The MSIV closure initiator, followed by a mechanical failure of the control rods to insert, places a demand on three safety systems which are provided to prevent degraded core conditions (see Section 1.5). The accident sequences which would result from a failure of each of these safety systems could be quite different in their impact on risk. The following discussion outlines the course of each accident sequence as developed in the Limerick analysis:

SEQUENCE	ANTICIPATED RESULT
$T_F C_M C_2$	This accident sequence is similar to that discussed for the turbine trip ATWS initiator followed by failure of liquid poison injection. The physical processes which are included in the modeling are: • Recirculation pumps trip and power level drops to ~ 25%. • The HPCI maintains adequate coolant inventory to the reactor. • All heat is being deposited in the suppression pool. • Both SLC pumps fail to inject poison. • Containment pressure rises to containment design pressure. • At this point we transfer to the Bridge tree (Figure 3.4.13), where the following functions are shown: a. COR is not used to relieve containment pressure because of the potential for radiation in containment from fuel failures with no SLC.

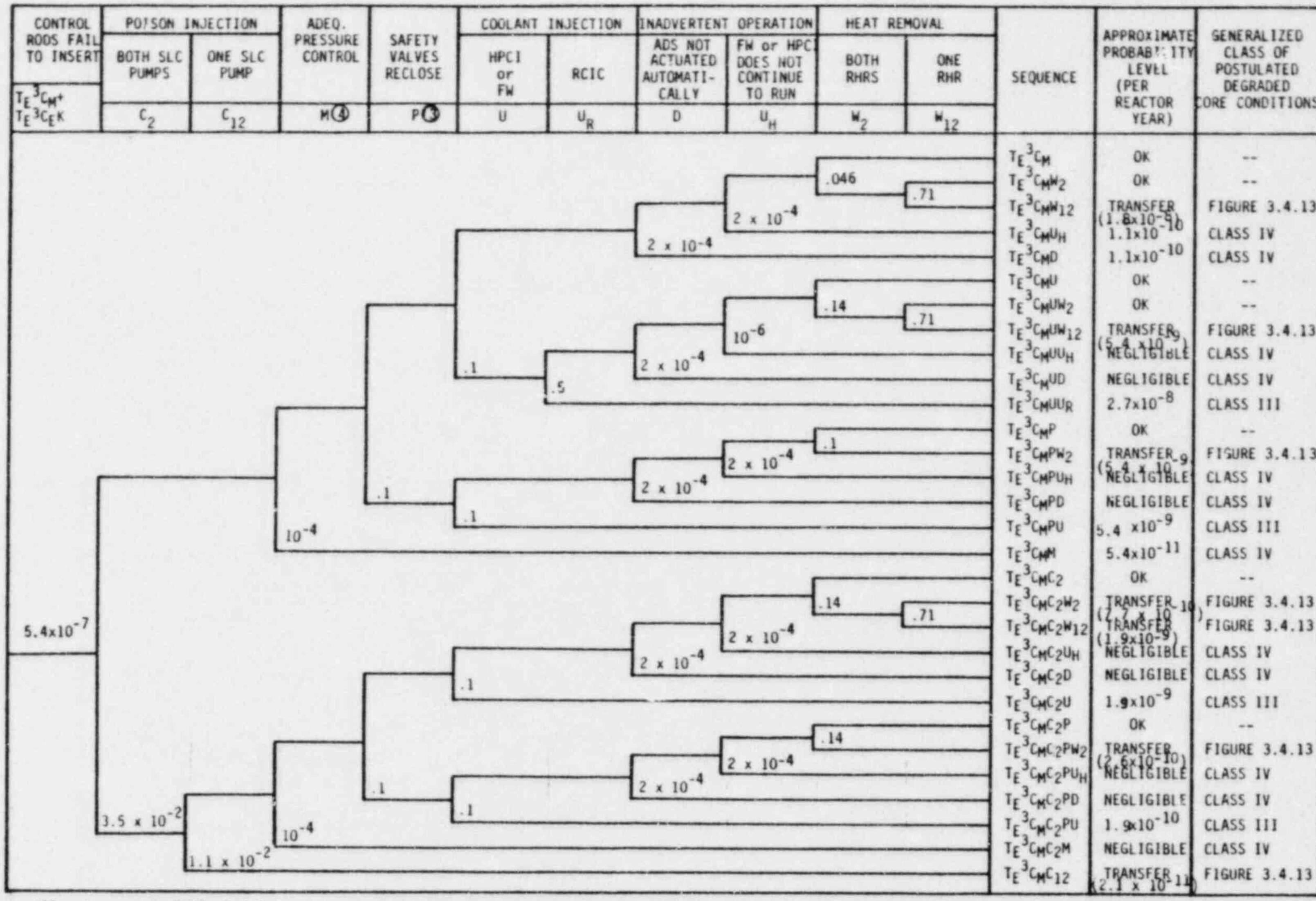
	b. HPCI may shut off due to high turbine exhaust pressure. • Based on the decisions made probabilistically in the bridge tree, containment may fail: a. prior to core melt initiation with no COR and HPCI continuing to run. This sequence is referred to as $T_FC_MC_2$ Mode 1, and is included in Class IV, since radioactive releases may be emitted directly to the atmosphere during the core vaporization. b. following core melt if HPCI terminates as it is designed to. This sequence is referred to as $T_FC_MC_2$ Mode 1/3, and is included in Class III since the containment still serves the function of retention and deposition of radioactive material during the core melt/vaporization process.
$T_FC_MUU_R$	This sequence is the failure to provide adequate coolant makeup to the reactor. The physical processes involved in this sequence are: • Recirculation pump trips and power level drops to ~ 25% • With the MSIVs closed, feedwater is unavailable • Both SLC pumps fail to inject poison • HPCI and RCIC fail • With no coolant injection, the core water level drops • With the loss of coolant inventory, core melt is initiated • Containment pressure is relatively low • Containment is calculated to fail in a manner similar to Class I -- after the core vaporization phase. The result of such a sequence is that initiation of degraded core conditions and postulated core melt would occur with the containment intact. This is typical of Class III accident sequences.
$T_FC_MC_{12}U$	This sequence is similar to that noted above for loss of coolant inventory, however, in this accident sequence one SLC pump leg is available for injection. RCIC is insufficient to maintain adequate coolant inventory (see the ATWS Success Criteria in Section 1.5). Therefore, for sequences with one SLC, where the HPCI system provides coolant injection, its failure leads to Class III events.
T_FC_MW	This sequence involves the inability to adequately remove heat from containment using the RHR. The physical processes involved are: • The recirculation pumps trip and power drops to ~ 25%. • Both HPCI and SLC start automatically to maintain coolant inventory and provide negative reactivity for core shutdown. • However, since there is no heat removal from containment (MSIVs are closed and RHR has not been initiated) containment pressure continues to rise, • The bridge tree (Figure 3.4.1.3) displays the options which may occur following this accident scenario, these include: a. Use of COR to relieve containment pressure and maintain its integrity by preventing rupture. b. Loss of HPCI due to high containment pressure* leading to core melt initiation with an intact containment. This is a Class III event. c. If HPCI continues to run and COR fails, failure of containment may occur prior to initiation of core melt. This sequence is a Class IV event.

*Turbine exhaust pressure trip.



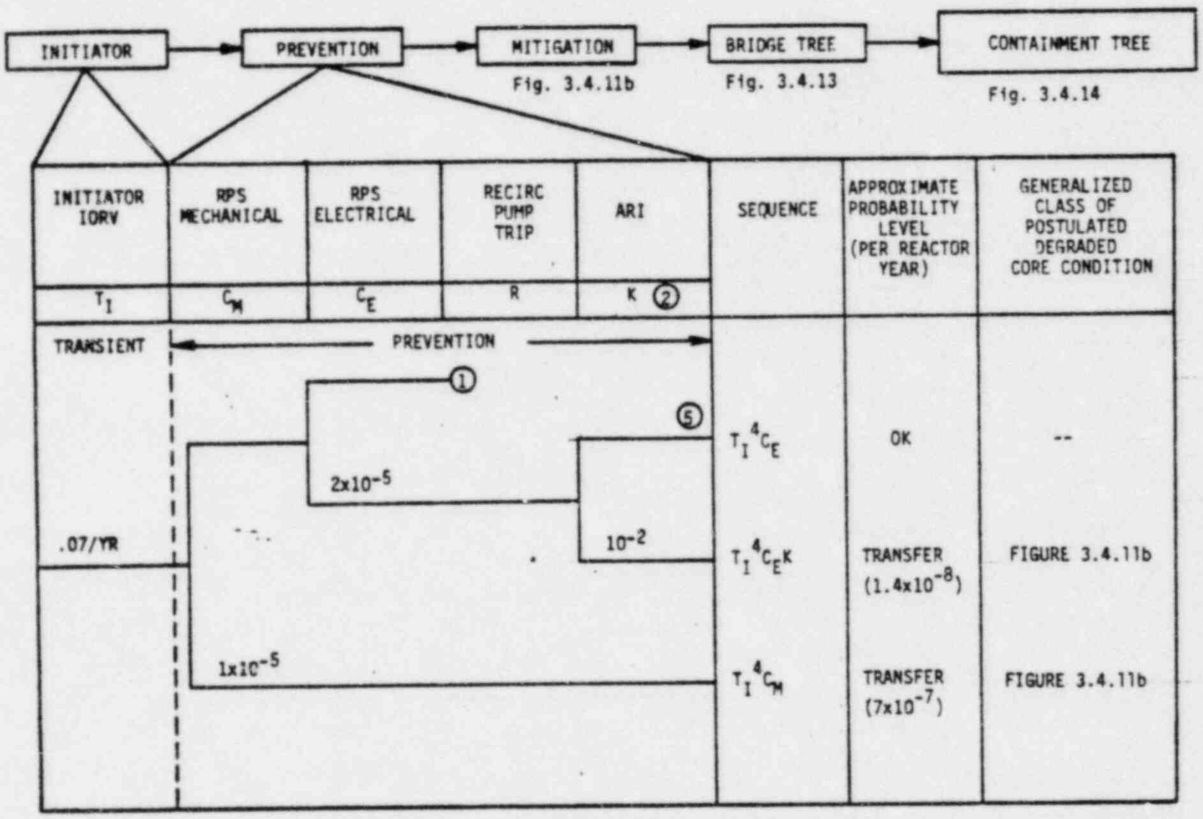
* For the loss of offsite power initiator, electrical faults leading to a failure to scram may be virtually zero for loss of offsite power incidents. However, since no detailed evaluation has been performed to verify this assertion, electrical RPS failures are included here for completeness. They are a small contribution to the overall probability of degraded core conditions.

Figure 3.4.10a Event Tree Diagram of Accident Sequences Following a Loss of Offsite Power Initiator



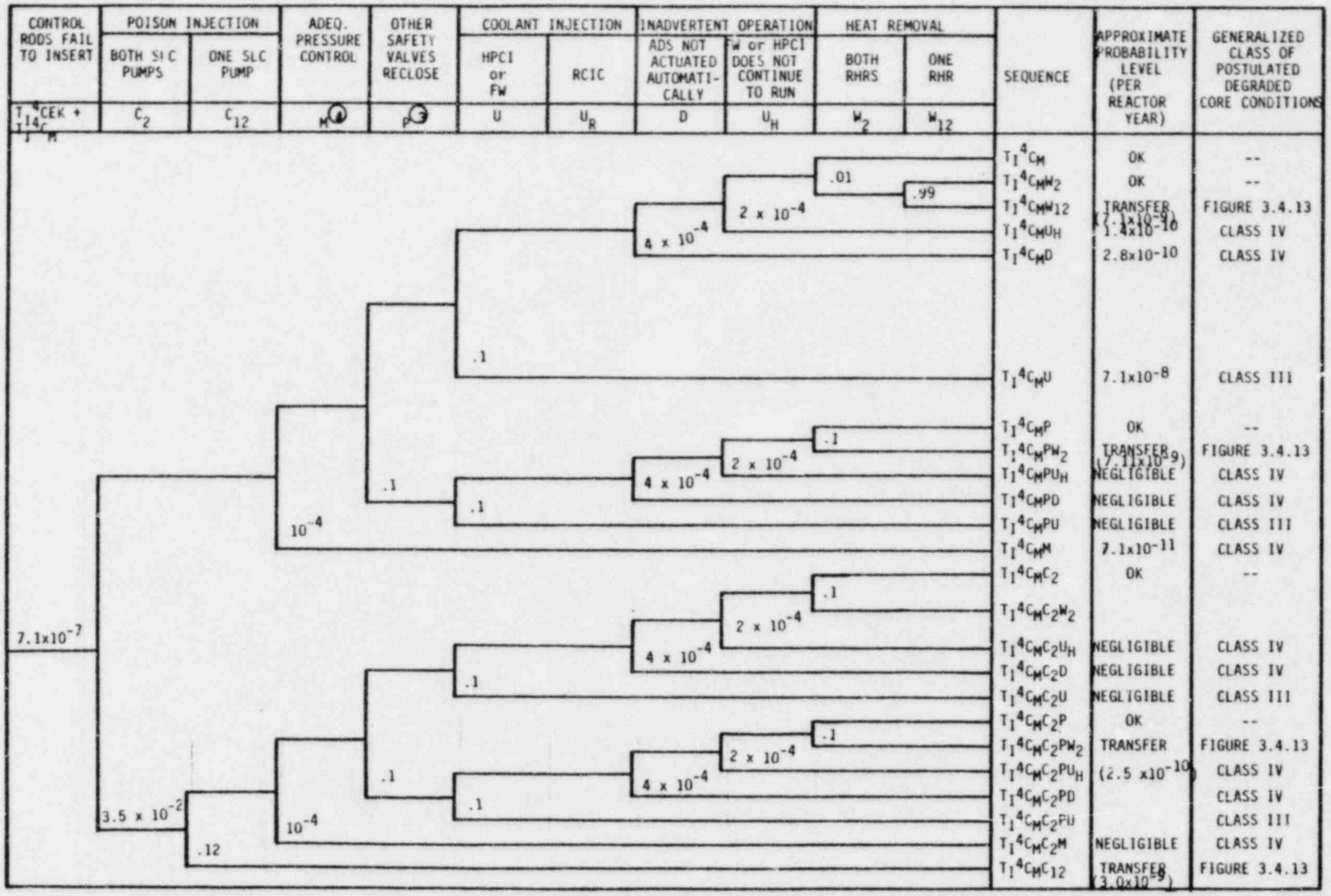
All notes are in Table 3.4.2a.

Figure 3.4.10b Event Tree Diagram of Postulated ATWS Accident Sequences Following a Loss of Offsite Power Initiator



All notes are in Table 3.4.2a.

Figure 3.4.11a Event Tree Diagram of Accident Sequences Following an IORV Initiator



All notes are in Table 3.4.2a

Figure 3.4.11b Event Tree Diagram of Postulated ATWS Accident Sequences Following an IORV Initiator

Table 3.4.2a

NOTES FOR ATWS EVENT TREES

1 Estimated probability for these branches has been considered in Figure 3.4.1.

2 Heat removal capability from the suppression pool may be coupled to "safety valves fail to reclose." Therefore, a different probability of failure is applicable depending upon which sequence occurs (i.e., PW) is treated as a distinct type of event.

3 For the inadvertently open relief valve transient, the scram and ARI are initiated manually; therefore, the probability of success must be modified to account for this.

4 Adequate pressure control has three meanings in these event trees. Each function must be fulfilled or unacceptable consequences result.

5 The scenario following control rod insertion is that developed in WASH-1400, however since ARI was not considered in WASH-1400, a success of this point represents a reduction in the probability of unacceptable consequences calculated in WASH-1400.

Table 3.4.2b

DEFINITIONS OF FUNCTIONS OF EACH SYSTEM APPLIED IN THE ATWS EVENT TREE DEVELOPMENT

DESIGNATION	SYSTEM	FUNCTION
C_M	Reactor Protection System (Mechanical)	The RPS has been divided into electrical and mechanical functions for this study. The mechanical function includes the operation of the CRD hydraulic system,, the physical insertion of a sufficient number of contol rods to bring the reactor subcritical, and other mechanical parts as required.
C_E	Reactor Protection System (Electrical)	This portion of the RPS includes proper generation of a scram signal from the sensors, processing the signal through the logic, and the De-energizing of the scram solenoids.
C_2	Poison Injection	Termination of nuclear fission is required to assure containment and core integrity. Note that delayed control rod insertion has not been assumed.
R	Recirculation Pump Trip	This system is designed to be completely diverse from the RPS (both electrically and mechanically) in order to guarantee that either the RPS or RPT will function. The RPT is intended to trip the recirculation pumps which will reduce the flow through the core and lead to reduced moderation in the core and lower core power level.
I	Containment Isolation	In the event of an ATWS provision must be made for preventing the release of radioactive material to the environment. One method is the isolation of containment which includes the closure of the MSIVs (see also secondary containment isolation).
V	Isolation of Balance of Plant Systems	One potential scenario which could occur following an ATWS is the removal of reactor heat through the condenser via turbine bypass valves. The secondary system will function properly to remove heat from the reactor as long as high radiation levels are not sensed, low level (level 1) is not reached, and low steam line pressure (e.g. LOCA) does not occur. Therefore, operation of the PCS to remove heat is dependent upon the severity of the ATWS event, namely, the fuel integrity and water level following an ATWS.

Table 3.4.2b (continued)

DESIGNATION	SYSTEM	FUNCTION
U	Coolant Injection	The coolant injection function is to put sufficient water in the reactor vessel to maintain the core covered. The methods available to perform this function vary with the transient and the assumptions used. The systems available for each case are delineated in the functional level fault trees.
W	Heat Removal	Heat is removed from the reactor by steam blowdown through the safety relief valves to the suppression pool. However, the heat must also be removed from the suppression pool or a failure of the suppression pool could result. The methods available to remove heat from the suppression pool are described in the function level fault trees for each transient.
K	Alternate Rod Insertion (ARI)	ARI has the following characteristics: a) ARI is not effective if there is a mechanical failure of control rods to insert. b) ARI is effective if and only if RPT is successful.
M	Adequate Pressure Control	This category considers the cases which involve failure to adequately control reactor pressure. The failure modes considered are: a) Overpressure protection fails b) ADS operates to depressurize the reactor when not required. c) A sufficient number of safety valves remain open (stuck or manually open) to depressurize the reactor when undesired. Failure modes a,b, or c may lead to the actuation of the low pressure injection systems and the flushing of boron from the reactor.
P	Safety Valves Reclose	Under this category, the effects of up to 2 relief valves remaining stuck open (SORV) during the transient are included.

3.4.3.1.3 Loss of Offsite Power ATWS Initiator

An accident initiator which directly affects the reliability of the mitigating systems is the loss of offsite power. This initiator leads to conditions similar to the MSIV closure, with the added requirement that some of the safety systems depend on a diesel generator for the necessary power.

The event tree for this initiator is given in Figure 3.4.10 and is nearly identical to that for MSIV closure, except that the initiator frequency* is lower than the MSIV closure frequency, but the probability of failure of the mitigation systems is higher, due to the dependency of the safety systems on the emergency diesels.

3.4.3.1.4 Inadvertent Open Relief Valve ATWS Initiator (See Figure 3.4.11)

The IORV accident initiator is a relatively frequent event that also requires manual initiation of scram and RHR. Therefore, the human error rate has a significant effect on the calculated probability of successful accident mitigation. (See the discussion in Section 3.4.1.)

3.4.3.2 Reactor Pressure Vessel Failure

Disruptive failure of the reactor pressure vessel is not incorporated in the Limerick analysis because potential failures of the reactor pressure vessel as an initiating event have a very low probability of expected occurrence.

3.4.3.3 Interfacing LOCA

The Reactor Safety Study (WASH-1400) identified the potential for a combination of a LOCA and check valve failure, which could lead to a di-

*Based upon operating experience data on the PJM grid rect release of radioactive material to the environment. The Limerick plant has been evaluated to determine if a similar set of conditions could arise. No case similar to the PWR interfacing LOCA conditions has been identified.

3.4.4 Bridge Event Tree

There are a number of postulated events which may lead to containment overpressure conditions due to the inability to remove heat from containment via either:

- The power conversion system (PCS), or
- The RHR system.

It was assumed in WASH-1400 that loss of the above two containment heat removal systems led directly to containment failure, and loss of all coolant injection. The Limerick Generating Station has more capability to withstand such challenges than assumed for the WASH-1400 BWR. The principal improvements affecting containment heat removal are:

1. Incorporation of Containment Overpressure Relief (COR) capability at LGS (see Appendix B). New emergency guidelines for BWRs provide criteria for containment overpressure relief when there is no significant radiation inside containment.
2. Incorporation of pumps not requiring a positive back pressure to pump at high suppression pool temperatures.

Maintaining containment integrity allows a great deal of flexibility to the plant operator for maintaining adequate core cooling from a variety of sources, despite the inability to remove heat from containment via the RHR or PCS. Containment pressure relief is of importance for all TW sequences.

Those sequences which may be affected by the controlled pressure relief of containment warrant additional analysis to demonstrate the circumstances under which the reactor core can be maintained adequately cooled despite the inability to maintain containment heat removal by either RHR or the power conversion system. To analyze this effect, the plant systems and containment interaction must be evaluated. There are cases where failure of suppression pool cooling may not lead to core melt. Such cases include:

- Containment pressure control by controlled pressure relief
- Containment leakage without containment failure
- Minor breaches of containment with subsequent reclosure of the concrete fissures.

For the LGS analysis, disruptive containment failure is assumed to lead to ultimate failure of the coolant makeup system to the reactor. The estimates of containment ultimate failure capability for Limerick are developed from the following:

- Analysis of the Mark II concrete structure response under high internal pressure loads (see Appendix J)
- Analysis of containment dome and hatch design response (see Appendix J)
- WASH-1400 estimates of the PWR concrete containment structure response
- Evaluations of pipe penetration integrity under high internal containment pressure.

The containment failure criterion used in the Limerick analysis is based upon exceeding 140 psig (approximately 2.7 times design pressure), which occurs at more than 30 hours after initiation of TW. However, the Limerick emergency guidelines used in this analysis specify the implementation of containment overpressure relief (COR) at a point near the containment design pressure, if there is no abnormal radiation level inside containment and the core is covered.

This section focuses on those sequences which can take advantage of COR (containment overpressure relief). Specifically the following three types of sequences are assessed:

1. Anticipated transients for which the RHR or PCS is not available (TW)

2. ATWS events for which the SLC system and HPCI operate, but the RHR and PCS are unavailable (ATWS-W). There are two types of such events: those where one SLC pump fails and those where both fail.

3. ATWS events for which the SLC system also fails (ATWS-C_2).

In Sections 3.4.1, 3.4.2, and 3.4.3 the dominant accident sequences have been developed on a basis similar to that used in WASH-1400. However, the sequences do not all lead to core melt as assumed in WASH-1400. For those TW-type sequences where containment overpressure is the failure mode, core melt will not occur if coolant makeup can be maintained during the subsequent long term overpressure relief of containment. Since the containment event trees developed in Section 3.4.5 use core melt as the initiating event, a connection must be made between the sequences developed analogous to WASH-1400 and those which lead to core melt (i.e., TW type). Figure 3.4.12 is a simplified flow chart showing the progression of accident sequences from initiators through containment pressure relief and to potential containment release pathways. The purpose of the bridge event tree, Figure 3.4.13, is to provide this connection between the containment overpressure sequences developed previously, and the containment event trees provided in Section 3.4.5.

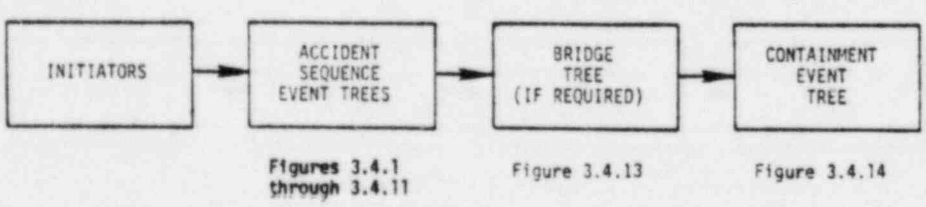


Figure 3.4.12 Flow Chart of the Event Trees Used to Define Accident Sequences

The applicability of the use of COR is assessed for each type of accident sequence. The following sections discuss how the bridge tree (Figure 3.4.13) is used to evaluate each of the different types of accident sequences:

1. Section 3.4.4.1: Loss of containment heat removal following an anticipated transient (TW sequences)

2. Section 3.4.4.2: Loss of containment heat removal (RHR) following an ATWS event (ATWS-W sequences)

3. Section 3.4.4.3: Mismatch between containment heat removal and heat production following an ATWS with loss of all SLC poison injection (ATWS-C_2 sequences)

4. Section 3.4.4.4: Combination of items (2) and (3) (ATWS-C_{12} sequences).

3.4.4.1 Bridge Tree Applied to Sequences Involving a Loss of Containment Heat Removal Following an Anticipated Transient. (TW Type Sequences)

Successful termination of a TW sequence without abnormal radioactive release to the environment will result if adequate reactor core cooling can be maintained by water from one of four sources using the associated motive forces shown in Table 3.4.2.

Table 3.4.2

SUMMARY OF SOURCES OF WATER AND PUMPS AVAILABLE FOR COOLANT MAKEUP TO THE REACTOR WHEN THE CONTAINMENT IS AT OR NEAR DESIGN PRESSURE

WATER SOURCE	PUMPS
1. CST	HPCI, CRD, LPCS
2. Suppression Pool	LPCS, HPCI, LPCI
3. Hotwell	Condensate
4. Spray Pond.	RHRSW

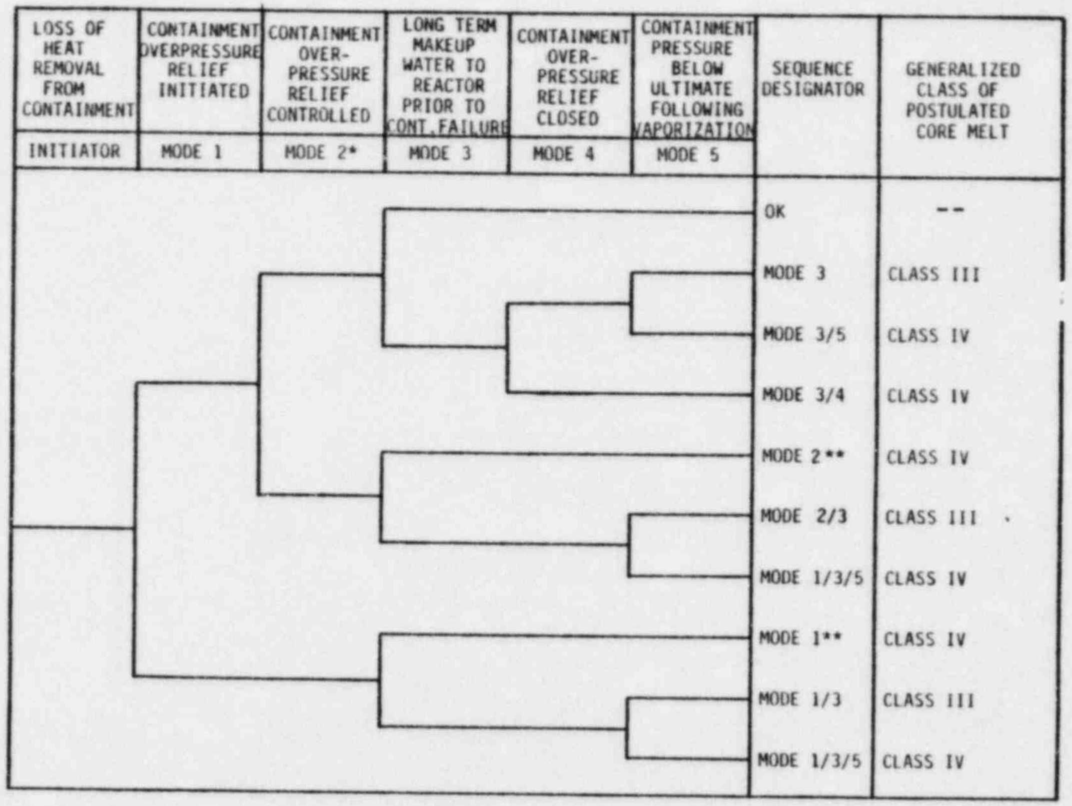
For the Limerick analysis, containment heat removal is successful with RHR and PCS unavailable if COR successfully operates. The heat would be removed from containment by steam passing from the reactor through the safety relief valves, through the suppression pool, through the drywell and directly out the Containment Overpressure Relief system to the atmosphere. It is assumed that once initiated, pressure relief from the reactor will continue to be successful during this process with appropriate reliability. Figure 3.4.13 is the bridge tree for the TW type sequences. The method of quantifying and evaluating is is presented below.

The following discussion of the bridge tree as applied to TW sequences is provided to clarify the event descriptions:

<u>TW -- Initiating Event</u>. For the TW event to occur, the RHR system and the Power Conversion System must be unavailable. For the RHR to be inoperable, either the RHRSW is not available to the RHR heat exchangers or the LPCI pumps are not operable. These two events are evaluated as approximately equally likely to occur. The availability of the LPCI pumps will affect the success criteria used in both the TW event trees from Section 3.4.1 and the bridge tree; therefore, in the TW-type events the common dependencies of "W" and Mode 3 functions need to be accounted for. This is accomplished by combining the entire sequence in a Boolean fashion (see Appendix B).

<u>Event Mode 1 -- Failure of Containment Overpressure Relief</u>. This represents the process of opening the containment vent as described in the emergency procedure guidelines. Failure of COR is assumed to result in containment failure sequences similar to the TW sequences described in WASH-1400. The overpressure relief procedure is assumed to require:

- Indication of high containment pressure
- Indication of RPV water level
- Indication of low radiation in the containment
- Operator action (manual action from control room)
- Power to COR valves (emergency power bus).



* Mode 2 is equivalent to Mode 1 in its impact on the containment.

** The assumption used in the LGS Risk Analysis is that containment failure leads to loss of long term coolant injection with a probability of one.

Figure 3.4.13 "Bridge" Event Tree Providing the Link Between Postulated Transient and LOCA Accident Sequences Which May Result in Containment Overpressure (see Figures 3.4.1 through 3.4.10) and the Containment Event Sequences Following Core Melt (see Figure 3.4.14).

Event Mode 2 -- Failure to Maintain Overpressure Relief Over the Long Term. The failure of any of the requirements of Mode 1 may result in closure of the COR valves. In addition, the COR valves may be closed to prevent rapid blowdown, and then fail to reopen.

Event Mode 3 -- Failure of Coolant Makeup to the Reactor Vessel. A functional fault tree in Appendix B describes the loss of reactor coolant makeup failure modes. Four sources of makeup water are available to the operator and all must be lost for an event Mode 3 to occur. The sources are: 1) the suppression pool via LPCS, HPCI, or LPCI pumps; 2) the condensate storage tank via HPCI, LPCS, RCIC*, or CRD: 3) the hotwell via the condensate pump; or 4) the spray pond via the RHR service water system. Availability of these systems varies according to the failure mode causing TW, the initiating transient, as well as closure of the COR valve to fail to relieve containment pressure.

Event Mode 4 -- Failure of COR Valves to Reclose. Once COR has been initiated, there is a possibility that conditions in the core may deteriorate (i.e., Mode 3) such that the COR valves should be reclosed to provide an intact containment. The failure to reclose the COR valves due to mechanical problems or human error is assessed in Mode 4.

Event Mode 5 -- Long-Term Makeup Fails and Containment Integrity Fails. Mode 5 is a decision point used to define the possibility that following a loss of long-term coolant injection (Mode 3) with the containment at relatively high pressure that the ensuing postulated core melt, RPV failure, molten core-concrete interaction, and containment heat load may all combine to lead to a containment failure prior to the radionuclide vaporization releases (see Section 3.6). This possibility is only considered for those sequences associated with high containment pressures prior to initiation of core melt and is assumed to lead to radionuclide releases comparable to that of Class IV.

Table 3.4.3 summarizes the effects of each of the bridge tree event sequences for those processed by the bridge tree.

In summary, preserving containment integrity is important to the evaluation of the TW sequence. Preserving containment integrity (through the incorporation of a pressure relief function) means that the only other

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*RCIC trips off automatically at high containment pressure.

Table 3.4.3

BRIDGE TREE EVENT SEQUENCES IMPACT

SEQUENCE	FAILURE MODE	IMPACT	TIME FRAME
	None	OK	NA
Mode 1	COR Fails	Delayed Core Melt	27 Hours
Mode 2	COR Fails	Delayed Core Melt	27 Hours
Mode 3	Coolant Makeup Fails	Core Melt (Similar to TQUV)	2-10 Hours
Mode 3/4*	COR Fails Open and Coolant Makeup Fails	Core Melt (Direct Release)	2-10 Hours
Mode 5*	Long Term Make-up Fails and Containment Integrity Fails	Potential Direct Release From Containment Following Core Melt	2-10 Hours

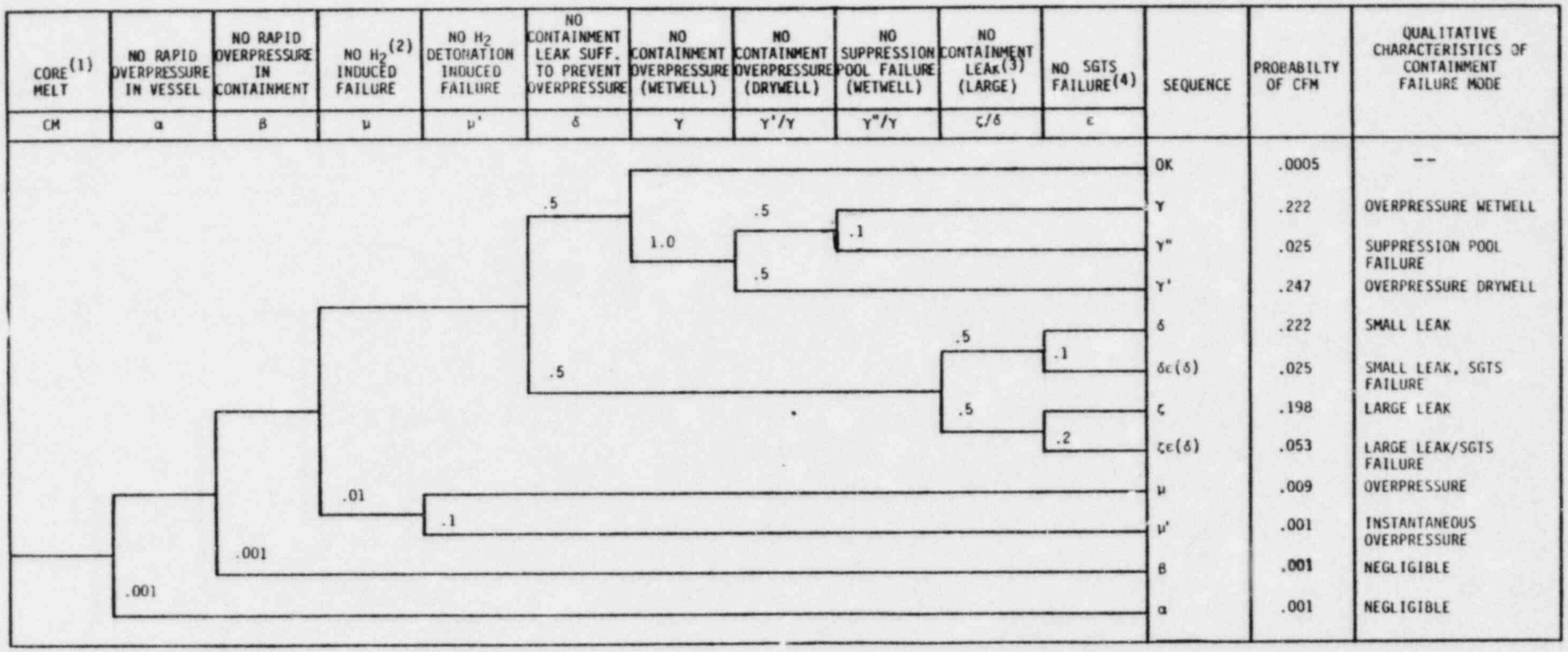
*Mode 4 is treated the same as Mode 3

function required to maintain core coverage is makeup water. This can be accomplished from a number of water sources as shown in Table 3.4.2. The functional fault tree for a long term coolant makeup to the reactor is described in Appendix B.

3.4.4.2 Loss of Containment Heat Removal (RHR) Following An ATWS Event (ATWS-W Type Sequences)

The bridge event tree is used for the ATWS sequences involving the inability to remove heat from containment. Figure 3.4.13 is again used to process those sequences for which containment heat removal fails following an ATWS event. The important features of the ATWS-W bridge tree are the following:

> ATWS-W. An ATWS plus loss of containment heat removal (W) does not necessarily lead to inadequate core cooling, since the inclusion of containment overpressure relief (COR) provides a viable alternative to maintain containment integrity and remove heat from containment if both liquid poison and coolant injection are not successful.



(1) Containment failure may have occurred prior to core melt. In those cases (Class II and Class IV), the containment failure modes are only used as mechanisms for release fraction determination.

(2) Assumes that H_2 explosion in containment causes overpressure failure with direct pathway to outside atmosphere.

(3) Leakage at 2400 volume percent/day.

(4) Failure standby gas treatment system.

Figure 3.4.14 Containment Event Tree for the Mark II Containment

Mode 1 failures are those involving the success of coolant makeup to the reactor despite failure to maintain containment pressure within design limits following an ATWS. This involves HPCI operating successfully beyond its normal limits (see Appendix B). Failures of this type are considered to lead to containment failure prior to core melt (Class IV), so that the fission product releases to the drywell have an immediate and direct path outside containment. This type of failure is considered similar to the TW sequences except that there may be more heat stored in the fuel resulting in a more energetic release, melt may occur more quickly, and a larger radioactive source term may result. Class IV has its own unique release fractions (see Appendices C and D).

Mode 3 and Mode 1/3 failures lead to accident scenarios similar to Class III accidents; that is, the containment is at elevated pressure prior to the initiation of a degraded core condition, but maintains its integrity throughout the core melt and vaporization phases.

Mode 3/4 failures are grouped into Class IV since they have similar effects to those noted above for Mode 1; that is, the containment is not intact when the postulated core melt occurs. The reason for the loss of containment integrity is the failure to isolate the COR system following initiation of core melt.

Mode 1/3/5 failures are similar to Mode 1 failures. Mode 5 implies that failure of coolant injection occurs but the core melt/core vaporization does not occur until after containment failure. This failure mode is assigned a low probability.

In summary, the ATWS-W bridge tree displays the possible outcomes of an ATWS event followed by a failure to remove the heat from containment. The outcomes are classed as: (1) acceptable for the cases which involve successful COR; (2) Class III events; (3) Class IV events involving a containment which is not intact prior to incipient core melt from relatively high power.

3.4.4.3 Mismatch of Containment Heat Removal and Heat Production Following an ATWS with Loss of all SLC Poison Injection (ATWS-C_2 Type Sequences) (See Figure 3.4.13)

The bridge event tree also assists in classifying the possible sequences resulting from an ATWS event in which the diverse shutdown mechanism (SLC) also fails. This type of event is evaluated to have a low probability; however, the consequences may be very high. The key features of the ATWS-C_2 bridge tree are as follows:

Mode 1 and Mode 2 -- Containment Overpressure Relief. The analysis is similar to that discussed above; however, by the nature of the accident it is assumed that there is a high probability of steam generation in excess of COR capacity or sufficient fuel failures may occur resulting in an automatic interlock preventing COR from operating. Therefore, because of these two factors, the probability of COR preserving the core integrity given ATWS-C_2 accident sequences is felt to be low.

Mode 3 -- Makeup Water to Reactor. The design of Limerick includes specific features to shut off both high pressure safety systems (HPCI and RCIC) on high containment pressure*. Since these features are included in the design, a high level of success is accorded the shut off of the high pressure systems for this sequence. However, the interlock or trip can be bypassed, so it is assumed that the possibility exists that the operator will ignore the interlock and restart HPCI.

Mode 3/4 -- COR Valves Fail to Reclose. Given that COR has operated and that coolant makeup water is lost, the COR valves may also not reclose. This is a low probability event and does not significantly contribute to the probability of Class IV events.

Mode 1/3/5 -- Loss of Containment Integrity Before Core Melt With Loss of Coolant Injection. Because of the relatively rapid increase in reactor pressure associated with ATWS, and failure of the SLC, the containment pressure is expected to rise sharply. Following HPCI shutoff on high turbine exhaust pressure, the containment may fail due to high internal pressure.

3.4.4.4 ATWS Events Coupled With Loss of One SLC Pump (ATWS-C_{12} Type Sequences)

When only one SLC pump is available for poison injection, the outcome of ATWS events may differ from the two pump case so these sequences have been treated separately to add specificity to the determination of ATWS events.

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*The shutoff is on high turbine exhaust pressure, and is for the purpose of protecting the turbine.

Mode 1 and Mode 2 -- Containment Overpressure Relief. The analysis is the same as discussed above; however, since some poison injection does occur, reactor subcriticality will take place in approximately 30 minutes. Therefore, the probability of success of COR and the failure modes are similar to those discussed under ATWS-W.

3.4.5 Containment Event Tree Description

The containment event tree developed for the Limerick analysis differs from the containment event tree appearing in WASH-1400 through differences in containment design and operation of safety systems. The changes are reflected in the following areas:

- Containment structural capability of the Mark II steel-lined concrete strucutre versus the Mark I/BWR steel shell containment used in WASH-1400
- The internal configuration of the drywell and its relationship to the wetwell
- The adequacy of the secondary containment building for processing any small leak releases from the primary containment
- The elevation of the release and the release fraction as a function of the various containment failure modes.

Figure 3.4.14 presents the containment event tree which describes the possible failure modes of the Mark II containment. For some core melt classes (Classes II and IV), the containment is taken to be failed prior to core melt. For these classes, the containment event tree represents the probability that the release of radioactive material following core melt is via a particular path. The failure modes used in the quantification of accident sequences for input to the ex-plant consequence analysis (CRAC) are calculated for the four types of core melt initiators.

Using the four classes of core melt initiators (as defined in Section 3.4.0), the containment event tree yields ten sequences for each of these classes, or on the order of forty sequences which can be analyzed by CRAC. Some collapsing of sequences into groupings by containment failure mode was performed and is discussed in Section 3.4.5.3.

3.4.5.1 Containment Event Tree -- Event Definition

The following discussion provides a qualitative description of each of the events considered in the containment event tree.

CM -- Core Melt. This is the initiating event used to enter the containment event tree. It provides a link between the containment event tree and the accident sequences developed in Sections 3.4.1 through 3.4.4. For the LGS risk assessment, the types of initiators used to enter the containment event tree have been divided into four classes as discussed in the introduction to Section 3.4.

β -- Containment Steam Explosion. Subsequent to reactor vessel melt-through, the molten core may fall on the diaphragm floor, melt through the floor, and fall into the suppression pool in such a state that a coherent steam explosion may occur. This phenomena may lead directly to failure of both primary and secondary containment. This event has been assigned a probability of .001. See Appendix H for a more detailed discussion of this phenomenon. This event focuses on the postulated scenario in which sufficient hydrogen is generated in a core melt sequence to allow potentially explosive mixtures of hydrogen to exist in containment. Hydrogen combustion is of concern if any of the following conditions exist:

- An accident occurs during a period when containment is deinerted.
- The containment inerting system fails undetected and sufficient oxygen accumulates in containment to allow an explosive mixture to be possible during a core melt.
- Subsequent to core melt a containment failure occurs which would result in oxygen in-flow into the primary containment.

μ' -- Hydrogen Detonation. While a combustible mixture of hydrogen may exist within containment for the reasons cited above, the conditional probability that the mixture would detonate (shock wave propagation) is felt to be less than the probability of deflagration.

γ -- Containment Overpressure. Given that no containment leakage occurs, the possibility exists of containment overpressure following a core melt. The LGS containment pressure capability is estimated to be approximately 140 psig (see Appendix J for further discussion). For core melt sequences where no leakage occurs, 140 psig may be reached, or the molten core interaction with the concrete diaphragm floor may lead to structural failures which in turn lead to a breach of containment.

γ'/γ -- Containment Overpressure (split between wetwell and drywell failure. Failure of containment due to overpressure has been divided into two types because of the potential difference in radioactive release terms for the case of failure in the drywell and direct release to the stack, versus a failure in the wetwell, where release would be through the suppression pool. Failure at very high containment pressure may occur with equal likelihood in the wetwell or drywell. Therefore, $\gamma'/\gamma=0.5$.

γ'' -- Overpressure Failure in the Wetwell Below the Suppression Pool Water Level. A rupture in the wetwell may be of sufficient size to lead to a loss of water from the suppression pool. Such a failure mode may lead to higher consequences than those calculated for γ, since no pool scrubbing is assumed for this failure mode. The probability of this occurrence is small.

ζ -- Large Leak. The size of the leak from the primary containment is important in determining the radioactive releases to the environment. Specifically, small leaks may be handled effectively by the standby gas treatment system. However, larger leaks may be too large to be effectively processed by the SGTS. (For the large leak, the conditional probability of the SGTS operating is assessed as a factor of two less than for the small leak.)

For the purposes of the LGS study the assumption is made that for Class IV event sequences the containment pressurization is sufficiently rapid to result in some form of overpressure rupture, that is, leaks (i.e., low release fraction sequences) are precluded in the Class IV analysis. This assumption is the best estimate of the containment response under these conditions.

<u>ε -- Standby Gas Treatment System -- Secondary Containment</u>. This event represents the capability of the SGTS to process the effluent of the primary containment to the secondary containment.

<u>$\varepsilon(\delta)$ -- Standby Gas Treatment System -- Secondary Containment, Given Primary Containment is Not Intact</u>. This event represents the capability of the SGTS to process the effluent of a leak in the primary containment due to containment isolation failure or vapor suppression failure. This event also includes sequences where containment leakage occurs. Success of this system is dependent on the primary containment leakage rate. Failure within the SGTS itself is also considered.

3.4.5.2 Additional Comments on the Containment Capability

One notable change from the method used to develop the containment event tree in WASH-1400 as compared to the LGS analysis is the treatment of containment leakage. Containment leakage was eliminated from the LGS accident sequence event trees (Section 3.4.1 through 3.4.4) and treated within the containment event tree (see Figure 3.4.14). Linkage of the containment event tree with core melt sequences is made directly. The containment event tree represents only the short term response of containment to the core melt. Long term effects which may occur over the period of many days, at elevated temperature and pressure, are not modeled in this analysis.

The LGS analysis is similar to WASH-1400 in that energetic failures of the primary containment (α,β,μ') are expected to result in immediate failure of the secondary containment. Therefore, for these sequences, no credit is taken for decontamination factors of the secondary containment. The Limerick reactor suppression pool is directly below the reactor vessel and is covered by a concrete diaphragm floor 3'6" thick. The diaphragm floor is drained directly by 4" lines to a sump and through an adjacent area by downcomer pipes to the suppression pool. The drainage capability eliminates the possibility of the molten core dropping in one large mass from the reactor vessel directly into a pool of water. The dropping of the molten core into the suppression pool was assumed for some LOCA cases in WASH-1400. In addition, the Mark II containment floor provides a greater potential for the molten core to spread across a large area increasing the probability that the diaphragm floor will remain intact. Portions of the melt would probably freeze on the floor, or drop through the drain lines to the sump or the downcomers to the suppression pool and be quenched.

3.4.5.3 Grouping of Similar Accident Sequences for Input to CRAC

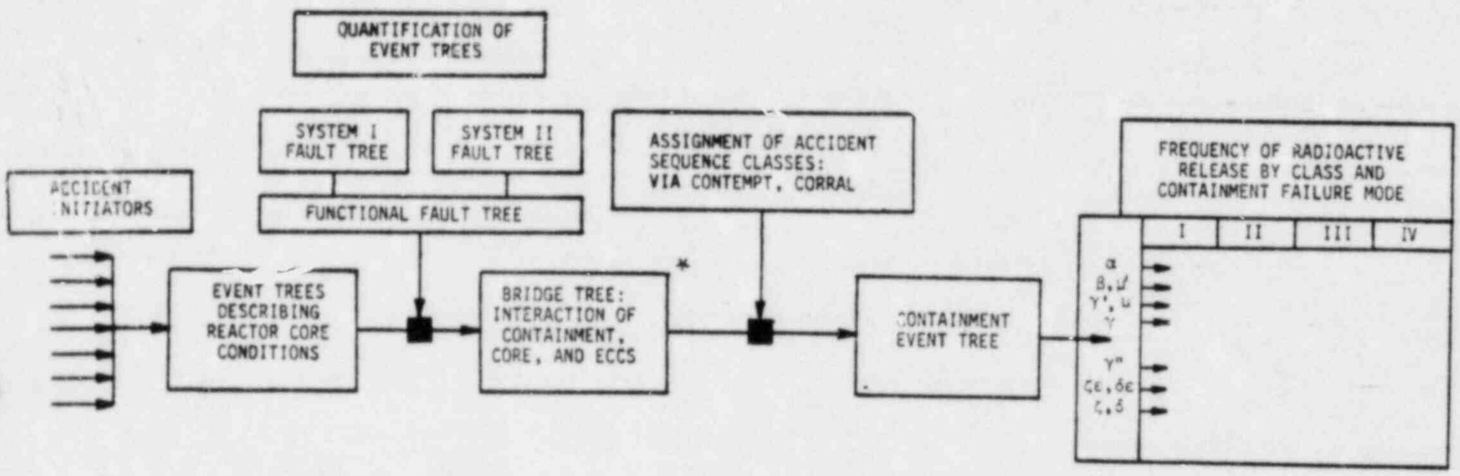
The grouping of similar accident sequences, or sequences which yield similar containment response and consequences, is necessary to reduce the number of unique CRAC runs which need to be performed. The major objective of grouping the sequences together is to make them narrow enough so that they represent a unique set of circumstances which can be realistically evaluated from accident initiators through consequence/risk calculation. This approach avoids the need to develop generalized categories, requiring some technique such as "smoothing"* to account for miscategorization of sequences.

*Smoothing is a technique of the WASH-1400 analysis used to ensure that possible miscategorization of event sequences did not cause the risk to be underestimated.

3.5 QUANTIFICATION OF ACCIDENT SEQUENCES AND IDENTIFICATION OF DOMINANT CONTRIBUTORS

3.5.1 Summary of Accident Sequence Quantification

The approach used in the Limerick risk assessment for quantifying risk is composed of several tasks which are shown schematically in Figure 3.5.1.



*If required

Figure 3.5.1 Flow Chart of Information and Calculations to Determine the Accident Frequency Input to CRAC

To determine the accident frequency to be input to CRAC, the following steps are performed:

1. The accident initiators are identified from operating experience data (see Section 3.2 and A.1)

2. The accident event tree sequences are defined which describe the reactor conditions leading to a postulated core melt (see Section 3.4).

3. The accident sequences are quantified through the use of functional fault trees which combine in Boolean form all the system interactions during the accident scenario.

4. If required, a bridge tree is used to quantify core and containment interactions (Section 3.5.3).

5. All accident sequences are quantified.

6. The dominant accident sequences are then placed into similar groups (Section 3.5.2).

7. The effects on containment due to the accident sequence are then quantified (Section 3.5.4).

8. Lastly, the probabilities associated with each containment failure mode (Class I, II, III or IV) are included (Section 3.5.5) to determine a frequency of release through each identified release path.

3.5.2 Quantification of the Dominant Accident Sequences by Class

The event trees from Section 3.4 identify the accident sequences which are considered in the Limerick risk analysis. In this section the results of the accident sequence quantification are presented, including the following:

1. The scheme used to place accident sequences into one of four classes which produce similar effects on containment

2. The accident sequences which appear in each class

3. The frequency of postulated degraded core conditions for:

    - Each dominant sequence

    - The summation over sequences within a class

4. A comparison between Limerick and WASH-1400 for:

    - Selected accident sequences

    - Overall frequency of degraded core conditions.

A detailed in-plant consequence evaluation has been performed for each of the accident sequences classes, including:

. Modeling of the core and containment physics associated with the postulated core meltdown process. The final result of this analysis is the definition of a set of containment conditions which exist at the time of release and can be used to provide isotopic inventory releases, rate estimates, release energies, and other pertinent radioactive release data.

2. Modeling of the leak path flow/deposition process, to calculate radioactive release fractions inside the primary containment, and fractional attenuation processes. The end result is an evaluation of the release term for each of the possible containment failure modes. The containment effects associated with each class are calculated to characterize the response for that class.

The Class I (C1) events can be characterized as transients involving loss of coolant makeup to the reactor core. For the Limerick analysis, these events are found to have the highest calculated frequency of occurrence. They involve successful control rod insertion; however, there is postulated to be a loss of both high pressure and low pressure injection. The physics model used in the consequence calculation represents the sequence designated TQUV. The CONTEMPT calculation (see Section 3.6 and Appendix C) for this sequence is then used to characterize the response for all of the sequences grouped together in Class I.

For Class II (C2), the sequence modeled is a transient with long-term loss of heat removal. For Limerick, this sequence involves the failure of the power conversion system, the RHR system, and the containment overpressure relief, i.e., TW-Mode 1, from Section 3.4.4. Also included in this class are other sequences, such as LOCAs accompanied by a failure of the containment heat removal systems, and inadvertently open relief valves with failure to remove heat from containment. The key feature in this class is that the containment is assumed to fail prior to core melt, but after a lengthy period of time into the accident. Postulated core melt begins with a relatively low decay heat source, leading to a slower core melt than anticipated for Classes I, III, or IV, but with a failed containment.

Classes III and IV are used primarily to assess the spectrum of potential consequences associated with ATWS sequences (transient followed by a failure to insert the control rods). There are several reasons for creating two separate classes to describe ATWS sequences, these are:

1. ATWS events produce unique interactions between containment and the reactor core, which may be sensitive to timing of events.

2. Two separate types of ATWS interactions with containment are considered*: (1) an ATWS followed by loss of coolant makeup (this is referred to as Class III); (2) an ATWS followed by poison injection failure, but continuing coolant injection (Class IV).

Class III is similar to Class I, except for the initial failure to scram. They both are basically a loss of adequate coolant inventory. Class IV is similar to Class II, since both of these involve a postulated containment failure prior to the initiation of a degraded core condition.

Table 3.5.2 summarizes the dominant failure sequences whose frequency of occurrence contribute cumulatively to the frequency of each class.

The results of quantifying of the dominant sequences for each class is displayed in Tables 3.5.3 through 3.5.6. These tables also show the calculated sequence frequencies by containment failure mode as defined in Section 3.5.4.

These tables provide a focal point for the accident sequence definition and quantification. Using these results, it is possible to:

- Determine the relative importance of the other sequences as they affect each class
- Determine the frequency input to the CCDF calculation using CRAC.

*See also the NRC BWR Rebaseline Case: E. J. Hanraham and L. Bickwit, Jr., "Report to the Commissioners, Subject: Report of the Task Force on Interim Operation of Indian Point", (Docket Nos. 50-247 and 50-286), June 12, 1980

Table 3.5.2

SUMMARY OF GENERAL TYPES OF ACCIDENT SEQUENCES*

	CLASS I	CLASS II	CLASS III	CLASS IV
	RELATIVELY RAPID CORE MELT WITH INTACT CONTAINMENT	RELATIVELY SLOW CORE MELT WITH FAILED CONTAINMENT	RELATIVELY RAPID CORE MELT WITH INCIPIENT CONTAINMENT FAILURE	RELATIVELY RAPID CORE MELT WITH CONTAINMENT FAILURE
PRIME EXAMPLE	TQUV	TW (MODE 1)**	TC_MC_2 (ATWS)	$TC_MC_2\bar{U}$ (MODE 1)
DOMINANT ACCIDENT SEQUENCES	T_TQUV T_TQUX T_FQUV T_FQUX T_EUV T_EUX T_IUV T_IUX S_1UV S_1UX T_MQUX T_MQUV	T_TW T_TPW T_FW $T_FQW(Q)$ T_FPW T_EW T_IW AJ+ S_1QUW S_2W	AI AE $T_T{}^1C_MPU$ $T_T{}^1C_MC_2$(MODE 1/3) $T_F{}^2C_MUU_R$ $T_F{}^2C_MC_2$(MODE 1/3) $T_E{}^3C_MU$ $T_E{}^3C_MC_2$(MODE 1/3) TW (MODE 3) $T_F{}^2C_MC_{12}U_R$ $T_F{}^2C_MPU$ $T_F{}^2C_MR+T_E{}^2C_ER$ $T_E{}^3C_MC_{12}U$ T_IC'' (MODE 1/3)	$T_F{}^2C_MW$ $T_E{}^3C_MW$ $T_I{}^4C_MW$ TW - MODE 3/4 $T_F{}^2C_MC_2\bar{U}$ $T_E{}^3C_MC_2\bar{U}$ $T_I{}^4C_MC_2\bar{U}$ $T_T{}^1C_MC_2\bar{Q}\bar{D}$ T_IC'' $T_T{}^1C_MU_H/D$ $T_T{}^1C_MM$ $T_T{}^1C_ER+T_T{}^1C_MR$ $T_F{}^2C_MU_H/D$

* Each of these types of accident sequences may lead to any of the types of containment failure modes identified in Section 3.4.5.

** See Figure 3.4.13

\+ AJ leads directly to a scenario equivalent to TW-MODE 1 since COR operation following a large LOCA is given a very low probability of occurring.

Table 3.5.3

SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE FOR DOMINANT SEQUENCES OF THE CLASS I VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	α .001	β,μ' .002	γ',μ .258	γ .222	γ'' .025	$\zeta\epsilon,\delta\epsilon$ .078	ζ,δ .42
T_TQUV	2.2×10^{-10}	4.4×10^{-10}	5.7×10^{-8}	4.9×10^{-8}	5.5×10^{-9}	1.7×10^{-8}	9.2×10^{-8}
T_TQUX	4.5×10^{-10}	9.0×10^{-10}	1.2×10^{-7}	1.0×10^{-8}	1.1×10^{-8}	3.5×10^{-8}	1.9×10^{-7}
T_MQUV	7.2×10^{-11}	1.4×10^{-10}	1.9×10^{-8}	1.6×10^{-8}	1.8×10^{-9}	5.6×10^{-9}	3.0×10^{-8}
T_MQUX	2.9×10^{-10}	5.8×10^{-10}	7.5×10^{-8}	6.4×10^{-8}	7.3×10^{-9}	2.3×10^{-8}	1.2×10^{-7}
T_FQUV	1.1×10^{-9}	2.2×10^{-9}	2.8×10^{-7}	2.4×10^{-7}	2.8×10^{-8}	8.6×10^{-8}	4.6×10^{-7}
T_FQUX	4.4×10^{-9}	8.8×10^{-9}	1.1×10^{-6}	9.8×10^{-7}	1.1×10^{-7}	3.4×10^{-7}	1.8×10^{-6}
T_EUV	4.0×10^{-9}	8.0×10^{-9}	1.0×10^{-6}	8.9×10^{-7}	1.0×10^{-7}	2.0×10^{-6}*	NA*
T_EUX	8.4×10^{-10}	1.7×10^{-9}	2.2×10^{-7}	1.9×10^{-7}	2.1×10^{-8}	4.2×10^{-7}*	NA*
T_IQUV	2.0×10^{-10}	4.0×10^{-10}	5.2×10^{-8}	4.4×10^{-8}	5.0×10^{-9}	1.6×10^{-8}	8.4×10^{-8}
T_IQUX	7.8×10^{-10}	1.6×10^{-9}	2.0×10^{-7}	1.7×10^{-7}	2.0×10^{-8}	6.1×10^{-8}	3.3×10^{-7}
S_1QUV	$7\ \times10^{-11}$	1.4×10^{-10}	1.8×10^{-8}	1.6×10^{-8}	1.8×10^{-9}	5.5×10^{-9}	2.9×10^{-8}
S_1QUX	1.4×10^{-12}	2.8×10^{-12}	3.6×10^{-10}	3.1×10^{-10}	3.5×10^{-11}	1.1×10^{-10}	5.9×10^{-10}
S_2QUV	$4\ \times10^{-12}$	$8\ \times10^{-12}$	1.0×10^{-9}	8.9×10^{-10}	1.0×10^{-10}	3.1×10^{-10}	1.7×10^{-9}
S_2QUX	8.0×10^{-13}	1.6×10^{-12}	2.1×10^{-10}	1.8×10^{-10}	2.0×10^{-11}	6.2×10^{-11}	3.4×10^{-10}
APPROXIMATE TOTAL PROBABILITY FOR CLASS I SEQUENCES	1.3×10^{-8}	2.6×10^{-8}	3.4×10^{-6}	2.9×10^{-6}	3.2×10^{-7}	3.0×10^{-6}*	3.4×10^{-6}*

*For loss of offsite power cases no credit is taken for the S GTS (which is powered from normal power supplies). However, since the S GTS would not be required until 24-40 hours following the loss of offsite power initiator, this assumption is judged to slightly overestimate the releases from these sequences.

The remainder of this section discusses the accident sequence calculations in various ways to provide a comparison with previous BWR probabilistic risk assessments. A summary chart of all the identified accident sequences within each class is given in Figure 3.5.2. This histogram provides a visual display of the calculated relative frequency of potential degraded core conditions for each of the classes of accident sequences. Also displayed on Figure 3.5.2 for comparison is the total frequency of postulated core melt taken from WASH-1400 (all values expressed as mean values).

Table 3.5.4
SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE FOR DOMINANT SEQUENCES OF THE CLASS II VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	.001	.002	.258	.222	.025	.078	.42
T_TW (mode 1)	$4.4x10^{-11}$	$8.8x10^{-11}$	$1.1x10^{-8}$	$9.8x10^{-9}$	$1.1x10^{-9}$	$3.4x10^{-9}$	$1.8x10^{-8}$
T_TPW (mode 1)	$8.8x10^{-11}$	$1.8x10^{-10}$	$2.3x10^{-8}$	$2.0x10^{-8}$	$2.2x10^{-9}$	$6.9x10^{-9}$	$3.7x10^{-8}$
T_MW (mode 1)	$3.6x10^{-11}$	$7.2x10^{-11}$	$9.3x10^{-9}$	$8.0x10^{-9}$	$9.0x10^{-10}$	$2.8x10^{-9}$	$1.5x10^{-8}$
T_FQW (Q)(mode 1)	$2.0x10^{-10}$	$4.0x10^{-10}$	$5.2x10^{-8}$	$4.4x10^{-8}$	$5.0x10^{-9}$	$1.6x10^{-8}$	$8.4x10^{-8}$
T_FW (mode 1)	$5.6x10^{-11}$	$1.1x10^{-10}$	$1.4x10^{-8}$	$1.2x10^{-8}$	$1.4x10^{-9}$	$4.4x10^{-9}$	$2.4x10^{-8}$
T_FPW (mode 1)	$3.9x10^{-11}$	$7.8x10^{-11}$	$1.0x10^{-8}$	$8.7x10^{-9}$	$9.8x10^{-10}$	$3.0x10^{-9}$	$1.6x10^{-8}$
T_EW_d (mode 1)	$1.2x10^{-11}$	$2.4x10^{-11}$	$3.1x10^{-9}$	$2.7x10^{-9}$	$3.0x10^{-10}$	$6.0x10^{-9}$*	NA*
T_EPW_d (mode 1)	$1.2x10^{-12}$	$2.4x10^{-12}$	$3.1x10^{-10}$	$2.7x10^{-10}$	$3.0x10^{-11}$	$6.0x10^{-10}$*	NA*
T_IW (mode 1)	$1.5x10^{-10}$	$3.0x10^{-10}$	$3.9x10^{-8}$	$3.3x10^{-8}$	$3.8x10^{-9}$	$1.2x10^{-8}$	$6.3x10^{-8}$
$T_IC'W$ (mode 1)	$1.5x10^{-11}$	$3.0x10^{-11}$	$3.9x10^{-9}$	$3.3x10^{-9}$	$3.8x10^{-10}$	$1.2x10^{-9}$	$6.3x10^{-9}$
T_IC'' (mode 1)	4×10^{-13}	$8.0x10^{-13}$	$1.0x10^{-10}$	$8.9x10^{-11}$	$1.0x10^{-11}$	$3.1x10^{-11}$	$1.7x10^{-10}$
AJ	$1.6x10^{-10}$	$3.2x10^{-10}$	$4.1x10^{-8}$	$3.6x10^{-8}$	$4.0x10^{-9}$	$1.2x10^{-8}$	$6.7x10^{-8}$
S_1W (mode 1)	$1.6x10^{-11}$	$3.2x10^{-11}$	$4.1x10^{-9}$	$3.6x10^{-9}$	$4.0x10^{-10}$	$1.2x10^{-9}$	$6.7x10^{-9}$
S_2W (mode 1)	negligible						
S_1QW (Q)	$5.6x10^{-11}$	$1.1x10^{-10}$	$1.4x10^{-8}$	$1.2x10^{-8}$	$1.4x10^{-9}$	$4.4-10^{-9}$	$2.4x10^{-8}$
APPROXIMATE TOTAL PROBABILITY FOR CLASS II SEQUENCES	$8.6x10^{-10}$	$1.7x10^{-9}$	$2.2x10^{-7}$	$1.9x10^{-7}$	$2.2x10^{-8}$	$7.2x10^{-8}$*	$3.5x10^{-7}$*

*For loss of offsite power cases no credit is taken for the SGTS (which is powered from normal power supplies). However, since the SGTS would not be required until 24-40 hours following the loss of offsite power initiator, this assumption is judged to slightly overestimate the releases from these sequences.

Table 3.5.5

SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE FOR DOMINANT SEQUENCES OF THE CLASS III VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	α .001	β,μ' .002	γ',μ .258	γ .222	γ'' .025	$\zeta\epsilon,\delta\epsilon$ .078	ζ,δ .42
$T_T^1C_MPU$	2.98×10^{-10}	5.96×10^{-10}	7.69×10^{-08}	6.62×10^{-08}	7.45×10^{-09}	2.32×10^{-08}	1.25×10^{-07}
$T_T^1C_MPW_2$ (MODE 1/3)	1.1×10^{-11}	2.2×10^{-11}	2.83×10^{-09}	2.44×10^{-09}	2.75×10^{-10}	8.58×10^{-10}	4.62×10^{-09}
$T_T^1C_MC_2PW_2$ (MODE 1/3)	3.8×10^{-12}	7.6×10^{-12}	9.80×10^{-10}	8.44×10^{-10}	9.5×10^{-11}	2.96×10^{-10}	1.60×10^{-09}
$T_T^1C_MC_2$ (MODE 1/3)	7.8×10^{-11}	1.56×10^{-10}	2.01×10^{-08}	1.73×10^{-08}	1.95×10^{-9}	6.08×10^{-9}	3.28×10^{-8}
$T_F^2C_MR+T_F^2C_ER$	6.60×10^{-12}	1.32×10^{-11}	1.70×10^{-09}	1.47×10^{-09}	1.65×10^{-10}	5.15×10^{-10}	2.77×10^{-09}
$T_F^2C_MUU_R$	2.86×10^{-10}	5.72×10^{-10}	7.38×10^{-08}	6.35×10^{-08}	7.15×10^{-09}	2.23×10^{-08}	1.20×10^{-07}
$T_F^2C_MPU$	2.50×10^{-10}	5.00×10^{-10}	6.45×10^{-08}	5.55×10^{-08}	6.25×10^{-09}	1.95×10^{-08}	1.05×10^{-07}
$T_F^2C_MC_{12}U$	6.30×10^{-11}	1.26×10^{-10}	1.63×10^{-08}	1.40×10^{-08}	1.57×10^{-09}	4.91×10^{-09}	2.65×10^{-08}
$T_F^2C_MW$ (MODE 1/3)	6.8×10^{-11}	1.36×10^{-10}	1.75×10^{-8}	1.51×10^{-8}	1.7×10^{-9}	5.30×10^{-9}	2.86×10^{-8}
$T_F^2C_MPW_2$ (MODE 1/3)	2.4×10^{-11}	4.8×10^{-11}	6.19×10^{-9}	5.33×10^{-9}	6.0×10^{-10}	1.87×10^{-9}	1.10×10^{-8}
$T_F^2C_MC_2$ (MODE 1/3)	4.8×10^{-11}	9.6×10^{-11}	1.24×10^{-8}	1.07×10^{-8}	1.20×10^{-9}	3.74×10^{-9}	2.02×10^{-8}
$T_E^3C_MUU_R$	2.70×10^{-11}	5.40×10^{-11}	6.97×10^{-09}	5.99×10^{-09}	6.75×10^{-10}	1.34×10^{-08}	NA*
$T_E^3C_MPU$	4.30×10^{-12}	8.60×10^{-12}	1.11×10^{-09}	9.55×10^{-10}	1.08×10^{-10}	2.14×10^{-09}	NA*
$T_E^3C_MC_{12}U$	1.50×10^{-12}	3.00×10^{-12}	3.87×10^{-10}	3.33×10^{-10}	3.75×10^{-11}	7.47×10^{-10}	NA*
$T_E^3C_MC_2$	4.30×10^{-12}	8.60×10^{-12}	1.11×10^{-09}	9.55×10^{-10}	1.08×10^{-10}	2.14×10^{-09}	NA*
$T_E^3C_MW$ (MODE 1/3)	7.5×10^{-12}	1.5×10^{-11}	1.94×10^{-9}	1.67×10^{-09}	1.88×10^{-10}	3.74×10^{-09}	NA*
$T_E^3C_MC_2W$ (MODE 1/3)	5.7×10^{-13}	1.14×10^{-12}	1.47×10^{-10}	1.27×10^{-10}	1.43×10^{-11}	2.84×10^{-10}	NA*
$T_I^4C_MW$ (MODE 1/3)	2.1×10^{-12}	4.2×10^{-12}	5.12×10^{-10}	4.66×10^{-10}	5.25×10^{-11}	1.64×10^{-10}	8.82×10^{-10}
T_IC'' (MODE 1/3)	1.3×10^{-11}	2.6×10^{-11}	3.35×10^{-9}	2.89×10^{-9}	3.25×10^{-10}	1.01×10^{-9}	5.46×10^{-9}
$T_I^4C_MC_2W$ (MODE 1/3)	7.3×10^{-13}	1.5×10^{-12}	1.88×10^{-10}	1.62×10^{-10}	1.82×10^{-11}	5.69×10^{-11}	3.07×10^{-10}
$T_I^4C_MC_2$ (MODE 1/3)	1.5×10^{-12}.	3.0×10^{-12}	3.87×10^{-10}	3.33×10^{-10}	3.75×10^{-11}	1.17×10^{-10}	6.30×10^{-10}
AE/AI	2.00×10^{-10}	4.00×10^{-10}	5.16×10^{-08}	4.44×10^{-08}	5.00×10^{-09}	1.56×10^{-08}	8.40×10^{-08}
APPROXIMATE TOTAL PROBABILITY FOR CLASS SEQUENCES	1.4×10^{-9}	2.8×10^{-9}	3.6×10^{-7}	3.1×10^{-7}	3.5×10^{-8}	1.3×10^{-7}*	5.7×10^{-7}*

Table 3.5.6

SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE FOR DOMINANT SEQUENCES OF THE CLASS IV VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	α .001	β,μ' .002	γ',μ .503	γ .443	γ'' .05	$\zeta\epsilon,\delta\epsilon$ .0002	ζ,δ .0002
$T_T^1C_MPW_2$ (MODE 1)	1.48×10^{-12}	2.96×10^{-12}	7.44×10^{-10}	6.56×10^{-10}	7.40×10^{-11}	2.96×10^{-13}	2.96×10^{-13}
(MODE 3/4)	4.60×10^{-13}	9.20×10^{-13}	2.31×10^{-10}	2.04×10^{-10}	2.30×10^{-11}	9.20×10^{-14}	9.20×10^{-14}
$T_F{}^2C_MW$ (MODE 3/4)	9.00×10^{-12}	1.80×10^{-11}	4.53×10^{-9}	3.99×10^{-9}	4.50×10^{-10}	1.80×10^{-12}	1.80×10^{-12}
(MODE 1)	2.85×10^{-12}	5.70×10^{-12}	1.43×10^{-09}	1.26×10^{-09}	1.42×10^{-10}	5.70×10^{-13}	5.70×10^{-13}
$T_E{}^3C_MW$ (MODE 1)	1.00×10^{-12}	2.00×10^{-12}	5.03×10^{-10}	4.43×10^{-10}	5.00×10^{-11}	4.00×10^{-13}	NA*
(MODE 3/4)	9.50×10^{-13}	1.90×10^{-12}	4.78×10^{-10}	4.21×10^{-10}	4.75×10^{-11}	3.80×10^{-13}	NA*
$T_I{}^4C_MW$ (MODE 1)	2.80×10^{-13}	5.60×10^{-13}	1.41×10^{-10}	1.24×10^{-10}	1.40×10^{-11}	5.60×10^{-14}	5.60×10^{-14}
(MODE 3/4)	9.00×10^{-15}	1.80×10^{-14}	4.53×10^{-12}	3.99×10^{-12}	4.50×10^{-13}	1.80×10^{-15}	1.80×10^{-15}
T_1C^* (MODE 1)	1.70×10^{-12}	3.40×10^{-12}	8.55×10^{-10}	7.53×10^{-10}	8.50×10^{-11}	3.40×10^{-13}	3.40×10^{-13}
(MODE 3/4)	5.50×10^{-13}	1.10×10^{-12}	2.77×10^{-10}	2.44×10^{-10}	2.75×10^{-11}	1.10×10^{-13}	1.10×10^{-13}
$T_TC_{M_1}C_2PW_2$ (MODE 1)	5.10×10^{-13}	1.02×10^{-12}	2.57×10^{-10}	2.26×10^{-10}	2.55×10^{-11}	1.02×10^{-13}	1.02×10^{-13}
(MODE 3/4)	1.62×10^{-13}	3.24×10^{-13}	8.15×10^{-11}	7.18×10^{-11}	8.10×10^{-12}	3.24×10^{-14}	3.24×10^{-14}
$T_F{}^2C_{M_1}C_2W_2$ (MODE 1)	3.00×10^{-12}	6.00×10^{-12}	1.51×10^{-09}	1.33×10^{-09}	1.50×10^{-10}	6.00×10^{-13}	6.00×10^{-13}
(MODE 3/4)	1.03×10^{-12}	2.06×10^{-12}	5.18×10^{-10}	4.56×10^{-10}	5.15×10^{-11}	2.06×10^{-13}	2.06×10^{-13}
$T_E{}^3C_{M_1}C_2W_2$ (MODE 1)	7.60×10^{-14}	1.52×10^{-13}	3.82×10^{-11}	3.37×10^{-11}	3.80×10^{-12}	3.04×10^{-14}	NA*
(MODE 3/4)	6.20×10^{-15}	1.24×10^{-14}	3.12×10^{-12}	2.75×10^{-12}	3.10×10^{-13}	2.48×10^{-15}	NA*
$T_IC_{M_1}C_2W_2$ (MODE 1)	1.00×10^{-13}	2.00×10^{-13}	5.03×10^{-11}	4.43×10^{-11}	5.00×10^{-12}	2.00×10^{-14}	2.00×10^{-14}
(MODE 3/4)	3.30×10^{-14}	6.60×10^{-14}	1.66×10^{-11}	1.46×10^{-11}	1.65×10^{-12}	6.60×10^{-15}	6.60×10^{-15}
$T_T^1C_MC_2$ (MODE 1)	2.97×10^{-11}	5.94×10^{-11}	1.49×10^{-08}	1.32×10^{-08}	1.49×10^{-09}	5.94×10^{-12}	5.94×10^{-12}
(MODE 1/3/5)	1.21×10^{-11}	2.42×10^{-11}	6.09×10^{-09}	5.36×10^{-09}	6.05×10^{-10}	2.42×10^{-12}	2.42×10^{-12}
$T_F{}^2C_MC_2$ (MODE 1)	1.80×10^{-11}	3.60×10^{-11}	9.05×10^{-09}	7.97×10^{-09}	9.00×10^{-10}	3.60×10^{-12}	3.60×10^{-12}
(MODE 1/3/5)	7.30×10^{-12}	1.46×10^{-11}	3.67×10^{-09}	3.23×10^{-09}	3.65×10^{-10}	1.46×10^{-12}	1.46×10^{-12}
$T_E{}^3C_MC_2$ (MODE 1)	1.20×10^{-12}	2.40×10^{-12}	6.04×10^{-10}	5.32×10^{-10}	6.00×10^{-11}	4.80×10^{-13}	NA*
(MODE 1/3/5)	4.80×10^{-13}	9.60×10^{-13}	2.41×10^{-10}	2.13×10^{-10}	2.40×10^{-11}	1.92×10^{-13}	NA*
$T_I{}^4C_MC_2$ (MODE 1)	5.60×10^{-13}	1.12×10^{-12}	2.82×10^{-10}	2.48×10^{-10}	2.80×10^{-11}	1.12×10^{-13}	1.12×10^{-13}
(MODE 1/3/5)	2.20×10^{-13}	4.40×10^{-13}	1.11×10^{-10}	9.75×10^{-11}	1.10×10^{-11}	4.40×10^{-1}	4.40×10^{-14}
$T_T^1C_MU_H/D$	1.46×10^{-11}	2.92×10^{-11}	7.34×10^{-9}	6.47×10^{-9}	7.30×10^{-10}	2.29×10^{-12}	2.29×10^{-12}
$T_T^1C_MM$	3.90×10^{-12}	7.80×10^{-12}	1.96×10^{-09}	1.73×10^{-09}	1.95×10^{-10}	7.80×10^{-13}	7.80×10^{-13}
$T_T^1C_ER+T_T{}'C_MR$	1.05×10^{-11}	2.10×10^{-11}	5.28×10^{-9}	4.65×10^{-9}	5.25×10^{-10}	2.10×10^{-12}	2.10×10^{-12}
$T_F{}^2C_MU_H/D$	9.00×10^{-12}	1.80×10^{-11}	4.53×10^{-09}	3.99×10^{-09}	4.50×10^{-10}	1.80×10^{-12}	1.80×10^{-12}
$T_F{}^2C_MM$	2.25×10^{-12}	4.50×10^{-12}	1.13×10^{-09}	9.97×10^{-10}	1.13×10^{-10}	4.50×10^{-13}	4.50×10^{-13}
$T_E{}^3C_MU_H/D$	2.20×10^{-13}	4.40×10^{-13}	1.11×10^{-10}	9.75×10^{-11}	1.10×10^{-11}	8.80×10^{-14}	NA*
$T_E{}^3C_MM$	5.40×10^{-14}	1.08×10^{-13}	2.72×10^{-11}	2.39×10^{-11}	2.70×10^{-12}	2.16×10^{-14}	NA*
$T_I{}^4C_MU_H/D$	4.20×10^{-13}	8.40×10^{-13}	2.11×10^{-10}	1.86×10^{-10}	2.10×10^{-11}	8.40×10^{-14}	8.40×10^{-14}
$T_I{}^4C_MM$	7.10×10^{-14}	1.42×10^{-13}	3.57×10^{-11}	3.15×10^{-11}	3.55×10^{-12}	1.42×10^{-14}	1.42×10^{-14}
APPROXIMATE TOTAL PROBABILITY FOR CLASS SEQUENCES	1.3×10^{-10}	2.7×10^{-10}	6.72×10^{-8}	5.9×10^{-8}	6.7×10^{-9}	2.7×10^{-11}*	2.5×10^{-11}*

WASH-1400 basically used one class of accident sequence and five containment failure modes to represent BWRs. Therefore, for the purposes of estimating the total calculated frequency of a potentially degraded core condition, the Limerick classes should be summed and compared with the value from WASH-1400. The Limerick evaluation produces a total estimate of degraded core conditions smaller than WASH-1400 (see also Figure 3.5.4). Figure 3.5.2 indicates that the events are all of relatively low probability.

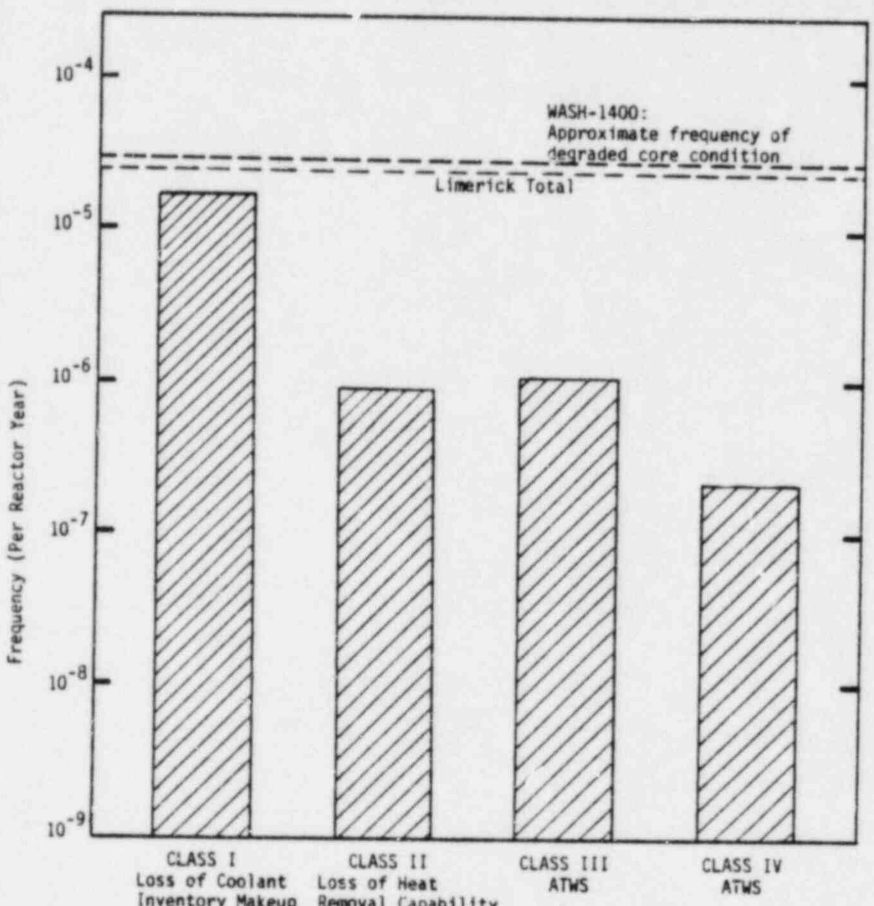


Figure 3.5.2 Summary of the Accident Sequence Frequencies Leading to Degraded Core Conditions Summed Over All Accident Sequences within a Class.

Each of the accident classes has been examined in further depth to determine the principal initiators and sequences which make up the individual classes. Figure 3.5.3 summarizes, in histogram format, some of the dominant sequences by class. The frequency of these sequences is displayed for each sequence. Note that the loss of coolant inventory sequences are calculated to have the highest frequency of potential degraded core conditions. Smaller contributors include ATWS events, large LOCA, and small LOCA. Loss of containment heat removal sequences have a relatively low probability when compared with WASH-1400 estimates primarily due to the inclusion of controlled containment overpressure relief (COR) at LGS.

The accident sequences which dominate the overall estimated frequency of postulated degraded core conditions are:

- Loss of offsite power (T_EQUV, T_EQUX)
- Loss of coolant makeup to the reactor following loss of feedwater or MSIV closure (T_FQUV, T_FQUX)
- ATWS events followed by a failure of high pressure coolant injection or poison injection ($T_F^2C_mC_2$, $T_F^2C_mU$)
- Large LOCA (AE, AJ)
- Medium LOCA (S, UV).

Table 3.5.7 provides a comparison of the calculated values for some dominant sequences from WASH-1400 versus the values calculated for the Limerick analysis. Figure 3.5.4 provides a graphical display of the calculated core melt frequencies from WASH-1400 and Limerick.

3.5.3 Quantification of the Bridge Event Tree*

The Limerick analysis was performed making use of a containment design feature which will prevent overpressure failures under certain circumstances. This containment overpressure relief (COR) feature consists

*This information is used in deriving the frequencies given in Section 3.5.2.

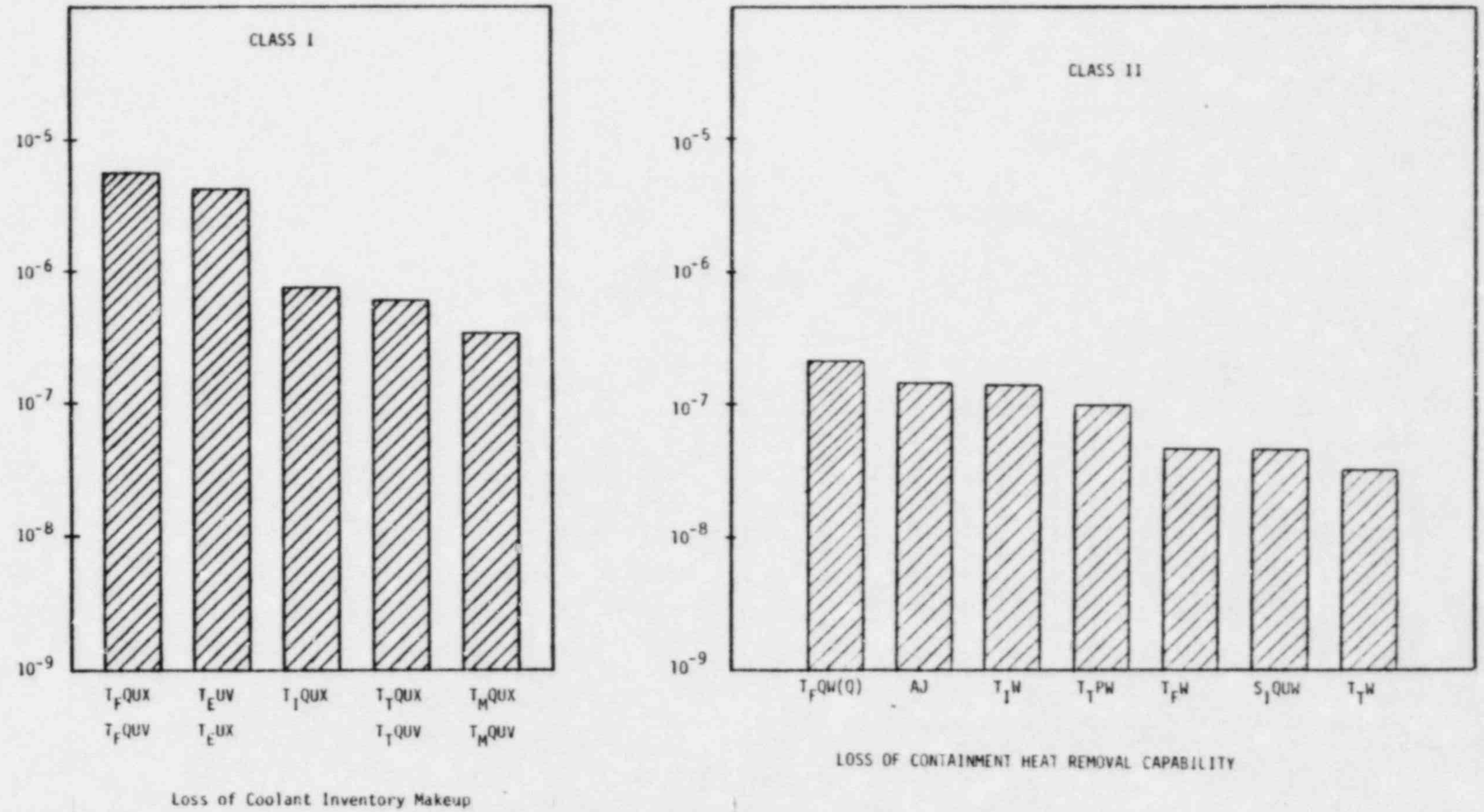
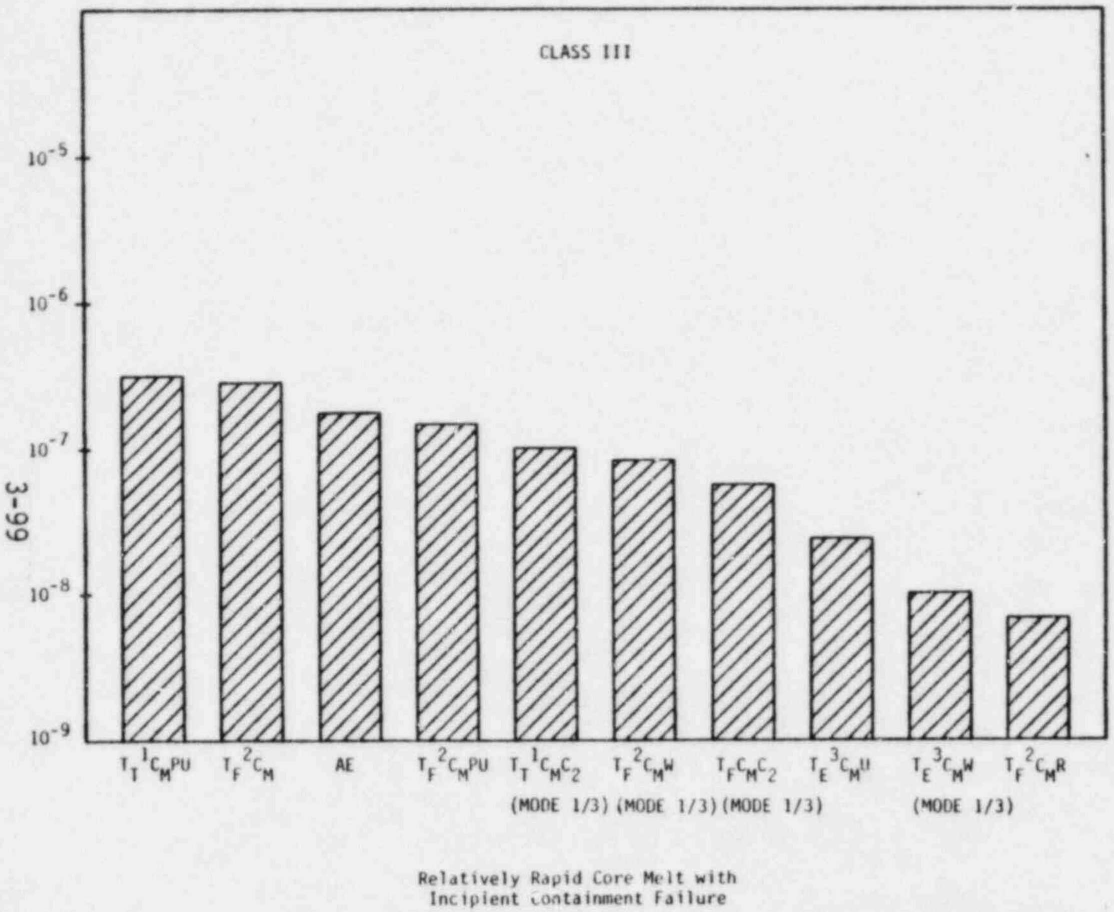


Figure 3.5.3a Summary of Dominant Accident Sequences Presented by Class



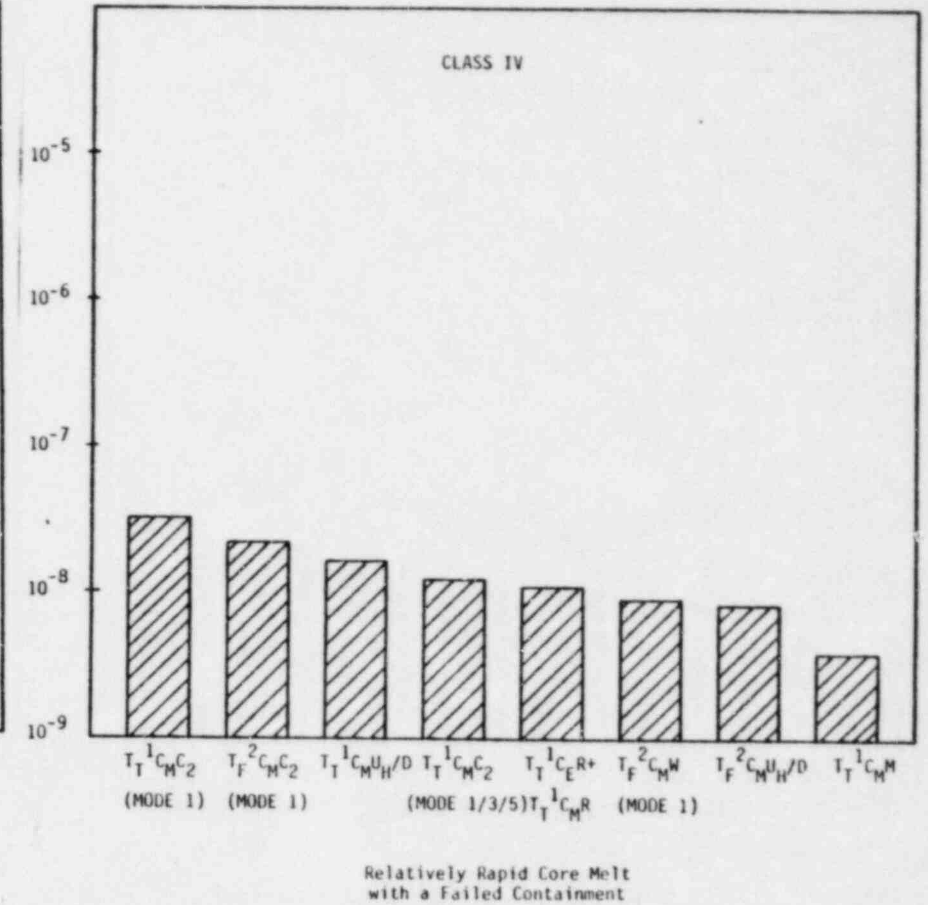


Figure 3.5.3b Summary of Dominant Accident Sequences Presented by Class

of a set of valves which can be operated from the control room to relieve pressure in the containment (see Appendix B). Since the valves are assumed to be interlocked to high radiation monitors, COR can only be utilized for cases where no significant radiation has been released to the drywell. These cases are generally the Class II and some Class IV types of sequences, involving the inability to remove heat from containment. For these two classes, the reactor core is adequately cooled; the major concern is maintaining the containment intact and within its pressure capability, while insuring no offsite consequences. (In considering COR, a conservative analysis, using 5% worst meteorology*, a semi-infinite cloud model**, and a realistic noble gas source, showed that offsite doses would be less than one five-hundredth of permissible guidelines (10CFR100), and would result in no offsite consequence, based on exposure of the population.)

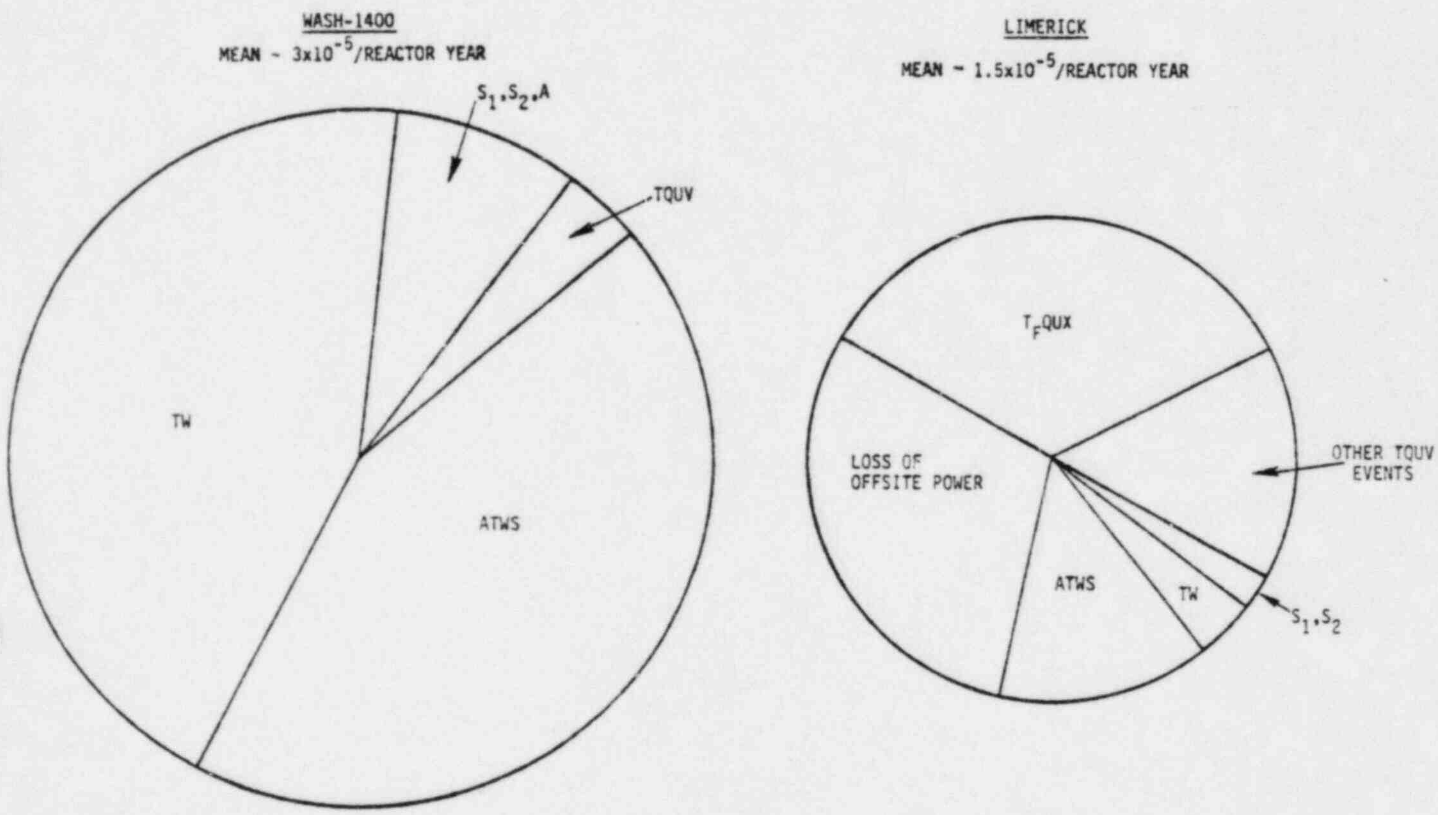
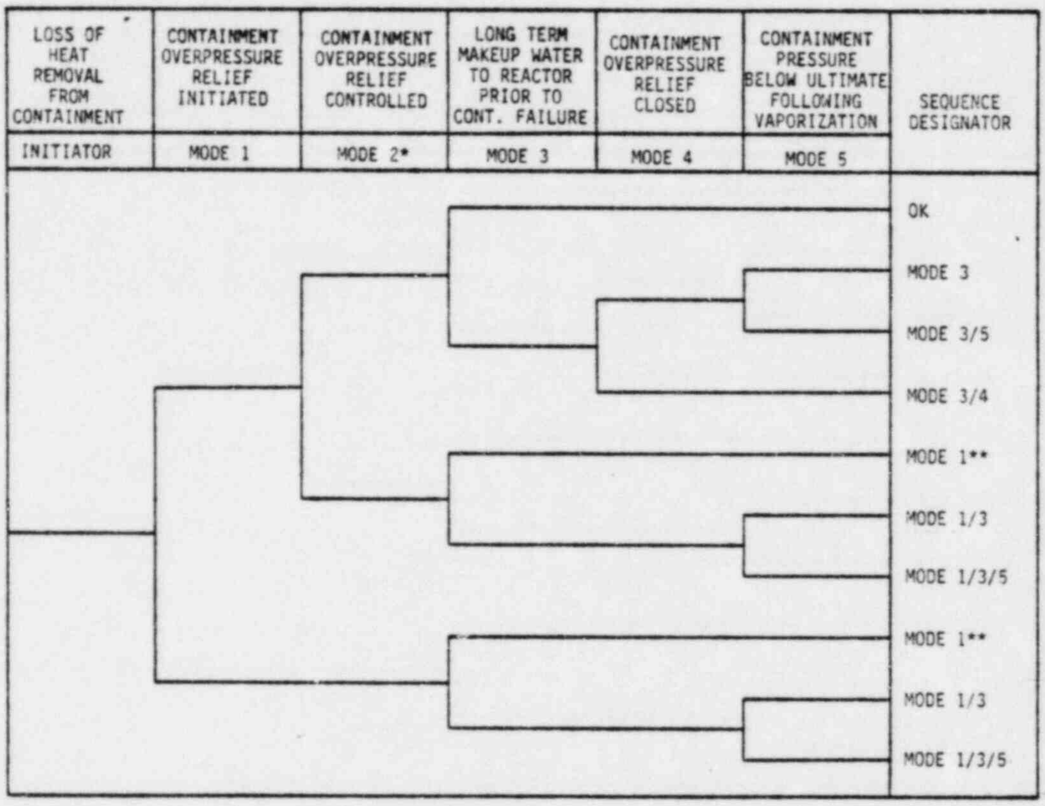


Figure 3.5.4 Comparison of the Contributing Accident Sequence to the Calculated Frequency of Core Melt from WASH-1400 and the Limerick Analysis (Area of "Pie Chart" is proportional to Mean Frequency)

*Worse conditions exist only 5% of the time
**Conservative by approximately a factor of three.

The bridge event tree (see Figure 3.5.5) is provided to connect the Class II and IV accident sequence event trees of Sections 3.4.1, 3.4.2, and 3.4.3 to the containment event tree of Section 3.4.5.



* Mode 2 is equivalent to Mode 1 in its impact on the containment.
** The assumption used in the LGS Risk Analysis is that containment failure leads to loss of long term coolant injection with a probability of one.

Figure 3.5.5 Bridge Event Tree, Characteristic of the Three Types of Events Discussed in Section 3.4. (same as Figure 3.4.13)

The quantification of the bridge tree requires the evaluation of the systems involved in each function for the conditions which exist during the demand on the containment and operator. In this analysis, four types of demands are investigated: (1) Anticipated transients with scram but a

Table 3.5.7

COMPARISON OF QUANTIFIED DOMINANT SEQUENCES: LIMERICK ANALYSIS+ VS. WASH-1400

$T_E QUV$ — LOSS OF OFFSITE POWER

Source	Initiation (Per Year)	Coolant Injection UV*	Total (Probability per Reactor Year)
WASH-1400	$4x10^{-2}$	$2x10^{-5}$	$8x10^{-7}$
Limerick Analysis	$8x10^{-3}$	$6x10^{-4}$	$4.8x10^{-6}$

*For loss of offsite power, main feedwater (Q) is unavailable and coolant injection unreliability is dominated by the common link to the emergency power buses (i.e., diesel reliability).

$T_F QUV$ — LOSS OF INVENTORY MAKEUP FOLLOWING A TRANSIENT: LOSS OF FEEDWATER

Source	Initiation (Per Year)	FW Q	High Pres. Injection U	Low Pres. Injection V	Total (Probability per Reactor Year)
WASH-1400	10	.01	$2x10^{-3}$	$2x10^{-3}$	$4x10^{-7}$
			$(3x10^{-7})$*		$(3x10^{-6})$*
Limerick Analysis	1.78	.22	$3.4x10^{-3}$	$2.1x10^{-3}$	$3x10^{-6}$

* From WASH-1400 Appendix I not located in summary tables of WASH-1400

ATWS — LOSS OF POISON INJECTION OR LOSS OF COOLANT INJECTION

Source	Initiation (per year)	Scram Failure		Total (per Demand)	ARI	Mitigation Systems	Total (Probability per reactor year)
		Mechanical (per Demand)	Electrical (per Demand)				
WASH-1400	10	--	--	$1x10^{-5}$	NA	.1	$1x10^{-5}$
NUREG-0460 Alternate 3A	6	$1.5x10^{-5}$	$1.5x10^{-5}$	$3x10^{-5}$	10^{-2}	.1	$9x10^{-6}$
Limerick Alternate 3A	3.5	$1.0x10^{-5}$	$2.0x10^{-5}$	$3x10^{-5}$	10^{-2}	SLC=.035 HPCI=.07	$2x10^{-6}$

TW — LOSS OF CONTAINMENT HEAT REMOVAL

Source	Initiator (per Year)	W (per Demand)	COR (per Demand)	Total (Probability per Reactor Year)
WASH-1400	10	$1x10^{-6}$	NA	$1x10^{-5}$
Limerick Analysis	7.2	$8x10^{-7}$	10^{-2}	$5.8x10^{-8}$

+ These summaries are approximate representations, only for the purpose of illustration, and do not reflect the precise values of the actual sequences analyzed in the Limerick analysis.

failure to remove heat from containment: these are referred to as TW-type sequences; (2) Cases involving a failure to scram along with a failure of containment heat removal, referred to as ATWS-W type sequences; (3) ATWS events for which there is a failure of the SLC coupled with continued injection of cooling water to the reactor, until containment fails, followed by a failure of all coolant injection, referred to as ATWS-C_2 type sequences; and (4) ATWS events for which one leg of the redundant SLC system fails, referred to as ATWS-C_{12} type sequences.

Table 3.5.8

SUMMARY OF THE CALCULATED REDUCTIONS IN THE FREQUENCY OF A RADIOACTIVE RELEASE DUE TO THE USE OF CONTAINMENT OVERPRESSURE RELIEF (REFLECTED IN THE BRIDGE TREE)

TYPE OF SEQUENCE	FAILURE OF CONTAINMENT OVERPRESSURE RELIEF	FAILURE OF MAKEUP WATER TO REACTOR MODE 3		FAILURE OF CONTAINMENT OVERPRESSURE RELIEF TO CLOSE	CONTAINMENT PRESSURE BELOW ULTIMATE FOLLOWING VAPORIZATION
	MODE 1, 2	MODE 3	MODE 1/3**	MODE 4+	MODE 5
Loss of Containment Heat Removal (TW) (Figure 3.4.13)	10^{-2}	$2x10^{-4}$	$1x10^{-3}$	$5x10^{-2}$	10^{-2}
Failure to Scram w/Loss of RHR (ATWS-W) (Figure 3.4.13)	2×10^{-2}	.28*	.8	$5x10^{-2}$	10^{-2}
Failure to Scram w/Loss of SLC (ATWS-C_2) (Figure 3.4. 13)	.9999	.28*	.8	$5x10^{-2}$	10^{-1}
Failure to Scram w/Loss of 1 SLC and 1 or 2 RHR (ATWS-C/RHR) (Figure 3.4.13)	10^{-1}	.28*	.8	$5x10^{-2}$	10^{-2}

* Loss of coolant make up probability is the combination of the following:
   a. Evaluated HPCI failure probability
   b. Increased likelihood of exceeding the pressure trip setpoint of 50 psig (Actual) due to exceeding the containment pressure design point
   c. Increased likelihood of the setpoint drifting low

** Mode 1/3 is the conditional probability of mode 3 occurring given that mode 1 (or mode 2) has occurred.

\+ Conditional failure probability of COR not reclosing given that coolant makeup to the core has failed.

COR Failure: The failure to maintain containment pressure below its pressure capability, using only COR, has a different probability, depending on the type of accident sequence (see Appendix B.4 for description of COR). The success of the containment overpressure relief (COR) function (Mode 1 or 2) is inversely related to the probability of high radiation in containment following an accident initiator. Following an ATWS initiator, there is a higher probability of some radiation being released to containment, while little if any is expected to be released during a TW transient.

- TW sequences: A low failure probability is calculated since the accident sequence is relatively slow in occurring. There is sufficient time for a well thought out operator response; and the probability of potential accident conditions complicating or defeating COR is minimal.
- ATWS-W sequences: A significantly higher failure probability is assigned to COR for these sequences, due to the higher probability of some radioactivity (fuel/clad gap primarily) reaching the containment drywell during the sequence.
- ATWS-C_2 sequences: Very little credit is assigned to COR for these sequences because of the estimated high probability of obtaining some radioactive releases to the containment and the relatively low capacity of the COR system.

Coolant Inventory Makeup: As with COR, the type of sequence can have a significant effect on the calculated probability used in the bridge tree for maintaining coolant injection over a long period of time.

- TW sequence: For LGS, most of the sources of coolant injection are available for use to maintain inventory (see fault tree in Appendix B). Therefore, the probability of loss of makeup is calculated to be relatively low.
- ATWS sequences: Since only HPCI is considered adequate for coolant inventory makeup during an ATWS condition with high internal containment pressure*, the failure probability for these sequences is simply the HPCI unreliability (taking into account the containment conditions which would exist during COR operation). During COR operation, it is estimated that the HPCI unre-

*Calculations indicate that RCIC alone is also adequate but was not evaluated in this analysis.

liability would double (failure probability increase by a factor of two). If COR is unsuccessful, an estimate of the HPCI unreliability is still important since it is a potential source of positive reactivity due to its high flow rate. HPCI is designed to trip on high turbine exhaust pressure, but since it is the only potentially successful high pressure makeup system available during the postulated ATWS with high containment pressure, the operator can bypass the HPCI trip circuits and restart the pump. Based on the above discussion, the availability of HPCI during an ATWS has been estimated to be higher than might be calculated using typical hardware failure and human error rates.

- Due to the environmental conditions which would be present when implementing COR, the HPCI reliability is anticipated to be lower during this period than under other accident conditions. For the purposes of this analysis, the unreliability of HPCI is estimated to be a factor of 2 higher during use of COR than under normal operation.

Containment Overpressure Relief Closed: For those cases in which COR is used to maintain containment pressure within its design limits, the long-term coolant makeup to the reactor may still fail despite the success of COR, and successful initial start and run of makeup systems. In such cases, core melt could occur with the COR valves open to the atmosphere. The failure to close these valves to ensure the containment integrity during the core melt and vaporization phases is assigned a relatively low probability.

Containment Pressure Below Ultimate Following Vaporization: Most accident scenarios involving the loss of coolant inventory are placed in Class I or III by virtue of the fact that an intact containment exists during the subsequent postulated core melt sequence. However, under those cases where coolant makeup failure occurs prior to containment failure, a spectrum of containment pressures may exist during core melt and vaporization. Likewise, the ultimate failure of the containment has some uncertainty associated with it. Considering these two uncertainties, accident sequences could exist for which the failure of coolant makeup may still lead to containment failure prior to core vaporization. The Limerick PRA assumes containment failure at 140 psig.

In the calculation of risk, the bridge tree has the effect of moving accident sequences from one class to another which thereby changes the calculated radioactive release fractions associated with the sequence. A set of tables has been prepared to clearly identify the contribution of the bridge tree to each class by accident sequence. Tables 3.5.9, 3.5.10, and 3.5.11 provide the summary of all the dominant sequences which are processed through the bridge tree. Included in the tables are the following:

- Input - the sequence frequencies
- Processing information - the reduction in these frequencies associated with each branch of the bridge tree
- Output - the frequency of potential degraded core for each class.

Table 3.5.9 summarizes the frequency of potential degraded core conditions associated with each class for accident sequence-types of the variety involving loss of containment heat removal (TW). Processing each of the TW-type sequences through the bridge tree results in the calculation of the frequency in each class of accident, depending upon the effect of the sequence on containment. The highest frequencies are calculated for the contributions to Class II, which are events involving degraded core conditions following a failure of containment. The contributions to the other classes are small by comparison to the other sequences in these classes.

Table 3.5.10 summarizes the contributions of ATWS-W sequences to each class as a result of the processing through the bridge tree. There are two important results from this processing:

1. There is a substantial contribution to the Class III types of sequences
2. The Class IV contribution appears small, however, the high radionuclide release fractions make this an important contributor to risk.

Table 3.5.11 summarizes the frequency of potential degraded core conditions associated with ATWS events which are compounded by the inability to inject liquid poison into the reactor.

Table 3.5.9

EXAMPLE SUMMARY OF TW* TYPE EVENT SEQUENCES WHICH ARE PROCESSED THROUGH THE BRIDGE TREE, FIGURE 3.4.13, TO DETERMINE THEIR SEQUENCE CLASSIFICATION

SEQUENCE TYPE	ACCIDENT SEQUENCE	ACCIDENT+++ SEQUENCE FREQUENCY (PER REACTOR YEAR)	REDUCTION++ THRU BRIDGE TREE	FREQUENCY (PER REACTOR YEAR) CONTRIBUTION TO EACH CLASS			
				CLASS I	CLASS II	CLASS III	CLASS IV
TW	T_TPW	$6.6x10^{-6}$	MODE 1† (2×10^{-2})	—	$1.4x10^{-7}$	—	—
			MODE 1/3 (2×10^{-5})	—	negligible	—	—
			MODE 3 $(1.9x10^{-4})$	$1.3x10^{-9}$	—	—	—
			MODE 3/4 (1×10^{-5})**	—	—	—	$6.6x10^{-11}$
	T_MW	$1.8x10^{-6}$	MODE 1 (2×10^{-2})	—	$3.6x10^{-8}$	—	—
			MODE 1/3 (4×10^{-6})	—	negligible	—	—
			MODE 3 $(1.9x10^{-4})$	$3.4x10^{-10}$	—	—	—
			MODE 3/4 (1×10^{-5})	—	—	—	$1.8x10^{-11}$
	T_FPW T_FQW	$1.5x10^{-5}$	MODE 1 (2×10^{-2})	—	$3.0x10^{-7}$	—	—
			MODE 1/3 (2×10^{-5})	—	negligible	—	—
			MODE 3 $(1.9x10^{-4})$	$2.9x10^{-9}$	—	—	—
			MODE 3/4 (1×10^{-5})	—	—	—	$1.5x10^{-10}$
	T_EPW_d	$6.5x10^{-7}$	MODE 1 (2×10^{-2})	—	$1.3x10^{-8}$	—	—
			MODE 2 (2×10^{-5})	—	negligible	—	—
			MODE 3 $(1.9x10^{-4})$	$1.2x10^{-10}$	—	—	—
			MODE 4 (1×10^{-5})	—	—	—	$6.5x10^{-12}$
	$T_IC'W$ T_IW	$8.3x10^{-6}$	MODE 1 (2×10^{-2})	—	$1.7x10^{-7}$	—	—
			MODE 2 (2×10^{-5})	—	negligible	—	—
			MODE 3 $(1.9x10^{-4})$	$1.6x10^{-9}$	—	—	—
			MODE 4 (1×10^{-5})	—	—	—	$8.3x10^{-11}$
TW	TOTAL			$6.3x10^{-9}$	$6.6x10^{-7}$	—	$3.2x10^{-10}$

* It must be noted that the large LOCA event tree contains a postulated accident sequence, AJ, which involves the large LOCA initiator coupled with the failure to remove heat from containment. For this particular ca[illegible] here is assumed to be sufficient radioactivity released to the containment atmosphere to caus[illegible] COR valves to be interlocked closed. The large and medium LOCA sequences then contribute dir[illegible]y to Class II and do not pass through the Bridge tree.

**Mode 5 effects are contribution to Class IV but are negligible relative to the mod 3/4 evaluation.

† Mode 1 includes the mode 2 (i.e. P--mode 1 + P--mode 2) failures since these have the same qualitative effect on containment and accident sequences.

†† The mode designators given in this Table are the sequence designators from Figure 3.5.5 and are formed by the product of the event proabability associated with the sequence designators. For example, mode 1/3. The probability of sequences designated mode 1/3 is calculated as the product of the probabilities of (mode 1)x(mode 1/3)x(mode 5) each of which is taken from Table 3.5.8; the number of such sequences is then multiplied times this product to determine the value down in the above table.

††† The accident sequence probabilities appearing in this example table are derived for sequences initiated above 25% power. The values appearing in Tables 3.5.5, 3.5.6, and 3.5.7 are the sum of Transients initiated from all powers.

Table 3.5.10

EXAMPLE SUMMARY TABLE OF ATWS-W TYPE EVENT SEQUENCES WHICH ARE PROCESSED THROUGH THE BRIDGE TREE, (FIGURE 3.4.13), TO DETERMINE THEIR SEQUENCE CLASSIFICATION

SEQUENCE TYPE	ACCIDENT SEQUENCE	ACCIDENT*** SEQUENCE PROBABILITY	REDUCTION** THRU BRIDGE TREE	FREQUENCY (PER REACTOR YEAR) CONTRIBUTION TO EACH CLASS			
				CLASS I	CLASS II	CLASS III	CLASS IV
ATWS-W (BOTH SLC PUMPS OPERATING)	$T_T{}^1C_MPW_2$	2.7×10^{-8}	MODE 1* (.04)	–	–	–	1.1×10^{-9}
			MODE 1/3 (.03)	–	–	8.1×10^{-10}	–
			MODE 3 (.27)	–	–	7.3×10^{-9}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	3.4×10^{-10}
	$T_F{}^2C_MW+$ $T_F{}^2(C_EK)W$	7.5×10^{-8}	MODE 1 (.04)	–	–	–	3.0×10^{-9}
			MODE 1/3 (.03)	–	–	2.2×10^{-9}	–
			MODE 3 (.27)	–	–	2.0×10^{-8}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	9.5×10^{-10}
	$T_E{}^3C_MW+$ $T_E{}^3(C_EK)W$	2.5×10^{-8}	MODE 1 (.04)	–	–	–	1.0×10^{-9}
			MODE 1/3 (.03)	–	–	7.5×10^{-10}	–
			MODE 3 (.27)	–	–	2.0×10^{-10}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	3.3×10^{-10}
	$T_I{}^4C_MW+$ $T_E(C_EK)W$	7.1×10^{-9}	MODE 1 (.04)	–	–	–	2.8×10^{-10}
			MODE 1/3 (.03)	–	–	2.1×10^{-10}	–
			MODE 3 (.27)	–	–	5.7×10^{-11}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	9.0×10^{-11}
	T_IC''	4.2×10^{-8}	MODE 1 (.04)	–	–	–	1.7×10^{-9}
			MODE 1/3 (.03)	–	–	1.3×10^{-9}	–
			MODE 3 (.27)	–	–	1.1×10^{-8}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	5.5×10^{-10}
ATWS-W	TOTAL	N/A	N/A	–	–	4.4×10^{-8}	9.3×10^{-9}

* Mode 1 includes the mode 2 (i.e. P--mode 1 + P--mode 2) failures since these have the same qualitative effect on containment and accident sequences.

** The mode designators given in this Table are the sequence designators from Figure 3.5.5 and are formed by the product of the event proababilites associated with the sequence designators. For example, mode 1/3. The probability of sequences designated mode 1/3 is calculated as the product of the probabilities of (mode 1)x(mode 1/3)x(mode 5) each of which is taken from Table 3.5.8; the number of such sequences is then multiplied times this product to determine the value down in the above table.

*** The accident sequence probabilities appearing in this example table are derived for sequences initiated above 25% power. The values appearing in Tables 3.5.5, 3.5.6, and 3.5.7 are the sum of Transients initiated from all powers.

Table 3.5.11

EXAMPLE SUMMARY TABLE OF ATWS-C_2[+] EVENT SEQUENCES WHICH ARE PROCESSED THROUGH THE BRIDGE TREE (FIGURE 3.4.13), TO DETERMINE THEIR SEQUENCE CLASSIFICATION

SEQUENCE TYPE	ACCIDENT SEQUENCE	ACCIDENT*** SEQUENCE PROBABILITY	REDUCTION** THRU BRIDGE TREE	FREQUENCY (PER REACTOR YEAR) CONTRIBUTION TO EACH CLASS			
				CLASS I	CLASS II	CLASS III	CLASS IV
ATWS-C_2 (ALL REACTIVITY SHUTDOWN MECHANISMS LOST)	$T_T{}^1C_MC_2$	1.1×10^{-7}	MODE 1* (2×10^{-1})	–	–	–	2.2×10^{-8}
			MODE 1/3 (7.2×10^{-1})	–	–	8×10^{-8}	–
			MODE 1/3/5 ($.8\times10^{-2}$)	–	–	–	9×10^{-9}
			MODE 3 (NEGLIGIBLE)	–	–	–	–
			MODE 3/4 (NEGLIGIBLE)	–	–	–	–
	$T_F{}^2C_MC_2$	3×10^{-8}	MODE 1 (2×10^{-1})	–	–	–	6×10^{-9}
			MODE 1/3 (7.2×10^{-1})	–	–	2.2×10^{-8}	–
			MODE 1/3/5 (8×10^{-2})	–	–	–	2.4×10^{-9}
	$T_E{}^3C_MC_2$	6×10^{-9}	MODE 1 (2×10^{-1})	–	–	–	1.2×10^{-9}
			MODE 1/3 (7.2×10^{-1})	–	–	4.3×10^{-9}	–
			MODE 1/3/5 (8×10^{-2})	–	–	–	4.8×10^{-10}
	$T_I{}^4C_MC_2$	2.8×10^{-9}	MODE 1 (2×10^{-1})	–	–	–	5.6×10^{-10}
			MODE 1/3 (7.2×10^{-1})	–	–	2×10^{-9}	–
			MODE 1/3/5 (8×10^{-2})	–	–	–	2.2×10^{-10}
ATWS-C	TOTAL	N/A	N/A	–	–	1×10^{-7}	4.2×10^{-8}

* Mode 1 includes the mode 2 (i.e. P--mode 1 + P--mode 2) failures since these have the same qualitative effect on containment and accident sequences.

** The mode designators given in this Table are the sequence designators from Figure 3.5.5 and are formed by the product of the event proababilites associated with the sequence designators. For example, mode 1/3. The probability of sequences designated mode 1/3 is calculated as the product of the probabilities of (mode 1)x(mode 1/3)x(mode 5) each of which is taken from Table 3.5.8; the number of such sequences is then multiplied times this product to determine the value down in the above table.

*** The accident sequence probabilities appearing in this example table are derived for sequences initiated above 25% power. The values appearing in Tables 3.5.5, 3.5.6, and 3.5.7 are the sum of Transients initiated from all powers.

[+] ATWS-C_2 Events are those ATWS events which include the failure of the SLC.

Table 3.5.12 completes the series of tables used to display the processing of accident sequences throught the bridge event tree. A separate table is developed for the ATWS-C_{12} sequences. The numerical values used in the quantification of the ATWS-C_{12} bridge event tree are the same as used in the ATWS-W sequences (see Table 3.5.8). As indicated by the total probability of the sequences for Class III and IV, this set of sequences makes a relatively small contribution to the overall frequency.

Table 3.5.12

EXAMPLE SUMMARY TABLE OF ATWS-C_{12}* EVENT SEQUENCES WHICH ARE PROCESSED THROUGH THE BRIDGE TREE, FIGURE 3.4.13 TO DETERMINE THEIR SEQUENCE CLASSIFICATION

SEQUENCE TYPE	ACCIDENT SEQUENCE	ACCIDENT SEQUENCE PROBABILITY†††	REDUCTION THRU BRIDGE TREE††	FREQUENCY (PER REACTOR YEAR) CONTRIBUTION TO EACH CLASS			
				CLASS I	CLASS II	CLASS III	CLASS IV
ATWS-W (1 SLC PUMP & 1 or 2 RHRs FAIL)	$T_T{}^1C_MC_{12}PW_2$	9.5×10^{-9}	MODE 1† (.04)	–	–	–	3.8×10^{-10}
			MODE 1/3 (.03)	–	–	2.8×10^{-10}	–
			MODE 3 (.27)	–	–	2.6×10^{-9}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	1.2×10^{-10}
	$T_F{}^2C_MC_{12}W_{12}$+ $T_F{}^2C_MC_{12}PW_2$	2.6×10^{-8}	MODE 1 (.04)	–	–	–	1×10^{-9}
			MODE 1/3 (.03)	–	–	7.8×10^{-10}	–
			MODE 3 (.27)	–	–	7.0×10^{-9}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	3.4×10^{-10}
	$T_T{}^3C_MC_{12}W_2$+ $T_E{}^3C_MC_{12}PW_{12}$	1.9×10^{-9}	MODE 1 (.04)	–	–	–	7.6×10^{-11}
			MODE 1/3 (.03)	–	–	5.7×10^{-11}	–
			MODE 3 (.27)	–	–	5.1×10^{-10}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	6.2×10^{-12}
	$T_1C_mC_{12}W_2$	2.5×10^{-9}	MODE 1 (.04)	–	–	–	1×10^{-10}
			MODE 1/3 (.03)	–	–	7.5×10^{-11}	–
			MODE 3 (.27)	–	–	6.7×10^{-10}	–
			MODE 3/4 (1.3×10^{-2})	–	–	–	3.3×10^{-11}
ATWS-W (1 SLC PUMP & 1 or 2 RHR FAIL)	TOTAL	N/A	N/A	–	–	1.2×10^{-8}	2.1×10^{-9}

* ATWS-C_{12} events are those ATWS events which include the failure of one SLC pump.

† Mode 1 includes the mode 2 (i.e. P--mode 1 + P--mode 2) failures since these have the same qualitative effect on containment and accident sequences.

†† The mode designators given in this Table are the sequence designators from Figure 3.5.5 and are formed by the product of the event proabability associated with the sequence designators. For example, mode 1/3. The probability of sequences designated mode 1/3 is calculated as the product of the probabilities of (mode 1)x(mode 1/3)x(mode 5) each of which is taken from Table 3.5.8; the number of such sequences is then multiplied times this product to determine the value down in the above table.

††† The accident sequence probabilities appearing in this example table are derived for sequences initiated above 25% power. The values appearing in Tables 3.5.5, 3.5.6, and 3.5.7 are the sum of Transients initiated from all powers.

3.5.4 Quantification of the Containment Event Tree

The two containment event trees which describe the possible paths of radioactive release from containment, and the numerical values used in the evaluation, are given in Figures 3.5.6a and b. The reason for two separate sets of numerical values for the containment event tree is that Class IV containment failures are assumed to be relatively rapid overpressures for which containment leakage before rupture is much less likely than for the relatively slow overpressure failures postulated for Class I, II and III. A discussion of probabilities used for each of the containment failures modes is provided below.

α -- Steam Explosion (In-Vessel). Full scale testing of the potential for coherent steam explosions when molten metal comes in intimate contact with water has not been performed. In an attempt to identify a probability for a coherent steam explosion inside the reactor vessel of sufficient energy to fail containment, the following evaluations were considered:

- Fauske Associates provided an analysis of the Limerick design to determine if the required conditions exist for a coherent steam explosion in the reactor vessel which would have sufficient energy to overpressurize containment. Their conclusion was that the coherent steam explosion appears to be impossible (see Appendix H).

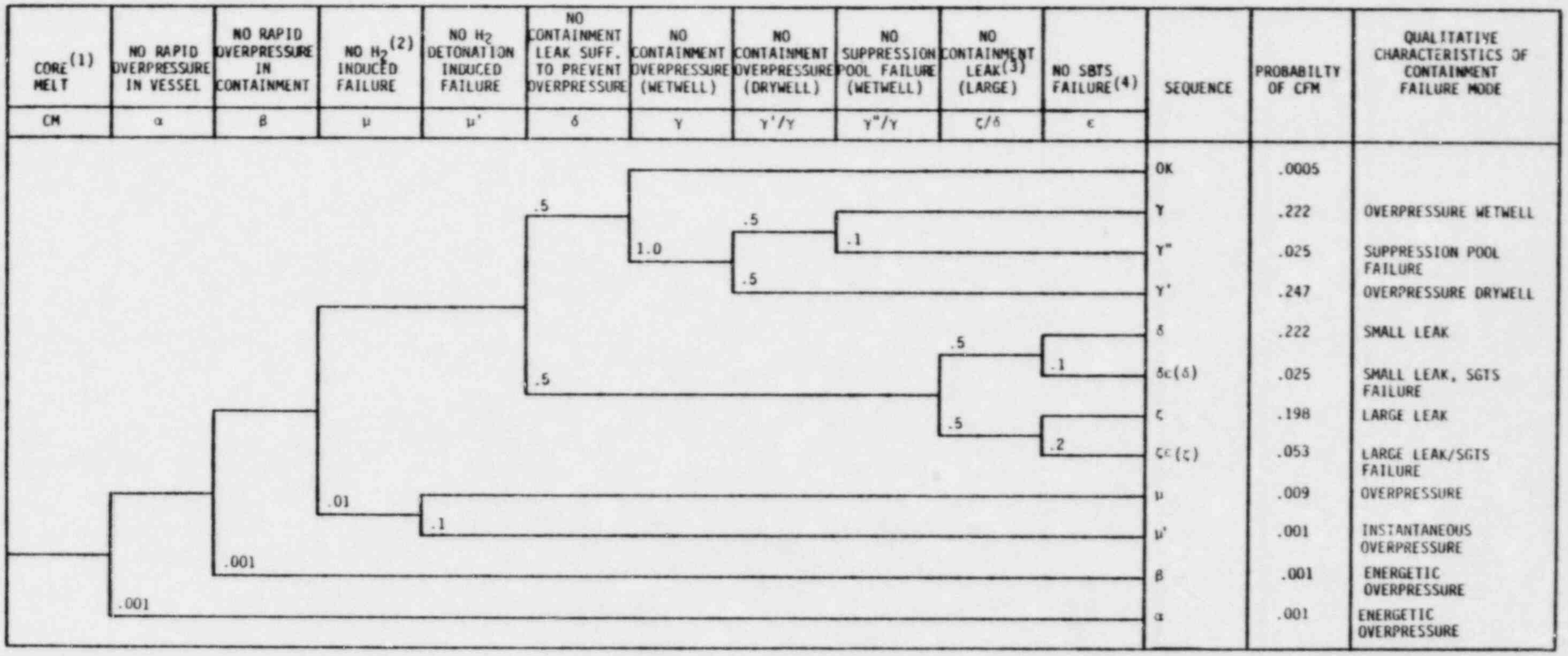
- Sandia Laboratories has performed analysis and small scale experiments with molten metal/water. Sandia has stated that steam explosions could occur in PWRs but probably of insufficient energy to overpressurize PWR containment. A similar statement was made for BWRs*. The WASH-1400 value (10^{-2}) with a reduction factor of 10 results in a value of 10^{-3} per demand, which was used in this analysis.

- The NRC**, in rebaselining the BWR, has used the following values to estimate the probability of an in-vessel steam explosion which overpressurize containment:

---

*Personal communication, Corradini (Sandia) to Burns and Parkinson (SAI).

**Personal communication between NRC (Taylor) and SAI (Burns).



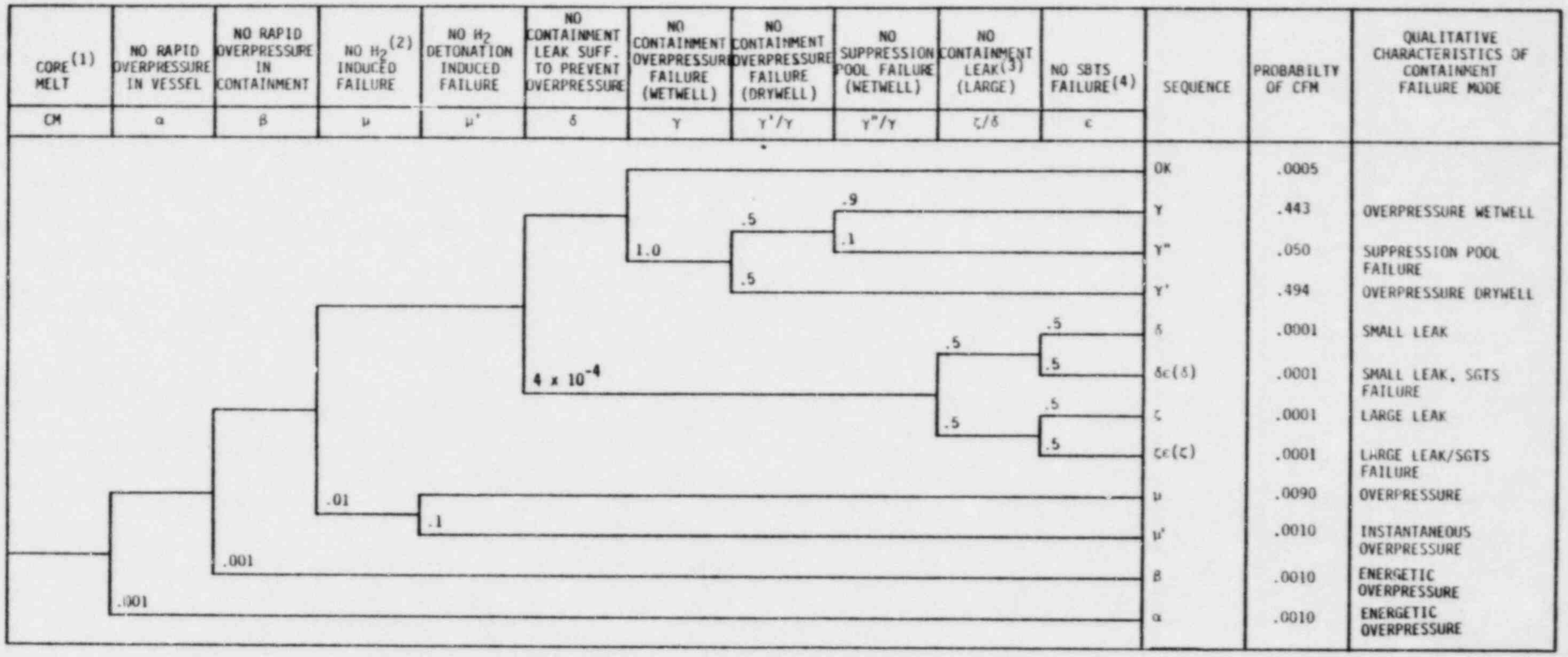
[1] Containment failure may have occurred prior to core melt. In those cases (Class II and Class IV), the containment failure modes are only used as mechanisms for release fraction determination.

[2] Assumes that H_2 explosion in containment causes overpressure failure with direct pathway to outside atmosphere.

[3] Leakage at 2400 volume percent/day.

[4] Failure standby gas treatment system.

Figure 3.5.6a Containment Event Tree for the Mark II Containment For Class I, II, and III Event Sequences.



(1) Containment failure may have occurred prior to core melt. In those cases (Class II and Class IV), the containment failure modes are only used as mechanisms for release fraction determination.

(2) Assumes that H_2 explosion in containment causes overpressure failure with direct pathway to outside atmosphere.

(3) Leakage at 2400 volume percent/day.

(4) Failure standby gas treatment system.

Figure 3.5.6b Containment Event Tree for the Mark II Containment for Class IV Event Sequences

- For LOCA events, a value of 10^{-2} was used, as in WASH-1400

- For non-LOCA events, a value of 10^{-4} was used since a steam explosion at high pressure is considered to have an extremely low probability.

The above evaulations were used to arrive at an estimate of α to be 10^{-3} per demand for a coherent in-vessel steam explosion which over-pressurizes containment (given a core melt).

<u>β -- Steam Explosion in Containment</u>. Containment steam explosions are less well understood than in-vessel steam explosions. However, they are generally considered to be low probability events. Fauske Associates included consideration of this event in their analysis (Appendix H).

<u>μ -- Hydrogen Burn or Explosion in Containment</u>. For the inerted Limerick containment, the possibility of a hydrogen detonation or burn appears quite remote; however, according to the tentative technical specification there may be short periods of time when the plant is operating at power and the containment is not fully inerted. This is anticipated to occur following reactor startups and prior to shutdowns. Based on past PECo experience and projected Limerick operating procedures, the probability of a hydrogen burn or detonation is considered to be 0.01. Relative to this 0.01 probability of not being inerted at power, if a core melt occurs during this time, then the probability of a burn or detonation sufficient to cause direct overpressure release, with a significant increase in the radioactive release fraction (i.e., comparable to a containment steam explosion) is no larger than 0.8*. This leads to a probability on the order of 10^{-3} for the μ' failure mode. However, the probability of some H_2 burn (μ) remains at ~0.01. This also leads to a drywell overpressure release and is included in the γ' containment failure mode.

*Any reduction of the hydrogen concentration by means of the hydrogen recombiners was not assumed due to the large amounts of hydrogen released during a core melt and the relatively small capacity of the recombiners.

δ -- Containment Leakage. Bechtel has performed a detailed containment analysis to define possible areas where containment may fail in the case of overpressure (see Appendix J). In addition, some effort was expended to identify potential areas where leakage before rupture would occur. Two items are noted:

1. Bechtel was unable to identify any specific areas where leakage would occur before rupture. Containment isolation valves are designed for much lower pressure, but have an expected capability much higher than design.
2. Containment leak rate testing has found that there is some degree of containment leakage at containment pressures below design pressure.

From the above considerations, it appears equally as likely for noticeable containment leakage to occur as not. Therefore, a value of 0.5 was used for this probability.

γ -- Containment Overpressure (No Leakage). Given that no containment leakage occurs, the possibility of containment overpressure without failure following a core melt is considered to be possible even though ultimate pressure is exceeded. Bechtel calculated the ultimate containment pressure capability to be 140 psig (approximately three times design pressure). For those core melt sequences where no leakage occurs, 140 psig is reached with a high probability (0.999) unless COR is initiated.

γ'/γ* -- Containment Overpressure (split between wetwell and drywell failure). Failure of containment due to overpressure has been divided into two types because of the potential difference in radioactive release terms. Failure in the drywell leads to direct release to the stack while a failure in the wetwell causes a release through the suppression pool. At present, evidence indicates failure at very high containment pressure may occur with equal likelihood in the wetwell or drywell. Therefore, $\gamma'/\gamma = .5$.

γ'' -- Wetwell Failure. The probability of a failure of containment which results in the loss of water in the suppression pool is evaluated based upon

*γ'/γ means γ' given γ.

the Bechtel analysis which indicates that the points of highest stress in the wetwell are near the nominal waterline in the suppression pool. It is assumed that the probability of a failure large enough* to drain the pool below the downcomers is approximately 10% of the probability that the failure will occur in the wetwell. Therefore, the probability of γ'' used in the Limerick analysis is 0.025.

ζ/δ -- Large Leak. If a leak in containment does occur prior to failure, then the question arises as to the size of the leak. ζ is the probability that the leak is greater than an equivalent 6" diameter hole in the drywell. This size hole is insufficient to fail the stack blowout panels, but does lead to overloading of the standby gas treatment system. The state of knowledge of the size of the postulated leaks is such that it leads us to estimate equal frequency of occurrence for both postulated leak sizes (ζ/δ = 0.5).

ε -- Standby Gas Treatment. The probability of standby gas treatment operating effectively in mitigating a radioactive release depends upon the size of the leak. For overpressure failures, the SGTS is assumed to be bypassed and the radioactive source escapes directly to the stack through the blowout panels. However, the SGTS is assumed effective to varying degrees for small and large leaks.

The containment failure modes developed in the Limerick probabilistic Risk Assessment use the same failure probabilities for each of the four Classes of accident types. While this is a simplification, the uncertainty in containment failure probability is much larger than the potential variability associated with the type of accident sequence.

*Either a failure below the elevation of the bottom of the downcomers or a containment wetwell failure which propagates to below the bottom elevation of the downcomers.

With the containment failure modes defined and quantified, the next step is to combine the dominant accident sequences under each failure mode. As noted previously, there ar_ four types of sequences considered for each containment failure mode.

3.5.5 Quantification of Accident Sequences by Containment Failure Mode

This section summarizes the information in the previous section and puts it into the format to be used in the ex-plant consequence calculation. It should be noted that WASH-1400 used five BWR release categories. Each category corresponded approximately to a containment failure mode, and all types of accident sequences were lumped together in these categories. For the Limerick analysis, there are seven distinct containment failure modes considered, and four classes of accident sequences. This leads to potentially twenty-eight separate ex-plant consequence calculations, compared with the five performed in WASH-1400.

Table 3.5.12 summarizes each of the containment failure moeds and provides, in capsule form, the information to be used in assessing the radioactive release fractions in Section 3.6, which in turn are input to the ex-plant consequence code, CRAC. In particular, in Table 3.5.12, the four (4) separate generic accident sequence classes, which are evaluated separately in terms of these containment failure modes, are cited.

Table 3.5.13 gives a summary of the probabilities associated with each containment failure mode leak path, and each of the accident classes. Table 3.5.14 provides the accident sequence probability which is input to the ex-plant consequence calculation. The radioactive source term for each of these sequences is calculated in Section 3.6.

Table 3.5.13

RELEASE TERM CALCULATIONS REQUIREMENTS[a]

CONTAINMENT FAILURE MODES		RADIOACTIVE RELEASE FRACTIONS			
Designator	Description	Class I (C1)	Class II (C2)	Class III (C3)	Class IV (C4)
α	Steam explosion in vessel	Note f	Note f	Note f	Note f
β	Steam explosion in containment	Note f	Note f	Note f	Note f
μ'	H_2 explosion induced containment failure	Note e	Note e	Note e	Note e
μ	H_2 deflagration sufficient to cause containment overpressure failure	Note b	Note b	Note b	Note g
δ	Overpressure small leaks (A_R=.05 ft^2)	X	X	X	Note h
γ'	Overpressure failure (A_R= 2.0 ft^2) Release through drywell	Note b	Note b	Note b	Note g
γ	Overpressure failure (A_R= 2.0 ft^2) Release through wetwell break	X	X	X	Note h
ζ	Overpressure, large leak (A_R = .2 ft^2)	X	X	X	Note h
$\zeta\varepsilon$	Overpressure, large leak, SGTS failure (A_R = .2 ft^2)	Note c	Note c	Note c	Note c
$\delta\varepsilon$	Overpressure, small leak, SGTS failure (A_R = .05 ft^2)	Note d	Note d	Note d	Note d
ε	Standby gas treatment system fails				

(a) An "X" under the heading indicates that a calculation of release fraction must be made for the particular accident involving a BWR/4 with a Mark II containment; all other cases can either be extrapolated from the set of calculations or can be extracted directly from WASH-1400.

(b) Can be extrapolated from γ release by assuming a different decontamination factor for room deposition. The principal difference between γ and γ' is that the γ release occurs with much of the release passing through the suppression pool. The γ' release occurs with much of the release occurring through the drywell.

(c) Can be extrapolated from equivalent ζ case by not using decontamination factor for SGTS (affects only portion of release flow)

(d) Can be extrapolated from equivalent δ case by not using decontamination factor for SGTS (affects all of release flow)

(e) Will be assumed to be equivalent to a β failure and same release fraction will be used.

(f) Release fractions will be extracted directly from WASH-1400 since the phenomenological nature of the accident does not change

(g) Release fractions similar to those developed by the NRC using March-Corral are used in the characterization of Class IV radioactive release fractions for γ'.

(h) Extrapolated from the Class I, II, III results.

Table 3.5.14

SUMMARY -- GENERIC ACCIDENT SEQUENCE/RELEASE PATH COMBINATIONS

CONTAINMENT FAILURE MODE \ CLASS	CLASS I	CLASS II	CLASS III	CLASS IV
α	1.3×10^{-8}	5.8×10^{-10}	1.4×10^{-9}	1.3×10^{-10}
β,μ'	2.6×10^{-8}	1.2×10^{-10}	2.8×10^{-9}	2.7×10^{-10}
γ',μ	3.4×10^{-6}	$1.5\ 10^{-7}$	3.6×10^{-7}	6.7×10^{-8}
γ	2.9×10^{-6}	1.3×10^{-7}	3.1×10^{-7}	5.9×10^{-8}
γ''	3.2×10^{-7}	1.5×10^{-8}	3.5×10^{-8}	6.7×10^{-9}
$\zeta\varepsilon,\delta\varepsilon$	3.0×10^{-6}	5.0×10^{-8}	1.1×10^{-7}	2.7×10^{-11}
ζ,δ	3.4×10^{-6}	1.9×10^{-7}	5.9×10^{-7}	2.7×10^{-11}
TOTAL PROBABILITY BY CLASS	1.3×10^{-5}	5.8×10^{-7}	1.4×10^{-6}	1.3×10^{-7}

Figure 3.5.7 indicates that the highest probability scenarios are those involving a coupling of core melt accident sequences with postulated containment overpressure failures. The in-vessel steam explosion and containment steam explosion scenarios both have significantly lower probability than the others. However, the consequences for these scenarios tend to be larger than for overpressure failures. The postulated leaks are of relatively high probability, but they have smaller consequences than the containment overpressure failures.

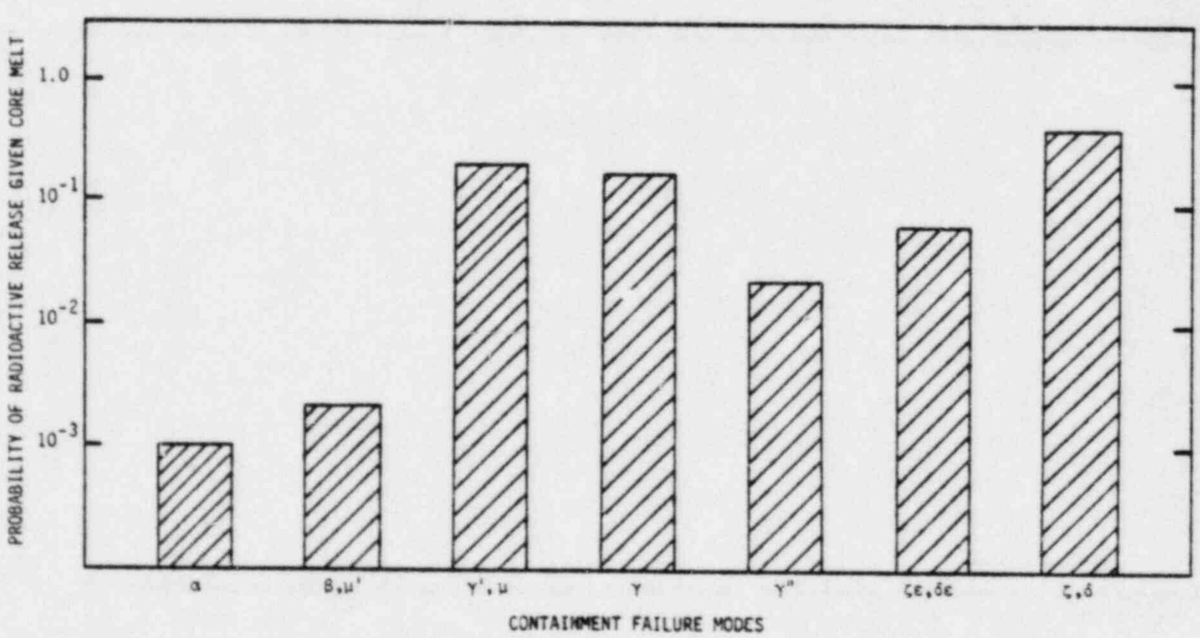


Figure 3.5.7 Probability of a Radioactive Release Given a Severe Degradation of Core Integrity -- Presented by Containment Failure Mode for All Classes.

3.6 RADIOACTIVE RELEASE FRACTIONS ASSOCIATED WITH DOMINANT ACCIDENT SEQUENCES

This section describes the radionuclide release fractions for the dominant accident sequences as used in the Limerick analysis. The release fractions of the key radionuclide isotopes are a portion of the input to the CRAC code (see Appendix E and Section 3.7).

The radionuclide release fractions are determined for each of the Mark II containment failure modes from the coupled calculations of INCOR and CORRAL and from assumptions considered in WASH-1400. INCOR (see Appendix C) calculates the thermodynamic conditions in the reactor system and inside containment plus the leak rates between containment compartments during postulated core melt scenarios. CORRAL (see Appendix D) takes these results and calculates the fission product removal rates as a function of time to determine the fission product concentration in each compartment. The final results from CORRAL are the cumulative radionuclide releases from containment to the atmosphere for each of the fission product species.

Included in this section are the following brief summaries of analyses for calculation of CRAC input:

- General radionuclide release discussion (Section 3.6.1)
- Summary of containment conditions (Section 3.6.2)
- Summary of radionuclide release fractions by failure mode (Section 3.6.3).

3.6.1 General Radionuclide Release Discussion

The amount of radioactivity released after an accident sequence is calculated by using the CORRAL computer code*. The boundary conditions for CORRAL are set by INCOR. CORRAL is used to trace the movement of radionuclides from their sources, through various nuclide removal steps, and ultimately to their release into the environment. The release fractions

*SAI-REACT was also used to verify the CORRAL results.

of the various radioactive isotope groups are then input into the CRAC program to calculate the offsite effects (see Section 3.7 and Appendix E).

Upon initiation of core melt, radionuclides will be released by all the potential physical mechanisms, but for the purposes of modeling and discussion it is useful to talk of four separate time phases of release. The four major radionuclide release phases considered in the CORRAL model are:

Gap: The nuclides are released as a result of the fuel rods breaking. This is the first release to occur in the accident. The radionuclides are passed to the containment via the safety relief valves or a reactor system leak or rupture.

Melt: This release occurs after the core has been uncovered and it begins to melt. Fission products are then released for one to two hours. At 80% core melt, the core is assumed to slump to the bottom of the vessel and begin to attack the lower head.

Vaporization: This release occurs after the RPV fails in the bottom head due to the attack by the molten core. The core remnants then fall to the diaphragm floor and interact with the concrete releasing nuclides to the drywell atmosphere. Therelease continues for several hours and decreases exponentially with time.

Oxidation: Particulate nuclides are released into the wetwell vapor region from molten core falling through the downcomers into the suppression pool and causing small scale steam explosions. This release is almost instantaneous.

The radionuclides emitted from the above releases are divided into seven species and further classified into one of three types because of their chemical properties. The seven species of nuclides, and their appropriate classifications, that are considered in the Limerick PRA analysis are chosen to parallel those chosen in WASH-1400 and are the following:

SPECIES	TYPE
Noble Gases	gas
Iodine (elemental, organic)	vapor
Cesium-Rubidium	particulate
Tellurium	particulate
Barium-Strontium	particulate
Ruthenium	particulate
Lanthanum	particulate

In the Limerick analysis, the radionuclide release fractions to the atmosphere for each postulated containment failure mode which are inputs to CRAC, are obtained in two steps:

1. The total release fractions to the containment (the fractional amount of each of the separate radionuclides that can be released to containment) are necessary. This is discussed below and is based solely on the WASH-1400 evaluation.

2. Each of the radionuclide groups are subjected to different times of release during the sequence, different processing as a function of the accident scenario, and different holdup times inside containment. Those features which determine the fraction finally released to the atmosphere are discussed in Section 3.6.3.

In determining the total fraction of each isotope released to containment, the basic research which was applied in the WASH-1400 analysis is also applied in the Limerick quantification of consequences. Table 3.6.1 summarizes the core fraction by radionuclide species which are released during each of the release phases. The iodine and tellurium releases are important in determining the sensitivity of the early fatality CCDFs. During the meltdown release (the core is still inside the reactor vessel) a substantial por-

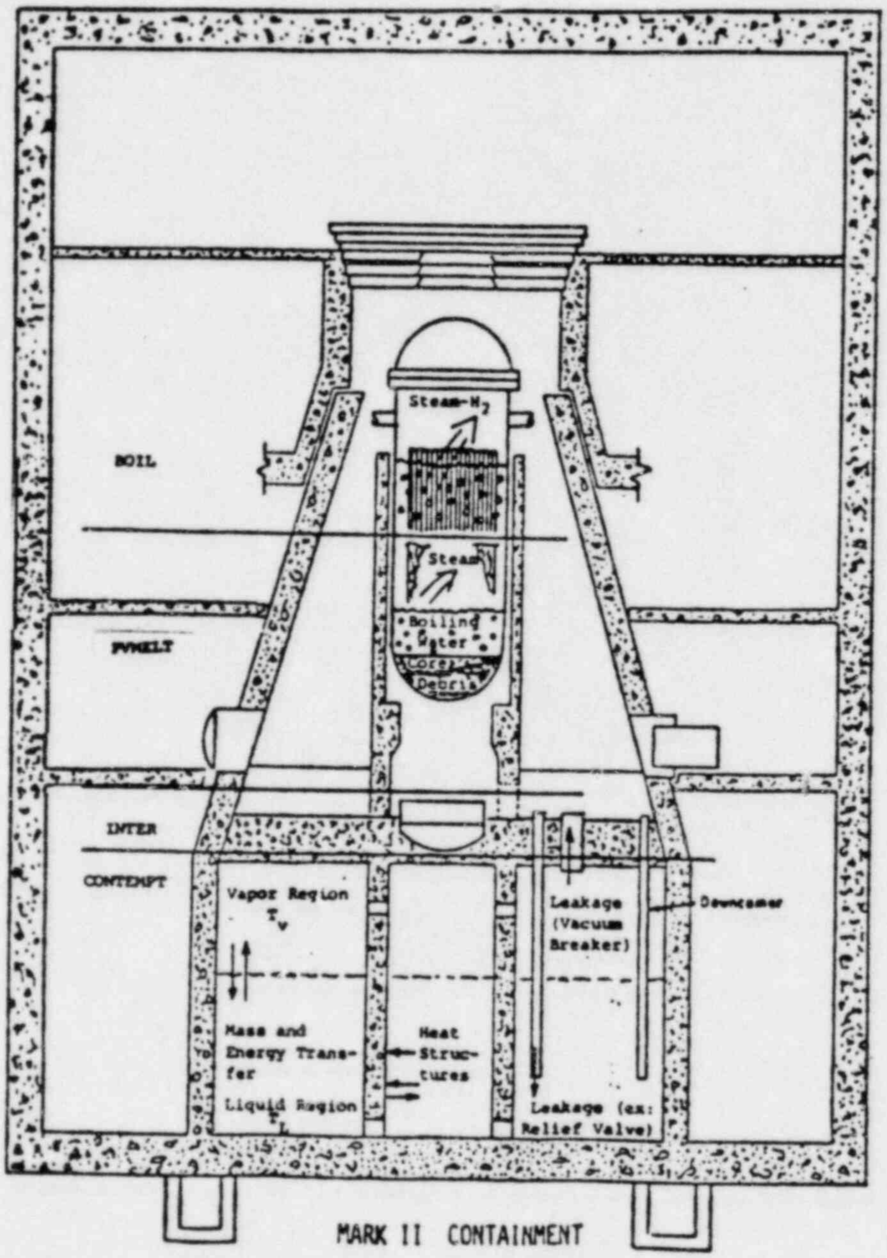


Figure 3.6.1 Schematic of the Limerick Containment

TABLE 3.6.1

FISSION PRODUCE RELEASE SOURCE SUMMARY-BEST ESTIMATE TOTAL CORE RELEASE FRACTIONS

Fission Product	Gap Release Fraction	Meltdown Release Fraction	Vaporization Release Fraction[d]
Xe, Kr	0.030	0.870	0.100
I, Br	0.017	0.883	0.100
Cs, Rb	0.050	0.760	0.190
Te[a]	0.0001	0.150	0.850
Sr, Ba	0.000001	0.100	0.010
Ru[b]	--	0.030	0.050
La[c]	--	0.003	0.010

(a) Includes Se, Sb

(b) Includes Mo, Pd, Rh, Tc

(c) Includes Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

(d) Exponential loss over 2 hours with halftime of 30 minutes. If a steam explosion occurs prior to this, only the core fraction not involved in the steam explosion can experience vaporization.

tion of the iodine is released. This results in processing the iodine through the suppression pool for non-LOCA sequences in which the pool is intact. The other key element to note is the tellurium which is released principally during the vaporization phase. The oxidation release which might occur during some sequences has a much larger release fraction associated with it. It is not shown in Table 3.6.1 but is the same as was used in WASH-1400. Reduction of the release fraction before exiting containment is discussed in Section 3.6.3; however, since the attenuation of the radionuclide releases is strongly sequence dependent, the containment conditions and accident sequence timing are important parameters which must be included in the analysis. Section 3.6.2 is used to summarize boundary conditions which effect the various methods of radionuclide removal and includes:

Active Safety System: The Standby Gas Treatment System (SGTS) is used to filter the reactor building air should the primary containment fail. This method is effective as long as there is no large leak due to reactor building overpressure.

Passive Safety System: The wetwell pool is a major removal method for radioactivity during an accident. The effectiveness of pool decontamination depends on the conditions of the water and requires that the radioactive material pass through the pool (this is calculated using INCOR).

Natural Removal: Radioactivity may be removed by natural deposition (plateout) or settling.

3.6.2 Summary of Containment Conditions Following a Core Melt Accident Sequence

The thermal hydraulic interaction of the molten core with the containment is calculated using the INCOR code package (see Appendix C). Figure 3.6.1 presents a schematic of the Limerick reactor vessel and containment, and identifies which portions of the INCOR code are used to calculate the thermodynamic conditions inside containment during each phase of the postulated accident sequence. The INCOR package includes:

BOIL: Core melting

PVMELT: Molten core interaction with the reactor vessel bottom head

INTER: Molten core and diaphragm floor interaction

CONTEMPT-LT: The flow rates, temperatures, and pressures in each compartment determined using the input from BOIL, PVMELT, and INTER.

Depending upon the core melt scenario, the containment can be in a variety of states during and after an accident sequence. Tables 3.6.2 and 3.6.3 list the assumptions determined from the INCOR analysis used to set the containment conditions for the selected sequences which were analyzed in detail using CORRAL. These conditions are used to characterize the failure modes in each accident sequence. Table 3.6.3 summarizes the key event sequence times used in the containment evaluation. The time of the radionuclide releases with respect to the time of containment failure may greatly influence the release fractions. Containment failure after the various releases (gap, melt, oxidation, vaporization) results in the radionuclides from each release accumulating in containment and then released to the atmosphere. If containment failure occurs prior to any of the radionuclide releases, without the benefit of longer accumulation in containment where natural deposition or plateout has a larger influence, the radioactivity may be released directly to the atmosphere.

The containment conditions which have the most effect on the radionuclide releases are described below for each of the accident sequence classes:

1. Class I: Cases involving a loss of cooling to the core are evaluated to result in a core melt with an intact containment. The containment conditions are such that the containment pressure just prior to meltthrough of the reactor vessel is slightly higher than atmospheric and the suppression pool is subcooled. Therefore, the radionuclide release fractions are calculated for containment conditions for which the containment is intact through a large portion of the core vaporization phase.

Table 3.6.2 SUMMARY OF CONTAINMENT CONDITIONS FOR THE DOMINANT ACCIDENT SEQUENCES

CLASS	TYPICAL SEQUENCE	PRINCIPAL ELEMENTS OF CONTAINMENT ANALYSIS					
		INITIAL PROBLEM	CORE POWER AT CORE UNCOVERY	CONTAINMENT PRESSURE AT MELT INITIATION	CONTAINMENT INTACT DURING VAPORATION	POOL TEMPERATURE	RELEASE FRACTIONS
1	T [illegible]	Loss of coolant inventory	≤2%	17 PSI	Yes	Subcooled	SAI
2	[illegible]	Containment pressure increase	≤.1%	140PSI-Atmospheric	No	Saturation	SAI
3	ATWS-C_z	Loss of coolant inventory	30%	25-65 PSI	Yes	Saturation	SAI
4	ATWS-$C_z\overline{U}$	Containment	30%	140PSI-Atmospheric	No	Saturation	NRC/Battll

TABLE 3.6.3

SUMMARY OF CONTAINMENT EVENTS DEVELOPED FROM THE INCOR ANALYSIS FOR THE RADIONUCLIDE RELEASE FRACTION CALCULATIONS

CLASS	CONTAINMENT FAILURE TIME CALCULATED BY INCOR	DIAPHRAGM FLOOR FAILURE TIME CALCULATED BY INCOR	CONTAINMENT FAILURE TIME USED IN ANALYSIS*
TQUV (C1)	6 hrs (small radius) 6.5hrs(large radius)	6 hrs (small radius) 6.5hrs(large radius)	6.5 hrs
TW (C2)	30 hrs	43.3 hrs	Failure prior to core melt
ATWS (C3)	6 hrs (small radius) 6.5hrs(large radius)	6 hrs (small radius) 6.5hrs(small radius)	6.5 hrs
ATWS (C4)	40 mn	6.5-7 hrs	Failure prior to core melt

*INCOR analyzed two cases for the Class 1 and Class 3 sequences. Small radius class denotes the molten core staying inside the pedestal region while the large radius indicates the molten core flows through the doorway and covers the entire diaphragm floor. However, for the release fraction calculations only the large radius case is analyzed.

2. Class II: These cases are different than the Class I events since the containment is considered to be failed prior to core melt due to failure to remove heat from containment. The containment conditions include a saturated suppression pool. The scenario involves:

   - The gap release and the melt release occurring through the safety relief valves to the saturated suppression pool.

   - The vaporization release occurring in the drywell with an open containment

3. Class III: This case is very similar to the Class I sequence of events. The major difference is that the suppression pool is saturated during the gap and melt radionuclide releases. Therefore, the decontamination factor for these releases is less than determined for Class I.

4. Class IV: This case parallels Class II except that the power level is significantly higher prior to loss of coolant inventory.

The effect of the suppression pool, as discussed above, is reflected in the Limerick CORRAL release fraction calculations by using a decontamination factor associated with the pool which varies with containment conditions. Table 3.6.4 lists the decontamination factors based on a survey of the available data.

TABLE 3.6.4

SUMMARY OF THE DECONTAMINATION FACTORS

Conditions	Meltdown Release	Vessel Failure and Vaporization
Containment Failure at End of Release	100	10*
Containment Failure Initiates Release	10*	10*

*Suppression pool considered saturated

For each of the accident sequence classes there is a set of containment failure modes which will also affect the magnitude of the radionuclide releases. The principal ways the containment failure modes affect these releases are the following:

1. Size of Containment Breach: The size of postulated containment failures determine the usefulness of the reactor enclosure and the standby gas treatment system for providing additional decontamination.
2. Location of the Breach: The location of the postulated containment failure affects the degree of difficulty of the path for the radionuclide release. The most important aspect of the location is in relation to the suppression pool; that is, for some sequences which include drywell failure (i.e., γ) the radionuclide release during vaporization will bypass the suppression pool.

In summary, the different types of containment failures may have an effect upon the attenuation and filtering on the radionuclides; therefore, the different failure paths have different release fractions. This evaluation of containment effect on release fractions is unique to the Mark II Limerick containment analysis although it follows the same methodology and logic used in WASH-1400.

3.6.3 Radionuclide Release Fractions from the Limerick Mark II Containment as a Function of Accident Sequence and Containment Failure Mode

Radionuclide release fractions from the containment to the atmosphere (obtained from CORRAL) along with other pertinent data are input to the CRAC code in order to determine the offsite consequences associated with the radionuclide release. The release fractions that are used in the Limerick risk analysis and the release parameters for the major release modes are summarized in Table 3.6.5

TABLE 3.6.5

EXAMPLES OF RADIONUCLIDE RELEASE PARAMETERS AND RELEASE FRACTIONS FOR DOMINANT ACCIDENT SEQUENCE CLASSES AND CONTAINMENT FAILURE MODES

PATHWAY SEQUENCE	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVALUATION (Hr)	ELEVATION OF RELEASE (METERS)	CONTAINMENT ENERGY RELEASE (10^6 BTU/Hr)	RELEASE FRACTIONS						
						Xe[a]	$I^{(b)}_2+CH_3I$	Cs[c]	Te[d]	Sr[e]	Ru[f]	La[g]
α												
$(C_1+C_3)\alpha$	2.0	0.5	1.0	82	40	1.0	0.40	0.40	0.50	0.05	0.50	3.0×10^{-3}
$C_2\alpha$	39.0	0.5	8.0	82	40	1.0	0.096	0.10	0.40	0.01	0.40	2.0×10^{-3}
$C_4\alpha$	2.0	0.5	1.5	82	40	1.0	0.096	0.10	0.40	0.01	0.40	2.0×10^{-3}
β												
$(C_1+C_2+C_3+C_4)(\beta+\mu')$	4.0	0.5	3.0	82	36	1.0	0.15	0.06	0.52	0.007	0.40	1.0x10[illegible]
γ'												
$C_1\gamma'$	7.0	2.0	6.0	82	36	1.0	0.11	0.09	0.016	0.011	3.2×10^{-3}	3.2×10^{-4}
$C_2\gamma'$	-	-	-	-	-	1.0	0.06	0.023	0.4	6.3×10^{-3}	0.069	4.7×10^{-3}
$C_3\gamma'$	-	-	-	-	-	1.0	0.04	0.024	0.073	2.7×10^{-3}	8.6×10^{-3}	9.1×10^{-4}
$C_4\gamma'$	1.5	2.0	1.0	82	.3	1.0	0.26	0.201	0.472	0.029	0.084	5.0×10^{-3}
γ												
$C_4\gamma$	1.5	2.0	1.0	82	.3	1.0	0.07	0.09	0.20	0.015	0.085	5.0×10^{-3}
γ''												
$C_4\gamma''$	1.5	2.0	1.0	0	.3	1.0	0.73	0.70	0.55	0.09	0.12	7.0×10^{-3}
$\zeta\epsilon$												
$C_2\zeta\epsilon$						.73	0.019	9.8×10^{-3}	0.046	1.6×10^{-3}	3.2×10^{-3}	5.8×10^{-4}

3.6.3.1 In-Vessel Steam Explosion

The postulated in-vessel steam explosion is assumed to lead to an oxidation reaction involving a large fraction of the molten core releasing the remaining volatile fission products. Therefore, a potentially high radionuclide release from the open containment may occur.

The radionuclide release fractions for each of the four accident sequences for in-vessel steam explosion are basically the same. The major differences in the release fractions between the sequences occur due to the amount of core melt that is assumed prior to the postulated steam explosion and the conditions of the suppression pool. These differences cause higher release fractions for the Class I and Class III sequences than for Class II and Class IV. In Class I and III sequences, the core is assumed to drop to the bottom of the RPV. Up until this time, all the radionuclides released have been passed through the suppression pool and therefore attenuated before they are released into the containment atmosphere. However, when the core drops, there is a sudden oxidation release (in-vessel steam explosion) of the radionuclides remaining in the core which breaches first the RPV then the containment. In the Class II and Class IV sequences, the core is not assumed to drop to the bottom of the vessel until it has reached 100% melt. Therefore, a large fraction of the volatile radionuclides have already been released and filtered through the suppression pool before the in-vessel steam explosion occurs which releases the remaining nuclides to the containment.

3.6.3.2 β -- In-Containment Steam Explosion

In the unlikely event of a core melt, it is conceivable that this molten fuel may interact with the suppression pool in a coherent manner. A significant portion of core material could then be released to the environment due to the molten core contacting the water of the suppression pool in a confined area, thereby causing a steam explosion (or oxidation release). It is assumed that the releases due to this containment failure mode occur from a combination of sources:

1. Material released from the fuel/cladding gap or during core melt which has not been discharged through the SRVs to the suppression pool at the time of vessel meltthrough

2. Material released during the vaporization stage due to the interaction between the molten core and concrete diaphragm floor

3. Material previously dissolved or suspended in the suppression pool which is revaporized (with the steam) or resuspended as a result of the steam explosion in the pool

4. Material released from the fuel during the oxidation process as a result of the steam explosion.

It was found that the β release fractions did not vary much from one class to another. The gap releases will vary among the different classes; however, the source term associated with these radionuclides is considered small relative to the dominant source, the vaporization and oxidation release.

The radionuclides suspended in containment following the oxidation release are assumed to be the same for each accident sequence. Effects due to the status of the suppression pool are considered to be negligible for this release. Some radioactivity in the suppression pool is resuspended as a result of the postulated steam explosion.

3.6.3.3 μ' -- Hydrogen Explosion

Hydrogen explosion is considered to be a low probability event for the Limerick containment since it is usually inerted. However, according to tentative technical specifications there may be times when the plant is operating at power with the containment deinerted. Therefore, the possibility of hydrogen combustion is considered and the release fractions due to this type of failure are taken from WASH-1400.

The hydrogen combustion (μ') and containment steam explosion (β) are combined because of the similar manner in which they fail the containment and the assumption that they both have similar impacts on the radionuclide release fractions.

3.6.3.4 γ, γ', γ'' -- Relatively Slow Overpressure Failures During Postulated Core Melt Scenarios (Class I through IV)

The containment may fail due to a relatively slow pressure buildup due to core melt (assessed as the most likely type of failure). The various locations for such a failure are differentiated as follows:

γ' - Drywell Failure

γ - Wetwell Failure

γ'' - Wetwell Failure below the suppression pool waterline.

These locations were chosen based upon a structural analysis of the LGS containment (see Appendix J).

The release fractions associated with γ (wetwell) failures are nearly identical for all the classes of accident sequences used in the Limerick PRA quantification.

3.7 CONSEQUENCES ASSOCIATED WITH ACCIDENT SEQUENCES

This section summarizes the calculation of offsite effects for the following:

- The calculational model used in the Limerick site-specific analysis (CRAC)
- The input data used in the CRAC evaluation
- The results of the CRAC calculation.

3.7.1 Ex-Plant Consequence Model

CRAC (calculation of reactor accident consequences) is a computer code which was used in the Reactor Safety Study (WASH-1400) to assess the impact of reactor accidents on public risk. The CRAC evaluation in WASH-1400 was applied to specific sites but in the final assessment was applied to a composite site with population density derived in a manner to approximate an average site in the United States. This section focuses on the application of the CRAC model to the site-specific evaluation of the Limerick Generating Station. A discussion of the various aspects of the CRAC model are provided in Appendix E.

The basic CRAC model as used in WASH-1400 was also used in the LGS analysis. The effect on public risk is determined by the behavior of the radionuclide cloud, the health effects induced by the radionuclides, and the population response. Specific aspects of the LGS CRAC model and additional comments are noted below.

1. Impacts on the dispersion of radionuclides from the reactor site is governed by the following:

   - The length of release* was modified from that used in WASH-1400 based on subsequent·data to produce a more lateral diffusion estimate.

   - A plant-specific terrain roughness* factor is used in the model calculation of plume dispersion to account for turbulence-producing ground effects.

   - The height of the release is varied as a function of the accident sequence (see Section 3.7.2) and the release energy rate.

   - A seasonal windrose is used to determine the weighting of the consequences as they are affected by the wind direction.

   - The wind speed and precipitation are determined using meteorological data gathered by PECo for the LGS site.

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*Both items are consistent with current NRC site review methods. See Appendix E for further discussion of radionuclide dispersion.

- All calculations were done using five years of weather data with risk estimates averaged to provide a best estimate.

2. The effect on public risk is determined by the behavior of the radionuclide cloud and by the population response. The population response model used for the LGS evaluation incorporated the following:

   - Population shielding factors characteristic of the Pennsylvania area

   - Evacuation speeds and effected area the same as used in WASH-1400

3. The health effects model used in the evaluation of early fatality risk is a threshold model, which requires the exposed person to be subjected to a specific level of radiation before an early fatality is recorded.

3.7.2 The Input Data Used in the Limerick Site-Specific CRAC Model

The CRAC computer code acts as a bookkeeping program to combine data to calculate public risk using the models discussed in Section 3.7.1 and Appendix E. The principal inputs used in the analysis include:

1. The accident sequence probabilities determined from the event tree/fault tree evaluation (Section 3.5)

2. The radionuclide fraction released for each of these postulated accident sequences (Section 3.6)

3. The warning time for evacuation determined as a function of the accident sequence type (Section 3.6)

4. The height of release and the duration of release to determine the plume characteristics (Section 3.6)

5. The population distribution (see Appendix E) determined from the following:

   - Philadelphia Electric Company data up to 50 miles from the plant

   - Census data on Northeast population density from 50 to 500 miles

6. Meteorological data from the Limerick site compiled by PECo over the period 1972 to 1976. The CRAC best estimate risk curve is an average of best estimate curves calculated independently for each of the five years.

3.7.3 Results of the Limerick Specific Offsite Consequence Evaluation

The analysis of risk involves both the estimation of the probability and the calculation of consequences that may occur due to identified accident sequences. The consequence analysis for LGS was performed using the CRAC code, as was done in WASH-1400.

The form used to present the results of the LGS probabilistic risk assessment is identical to that used in WASH-1400: the complementary cumulative distribution function (CCDF). The CCDF is a plot of the probability or frequency of equalling or exceeding a given parameter versus the parameter in question. The parameter analyzed in this study is early or latent fatalities due to postulated nuclear reactor accidents.

The early and latent CCDFs for LGS are presented in Figures 3.7.1 and 3.7.2, respectively. For the early fatalities, a comparison is presented which indicates that the risk due to LGS is several orders of magnitude lower than that encountered by the general public from various non-voluntary activities (i.e., activities undertaken without a conscious decision).

The LGS CCDF is a best estimate curve calculated to allow comparison to other curves generated in a similar manner. To properly evaluate this curve and make meaningful comparisons to other curves of the same type, an understanding of the uncertainties associated with the CCDF is essential. Section 3.8 discusses the uncertainties involved in probabilistic risk assessment, and Section 4 compares LGS and the WASH-1400 BWR CCDFs

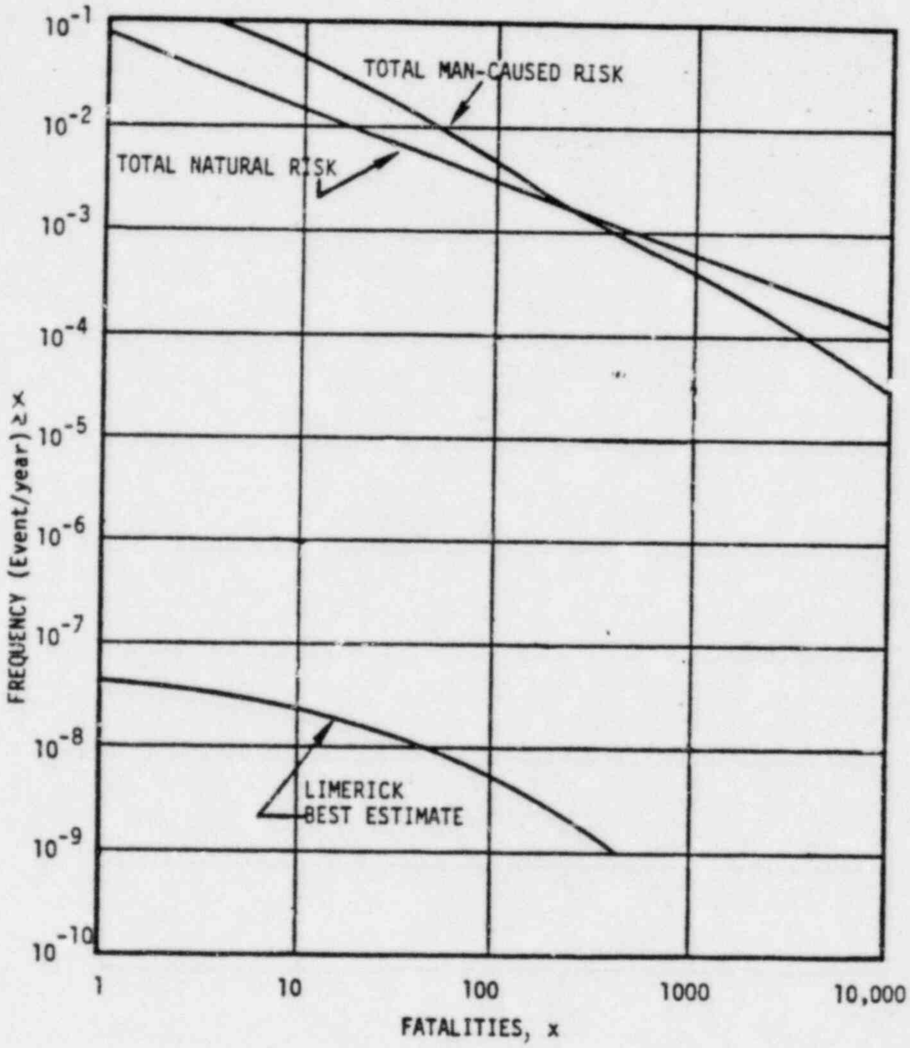

Figure 3.7.1 Summary of Risks Assumed by the Population Surrounding the LGS compared to the CCDF for Early Fatalities for Limerick.

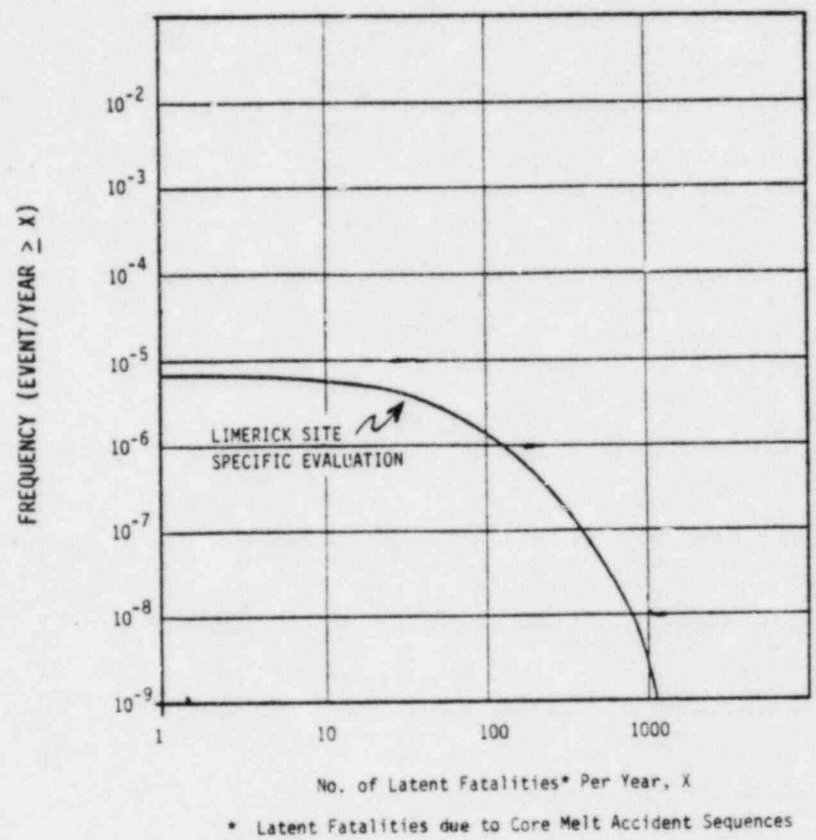

Figure 3.7.2 Latent Fatality CCDF for LGS

3.8 CHARACTERIZATION OF UNCERTAINTIES

The Limerick risk analysis has provided a best estimate of the cumulative complementary distribution functions for consequences representing early and latent fatalities. The calculated curves are determined from a broad range of factors including failure rate data, event sequence definition, physical process modeling, meteorological effects, and source term determination. Each of these factors has an associated uncertainty. This section presents a characterization and discussion of these uncertainties.

Based upon comments by the Lewis Committee, it is considered inappropriate to provide only a best estimate curve without a characterization of the uncertainties. Therefore, an evaluation of the principal items which may affect the uncertainty band in the Limerick PRA is provided. The uncertainty band was assessed based upon a combination of (a) quantification of uncertainties in the probabilities of accident sequences using Monte Carlo simulation of the LGS fault tree models for selected sequences; (b) subjective assessments of other potential contributors, and (c) subjective evaluation of consequence variations based upon limited sensitivity analysis.

The principal topic areas of this section include:

- Evaluation of WASH-1400 Uncertainty Bounds (3.8.1)
- Limerick Accident Sequence Uncertainty Characterization (3.8.2)
- Limerick Uncertainty Band (3.8.3)
- Qualitative Summary of Contributors to the Uncertainty in the Evaluation of Risk (3.8.4).

3.8.1 Evaluation of WASH-1400 Uncertainty Bounds

WASH-1400 characterized the uncertainty on their final CCDF Curves to be approximately a factor of 4 to 5 above and below the best estimate curves. WASH-1400 estimated accident sequence probability uncertainties by propagating the uncertainties of failure probabilities through the fault tree logic models. It should be noted, however, that the consequence portion of the calculation (in-containment and ex-plant) was then treated in a deterministic manner; i.e., uncertainty was not evaluated for that part of the analysis. Therefore, the final summary curves are effectively characterized by the uncertainties in the accident sequence probabilities.

The Lewis Committee subsequently stated that the uncertainties in WASH-1400 were understated. Evaluations over the past 5 to 7 years would indicate that, indeed, the uncertainties are larger than previously cited in WASH-1400. Documentation of some of these evaluations can be found in EPRI publications.

Following the release of WASH-1400 in August 1974, the question of uncertainty on the final risk curve was investigated through an in-depth review of those potential risk contributors identified by the Reactor Safety Study group. Among those potential contributors, twenty-seven items were identified as having been inadequately treated and were further explored (EPRI 217-2-3). Each of these was evaluated for its quantitative impact on the risk calculation (both probability and consequence of core melt).

Following formal issue of the Lewis report, further study was performed by EPRI. The estimation procedure employed involved significant dependence on engineering judgment. Also, the underlying assumptions in combining the uncertainty factors were that both probability and consequence of each risk contributor were log-normally distributed, and that the total risk was the product of all risk contributors. The results of EPRI-1130 approximately doubled the WASH-1400 uncertainty band.

3.8.2 Limerick Accident Sequence Uncertainty Characterization

The quantification of uncertainties is difficult due to the lack of knowledge of specific parameter bounds and the resulting cumulative sensitivity of the calculations. In lieu of rigorous quantification, some of the methods of WASH-1400 are adopted in this analysis for the characterization of accident sequence probability uncertainty ranges. These methods include:

- The assumption of log-normal distributions for most hardware failures
- Uncertainty factors of ~3 for most hardware failures
- Assumptions regarding short term operability of equipment and allowable credit for operator action utilizing the operator action stress curve from WASH-1400.

Uncertainties associated with the calculation of accident sequence probabilities can be evaluated by propagation of input uncertainties through each accident sequence, and combination of all accident sequences to determine the overall uncertainty range for each accident class.

This process can be simplified for the Limerick analysis due to the following:

1. There is a single dominant accident sequence with probability much larger than the other sequences. This allows the evaluation of a single sequence to characterize the uncertainties in that particular accident class.
2. Each of the accident sequences in the class have similar probabilities, and the uncertainty ranges associated with each are nearly the same. This allows the use of the uncertainty range determined from the explicit calculation for one sequence to represent the range for the Boolean sum of the class.

Based on the above discussion, the following evaluation of individual accident sequences is presented:

MSIV Closure and Loss of Coolant Injection: T_FQUX

This sequence is one of the major contributors to the Class I event sequence probability. Other sequences which contribute to the Class I probability have similar system and operator interactions, and therefore similar uncertainty ranges. The MSIV Closure with the subsequent failure of high pressure coolant injection, and the inability to return feedwater to service, coupled with the failure to depressurize is modeled in fault tree format. The component unavailabilities are input along with their probability distributions (assumed to be log-normal in most cases). The uncertainties are propagated through the fault tree model using Monte Carlo simulation (see Appendix K). This uncertainty bound is transferred through the calculations to the CCDF for those consequences affected by Class I. However, it should be noted that some measures of consequences (e.g. early fatalities) are not strongly affected by Class I sequences.

ATWS Accident Sequences

ATWS sequences, as evaluated in the LGS analysis, are important in the calculation of the CCDF for early fatalities contributing to Class IV sequence probabilities. The uncertainty distribution associated with the scram failure probability is one of the most important single elements in the estimation of the confidence bounds on ATWS accident sequence probabilities. A simplified approach is used to define the probability distribution to be assigned to scram failure.

Bayes' theorem makes it possible to update the state-of-knowledge of a given event by incorporating any available operating experience data into the prior distribution. Acknowledging the existence of different sets of experience data from different sources, Apostalokis et.al. (3-5) computed

*Note mean values are used in all accident sequence calculations.

a posterior distribution for each data source and then combined them to obtain the final composite posterior distribution of the scram failure frequency. It must be carefully noted that the uncertainty distribution constructed here depends heavily on the published values from Apostalokis, et.al. (3-5). However, these are only used to provide the relative distribution about the mean. The mean scram failure probability value is taken from the published NRC value of 3×10^{-5}/demand (NUREG-0460). The result is plotted as a histogram in Figure 3.8.1. It should be noted that a recent analysis by GE indicates that a significantly lower probability for failure to scram may be more realistic.

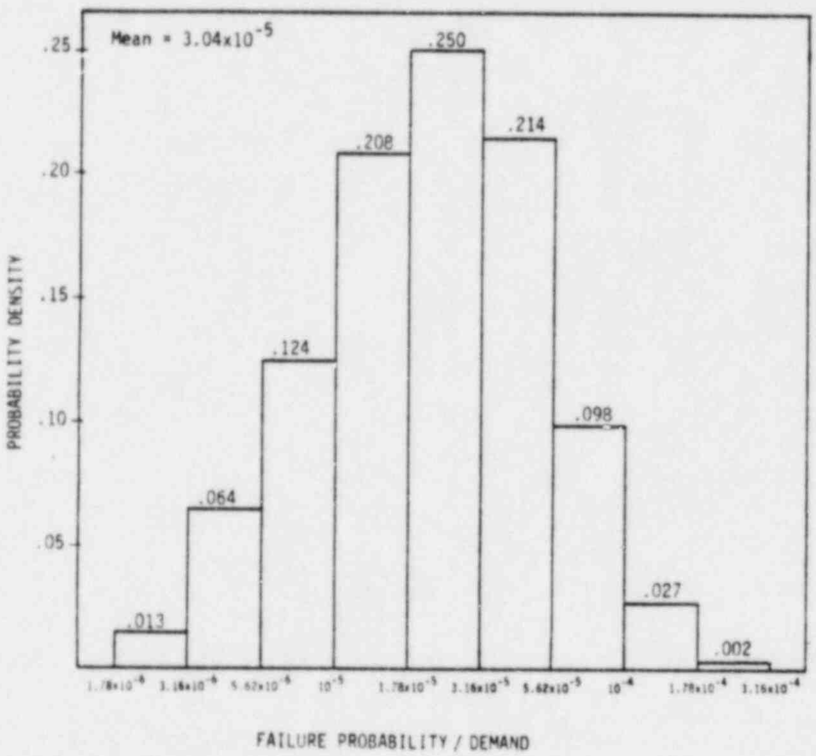

Figure 3.8.1 Scram System Failure Probability/Demand

Once the distribution for the scram failure probability is determined, the probability distribution for an ATWS sequence is generated using Monte Carlo simulation techniques. The sequence chosen here for the purpose of developing an uncertainty band representative of Class IV is the sequence which may lead to the largest release of radionuclides to the environment. This sequence is an ATWS followed by the total failure of the poison injection system coupled with continued operation of HPCI despite high containment pressure. The Monte Carlo simulation of this sequence was performed to estimate the uncertainty bounds for the entire Class IV sequence.

It should be noted that Class III accident sequences are also dominated by ATWS events; however, Class III sequences involve the loss of the coolant injection function prior to containment failure. The uncertainty bounds for these sequences are estimated to be comparable to those calculated above for the typical Class IV accident.

LOCA Sequences

One of the accident sequence contributors to Class II is the large LOCA initiator followed by a failure to provide adequate containment heat removal capability. For the Limerick analysis a mean value for large LOCAs is determined to be comparable to that used in WASH-1400 (see Appendix A). The probability distribution is assumed to be similar to that evaluated for WASH-1400; i.e., log-normally distributed, with an uncertainty factor of 10*.

* WASH-1400 states that the error factor on LOCA initiators is 30. The actual implementation of the data in accident sequences and the evaluation of their uncertainty do not reflect error bands of this magnitude. (A value of ~7 appears to have been used.)

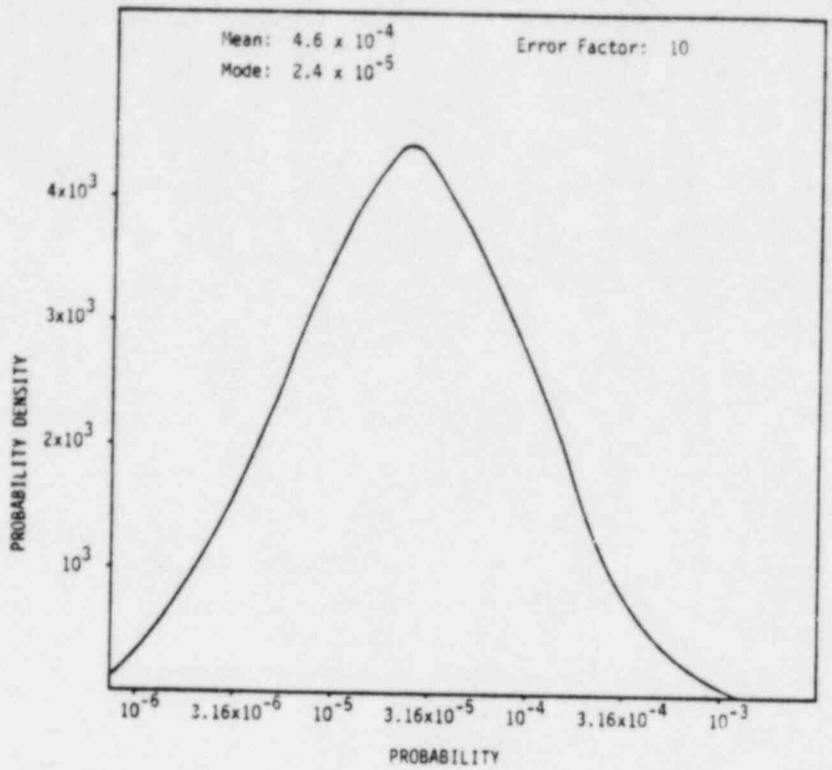

Figure 3.8.2 Probability Density Function for the Large LOCA Initiator

When the large LOCA initiator distribution is combined with the probability distribution of unsuccessful containment heat removal, the accident sequence probability distribution is determined to provide the uncertainty bounds for Class II.

3.8.3 Limerick Uncertainty Band

The CCDF uncertainty band for potential early fatalities associated with remote accident sequences at LGS has been established and is shown in Figure 3.8.3 to provide a perspective on the best estimate curve. This uncertainty band is a judgment of the potential variations due to the identified areas of uncertainty (see Section 3.8.4). The following factors have the major affect on the uncertainty bands established in Figure 3.8.3:

1. The quantitative evaluation of the uncertainty of accident sequences with the potential of leading to core melt.

2. The evaluation of many of the hardware and data uncertainties associated with accident sequences.

3. The completeness of the accident sequences analyzed.

4. Uncertainties in the consequence evaluation.

With regard to item 4, it is concluded that the uncertainties in the consequence evaluation are approximately of the same order of magnitude as the uncertainties in the accident sequence frequencies. This judgment is based upon estimates of the effects of the several contributors to consequence uncertainty, specifically in the areas of containment failure modes and offsite releases. The quantity of radionuclides released to the environment is uncertain due to lack of understanding of core melt phenomenology and interaction with the containment, probabilities of release fractions characteristic of steam and hydrogen explosions, and uncertainties in decontamination factors. Offsite distribution and effects of a postulated radionuclide release are uncertain due to uncertainties in health effects models, accuracy of past weather patterns to represent future patterns, and the potential dispersion of radionuclides in various weather patterns.

3.8.4 Qualitative Summary of Contributors to the Uncertainty in the Evaluation of Risk

The Limerick Generating Station analysis has been carried out using the groundrules provided in Section 1. Given these groundrules, best estimate CCDFs were calculated (for early and latent fatalities) and presented in Section 3.7. The principal areas of uncertainty in the best estimate early fatalities curve are identified in this section. A detailed sensitivity study comparing the CCDF variations with each identified parameter has not been performed. However,an attempt has been made to qualitatively assess the impact of each identified area.

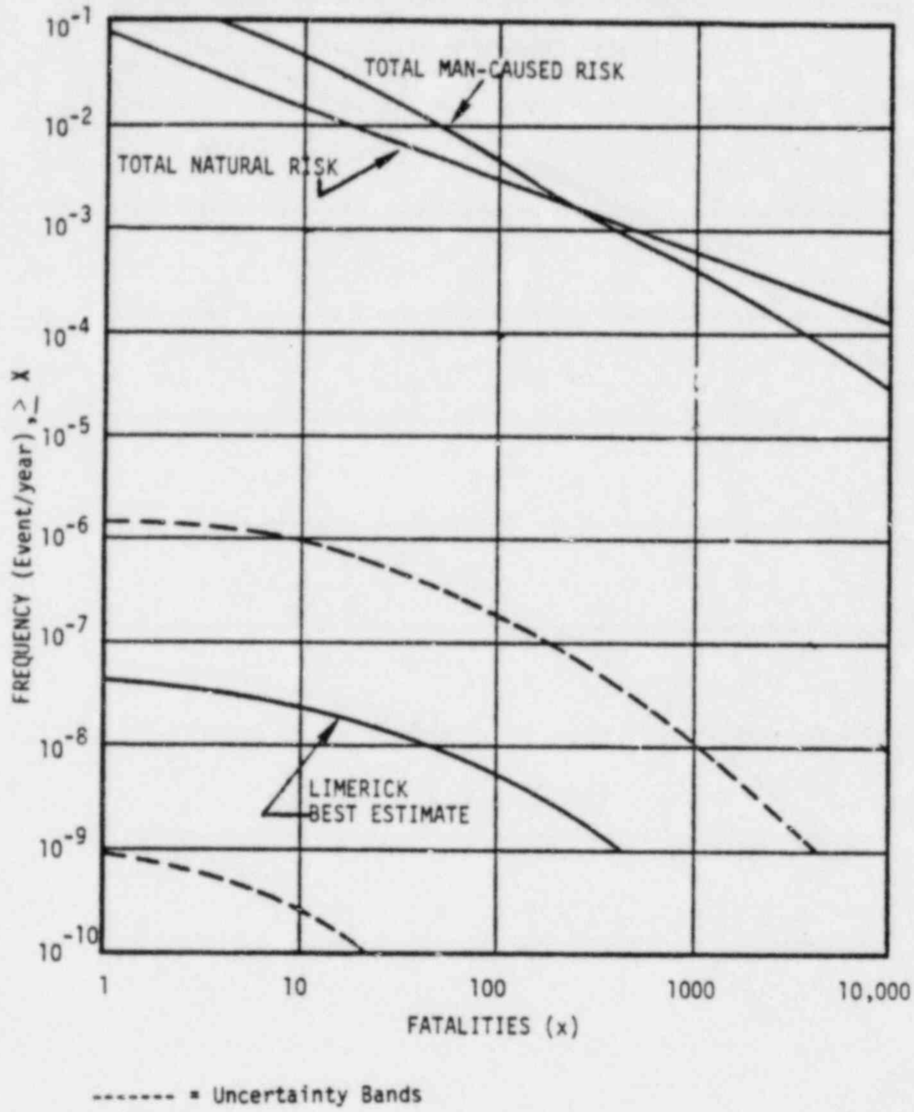

Figure 3.8.3 Summary of Risks Assumed by the Population Surrounding the LGS for Early Fatalities with the Estimated Uncertainty Band.

Tables 3.8.2 to 3.8.4 are provided as summaries of those items which contribute to the uncertainty in the best estimate CCDF. The distribution of uncertainties into categories of minor, moderate, and potentially significant are based upon a subjective evaluation of the effect of each item taken individually on the CCDF for early fatalities. However, since the calculation of CCDFs is a complex process the effects of each of the items is not strictly independent of all other items.

As previously noted, the Limerick probabilistic risk assessment has been performed as a best estimate analysis. The factors contributing to the uncertainty in the resulting CCDF curve for early fatalities in Tables 3.8.2 to 3.8.4 have not been individually quantified. Based on subjective consideration of these effects and the other considerations identified in Section 3.8, the uncertainty band shown in Figure 3.8.3 was constructed.

TABLE 3.8.2

SUMMARY OF AREAS OF UNCERTAINTY HAVING
A MINOR EFFECT ON THE LGS EARLY FATALITY CCDF

SUBJECT	ASSUMPTION USED IN LGS ANALYSIS	IMPACT
METHODOLOGY:		
Success Criteria	The success criteria are based on realistic calculations or estimates of system capability during accident conditions. Future changes in modeling, to more accurately reflect reality, may alter success criteria; therefore, there is an uncertainty associated with success criteria.	P*
Degraded Core Leads Directly to Core Melt	The assumption used in WASH-1400 and in the LGS analysis is that once a core loses identified methods of cooling, it will melt. This may be conservative.	P,C**
No Repair of Failed Systems	As in WASH-1400 very little, or no credit is given to the operator for restoring a system to service if it is failed or in maintenance.	P
Accident Sequences Characterization	Accident sequences are characterized by the most severe conditions associated with the event. There may be conservatisms in the sequence evaluation.	C
Common-mode Failures	Common connections and dependencies among systems were included based upon design drawings and proposed environmental qualification. The as-built plant may have interdependencies not modeled.	P
Constant Wind Direction in the CRAC Code	The wind direction is assumed constant throughout the accident sequence.	C
DATA:		
Plant/Component Age	Data for plants with a long operating history are not available. Therefore component failure rate data are in general an average of failure rates over the initial 5 to 10 years of plant operation.	P
Constant Failure Rate Assumption	The failure rate is assumed to be a constant. The time variation of component failure rates is not known. Recent EPRI work has shown that higher than normal failure rates may be expected during the initial year of plant operation. There is currently no characterization of the end of life performance of major plant components, i.e., pipes, pumps, valves.	P
Component Failure Rate Distribution	Log-normal distributions are assumed to describe component failure probability distributions. However, sufficient data does not exist to fully justify this assumption.	P
Human Error Probabilities	The only data used are data cited in WASH-1400 and the Human Reliability Handbook (Swain and Guttman).	P

*P = Probability **C = Consequence

TABLE 3.8.2 (continued)

SUBJECT	ASSUMPTION USED IN LGS ANALYSIS	IMPACT
EQUIPMENT:		
CRD Injection Water	Due to their relatively small capacity, the CRD pumps are not included in the analysis. There are, however, some conditions which would benefit from the CRD pump flow: • Manual shutdowns with gradual power reductions • Injection after decay heat has been reduced	P,C
MSIVs Reopened	During transients which result in MSIV closure it is assumed in the analysis (based upon limited operating experience) that the MSIVs can be reopened in sufficient time to restore feedwater flow to the reactor for some accident sequences.	P
CONTAINMENT:		
Containment Integrity at High Temperature and Pressure	Containment integrity is assumed to be with the pool temperature at 290°F and at internal pressures in the range 50 psig, with safety relief valves blowing down to the suppression pool. These conditions may result in containment loads that have not been proven to be acceptable.	P
Containment Failure	Lower pressures that used, for containment failure lead to: a. shorter retention time for fission products b. shifting of Class III events to Class IV	C
Molten Core Reaction	An area of uncertainty is the deposition molten core after it fails the RPV. It is uncertain what portion of the molten core may: • drop onto the diaphragm floor in one coherent mass • fragment and disperse around containment from blowdown of RPV if a large blowdown force occurs • stay inside the pedestal region of the diaphragm floor • melt through the diaphragm floor vents and drop into the suppression pool causing steam explosion(s).	C
Molten Core	In some of the dominant sequences, the oxide layer is predicted to freeze. The implication of this layer is uncertain. In the Limerick analysis, the vaporization release period is considered to occur whether or not the oxide layer freezes; therefore, the radioactivity release fractions are larger for those cases with the oxide	C

Table 3.8.2 (cont.)

SUBJECT	ASSUMPTION USED IN LGS ANALYSIS	IMPACT
RELEASE FRACTION REACT/CORRAL MODEL:		
Melt Release	The REACT model assumed that only 50% of the available radionuclides could be released. This assumed to cover plateout, etc. This has little effect since the bulk of the material release occurs from the vaporization release.	C
EX-PLANT EFFECTS		
Plume dispersion	The model used to define the narrowness of the plume as it traverses large distances (20 miles) has not been verified experimentally.	C
Evacuation model	The assumption that large numbers of people can be informed, motivated, and actually move away from a site has not been demonstrated for a large metropolitan area.	C
Shielding effectiveness	An appreciable portion of the effects on the public comes from gamma ray cloudshine. The degree of shielding is a function of the location of the population and the type of structures they occupy.	C

Table 3.8.3

SUMMARY OF AREAS OF UNCERTAINTY HAVING A MODERATE EFFECT ON THE LGS EARLY FATALITY CCDF

SUBJECT	ASSUMPTION USED IN LGS ANALYSIS	IMPACT
METHODOLOGY:		
Incomplete or Missing Accident Sequences	All possible accident sequences are not included. Becuase of the infinite number of possibilities that accident sequences could take, and because not all these sequences have been included in the quantification effort, it is possible that a sequence with a low probability of occurence may not be lepresented.	P
Containment Failure Leads Directly to Core Melt	Several potential mechanisms connecting containment failure with eventual core melt have been identified. However, this remains an assumption and an area of potential conservatism.	P
DATA:		
Meteorological Data	A five year sample of data (1972-1976) is used to characterize the LGS weather patterns. Sharp changes in future weather patterns are not included.	C
ADS Initiation by Operator	For some accident sequences manual depressurization is required. The probability of failure is estimated as 1/500 demands. Because of the uncertainty in the human error probabilities, this operation is assumed to have a larger uncertainty than typical hardware failures.	P
EQUIPMENT:		
Improvement in Hardware Based upon Operating Experience	Operating problems have resulted in selective improvements in component design. This is the case for diesels, relief valves, scram discharge volume, etc. Some of these improvements are not reflected in the analysis since failure rates are based upon the total available data.	P
CONTAINMENT:		
RPV Failure	The manner in which the RPV fails is uncertain. The INCOR method, modeled for a PWR, assumes that the RPV ruptures from the stress of the molten core rather than melting through. This model allows the entire bottom head of the vessel to fail at one instant. Other methods assume failure from melting, but the manner of melting is also uncertain.	C

Table 3.8.3 (cont.)

SUBJECT	ASSUMPTION USED IN LGS ANALYSIS	IMPACT
Steam Explosion	The probability of a steam explosion (in-vessel or in-containment) is the subject of controversy. The values from WASH-1400 are expected to be an upper bound. The values used in the LGS are viewed as high, however, a lower value appears to be difficult to justify based upon operating experience or test data.	P
Hydrogen Explosion	It is considered possible that a hydrogen explosion of sufficient magnitude to result in radioactivity releases comparable to an in-vessel steam explosion may occur. The probability is estimated to be 10% of the time a core melt occurs with the containment not inerted.	P
Containment Failure	All containment failures due to high internal pressure result in loss of coolant inventory makeup.	
RELEASE FRACTION REACT/CORRAL MODEL:		
Radioactive Releases	Both REACT and CORRAL use the WASH-1400 values for best estimate percent releases for each group of radionuclides. Theses values are uncertain, and recent experimental data indicate the larger numbers are conservative and the low estimates may be low. Group 4, tellurium, is especially considered to be uncertain since its release in WASH-1400 is for LOCA events. This directly effects the amount of the release, for it determines the cladding reaction, which determines the amount of tellurium that will be released. The values for tellurium from WASH-1400 used in the Limerick analysis may be overestimated.	C
EX-PLANT EFFECTS:		
Threshold effect in early fatalities	The applicability of a given threshold is strongly dependent upon the health of a person and the degree of medical attention received once exposed. In addition, changes in the threshold may affect the calculated number of early fatalities.	C
Duration of radionuclide release	The release of all the radionuclides calculated by CORRAL to escape for each containment failure mode and accident sequence is assumed to occur over a 30 minute period. This is longer than the WASH-1400 3 minute "puff"; however, the actual release for most accident sequences may be even longer.	C

Table 3.8.4

SUMMARY OF AREAS OF UNCERTAINTY HAVING A POTENTIALLY SIGNIFICANT EFFECT ON THE LGS EARLY FATALITY CCDF.

SUBJECT	ASSUMPTION USED IN LGS ANALYSIS	IMPACT
DATA:		
ATWS Frequency	Operating experience is insufficient to adequately characterize the potential for ATWS. The frequency used for the LGS analysis is that derived from NUREG-0460.	P
CONTAINMENT:		
Decontamination factors	Despite continued research into the behavior of different radionuclide species under postulated accident conditions, there is insufficient experimental information available to precisely define the decontamination factors. The values utilized in the LGS analysis appear to be conservative.	C

REFERENCES

3-1 A Risk Assessment of A Pressurized Water Reactor for Class VII - VIII, R. E. Hall, et. al. Brookhaven National Laboratory, NUREG CR/0603, October 1979.

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3-3 F. L. Leverenz, J. M. Koren, R. C. Erdmann, and G. S. Lellouche, ATWS: A Reappraisal, Part III, Frequency of Anticipated Transients, EPRI NP-801, July 1978.

3-4 Anticipated Transients Without Scram For Light Water Reactors NUREG-0460, Vol. 3, Staff Report, USNRC, December 1978.

3-5 G. Apostolaks, S. Kaplan, B. J. Garrick, and W. Dicksen, "Assessment of the Frequency of Failure to Scram in Light-Water Reactors," Nuclear Safety, Vol. 20, No. 6, November-December 1979, pp. 690-705.

Section 4

COMPARISON OF LIMERICK PRA WITH WASH-1400

The reactor Safety Study (WASH-1400) and the Limerick PRA are similar studies in that they both analyze the risk to the general public associated with the operation of a BWR. The methodology used in both studies is basically the same (probabilistic event/fault tree analysis) and the results are presented in the same manner (complementary cumulative distribution functions of offsite consequences).

While the Limerick PRA and WASH-1400 employed similar techniques to quantify risk, the details of the two analyses are substantially different. The differences fall into four major categories:

1. Plant siting
2. Design
3. Data base
4. Methodology.

The impact of the differences in the two analyses is evidenced by the differences in the calculated offsite consequences (CCDF curves). These results are compared on two bases (Section 4.2):

1. The effect of the Limerick site if no changes are made in the plant design, the evaluation methodology, or the input data from that used in WASH-1400. This comparison with WASH-1400 is referred to as the WASH-1400 BWR at the Limerick site.
2. The Limerick site-specific evaluation compared with WASH-1400 for cases where:
   - Only data and methodological differences are included
   - Only design differences are included.

4.1 DISCUSSION OF DIFFERENCES IN THE ANALYSES

4.1.1 Site Differences

The Limerick PRA considered the actual site population and meteorology associated with the Limerick site while WASH-1400 used a composite site with data averaged over all U. S. reactor sites. The Limerick site has a higher population density than that used in WASH-1400 and the combination of rain, wind, and temperature patterns is substantially different. Therefore, site-specific meteorology and population is used (see Appendix E for further details).

4.1.2 Design Differences

The major differences in design between the WASH-1400 BWR and Limerick are summarized in Table 4.1. The most important design features from the viewpoint of risk are:

1. The reinforced concrete steel lined containment, which has a different suppression pool configuration and different radioactive release pathways and containment failure modes.

2. The containment overpressure relief system, which provides the capability to avoid containment overpressure and failure for certain accident sequences.

3. An improved shutdown system that provides a more reliable and diverse means to insert negative reactivity into the core to shut down the reactor.

4. More reliable offsite power supplies are available for the Limerick site.

Additionally, the larger standby gas treatment system, new safety relief valves, spray pond, improved pumps and diesel generator capability designed into the Limerick plant provide improved reliability over that associated with the WASH-1400 BWR.

Table 4.1

COMPARISON OF WASH-1400 AND
LIMERICK DESIGN DIFFERENCES

DESIGN FEATURE	WASH-1400	LIMERICK	EFFECT
RHR Connections	4 dedicated RHR Heat Exchangers	2 RHR Heat Exchangers with the ability to cross connect the Unit 2 pumps	A slight reduction in the RHR reliability, but affecting a sequence which has a low probability of occurrence at Limerick
Containment Design	Mark I	Mark II	Different suppression pool configuration, radioactive release pathways and containment failure modes
Off Site Power Supplies	2 Redundant Offsite Power Supplies	Five Offsite Power Supplies	Decreases the probability of loss of power. This results in a lower probability of core melt for certain sequences.
Containment Overpressure Relief	Not Included	Included to provide a method of maintaining containment pressure below design	Increase the probability of maintaining the containment intact. This results in a reduction in core melt probability
NPSH Requirement On Low Pressure ECCS Pumps	Pumps were failed at saturation conditions	Pumps are designed to pump at saturated conditions	Increases the probability of successful coolant injection under adverse containment conditions, thereby reducing the probability of a core melt
ATWS Mitigation (Alternate 3A)	RPT and Manual SLC intiation	a. ARI b. RPT c. Automatic SLC d. FW Runback e. Scram discharge instrument volume modifications	Reduces the probability of an ATWS by prevention (ARI+RPT). Also reduces the probability of an ATWS leading to a core melt by mitigation once it has occurred

4.1.3 Data Differences

Since the publication of WASH-1400, a significant increase in the data associated with nuclear reactor operation has been accumulated. This newer data provides enhanced understanding of maintenance and component reliability quantifications as well as frequency of potential transient initiators to be investigated. In addition, the use of plant or site-specific data which is directly applicable to Limerick provides a more accurate representation than that from the more generic data used in WASH-1400. Some of the more important aspects of these differences are summarized in Table 4.2.

4.1.4 Methodology Differences

The methodology used in Limerick represents an update of the WASH-1400 approach. The computer models have been improved to provide a more comprehensive treatment of the complex physical interactions that can take place during a nuclear reactor accident.

The event trees that were used to describe transient initiated accidents in the Limerick PRA are larger in number and contain more detail than was possible in WASH-1400. For example, four transient events were analyzed in the Limerick PRA compared to only one in WASH-1400. In addition, ATWS sequence evaluations received increased attention.

Experimental investigations into the mechanisms of core melt, reactor pressure vessel failure, and hot metal-water interactions have provided an improved understanding of the performance of degraded cores. This knowledge has been incorporated into the core performance modeling codes to produce an improved representation of containment pressure rises due to rapid steam generation and hydrogen burning.

A summary of the more important methodology differences between WASH-1400 and the Limerick PRA is presented in Table 4.3.

Table 4.2
COMPARISON OF WASH-1400 AND LIMERICK
DATA BASE DIFFERENCES

DATA INPUT ITEM	WASH-1400	LIMERICK	EFFECT
•Maintenance Data	Extrapolated Data from Dresden and Quad Cities	PECo Experience (Peach Bottom Specific)	Reduction in the impact on system unavailability. Very small net effect on risk
•Component Reliability	Multitude of Data sources	Used the following sources in the order given: a. Peach Bottom or PECo Grid Specific b. NRC evaluated component reliability c. GE Data d. Wash-1400	Some changes in the calculated frequency of core melt. Some positive and negative contributions
•Transient Initiator Probability	Estimate of nuclear operating experience 1972-1973	EPRI survey of utilities summarizing operating experience through 1977. Also later GE Data.	Better definition of types of transients. Net effect increases the calcuated measures of risk
•Scram System Failure Probability	Fault Tree Evaluation	NUREG-0460	Increases the calcuated risk
•Human Error Probabilities	Swain	Swain & Guttmann and WASH-1400	Approximately the same
• Probability of coherent in-vessel steam explosion	Due to lack of data a rather high probability was estimated (~ .01)	Additional experimental data and analysis indicates the probability may be much lower than identified in WASH-1400	This, coupled with the elimination of smoothing (see methodology discussion), leads to a reduction in the early fatality CCDF

Table 4.3

COMPARISON OF WASH-1400 AND LIMERICK METHODOLOGY DIFFERENCES

METHODOLOGY ITEM	REACTOR SAFETY STUDY	LIMERICK PRA	QUALITATIVE EFFECT
ACCIDENT SEQUENCE DEFINITION •Transient Event Trees	One transient event tree	Four transient event trees	Better definition of system effects following special transient initiators. Net effect is to increase slightly the calculated risk, as measured by the CCDF for latent fatalities
•ATWS Event Trees	Part of transient event tree	Seperate event trees with significant increase in the level of detail	Better definition of system interactions during an ATWS
•Emergency Operator Guidelines	Plant specific procedures	General Electric Emergency Procedure Guidelines	Redefinition of plant status and operator interaction
•Success Criteria	Licensing Basis	Realistic estimate of system capability	Reduction in probabilities of accident sequences leading to degraded core conditions
CONTAINMENT ANALYSIS •Code Package	Hand calcuations plus CORRAL	Contempt LT plus CORRAL	Small effect in reducing calculated risk in LGS PRA
•Suppression Pool Effectiveness (Decontamination Factors)	No credit for suppression pool DF if saturated	Some credit taken for suppression pool decontamination, even if saturated	Decreases early radionuclide release
•Fission Product Retention in Reactor Enclosure	No credit	Minimal credit	Negligible effect on early fatality CCDF
•Reactor Vessel Failure Mechanism During Melt Through Process	Intantaneous at bottom head creep rupture failure	Failure at the penetration in the bottom Head	No instantaneous failure of containment following RPV failure, therefore a longer fission product holdup in containment before postulated containment failure
OFFSITE EFFECTS •Population Shielding	Composite Site	Typical Pennsylvania homes used in estimating effective shielding	Reduction in risk
•Radionuclide Puff Duration	Short Duration (~3 min) for release of all radionuclides	Slightly longer duration (~30 min) for release of all radionuclides	Reduction in cloud concentration with offsetting increase in exposure time
•Smoothing* of Accident Sequence Probabilities	Included smoothing to account for miscategorization	No smoothing, because sequences are defined more precisely	Noticeable decrease in the high end of the early fatality CCDF

* Placing a portion of the probabilities associated with a given category in adjacent categories to account for miscategorization

4.2 EFFECTS OF THE DIFFERENCES

In Section 3 the offsite effects for the Limerick Generating Station PRA are presented, and compared with similar effects calculated for the Reactor Study (WASH-1400). The purpose of this section is to attempt to isolate which effects lead to changes in the offsite CCDF consequence curves.

4.2.1 WASH-1400 BWR at Limerick (Site Effects)

Figure 4.1 is the complementary cumulative distribution function (CCDF) for the WASH-1400 BWR at the Limerick site. As can be seen, due to the increased population density at Limerick, there is a potential increase in risk associated with this site. Even with the higher risk associated with the WASH-1400 BWR at the Limerick site as postulated, it can be seen that the risk is still approximately 100,000 times less than the size of other risks.

In performing this analysis, the following were utilized directly from WASH-1400:

- The accident sequence probabilities. (The methodology, data, and plant design were not changed.)
- The radionuclide release fractions. (The containment response and release fraction calculations were not changed.)
- The offsite effects model (CRAC). (The basic model from WASH-1400 was implemented precisely as it was in the Reactor Safety Study.)

It must be emphasized that this calculation was made to provide the reader with an approximate measure of the impact of the Limerick site on the public. Two key aspects of the problem are not included which make this calculation useful only on a relative basis and not as an absolute measure of risk to the public. These two aspects are:

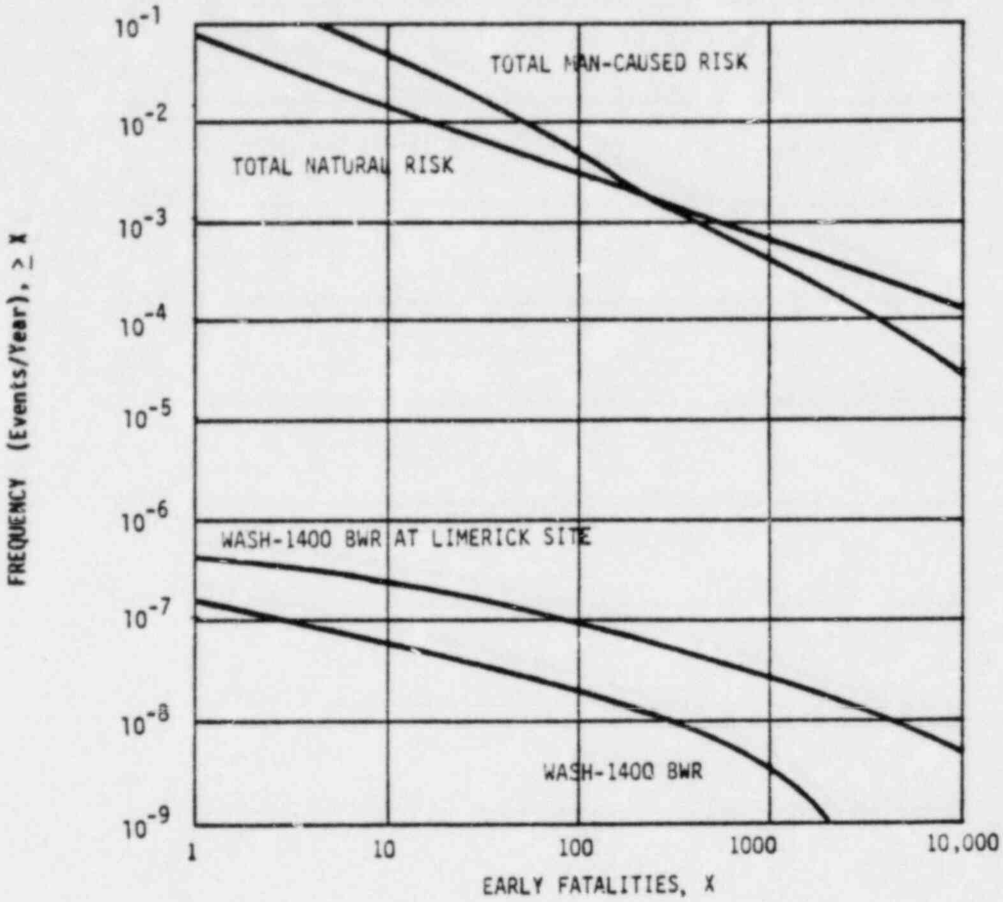

Figure 4.1 - Comparison of WASH-1400 at Limerick with WASH-1400 (Site Effects)

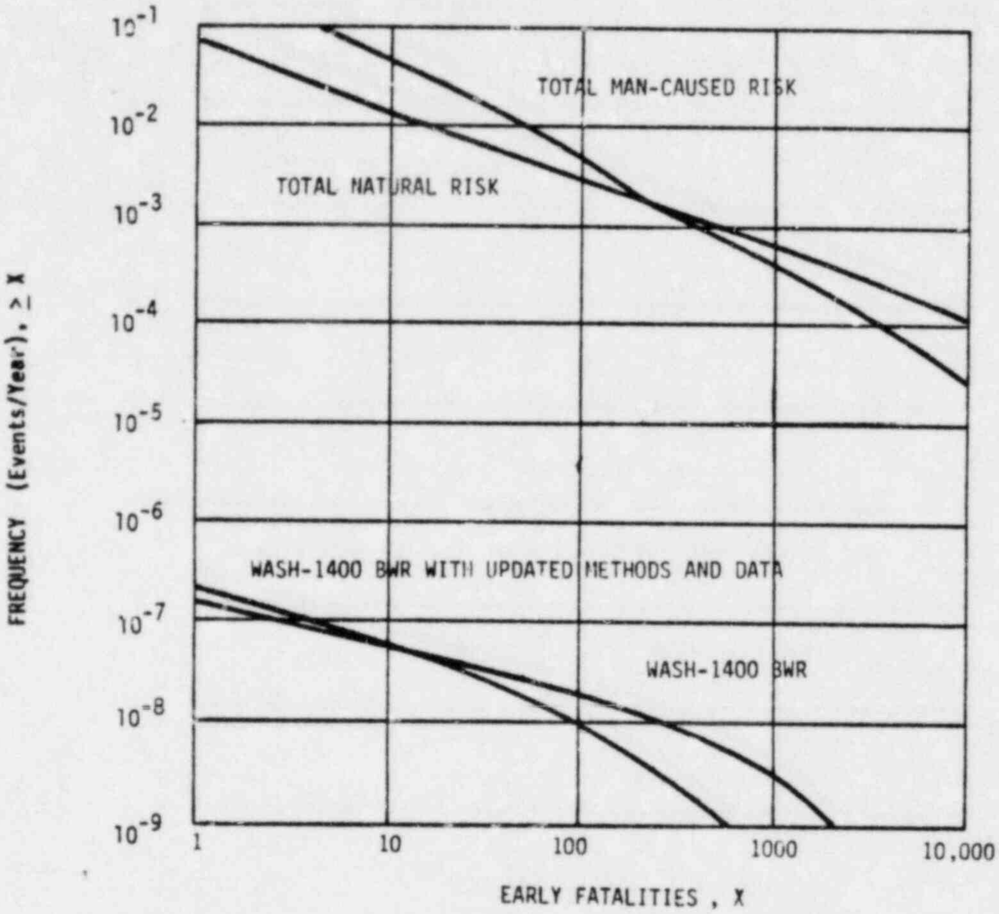

Figure 4.2 Effects of Data and Methods Differences

1. There is no such plant-site configuration. The actual plant design is different, and the evaluated offsite CCDFs are reduced because of these differences, and

2. There are several assumptions in the WASH-1400 methodology which are either overconservative* or in error.

4.2.2 Effects of Data and Methodology Differences

Figure 4.2 shows the CCDF for early fatalities for the WASH-1400 BWR with updated data and methods, as used in the Limerick PRA. The principal changes in methodology which lead to the net downward shift in the CCDF are the following:

1. Reduction in the probability of an in-vessel steam explosion leading to reactor vessel failure and containment failure. The probability is reduced by a factor of ten (see Appendix J).

2. Elimination of the use of smoothing based upon the Lewis Committee comments, plus the use of a better definition of accident sequences and their release fractions.

Both of these effects lead to a reduction in the probability of high consequence events. These high consequence events were characterized in WASH-1400 as the BWR Category 1 sequences.

It must be noted that while there were methodology changes which tend to reduce the offsite effects as noted above, there were also changes in methodology which tend to increase the calculated offsite effects. The principal methodology change which tends to increase the CCDF for early fatalities is the redefinition of ATWS accident sequences to include the possibility of containment failure prior to core melt. This event sequence leads to a substantial increase in the calculated CCDF for early fatalities.

*Overconservatisms are inappropriate in risk evaluation because they may focus attention on sequences or problems which should not be the principal areas of interest.

4.2.3 Effects of the Limerick Design

Figure 4.3 shows the early fatality CCDF for the Limerick design at the WASH-1400 composite site using WASH-1400 data and methods. The difference between this curve and the WASH-1400 curve represents the effects of the design differences (discussed in Section 4.1.2) between Limerick and the WASH-1400 BWR. The principal design differences between Limerick and the WASH-1400 BWR which affect the early fatality CCDF are the following:

- ATWS Alternate 3A implementation
- Containment overpressure relief
- Mark II containment
- PJM grid reliability.

4.3 SUMMARY

Figure 4.4 summarizes the effects of the differences between the WASH-1400 analysis and the Limerick PRA. Examination of Figures 4.1, 4.2, and 4.3 reveals the following:

- The Limerick site acts to increase the risk
- The effect of updated data and methods is to increase the risk estimate for low consequence levels and decrease the risk estimate for higher consequence levels.
- Design features incorporated into Limerick offset the site effects, to produce a net reduction in risk for Limerick relative to the WASH-1400 BWR.

The effects of the differences between the WASH-1400 analysis and the Limerick analysis have been evaluated in terms of the best estimate value of early fatality CCDFs. The effects on latent fatality CCDFs would be similar, but the differences would be relatively smaller.

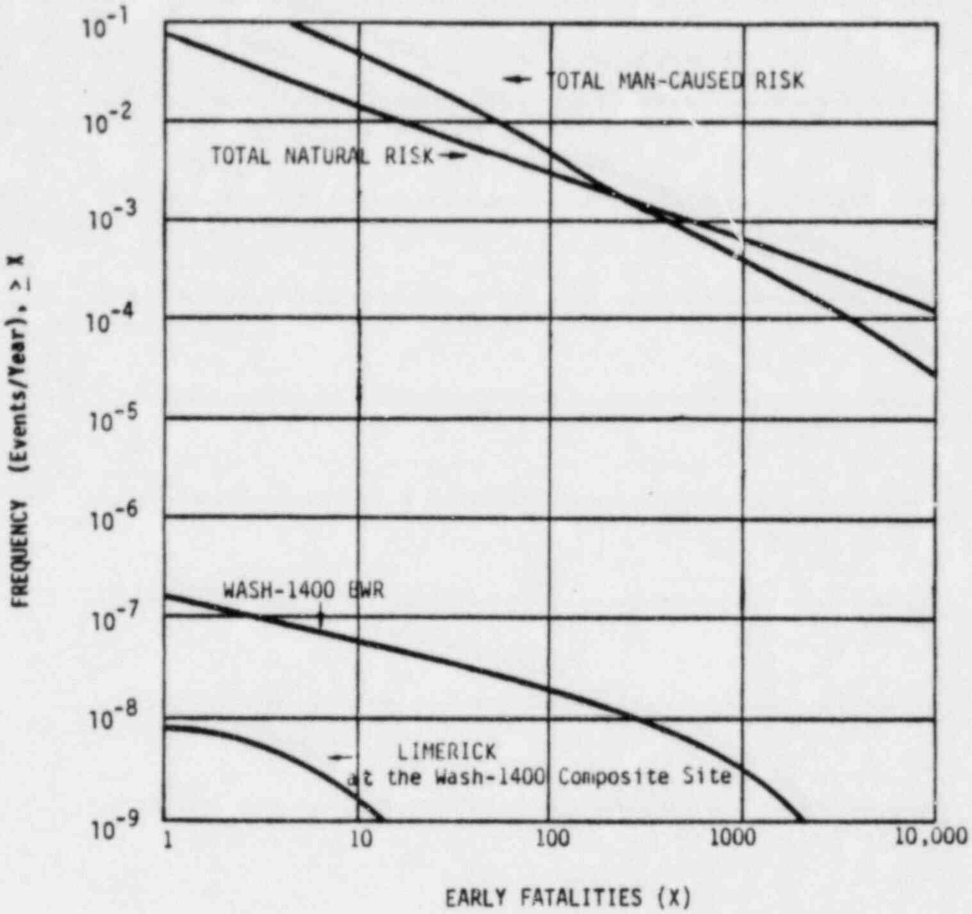

Figure 4.3 Effects of Design Differences

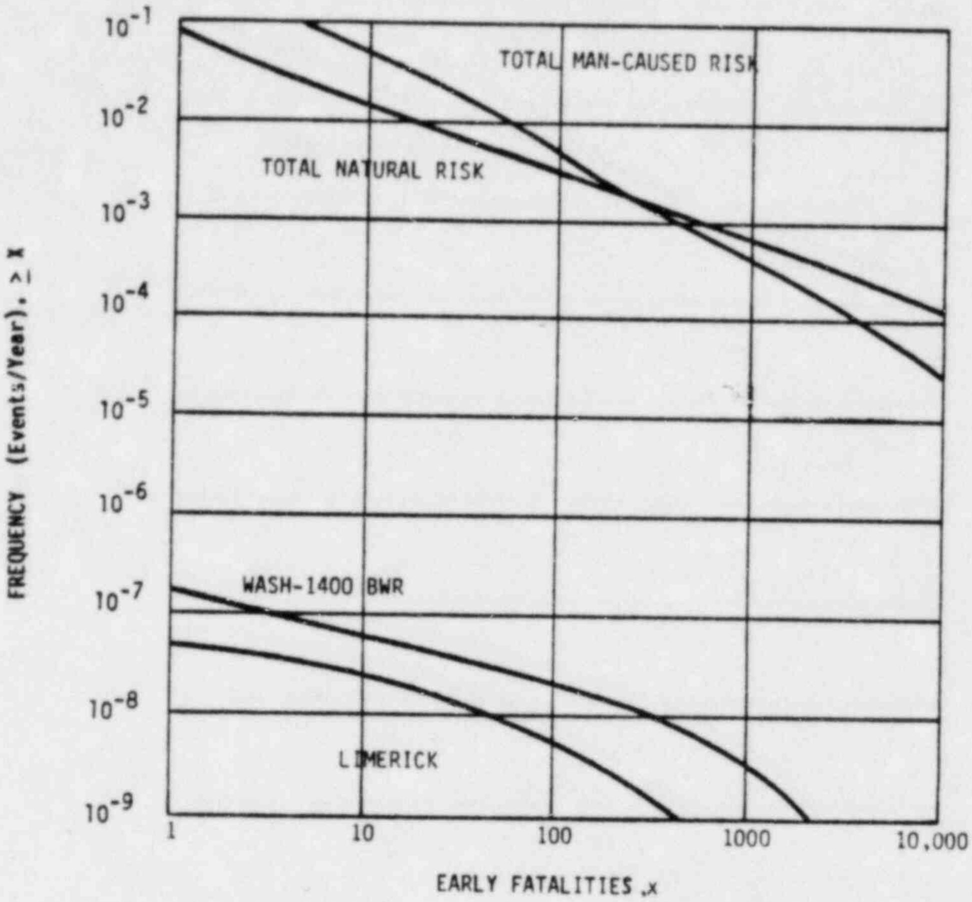

Figure 4.4 - Comparison of WASH-1400 and Limerick

Section 5

CONCLUSION AND SUMMARY

The Limerick Generating Station Probabilistic Risk Assessment represents the implementation of the best analysis tools available for the identification of potential accident scenarios and evaluation of the attendant level of risk to the public. The key features of the methods used include the use of fault trees and event trees to develop and quantify the probability of postulated accident sequences, an accident analysis code package developed through EPRI to evaluate plant thermal-hydraulic responses and radioactivity releases for severe accidents, and the use of CRAC to calculate the consequences to the public of releases of radionuclides to the environment. The analysis includes the same general types of accident initiators as evaluated in WASH-1400, i.e., transients and LOCAs under operating conditions, with and without scram. Excluded from the assessment are event sequences associated with external events, such as seismic, tornado and flood; fires; sabotage; and operator errors of commission. The risk evaluation techniques used involve several potentially important uncertainties, which are incorporated into uncertainty bands around the best estimate calculations. These uncertainties, and how they are treated in the Limerick analysis, are discussed in Section 3.8 and Appendix I. The LGS analysis can be compared directly to, and on the same basis as, the WASH-1400 evaluation.

The results of the LGS evaluation are shown in Figure 5.1 and 5.2, and can be summarized as follows:

1. The calculated core melt frequency for Limerick is approximately one half that calculated in WASH-1400. The accident sequence contributors identified and evaluated in the Limerick analysis are of a different nature, and more numerous than those identified and assessed in WASH-1400.

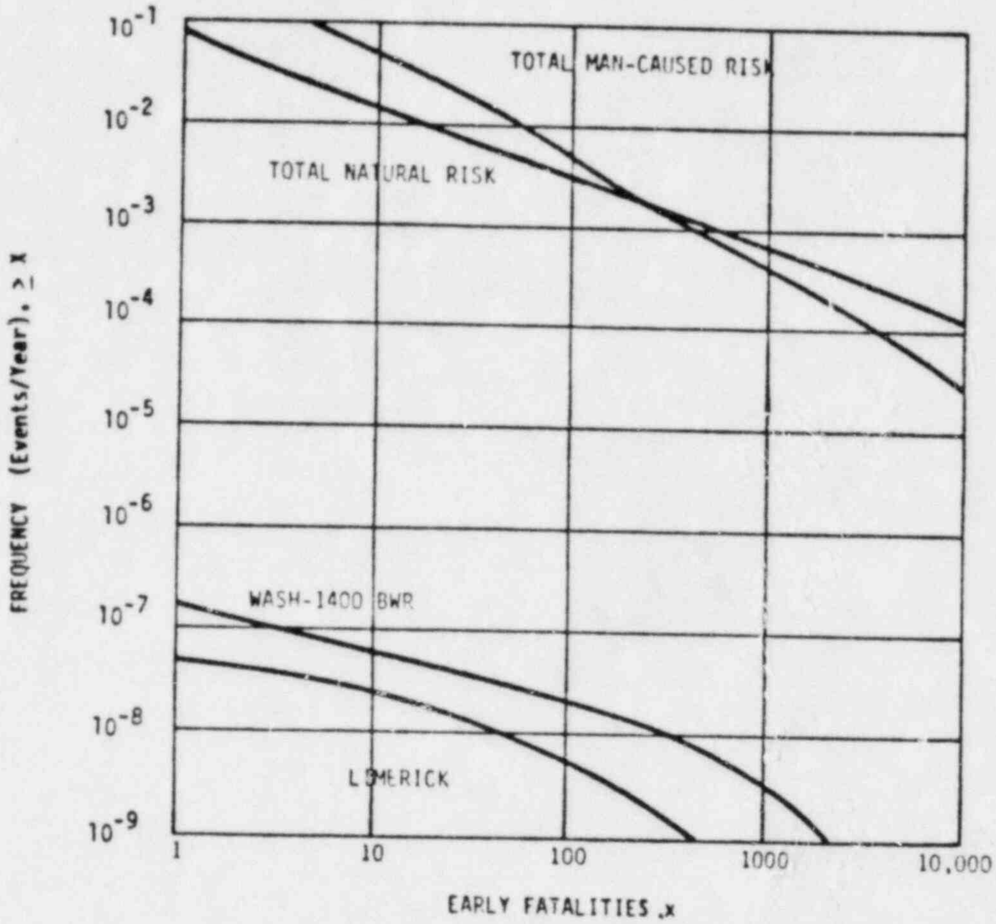

Figure 5.1 - Comparison of WASH-1400 and Limerick

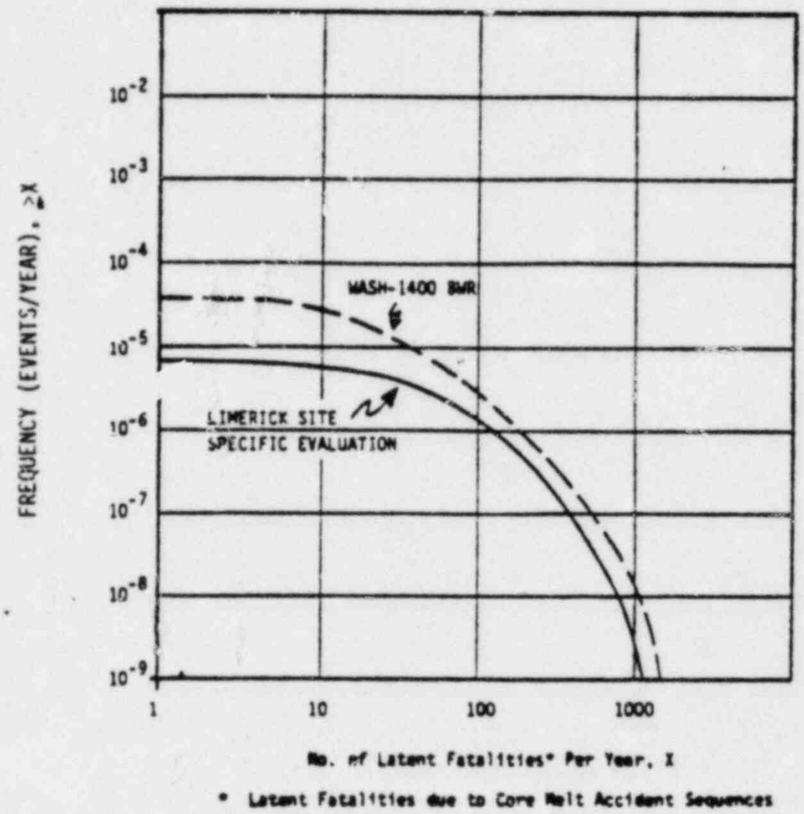

Figure 5.2 Comparison of the CCDF for Latent Fatalities for the Population Surrounding the LGS: WASH-1400 Composite Site BWR and the Limerick Site Specific Evaluation

2. The Limerick best estimate CCDF curves are below the published WASH-1400 CCDF curves for both early fatalities and latent fatalities for all calculated consequences.

3. The Limerick best estimate CCDF for early fatalities is several orders of magnitude below the CCDFs due to all natural and man-made risks.

4. Even with the uncertainties involved in the analysis, the Limerick Generating Station is not expected to represent any undue or disproportionate risk to the public.

The LGS analysis includes a major reevaluation of the assumptions and techniques used in WASH-1400, taking into account significant comments from the Lewis Committee review of WASH-1400. The following items are key differences in the LGS analysis with respect to WASH-1400:

1. The estimate of the probability of the in-vessel steam explosion leading directly to containment failure was reassessed (see Appendix H). The probability of a steam explosion leading directly to a containment failure is reduced by a factor of ten.

2. The use of accident categories in WASH-1400 required lumping accident sequences having major differences in potential consequences into the same category, for consequence evaluation. For the LGS evaluation, the use of categories was eliminated, and each unique sequence type was evaluated separately.

3. WASH-1400 used a concept of smoothing of probabilities between categories, to account for miscategorization and other uncertainties. This procedure was unnecessary in the LGS evaluation, because of better definition of accident sequence consequence evaluation.

4. Accident sequences were reevaluated, and additional sequences affecting offsite consequences were identified.

5. Component failure rate data were reevaluated based upon the latest operating nuclear data.

6. Philadelphia Electric Company nuclear experience data for maintenance operations, diesel reliability, and offsite power availability were all used in the LGS evaluation. These data are believed to be more applicable to Limerick than the broader-based WASH-1400 data.

Philadelphia Electric Company has incorporated several design changes in the Limerick Generating Station that make the plant safer. These design changes include:

1. Containment overpressure relief to offer an alternate method of containment heat removal, and prevent containment rupture when no abnormal radioactivity exists in containment.

2. An automatic boron injection system as a backup to the control rods, in the event of a postulated ATWS condition.

3. Low pressure pumps which do not require a positive back pressure, for operation at high suppression pool temperature.

4. Multiple interconnections to offsite power sources.

5. Four diesels per unit. (Total of eight for the station.)

The analysis is characterized as a best-estimate of the probabilities and consequences of postulated accidents which could occur at the Limerick Generating Station.

Despite the attempts to perform a realistic analysis, there remain some potentially conservative assumptions, data and methods. These include:

1. Plume dispersion, which impacts the calculation of early effects, may be conservatively modeled in CRAC.

2. Containment failure is assumed to lead directly to unacceptable core conditions.

3. The scram failure probability from the NRC NUREG-0460 is used. Recent analysis by GE indicates a significantly lower value may be appropriate.

4. Recovery of failed equipment is generally not included; even failure modes which could be remedied through very minor operations.

5. The radionuclide release fractions which were used may be higher than the true mean value.

The best estimate can only be used effectively if the range of uncertainty associated with it can be characterized. Figure 5.3 shows the uncertainty bounds on the best estimate CCDF for early fatalities. Uncertainty bands for the latent fatality CCDF would be similar.

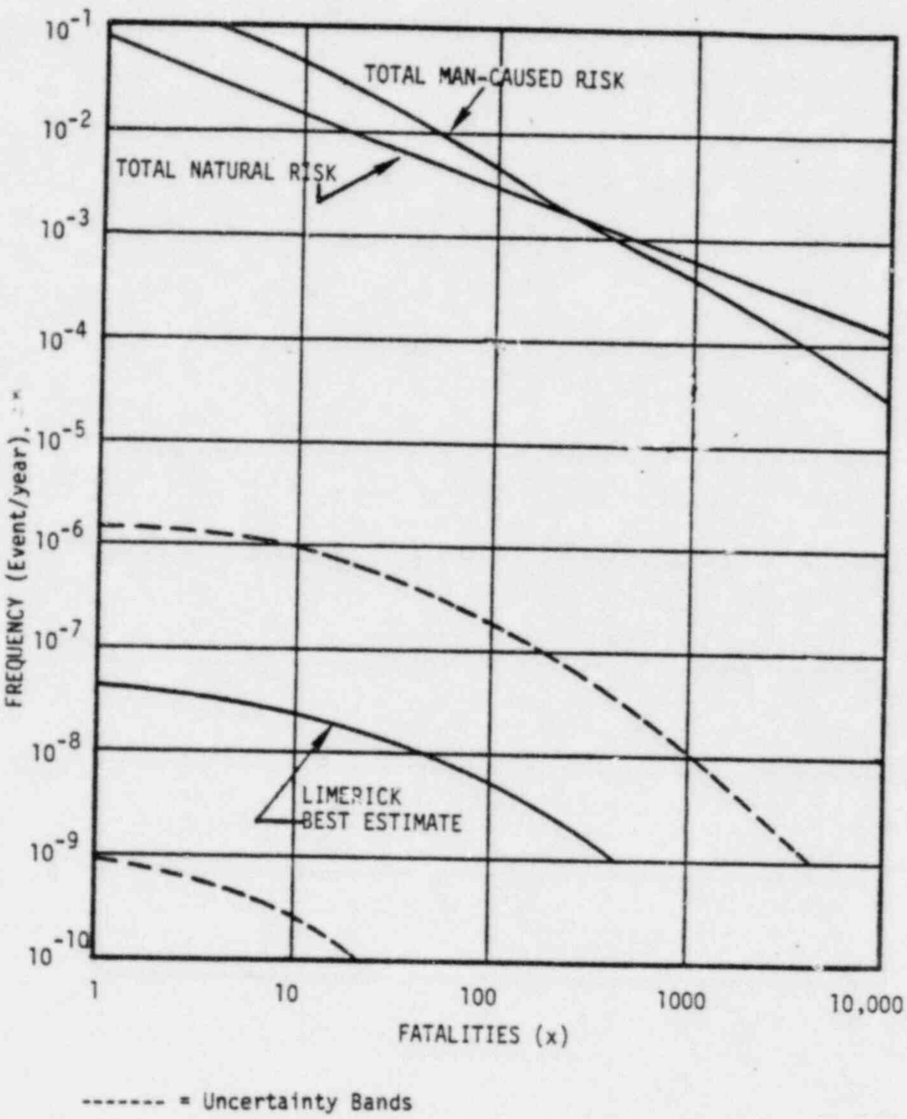

Figure 5.3 Uncertainty Bands on Limerick Best Estimate CCDF

Section 6

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