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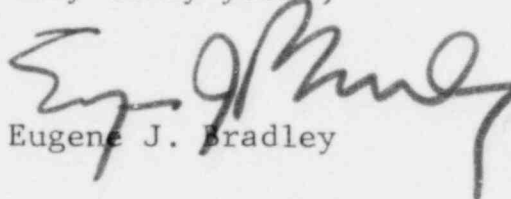
Mr. Darrell G. Eisenhut, Director
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Limerick Generating Station Units 1 and 2
Docket Nos. 50-352 and 50-353
Probabilistic Risk Assessment

Dear Mr. Eisenhut:

Your July 6, 1981 letter to E. G. Bauer, Jr., requested additional information on the Limerick Generating Station Probabilistic Risk Assessment. Transmitted herewith are 50 copies of the Company's responses to questions 720.1, 720.2 and 720.3.

Very truly yours,


Eugene J. Bradley

EJB:mk
Enclosures

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A site review was a major part of the analysis. This consisted of a review of important weather conditions to determine prevailing wind directions 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

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 $\geq 0.08\text{ft}^2$ Liquid Break $\geq 0.1\text{ft}^2$	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 Liquid Break 0.004 to 0.1ft^2	HPCI OR 1 of 4 LPCI Pumps OR 1 of 2 CS Subsystems } and ADS*	1 RHR OR COR
Small LOCA: Steam Break $< 0.016\text{ft}^2$ Liquid Break $< .004\text{ft}^2$	HPCI OR RCIC OR 1 Feedwater Pump OR 1 of 2 CS Subsystems } and OR 1 of 4 LPCI Pumps } OR 1 Condensate Pump } 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.

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
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

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 should perform during an accident sequence. In addition, those maintenance recovery actions which may be in error and which would adversely affect the system operation 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.

TABLE 1.4 (continued)

MMH	Monorail Mounted Hoist
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
NC	Normally Closed
NEO	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	Recirculation 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
SORV	Stuck Open Relief Valve
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

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 traveled 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 other 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.

*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.

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 sub-

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 Transmission Lines)

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)
<u>MSIV Closure</u>	<u>1.08</u>
Closure of all MSIVs	1.00
Turbine Trip Without Bypass	0.01
Loss of Condenser	0.067
<u>Turbine Trip</u>	<u>3.38</u>
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
<u>Loss of Offsite Power</u>	<u>.38</u> *
<u>Inadvertent Open Relief Valve</u>	<u>.06</u>
<u>Loss of Feedwater</u>	<u>.70</u>
TOTAL	6.2
MANUAL SHUTDOWNS	3.2

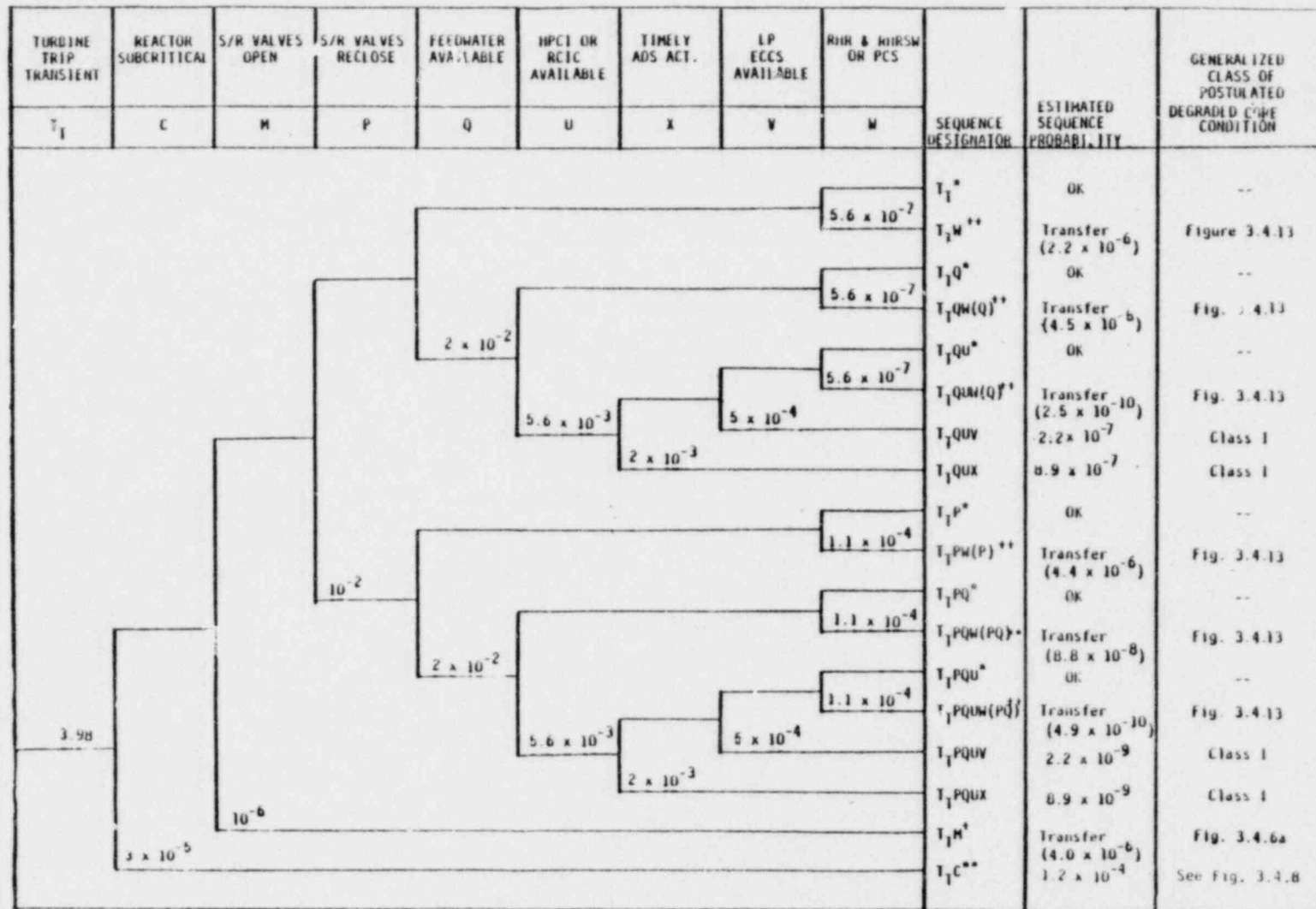
*Not used in the Limerick PRA. Limerick site-specific data was used.

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 ≥ 4" Diam.	4.0×10^{-4}
Medium Pipe < 4" Diam. ≥ 1" Diam.	2.0×10^{-3}
Small Pipe < 1" Diam.	1.0×10^{-2}

3-14



*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

- 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.

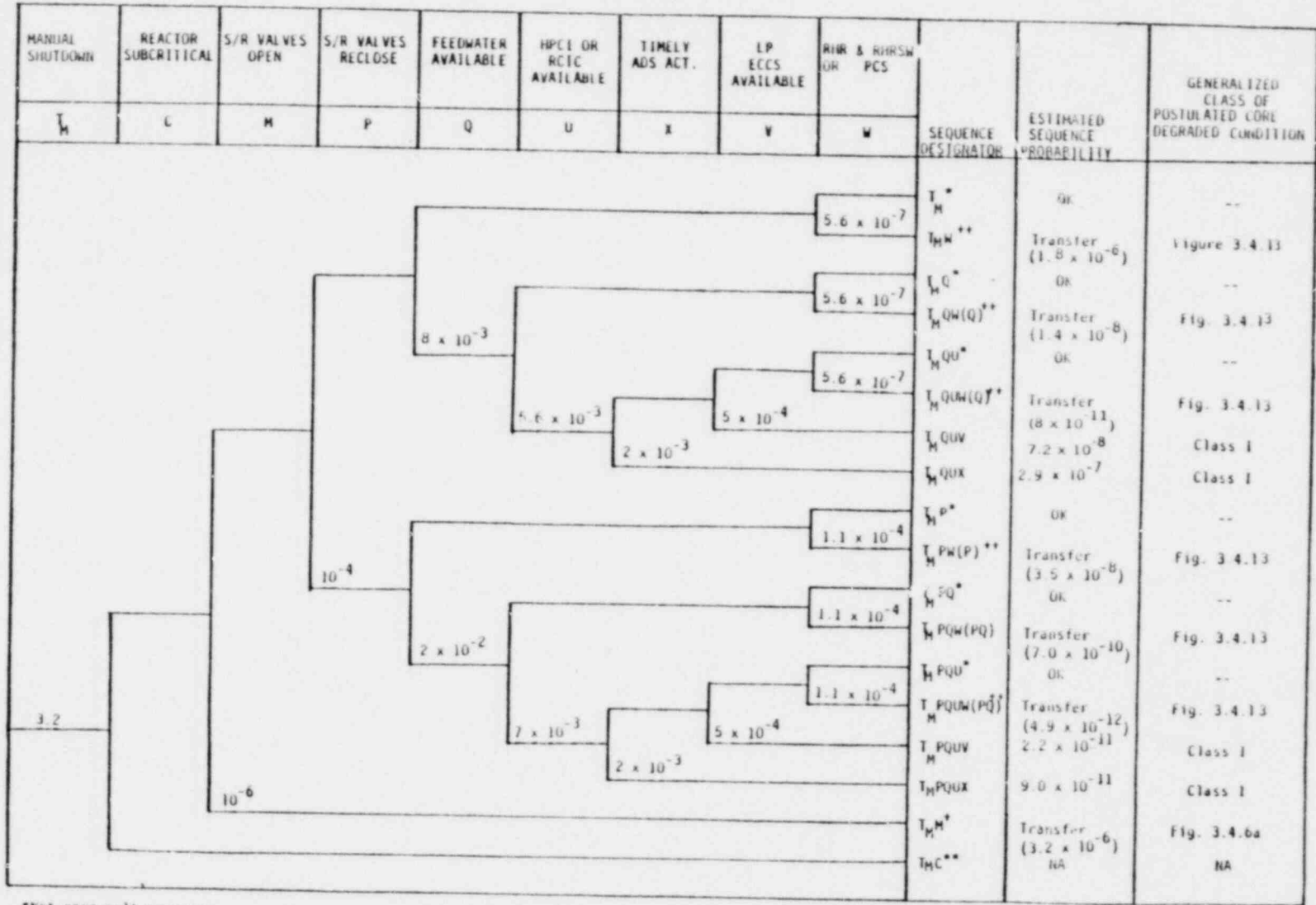
T_M -- Functions in Event Tree

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 3.4.4.

**The same value used for the turbine trip transient was also used (conservatively) for the manual shutdown.



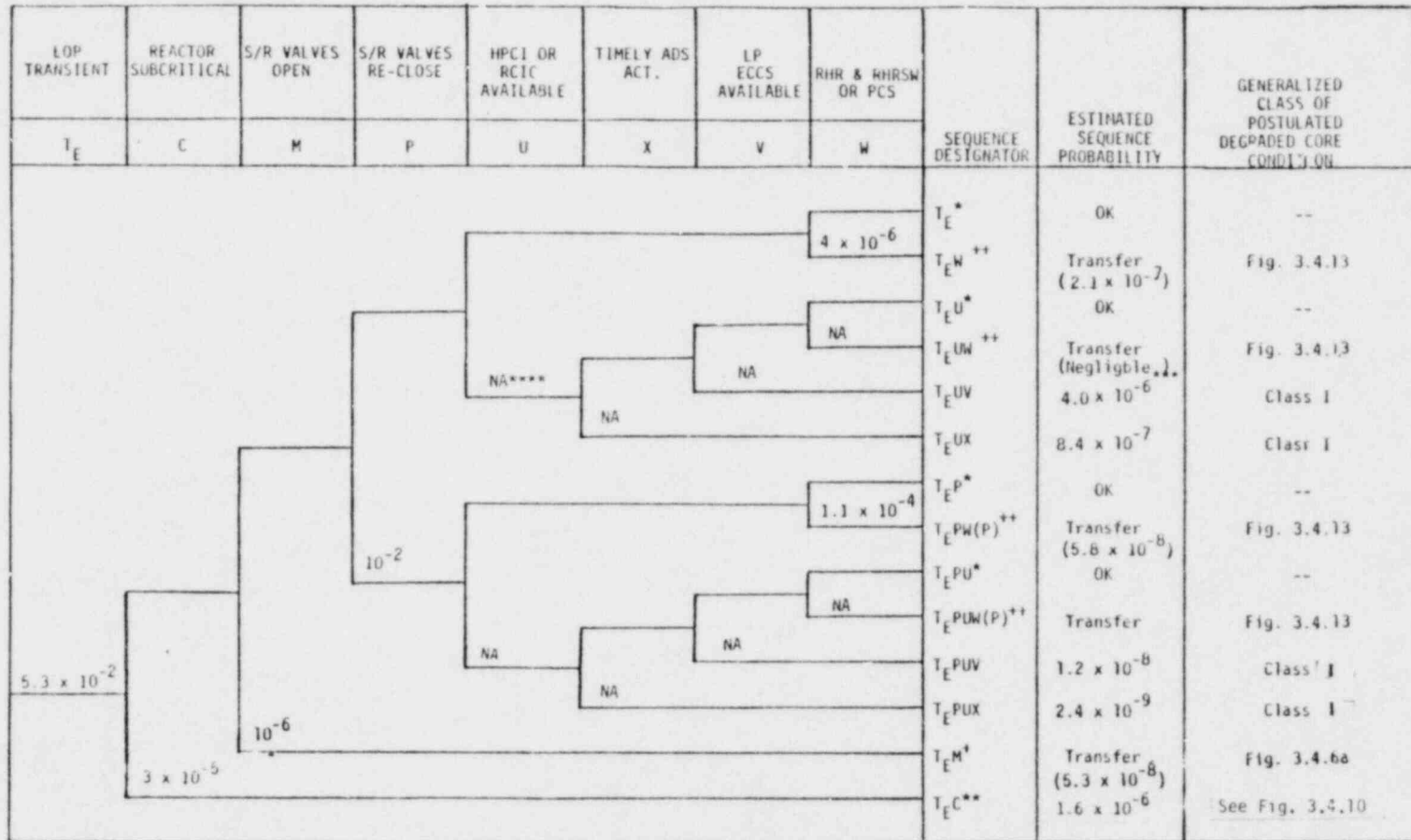
*Not core melt sequence
 **ATWS is judged not to be risk contributor for manual shutdowns.
 *Transfer to large LOCA event tree
 **Transfer to bridge tree

Figure 3.4.2 Manual Shutdown Event Tree

MSIV CLOSURE AND LOSS OF FEEDWATER	REACTOR SUBCRITICAL	S/R VALVES OPEN	S/R VALVES RECLOSE	FEEDWATER RECOVERED	HPCT OR RECIC AVAILABLE	TIMELY ADS ACT.	LP ECCS AVAILABLE	RHR & MHRSM OR PCS	SEQUENCE DESIGNATOR	ESTIMATED SEQUENCE PROBABILITY	GENERALIZED CLASS OF POSTULATED DEGRADED CORE CONDITION
T_f	C	M	P	Q	U	X	V	M	T_f^*	OK	--
									T_f^{M**}	Transfer (2.8×10^{-6})	Fig. 3.4.13
									T_f^Q	OK	--
									$T_f^{QM(Q)**}$	Transfer (7.8×10^{-6})	Fig. 3.4.13
									T_f^{QU*}	OK	--
									$T_f^{QM(Q)**}$	Transfer (Negligible)	Fig. 3.4.13
									T_f^{QUV}	1.1×10^{-6}	Class I
									T_f^{QUX}	4.4×10^{-6}	Class I
									T_f^{P*}	OK	--
									$T_f^{PM(P)**}$	Transfer (1.5×10^{-6})	Fig. 3.4.13
									T_f^{PQ*}	OK	--
									$T_f^{PQM(PQ)**}$	Transfer (4.3×10^{-7})	Fig. 3.4.13
1.78	C	M	P	Q	U	X	V	M	T_f^{PQU*}	Transfer (2.4×10^{-9})	Fig. 3.4.13
									$T_f^{PQM(PQ)**}$	Transfer (1.1×10^{-8})	Class I
									T_f^{PQUX}	4.4×10^{-8}	Class I
									T_f^{M*}	Transfer (1.78×10^{-6})	Fig. 3.4.6a
									T_f^{C**}	5.3×10^{-5}	See Fig. 3.4.9
									3×10^{-5}		
10^{-6}	C	M	P	Q	U	X	V	M	T_f^*	OK	--
									T_f^{M**}	Transfer (2.8×10^{-6})	Fig. 3.4.13
									T_f^Q	OK	--
									$T_f^{QM(Q)**}$	Transfer (7.8×10^{-6})	Fig. 3.4.13
									T_f^{QU*}	OK	--
									$T_f^{QM(Q)**}$	Transfer (Negligible)	Fig. 3.4.13
									T_f^{QUV}	1.1×10^{-6}	Class I
									T_f^{QUX}	4.4×10^{-6}	Class I
									T_f^{P*}	OK	--
									$T_f^{PM(P)**}$	Transfer (1.5×10^{-6})	Fig. 3.4.13
									T_f^{PQ*}	OK	--
									$T_f^{PQM(PQ)**}$	Transfer (4.3×10^{-7})	Fig. 3.4.13
10^{-2}	C	M	P	Q	U	X	V	M	T_f^{PQU*}	Transfer (2.4×10^{-9})	Fig. 3.4.13
									$T_f^{PQM(PQ)**}$	Transfer (1.1×10^{-8})	Class I
									T_f^{PQUX}	4.4×10^{-8}	Class I
									T_f^{M*}	Transfer (1.78×10^{-6})	Fig. 3.4.6a
									T_f^{C**}	5.3×10^{-5}	See Fig. 3.4.9
									3×10^{-5}		

*Not core melt sequence
 **AIMS 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



*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 = 1.08 x 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.

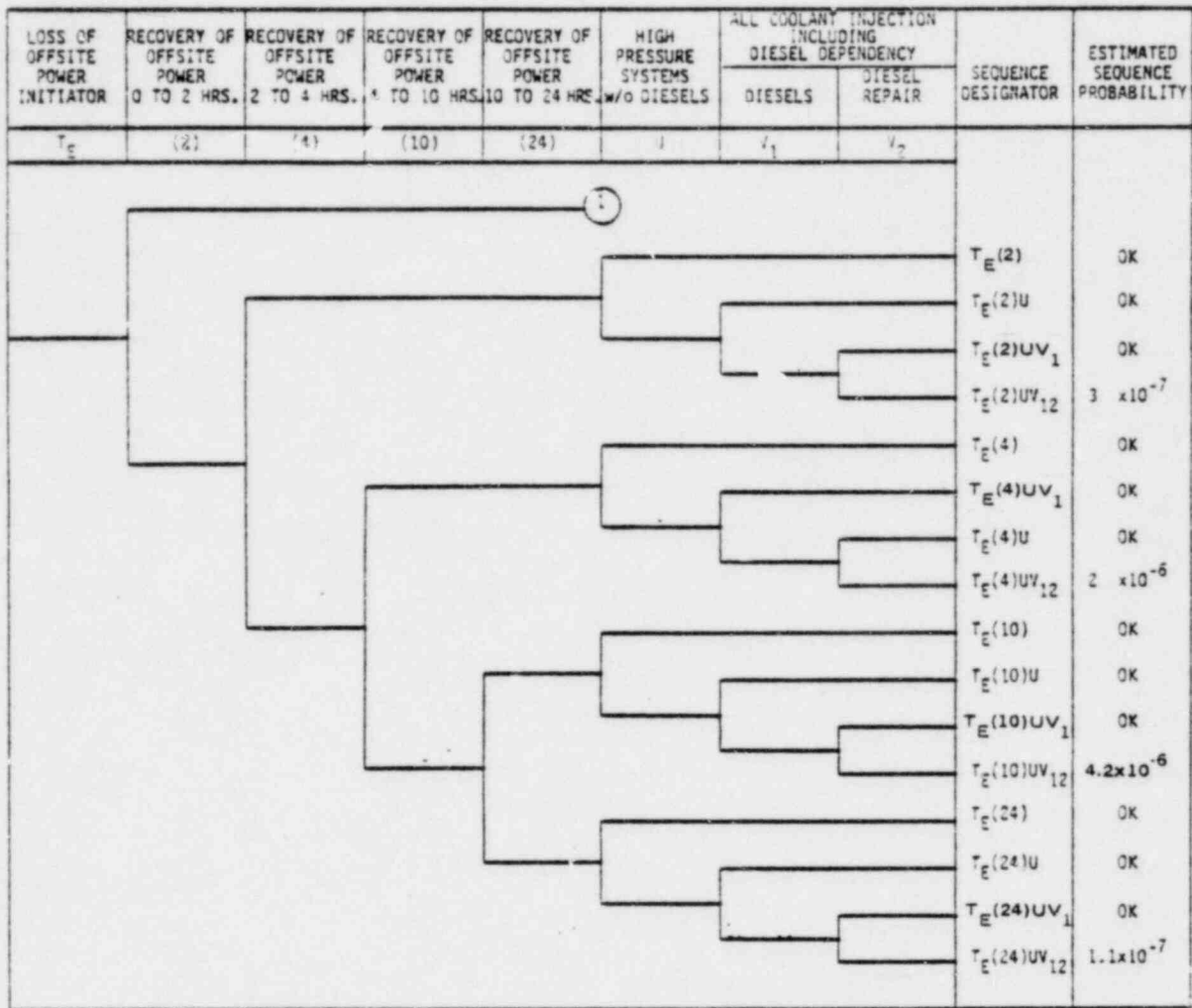
The transient event tree for loss of offsite power describes the interaction of systems and their response for various time periods ranging from 2 to 6 hours following a loss of offsite power. System AC power requirements are time dependent, so failure rates vary with time.

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.b
- Containment Heat Removal, Figure 3.4.4.c.

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

C -- Reactor Subcritical. 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.



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 time 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**†	FAILURE OF HIGH PRESSURE SYSTEMS U	FAILURE OF 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}	.56	3×10^{-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 failure of ventilation of HPCI rooms coupled with the probability of failure of operators to establish a natural circulation ventilation path for these rooms.

**Conditional probability of failure 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 for 30 minutes, 2 hours, 4 hours, and 10 hours.

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 (Figure 3.4.4c) for a 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 20 hours. The frequency of loss of offsite power for greater than 20 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.

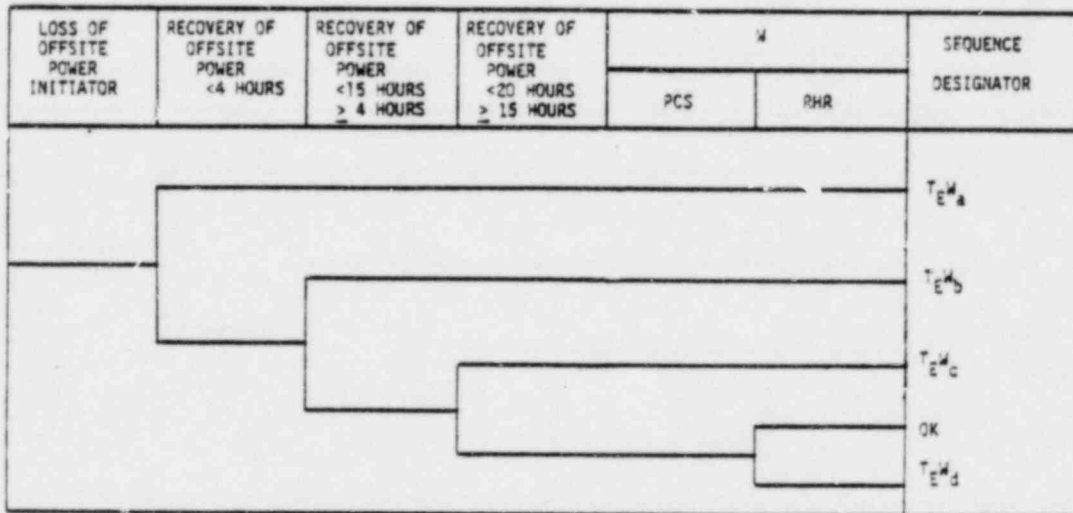
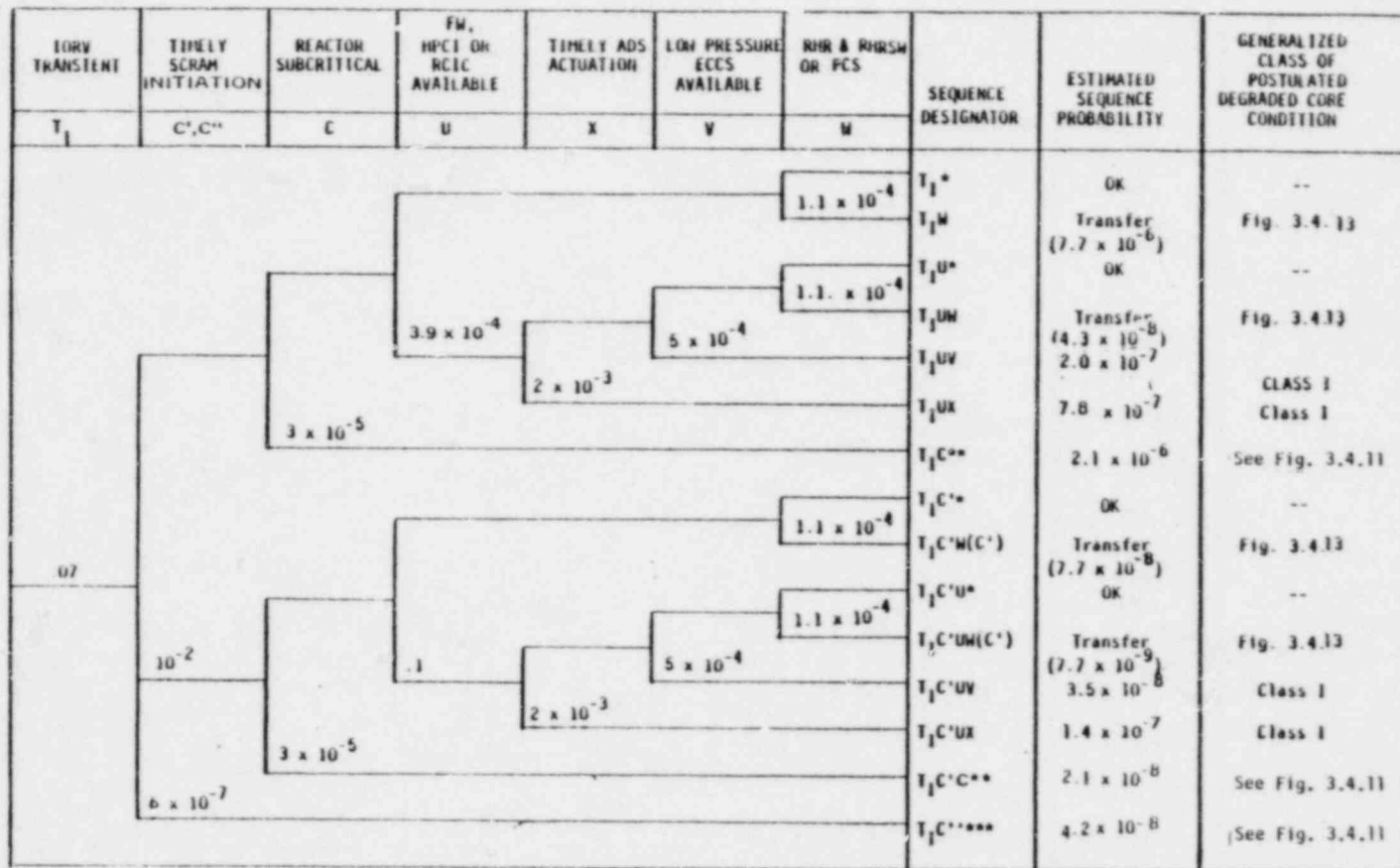


Figure 3.4.4c Time Phased Event Tree for Calculating Containment Heat Removal Capability Following a Loss of Offsite Power

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

**ATHS 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.

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 a stuck-open relief valve (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:

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.

C' -- Timely Scram. 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 exchangers to be operational.

C". Failure to scram (either manually or automatically from high drywell pressure) before the suppression pool reached 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 -- FW, 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°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 functionality*, has also been removed from the event tree, since there was no identified physical basis for this event.

The medium LOCA and small LOCA event trees (see Figures 3.4.6b 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.

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. 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 S₁ - 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 ft² for a liquid line, and between 0.016 and 0.08 ft² 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 is assumed to be unavailable for coolant injection.

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

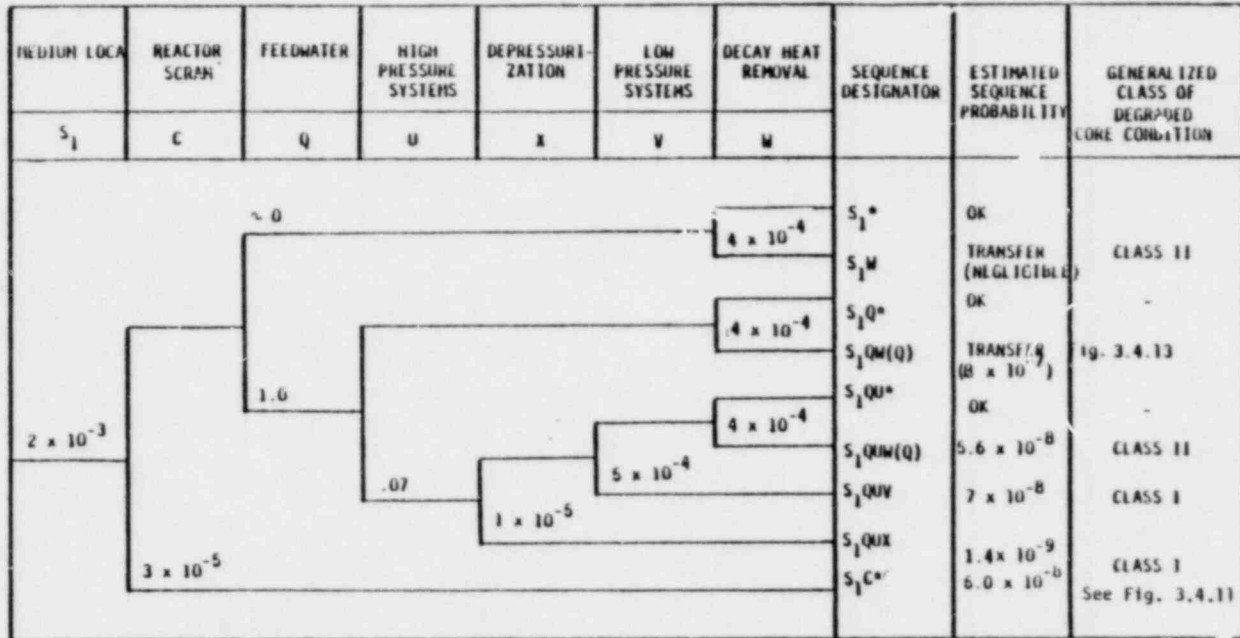
Event U - High Pressure Systems: A functional level fault tree depicting the failure of U for a medium LOCA is simply a failure of HPCI.

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.

3-4-5



*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 is.

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 3.4.6b, 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 was constructed, using the same requirements 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.

3-47

SMALL LOCA	REACTOR SHUTDOWN	HIGH PRESSURE SYSTEM	DEPRESSURIZATION	LOW PRESSURE SYSTEM	DECAY HEAT REMOVAL	SEQUENCE DESIGNATOR	ESTIMATED SEQUENCE PROBABILITY	GENERALIZED CLASS OF POSTULATED DEGRADED CORE CONDITION
S ₂	C	U	X	V	W			
						S ₂	OK	--
						S ₂ M	Transfer (2×10^{-9})	Fig. 3.4.13
						S ₂ U*	OK	--
						S ₂ UW	Transfer 1.6×10^{-12}	Fig. 3.4.13
						S ₂ UV	4.0×10^{-9}	Class I
						S ₂ UX	1.6×10^{-8}	Class I
						SC**	7×10^{-7}	See Fig. 3.4.11

*Not a core melt sequence

**Treated in ATIS trees

Figure 3.4.6c. Limerick Small LOCA Event Tree (S₂)

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:

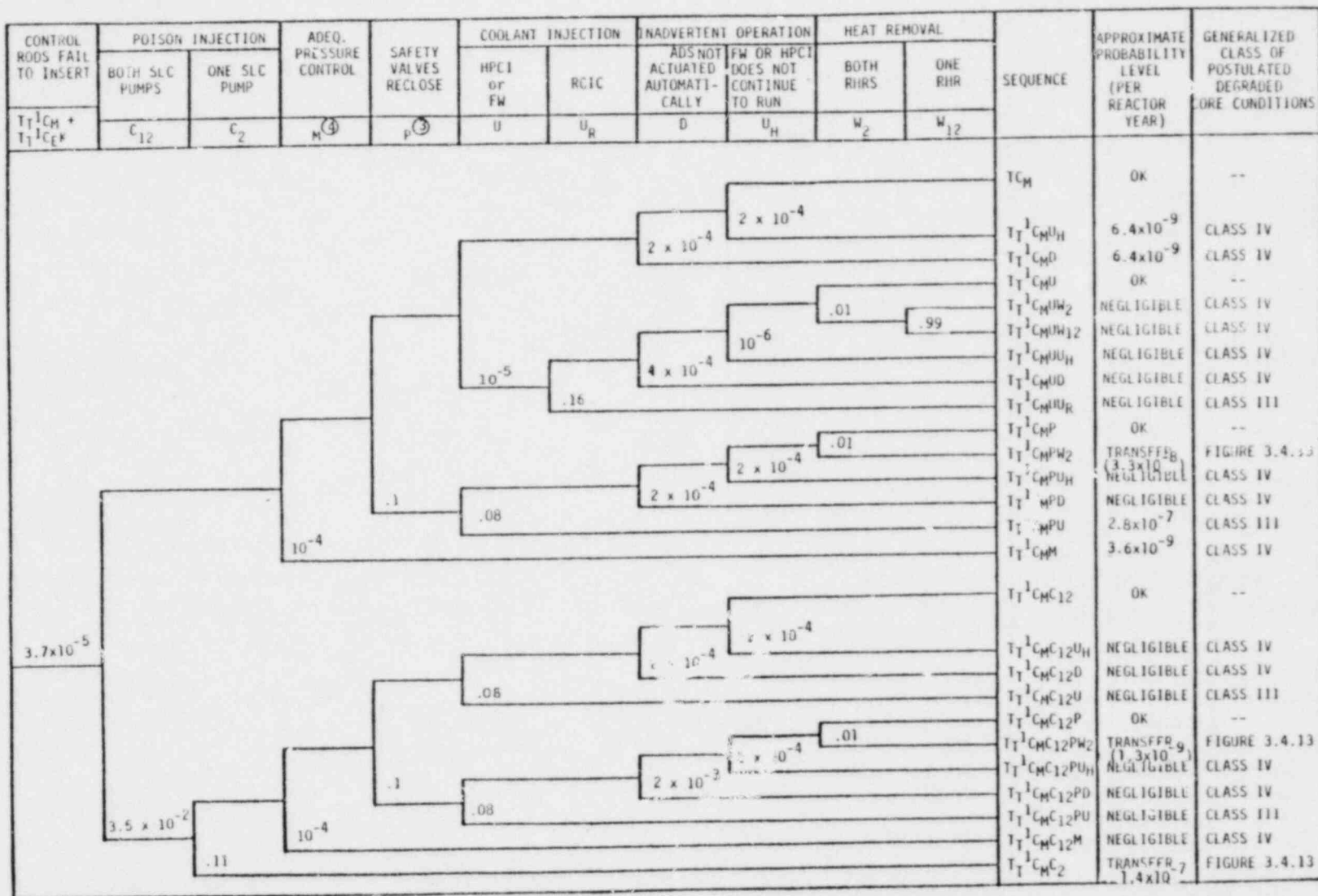
C_M -- The mechanical redundancy of the control rod drive mechanisms makes the common-mode failure of multiple adjacent control rods unlikely.

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.

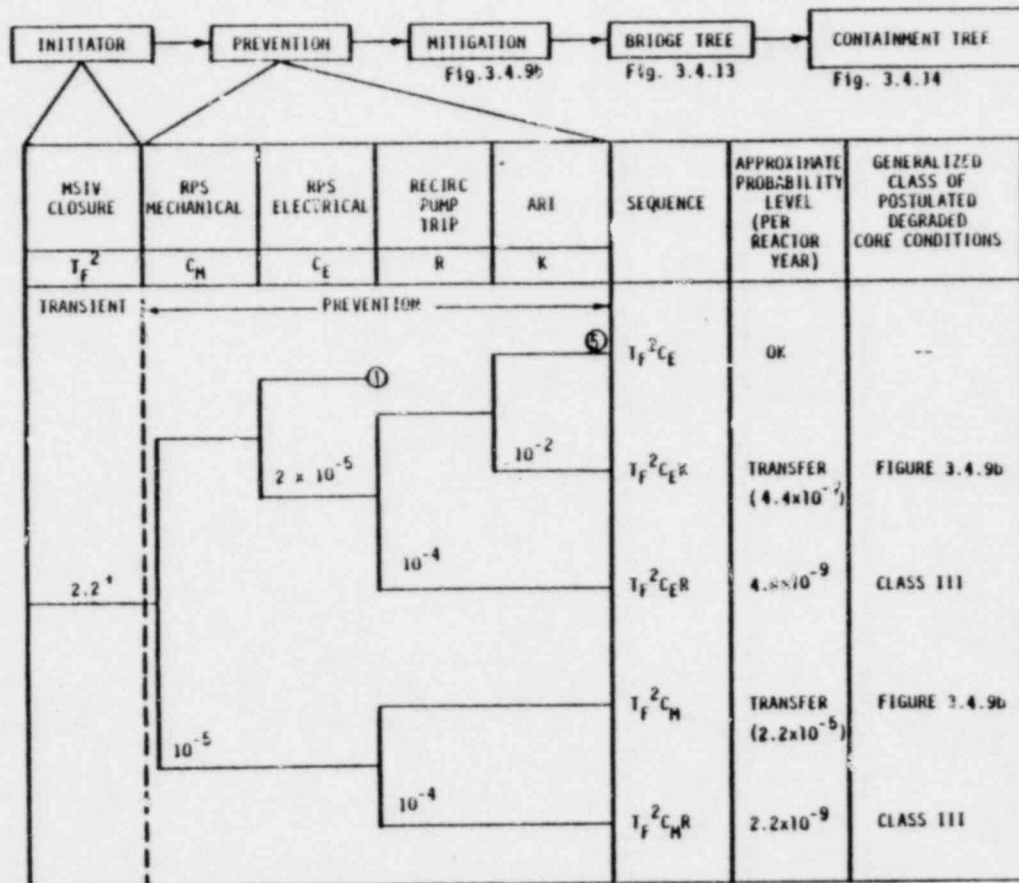
K -- Alternate Rod Insertion (ARI) incorporates a number of changes including additional sensors, additional logic, and additional solenoid valves on each mechanism to provide added assurance that the postulated electrical failures will not prevent control rod insertion.

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:



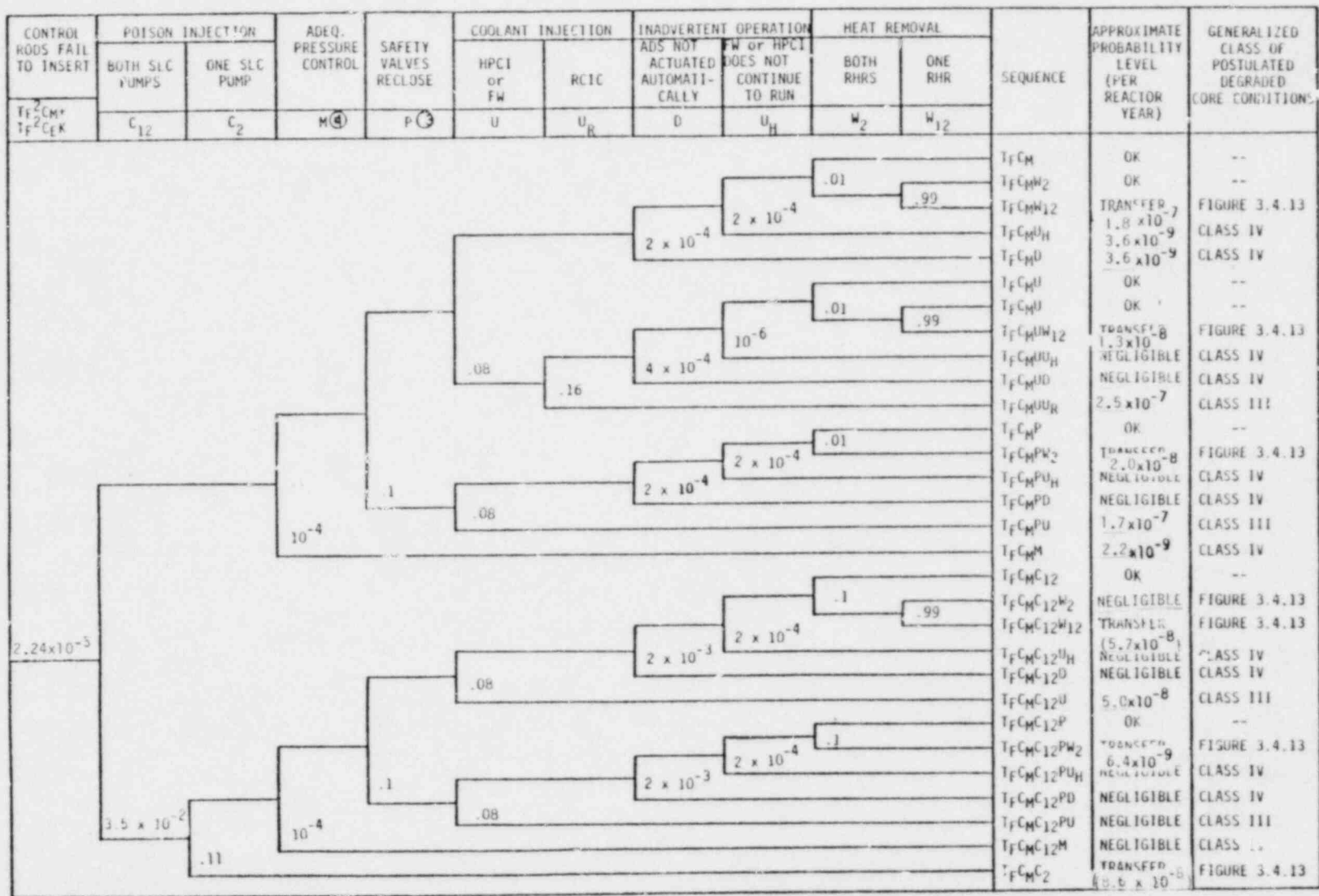
All Notes are in Table 3.4.2a.

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



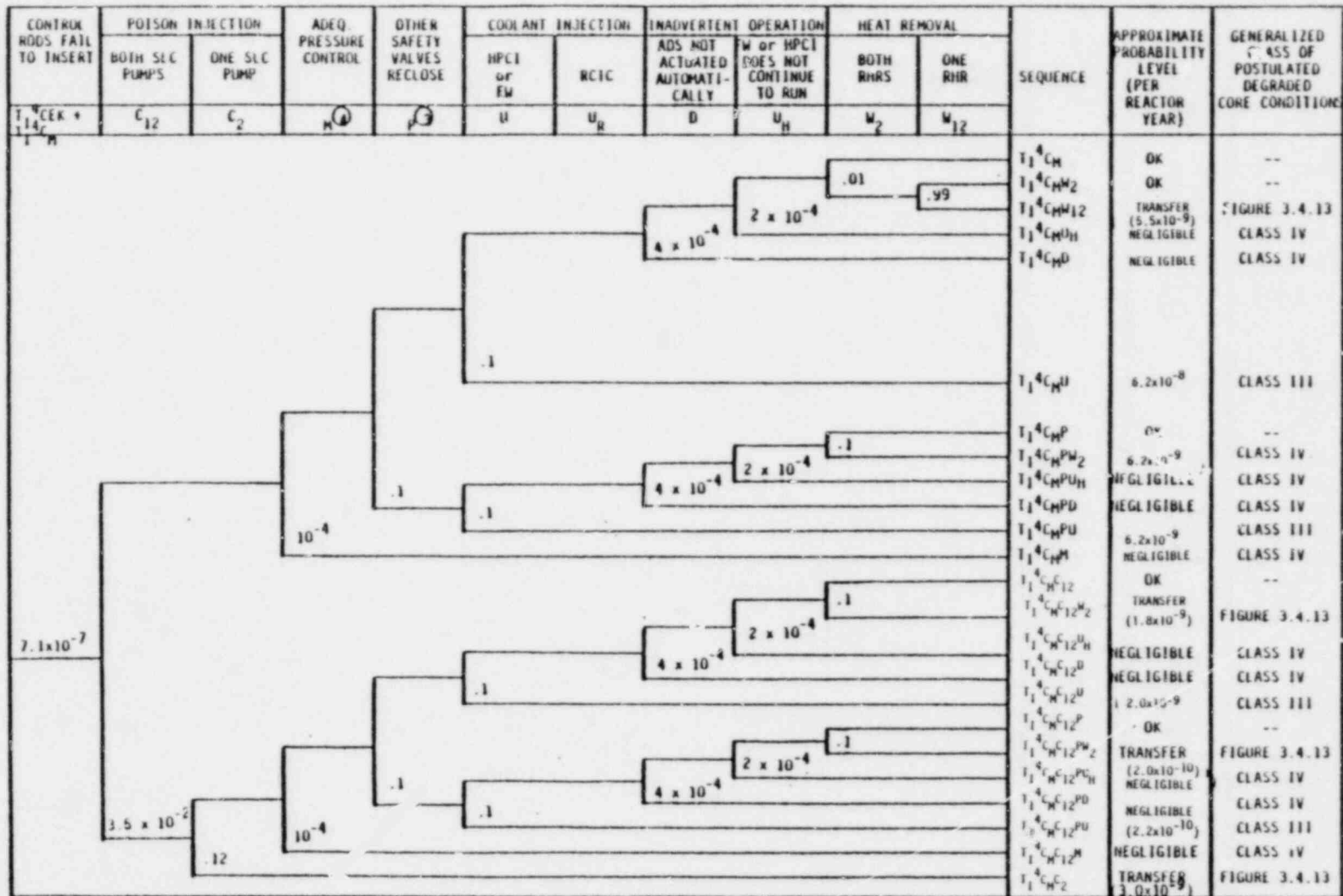
All notes are in Table 3.4.2g.
 *This Initiator is the sum of Initiators considered in Figure 3.4.3 and those transferred from Figure 3.4.7.

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



All notes are in table 3.4.2a

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

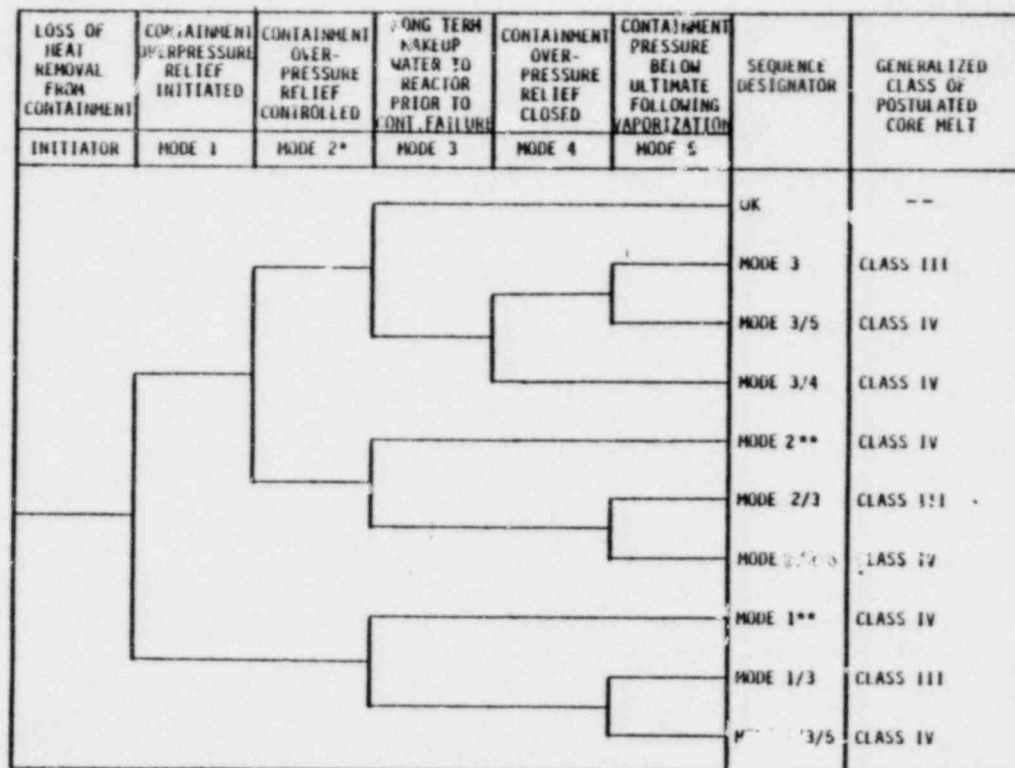
For the Limerick analysis, containment heat removal is successful with RHR and PCS unavailable if COR successfully operates. The heat would be removed from containment by steam passing from the reactor through the safety relief valves, through the suppression pool, through the drywell and directly out 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 it is presented below.

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

TW -- Initiating Event. 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).

Event Mode 1 -- Failure of Containment Overpressure Relief. 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 TI sequences described in WASH-1400. The overpressure relief procedure is assumed to require:

- Indication of high containment pressure
- Indication of RPV water level L1 or above
- 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 was constructed for 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

*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 TQIV)	2-10 Hours
Mode 3/4*	COR Fails Open and Coolant Makeup Fails	Core Melt (Direct Release)	2-10 Hours
Mode 5	Long Term Makeup 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. A functional fault tree for long term coolant makeup to the reactor was constructed.

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.

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-C₂ 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 mech-

analysis (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-C₂ 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-C₂ 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-C₁₂ 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.

*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 structure 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 enclosure 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

(1) CORE MELT	NO RAPID OVERPRESSURE IN VESSEL CONTAINMENT	(2) NO H ₂ INDUCED FAILURE	NO H ₂ DETONATION INDUCED FAILURE	NO CONTAINMENT LEAK SUFF. TO PREVENT OVERPRESSURE	NO CONTAINMENT OVERPRESSURE	NO CONTAINMENT OVERPRESSURE (NETWELL)	NO SUPPRESSION POOL FAILURE (NETWELL)	NO CONTAINMENT LEAK (LEAK)	(5) NO SPTS FAILURE (4)	SEQUENCE	PROBABILITY OF CFM	QUALITATIVE CHARACTERISTICS OF CONTAINMENT FAILURE MODE
CM	α	μ	μ'	δ	γ	γ'/γ	γ''/γ	ζ/δ	ϵ			
										OK	.0005	---
				.5						γ	.222	OVERPRESSURE
					0.999			.1		γ''	.025	SUPPRESSION POOL FAILURE
						.5				γ'	.247	OVERPRESSURE NETWELL
								.5		δ	.222	SMALL LEAK
									.1	$\delta\epsilon$.025	SMALL LEAK, SPTS FAILURE
									.5	$\delta\zeta$.198	LARGE LEAK, SPTS ADEQUATE
									.2	$\delta\zeta\epsilon$.055	LARGE LEAK, SPTS INADEQUATE
										μ	.009	OVERPRESSURE
									.01	$\mu\mu'$.031	INSTANTANEOUS OVERPRESSURE
									.001	β	.001	ENERGETIC OVERPRESSURE
									.001	α	.001	ENERGETIC OVERPRESSURE

(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 H₂ 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.

(5) VALUES SHOWN HERE ARE FOR CLASS I, II, AND III. SEE SECTION 3.5.4 FOR DISCUSSION AND VALUES FOR CLASS IV.

FIGURE 3.4.14 CONTAINMENT EVENT TREE FOR THE MARK II CONTAINMENT

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. Quantification of the tree is discussed in Section 3.5.4.

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.

α -- In-Vessel Steam Explosion. Prior to vessel melt-through, the molten core may drop into water in the bottom of the reactor vessel causing a steam explosion of sufficient energy to cause vessel rupture. The probability of this is believed to be extremely remote for a BWR, but was assigned a probability of 0.001 for this analysis.

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 0.001. See Appendix H for a more detailed discussion of this phenomenon.

μ -- Hydrogen Combustion. 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.

δ -- Containment Leakage. Above containment design pressure, sufficient leakage may occur to stabilize pressure.

γ -- Containment Overpressure. Given that no containment leakage occurs, containment overpressure following a core melt is assumed. 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 could, 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''/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.

ε/δ -- 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.

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

TABLE 3.5.2
SUMMARY OF GENERAL TYPES OF ACCIDENT SEQUENCES*

	CLASS I RELATIVELY RAPID CORE MELT WITH INTACT CONTAINMENT	CLASS II RELATIVELY SLOW CORE MELT WITH FAILED CONTAINMENT	CLASS III RELATIVELY RAPID CORE MELT WITH INCIPIENT CONTAINMENT FAILURE	CLASS IV RELATIVELY RAPID CORE MELT WITH CONTAINMENT FAILURE
PRIME EXAMPLE	TQV	TW (MODE 1)**	TC _{MC2} (ATWS)	TC _{MC2} ^U (MODE 1)
	T _T QUV	T _T W (MODE 1)	T _T ¹ C _M PU	T _T ¹ C _M PW ₂ (MODE 1) (MODE 3/4)
	T _T QUX	T _T PW (MODE 1)	T _T ¹ C _M PW ₂ (MODE 1/3)	T _F ² C _M W (MODE 3/4) (MODE 1)
	T _M QUV	T _M W (MODE 1)	T _T ¹ C _{MC2} PW ₂ (MODE 1/3)	T _E ³ C _M W (MODE 1) (MODE 3/4)
	T _M QUX	T _F QW (Q) (MODE 1)	T _T ¹ C _{MC2} (MODE 1/3)	T _I ⁴ C _M W (MODE 1) (MODE 3/4)
	T _F QUV	T _F W (MODE 1)	T _F ² C _{MR} +T _F ² C _{ER}	T _I C* (MODE 1) (MODE 3/4)
	T _F QUX	T _F PW (MODE 1)	T _F ² C _M UUR	T _T ¹ C _M ¹ 2PW ₂ (MODE 1) (MODE 3/4)
DOMINANT ACCIDENT SEQUENCES	T _E UV	T _E W _d (MODE 1)	T _F ² C _M PU	T _F ² C _M ¹ 2W ₂ (MODE 1) (MODE 3/4)
	T _E UX	T _E PW _d (MODE 1)	T _F ² C _{MC12} U	T _E ³ C _M ¹ 2W ₂ (MODE 1) (MODE 3/4)
	T _I QUV	T _I W (MODE 1)	T _F ² C _M W (MODE 1/3)	T _I ¹ C _M ¹ 2W ₂ (MODE 1) (MODE 3/4)
	T _I QUX	T _I C'W (MODE 1)	T _F ² C _M PW ₂ (MODE 1/3)	T _T ¹ C _{MC2} (MODE 1) (MODE 1/3/5)
	S ₁ QUV	T _I C* (MODE 1)	T _F ² C _{MC2} (MODE 1/3)	T _F ² C _{MC2} (MODE 1) (MODE 1/3/5)
	S ₁ QUX	AJ+	T _E ³ C _M UUR	T _E ³ C _{MC2} (MODE 1) (MODE 1/3/5)
	S ₂ QUV	S ₁ W (MODE 1)	T _E ³ C _M PU	T _I ⁴ C _{MC2} (MODE 1) (MODE 1/3/5)
	S ₂ QUX	S ₂ W (MODE 1)	T _E ³ C _{MC12} U	T _T ¹ C _M UH/D
		S ₁ QW (Q)	T _E ³ C _{MC2}	T _T ¹ C _M
			T _E ³ C _M W (MODE 1/3)	T _T ¹ C _{ER} +T _T ¹ C _{MR}
			T _E ³ C _{MC2} W (MODE 1/3)	T _F ² C _M UH/D
			T _I ⁴ C _M W (MODE 1/3)	T _F ² C _M M
			T _I C'' (MODE 1/3)	T _E ³ C _M UH/D
			T _I ⁴ C _{MC2} W (MODE 1/3)	T _E ³ C _M M
			T _I ⁴ C _{MC2} (MODE 1/3)	T _I ⁴ C _M UH/D
			AE/AI	T _I ⁴ C _M M

*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						
	a .001	B, u' .002	Y', u .258	Y .222	Y" .025	CC, Cc .078	C, C .42
T ₁ QUV	2.2x10 ⁻¹⁰	4.4x10 ⁻¹⁰	5.7x10 ⁻⁸	4.9x10 ⁻⁸	5.5x10 ⁻⁹	1.7x10 ⁻⁸	9.2x10 ⁻⁸
T ₂ QUX	8.9x10 ⁻¹⁰	1.8x10 ⁻⁹	2.3x10 ⁻⁷	2.0x10 ⁻⁷	2.2x10 ⁻⁸	6.9x10 ⁻⁸	3.7x10 ⁻⁷
T ₃ QUV	7.2x10 ⁻¹¹	1.4x10 ⁻¹⁰	1.9x10 ⁻⁸	1.6x10 ⁻⁸	1.8x10 ⁻⁹	5.6x10 ⁻⁹	3.0x10 ⁻⁸
T ₄ QUX	2.9x10 ⁻¹⁰	5.8x10 ⁻¹⁰	7.5x10 ⁻⁸	6.4x10 ⁻⁸	7.3x10 ⁻⁹	2.3x10 ⁻⁸	1.2x10 ⁻⁷
T ₅ QUV	1.1x10 ⁻⁹	2.2x10 ⁻⁹	2.8x10 ⁻⁷	2.4x10 ⁻⁷	2.8x10 ⁻⁸	8.6x10 ⁻⁸	4.6x10 ⁻⁷
T ₆ QUX	4.4x10 ⁻⁹	8.8x10 ⁻⁹	1.1x10 ⁻⁶	9.8x10 ⁻⁷	1.7x10 ⁻⁷	3.4x10 ⁻⁷	1.8x10 ⁻⁶
T ₇ UV	4.0x10 ⁻⁹	8.0x10 ⁻⁹	1.0x10 ⁻⁶	8.9x10 ⁻⁷	1.0x10 ⁻⁷	2.0x10 ^{-6*}	NA*
T ₈ UX	8.4x10 ⁻¹⁰	1.7x10 ⁻⁹	2.2x10 ⁻⁷	1.9x10 ⁻⁷	2.1x10 ⁻⁸	4.2x10 ^{-7*}	NA*
T ₉ QUV	2.0x10 ⁻¹⁰	4.0x10 ⁻¹⁰	5.2x10 ⁻⁸	4.4x10 ⁻⁸	5.0x10 ⁻⁹	1.6x10 ⁻⁸	8.4x10 ⁻⁸
T ₁₀ QUX	7.8x10 ⁻¹⁰	1.6x10 ⁻⁹	2.0x10 ⁻⁷	1.7x10 ⁻⁷	2.0x10 ⁻⁸	6.1x10 ⁻⁸	3.3x10 ⁻⁷
S ₁ QUV	7 x10 ⁻¹¹	1.4x10 ⁻¹⁰	1.8x10 ⁻⁸	1.6x10 ⁻⁸	1.8x10 ⁻⁹	5.5x10 ⁻⁹	2.9x10 ⁻⁸
S ₁ QUX	1.4x10 ⁻¹²	2.8x10 ⁻¹²	3.6x10 ⁻¹⁰	3.1x10 ⁻¹⁰	3.5x10 ⁻¹¹	1.1x10 ⁻¹⁰	5.9x10 ⁻¹⁰
S ₂ QUV	4 x10 ⁻¹²	8 x10 ⁻¹²	1.0x10 ⁻⁹	0.9x10 ⁻¹⁰	1.0x10 ⁻¹⁰	3.1x10 ⁻¹⁰	1.7x10 ⁻⁹
S ₂ QUX	1.6x10 ⁻¹¹	3.2x10 ⁻¹¹	4.1x10 ⁻⁹	3.6x10 ⁻⁹	4.0x10 ⁻¹⁰	1.2x10 ⁻⁹	6.7x10 ⁻⁹
APPROXIMATE TOTAL PROBABILITY FOR CLASS I SEQUENCES	1.3x10 ⁻⁸	2.6x10 ⁻⁸	3.4x10 ⁻⁶	2.9x10 ⁻⁶	3.2x10 ⁻⁷	3.0x10 ^{-6*}	3.4x10 ^{-6*}

*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 slightly overestimates the releases from these sequences. All loss of offsite power accident sequences involving containment leakage are included in the column for failed SGTS (cc, Cc). The total conditional failure probability for these sequences is 0.5 (0.078 + 0.42).

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						
	α .001	β, β' .002	γ, γ' .256	γ .222	γ'' .025	δ, δ' .078	δ, δ' .42
T_{TW} (mode 1)	4.4×10^{-11}	8.8×10^{-11}	1.1×10^{-8}	9.8×10^{-9}	1.1×10^{-9}	3.4×10^{-9}	1.8×10^{-8}
T_{TPW} (mode 1)	8.8×10^{-11}	1.8×10^{-10}	2.3×10^{-8}	2.0×10^{-8}	2.2×10^{-9}	6.9×10^{-9}	3.7×10^{-8}
T_{NH} (mode 1)	3.6×10^{-11}	7.2×10^{-11}	9.3×10^{-9}	8.0×10^{-9}	9.0×10^{-10}	2.8×10^{-9}	1.5×10^{-8}
T_{FQH} (Q) (mode 1)	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_{FH} (mode 1)	5.6×10^{-11}	1.1×10^{-10}	1.4×10^{-8}	1.2×10^{-8}	1.4×10^{-9}	4.4×10^{-9}	2.4×10^{-8}
T_{FPW} (mode 1)	3.9×10^{-11}	7.8×10^{-11}	1.0×10^{-8}	8.7×10^{-9}	9.8×10^{-10}	3.0×10^{-9}	1.6×10^{-8}
T_{EWd} (mode 1)	1.2×10^{-11}	2.4×10^{-11}	3.1×10^{-9}	2.7×10^{-9}	3.0×10^{-10}	6.0×10^{-9} *	NA*
T_{EPWd} (mode 1)	1.2×10^{-12}	2.4×10^{-12}	3.1×10^{-10}	2.7×10^{-10}	3.0×10^{-11}	6.0×10^{-10} *	NA*
T_{LW} (mode 1)	1.5×10^{-10}	3.0×10^{-10}	3.9×10^{-8}	3.3×10^{-8}	3.8×10^{-9}	1.2×10^{-8}	6.3×10^{-8}
T_{LCW} (mode 1)	1.5×10^{-12}	3.0×10^{-12}	3.9×10^{-10}	3.3×10^{-10}	3.8×10^{-11}	1.2×10^{-10}	6.3×10^{-10}
T_{LC}^* (mode 1)	4×10^{-13}	8.0×10^{-13}	1.0×10^{-10}	8.9×10^{-11}	1.0×10^{-11}	3.1×10^{-11}	1.7×10^{-10}
S_1	1.6×10^{-10}	3.2×10^{-10}	4.1×10^{-8}	3.6×10^{-8}	4.0×10^{-9}	1.2×10^{-8}	6.7×10^{-8}
S_{1W} (mode 1)	1.6×10^{-11}	3.2×10^{-11}	4.1×10^{-9}	3.6×10^{-9}	4.0×10^{-10}	1.2×10^{-9}	6.7×10^{-9}
S_{2W} (mode 1)	negligible	negligible	negligible	negligible	negligible	negligible	negligible
S_{1QW} (Q)	5.6×10^{-11}	1.1×10^{-10}	1.4×10^{-8}	1.2×10^{-8}	1.4×10^{-9}	4.4×10^{-9}	2.4×10^{-8}
APPROXIMATE TOTAL PROBABILITY FOR CLASS II SEQUENCES	8.6×10^{-10}	1.7×10^{-9}	2.2×10^{-7}	1.9×10^{-7}	2.2×10^{-8}	7.2×10^{-8} *	3.5×10^{-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 slightly overestimates the releases from these sequences. All loss of offsite power accident sequences involving containment leakage are included in the column for failed SGTS (δ, δ'). The total conditional failure probability for these sequences is 0.5 (0.078 + 0.42)

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 .256	γ .222	γ'' .025	$\zeta, \delta e$.078	ζ, δ .42
$T_{C1}^1 C_{PH}^1$	2.98×10^{-10}	5.96×10^{-10}	7.69×10^{-8}	6.62×10^{-8}	7.49×10^{-9}	2.32×10^{-8}	1.25×10^{-7}
$T_{C1}^1 C_{PH}^1$ (MODE 1/3)	1.1×10^{-11}	2.2×10^{-11}	2.83×10^{-9}	2.44×10^{-9}	2.75×10^{-10}	8.58×10^{-10}	4.62×10^{-9}
$T_{C1}^1 C_{PH}^1 C_{PH}^1$ (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^{-9}
$T_{C1}^1 C_{PH}^1$ (MODE 1/3)	7.8×10^{-11}	1.56×10^{-10}	2.01×10^{-8}	1.73×10^{-8}	1.95×10^{-9}	6.08×10^{-9}	3.28×10^{-8}
$T_F^2 C_{PH}^2 + T_F^2 C_{ER}^2$	6.60×10^{-12}	1.32×10^{-11}	1.70×10^{-9}	1.47×10^{-9}	1.65×10^{-10}	5.15×10^{-10}	2.77×10^{-9}
$T_F^2 C_{PH}^2$	2.86×10^{-10}	5.72×10^{-10}	7.38×10^{-8}	6.35×10^{-8}	7.15×10^{-9}	2.23×10^{-8}	1.20×10^{-7}
$T_F^2 C_{PH}^2$	2.50×10^{-10}	5.00×10^{-10}	6.45×10^{-8}	5.55×10^{-8}	6.25×10^{-9}	1.95×10^{-8}	1.05×10^{-7}
$T_F^2 C_{PH}^2 C_{PH}^2$	6.30×10^{-11}	1.26×10^{-10}	1.63×10^{-8}	1.40×10^{-8}	1.57×10^{-9}	4.91×10^{-9}	2.65×10^{-8}
$T_F^2 C_{PH}^2$ (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}
$T_F^2 C_{PH}^2$ (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}
$T_F^2 C_{PH}^2$ (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}
$T_E^3 C_{PH}^3$	2.70×10^{-11}	5.40×10^{-11}	6.97×10^{-9}	5.99×10^{-9}	6.75×10^{-10}	1.34×10^{-8}	NA*
$T_E^3 C_{PH}^3$	4.30×10^{-12}	8.60×10^{-12}	1.11×10^{-9}	9.55×10^{-10}	1.08×10^{-10}	2.14×10^{-9}	NA*
$T_E^3 C_{PH}^3 C_{PH}^3$	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*
$T_E^3 C_{PH}^3$	4.30×10^{-12}	8.60×10^{-12}	1.11×10^{-9}	9.55×10^{-10}	1.08×10^{-10}	2.14×10^{-9}	NA*
$T_E^3 C_{PH}^3$ (MODE 1/3)	7.5×10^{-12}	1.5×10^{-11}	1.94×10^{-9}	1.67×10^{-9}	1.88×10^{-10}	3.74×10^{-9}	NA*
$T_E^3 C_{PH}^3$ (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*
$T_I^4 C_{PH}^4$ (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}	8.32×10^{-10}
$T_I^4 C_{PH}^4$ (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}
$T_I^4 C_{PH}^4$ (MODE 1/3)	7.3×10^{-13}	1.5×10^{-12}	1.88×10^{-10}	1.62×10^{-10}	1.82×10^{-11}	3.69×10^{-11}	3.07×10^{-10}
$T_I^4 C_{PH}^4$ (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^{-8}	4.44×10^{-8}	5.00×10^{-9}	1.56×10^{-8}	8.40×10^{-8}
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} *

*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 slightly overestimates the releases from these sequences. All loss of offsite power accident sequences involving containment leakage are included in the column for failed SGTS ($\zeta, \delta e$). The total conditional failure probability for these sequences is 0.5 (0.078 + 0.42)

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						
	a .001	B,u' .002	Y',u .503	Y .443	Y° .05	Cc,δc .0002	C,δ .0002
T ₁ ¹ C _M PH ₂ (MODE 1)	1.48x10 ⁻¹²	2.96x10 ⁻¹²	7.44x10 ⁻¹⁰	4.56x10 ⁻¹⁰	7.40x10 ⁻¹¹	2.96x10 ⁻¹³	2.96x10 ⁻¹³
(MODE 3/4)	4.60x10 ⁻¹³	9.20x10 ⁻¹³	2.31x10 ⁻¹⁰	2.04x10 ⁻¹⁰	2.30x10 ⁻¹¹	9.20x10 ⁻¹⁴	9.20x10 ⁻¹⁴
T _F ² C _M H (MODE 3/4)	9.00x10 ⁻¹²	1.80x10 ⁻¹¹	4.53x10 ⁻⁹	3.99x10 ⁻⁹	4.50x10 ⁻¹⁰	1.80x10 ⁻¹²	1.80x10 ⁻¹²
(MODE 1)	2.85x10 ⁻¹²	5.70x10 ⁻¹²	1.43x10 ⁻⁹	1.26x10 ⁻⁹	1.42x10 ⁻¹⁰	5.70x10 ⁻¹³	5.70x10 ⁻¹³
T _E ³ C _M H (MODE 1)	1.00x10 ⁻¹²	2.00x10 ⁻¹²	5.03x10 ⁻¹⁰	4.43x10 ⁻¹⁰	5.00x10 ⁻¹¹	1.00x10 ⁻¹³	NA*
(MODE 3/4)	9.50x10 ⁻¹³	1.90x10 ⁻¹²	4.78x10 ⁻¹⁰	4.21x10 ⁻¹⁰	4.75x10 ⁻¹¹	3.80x10 ⁻¹³	NA*
T _I ⁴ C _M H (MODE 1)	2.80x10 ⁻¹³	5.60x10 ⁻¹³	1.41x10 ⁻¹⁰	1.24x10 ⁻¹⁰	1.40x10 ⁻¹¹	5.60x10 ⁻¹⁴	5.60x10 ⁻¹⁴
(MODE 3/4)	9.00x10 ⁻¹⁵	1.80x10 ⁻¹⁴	4.53x10 ⁻¹²	3.99x10 ⁻¹²	4.50x10 ⁻¹³	1.80x10 ⁻¹⁵	1.80x10 ⁻¹⁵
T _I ¹ C ^u (MODE 1)	1.70x10 ⁻¹²	3.40x10 ⁻¹²	8.55x10 ⁻¹⁰	7.33x10 ⁻¹⁰	8.50x10 ⁻¹¹	3.40x10 ⁻¹³	3.40x10 ⁻¹³
(MODE 3/4)	5.50x10 ⁻¹³	1.10x10 ⁻¹²	2.77x10 ⁻¹⁰	2.44x10 ⁻¹⁰	2.75x10 ⁻¹¹	1.10x10 ⁻¹³	1.10x10 ⁻¹³
T _F ² C _M PH ₂ (MODE 1)	5.10x10 ⁻¹³	1.02x10 ⁻¹²	2.57x10 ⁻¹⁰	2.26x10 ⁻¹⁰	2.55x10 ⁻¹¹	1.02x10 ⁻¹³	1.02x10 ⁻¹³
(MODE 3/4)	1.62x10 ⁻¹³	3.24x10 ⁻¹³	8.15x10 ⁻¹¹	7.18x10 ⁻¹¹	8.10x10 ⁻¹²	3.24x10 ⁻¹⁴	3.24x10 ⁻¹⁴
T _F ² C _M C ₂ H ₂ (MODE 1)	3.00x10 ⁻¹²	6.00x10 ⁻¹²	1.51x10 ⁻⁹	1.33x10 ⁻⁹	1.50x10 ⁻¹⁰	6.00x10 ⁻¹³	6.00x10 ⁻¹³
(MODE 3/4)	1.03x10 ⁻¹²	2.06x10 ⁻¹²	5.18x10 ⁻¹⁰	4.56x10 ⁻¹⁰	5.15x10 ⁻¹¹	2.06x10 ⁻¹³	2.06x10 ⁻¹³
T _E ³ C _M C ₂ H ₂ (MODE 1)	7.60x10 ⁻¹⁴	1.52x10 ⁻¹³	3.82x10 ⁻¹¹	3.37x10 ⁻¹¹	3.80x10 ⁻¹²	3.04x10 ⁻¹⁴	NA*
(MODE 3/4)	6.20x10 ⁻¹⁵	1.24x10 ⁻¹⁴	3.12x10 ⁻¹²	2.75x10 ⁻¹²	3.10x10 ⁻¹³	2.48x10 ⁻¹⁵	NA*
T _I ⁴ C _M C ₂ H ₂ (MODE 1)	1.00x10 ⁻¹³	2.00x10 ⁻¹³	5.03x10 ⁻¹¹	4.43x10 ⁻¹¹	5.00x10 ⁻¹²	2.00x10 ⁻¹⁴	2.00x10 ⁻¹⁴
(MODE 3/4)	3.30x10 ⁻¹⁴	6.60x10 ⁻¹⁴	1.66x10 ⁻¹¹	1.46x10 ⁻¹¹	1.65x10 ⁻¹²	6.60x10 ⁻¹⁵	6.60x10 ⁻¹⁵
T _I ¹ C _M C ₂ (MODE 1)	2.97x10 ⁻¹¹	5.94x10 ⁻¹¹	1.49x10 ⁻⁸	1.32x10 ⁻⁸	1.49x10 ⁻⁹	5.94x10 ⁻¹²	5.94x10 ⁻¹²
(MODE 1/3/5)	1.21x10 ⁻¹¹	2.42x10 ⁻¹¹	6.09x10 ⁻⁹	5.36x10 ⁻⁹	6.05x10 ⁻¹⁰	2.42x10 ⁻¹²	2.42x10 ⁻¹²
T _F ² C _M C ₂ (MODE 1)	1.80x10 ⁻¹¹	3.60x10 ⁻¹¹	9.05x10 ⁻⁹	7.97x10 ⁻⁹	9.00x10 ⁻¹⁰	3.60x10 ⁻¹²	3.60x10 ⁻¹²
(MODE 1/3/4)	7.30x10 ⁻¹²	1.46x10 ⁻¹¹	3.67x10 ⁻⁹	3.23x10 ⁻⁹	3.65x10 ⁻¹⁰	1.46x10 ⁻¹²	1.46x10 ⁻¹²
T _E ³ C _M C ₂ (MODE 1)	1.20x10 ⁻¹²	2.40x10 ⁻¹²	6.04x10 ⁻¹⁰	5.32x10 ⁻¹⁰	6.00x10 ⁻¹¹	4.80x10 ⁻¹³	NA*
(MODE 1/3/5)	4.80x10 ⁻¹³	9.60x10 ⁻¹³	2.41x10 ⁻¹⁰	2.13x10 ⁻¹⁰	2.40x10 ⁻¹¹	1.92x10 ⁻¹³	NA*
T _I ⁴ C _M C ₂ (MODE 1)	5.60x10 ⁻¹³	1.12x10 ⁻¹²	2.82x10 ⁻¹⁰	2.48x10 ⁻¹⁰	2.80x10 ⁻¹¹	1.12x10 ⁻¹³	1.12x10 ⁻¹³
(MODE 1/3/5)	2.20x10 ⁻¹³	4.40x10 ⁻¹³	1.11x10 ⁻¹⁰	9.75x10 ⁻¹¹	1.10x10 ⁻¹¹	4.40x10 ⁻¹³	4.40x10 ⁻¹⁴
T _I ¹ C _M H/D	1.46x10 ⁻¹¹	2.92x10 ⁻¹¹	7.34x10 ⁻⁹	6.47x10 ⁻⁹	7.30x10 ⁻¹⁰	2.92x10 ⁻¹²	2.92x10 ⁻¹²
T _I ¹ C _M H	3.90x10 ⁻¹²	7.80x10 ⁻¹²	1.96x10 ⁻⁹	1.73x10 ⁻⁹	1.95x10 ⁻¹⁰	7.80x10 ⁻¹³	7.80x10 ⁻¹³
T _I ¹ C _E R+T _I ¹ C _M R	1.05x10 ⁻¹¹	2.10x10 ⁻¹¹	5.28x10 ⁻⁹	4.65x10 ⁻⁹	5.25x10 ⁻¹⁰	2.10x10 ⁻¹²	2.10x10 ⁻¹²
T _F ² C _M H/D	9.00x10 ⁻¹²	1.80x10 ⁻¹¹	4.53x10 ⁻⁹	3.99x10 ⁻⁹	4.50x10 ⁻¹⁰	1.80x10 ⁻¹²	1.80x10 ⁻¹²
T _F ² C _M H	2.25x10 ⁻¹²	4.50x10 ⁻¹²	1.13x10 ⁻⁹	9.97x10 ⁻¹⁰	1.13x10 ⁻¹⁰	4.50x10 ⁻¹³	4.50x10 ⁻¹³
T _E ³ C _M H/D	2.20x10 ⁻¹³	4.40x10 ⁻¹³	1.11x10 ⁻¹⁰	9.75x10 ⁻¹¹	1.10x10 ⁻¹¹	8.80x10 ⁻¹⁴	NA*
T _E ³ C _M H	5.40x10 ⁻¹⁴	1.08x10 ⁻¹³	2.72x10 ⁻¹¹	2.39x10 ⁻¹¹	2.70x10 ⁻¹²	2.16x10 ⁻¹⁴	NA*
T _I ⁴ C _M H/D	4.20x10 ⁻¹³	8.40x10 ⁻¹³	2.11x10 ⁻¹⁰	1.86x10 ⁻¹⁰	2.10x10 ⁻¹¹	8.40x10 ⁻¹⁴	8.40x10 ⁻¹⁴
T _I ⁴ C _M H	7.10x10 ⁻¹⁴	1.42x10 ⁻¹³	3.57x10 ⁻¹¹	3.15x10 ⁻¹¹	3.55x10 ⁻¹²	1.42x10 ⁻¹⁴	1.42x10 ⁻¹⁴
APPROXIMATE TOTAL PROBABILITY FOR CLASS SEQUENCES	1.3x10 ⁻¹⁰	2.7x10 ⁻¹⁰	6.72x10 ⁻⁹	5.9x10 ⁻⁹	6.7x10 ⁻⁹	2.7x10 ⁻¹¹	2.5x10 ⁻¹¹ *

*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 slightly overestimates the releases from these sequences. All loss of offsite power accident sequences involving containment leakage are included in the column for failed SGTS (cc, δc). The total conditional failure probability for these sequences is 0.5 (0.078 + 0.42).

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 ($T_{Fm}^2C_U$, $T_{Fm}^2C_2$)
- 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.

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.)

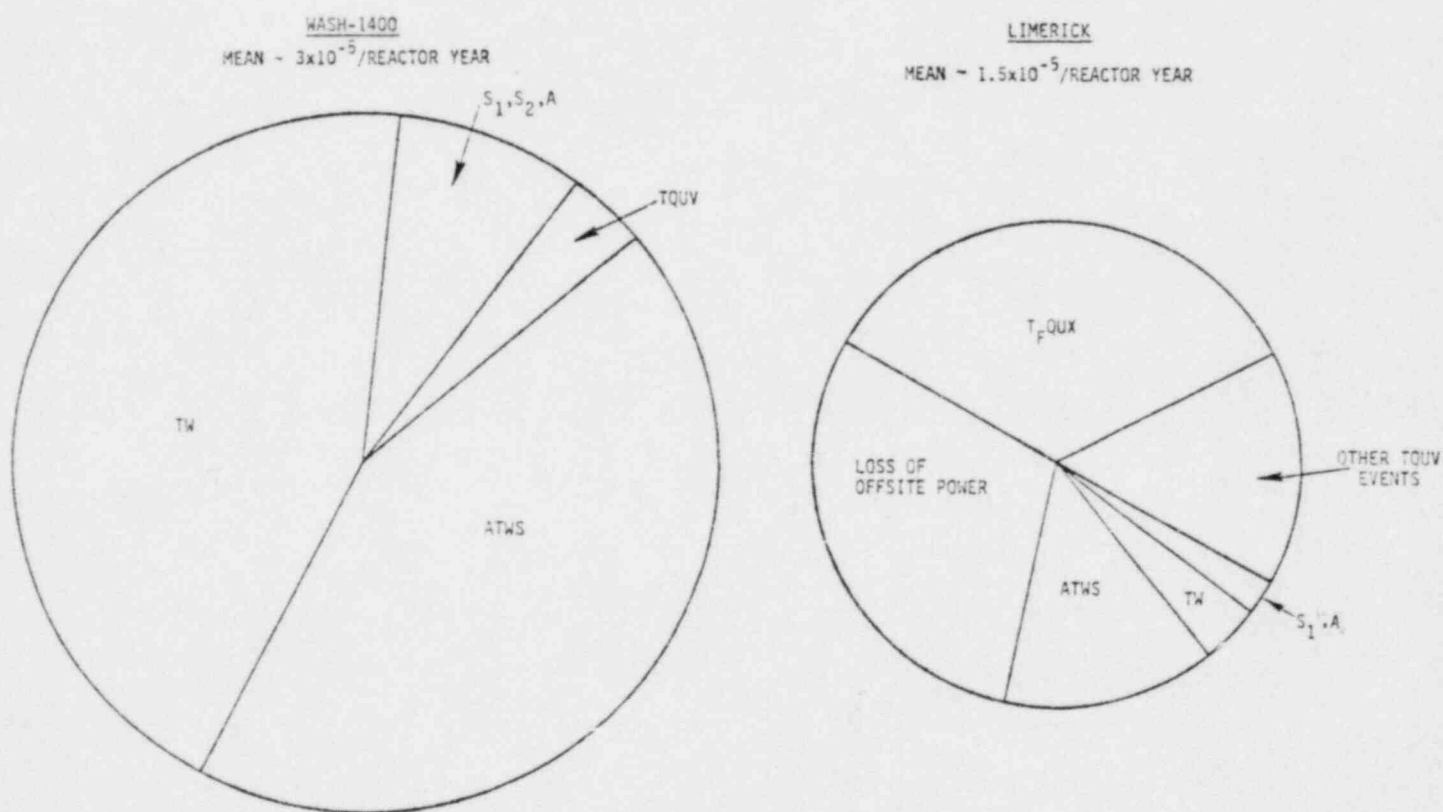
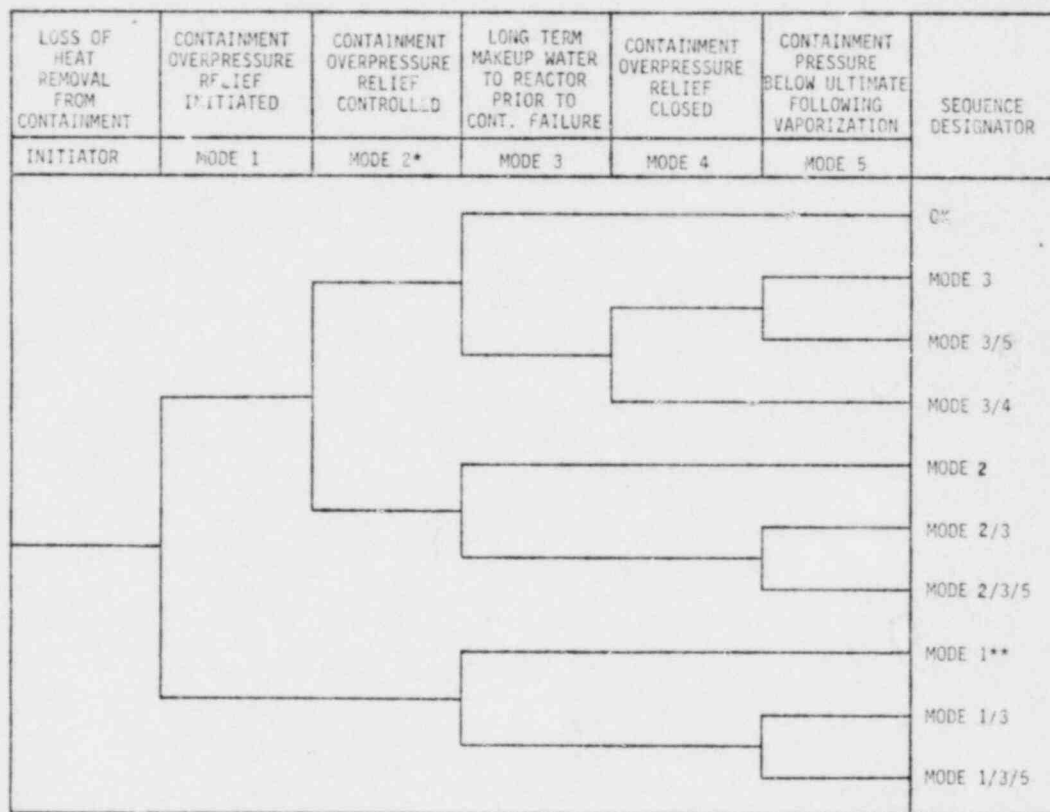


Figure 3.5.4 Comparison of the Contributing Accident Sequence to the Calculated Frequency of Core Melt from WASH-1400 and the Limerick Analysis (Area of "Pie Chart" is proportional to Mean Frequency)

*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 ₂ QUV		LOSS OF OFFSITE POWER	
Source	Initiation (Per Year)	Coolant Injection UV*	Total (Probability per Reactor Year)
WASH-1400	4×10^{-2}	2×10^{-5}	8×10^{-7}
Limerick Analysis	5.3×10^{-2}	7.5×10^{-5}	4.0×10^{-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 ₂ 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	2×10^{-3} $(3 \times 10^{-7})^*$	2×10^{-3}	4×10^{-7} $(3 \times 10^{-6})^*$
Limerick Analysis	1.78	.22	3.4×10^{-3}	2.1×10^{-3}	3×10^{-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	--	--	1×10^{-5}	NA	.1	1×10^{-5}
NUREG-0463 Alternate 3A	6	1.5×10^{-5}	1.5×10^{-5}	3×10^{-5}	10^{-2}	.1	9×10^{-6}
Limerick Alternate 3A	3.5	1.0×10^{-5}	2.0×10^{-5}	3×10^{-5}	10^{-2}	SLC=.035 HPCI=.07	2×10^{-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	1×10^{-6}	NA	1×10^{-5}
Limerick Analysis	7.2	8×10^{-7}	10^{-2}	5.8×10^{-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-C₂ type sequences; and (4) ATWS events for which one leg of the redundant SLC system fails, referred to as ATWS-C₁₂ 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)	3.6 x 10 ⁻²	.28*	.3	5x10 ⁻²	10 ⁻²
Failure to Scram w/Loss of SLC (ATWS-C ₂) (Figure 3.4.13)	~1	.28*	.8	5x10 ⁻²	10 ⁻¹
Failure to Scram w/Loss of 1 SLC and 1 or 2 RHR (ATWS-C ₁₂) (Figure 3.4.13)	3.6 x 10 ⁻²	.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-C₂, C₁₂ 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. 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.

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 ₁ PW	5.6x10 ⁻⁶	MODE 1 † (2 x 10 ⁻²) MODE 1/3 (2 x 10 ⁻⁵) MODE 3 (1.9x10 ⁻⁴) MODE 3/4 (1 x 10 ⁻⁵)	— — 1.3x10 ⁻⁹ —	1.4x10 ⁻⁷ negligible — —	— — — —	— — — 6.6x10 ⁻¹¹
	T ₂ PW	1.8x10 ⁻⁶	MODE 1 (2 x 10 ⁻²) MODE 1/3 (2 x 10 ⁻⁵) MODE 3 (1.9x10 ⁻⁴) MODE 3/4 (1 x 10 ⁻⁵)	— — 3.4x10 ⁻¹⁰ —	3.6x10 ⁻⁸ negligible — —	— — — —	— — — 1.8x10 ⁻¹¹
	T ₃ PW T ₄ PW	1.5x10 ⁻⁵	MODE 1 (2 x 10 ⁻²) MODE 1/3 (2 x 10 ⁻⁵) MODE 3 (1.9x10 ⁻⁴) MODE 3/4 (1 x 10 ⁻⁵)	— — 2.9x10 ⁻⁹ —	3.0x10 ⁻⁷ negligible — —	— — — —	— — — 1.5x10 ⁻¹⁰
	T ₅ PW _d	5.5x10 ⁻⁷	MODE 1 (2 x 10 ⁻²) MODE 1/3 (2 x 10 ⁻⁵) MODE 3 (1.9x10 ⁻⁴) MODE 3/4 (1 x 10 ⁻⁵)	— — 1.2x10 ⁻¹⁰ —	1.3x10 ⁻⁸ negligible — —	— — — —	— — — 6.5x10 ⁻¹²
	T ₁ C'W T ₁ W	8.3x10 ⁻⁶	MODE 1 (2 x 10 ⁻²) MODE 1/3 (2 x 10 ⁻⁵) MODE 3 (1.9x10 ⁻⁴) MODE 3/4 (1 x 10 ⁻⁵)	— — 1.6x10 ⁻⁹ —	1.7x10 ⁻⁷ negligible — —	— — — —	— — — 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 case there is assumed to be sufficient radioactivity released to the containment atmosphere to cause the GDR valves to be interlocked closed. The large and medium LOCA sequences then contribute directly 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 mode 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 probabilities associated with the sequence designators. For example, the probability of sequences designated mode 1/3 is calculated as the product of the probabilities of (mode 1) x (mode 3 given mode 1) x (not 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 shown 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 3.5.4, 3.5.5, and 3.5.6 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)	$T_T^1 C_M^{PW_2}$	2.7×10^{-8}	MODE 1* (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3×10^{-2})	-	-	- 8.1×10^{-10} 7.3×10^{-9} -	1.1×10^{-9} - - 3.4×10^{-10}
	$T_F^2 C_M^{W+}$ $T_P^2 (C_E K) W$	7.5×10^{-8}	MODE 1 (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3×10^{-2})	-	-	- 2.2×10^{-9} 2.0×10^{-8} -	3.0×10^{-9} - - 9.5×10^{-10}
	$T_E^3 C_M^{W+}$ $T_E^3 (C_E K) W$	2.5×10^{-8}	MODE 1 (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3×10^{-2})	-	-	- 7.5×10^{-10} 2.0×10^{-10} -	1.0×10^{-9} - - 3.3×10^{-10}
	$T_I^4 C_M^{W+}$ $T_E (C_E K) W$	7.1×10^{-9}	MODE 1 (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3×10^{-2})	-	-	- 2.1×10^{-10} 5.7×10^{-11} -	2.8×10^{-10} - - 9.0×10^{-11}
	$T_I C^{**}$	4.2×10^{-8}	MODE 1 (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3×10^{-2})	-	-	- 1.3×10^{-9} 1.1×10^{-8} -	1.7×10^{-9} - - 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 probabilities associated with the sequence designators. For example, 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 3) 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 shown 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 3.5.4, 3.5.5, and 3.5.6 are the sum of transients initiated from all powers.

Table 3.5.11

EXAMPLE SUMMARY TABLE OF ATWS-C₂[†] 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 ¹ C _N C ₂	1.1x10 ⁻⁷	MODE 1* (2x10 ⁻¹) MODE 1/3 (7.2x10 ⁻¹) MODE 1/3/5 (.8x10 ⁻²) MODE 3 (NEGLIGIBLE) MODE 3/4 (NEGLIGIBLE)	-	-	3x10 ⁻⁸	2.2x10 ⁻⁸ - 9x10 ⁻⁹ - -
	T _F ² C _N C ₂	3x10 ⁻⁸	MODE 1 (2x10 ⁻¹) MODE 1/3 (7.2x10 ⁻¹) MODE 1/3/5 (8x10 ⁻²)	-	-	2.2x10 ⁻⁸	6x10 ⁻⁹ - 2.4x10 ⁻⁹
	T _E ³ C _N C ₂	6x10 ⁻⁹	MODE 1 (2x10 ⁻¹) MODE 1/3 (7.2x10 ⁻¹) MODE 1/3/5 (8x10 ⁻²)	-	-	4.3x10 ⁻⁹	1.2x10 ⁻⁹ - 4.8x10 ⁻¹⁰
	T _I ⁴ C _N C ₂	2.8x10 ⁻⁹	MODE 1 (2x10 ⁻¹) MODE 1/3 (7.2x10 ⁻¹) MODE 1/3/5 (8x10 ⁻²)	-	-	2x10 ⁻⁹	5.6x10 ⁻¹⁰ - 2.2x10 ⁻¹⁰
ATWS-C ₂	TOTAL	N/A	N/A	-	-	1x10 ⁻⁷	4.2x10 ⁻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 probabilities associated with the sequence designators. For example, 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 shown 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 3.5.4, 3.5.5, and 3.5.6 are the sum of transients initiated from all powers.

† ATWS-C₂ 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 through the bridge event tree. A separate table is developed for the ATWS-C₁₂ sequences. The numerical values used in the quantification of the ATWS-C₁₂ 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-C₁₂* 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 ₁₂ (1 SLC PUMP & 1 or 2 RHRs FAIL)	T _T ¹ C _M C ₁₂ PM ₂	9.5x10 ⁻⁹	MODE 1 ⁺ (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3x10 ⁻²)	-	-	2.8x10 ⁻¹⁰ 2.6x10 ⁻⁹	3.8x10 ⁻¹⁰ - - 1.2x10 ⁻¹⁰
	T _F ² C _M C ₁₂ M ₁₂ ⁺ T _F ² C _M C ₁₂ PM ₂	2.6x10 ⁻⁸	MODE 1 (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3x10 ⁻²)	-	-	7.8x10 ⁻¹⁰ 7.0x10 ⁻⁹	1x10 ⁻⁹ - - 3.4x10 ⁻¹⁰
	T _T ³ C _M C ₁₂ M ₂ ⁺ T _E ³ C _M C ₁₂ PM ₂	1.9x10 ⁻⁹	MODE 1 (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3x10 ⁻²)	-	-	5.7x10 ⁻¹¹ 5.1x10 ⁻¹⁰	7.6x10 ⁻¹¹ - - 6.2x10 ⁻¹²
	T ₁ C _M C ₁₂ M ₂	2.5x10 ⁻⁹	MODE 1 (.04) MODE 1/3 (.03) MODE 3 (.27) MODE 3/4 (1.3x10 ⁻²)	-	-	7.5x10 ⁻¹¹ 6.7x10 ⁻¹⁰	1x10 ⁻¹⁰ - - 3.3x10 ⁻¹¹
ATWS-C ₁₂ (1 SLC PUMP & 2 RHR FAIL)	TOTAL	N/A	N/A	-	-	1.2x10 ⁻⁸	2.1x10 ⁻⁹

* ATWS-C₁₂ 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 probabilities associated with the sequence designators. For example, 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 shown 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 3.5.4, 3.5.5, and 3.5.6 are the sum of transients initiated from all powers.

3.5.4 Quantification of the Containment Event Tree

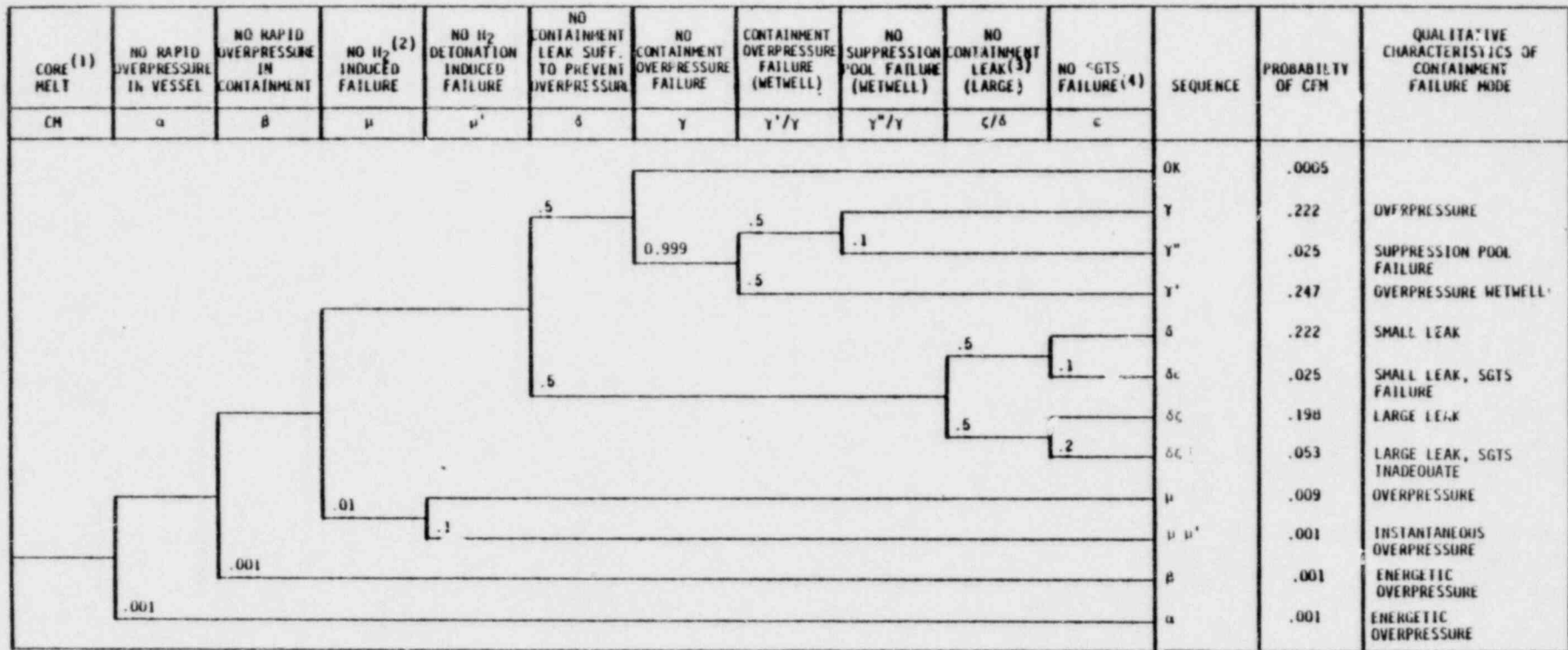
The two containment event trees which describe the possible paths of radioactive release from containment, and the numerical values used in the evaluation, are given in Figures 3.5.6a and b. The reason for two separate sets of numerical values for the containment event tree is that Class IV containment 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.

a -- 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 would 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 H₂ 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.

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CORE MELT (1)	NO RAPID OVERPRESSURE IN VESSEL	NO RAPID OVERPRESSURE IN CONTAINMENT	NO H ₂ INDUCED FAILURE (2)	NO H ₂ DETONATION INDUCED FAILURE	NO CONTAINMENT LEAK SUFF. TO PREVENT OVERPRESSURE	NO CONTAINMENT OVERPRESSURE FAILURE	CONTAINMENT OVERPRESSURE FAILURE (WETWELL)	NO SUPPRESSION POOL FAILURE (WETWELL)	NO CONTAINMENT LEAK (3) (LARGE)	NO SGTS FAILURE (4)	SEQUENCE	PROBABILITY OF CFM	QUALITATIVE CHARACTERISTICS OF CONTAINMENT FAILURE MODE
CM	a	β	μ	μ'	δ	γ	γ'/γ	γ''/γ	ε/δ	c			

(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 H₂ 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 evaluations were used to arrive at an estimate of α to be 10^{-3} per demand for a coherent in-vessel steam explosion which overpressurizes containment (given a core melt).

B -- Steam Explosion in Containment. 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).

μ, μ' -- Hydrogen Burn or Explosion in Containment. 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 0.1*. 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 may lead 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 for Class I, II, and III. For Class IV, a much lower value (4×10^{-4}) was used for the probability of leak before rupture.

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 is 0.025 for Class I, II, and III, and 0.05 for Class IV.

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 to an estimate of equal frequency of occurrence for both postulated leak size. ($\zeta/\delta = 0.5$).

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 the first three Classes of accident types. While this is a simplification, the uncertainty in containment failure probability is much larger than the potential variability associated with these three types 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 are 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 3.5.13 summarizes each of the containment failure modes 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 3.5.13, the four (4) separate generic accident sequence classes, which are evaluated separately in terms of these containment failure modes, are cited.

Table 3.5.14 gives a summary of the probabilities associated with each containment failure mode leak path, and each of the accident classes. This table 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)
a	Steam explosion in vessel	Note f	Note f	Note f	Note f
b	Steam explosion in containment	Note f	Note f	Note f	Note f
u'	H ₂ explosion induced containment failure	Note e	Note e	Note e	Note e
u	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 ^x	X	X	X	Note g
γ	Overpressure failure (A _R = 2.0 ft ²) Release through wetwell break	Note b	Note b	Note b	Note h
γ''	Overpressure failure (A _R = 2.0 ft ²) Wetwell pool drained	X	X	X	Note h
c	Overpressure, large leak (A _R = .2 ft ²)	X	X	X	Note h
cc	Overpressure, large leak, SGTS failure (A _R = .2 ft ²)	Note c	Note c	Note c	Note c
dc	Overpressure, small leak, SGTS failure (A _R = .05 ft ²)	Note d	Note d	Note d	Note d

- (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 c 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 b 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}
$\zeta, \delta e$	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}	2.5×10^{-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.

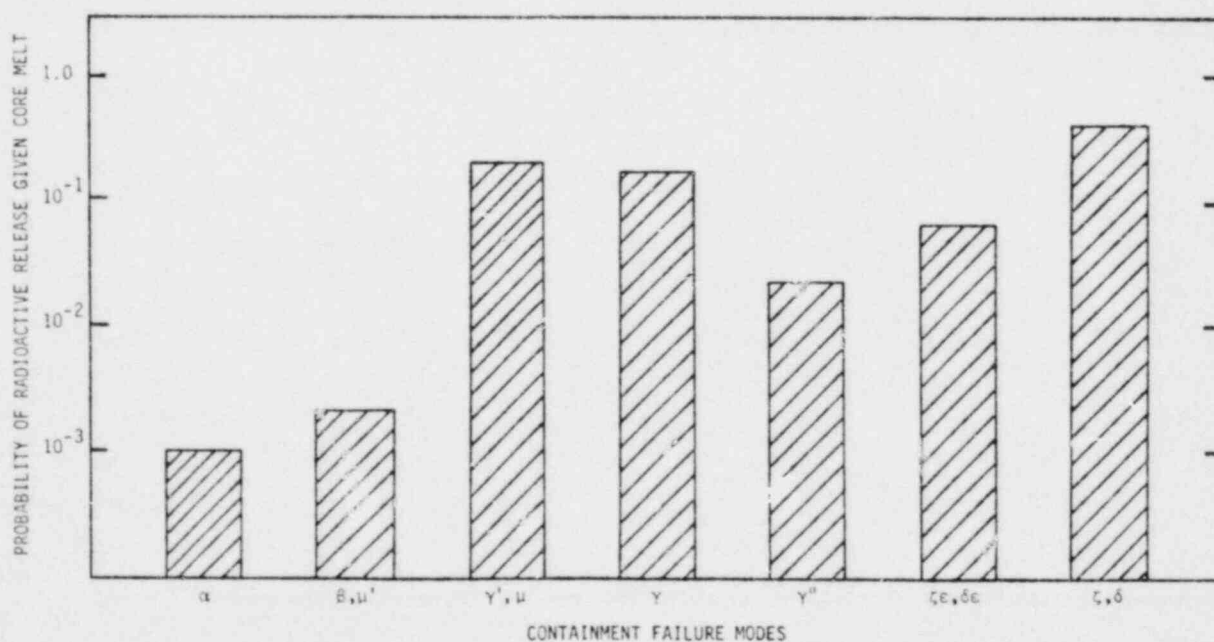


Figure 3.5.7 Probability of a Radioactive Release Given a Severe Degradation of Core Integrity -- Presented by Containment Failure Mode for All Classes.

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. (See Figure 3.6.1) 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.

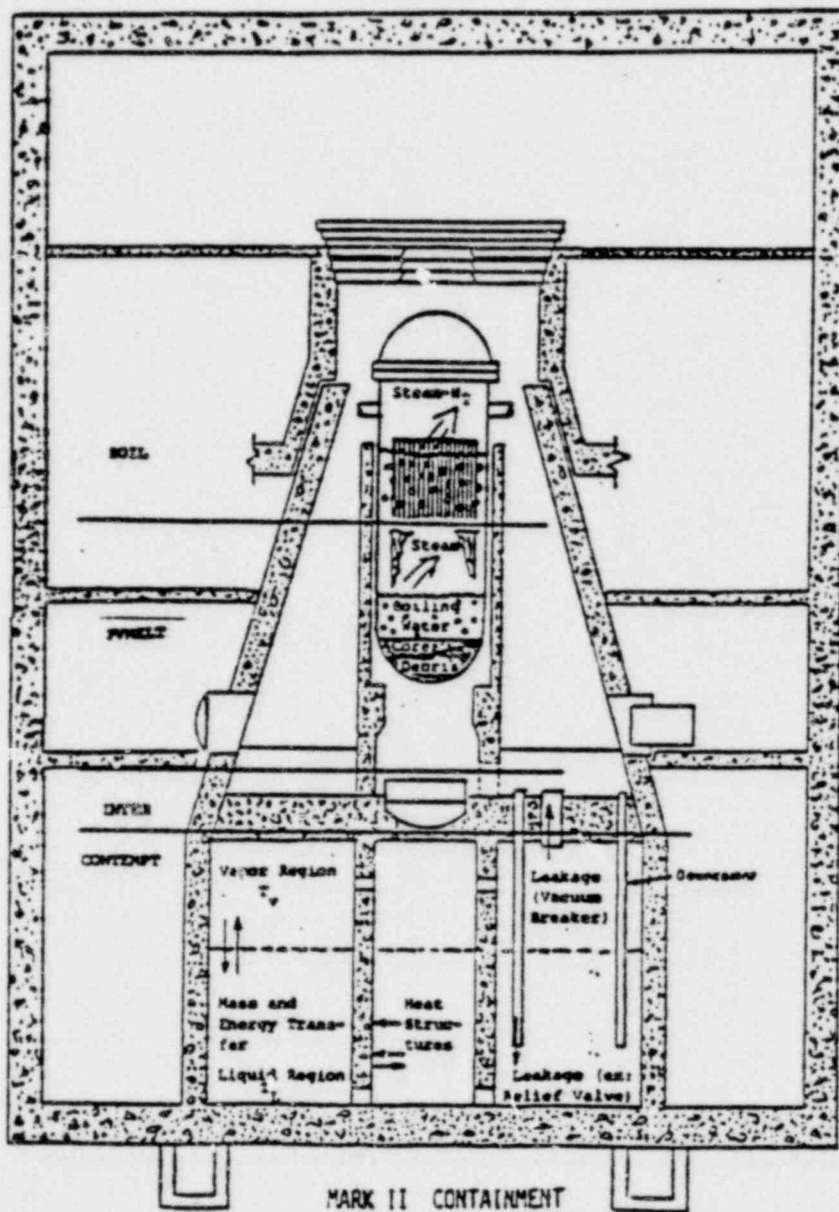


Figure 3.6.1 Schematic of the Limerick Containment Detailing INCOR Analysis

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. The release 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 which can be released to containment) are necessary. This fraction 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 melt-down release (the core is still inside the reactor vessel) a substantial por-

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 enclosure air should the primary containment fail. This method is effective as long as there is no large leak from the reactor enclosure.

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:

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
I	T QUV	Loss of coolant inventory	42%	17 PSI	Yes	Subcooled	SAI
II	TW	Containment pressure increase	4.1%	140PSI-Atmospheric	No	Saturation	SAI
III	ATWS-C ₂	Loss of coolant inventory	30%	25-65 PSI	Yes	Saturation	SAI
IV	ATWS-C ₂ U	Containment Pressure Increase	30%	140PSI-Atmospheric	No	Saturation	NRC/Battelle

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.

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						
						X ₀ ^(a)	I ₁ ^(b) + CH ₃ I	C ₅ ^(c)	I _e ^(d)	S _r ^(e)	R ₀ ^(f)	I ₀ ^(g)
a (C ₁ +C ₃) ^a	2.0	0.5	1.0	82	40	1.0	0.40	0.40	0.50	0.05	0.50	3.0x10 ⁻³
C ₂ ^a	39.0	0.5	8.0	82	40	1.0	0.096	0.10	0.40	0.01	0.40	2.0x10 ⁻³
C ₄ ^a	2.0	0.5	1.5	82	40	1.0	0.096	0.10	0.40	0.01	0.40	2.0x10 ⁻³
b (C ₁ +C ₂ +C ₃ +C ₄) (B ⁹⁹)	4.0	0.5	3.0	82	36	1.0	0.15	0.06	0.52	0.007	0.40	1.0x10 ⁻³
C ₁ ^b	7.0	2.0	6.0	82	36	1.0	0.11	0.09	0.016	0.011	3.2x10 ⁻³	3.2x10 ⁻³
C ₂ ^b	-	-	-	-	-	1.0	0.06	0.022	0.4	6.3x10 ⁻³	0.069	4.7x10 ⁻³
C ₃ ^b	-	-	-	-	-	1.0	0.04	0.024	0.073	2.7x10 ⁻³	8.6x10 ⁻³	9.1x10 ⁻⁴
C ₄ ^b	1.5	2.0	1.0	82	.3	1.0	0.26	0.201	0.472	0.029	0.084	5.0x10 ⁻³
c C ₄ ^c	1.0	2.0	1.0	82	.3	1.0	0.07	0.09	0.20	0.015	0.085	5.0x10 ⁻³
d C ₄ ^d	1.5	2.0	1.0	0	.3	1.0	0.73	0.70	0.55	0.0%	0.12	7.0x10 ⁻³
e C ₂ C ₅	.73					.73	0.019	9.8x10 ⁻³	0.046	1.6x10 ⁻³	3.2x10 ⁻³	5.8x10 ⁻³

(a) Includes Xe, Kr

(b) Includes I (elemental), Br

(c) Includes Cs, Rb

(d) Includes Te, Se, Sb

(e) Includes Sr, Ba

(f) Includes Ru, Rh, Pd, Mo, Te

(g) Includes La, Y, Zr, Nb, Ce, Pr, Nd, Np, Pu, Sm, Eu, Pu

1. Material released from the fuel/cladding gap or during core melt which has not been discharged through the SRV 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 gas 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, 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 is 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 wind rose 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.

*Both items are consistent with current NRC site review methods. See Appendix E for further discussion of radionuclide dispersion.

Tables 3.8.1 to 3.8.3 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 3.8.1

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 conservatism 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 3.8.1 (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: <ul style="list-style-type: none"> <li data-bbox="707 483 1113 533">• Manual shutdowns with gradual power reductions <li data-bbox="707 555 1125 604">• 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 than used for containment failure lead to: <ol style="list-style-type: none"> <li data-bbox="712 1200 1166 1249">a. shorter retention time for fission products <li data-bbox="712 1272 1130 1321">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: <ul style="list-style-type: none"> <li data-bbox="657 1469 1139 1518">• drop onto the diaphragm floor in one coherent mass <li data-bbox="657 1541 1118 1615">• fragment and disperse around containment from blowdown of RPV if a large blowdown force occurs <li data-bbox="657 1637 1118 1686">• stay inside the pedestal region of the diaphragm floor <li data-bbox="657 1709 1130 1783">• 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 layer freezing.	C

Table 3.8.1 (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 effective- ness	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 3.8.2

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. Because 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 occurrence may not be represented.	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 3.8.2 (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. These 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 3.8.3

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 <u>conservative</u> .	C

Table A.1.5

PIPE SIZE VERSUS FAILURE MODE DISTRIBUTION
FOR PIPE FAILURES WITHIN BWR PLANTS*

FAILURE MODE	PIPE SIZE CATEGORY BY DIAMETER				ROW SUM
	(<1")	(>1", <=6")	(>6", <=10")	(>10")	
Vibration	15(53.6)**	7(17.9)	1(16.7)	0	23
Thermal & Cyclic Fatigue	4(14.3)	3(7.7)	0	1(50.0)	8
Fabrication	6(21.4)	4(10.3)	1(16.7)	0	11
Corrosion	1(3.6)	3(7.7)	0	1(50.0)	5
Erosion	1(3.6)	7(17.9)	1(16.7)	0	9
Stress Corrosion Cracking	1(3.6)	15(38.5)	3(50.0)	0	19
Column Sums	28	39	6	2	75

*Note: The entries in this table only include the 75 failures out of 121 for which the failure mode and pipe size were both specified.

**Entries in parenthesis represent percentage of column sum.

the NUREG-0460 value of 3×10^{-5} per demand. This value is felt to be conservative since recent GE analysis indicates that a significantly lower probability may be appropriate.

Limerick ATWS logic will respond to ATWS events for all transient and LOCA initiators throughout the range of initial operating power levels. For the Limerick PRA, ATWS events from all power levels are treated the same, and the ATWS frequency used includes all cases.

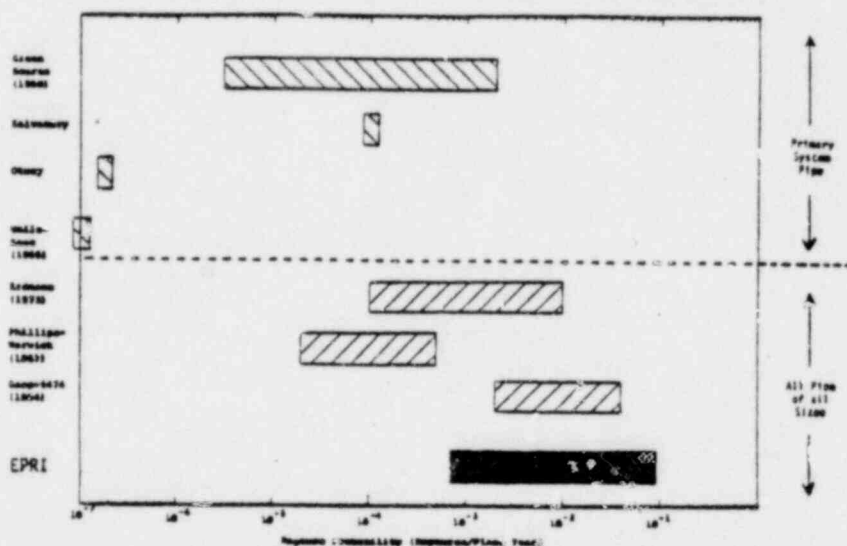


Figure A.1.3 Comparison of Evaluated Rupture Probabilities of Pipe to Estimate Nuclear Power Plant Rupture Probabilities

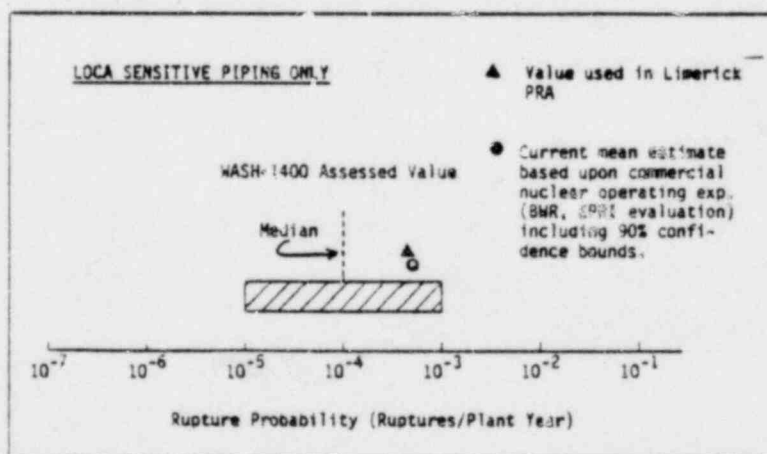


Figure A.1.4 Estimates of LOCA Initiated by A Large Pipe Break

Table A.1.5

PIPE SIZE VERSUS FAILURE MODE DISTRIBUTION
FOR PIPE FAILURES WITHIN BWR PLANTS*

FAILURE MODE	PIPE SIZE CATEGORY BY DIAMETER				ROW SUM
	(<1")	(>1", <=6")	(>6", <=10")	(>10")	
Vibration	15(53.6)**	7(17.9)	1(16.7)	0	23
Thermal & Cyclic Fatigue	4(14.3)	3(7.7)	0	1(50.0)	8
Fabrication	6(21.4)	4(10.3)	1(16.7)	0	11
Corrosion	1(3.6)	3(7.7)	0	1(50.0)	5
Erosion	1(3.6)	7(17.9)	1(16.7)	0	9
Stress Corrosion Cracking	1(3.6)	15(38.5)	3(50.0)	0	19
Column Sums	28	39	6	2	75

*Note: The entries in this table only include the 75 failures out of 121 for which the failure mode and pipe size were both specified.

**Entries in parenthesis represent percentage of column sum.

1. Based on estimates from other sources (e.g., RSS, Bush), a pipe rupture leading to a LOCA is assessed to occur once every 10,000 plant years. With this estimated frequency and the fact that U. S. cumulative reactor experience is only approximately 260 reactor years, it is difficult to assess the probability of a LOCA accurately by only considering LOCA-sensitive pipes because sufficient operating experience has not accrued;
2. There have been several pipe ruptures reported in high integrity piping in secondary systems of nuclear plants.

Based upon these rupture failures, a probability of a pipe rupture failure in LOCA-sensitive piping can be estimated using the fact that LOCA-sensitive piping represents approximately 10% of the piping in a reactor plant.

3. For the calculation of small LOCA probability, several incidents of reactor system failures have occurred which can be interpreted as being similar to a small LOCA.

Figure A.1.3 presents comparisons of various estimates of pipe rupture failures from various sources. It is apparent from this figure that there is an appreciable overlap in the error bounds of the estimated pipe failure probabilities.

Based on Figure A.1.4, the probability of a large LOCA may be considered to be slightly higher than that used in WASH-1400 since there is not adequate data to support a lower value. Therefore, the values in Table A.1.6 were used.

A.1.3.3 ATWS Accident Initiators

Since ATWS events are by definition transients, the ATWS initiator frequencies are calculated by using the transient initiator frequencies, as determined in section A.1.3.1, and multiplying them by the scram failure probability. The scram failure probability used in the Limerick analysis is

the NUREG-0460 value of 3×10^{-5} per demand. This value is felt to be conservative since recent GE analysis indicates that a significantly lower probability may be appropriate.

Limerick ATWS logic will respond to ATWS events for all transient and LOCA initiators throughout the range of initial operating power levels. For the Limerick PRA, ATWS events from all power levels are treated the same and the ATWS frequency used includes all cases.

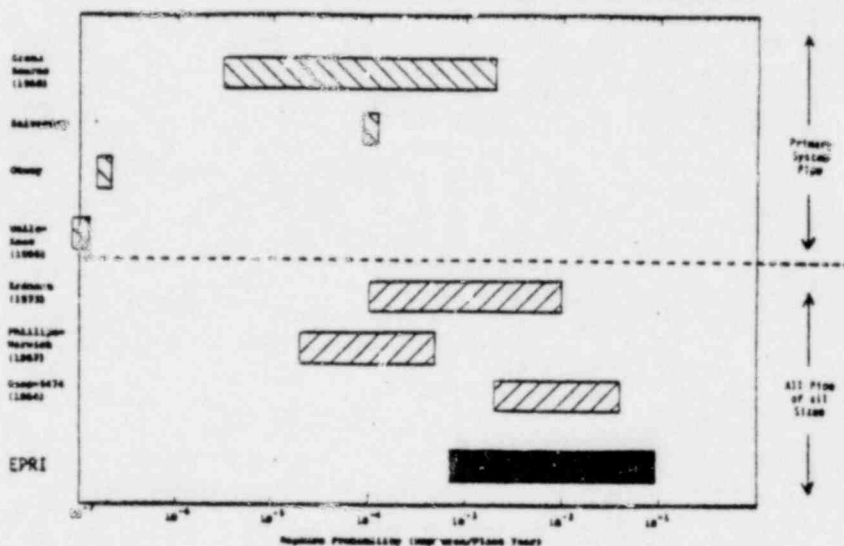


Figure A.1.3 Comparison of Evaluated Rupture Probabilities of Pipe to Estimate Nuclear Power Plant Rupture Probabilities

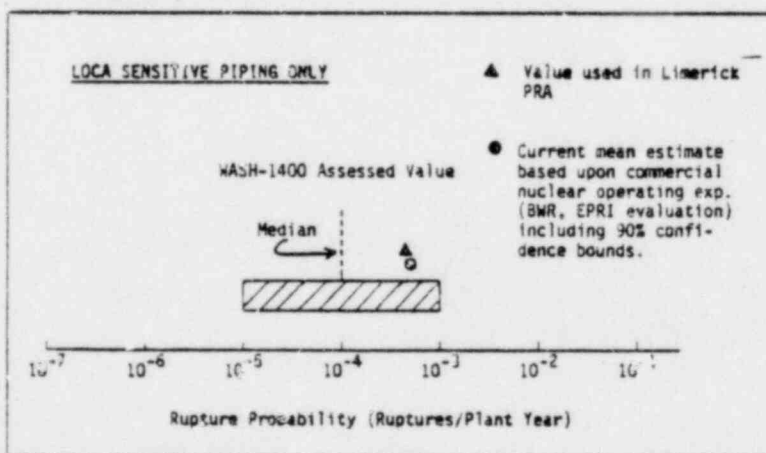


Figure A.1.4 Estimates of LOCA Initiated by A Large Pipe Break

Table A.1.6

PROBABILITY OF A LOCA

PIPE SIZE	BWR* EPRI VALUATION (MEAN)	WASH-1400 (MEDIAN)	VALUE USED IN THE LIMERICK PRA
Large Pipe $\geq 4''\phi$	7.7×10^{-4}	$1.0 \times 10^{-4**}$	4.0×10^{-4}
Medium Pipe $\geq 4''\phi$	3.0×10^{-3}	$3.0 \times 10^{-4**}$	2.0×10^{-3}
Small Pipe $\geq 1''\phi$	8.0×10^{-3}	$1.0 \times 10^{-3**}$	1.0×10^{-2}

* This probability estimate is based upon all high pressure plant piping. Since only the LOCA sensitive pipe is of concern here, the probabilities are reduced by approximately a factor of ten (LOCA sensitive piping is approximately 10% of the plant piping).

** Mean values are approximately three times larger.

Pipe, Heat Exchangers, Pressure Containing Members: Because of their relatively low failure rates, the probability of rupture or flow blockage in these components during an accident sequence has very little effect on the probabilistic quantification of the system level fault trees. The system unavailabilities are dominated by active component failures, human errors, and maintenance outages.

Instrumentation: Instrumentation component failure rates are relatively low. Also, the components are combined in a logic which allows proper operation despite some failures. However, since some instrumentation sensors may be used for actuation of more than one safety system, a failure of the instrumentation can lead to a failure to automatically initiate several systems.

A.2.2 Application of the Failure Rate Data

The application of failure rate data to the quantification of the fault tree model can be done in several ways. WASH-1400 made use of the concept of demand failure rates for the purpose of quantifying component failures in safety systems which were in a standby status and required to begin operation following an accident initiation. A General Electric reliability assessment of safety systems used the concept of a constant hourly failure rate for components in standby. The probability of failure was calculated as $1 - e^{-\lambda\theta/2}$ where λ is the failure rate and θ is the scheduled time between system tests (e.g., monthly, annually, etc.).

Table A.2.2 gives a comparison of the failure rates from three available sources. The demand failure rates will be used to describe such failures as:

- Pump or turbine fails to start
- Valve fails to change state (position)
- Human errors, such as miscalibration of instrumentation
- Relief valve fails to open.

Table A.2.2

FAILURE RATE DATA COMPARISON

GENERAL GROUPING	COMPONENT	FAILURE MODE	FAILURE RATES								
			WASH-1400			GE			NRC		
			MEDIAN	EF*	R*		R*	BWR	PWR	R*	
VALVES	MOV (Motor operated valve)	NO FO	$1 \times 10^{-3}/d$	3	A	$1.6 \times 10^{-6}/hr$	B	$3 \times 10^{-3}/d$	$2 \times 10^{-3}/d$	F	
		NC FC	$1 \times 10^{-4}/d$	3	A	$1.5 \times 10^{-6}/hr$	B				
		NO FC	$1 \times 10^{-4}/d$	3	A	$0.15 \times 10^{-6}/hr$	B	$1 \times 10^{-3}/d$	$5 \times 10^{-4}/d$	F	
		NC FC	$1 \times 10^{-4}/d$	3	A	$0.16 \times 10^{-6}/hr$	B				
	Check Valve	Falls to permit flow	$1 \times 10^{-4}/d$	3	A	$0.15 \times 10^{-6}/hr$	B	$1 \times 10^{-4}/d$	$2 \times 10^{-4}/d$	F	
		Falls to prevent flow	-	-	-	$1.6 \times 10^{-6}/hr$	B	$5 \times 10^{-4}/d$	$2 \times 10^{-4}/d$	F	
		Rupture, λ_s	$1 \times 10^{-8}/hr$	10	A						
	Manual Valve	Failure to remain open	$1 \times 10^{-4}/d$	3	A						
		Rupture, λ_s	$1 \times 10^{-8}/hr$	10	A						
	Valve, Check Testable	Falls to permit flow				$0.22 \times 10^{-6}/hr$	B				
		Falls to prevent flow				$2.2 \times 10^{-6}/hr$	B				
	ADS Depressurization Valve	Falls to operate				$7 \times 10^{-6}/hr$	B				
Solenoid Operated Valves	Failure to operate	$1 \times 10^{-3}/d$	3	A							
	Failure to remain open	$1 \times 10^{-4}/d$	3	A							
	Rupture, λ_s	$1 \times 10^{-8}/hr$	10	A							
Air-Fluid Operated	Failure to operate	$3 \times 10^{-4}/d$	3	A							
	Failure to remain open	$1 \times 10^{-4}/d$	3	A							
	λ_s	$3 \times 10^{-7}/d$	3	A							
	Rupture, λ_s	$1 \times 10^{-8}/hr$	10	A							
Relief Valves	Failure to open	$1 \times 10^{-5}/d$	3	A			$5 \times 10^{-3}/d$	$1 \times 10^{-3}/d$	F		
	Premature open	$1 \times 10^{-5}/d$	3	A			$4 \times 10^{-3}/d$	$3 \times 10^{-3}/d$	F		

* EF: Error Factor
R: Reference (P. A-28)

Table A.2.2
(Continued)

GENERAL GROUPING	COMPONENT	FAILURE MODE	FAILURE RATES								
			WASH-1400			GE		NRC			
			MEDIAN	EF*	R*		R*	BWR	PWR	R*	
BATTERY	Battery Power Systems: (wet cell)	Failure to provide proper output, λ_0	$3 \times 10^{-6}/hr$	3	C						
CIRCUIT BREAKERS	Fuses	Failure to open	$1 \times 10^{-5}/d$	3	C						
		Premature open, λ_0	$1 \times 10^{-6}/hr$	3	C						
	Motors Relays	Failure to energize	$1 \times 10^{-4}/d$	3	C						
		Failure of NC contacts to close given energizers	$3 \times 10^{-7}/hr$	3	C						
		Failure of NC contact by opening-not energizing	$1 \times 10^{-7}/hr$	10	C						
		Short across NC/NC contact, λ_0	$1 \times 10^{-8}/hr$	10	C						
		Coil open, λ_0	$1 \times 10^{-7}/hr$	10	C						
Coil short to power, λ_0	$1 \times 10^{-8}/hr$	10	C								
Circuit Breakers	Failure to transfer Premature transfer	$1 \times 10^{-3}/d$ $1 \times 10^{-6}/hr$	3 3	C C							
INSTRUMENTATION AND CONTROL	Switch contacts Relays (HFA)	Coil fails to operate	$1 \times 10^{-4}/d$	10	C	$0.4 \times 10^{-6}/hr$	B				
		Coil fails to open	$2 \times 10^{-7}/hr$	10	C	$0.98 \times 10^{-6}/hr$	B				
	Temperature Switch	Fails to operate Fails closed				$2.3 \times 10^{-6}/hr$ $0.33 \times 10^{-6}/hr$	B B				
Pressure Sensor (Reactor and Containment)	Fails to operate	$1 \times 10^{-4}/d$	3	C	$1.1 \times 10^{-6}/hr$	B					

* EF: Error Factor
R: Reference (p. A-28)

Table A.2.2
(Continued)

GENERAL GROUPING	COMPONENT	FAILURE MODE	FAILURE RATES								
			WASH-1400			GE		NRC			
			MEDIAN	EF*	R*	R ^W	BWR	PWR	R*		
INSTRUMENTATION AND CONTROL** (cont.)	Flow Controller	Fail to operate				$4.2 \times 10^{-6}/hr$	B				
	Manual	Fail to transfer	$1 \times 10^{-5}/d$	3	C						
	Reactor Water Level Sensor	Fail to operate				$3.9 \times 10^{-6}/hr$	B				
	Limit Switch		$3 \times 10^{-4}/hr$	3	C						
	Flow Switch	Failure to operate				$0.26 \times 10^{-6}/hr$	B				
	E/S Converter	Fail to operate				$4.2 \times 10^{-6}/hr$	B				
	Square Root Converter	Fail to operate				$4.2 \times 10^{-6}/hr$	B				
	Power Supply	Fail to operate				$4.2 \times 10^{-6}/hr$	B				
	Solid State Dev. HI Power App.	Falls to function, λ_0 Falls shorted, λ_0		$3 \times 10^{-6}/hr$	10	C					
				$1 \times 10^{-6}/hr$	10	C					
	Solid State Dev. Power App.	Falls to function, λ_0 Falls shorted, λ_0		$1 \times 10^{-6}/hr$	10	C					
				$1 \times 10^{-7}/hr$	10	C					
Instrumentation General (includes transmitter, amplifier and output devices)	Failure to operate, λ_0 Shift in calibration, λ_0		$1 \times 10^{-6}/hr$		C						
			$3 \times 10^{-5}/hr$		C						

*EF: Error Factor

R: Reference (p. A-28)

**General Electric and PECO information on Logic and Sensor Testing indicate six-month test interval for logic and sensor testing, except for ADS which is 18 months. (Therefore, failure probability is a factor of three larger for ADS logic and sensors.)

Table A.2.2
(Continued)

GENERAL GROUPING	COMPONENT	FAILURE MODE	FAILURE RATE						
			WASH-1400			GE		NRC	
			MEDIAN	EF*	R*	R	BWR	PWR	R*
TRANSFORMERS	Transformers:	Open circuit, primary or secondary	1×10^{-6} /hr	3	C				
		Short primary to secondary, λ_0	1×10^{-6} /hr	3	C				
ELECTRICAL DISTRIBUTION	Wires (typical Circuits, Several Joints)	Open circuit, λ_0	3×10^{-7} /hr	3	C				
		Short to ground, λ_0	3×10^{-7} /hr	10	C				
		Short to power, λ_0	1×10^{-5} /hr	10	C				
	Terminal Boards	Open connection, λ_0	1×10^{-7} /hr	10	C				
Short to adjacent circuit, λ_0		1×10^{-8} /hr	10	C					

*EF: Error Factor R: Reference

Note: Reference Table 4.12 from GE DOCUMENT, NEDE-24809

"Probabilistic Analysis of the Reliability of BWR/4 Systems for Small LOCA Events," April 1980:

- References:
- A. WASH-1400: Appendix 3, Table III 4-1
 - B. GE Evaluation
 - C. WASH-1400: Appendix 3, Table III 4-2
 - D. WASH-1400: Appendix 3, Table III 4-3
 - E. W. H. Sullivan and J. P. Poloski, "Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants- January 1972 to April 1978," prepared for U.S. Nuclear Regulatory Commission by EG&G. NUREG/CR-1205, dated January 1980.
 - F. EG&G report for USNRC, July 1979, Table I
 - G. "Summaries of Failure Rate Data," Government-Industry Data Exchange Program (GIDEP), Volume 1, dated January 1974.
 - H. WASH-1400: Appendix 3, Table III 6-1
 - I. EG&G report USNRC, August 1979

Demand failure rates are converted to failure probabilities by multiplying by the number of demands. This applies to all situations except the HPCI fault tree, where subsequent* automatic starts are possible.

Hourly failure rates are used to describe standby systems that are subject to continuous exposure:

- Instrumentation sensors
- Pipe and pressure containing member failures.

Hourly failure rates are converted to failure probabilities for successful initiation of a component using the following failure probability equation. For equipment in a standby condition which can fail, the probability is given by:

$$\text{failure probability} = \text{failure rate (per hour)} \times \frac{\text{exposure time}^{**} \text{ (hours)}^{\dagger}}{2}$$

In addition, there is a probability of failure associated with a component's failure to run for the duration of the accident analysis. For the purposes of failure probability calculations, the exposure/run time for the "accident" is taken to be 20 hours. This time is consistent with the definition of the accident sequence given in Section 1.5.

A.2.3 Comparisons of Data Evaluation Methods

There have been some efforts in recent probabilistic risk analyses to develop a formalized method of calculating the component failure rates on a plant-specific basis. These efforts have focused on the application of a Bayesian statistics approach to the use of available data.

*The determination of proper data input for this analysis is complicated by the fact that most applications of data assume a demand failure probability for a single start or demand. In this analysis, the potential unavailability of a system to respond to multiple demands to start is incorporated. The failure probability for these subsequent demands is judged to be less than the demand failure probability for the initial start.

** Interval between tests verifying operation.

† When failure probability is less than .1.

Probabilistic risk assessment methods allow not only determination of a best estimate frequency for various events, but also the "probability" of this frequency; that is, the frequency results can be expressed not only as an expected value, but also as a probability distribution about this expected value. Determination of the probability distribution for the failure rate parameters which describe the component behavior can be accomplished by utilizing component histories on a plant-specific basis.

The expected value of specific component failure rates as a function of various methods of data assessment were compared. Basically, two methods were investigated: Bayesian and classical statistics. The example chosen for comparison is pump failure to start on demand. Data for this event is taken from the EG&G data summary on pumps (A.2.1). This comparison used the pump experience from BWR's as given in Table A.2.3.

The methodology utilized can be found in several sources. Reference A.2.5 contains information on Bayesian methods and Reference A.2.6, classical statistics.

It is assumed for the following that all the plants in the EG&G study have an underlying pump failure probability in common. (If not, then the data of plants believed to be outliers could be discarded.) By means of some preliminary analysis, a common probability distribution model is fit to the reported pump failure data of each plant in the EG&G study. An adequate prior distribution on the parameter (s) of the model can then be chosen based on the distribution model above. As the Bayesian inference has an inherent sequential nature, it can be used to analyze the data of the plants successively to derive a posterior distribution representative of the current "state of knowledge" regarding pump failure probability. The model set-up required for the Bayesian inference includes:

- A prior distribution representative of our previous knowledge.
- A likelihood based on consideration of all the data

Table A.2.3

PUMP DATA FROM REFERENCE A.2-1

STANDBY PUMPS - MOTOR DRIVEN - DOES NOT OPERATE - 1/72 THRU 4/78 - WITHOUT COMMAND FAULTS
GENERAL ELECTRIC PLANTS

PLANT	CRIT. HRS.	POPULATION	DEMANDS	FAILURES	PJP. HOURS	POP. DEMANDS	STANDBY HOUR RATE	DEMAND RATE
BF1	15553.0	10	57	1	195536.0	570	6.4E-06	1.8E-03
BF2	12326.0	10	45	0	123260.0	450	2.4E-05*	6.7E-03*
BF3	11545.0	10	21	0	115450.0	210	2.6E-05*	1.4E-02*
BP1	38622.0	5	76	0	193110.0	360	1.6E-05*	7.9E-03*
BM1	7544.0	6	19	1	60352.0	152	1.7E-05	6.6E-03
BR2	14576.0	8	36	0	116608.0	208	2.6E-05*	1.0E-02*
COL	27641.0	8	49	2	221128.0	392	9.0E-06	5.1E-03
DA1	22487.0	8	46	0	179896.0	384	1.7E-05*	7.8E-03*
DA1	35600.0	8	76	3	284800.0	608	1.1E-05	4.9E-03
DA2	39390.0	11	76	0	673290.0	836	6.9E-06*	3.6E-03*
DR3	41057.0	11	76	0	451627.0	836	6.6E-06*	3.6E-03*
EN1	23634.0	6	44	0	141804.0	264	2.1E-05*	1.1E-02*
FPL	18189.0	8	41	1	145512.0	328	6.9E-06	3.0E-03
M11	39560.0	13	76	3	514280.0	988	5.8E-06	3.0E-03
MO1	44190.0	8	76	1	353520.0	608	2.8E-06	1.6E-03
NH1	41084.0	19	76	3	780596.0	1444	3.8E-06	2.1E-03
UC1	42170.0	20	76	2	843400.0	1520	2.4E-06	1.3E-03
PH2	24563.0	10	56	0	245630.0	560	1.2E-05*	5.3E-03*
EB3	22737.0	10	45	2	227370.0	450	8.8E-06	4.4E-03
PI1	31756.0	8	76	1	254048.0	608	3.9E-06	1.6E-03
QC1	37446.0	8	76	1	299568.0	608	3.3E-06	1.6E-03
QC2	39780.0	8	76	0	318740.0	576	9.4E-06*	5.2E-03*
VY1	39826.0	8	73	0	318608.0	584	9.4E-06*	5.1E-03*
TOTALS				21	6777627.0	13644		

* DENOTES UPPER 95 PERCENT CONFIDENCE BOUND WHEN NO FAILURES RECORDED

To assess the sensitivity to various priors and distribution models, the Bayesian methods were applied to the data in the following four combinations of the above two items:

1. A uniform prior* distribution: plant-specific data modeled as a binomial distribution**
2. The Reactor Safety Study log-normal distribution for pumps to fail on demand as the prior and the plant-specific data as binomial distributions
3. A "flat" prior distribution: the plant-specific data being modeled as log-normal distributions with mean and variance the same as the equivalent binomial distribution
4. Chi-square distribution (upper bound).

The results of these various assessments are given in Table A.2.4. As shown, the mean value is relatively insensitive to the method of data assessment. Moreover, those two cases which have the more consistent assumptions, Case 1 and Case 4, agree to three significant figures.

Table A.2.4

EXPECTED VALUE	CASE 1		CASE 2		CASE 3		CASE 4
	Prior	Data	Prior	Data	Prior	Data	Classical
	Flat	Binomial	RSS	Binomial	Flat	Log-Normal	
	1.61×10^{-3}		1.48×10^{-3}		2.01×10^{-3}		1.61×10^{-3}

* A uniform or "flat" prior distribution indicates that each failure probability value is equally likely. This generally indicates a lack of prior knowledge of the "real" distribution.

**The binomial distribution is constructed based upon the number of demands at each plant.

As shown above, the failure probability to be used in the assessment of risk at Limerick can be estimated by a number of methods. The above example demonstrates that data from individual plants can be characterized by a common distribution model, and the total population of plants combined in a Bayesian fashion, to determine a posterior distribution representing the current state of knowledge. However, it appears from the results that this method, which may be rather time consuming, produces point value results which are similar to the classical statistical approach. While the establishment of a specific method of combining component data from various sources is desirable, there are a number of variations in the currently available basic data used in the quantification of the accident sequences. These variations may tend to obscure any usefulness which could be gained by establishing a rigorous method of combining existing data. These potential variations in the data are due to such items as:

1. Lack of specificity as to the function/type of component (e.g., main circulating water pump or RHR pump). All types of pumps are treated together because of the very small population available.
2. Age of the components is generally not considered.
3. Variations which occur among different manufacturers are not included in the LER reporting scheme.
4. Local plant test and maintenance procedures, training programs, and management/personnel factors may vary.

Compared with variations arising from the above listed items which may be encountered at a specific plant, the calculated "expected values" from Table A.2.4 show very small differences which do not warrant an extensive Bayesian analysis. See Appendix F for further discussion of the statistical treatment used in the analysis.

References

- A.2.1 W. H. Sullivan and J.P. Poloski, Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants January 1972 to April 1978, prepared for U.S. Nuclear Regulatory Commission by EG&G. NUREG/CR-1205, January 1980.
- A.2.2 J.P. Poloski and W.H. Sullivan, Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants - January, 1976 to December 31, 1978, prepared for U.S. Nuclear Regulatory Commission by EG&G. NUREG/CR-1362 March 1980.
- A.2.3 Warren H. Hubble and Charles F. Miller, Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants - January 1, 1976 - December 31, 1978, prepared for U.S. Nuclear Regulatory Commission by EG&G. NUREG/CR-1362 June 1980.
- A.2.4 Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, U.S. NRC. WASH-1400.
- A.2.5 IEEE Transactions on Reliability, Vol. R-21, No.3, August 1972. (The whole issue deals with Bayesian Questions.)
- A.2.6 A. Hald, Statistical Theory With Engineering Applications, John Wiley and Sons, Inc., 1952.

A.3 HUMAN FAILURE RATE DATA

The safety systems provided in boiling water reactors to prevent and mitigate accidents are generally designed to operate automatically during the initial states of accident sequences. Information on the conditions in the reactor and on the operation of the safety systems during an accident would be displayed in the control room so that the operator would be able to follow the sequence of events, but no direct human action would usually be required until the accident were brought automatically under control. However, human intervention would be required in case of malfunctions in the automatic systems, and human interaction with the system exists in routine plant operation, testing, and maintenance. Therefore, human reliability plays a very large role in safety system reliability.

Although data on human reliability are sparse and difficult to apply to specific situations, many attempts along these lines have been published, usually using subjective estimates by experts in related fields. Section A.3.1 discusses some of the factors that effect human failure rates. A brief summary of several data sources, in particular the Handbook of Human Reliability by Swain and Guttman, is given in Section A.3.2. Finally, data and evaluations used for the Limerick analysis appear in Section A.3.3.

A.3.1 General Discussion of Causes of Human Error

Many causes and preventions of human errors must be considered in order to obtain a reasonable estimate of human failure rates. Some of the major considerations are:

- Plant design
- Training and experience
- Procedures
- Stress.

The following discussion of these items outlines the considerations which have been incorporated in the Limerick assessment to determine the effect of human error on each accident sequence.

Plant Design: Some reactor control rooms may have potentially confusing arrangements and labeling of controls. In particular, many labels are very long and differ in only a few letters or digits. This may lead to fairly high error rates for manipulating the wrong switch when controls and displays are close together without separation by functional flow lines on the panels, especially in emergency conditions. WASH-1400 found that the control arrangements in the plants it studied deviated from human engineering standards such as those used in the military. Errors can also be increased due to multiple alarms (several alarms due to one cause) that confuse the true cause. Human error rates for Limerick are not subject to high failure rates due to this cause because of improved plant design such as functional flow lines, annunciators for locked valves, and improved testing options.

Training and Experience: WASH-1400 stated that the training of nuclear plant personnel was outstanding, thereby assuming high reliability for routine maintenance, calibration, and control room operation. However, WASH-1400 found that operators were not able to "talk through" appropriate emergency procedures without hesitation or indecision. This led to the assumption of less reliability in major emergencies. An additional point concerns experience. Although experience leads to improvement, it can also increase error. For example, a technician or operator may become so used to seeing a correct instrument reading that when an out-of-tolerance condition occurs, he may still "see" the correct reading.

Procedures: Written procedures such as check lists are a definite aid in human reliability. However, WASH-1400 found that the written instructions do not conform to established principles of good

writing; they are more typical of military maintenance procedures of approximately 20 years ago. WASH-1400 also found poor printing quality, inappropriate indexing, and poor format which could contribute to potential human error. Check lists are also not always used correctly; some operators or technicians will perform several tasks and then check them all off instead of checking each task as performed. This may be partially prevented and the reliability improved by using checklists that require information (such as a meter reading) to be written down. Additional reliability may be obtained through verification (a second person verifying that the performance of the first person is correct).

Stress: Reported data on stress and human behavior indicate that the error rate for a task has a relationship to the stress level perceived by the operator. A hypothetical relationship is shown in Figure A.3.1.

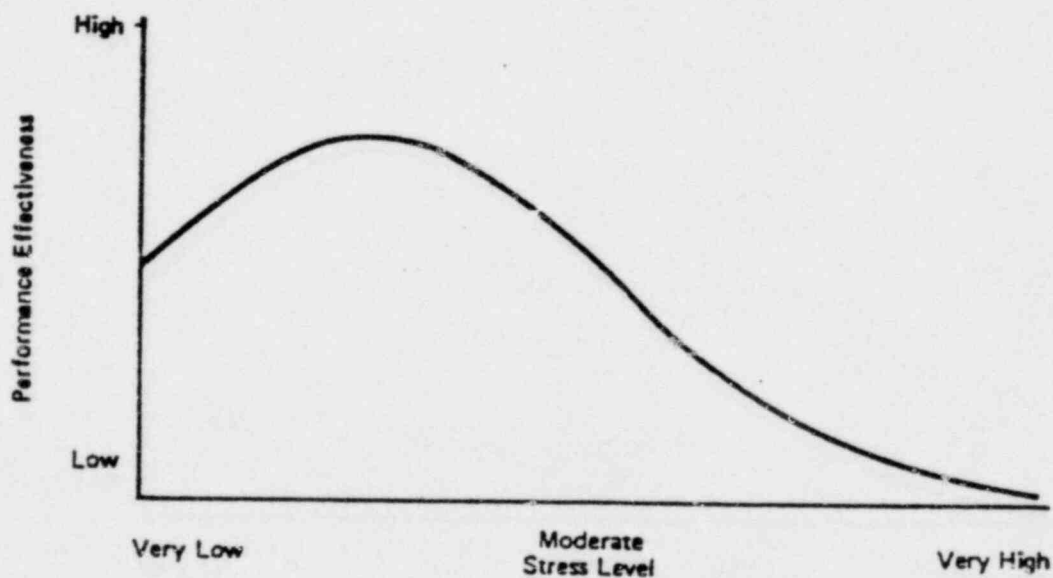


Figure A.3-1. Hypothetical Relationship Between Performance and Stress

As shown, when stress is low, a task is so dull and unchallenging that most operators would not perform at their optimal level. Passive-type inspection tasks are often of this type and can be associated with error rates of .5 or higher. When the stress level of a job is somewhat higher (high enough to keep the operator alert), optimum performance levels are reached. WASH-1400 determined that control room, maintenance, and calibration jobs were sufficiently challenging to maintain moderate stress and therefore maximum performance. When stress levels are still higher, performance begins to decline again, this time due to the effects of worry, fear, or other psychological responses to stress. At the highest level of stress, human reliability would be at its lowest level. WASH-1400 obtained a value of .2 to .3 as the average error rate for nuclear power plant personnel in a (continuing) high-stress situation such as the time following a large LOCA. This was considered to be conservative and variable for different situations.

Following a major accident, human error would be even higher than the high-stress value due to a probable incredulity response. Since the probability of a major accident is so small, for some moments a potential response would be to disbelieve panel indicators. Under such conditions, especially if false alarms had occurred frequently in the past, no action might be taken at all for at least one minute, and if any action were taken it would likely be inappropriate. WASH-1400 assessed that the error rate is .9 five minutes after an accident, .1 after thirty minutes, and .01 after several hours (Figure A.3.2). It was estimated that by 7 days after an accident there would be a complete recovery to normal, steady-state error rates. These values are based on the assumption that the nuclear plant is appropriately brought under control. For those cases where the situation persists or becomes worse, Swain (A.3-2) suggests that the error rate levels off at .25 after the initial peak.

A.3.2 Sources of Data

There are many sources of estimated human failure rates, but most are too specific or too general to allow easy comparison or averaging of values. Some of these data sources are presented here:

Nuclear Experience Data: Fullwood and Gilbert (A.3-3, A.3-4) used data reported to the NRC under the requirements of Regulatory Guide 1.16 in the form of all BWR and PWR citations in the Nuclear Safety Information Center (Oak Ridge, TN) up to the time of the search. There were 7,038 citations for LWRs, which were individually read to avoid misclassifications, and 1,490 or 21% contained identifiable human errors. Of this number, only 28 or 1.9% of the human factors citations were for compound human error. These events were grouped as appeared natural to the event descriptions.

The accumulated experience represented in the reports for 61 plants was 260 plant-years. The unprocessed categorization of data is presented in Table A.3.2. By using data on the number of people involved in various operational phases and making estimates as to task frequency, the data of Table A.3.2 were normalized and are presented in Table A.3.3.

Human Reliability Handbook: Swain and Guttman have prepared a handbook to aid in analyzing the reliability of human actions in a power plant. This handbook explains the basic terms, performance-changing factors, and human performance models. It also provides numerous examples of the application of analysis methods. Table A.3.4 summarizes some of the derived Human Error Probabilities (HEPs) for tasks in a nuclear plant. It should be pointed out that these data are not from objective observations, but instead are estimated from related (sometimes only marginally so) measurements of human performance. In general, Swain's handbook contains such a large amount of information on human failures that there may be several interpretations of the data or methods used to apply the data. Each interpretation may result in a different failure probability.

Table A.3.2

UNITED STATES NUCLEAR POWER PLANT EXPERIENCE CATEGORIZED BY TYPE OF HUMAN ERROR*

	Maintenance error	Failure to comply/complete procedure	Incorrect Instrument settings	Incorrect procedural sequence	Incorrect instrument used	Judgmental error	Misunderstanding of task	Communications failure	Failure to respond to alarm	Unidentified operator error	Clerical error	Administrative error	Procedural Deficiency	Analysis error	Design error	Fabrication error	Installation error	TOTALS
Reactor Type	340	277	137	40	29	43	26	9	28	27	15	48	53	27	193	77	121	1490
PWR	172	150	69	22	14	24	10	5	20	10	4	35	32	17	130	53	70	837
BWR	168	127	68	18	15	19	16	4	8	17	11	13	21	10	63	24	51	653
Reactor Mfg Co.																		
AC	9	4	5	3	3	2	-	-	-	3	-	-	1	-	4	2	-	36
B&W	42	36	20	11	5	10	3	-	5	3	1	14	12	5	34	14	16	231
CE	20	19	9	4	2	5	2	-	2	-	1	8	4	1	32	9	18	136
GE	159	123	63	15	12	17	16	4	8	14	11	13	20	10	59	22	51	617
WEST	110	95	40	7	7	9	5	5	13	7	2	13	16	11	64	30	36	470
Reactor Function Mode																		
Unknown or N/A	133	115	40	13	18	22	11	5	11	24	12	33	29	22	76	30	48	642
Preoperational	3	5	1	1	1	1	-	-	1	-	-	6	2	-	26	22	21	90
Operating unknown level	7	20	13	5	3	7	-	3	6	1	1	-	3	6	6	-	2	77
Startup	11	25	9	7	2	3	3	-	4	-	-	1	2	1	6	3	7	84
Run	112	72	48	7	2	7	10	2	5	1	3	6	12	4	49	10	26	376
Standby	3	2	-	-	1	-	-	-	1	-	-	-	-	-	-	-	1	8
Shutdown	46	27	20	4	2	6	2	2	1	-	-	3	6	-	24	7	14	164
Refueling	27	11	7	3	1	1	-	1	1	1	-	-	3	-	8	5	8	77
Totals	342	277	138	40	30	47	26	13	30	27	16	49	57	27	195	77	127	1518
Ranking	1	2	4	10	11	9	15	17	12	13	16	8	7	14	3	6	5	

*Reference A.3-3, A.3-4

Table A.3.3

ASSESSMENT OF HUMAN FAILURE RATES FROM
UNITED STATES POWER REACTOR EXPERIENCE*

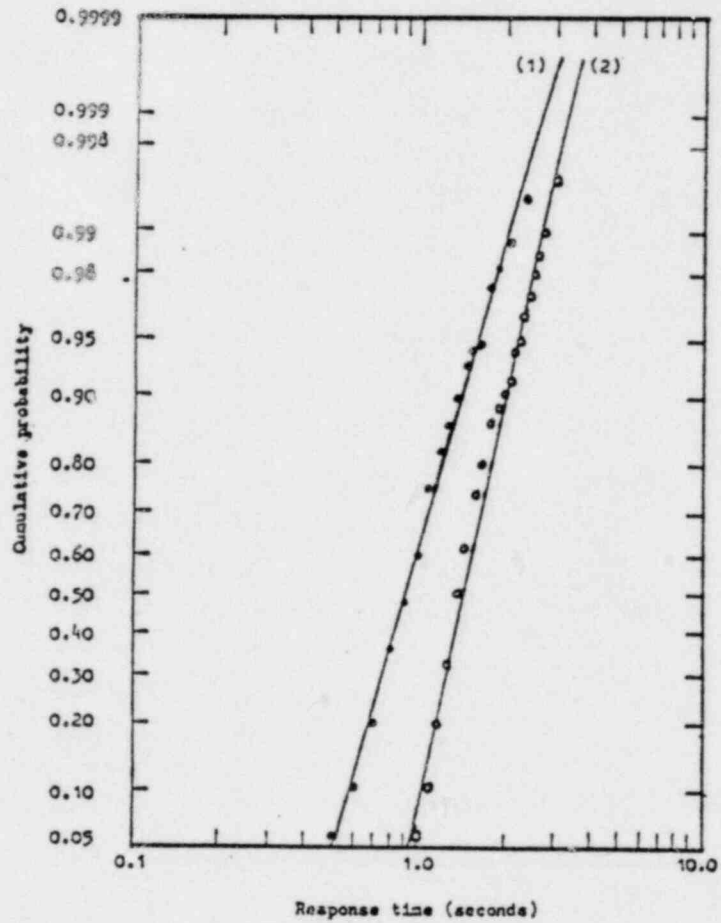
Item No.	Error Type	Events	Personnel/ Reactor-Year	Error Rate Per Man-Year	Error Rate Per Task
1	Maintenance	342	1368	2.5×10^{-1}	1.4×10^{-3}
2	Failure to Comply/ Complete	277	6156	4.5×10^{-2}	1.8×10^{-4}
3	Design	195	8291	2.4×10^{-2}	1.2×10^{-4}
4	Incorrect Settings	138	1368	1.0×10^{-1}	6.2×10^{-4}
5	Installation	127	1.9×10^4	6.7×10^{-3}	1.3×10^{-4}
6	Fabrication	77	1.9×10^4	4.1×10^{-3}	7.9×10^{-5}
7	Procedural Deficiency	57	6156	9.3×10^{-3}	3.8×10^{-5}
8	Administrative	49	6156	8.0×10^{-3}	3.2×10^{-5}
9	Judgmental	47	6156	7.6×10^{-3}	3.0×10^{-5}
10	Incorrect Sequence	40	6156	6.5×10^{-3}	2.6×10^{-5}
11	Wrong Instrument	30	1368	2.2×10^{-3}	1.4×10^{-5}
12	Failure to Respond	30	6156	4.9×10^{-3}	2.0×10^{-5}
13	Unspecified Operator Error	27	6156	4.4×10^{-3}	1.8×10^{-5}
14	Analysis	27	6156	4.4×10^{-3}	1.8×10^{-5}
15	Misunderstanding	26	6156	4.2×10^{-3}	1.7×10^{-5}
16	Clerical	16	6156	2.6×10^{-3}	1.1×10^{-5}
17	Communications	13	6156	2.1×10^{-3}	2.0×10^{-6}
18	Compound Personnel Errors	28	6156	4.5×10^{-3}	1.8×10^{-5}

*Reference A.3-3, A.3-4

Table A.3.4
HEPS FOR SELECTED TASKS ABOUT A NUCLEAR POWER PLANT*

Task	HEP
Walkaround inspections. Failure to recognize an incorrect status, using checklist correctly.	.01 (.003 - .03)
Walkaround inspections. Failure to recognize an incorrect status, using checklist incorrectly.	.1 (.05 - .5)
Walkaround inspections. Failure to recognize an incorrect status, no checklist. First walkaround.	.9 (.5 - .99)
Failure to use checklist correctly.	.5 (.1 - .8)
Failure to follow established policies or procedures.	.01 (.003 - .03)
Passive inspection.	.1 (.02 - .2)
Failure to respond to an annunciator (1 of 1).	.0001 (.00005 - .001)
Read annunciated lamp.	.001 (.0003 - .003)
Read digital display	.001 (.0003 - .003)
Read analog meter	.003 (.001 - .01)
Read analog chart recorder	.006 (.002 - .02)
Read a graph	.01 (.003 - .03)
Read printing recorder (cluttered)	.05 (.01 - .2)
Record more than 3 digits	.004 (.001 - .01)
Arithmetic errors	.03 (.01 - .1)
Failure to detect a deviant analog display during initial audit (with limit marks)	.001 (.003 - .003)
Check-read specific meters with limit marks.	.001 (.0003 - .003)
Check-read specific meters without limit marks.	.001 (.0003 - .003)
Check wrong indicator lamp in a group of similar lamps.	.003 (.001 - .01)
Failure to note incorrect status of an indicator lamp (in a group)	.99 (.97 - .997)
Failure to note incorrect status of a legend lamp (in a group)	.98 (.94 - .994)
Failure to remember oral instructions, 1 of 1	.001 (.0003 - .03)
Select wrong panel control:	
a. Among a group of similar controls	.003 (.001 - .01)
b. If functionally grouped	.001 (.0003 - .03)
c. If part of a mimic type panel	.0005 (.00005 - .005)
Set a multiposition switch	.001 (.0001 - .1)
Make a connector	.01 (.001 - .05)

*Reference A.3.2

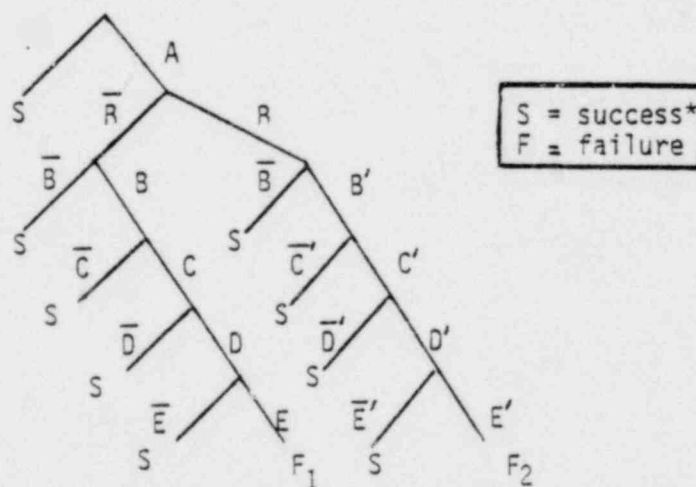


- (1) Lamp indication and audible alarm in research reactor control room - mean signal rate 1.5 per hour
- (2) Lamp indication and audible alarm in power reactor control room - mean signal rate 0.35 per hour

Figure A.3.3. Operator Response Tests in Reactor Control Room

is 10^{-2} . It is further assumed that half of the time this error leads to small (not unusual) miscalibrations on all future measurements, and half of the time it leads to large (very unusual) miscalibrations. Recovery of the setup error is entered into the estimate as follows: it was reasoned that when the technician discovered a small miscalibration of the first sensor, he would change the calibration. It was further reasoned that when the second sensor was also incorrect by a small amount, 30% of the time he would be suspicious and would recheck the setup. If the technician was not suspicious and the third sensor also had a small miscalibration, then 50% of the time he would become suspicious and recheck the setup. Finally, it is assumed that if the technician had not yet discovered the error, then he would not become suspicious after the fourth sensor. Similarly, if the test setup led to a large miscalibration error, the technician would recheck the setup 90% of the time after the second, and 99.9% of the time after the third. Again, if the technician did not discover the error after the first three sensors, he would also err on the fourth. The final assumption is that if the technician rechecked the setup, he would find the error and make the appropriate correction.

Using the probability tree in Figure A.3.4, the probability of miscalibrating all four sensors is approximately 2×10^{-3} .



*Key is on next page.

Figure A.3.4 Human Error Probability Tree
Describing Sensor Miscalibration

This evaluation is used for Limerick analyses for cases of normal transients and small LOCAs where it is presumed that minimal stress exists on the operator. For other, more stressful situations, the WASH-1400/Swain stress curve is used to establish the time-dependent operator behavior.

A.3.3.3 Operator Fails to Manually Initiate the Second Safety System Under Scram Conditions (Within 30 Minutes)

The handbook assumes that initiation of the first and second safety system is cued by the same or similar indications. The conservative assumption is made that failure to initiate the first system implies that the operator will not respond to the second, either. Similarly, it is assumed that if the operator initiates the first safety system, he will respond to the indication for the second with certainty. Thus, the probability of not responding to the indications of the second safety system is 10^{-3} .

Similarly, the probability for the operator to make an incorrect response is 10^{-3} and we can compute the probability of failure to initiate the second safety system can be computed as follows:

$$10^{-3} + (1 - 10^{-3}) \times 10^{-3} = 2 \times 10^{-3}$$

A.3.3.4 Initiate a Normal Plant Function Following a Reactor Scram

The annunciator response model of Table 20-4 in Swain is used. It is assumed that the operator may tend to focus on other displays, so the failure probability of initiating a normal plant function within a certain length of time is higher than that for initiating the safety systems. The probabilities are:

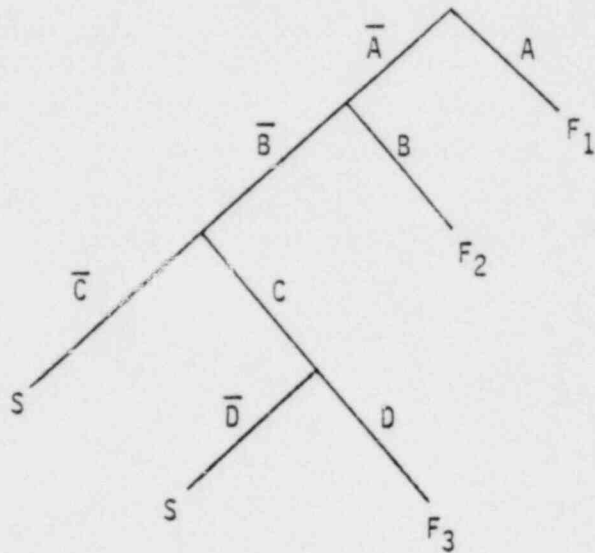
Failure to Initiate Within:	Human Error Probabilities
20 min.	.25
2 hrs.	.025
20 hrs.	.025

A.3.3.5 Turn Off Emergency System

An error-of-commission involving emergency systems cannot be assigned a probability using the techniques of the Swain handbook. As a gross estimate, the basic error-of-commission probability of 0.001 can be used. This number should be doubled for stressful situations. The Limerick analysis does not explicitly evaluate accident scenarios involving errors-of-commission by the operator.

A.3.3.6 Open a Single Manual Valve Within 30 Minutes

The failure probability of not properly opening a single manual valve is 0.02 under stressful conditions such as a 30 minute time limit. The probability of selecting the wrong manual valve from a group of similar valves is 0.01 under stress, and the probability of the valve failing is 0.001. In addition, the operator may fail to notice that the valve failed (probability 0.02, again under stress). Using the probability tree given in Figure A.3.6, the probability of failing to open the manual valve is determined to be 0.03. However, in reality, the manual valve may be some distance from the control room. Because of this, plus the time required to reach the decision that the valve must be open (especially if the operator does not have clear indication that the valve needs to be opened), the error probability is often taken to be approximately .9 to 1.0.



- A: Probability that the operator does not follow procedure (stress) = .02
- B: Probability that the operator selects incorrect valve from group (stress) = .01
- C: Probability that the valve fails = .001
- D: Probability that the operator fails to notice and to correct the failed valve (stress) = .02

Probability of failure (valve is not opened) =

$$\begin{aligned}
 F_1 + F_2 + F_3 &= A + (\bar{A} \times B) + (\bar{A} \times \bar{B} \times C \times D) \\
 &= .02 + (.98 \times .01) + (.98 \times .99 \times .001 \times .02) \\
 &\approx .03
 \end{aligned}$$

Figure A.3.6 Human Error Probability Tree Describing Failure to Open Single Manual Valve

A.3.3.7 Ensure Valves Are in Correct Line-Up

The failure of an operator to return both of two manual valves to the correct position is discussed as example 1.5 in Chapter 21 of Swain's handbook. Two major types of error are possible: (1) neglecting to reposition a valve following a TCIT; and (2) not opening a valve completely.

The handbook assumes that the line-up is scheduled using tags and checked by a second operator. The scheduled activity is regarded as an oral instruction, with a probability of .001 for failure of an operator to initiate the task. With only two valves, the probability of failure to restore the second valve is .003.

The probability of a valve failing is assumed to be .001, and the probability that an operator corrects for the failure is .01. According to Swain, the probability of a checking operator to make the same mistakes as the first operator is 10 times larger.

Using the probability tree given in Figure A.3.7, the probability of not ensuring the correct line-up is approximately .0001.

Given the events shown in Figure A.3.7, the probability of failure (both valves not lined-up) is:

$$\begin{aligned} &= F_1 + F_2 + F_3 \\ &= (AxA) + (\bar{A}xBxB') + (\bar{A}\bar{B}xCxD\bar{D}') \\ &= (.001 \times .01) + (.999 \times .003 \times .03) + (.999 \times .997 \times .001 \times .01 \times .1) \\ &\approx .0001 \end{aligned}$$

A.4 SYSTEM UNAVAILABILITY DUE TO MAINTENANCE DURING REACTOR OPERATION

One of the principal contributors to safety system unavailability may be the outage time associated with maintenance operations.* The reason for this unavailability is that while maintenance is occurring on one component in a system (or one leg of a two-leg system) that system may be incapacitated.** One available source of maintenance frequencies and durations based upon operating experience at nuclear power plants is WASH-1400. However, WASH-1400 does not state whether or not simultaneous maintenance activities are included and does not differentiate between on-line and off-line maintenance. The WASH-1400 data also include several startup problems in early BWRs which have been subsequently corrected. General Electric does not believe the data to be representative of present-day conditions. General Electric has performed a search of their Component Information Retrieval system and completed an analysis of the data. In addition, Philadelphia Electric has provided detailed maintenance information on each of the Peach Bottom 2 and 3 safety systems, based upon operating experience. Both of these sources confirm that maintenance unavailabilities are substantially less than assumed in WASH-1400. In addition, Philadelphia Electric Nuclear Plant operating philosophy dictates that all normal maintenance of safety systems be performed during outages when there will be no demand for the safety system. General Electric maintenance availability values were used in this analysis.

A.4.1 Calculated Maintenance Unavailabilities

For comparison purposes, maintenance information from WASH-1400 is provided. WASH-1400 assessed the mean time between component failures to be 4.55 months (.22 failures/month), while the mean time to repair (MTTR) is a function of both:

*WASH-1400 found on-line maintenance to be a major contributor to individual system unavailability; however, operating experience and maintenance philosophy of the operating PECO BWRs support a significantly lower estimate.

**Note that there may be some maintenance acts from which the operator can recover the system for use as a safety system; however, this is not considered in this model.

1. The component, and
2. The upper bound on the allowed system outage time (e.g., for HPCI it is 7 days with RCIC operational). The technical specifications define these combinations of system unavailabilities. The Limiting Conditions of Operation from the technical specifications for Peach Bottom* are reproduced in Table A.4.1.

The WASH-1400 evaluation of MTTR for components which are anticipated to lead to maintenance outages is summarized in Table A.4.2.

Table A.4.2

SUMMARY OF MEAN TIME TO REPAIR (HOURS) BY COMPONENT TYPE ASSUMING A LOG NORMAL DISTRIBUTION OF REPAIR TIMES AND A MAXIMUM ALLOWED OUTAGE TIME OF 7 DAYS (WASH-1400 ANALYSES)

Component Type	MTTR (hours)
Pumps	19
Valves	19
Diesels	21
Instrumentation (I&C)	7

The following brief summaries of each system provide the number of components by type, the evaluated mean time to repair (MTTR) for each component, and the calculated unavailability of safety systems due to maintenance while the plant is operating.

*As noted in the groundrules for this study, the Limerick technical specifications are not written or approved and therefore Peach Bottom technical specifications are used as typical.

Table A.4.1
LIMITING CONDITIONS OF OPERATION FOR THE PLANT USED IN THIS ANALYSIS

System/Component Out of Service	SYSTEMS REQUIRED TO BE OPERATIONAL											Shielding	
	Redundant Core Spray Subsystems	LPCI	Both Core Spray Subsystems	Remaining LPCI Subsystems	Remaining Containment Cooling Subsystems	Remaining Diesel Generators	Remaining Cooling Subsystems	Remaining Relief Valves	Remaining Recirc. Loop	Allowable Out of Service Time (days)	Shielding		
One Core Spray Subsystem	X											7	X
Both Core Spray Subsystems		X	X									2	X
One LPCI Pump		X	X									7	X
One LPCI Sub-system		X	X									7	X
Both LPCI Subsystems				X								2	X
Two RPSM Pumps				X	X							30	X
Three RPSM Pumps				X	X							15	X
Three Containment Cooling Subsystem Loops				X	X							7	X
All Containment Cooling Subsystem Loops												1	X
RPCI		X	X						X			7	X
ACIC								X				7	X
One ADS Valve									X			7	X
More Than One ADS Valve									X			7	X
One Diesel Generator		X	X							X		7	X
More Than One Diesel										X		30	X
Safety Valve Function of One Relief Valve										X		7	X
Safety Valve Function of Two Relief Valves										X		7	X
Safety Valve Function of More Than Two Relief Valves										X		1	X
One Recirculation Loop											X	1	X
More Than One Inoperable Control Rod in a 3x3 Array												1	X
Withdrawn Control Rod Which Can't be Moved -because of Possible Collet Housing Failure												2	X

NOTES

1. With irradiated fuel in reactor vessel and reactor in cold shutdown condition, both core spray subsystems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.
2. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of thirty days.

In the determination of the maintenance unavailabilities for HPCI, RCIC, and RHR, the number of components used in assessing the maintenance outages are for the specific system. Components involved in the room cooling and ventilation are not included in this estimate of maintenance unavailability.

HPCI: HPCI is a single leg series system as described in Appendix B. The following evaluation of HPCI unavailability due to maintenance is based upon the data and assumptions used in WASH-1400; a General Electric assessed value for HPCI maintenance unavailability (from BWR operating data available to General Electric); and Peach Bottom-specific data (see Table A.4.3).

Table A.4.3

SUMMARY OF MAINTENANCE ASSUMPTIONS USED
IN EVALUATING HPCI UNAVAILABILITY

Component	Number	MTTR (Hours)	Unavailability*
Turbine	1	19	5.80×10^{-3}
Pump	1	19	5.80×10^{-3}
Valves			
MOV	8	19	4.64×10^{-2}
AOV	1	19	5.80×10^{-3}
Turbine	2	19	1.16×10^{-2}
C&I	1 set	7	2.10×10^{-3}
TOTAL (WASH-1400 Assumptions)			7.75×10^{-2}
GENERAL ELECTRIC ASSESSED VALUE			1.0×10^{-2}

* Based on .22 failures per calendar month

Table A.4.8 gives a summary of safety-related systems required for normal operation. The operating experience data assembled to represent the maintenance on systems is compiled from data which results from normal plant operation where the maintenance represents operations carried out in accordance with the LCOs. For example, the maintenance unavailability for RCIC from operating experience data represents the unavailability associated with RCIC when HPCI is operating. Therefore, there is a condition on the RCIC operation requiring HPCI to be available which must be reflected in the fault tree model.

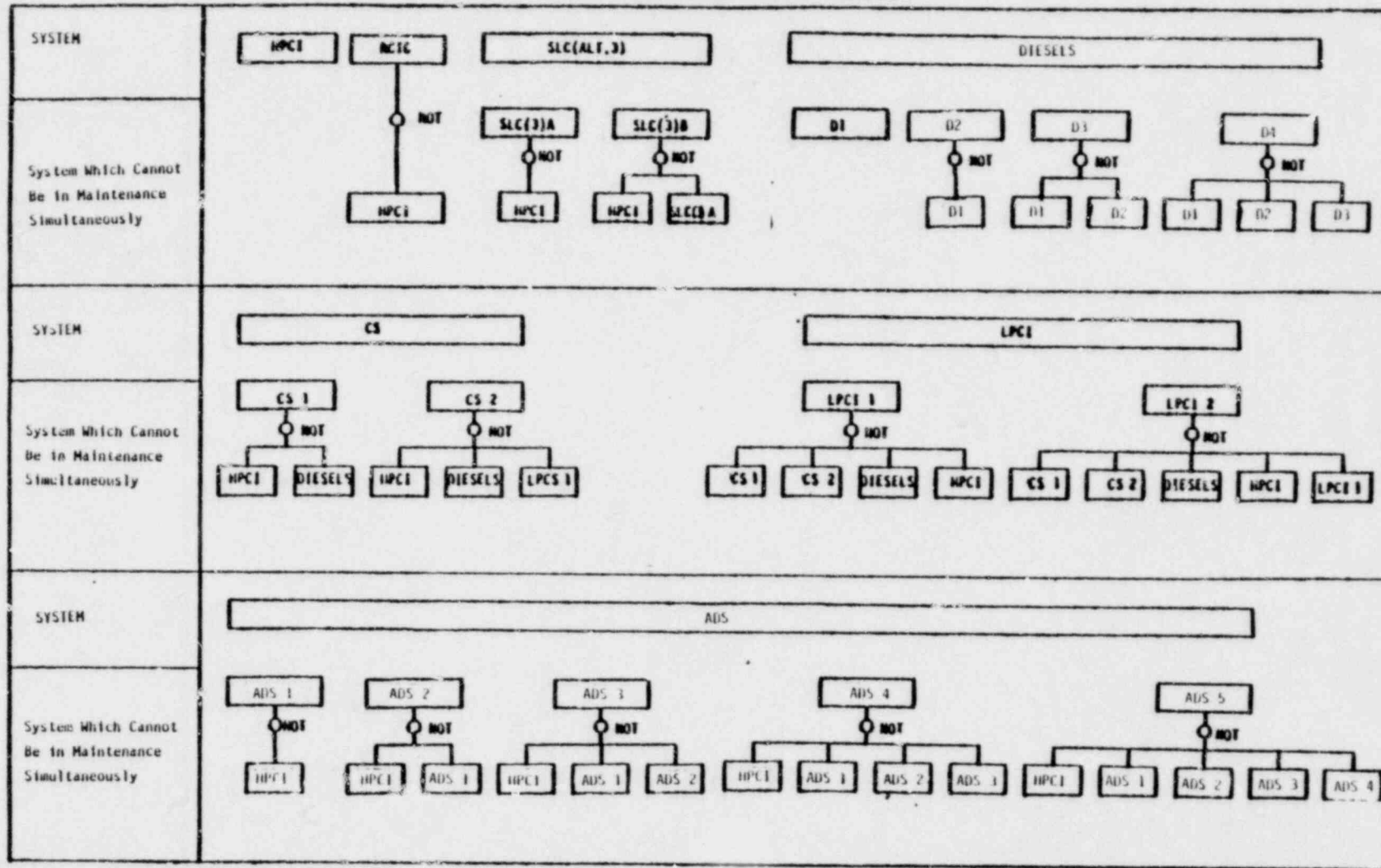
This section summarizes the relationship of the fault tree model logic to the probability that a system is in maintenance. Because of the dependencies among systems (summarized in the Limiting Conditions of Operation) and the "NOT" gate formalism used in this analysis, the input probabilities to the fault tree are not immediately obvious. Therefore, this section presents the boolean algebra used to derive these input probabilities in terms of the known probabilities.

The maintenance evaluation included in this analysis includes two contributions to system unavailability. These two contributions are:

1. The dependent portion of systems unavailable due to maintenance, which provides that a portion of the safety systems may be unavailable if the remainder of the safety systems are operational.
2. An independent portion of system unavailability which can be attributed to those cases in which both legs of a system are found to require maintenance. In general, the allowed outage time associated with these conditions is 24 to 48 hours. For this analysis, this condition is assumed to occur once every ten years, which corresponds to an unavailability of 2.74×10^{-4} to 5.48×10^{-4} .

Table A.4.8

SUMMARY OF THE MAINTENANCE LIMITING CONDITIONS OF OPERATION*
FROM THE PLANT TECHNICAL SPECIFICATION



*Safety Related Systems which are required to be Operational for Power Operation. This chart represents the Boolean Algebra for those Limiting Conditions of Operation having an exclusive NOT function.

The following discussion summarizes the derivation of dependent fault tree input parameter values for each system.

RCIC: The RCIC unavailability due to maintenance (Figure A.4.1) is combined with no HPCI maintenance. The assessed General Electric data, as shown in Table A.4.7, yields an estimation of the probability associated with the top event "RCIC in maintenance" (RCICTM). In order to determine RTM (Figure A.4.1), the following relationship is used:

$$RTM = \frac{RCICTM}{(1 - HTM)} = \frac{1.10 \times 10^{-2}}{1 - 1.0 \times 10^{-2}} = 1.1 \times 10^{-2}$$

The above is a simple example of a two-tiered relationship; however, as can be seen from the LCOs or the fault tree model, other systems have more complex relationships in defining the allowable maintenance which can be performed with the plant remaining operational. The remainder of this section is devoted to defining the input probabilities required for the fault tree model in terms of the known values from Table A.4.7.

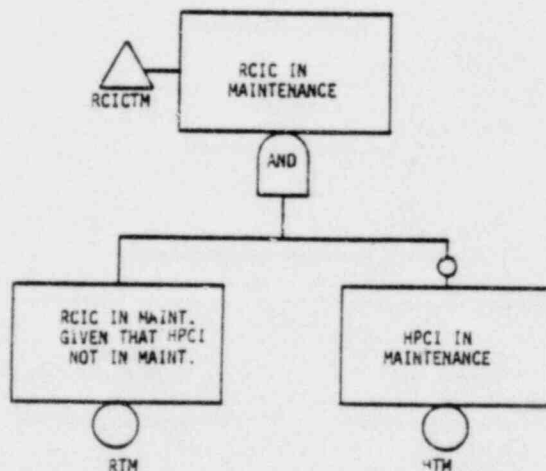


Figure A.4.1 The Structure of the RCIC Maintenance Unavailability in the RCIC Fault Tree

Diesels: The maintenance representation used for the Diesels is given in Figure A.4.2

Assuming the diesels are statistically symmetrical, each has an equal probability of being in maintenance as defined in Table A.4.7.

$$E1M \equiv E2M \equiv E3M \equiv E4M = 1.0 \times 10^{-3} \quad (A-1)$$

From the fault tree (see Figure A.4.2)

$$E4M = E4MM * \overline{ECUM3} \quad (A-2)$$

$$E3M = E3MM * \overline{ECUM2} \quad (A-3)$$

$$E2M = E2MM * \overline{E1M} \quad (A-4)$$

Where

$$\boxed{E1M = E1MM = 1.0 \times 10^{-3}} \quad (A-5)$$

$$ECUM2 = E2MM + E1M - (E2MM)(E1M) \quad (A-6)$$

$$\overline{ECUM2} = \overline{E2MM} * \overline{E1M} \quad (A-7)$$

Similarly,

$$\overline{ECUM3} = \overline{E3MM} * \overline{ECUM2} \quad (A-8)$$

$$\overline{ECUM4} = \overline{E4MM} * \overline{ECUM3} \quad (A-9)$$

Using (A-4):

$$E2M = E2MM * \overline{E1M}$$

$$\boxed{E2MM = \frac{E2M}{\overline{E1M}} = 1.0 \times 10^{-3}} \quad (A-10)$$

The probability of restoration of one DG before four hours is $11 \div 27 = 0.41$ based on the Peach Bottom data or 0.5 if a smooth (eye-fit) curve is used for statistical inference. These data do not vary significantly compared with the statistical uncertainties.

The inferred probability of restoration before thirty minutes is more uncertain. Directly from the data, the probability is 0.04; however, there are two problems. One is that there must be a minimum restoration time for personnel to arrive at the DG if the DG cannot be started from the control room. Thus, the probability should decrease at short times. The second problem is that the recording of short restoration times tends to be inaccurate because of a tendency to round off to larger values (less than one hour). This is suspected to be a reason for the small amount of data below 0.5 hours. However, because of the large uncertainty and small benefit perceived, no credit is taken for diesel restoration within thirty minutes.

These data were collected under fairly routine plant conditions, i.e., in no case was there an accident that could lead to hazardous conditions. The question arises as to how much faster could a DG be restored under "heroic" action. Based on human factor studies in the literature, it appears that the probability of restoration may increase greatly under accident conditions. People would move faster, but there are certain minimum transit times. There is also a negative effect; the increased stress could accentuate the error rate, causing more mistakes.

The emergency diesel data from Peach Bottom has been included in this analysis as the most appropriate to determine the basic diesel failure rate because of the following items:

1. The diesels are approximately the same size (2.6MW at Peach Bottom, 2.8MW at Limerick) and made by the same manufacturer (however, the details of design and auxiliaries are different).

2. Since Peach Bottom is also owned and operated by PECO, it has been subject to the same maintenance practices anticipated at Limerick.
3. Environmental conditions (seasonal variation in humidity and external temperatures) are approximately the same.
4. The data is based upon a detailed search of both tests (demands) and failures by the utility staff who have direct access to the plant logs.

Diesel data from several sources were obtained and analyzed to provide conditional failure rates for characterizing possible multiple diesel failures. These sources are discussed in the remaining sections of this appendix.

A.5.2 Plant X Data Assessment

In general, most sources of data do not provide detailed information as to the exact number of demands of diesels (success plus failures). One source does, however, provide detailed demand data on single, double, triple, and quadruple combinations of diesel demands and failures. In this assessment, these data will be referenced as Plant X.

Data were obtained from Plant X for its four-diesel population. These data yielded:

Diesels involved	Number of Demands	Number of Failures
1	133	25
2	23	2
3	71	1
4	67	0

These data can then be expanded to count trials and failure combinations. That is, there are two single trials in every double demand. Therefore, the following table can be formed:

	Number of Trials	Number of Failure Combinations
Singles	660	32
Doubles	638	5
Triples	339	1
Quadruples	67	0

Using this information, upper bound estimates can be obtained. These upper bound estimates assume a failure occurs on the next trial. Therefore, the probability of failure on demand becomes:

$$P(s) = P(\text{Single}) = \frac{33}{661} = 4.99 \times 10^{-2}$$

$$P(d) = P(\text{Double}) = \frac{6}{639} = 9.39 \times 10^{-3}$$

$$P(t) = P(\text{Triple}) = \frac{2}{340} = 5.88 \times 10^{-3}$$

$$P(q) = P(\text{Quadruple}) = \frac{1}{68} = 1.47 \times 10^{-2}$$

Note that these assumptions have resulted in an impossible result for quadruple failures. That is, quadruple failures cannot be more likely than lesser combinations. Therefore, another estimate must be obtained. This other estimate can be obtained by recognizing that there was one triple failure where a quadruple failure did not occur. Thus, there was one chance for a fourth failure given three failures. Again, using an upper bound approximation, gives:

$$P(\text{fourth given three failures}) = \frac{1}{2} = .5$$

Other conditional failures can be found as

$$P(\text{second given one failure}) = \frac{P(d)}{P(s)} = .188$$

$$P(\text{third given two failures}) = \frac{P(t)}{P(d)} = .626$$

$$P(\text{fourth given three failures}) = .5 \text{ (from above)}$$

These results are summarized under Plant X in the results tabulated in Table A.5.9.

A.5.3 Diesel Data Assessment From LERs and Direct Utility Response

A.5.3.1 Assessment of Utility Responses to EPRI Diesel Evaluation

Utility data were used directly to assess Zion and Cook diesels. A β factor approach was used as these plant data did not allow distinguishing multiple demands. Data were also studied for Zion and Cook from LERs to determine more precisely the course of failure. A rigorous statistical data analysis is not likely here due to the nature of the data base available.

Data were then collated to distinguish between independent and common-cause diesel failures. Three primary contributors to potential common-mode failures were identified:

- Human error (H)
- Design, fabrication, and installation errors (D)
- Procedural deficiencies (P).

Factors were therefore defined to account for the fraction of failures which were due to the common-cause contributors;

$$\beta_i = \frac{\Delta \text{ the number of common diesel failure causes of the } i^{\text{th}} \text{ type}}{\text{the total number of diesel failure causes}}$$

where:

β_H = Human contribution

β_D = Design, fabrication, and installation contribution

β_P = Procedural contribution.

Three β factors were defined because the plant-specific data showed that principal common-cause contributions varied among the facilities. In this manner, each plant could be assigned its own relative β factors.

In addition, the following general assumptions were made for multiple diesel plants of population N:

- Single unit demands are primarily due to tests.
- N unit demands are primarily due to non-test actuation signals.
- For $N > 2$; N-1 unit demands are actually N unit demands when 1 diesel is already unavailable.
- For $N > 2$; N-2 unit demands are primarily due to testing.

Therefore, based on the ratio of multiple to single demands from Plant X (given below), the average number of demands/diesel year (65.4), and the above assumptions, the following number of general multiple diesel demands were estimated:

Ratio of Multiple Diesel Demands

$$1:\text{Total} = .20$$

$$2:\text{Total} = .03$$

$$3:\text{Total} = .11$$

$$4:\text{Total} = .10$$

$$4+3+2:\text{Total} = .24$$

Estimation of Diesel Demands/Year

N=2

$$D(\text{total}) \approx (65.4 \times N) \approx 130$$

$$D(\text{double}) \approx .24 \times D(\text{total}) \approx 31$$

N=3

$$D(\text{total}) \approx (65.4 \times N) \approx 196$$

$$D(\text{double}) \approx .12 \times D(\text{total}) \approx 24$$

$$D(\text{triple}) \approx .12 \times D(\text{total}) \approx 24$$

$$D(\text{double+triple}) \approx .24 \times D(\text{total})$$

N=4

$$D(\text{total}) = (65.4 \times N) = 262$$

$$D(\text{double}) = .03 \times D(\text{total}) = 8$$

$$D(\text{triple}) = .11 \times D(\text{total}) = 29$$

$$D(\text{quad.}) = .10 \times D(\text{total}) = 26$$

$$D(\text{double+triple+quad.}) = .24 \times D(\text{total})$$

Using unavailabilities (based on multiple diesel failures) the following failure rate analysis can be made: (See Table A.5.8)

$$Q(2|2) = .10/\text{year}$$

$$\lambda_{31}(2|2) = \frac{.10}{31} = .0032/\text{demand}$$

$$Q(2|3) = .14/\text{year}$$

$$\lambda_{24}(2|3) = .0058/\text{demand}$$

$$Q(2|4) = .13/\text{year}$$

$$\lambda_8(2|4) = .016/\text{demand}$$

$$Q(3|3) = .04/\text{year}$$

$$\lambda_{24}(3|3) = .0017/\text{demand}$$

$$Q(3|4) = .03/\text{year}$$

$$\lambda_{29}(3|4) = .001/\text{demand}$$

$$Q(4|4) = .03/\text{year}$$

$$\lambda_{26}(4|4) = .0012/\text{demand}$$

A.6.2 WASH-1400 Assessment of Complete Loss of Offsite Power

The WASH-1400 estimate for loss of offsite power was larger than the estimate for the PJM interconnection, as shown below:

SOURCE	COMPLETE LOSS OF OFFSITE POWER INITIATOR
WASH-1400	0.18 per plant year
PJM Grid (see Section A.6.1)	0.053 per plant year

The PJM value (0.053) was used in this study.

WASH-1400 estimated the frequency of offsite power loss at 2×10^{-5} /hr. based on three occurrences in 1972 for 150,000 hours of operation. If it is assumed that plants are 100% available, this is equivalent to 17.1 plant-years of experience giving an estimated 0.18 offsite power loss/plant year. The estimate becomes 0.12 offsite power loss/plant year if the plants are assumed to be 70% available. Figure A.6.2 can then be utilized to predict frequency/duration characteristics of the offsite power outages. Results are as follows (assuming 70% average availability):

Frequency of offsite power loss (P_o) which is restored in less than 0.01 hour = 0.001/plant-yr.

P_o restored in 0.01 - 0.032 hours = 0.007/plant-yr.

P_o restored in 0.032 - 0.1 hour = 0.023/plant-yr.

P_o restored in 0.1 - .32 hour = 0.046/plant-yr.

P_o restored in 0.32 - 1.0 hour = 0.015/plant-yr.

P_o restored in 1.0 - 3.2 hour = 0.014/plant-yr.

P_o restored in 3.2 - 10.0 hour = 0.01/plant-yr.

P_o restored in greater than 10 hours = 0.004/plant-yr.

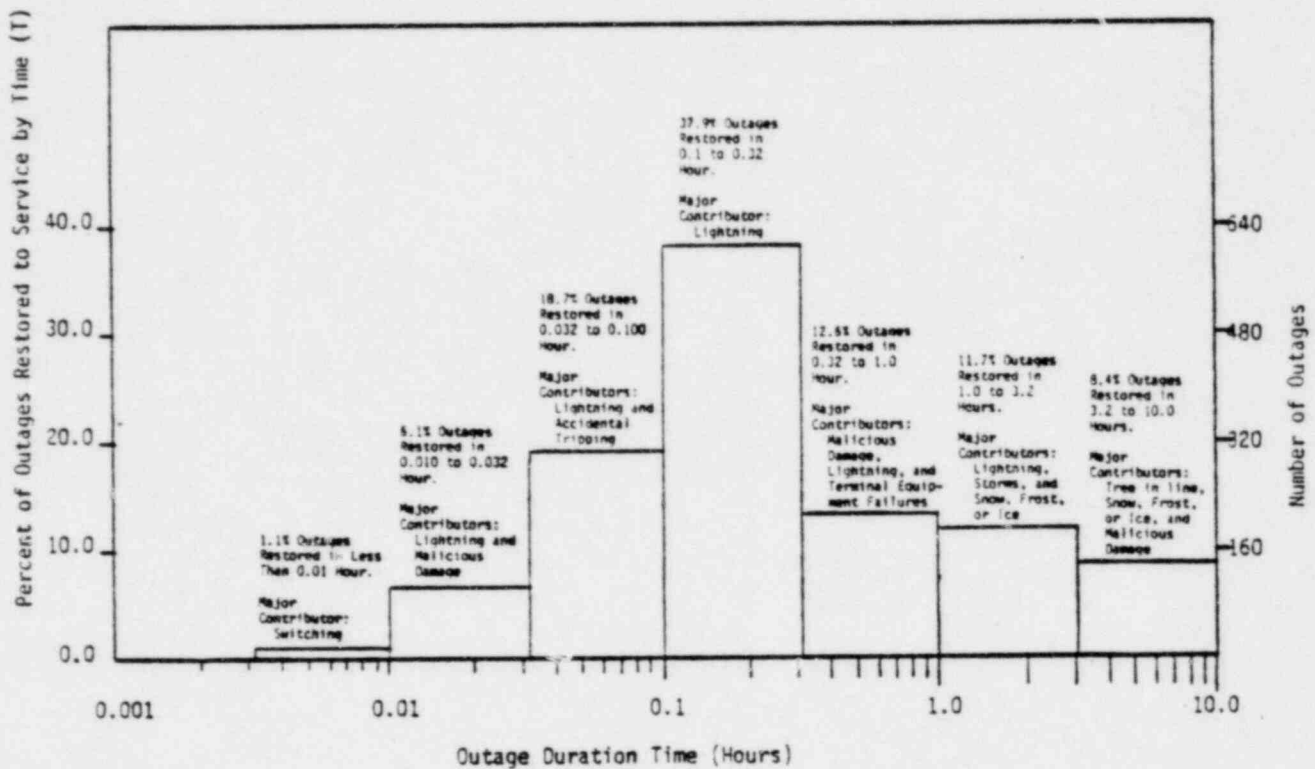


Figure A.6.2 Histogram -- Restoration of Transmission Line Outages

A.6.3 Loss of Offsite Power Resulting from Turbine/Generator Trip

In-plant transient events causing a turbine or generator trip result in a sudden loss of grid generating capacity. If the sudden loss of generator exceeds the transient stability limit of the local or regional grid system, then all offsite power to the plant could be lost. Based upon information developed for WASH-1400, the probability for complete loss of offsite power following a turbine or generator trip is assumed to be 1×10^{-3} . The probability for any particular plant could be lower depending on the transmission systems, the transient stability limit resulting from high installed capacity, extensive grid connections with other large utilities, and the number of 500 and 230 kV transmission lines connecting the plant to the grid. The probability of 1×10^{-3} is conservative for LGS because of the PJM Interconnection system and the use of five plant transmission lines and is included in the analysis.

APPENDIX B

The purpose of this appendix is to present, in a consolidated form, the following:

- A system description of the key systems which contribute to plant safety
- A schematic of the system arrangements
- The system level fault tree logic models and identification of the top level functional fault trees used in the evaluation of system and plant reliability
- Fault trees for generic components such as pumps, valves and turbines.

Included in this appendix are descriptions of the following systems, as identified by section:

B.1 High Pressure Coolant Systems

- B.1.1 HPCI
- B.1.2 RCIC
- B.1.3 CRD
- B.1.4 Condensate and Feedwater

B.2 Low Pressure Coolant Systems and Pressure Reduction System

- B.2.1 ADS
- B.2.2 LPCI
- B.2.3 CS

B.3 Decay Heat Removal Systems

- B.3.1 RHRSW
- B.3.2 Condenser
- B.3.3 Ultimate Heat Sink

B.4 Containment Systems

- B.4.1 Containment Over-Pressure Relief
- B.4.2 Containment Inerting

B.5 Electric Power System and Instrumentation

- B.6 Emergency Service Water System -- HVAC Pump Room Cooling
- B.7 Reactor Protection System
- B.8 Standby Liquid Control System

In addition, there are two summary sections which organize the individual system trees into the following:

- B.9: Generic component fault trees for potential failure modes of concern
- B.10: Functional level fault trees which combine system trees together to reflect the success criteria for various accident sequences

B.1 HIGH PRESSURE COOLANT SYSTEMS

B.1.1 High Pressure Coolant Injection System (HPCI)

Purpose

The primary purpose of the high pressure coolant injection (HPCI) is to maintain the reactor vessel water inventory under conditions which do not depressurize the reactor vessel.

Hardware Description

The HPCI system consists of a steam turbine-driven, constant-flow pump assembly and associated system piping, valves, controls, and instrumentation (see schematic in Figure B.1.1). Suction piping comes from both the condensate storage tank (CST) and the suppression pool (SP). Initially, water from the CST is used. Injected water is piped to the reactor vessel by way of the core spray loop B pipe. The steam supply for the turbine is piped from the main steam line in the primary containment. The steam piping has an isolation valve on each side of the primary containment. Remote controls for valve and turbine operation are provided in the main control room.

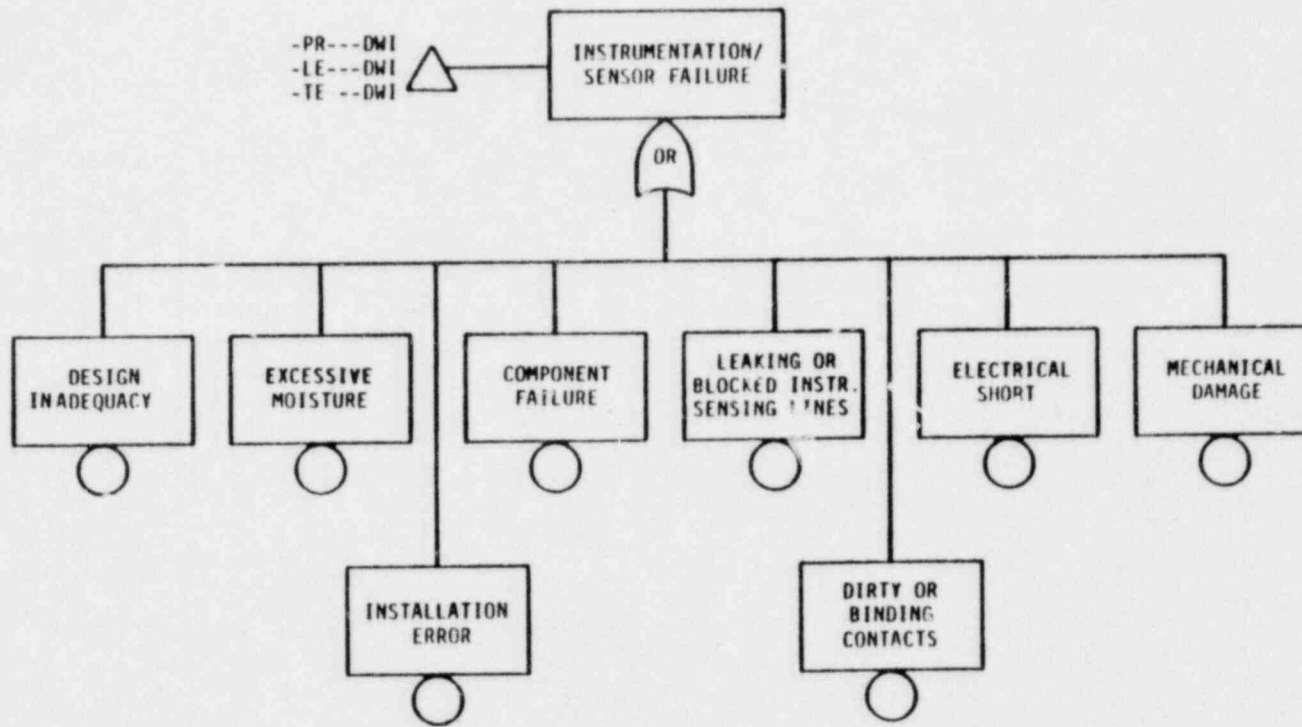


Figure B.9.8. Generic Instrumentation Fault Tree

Table B.9.3

INSTRUMENTATION/SENSOR FAILURE MODES
AND APPORTIONED FAILURE RATES FOR EACH

Rank	Failure Mode	Approximate Apportioning of Failure Rate
1	Component Failure	56%
2	Installation Error	14%
3	Dirty or Binding Contacts	13%
4	Leaking or Blocked Instrumentation Sensing Lines	6%
5	Excessive Moisture	5%
6	Design Inadequacy	3%
7	Electrical Short	2%
8	Mechanical Damage	1%
OVERALL		100%

REFERENCES

- B.9-1 W. H. Sullivan and J. P. Poloski, Data Summaries of Licensee Event Reports of Pumps at U. S. Commercial Nuclear Power Plants-January 1972 to April 1978, prepared for U. S. Nuclear Regulator Commission by EG&G. NUREG/CR-1205, January 1980.
- B.9-2 S. L. Basin and E. T. Burns, Characteristics of Instrumentation and Control System Failures in Light Water Reactors, EPRI NP-443, August 1977.
- B.9-3 E. Y. Lim, E. T. Burns, and R. J. Wilson, Component Failures that Lead to Reactor Scrams, prepared for Sandia Laboratories by Science Applications, Inc., May 1979.
- B.9-4 R. A. Hartfield, Setpoint Drift in Nuclear Power Plant Safety-Related Instrumentation, office of Operations Evaluation, U. S. AEC Report OOE-ES-003, August 1974.

B.10 FUNCTIONAL LEVEL FAULT TREES

The event trees are used to tie together the key system functions whose performance are required following the accident initiators. The functions appearing in the event trees may be simple or complex. This section discusses the fault tree model representation for the correct Boolean combination of these systems or functions. The Boolean combination is necessary in those instances where there are common dependencies among systems or functions. Some examples of such dependencies are: (1) the requirement that maintenance on one safety system be carried out exclusive of maintenance on certain other safety systems and (2) that sensors used in the initiation of one safety system are also used for another system (i.e., LPCI and ADS).

B.10.1 Transient Event Tree Functions

The first set of functional level fault trees are constructed to define, in fault tree format, the system success criteria for each of the functions of the transient event trees.

Initiators: These are input values determined based upon operating experience data.

Reactor Shutdown: This is treated separately in the ATWS event tree discussion. It is not developed as a system fault tree because of the criticisms such evaluations have received in the past. The single exception to this is the estimation of the failure to manually initiate a scram during an inadvertent open relief valve (IORV) incident.

Safety/Relief Valves Fail to Open: Many of the transients which normally occur during the course of a reactor plant life do not require the safety/relief valves to operate. However, there are a few transients which may demand that these valves operate successfully in order to protect the plant against ultimate reactor over-pressure. A fault tree description is used in assessing the likelihood of a failure to perform this function.

Loss of Coolant Makeup to the Reactor: The functional level fault tree for the loss of coolant makeup to the reactor is a combination of four functions listed in the event tree; these are:

- Feedwater availability
- HPCT or RCIC availability
- ADS operation
- Low pressure system operation.

A functional fault tree is used to combine these functions. The principal items to note are that:

1. The quantification of the fault tree depends upon the accident sequence being evaluated. For example, CRD coolant injection alone is not considered successful for any accident sequence evaluated for LGS. Also, feedwater has a lower probability of success during an MSIV closure than during a turbine trip.
2. The depressurization function (ADSX) is defined explicitly. This function provides the only access to the low pressure system capability.

Loss of Containment Heat Removal: The final function provided in the transient accident sequence event trees is the removal of heat from containment. This function can be fulfilled in the following ways:

1. The power conversion system (PCS) can be used to remove decay heat through the main steam lines to the condenser.
2. The RHR system can be used to remove heat from the suppression pool, using the safety/relief valves to provide the path from the reactor to the suppression pool, plus the RHR service water to remove heat from the RHR heat exchangers.
3. The RCIC system can be used in the steam condensing mode in conjunction with the RHR heat exchangers and RHR service water system to provide methods of:
 - High pressure coolant makeup
 - Direct heat removal from the primary system
4. In addition to the above methods of containment heat removal, there is the containment overpressure relief function which will satisfy this need temporarily (i.e., for periods not in excess of 3 days in certain accident sequences). This function is logically placed in the bridge tree so that the timing and constraints on its use can be understood by the reader. However, for quantitative evaluation it is included in the functional fault tree for containment heat removal.

A containment heat removal functional fault tree is used with the transient accident initiators. These systems have some interdependencies which require the fault tree evaluation of the systems. Similarly, accident sequences and groups of accident sequences require the same type of simultaneous evaluation to ensure that dependencies are properly evaluated.

Medium and small LOCAs are incorporated in the estimated MSIV closure initiator frequency for the calculation of risk due to failure to scram. An area of large uncertainty is the method of bringing the reactor from hot shutdown to cold shutdown, but this is not addressed in the current analysis.

Coolant Injection: The event tree function associated with coolant injection is governed by a functional fault tree which is identical to that given for transient events. The distinction to be drawn is in the evaluation of the fault tree. The differences in the quantification can be summarized as follows:

1. For Large LOCA:

System	Failure Probability	Reason
RCIC	1.0	Insufficient flow
HPCI	1.0	Insufficient flow
Feedwater	1.0	Unavailable due to MSIV closure
ADS	0.0	Not needed for large LOCA

2. For Medium LOCA:

System	Failure Probability	Reason
RCIC	1.0	Insufficient flow
FW	1.0	Isolation due to low reactor water level

3. For Small LOCA: the reliability of all systems is the same as used in the transient event trees with the exception that HPCI automatic initiation reliability is improved since high drywell pressure will occur.

Containment Heat Removal: The removal of heat from containment is a vital function in assuring the safe condition of the plant following a LOCA. The removal of heat from containment follows the same functional fault tree as developed for the transient events with the following exceptions:

1. Large LOCA:

System	Failure Probability Used in the LGS Analysis	Reason
Power Conversion System	1.0	Isolation of containment from the main condenser on low reactor water level
RCIC in the Steam Condensing Mode	1.0	Loss of steam to the RCIC turbine
Containment Overpressure Relief(COR)	1.0	Potential for radiation inside containment

2. Medium LOCA:

System	Failure Probability Used in the LGS Analysis	Reason
Power Conversion System	Reduced from transient event tree	Isolation immediately following LOCA: increased probability of failure to recover from isolation
RCIC in Steam Condensing Mode	1.0	Loss of steam to the RCIC turbine following depressurization

3. Small LOCA: the quantification of the small LOCA containment heat removal function is the same as that for the turbine trip event tree.

B.10.3 ATWS Functional Fault Trees

The functional events considered in the ATWS event tree evaluation are similar in most ways with those noted for transients. There are, however, some unique differences which require separate evaluation and these are discussed in this section. It should be carefully noted that the success criteria used in the construction of the functional level fault trees is that identified in Section 1.5 and reflects the General Electric evaluation of BWR/4 systems capability under ATWS conditions. Specifically, the system capability is based upon unpublished GE analysis and takes advantage of best estimate values for system flow and performance capability rather than the usual conservative values used in design basis analyses. In addition, containment capability beyond that usually acknowledged in design basis evaluations has been utilized.

Initiators: These input values are based upon the General Electric evaluation of operating experience data and include demands from all power levels. There is a discrimination among the types of transients in order to treat the dependent effects of the plant system on the initiating event as precisely as possible.

Reactor Shutdown: The ATWS event trees provide the vehicle in the Limerick analysis for treating the consequences associated with reactor shutdown, i.e., insertion of sufficient negative reactivity into the reactor core. Reactor shutdown for the LGS plants can be accomplished successfully through any of the following:

1. Insertion of the control rods by automatic action of the reactor protection system or by manual operator action. This is event item C_E in the ATWS event trees.

2. Insertion of the control rods by automatic action of the diverse and redundant backup system known as the Alternate Rod Insertion (ARI) system. This is referred to as Event V in the event trees. This backup system requires that the following additional features operate:

- The control rod mechanical operation be functional
- The recirculation pump trip (RPT) be functional.

At present, the design details of the logic, power, and sensors for RPS and RPT are not available for Limerick; therefore, possible dependencies between RPS and RPT have not been analyzed in detail. The possibility of a commonality between the logic and sensors has not been explicitly evaluated. The assumptions made in the analysis are:

- RPS and RPT are separate and diverse.
- The reliability of the RPS is as specified by the NRC characterizations in NUREG-0460.
- The RPT reliability is that specified by GE and assumed by the NRC in NUREG-0460.

3. Insertion of negative reactivity via the Standby Liquid Control (SLC) system which injects a sodium pentaborate solution into the reactor. This backup system is designed to safely shutdown the reactor in the unlikely event that the control rods cannot be inserted into the core.

Poison Injection: While poison injection was discussed briefly above under Reactor Shutdown, it needs further discussion since its partial operation can also be successful if other systems operate successfully. The specific points to be gleaned from an examination of the ATWS event trees are:

1. Loss of all SLC system capability and loss of control rod insertion will lead to a Class IV type event sequence.
2. Loss of one SLC system pump (i.e., half capacity) makes RCIC alone unacceptable to maintain adequate core coolant inventory and requires RHR initiation in a very short time, i.e., on the order of 10 minutes.

Adequate Pressure Control(N): For ATWS events the operation of the safety relief valves is required since there is a rapid pressure rise in the primary system. The number of valves required for operation under this remote postulated event is larger than that used in the typical transient event, but the dominant failure mode remains a common-mode failure of a large number of valves.

Safety Valves Reclose(P): Stuck-open safety valves are an undesirable event at any time and during an ATWS will tend to aggravate the situation by:

- Eliminating RCIC as a successful injection mode
- Requiring both RHR heat exchangers for successful containment heat removal.

The probability of failure of the safety relief valves in the stuck-open position following an ATWS is estimated to have a higher probability than during a normal transient.

Coolant Injection: The success of coolant injection during an ATWS event requires operation of one of the high pressure injection systems: HPCI, RCIC, or Feedwater. The use of low pressure systems may result in unacceptable dilution of the boron (LPCI overfilling the vessel or ADS initiation) which may lead to an unacceptable plant condition such as high reactor or containment pressure.

The probabilities of each of the system level functions in the fault tree are dependent on the ATWS initiator. In addition, considerations beyond those included in the transient and LOCA event trees are included in the ATWS quantification for the following reasons:

1. For ATWS accident scenarios, HPCI system has been modified to include a failure of the HPCI to restart once it has been turned off (i.e., isolated) by the high pressure spike which accompanies ATWS. In addition, there is an increased probability of premature HPCI shutoff due to high suppression pool pressure.
2. RCIC pump seals are generally considered marginal at elevated suppression pool temperature. Therefore, the RCIC system unreliability is evaluated as higher than normal since it is available during an ATWS event where suppression pool temperature may be above 140°F. In addition, the same logic consideration noted above for HPCI applies to RCIC.
3. During ATWS events where RCIC alone is a successful coolant makeup source, RCIC could be lost due to high containment pressure (25 psig) leading to a Class III event. RCIC loss for non-ATWS transients leads to Class II events.
4. IORV-initiated transients during which control rods cannot be inserted have some special characteristics which make the quantitative evaluation slightly different than for other transients. The normal method of recovering from an IORV is to depressurize, but ADS should not be initiated during an ATWS. Thus, the probability that the operator will fail to inhibit ADS needs to be assessed.

Containment Heat Removal: The evaluation of adequate containment heat removal following an ATWS is strongly dependent upon the ATWS initiator. For turbine trip cases where the MSIVs remain open, it is assumed that the power conversion system is more than adequate to remove the heat from containment. However, for all other cases (i.e., MSIVs closed) it is assumed that the power conversion system is unavailable due to the inability to reopen the MSIVs in time for successful PCS operation. The two paths available for adequate containment heat removal are:

1. The RHR system
2. The containment overpressure relief system.

Successful operation of these systems is dominated by the reliability associated with correct operator action.

rapidly over a central section and that the molten core appears in the lower head simultaneously with completion of core melt. The model also includes a physical representation for the water displaced by the molten debris and a film boiling process occurring at the molten debris-water interface. It does not assume any fragmentation of the molten core so that there is no intimate contact of the molten debris with the water. The water in the bottom head is displaced to the top of the molten core and is boiled off. The metal/water reaction and consequent hydrogen production is not modeled in this sub-code.

The PVMELT model assumes that conduction dominates the transfer of decay to debris/water and debris/wall interfaces and that constant thermo-physical properties exist. The heat transfer calculations are performed through a one-dimensional transient analysis at the RPV centerline and as a lumped parameter analysis for the vessel insulation. The PVMELT model assumes that the molten steel is promptly transferred to the upper surface of the molten debris layer and that thermal resistance of the transferred steel is negligible.

PVMELT is called by CONTEMPT-LT when BOIL has calculated (80%) core melt. This assumption is the same as was used in WASH-1400. The molten core appears at the bottom of the RPV simultaneously with the 80% core melt. Once PVMELT has calculated RPV failure, the primary system compartment disappears with its releases being incorporated into the drywell compartment.

PVMELT printout frequency is specified by CONTEMPT-LT with output as follows:

- Normalized wall thickness of RPV head
- Water overburden thickness
- Applied stress
- Yield stress.

C.2.4 INTER

INTER, the last of the major sub-codes called by CONTEMPT-LT, models core-concrete interactions by calculating the rate of penetration of concrete by a molten LWR core, and simultaneous generation of gases.

INTER assumes convective stirring of the melt by evolved gases, admixture of concrete decomposition products to the melt, chemical reactions, radiative heat losses, and variation of heat transfer coefficients with local pressure. INTER models the molten core as a two-phase melt (metallic and oxidic). Each layer is considered to be well-mixed and isothermal in its interior as long as the layer is molten. Heat transfer from layer to layer takes place across a boundary layer or film whose thickness varies with the violence of mixing. The two main layers are assumed to be in intimate contact with each other (there can be a vapor layer at the interface with the decomposing concrete). The thickness of the boundary layer can be different for each main layer; however, in each layer, it is uniform around the periphery of the layer. Heat is radiated to the containment, conducted into the concrete, and interchanged between the layers.

INTER models the molten core as a hemispherical segment intersected by a cylinder with geometry changes as the problem advances by material interchange between the layers. Iron oxides created by reaction of the steam with iron in the metallic part of the melt are assumed to be rapidly incorporated into the oxide layer. Solid or liquid decomposition products are assumed always to go promptly to the appropriate melt layer. However, gaseous products will not pass through the melt if the interface with the concrete is vertical.

INTER also assumes gas-induced insulation cells (Figure C.2). In a normal cell, more gas passes through the outside of the melt and circulation follows as shown in Figure C.2(a). However, if the lower layer is hotter than the upper layer, more gas flows through the center and the circulation direction can be reversed. A circulation cell will be modeled if the material is molten and there is appreciable gas flow. The intensity

- The MSIVs are assumed closed and feedwater to the RPV is lost.
- In addition to the failure of the control rods to insert, the liquid poison injection system is also postulated to fail for the purposes of this sequence calculation.
- HPCI is modeled to turn on at low RPV water level and turn off permanently on high pressure turbine exhaust. This is as designed and, therefore, is a high probability occurrence.
- The RPV is assumed to be at high pressure and maintained below a certain pressure (high setpoint of the relief valves) by relieving to the wetwell.
- The containment is assumed to be intact and conditions in the various compartments to be normal operating conditions at the start of the calculations.
- Normal leakage from the containment to the reactor building is modeled.

The code has been modified in this sequence to track the water level in the RPV. The code considers that the part of the core which is covered is at 30% power while the exposed core follows the decay power curve. The core melt occurs due to loss of coolant inventory. The molten core from the ruptured RPV is modeled in two ways: (1) drop on the diaphragm floor, interacting only with concrete inside the pedestal; or, (2) drop on the diaphragm floor and flow through the doorway to interact with concrete of the entire floor. (See Table C.1 for inputs).

C.3.4 Failure of Coolant Inventory Makeup Following an ATWS with Containment Failure Prior to Core Melt (Anticipated Transient without Scram - Case 2)

The last type of sequence calculated using the INCOR package is an ATWS without adequate containment heat removal which leads to containment overpressure failure prior to core melt, referred to as a Class IV accident sequence.

The assumptions in the sequence are similar to the third sequence with the following exceptions: HPCI is allowed to stay on even after the high exhaust pressure is reached for the HPCI turbine. The pressure in containment increases until failure pressure is reached. It is assumed that at containment failure, HPCI fails. Therefore, containment failure occurs prior to Core Melt, RPV Meltthrough, and Core/Concrete Interaction for this ATWS case.

The code has also been modified in this sequence to track the water level so that the part of the core which is covered is at 30% power while the exposed portion follows the decay power curve. The core melts due to loss of coolant inventory and eventually melts the RPV to drop onto the diaphragm floor. The molten core is only modeled to drop onto the diaphragm floor and flow through the doorway to interact with the entire floor. (See Table C.1 for inputs).

C.4 INCOR RESULTS: OVERVIEW OF PRESSURES AND TEMPERATURES CALCULATED TO OCCUR WITHIN CONTAINMENT FOLLOWING CORE MELT SCENARIOS

INCOR results are only used to calculate the containment conditions, which are then used for the radionuclide release fraction calculations. In the cases where INCOR did not perform the calculations out to diaphragm floor failure, some extrapolations using simplified models were done to predict: (1) the pressure in containment by estimating the steam pressure and the pressure due to the gases produced from the melting concrete; and (2) the time of floor failure by comparing the penetration rate and decomposition rate from various INCOR runs.

C.4.1 Containment Pressure Temperature Response During Postulated Core Melt

The RPV pressure for each sequence oscillates as shown in Figure C.7. This pressure oscillation is around the SRV setpoint. The size of the oscillation is dependent on the timestep used in the calculation and

the number of SRVs. A larger timestep or a larger number of SRVs depletes a greater amount of steam from the RPV, resulting in a larger calculated pressure reduction. Also, as the accident progresses, the steam generation rate for each sequence decreases with time since the core power is decreasing. For some sequences, the calculated steam volume is drastically reduced in a timestep so that the steam generated in later timesteps is not enough to make up the depleted volume and the total pressure decreases. Once enough steam is generated to overcome this calculated depletion, the pressure starts to increase until the SRV setpoint is reached. This phenomena also occurs during core melting. However, the pressure in this case decreases until the molten core drops to the bottom of the reactor vessel where film boiling (steam generation) of the water occurs and the pressure then starts to increase once again.

The steam relieved from the RPV is dumped into the suppression pool. In each sequence, the pool is subcooled. The containment pressure increase is very slow until saturation is reached at which point evaporation starts to occur and drives the pressure at a faster rate. The time at which saturation is reached is dependent upon the amount of flow through the SRVs from the RPV. Those sequences which have a larger number of SRVs actuated (ATWS with HPCI unavailable and ATWS with containment failure) produce a much greater steam flow to the wetwell pool and saturation is reached very quickly (see Figures C.12 and C.13). Also, for these two ATWS sequences, makeup water (HPCI) is continually added to the RPV during a portion of the sequence. The HPCI eventually becomes unavailable in both ATWS sequences due to high exhaust turbine pressure trip or containment failure at which point the water boils off in a relatively short time and the core reaches its melting temperature. Since the core is totally uncovered, the steam generation rate is very slight and is insufficient to overcome the depletion resulting in total pressure decrease. During this time there is no SRV flow and the pressure and temperature in the wetwell remains fairly constant. However, for the LOUV sequence, core melt is initiated soon after containment isolation and the total steam generation to the wetwell pool is smaller than in the ATWS sequences and

saturation is not reached prior to the RPV rupture; therefore, the pressure rise is very slight (see Figures C.8 and C.9). In the TW sequence the pressure rises slightly until saturation is reached and then increases rapidly (see Figures C.10 and C.11).

C.4.2 Containment Pressure and Temperature Response During Reactor Pressure Vessel Failure Phase

The PVMELT module of INCOR calculates RPV failure by creep rupture failure of the bottom head of the vessel.* This type of rupture can create a large pressure spike in containment. However, it has also been postulated** that the molten core preferentially melts through the BWR control rod penetrations in the RPV bottom head creating small holes and therefore slowly reducing the pressure in the reactor system during RPV melt, so that in failure there is no large pressure spike in containment. The second assumption is used in the LGS analysis to determine the pressure rise in containment since it appears more appropriate for BWR reactor vessel designs.

C.4.3 Containment Pressure-Temperature Response During the Phase of Corium-Concrete Interaction

The next calculational phase involves the time once the molten core has failed the RPV and dropped onto the diaphragm floor in the pedestal region directly below the RPV. In this phase, which includes the corium-concrete interaction and the reaction by-products, the conditions of the core following RPV failure are fairly similar for the sequences considered in the LGS analysis. Therefore, the evaluation of the concrete-corium interaction is performed with INCOR and used for each class of accident sequence. There are two bounding cases which have been run to establish the range of potential uncertainty during this phase of the INCOR calculation.

The pedestal wall surrounding the RPV has a doorway flush with the diaphragm floor. It is uncertain whether the molten core will: (1) case 1:

*Applicable primarily to PWRs which have few penetrations of the bottom head.

**Appendix H discusses the more likely mode of RPV failure for BWRs.

stay inside the pedestal region; or, (2) case 2: flow through the doorway and spread across the floor outside the pedestal. Both cases are analyzed. It is assumed that the diaphragm floor will fail structurally at approximately 70 cm or two thirds of floor penetration (before the molten core melts through the floor).

The molten core interacting with the diaphragm floor inside the pedestal (Case 1) versus the entire diaphragm floor (Case 2) produces different concrete penetration, concrete decomposition, and non-condensable concrete generation rates, thus affecting the pressure-temperature conditions inside containment.

At RPV rupture, molten core conditions for Case 2 in each of these sequences are similar. It is therefore assumed that the INTER calculations for the ATWS sequence with containment failed are applicable for the TQUV and ATWS (with HPCI Failure) Case 2 sequences. Therefore, one INCOR calculation of the time to diaphragm floor failure is used to characterize the following Case 2 sequences:

- ATWS - with containment failure occurring prior to core melt initiation (Class IV)
- ATWS - with HPCI unavailable during the sequence and with containment intact (Class III)
- TQUV - with loss of coolant inventory and core melt initiation with containment intact.

The INCOR results of the corium-concrete interaction phase are exemplified in Figures C.14 to C.16. One assumption made in these analyses which differs from that used in the REACT/CORRAL calculation for radionuclide release is that in the CORRAL calculations ten percent of the core is taken to directly interact with the suppression pool leading to an oxidation release*. However, the INCOR calculations do not include the pressure rise due to the steam generation from the ten percent steam-core interaction.

*The possibility of a coherent steam explosion which would lead to immediate containment failure is considered unlikely and is treated separately in the release fraction calculation discussed in Appendix D.

It should also be noted that the amount of hydrogen generated during a postulated core melt scenario is an area of uncertainty. The INCOR hydrogen production model during INTER (see Section C.2.4) is similar to that used in WASH-1400 and may tend to underpredict hydrogen production during such sequences. The pressure and temperature curves for each sequence consider neither RPV pressure relief during RPV melt nor steam explosion after RPV rupture. The large temperature rise seen on Figure C.13 is caused by the dumping of the mass and heat from the reactor system into the drywell.

The drywell conditions for each sequence (Figures C.3 - C.6) parallel those in the wetwell (for example, see Figures C.4, C.10, and C.11). As the pressure builds up inside the wetwell, it is relieved to the drywell through vacuum breakers once a specified pressure differential is reached. Therefore, a pressure increase in the wetwell causes a pressure (and temperature) increase in the drywell. Upon RPV rupture, drywell conditions are no longer determined by the wetwell. (The uncertainties, assumptions and considerations mentioned previously are also applicable to the drywell.) The molten core/concrete interaction now exerts the greater influence on containment conditions. The rate of concrete decomposition/penetration and the surface area over which the molten core is acting mainly determines the amount of non-condensibles generated. These gases control the pressure during this period of the analysis. As the core penetrates the diaphragm floor, its cooling and its rate of decomposition and penetration decreases, thereby decreasing the production of gases until the pressure inside containment becomes fairly stable (as demonstrated in Figures C.14 - C.16).

C.4.4 Summary of Individual Sequence Results

Using the assumptions and methodology described in sections C.4.1 through C.4.3, the results of the individual accident sequences are summarized in the following sections.

REFERENCE

- C.1 "ANS Decay Power Curve", Nuclear Safety, Volume 18, No. 5
Figure 14, September - October 1977.

TABLE C.1
SIGNIFICANT INCOR INPUTS

ANALYSIS ITEM	VALUES
Compartment Description Cards	
Total compartment volume (ft ³)	
RPV	21195.
Wetwell	2.80E5
Drywell	2.484E5
Rx Building	2.0E6
Volume of liquid pool (ft ³)	
RPV	11770.9
Wetwell	1.186E5 (1)
Drywell	0.
Rx Building	0.
Temperature of vapor region (°F)	
RPV	545. (2)
Wetwell	95. (2)
Drywell	150. (2)
Rx Building	90. (2)
Temperature of liquid pool region (°F)	
RPV	545. (2)
Wetwell	95. (2)
Drywell	150. (2)
Rx Building	90. (2)
Total compartment absolute pressure (psia)	
RPV	1020. (3)
Wetwell	15.45 (3)
Drywell	15.45 (3)
Rx Building	14.624 (3)
Relative humidity of vapor region (%)	
RPV	100
Wetwell	100
Drywell	20 (4)
Rx Building	45
Horizontal cross-sectional area of compartment (ft ²)	
RPV	0.
Wetwell	5.7E3
Drywell	5266.
Rx Building	0.

ANALYSIS ITEM	VALUES
Penetration Leakage Specification Cards	
Relief Valves	(5)
Identification number of compartment that leakage is into	2
Area of throat for leakage calculation(ft ²)	.5732 (6)
Ratio of throat area to exit area	0.0
Ratio of throat area to inlet area	1.0
Constant multiplier for leakage calculation	1.0
Drywell	
Identification number of compartment that leakage is into	5
Area of throat for leakage calculation(ft ²)	.208 (7)
Ratio of throat area to exit area	0.0
Ratio of throat area to inlet area	1.0
Constant multiplier for leakage calculation	1.0
Rx Building	
Identification number of compartment that leakage is into	0
Area of throat for leakage calculation(ft ²)	.429(8) 81.0 (9)
Ratio of throat area to exit area	0.0
Ratio of throat area to inlet area	1.0
Constant multiplier for leakage calculation	1.0
Vertical Vent (Mark I and Mark II) Pressure Suppression System	
System Control Card	
Number of downcomers in normal vent system	87
Ratio of fraction of liquid water entering normal vent system to fraction of liquid water in drywell atmosphere region	.5
Convergence criterion for vent flow	0.1
Miscellaneous Vent Data Card	
Vent submergence (ft)	10 (10)
Absolute roughness of inside wall of vent exit pipe (ft)	1.5E-4
Irreversible energy loss coefficient for incompressible single-phase flow	2.5
Inside diameter of vent opening (ft)	1.9375

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TABLE C.1 (continued)

ANALYSIS ITEMS	VALUES	ANALYSIS ITEMS	VALUES
Vacuum Relief System Card		Initial thickness of pressure vessel bottom head (ft)	.7381
Pressure difference (lb _f /in ²) at which vacuum breakers between wetwell and drywell open	1.75	Thermal conductivity of RPV bottom head material (Btu/hr-ft-°F)	18.0
k_{VR} , single-phase irreversible loss coefficient for vacuum relief system	5.7 (11)	Density of bottom head material (lbm/ft ³)	460.0
A, flow area of one vacuum breaker (ft ²)	2.05	Specific heat (Btu/lbm-°F)	0.15
N_{VR} , number of vacuum breakers in system	4	Latent heat of fusion (Btu/lbm)	118.0
Reactor Vessel and Gore Description Card		Fuel Properties Card	
Decay Power and Time Definition Card		Initial thickness of molten fuel layer (ft)	4.4
Initial core power level (BTU/hr)	1.124E10	Thermal conductivity (Btu/hr-ft-°F)	1.5
Time from shutdown to start of calculations (sec)	0.0 (12)	Density (lbm/ft ³)	550.
Reactor Core Physical Description Card		Specific heat (Btu/lbm-°F)	.123
Active fuel height (ft)	12.5	Emissivity of molten layer	.8
Fuel Rod Diameter (ft)	4.025E-2	Reference Temperatures (°F) Card	
Pellet diameter (ft)	3.417E-2	Melting point for steel	2605.
Core diameter (ft)	16.70	Failure point for insulation	3040.
Thickness of Zircalloy cladding (ft)	2.667E-3	Concrete Properties Card	
Fuel rod volumetric heat capacity (Btu/ft ³ /°F)	54.2	Concrete thermal conductivity (J/sec/cm ² /°K)	.0182
Flow channel hydraulic diameter (ft)	3.70E-1	Concrete specific heat (J/gm/°K)	.6525
Flow area in core (ft ²)	84.0	Concrete density (gm/cm ³)	2.405
Flow area in vessel (ft ²)	154.0	Mass fraction of CaCO ₃	.5071
Reactor Hardware Description Card		Mass fraction of Ca(OH) ₂	.0867
Radiation interchange factor between top of core and heat sink above	.465	Mass fraction of SiO ₂	.3777
Environment Parameters Card		Mass fraction of free H ₂ O in concrete	.0285
Initial zirconium oxide thickness (ft)	3.28E-6	Rebar to concrete mass ratio	.164
Melting temperature of fuel plus temperature equivalent of heat of fusion (°F)	6.343E3	Interface Heat Transfer Coefficient Card	
Melting temperature of fuel (°F)	5080.	Metal/concrete heat transfer coefficient (J/sec/cm ² /°K)	.009
Reactor Pressure Vessel Description Card		Oxide/concrete heat transfer coefficient (J/sec/cm ² /°K)	.050
		Materials Initial Temperature Card	
		Concrete temperature (°K)	322.
		Oxide layer temperature (°K)	3130.
		Metal layer temperature (°K)	2560.

C-2E

(13)
(13)

NOTES TO TABLE C.1

- (1) TQUV and ATWS only; volume of liquid pool is $1.308E5 \text{ ft}^3$ for TW.
- (2) TQUV and ATWS only; RPV temperature is 431.75°F , wetwell temperature is 365°F , and drywell temperature is 363°F for TW.
- (3) TQUV and ATWS only; RPV pressure is 351 psia, wetwell pressure is 165 psia, and drywell pressure is 160 psia for TW.
- (4) TQUV and ATWS only; drywell RH is 100% for TW.
- (5) Models flow through the SRVs.
- (6) TQUV and TW only (models 4 SRVs); throat area is 2.0062 ft^2 for ATWS (models 14 SRVs).
- (7) TW only; models break in containment (drywell). For ATWS with containment failure prior to core melt, break size of 3.14 ft^2 .
- (8) TW only; models normal seepage from Rx Building to outside.
- (9) ATWS with containment failure prior to core melt; models break in Rx building (blowout panels).
- (10) TQUV and ATWS only; vent submergence is 12.25 ft for TW.
- (11) ATWS and TQUV only; loss coefficient is 1.5 for TW.
- (12) ATWS and TQUV only; time from shutdown to start of calculations is $1.08E5 \text{ sec}$ (30 hours) for TW.
- (13) TQUV, TW, and ATWS sequences for molten core over entire diaphragm floor; for sequences with molten core in pedestal, initial oxide layer temperature is 3630°K and initial metal layer temperature is 3060°K .

The addition from sources, $S(t)$, may be modeled as any function of time, t , such as a constant rate, an impulse at time zero, or some other function. The removal from sinks, $\alpha(t) * C(t)$ is expressed in terms of the magnitude of the concentration at time, t , since the removal process acts within the compartment and treats the compartment volume as a whole. The source is independent of compartment concentration; however, it may be dependent on another compartment's concentration if leakage is occurring from one compartment to another.

An exponential solution to Equation D-1 is implied since the change in concentration with respect to time is proportional to the concentration at that time:

$$\frac{\partial C(t)}{\partial t} = S(t) - \alpha(t) * C(t); \quad S(t) = 0$$

$$\frac{\partial C(t)}{C(t)} = -\alpha(t)dt$$

$$C(t) = C_0 e^{-\int \alpha(t)dt} \quad (D-2)$$

Since Equation D-1 is a linear differential equation, the solution for any particular source can be found as a sum of the homogeneous solution ($S(t) = 0$) and the particular solution. This implies that concentrations as a function of time, developed from different sources or release mechanisms, can be combined by summing individual solutions to form an overall solution at any time of interest.

D.2.2 Radionuclide Release Mechanisms

In a core melt accident, there are four basic mechanisms for release of radioactivity:

- Gap Release -- occurs when the cladding ruptures and fission products are released to the reactor coolant system.
- Melt Release -- occurs when the fuel reaches its melting point, resulting in volatilization of fission products from the melting core.

- Oxidation Release -- occurs when part of the molten core drops into the suppression pool, causing a steam explosion, dispersing the hot core particles to the containment atmosphere.
- Vaporization Release -- occurs when the molten core drops onto the diaphragm floor and interacts with concrete, generating gases at the molten core/concrete interface.

These mechanisms are similar to those postulated in WASH-1400.

D.2.2.1 Gap Release

The gap release is chronologically the first release to occur in an accident sequence and the amount of radioactivity released is small compared to the other release mechanisms. Consequently, the overall release fractions of a gap release are negligible and are ignored for simplicity. The gap release is composed principally of noble gases.

D.2.2.2 Melt Release (Core and RPV Melt)

The core melt release starts after the core has uncovered and the fuel has heated to its melting point. As the fuel melts, core fission products are released. In the Limerick PRA it was assumed that the core melt release occurs linearly as 0% to 80% of the core melts. This is a simplification over WASH-1400, which assumed that fission product release occurred at the rate of core melting. The results of the BOIL code indicate that this simplification is reasonable. At 80% core melt, the core grid plate is assumed to fail. At this time, the core is assumed to fall into the lower head and further radioactive release is terminated. When the core falls to the bottom of the RPV, the surface area for release is noticeably smaller. Additionally, either a crust may form due to the water on top of the molten corium or the metal in the melt could migrate to the top due to density differences. Either of these would provide a barrier inhibiting the flow of fission products. The release fractions into other compartments during the core melt phase are the same as used in the Reactor Safety Study and are taken from Table VII 1-3 of WASH-1400.

D.2.5 Specific Sequence Calculations

Calculations for the radionuclide release fractions were done for three of the sequences modeled in the INCOR analysis -- TQUV, TW, and ATWS with HPCI failure (see Appendix C for details). These three sequences can be divided into two categories for the methodology used to calculate the release fractions: (1) Core Melt -- RPV Melthrough -- Core/Concrete Interaction Prior to Containment Failure (TQUV and ATWS with HPCI Failure); and (2) Containment Failure Prior to Core Melt -- RPV Melthrough -- Core/Concrete Interaction (TW).

D.2.5.1 Core Melt -- RPV Melthrough -- Core/Concrete Interaction Prior to Containment Failure

The basic methodology used in this type of sequence is to accumulate the radioactivity in the containment from the various releases (see Section D.2.2) and then to release the accumulated radioactivity from the containment either directly to the atmosphere or through the reactor building to the atmosphere. For each type of release, an equation is set up to define the concentration of radioactivity in the compartment being considered using general Equation D-1. This equation is then solved for two amounts: (1) the fraction of radioactivity available for release (f'_{leaked}) and (2) the fraction of available radioactivity remaining (f'_{left}) or the fraction of radioactivity available for release at a later time. The actual fractions of radioactivity released (f_{leaked}) and left (f_{left}) are equal to the product of the available radioactivity remaining/left and the percentage of radioactivity that is released for that type of release.

The first release considered is the melt release which is divided into four parts: Core melt, RPV Melt, RPV failure, and Blowdown. A constant release is assumed to occur during core melt which implies a constant source of radioactivity over time; therefore, general Equation D-1 takes on the form:

$$\frac{\partial C(t)}{\partial t} = S_0 - \alpha C(t) \quad (D-3)$$

Equation D-3 is then solved for the two sources: the radioactivity available for release to the wetwell pool through the SRVs (f'_{leaked}), and the available radioactivity remaining in the RPV for a later release (f'_{left}). The first solution is:

$$f'_{\text{leaked}} = \frac{\lambda_2}{\alpha^2 t} (\alpha t + e^{-\alpha t} - 1) \quad (\text{D-4})$$

where $\alpha = \lambda_2 + \lambda_{\text{NR}}$ and λ_2 is the leakage removal rate and λ_{NR} is the natural removal rate. It is assumed that natural deposition does not occur in the RPV; therefore, the total removal rate is only equal to the leakage removal rate ($\alpha = \lambda_2$). Equation D-4 then reduces to

$$f'_{\text{leaked}} = \frac{1}{\alpha t} (\alpha t + e^{-\alpha t} - 1) \quad (\text{D-5})$$

The second solution to Equation D-3 is:

$$\begin{aligned} f'_{\text{left}} &= (1 - f'_{\text{leaked}}) \\ &= 1 - \frac{1}{\alpha t} (\alpha t + e^{-\alpha t} - 1) \\ &= \frac{1}{\alpha t} (1 - e^{-\alpha t}) \end{aligned} \quad (\text{D-6})$$

At the time of core grid plate failure, it is assumed that no radionuclides are added to the RPV from the molten fuel (see Section D.2.2.2). The only radioactivity available for release through the SRVs during RPV melt is that left during core melt. However, the actual radioactivity available for release during RPV melt is the product of that which is available from core melt (CM) and the solution to the equation for the concentration of radioactivity. General Equation D-3 for a no-source release takes on the form:

$$\frac{\partial C(t)}{\partial t} = -\alpha C(t)$$

and with the same solutions:

$$f'_{\text{leaked}} = \frac{1}{\alpha t} (\alpha t + e^{-\alpha t} - 1)$$

$$f'_{\text{left}} = \frac{1}{\alpha t} (1 - e^{-\alpha t})$$

Natural deposition is still assumed not to occur in the RPV.

The radioactivity is constantly being released to the suppression pool, being scrubbed, and then released to the wetwell atmosphere. This release from the suppression pool acts as a constant source for the wetwell atmosphere, and, therefore, the same equations still apply. However, since natural deposition is assumed to occur, the equations for the radioactivity available for release from containment to the reactor building, and the available radioactivity left for later release are:

$$f'_{\text{leaked}} = \frac{\lambda_f}{\alpha - \lambda_f} (\alpha t + e^{-\alpha t} - 1) \quad (D-10)$$

$$f'_{\text{left}} = \frac{1}{\alpha t} (1 - e^{-\alpha t}) \quad (D-11)$$

Again, the fission products constantly being released from the wetwell atmosphere to the reactor building act as a constant source for the reactor building. Therefore, Equation D-10 is also applicable for the radioactivity available for release to the atmosphere. However, Equation D-11 is assumed zero since the radioactivity that is not naturally deposited is assumed to be totally released to the atmosphere.

The next step is a release of the available radioactivity left in the wetwell atmosphere to the reactor building and then to the atmosphere. Since this is remaining radioactivity, it is assumed there is no source.

It is also assumed that the remaining radioactivity which is not naturally deposited is released during a time period assumed to be infinity. The solution for a no-input source with an infinite time period then reduces to:

$$f'_{\text{leaked}} = \frac{\lambda_2}{\alpha} \quad (D-12)$$

In the next release, RPV melt, the fraction of radioactivity available for release from the RPV to the wetwell pool is subjected to the same assumptions and conditions as mentioned in Section D.2.5.1. Therefore, Equation D-7 is applicable and the fraction of radioactivity available for release is equal to:

$$f'_{\text{leaked}} = f'_{\text{left}_{\text{CM}}} (1 - e^{-\alpha t})$$

At RPV failure, the radioactivity remaining in the RPV available for release to the drywell is equal to:

$$f'_{\text{leaked}} = 1 - f'_{\text{leaked}_{\text{CM}}} - f'_{\text{leaked}_{\text{PVM}}}$$

Also at RPV failure, part of the radioactivity is released from the drywell to the wetwell pool due to a pressure increase. This release happens in so short a space of time that it is assumed to be an instantaneous release, and therefore, natural deposition is not assumed. Since it is also a no-source release, the radioactivity available for release to the suppression pool (where it is scrubbed) is equal to:

$$f'_{\text{leaked}} = 1 - e^{-\alpha t}$$

While the radioactivity is being released to both the wetwell atmosphere and the drywell, it is also being released from these compartments to the reactor building and then to the atmosphere. These releases are

- Escaped fractions released (for each release: elemental and organic iodines, particulates) at time, t
- Escape fractions of the core for any desired isotope
- Dose reduction factors for each release (elemental iodine and particulates) at time, t
- Overall dose reduction factor (elemental iodine and particulates) at time, t
- Total fraction of core iodine escaped and core particulates escaped up to time, t.

The input data for CORRAL includes two main types: constants and variables. The constant inputs are:

- Core fractions for each release -- (CFR(I,J))
- Number of compartments -- N
- Volumes -- (V(I)); wall areas -- AW(I); floor areas -- AF(I); and heights -- HT(I) of each compartment
- Spray parameters (see Footnote 1, Table D.1)
- Times of events (see Footnote 8, Table D.1)
- Compartment filter decontamination rates -- FDP(I) (see Footnote 6, Table D.1)
- Fractions of compartments released due to a puff release (see Footnote 2, Table D.1)
- Option to select gas flow through the drywell on annulus from a selected compartment - MANN (see Footnote 9, Table D.1).

The variable inputs (those that change with time) are (see Footnote 3, Table D.1):

- Thermodynamic conditions of each compartment: pressure -- PI(J,I); temperature -- TMY(J,I); water vapor content -- VAPI(J,I); and temperature difference between bulk gas and walls -- DELTTI(J,I)

- Flow rates between compartments -- GI(J,K,I)
- Decontamination factors between compartments -- EP(J,K,I) (see Footnote 4, Table D.1)
- Particle sizes -- DPE, DPL, TD
- Leak rates to atmosphere (leak decontamination factors) -- ELKP(J,I) (see Footnote 5, Table D.1).

Release fractions were calculated by CORRAL for each sequence analyzed by INCO.

D.4 SAI-REACT MARK II AND CORRAL RESULTS

Tables D.2 and D.3 give several examples of REACT calculations for various release producing events in which the containment fails due to overpressure. Table D.4 gives examples of data produced with the CORRAL code. The REACT calculations served two purposes:

- 1) to verify the reasonableness of CORRAL-produced release fractions
- 2) to estimate the effect of decreasing containment overpressure failure size as well as the influence of the secondary containment.

For the first purpose, it can be seen that the REACT release fractions, as expected, do not exactly match the CORRAL data. Absolute and relative magnitudes of release fractions, especially elemental iodine, are generally comparable except for tellurium. The REACT value for tellurium for the TQUV γ' sequence is markedly high for a subcooled pool, and is believed to be erroneous. The CORRAL value (.016) was used in the analysis.

For the second purpose, it is evident that the $\xi\epsilon$ release size produces comparable releases to the γ' event, but that other releases do not. Therefore, a conservative ex-plant evaluation approach was used to include the probability of a $\xi\epsilon$ occurrence with that of γ' mode and perform consequence evaluations using γ' release fractions, as defined with CORRAL.

Table D.2
 REACT RADIONUCLIDE RELEASE FRACTIONS FOR TQUV
 (with df of 10 for pool during oxidation release and resuspension of
 fission products from pool at containment failure)

FAILURE MODE	RADIONUCLIDE GROUPS								
	1 Noble Gases Xe, Kr	2A Elemental Iodine I, Br	2B Organic Iodide	3 Cs, Rb	4 Te (a)	5 Ba, Sr	6 Noble Metals Ru (b)	7 Rare Earths La (c)	8 Zr, Nb
γ	1	.067	.0003	.130	.350*	.0064	.032	.0042	.0042
γ	1	.064	.0003	.075	.115	.0064	.019	.0013	.0013
ζ	1	.002	.00007	.0039	.011	.0002	.001	.00012	.00012
δ	1	.0007	.00006	.0013	.0035	.00006	.0003	.00004	.00004
ζε	1	.017	.0003	.107	.290	.0053	.026	.0034	.0034
δε	1	.0036	.00032	.059	.159	.0029	.0145	.0019	.0019

(a) Includes Se, Sb

(b) Includes Rh, Pd, Mo, Tc

(c) Includes Y, Ce, Pr, Nd, Pm, Sm, Eu, Np, Pu

* Possible error in this computation.

Table D.3

REACT RADIONUCLIDE RELEASE FRACTIONS FOR TM

FAILURE MODE	RADIONUCLIDE GROUPS								
	1 Noble Gases Xe, Kr	2A Elemental Iodine I, Br	2B Organic Iodide	3 Cs, Rb	4 Tc (a)	5 Ba, Sr	6 Noble Metals Ru (b)	7 Rare Earths La (c)	8 Zr, Nb
Y	1	.018	.0023	.117	.414	.0002	.034	.0056	.0056
Y	NC	NC	NC	NC	NC	NC	NC	NC	NC
C	1	.0013	.0005	.0035	.014	.00016	.001	.00017	.00017
B	1	.00018	.00046	.0012	.0048	.000005	.00034	.00096	.00006
CC	NC	NC	NC	NC	NC	NC	NC	NC	NC
BC	NC	NC	NC	NC	NC	NC	NC	NC	NC

(a) Includes Se, Sb
 (b) Includes Rh, Pd, Mo, Tc
 (c) Includes Y, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Pu
 NC - NOT CALCULATED

The results of the CRAC consequences model are displayed as a set of complementary cumulative distribution functions (CCDF) for specific consequences. These distributions are determined from the calculated magnitude of each consequence for each combination of postulated accident release, weather, and population as well as the probability of each such combination.

The consequence Code (CRAC) used in WASH-1400 is summarized in Table E.1. The basis CRAC methodology, the dosimetric model, and the health effect models, were adopted for this analysis. Some modifications were made to adapt CRAC to the Limerick site-specific requirements. These site-specific effects include:

- The probabilities and release fraction input data (see 3.5 and 3.6)
- The Limerick site meteorology
- The population for the Limerick area.

The remaining subsections of this Appendix discuss the various models used in the Limerick CRAC calculation and how they were implemented.

E.2 BEHAVIOR OF RADIONUCLIDES IN THE ATMOSPHERE

In the event of radioactive release, radionuclides are released into the air and dispersed. Figure E.4 shows a side view of this process. The population in the area under the plume receive radiation in three ways:

1. From external radiation received directly from the radioisotopes in the cloud (cloudshine)
2. From radiation received following inhalation
3. From radiation received from material deposited on the ground (groundshine).

Table E.1
Consequence Code Model Details
as Applied in WASH-1400

ITEM/MODEL	
I Release Category	9 PWR categories* 5 BWR categories *PWR 1 was subdivided further to represent 2 distinct heat releases; PWR 1A & PWR 1B
No. of Isotopes	54 Isotopes
Fission Product Inventory	Initial source strength of the potential radioactive source was calculated by using ORIGEN code (Bell, 1973)
II Weather Data	6 Stability Classifications, A-F 8 Wind Velocity Groups per classification each, with rain/no rain condition, and associated probabilities. 6 distinct composite sites Each site had at least one year of complete recorded weather data including hourly data on rain occurrence. The consequence code used completely stratified samples. In order to ensure complete coverage, every four days plus one hour starting time was selected and the weather condition for the next 10 to 30 hours was updated every hour. Thus, 90 weather samples are utilized.

Item/Model	
III Atmospheric Dispersion Model Note: The radioactive cloud concentration is corrected for plume rise, allowance for release durations of greater than 0.5 hour, and a spatial modification for cloud depletion by dry and wet deposition and finally for radioactive decay.	In general $\frac{x}{y} = \left[\frac{u}{\sigma_y \sigma_z} \right]^{-1} \left[\frac{y^2}{2\sigma_y^2} - \frac{z^2}{2\sigma_z^2} \right]$ where the lateral concentration is assumed constant, the lateral width is taken as $\sigma_y = \sqrt{2R} \sigma_y e^{\frac{y^2}{2\sigma_y^2}} \text{ for } -1.5\sigma_y < y < 1.5\sigma_y$ For building effects when $h = 0$ $\frac{xu}{Q} = \frac{1}{\sigma_y \sigma_z + CA}$ The standard deviation σ_y & σ_z are evaluated at each y radial position as $\sigma = A x^B + C$ where A, B, C are parameters associated with each pasquill stability category, (Martin-Tikvart coefficients). The vertical diffusion, σ_z is not allowed to exceed a maximum, in this case, 0.8 L, where L is the mixing height (Holizworth 1972)
Plume Rise	The plume centerline height, h is determined by using a relationship developed by Briggs (1969) For unstable conditions $h = 1.6F^{1/3} - 1 x^{2/3}$ out to $x = 1.25Q_h^{2/5}$ For stable conditions: $\Delta h = 2.9(F/us)^{1/3}$ out to $x = 2.4u(S)^{-1/2}$ In these equations h is not allowed to exceed the mixing layer depth L. Δh = plume height above an initial emission height F = buoyancy flux = $3.7 \times 10^{-5} Q_h$ Q_h = energy release rate (calories/sec) S = stability parameter (sec ⁻²)

m
1
01

wind shear with altitude were not considered in WASH-1400. (It has since been shown that (E-5) wind shear variations do not significantly affect the plume dispersion calculations.) The Limerick analysis, uses seasonally varying wind roses, stability, and wind speed.

The wind measurements used in the Limerick consequence calculations are determined at the start of the radioactive release. No subsequent variations are accounted for.

E.2.3.2 Precipitation

Another consideration is the effect of precipitation on the dispersion of the plume (E-4). As rain falls through the plume, radioactive material falls with the rain to the ground. Thus, ground concentration of radiocativity is raised. The effects of a rainstorm on dispersion are controlled by the following variables:

- Washout coefficient - the amount of radioactivity interacting the rain.
- Runoff - the amount of water not absorbed into the ground
- Rain intensity - the variation with time
- Intersection - the distance from the reactor at which the plume intersect with the rainstorm.

Occurrence of rain will tend to increase the number of early fatalities, and decrease latent fatalities since the radioactivity is dispersed in a smaller area in more concentrated amounts. The WASH-1400 precipitation model, which does not consider runoff or time-varying rainfall intensity, was used in the Limerick analysis.

E.3 PUBLIC RESPONSE MODEL

Since the consequences of a nuclear power plant accident are dependent on public response, a response model must be included in consequence calculations. The public response model used in this study is the same as that used in WASH-1400, which considered two main facets of the public response: evacuation and shielding.

E.3.1 Evacuation

The quickness and effectiveness of an evacuation are mainly controlled by the following:

- Public participation in the evacuation
- Warning time for evacuation
- Speed of evacuation
- Population density of evacuated area
- Emergency preparedness.

In the WASH-1400 evacuation model, the evacuated area is in the shape of a keyhole centered on the prevailing wind heading at the time of release (Figure E.6)..

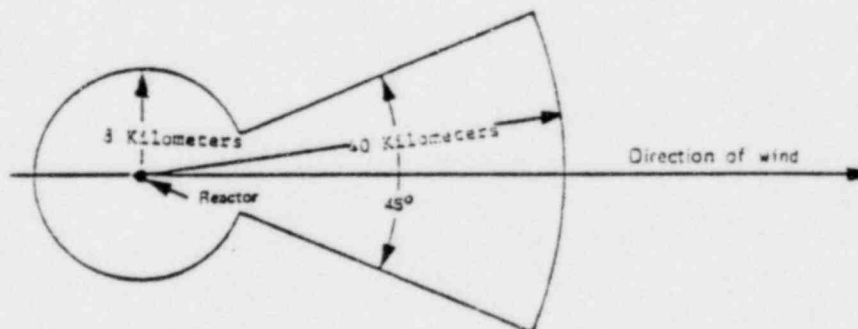


Figure E.6. Evacuation Area -- WASH-1400

In the Limerick analysis, the site-specific population distribution (Table E.7) and warning time are used to develop inputs to the CRAC code. The warning time depends upon the accident sequence. The population response is fixed, in that all persons in the evacuation portion of the effected zone head outward radially and all persons in the keyhole portion head tangentially at a constant rate of speed. In WASH-1400 the intention was to divide population into 3 groups with 3 effective evacuation speeds, in order to adequately model different levels of population participation in an evacuation. Thus 30% of the population would move with an effective speed of 0.2 mph, 40% would move with an effective speed of 1.2 mph, and 30% would move with an effective speed of 7 mph. In fact, analysis of several sites showed that only the medium speed (1.2 mph) need be used if the resulting casualties were scaled by a factor of 1.5. The Limerick analysis uses the same procedure actually used in WASH-1400, i.e., a medium evacuation speed of 1.2 mph and a multiplying factor of 1.5.

E.3.2 Shielding

E.3.2.1 Cloudshine and Ground shine

People caught within or under a radioactive cloud will receive an external dose to the whole body due to gamma radiation. Buildings offer some attenuation of doses since the walls of the building will absorb and scatter gamma radiation. Recent EPRI studies (E-6) have shown that the benefit of shielding in some areas of the country may outweigh the benefits of evacuation for much of the population.

In the Limerick ex-plant consequence model, dose assessment includes consideration of cloudshine and groundshine shielding. The form of the shielding model used in Limerick is the same as that used for WASH-1400. People in structures at the time of exposure receive a lower whole body dose than those that are unprotected. A shielding factor (SF) is defined, which is the ratio of the interior dose to the dose that would have been received with no protection.

Since structures have regionally related characteristics, an assessment was made for the area around Limerick. The methodology for determining the overall shielding factor involves weighted averaging of shielding factors. These shielding factors were developed for various human situations. This model assumed that people who were outdoors or commuting would not seek shelter. Additionally, 5% of the people were assumed to take no action, even if advised to. This model also uses regional data on the percentage of brick houses. The values for groundshine and cloudshine dose shielding factors used in the Limerick analysis are found in Table E.2a and E.2b respectively. When compared with the shielding factors of 0.33 for groundshine and 0.75 for cloudshine, as used in WASH-1400, the Limerick shielding factors are enhanced somewhat, principally because of the effect of more adequate shielding.

E.3.2.2 Inhalation

The effective inhalation rate for the population affects the latent consequences of a nuclear accident. When the radioactive plume passes over a populated area, people may inhale radionuclides from the passing cloud. The breathing rate input to the CRAC code is an effective breathing rate; it is a measure of how much radiation the public receives through inhalation. The breathing rate used in WASH-1400 and the Limerick PRA was $2 \times 10^{-4} \text{ m}^3/\text{s}$.

Credit was given only for the moderate reductions in inhalation dose as a result of sheltering with some subsequent effective ventilation action. The values for these sheltering factors were taken from Reference E-14. For ventilation rates consistent with closed windows, shut-down outside ventilation systems, and the reduced leakage consistent with houses equipped for energy conservation (typical of the Northeast), the indoor dose ratio, or inhalation shielding factor, is 0.53.

Table E.3

Shielding Factors For Inhalation Doses Given Sheltering

People at Home

	<u>Basement</u>	<u>No Basement</u>
Fraction of total	0.63	0.06
SF	0.48	0.53

People at Work

	<u>Large Building, Brick or Wood</u>	<u>Basement</u>
Fraction of total	0.078	0.118
SF	0.53	0.48

People Commuting or Outdoors

	<u>Communting</u>	<u>Outdoors</u>
Fraction of total	0.05	0.062
SF	1.0	1.0

TOTAL SF - 0.544

WASH-1400 + 1.0
(No Sheltering)

$$SF = 0.95 * (0.544) + 0.05 (1.0) = \underline{0.57}$$

*Portion of total population participating in emergency response

Table E.3 gives the calculations for the final shielding factor. It should also be noted, from Reference E-14, that approximately a 10% reduction is assumed for persons in their basements. Consequently, given that 95% of those inside follow directions, a total shielding factor of 0.57 is calculated.

E.4 HEALTH EFFECTS MODEL

One of the measures of consequences of a nuclear power plant accident is health effects on the public. Figure E.7 gives a summary of the isotopes which affect the consequences to the public. Health effects can be divided into two categories; short term (early) effects, which are apparent within one year from exposure, and long term (latent) effects, which can show up during the remainder of a lifetime. Cumulative Complementary Distribution Functions (CCDF) are ultimately obtained in the Limerick analysis for early and latent fatalities.

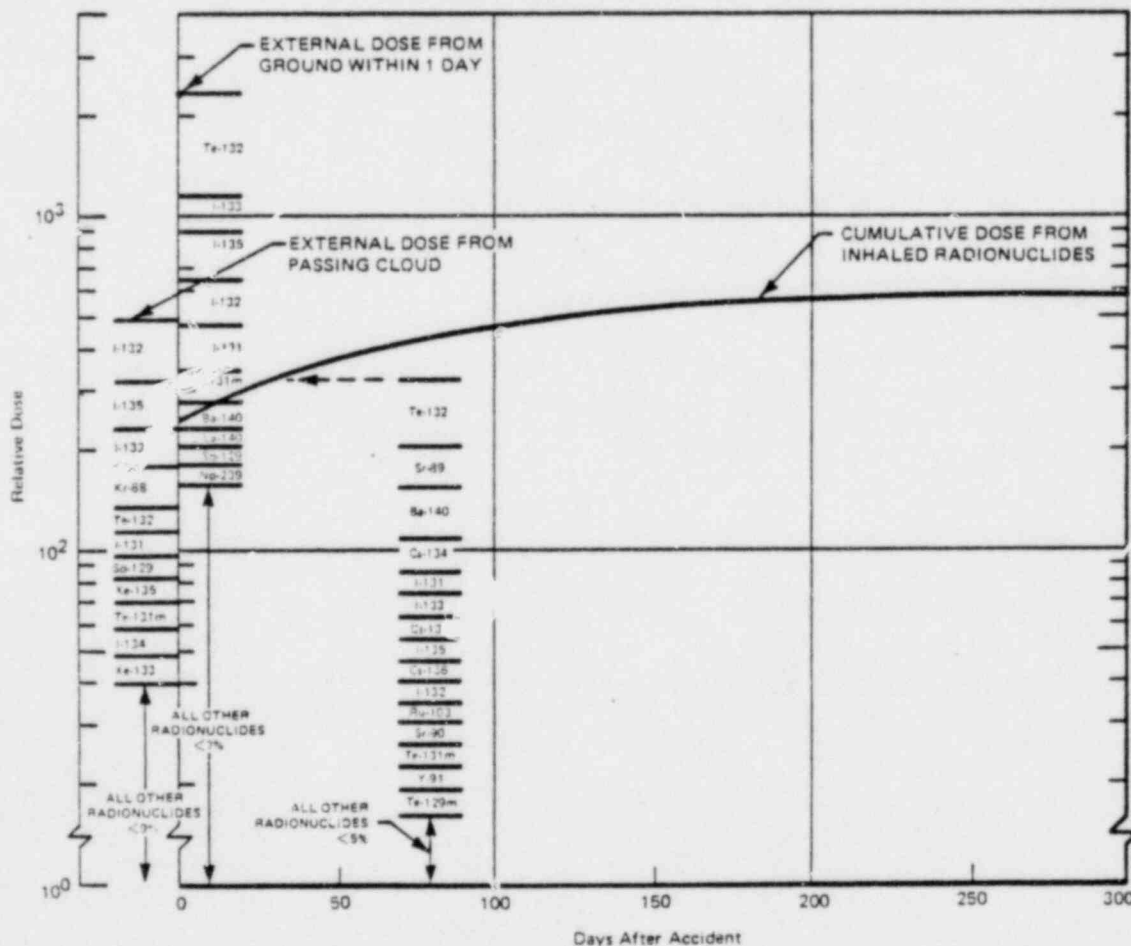


Figure E.7. Relative Doses to Bone Marrow at 0.5 Miles from Reactor (Reference E-1, App. VI, Fig. VI 13-1)

Table E.4
COMPARISON OF THE WASH-1400 COMPOSITE SITE DATA WITH THAT FOR LIMERICK

RANK OF SECTOR BY POPULATION	WASH-1400 COMPOSITE SITE				LGS SITE-SPECIFIC		
	ORIGIN OF SECTOR	CONDITIONAL PROBABILITY OF SECTOR BEING EXPOSED	WIND ROSE	TOTAL CONDITIONAL PROBABILITY OF SECTOR BEING EXPOSED	Ranking of Sectors	WINDROSE* 30 FT SUMMER 1974	170 FT WINTER 1974
1	1	.00446	1.0	.00446	(G) **	.06	.2
2	2	.00446	1.0	.00446	(F)	.12	.15
3	3, 4	.00893	1.0	.00893	(H)	.05	.05
4	5, 6	.00893	1.0	.00893	(P)	.03	.04
5	AVG. of NEXT 6	.0268	1.0	.0268	(B)	.04	.04
6	AVG. of NEXT 6	.0268	1.0	.0268	(E)	.10	.06
7	AVG. of NEXT 12	.0536	1.0	.0536	(J)	.04	.06
8	AVG. of NEXT 22	.0982	1.0	.0982	(D)	.07	.04
9	AVG. of NEXT 22	.0982	1.0	.0982	(A)	.09	.04
10	AVG. of NEXT 23	.1030	1.0	.1030	(K)	.03	.05
11	AVG. of NEXT 22	.0982	1.0	.0982	(C)	.06	.05
12	AVG. of NEXT 22	.0982	1.0	.0982	(Q)	.05	.04
13	AVG. of NEXT 20	.0893	1.0	.0893	(R)	.05	.02
14	AVG. of NEXT 20	.0893	1.0	.0893	(M)	.05	.05
15	AVG. of NEXT 21	.0948	1.0	.0948	(N)	.07	.08
16	AVG. of NEXT 22	.0982	1.0	.0982	(L)	.06	.03

*Also conditional probability of sector being exposed

**Population sector designator (Table E.7)

with the wind blowing in the direction of highest population. Table E.5 reflects the approximate increase in the conditional probability of the wind blowing in the direction of highest population.

Table E.5
TOP TWO(2) SECTORS WITH MAXIMUM POPULATION

RANK BY POPULATION	CONDITIONAL PROBABILITY OF SECTOR BEING EXPOSED		
	COMPOSITE SITE FROM WASH-1400	LIMERICK SITE	FACTOR LARGER FOR LIMERICK
1	.00446	.2	-45
2	.00446	.15	-34

E.6 CRAC INPUT

The inputs to the CRAC code are summarized in Table E-6.

Wind roses for the LGS site are shown on Figures E.8 and E.9.

Table E-7 shows the sector designations and the population by sector.

Table E-8 compares the radiological core inventory used in the Limerick analysis to that used in WASH-1400. The amounts are similar for the majority of the isotopes between Limerick and WASH-1400. The major difference is seen in the particulates. The Cesium (Cs), Antimony (Sb) and Tellurium (Te) isotopes are generally greater for WASH-1400 than Limerick. However, the Rubidium (Rb), Ruthenium (Ru), and Americium (Am), isotopes are generally greater for Limerick than WASH-1400 isotopes.

TABLE E.6 INPUTS TO CRAC CODE

Data Name	Value
Maximum Distance of Evacuation (mi)	25
Evacuation Velocity (mph)	1.2
Time Lag before Evacuation (days)	0
Travel Distance while Evacuating (m)	8000
Angle of Evaluated Downwind Sectors	45 ^c
Criteria of Duration of Release for Evacuation	3
Cloud Shielding with Sheltering	0.54
Cloud Shielding without Sheltering	0.71
Ground Shielding with Sheltering	0.15
Ground Shielding without Sheltering	0.29
Breathing Rate m ³ /s	2.0 x 10 ⁻⁴
Release Height-high (m)	25
Release Height-low (m)	0
Isotopes	Limerick core inventory*
Early Health Effects	Same as WASH-1400*
Latent Health Effects	Same as WASH-1400*
Spatial Mesh Description	Limerick site-specific*
Population Data	1970 census & PECO data (see Table E.7)
Meteorological Data	Limerick wind roses shown on Figures E.8 and E.9

*Input values on computer tape

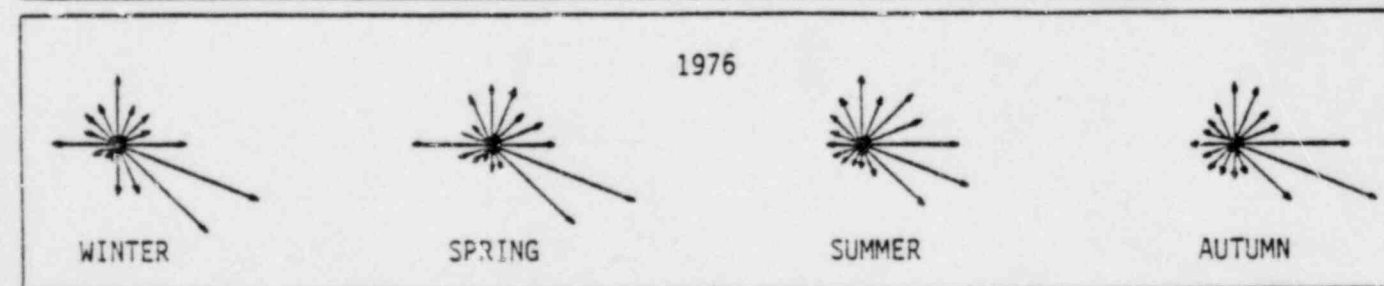
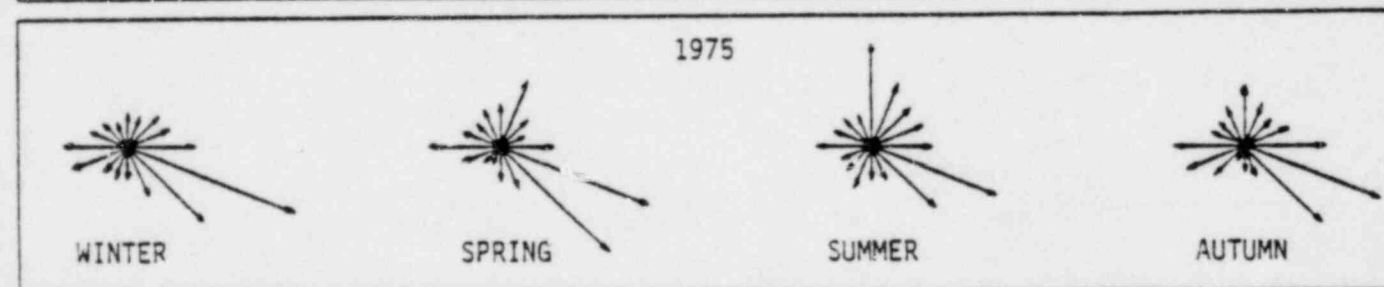
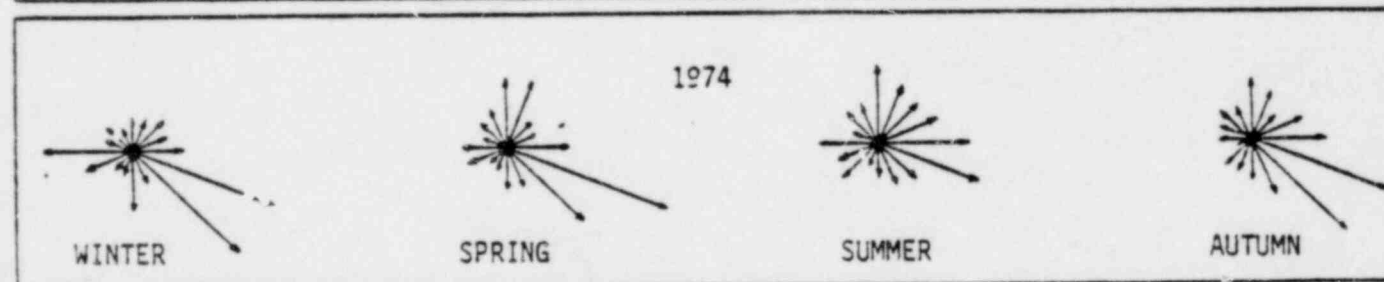
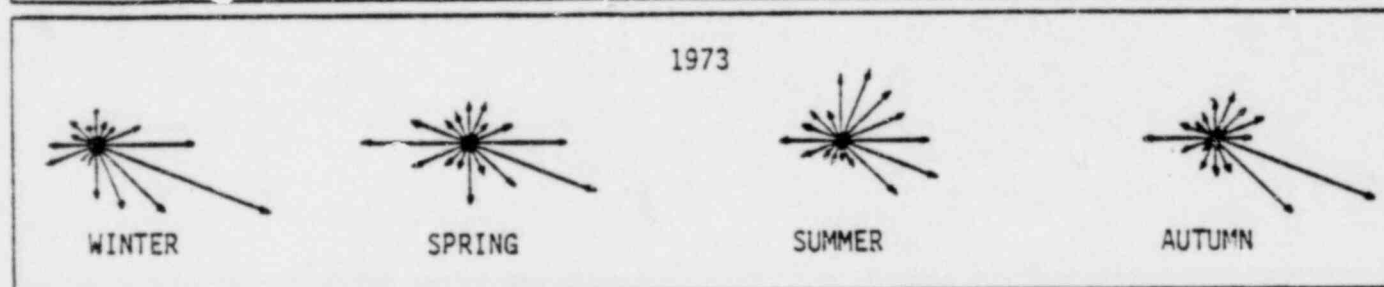
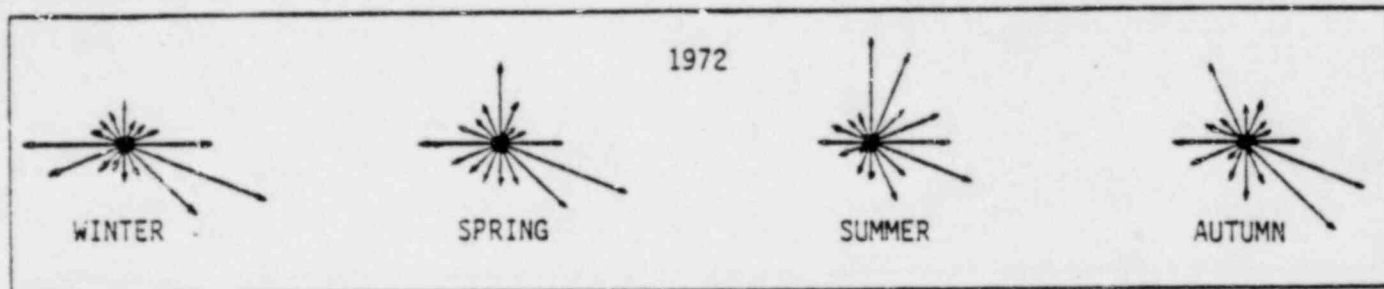
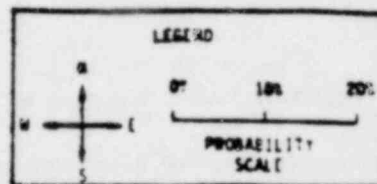


Figure E.8 Wind rose* for the Limerick Generating Station for five years of Meteorological Data by season at a Tower Height of 30 feet

* Probability of wind bearing during each season

a random variable that follows a log-normal distribution. Despite lack of conclusive reasons, WASH-1400 has led many to believe the log-normal is the best distribution to use (for example see F-5). WASH-1400 is correct in using "assessment ranges", since these accurately specify the state of knowledge or uncertainty.

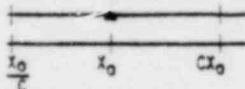
There is no theoretical foundation for or against either the gamma or log-normal distributions, and there is little data for either choice, particularly if the shape is dependent on the grouping. The log-normal distribution has the advantage that it is easy to obtain the distribution parameters from the 5% and 95% values, whereas this is difficult for the gamma. On the other hand, the gamma distribution forms a conjugate prior when using the exponential model (i.e., a gamma prior leads to a gamma posterior), and this simplifies Bayesian calculations, whereas, the log-normal is difficult to work with analytically. The choice between log-normal and gamma therefore becomes: "Do we just desire a distribution?" (the log-normal has easily obtained parameters); or "Will we perform Bayesian updating?" (the gamma is easier to work with analytically).

Note: In addition, a possible, but not necessarily correct, justification of log-normal can be the following:

If we took a "sufficiently large group of "relatively knowledgeable" people and asked them to estimate a number (for example, the length of a room in feet), the results of each estimate might be approximately normally distributed around some (maybe even correct) mean. Now, if we asked a similar group to estimate a very small or very large number in power of 10 (for example, the length of a pencil in miles), the exponents of 10 might be normally distributed, and therefore the estimates would be log-normal (F-6). Therefore, some might be inclined to consider the log-normal as a reasonable distribution for uncertainty in our collective knowledge. Experiments might be conducted to see if groups of people really estimate in this fashion.

Table F.2

WASH-1400 REASONS FOR USING LOG-NORMAL

Reason	Relation to Gamma
1. "Where sufficient data did exist," the data fit a log-normal.	Rarely, if ever, was there sufficient data.
2a. Using error factors, the range  can transform to $\ln x_0 \pm \ln c$ -- like using normal error spreads. 2b. Data is expressed as 10^{-c} , so if data is log-normal, then "c" is normal.	Somewhat circular logic. There is no reason to expect this.
3. Log-normal has two parameters.	Gamma has two parameters.
4a. Log-normal is positively skewed. 4b. Mean > Median > Most Probable (mode) is propagated "thus providing a protective, positive type bias".	Gamma has the identical properties.
5. "If the probabilities are decomposed into products of probabilities representing requisites for failure, then [if] when the central-limit theorem is applicable, the log-normal is the resulting distribution."	A big <u>IF</u> .
6. "The log-normal can become near normal or near exponential in certain situations.	The Gamma can become near normal, and the Gamma has exponential and chi-squared as special cases.
7. "Its application as a general distribution for...reliability processes is established and has often been validated."	Gamma is often applied to reliability processes also.

*Appendix II, p. 42

APPENDIX G

The question arises as to whether the mean or median should be used to characterize or display the point estimate results of the accident sequence probability calculations. The following points are important in this discussion:

1. The fault tree quantification is performed using mean values for all input parameters in the point value calculation. The mean values propagate through the fault tree to accurately represent the top event for independent input events (see Section G.2). However, in addition, in order to present a consistent set of values with those presented in WASH-1400, an estimate of the median (along with the appropriate error range) is provided for display purposes.
2. The question of which measure of central tendency to display (mean or median) is more philosophical than mathematical. In the case of a normal (Gaussian) distribution, either one would suffice since the mean = median. This is not true for asymmetric distributions (see Figure G.1).
3. This appendix discusses the applicability of using the mean or median value of a probability distribution as the point estimate in accident sequence probability calculations for the Limerick PRA. The following points are important in this discussion:

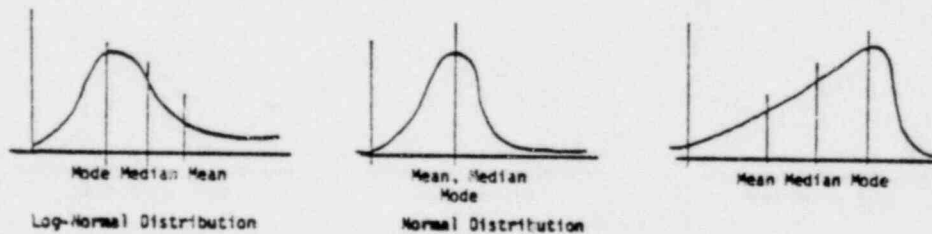


Figure G.1 Schematic Comparison of Three Possible Distributions Which Would Alter the Relationship Between the Mean and Median.

G.1 COMPARISON OF MEAN AND MEDIAN VALUES

Mean

There are a number of advantages associated with using mean values as opposed to median values. The mean value has several properties which make it suitable for use in the calculational phase of a problem. For example, means will propagate through the Boolean algebra calculation required to combine "a group of sequences" to determine the final probability (see Section G.2). Medians cannot, in general, be used in this calculational phase. They can be multiplied (AND gates) if the distribution is known to be log-normal, but they cannot be added (OR gates).

Also, mean values provide more information than do median values about the effect of extreme values which may be present in a skewed distribution (such as hypothesized nuclear power plant risk curves).

The ultimate use of the failure rate, however, may be in a value-impact analysis. In such an analysis, where consequences associated with failures are combined with the probabilities, the distribution may be skewed. In such cases (where a value-impact analysis is involved), it appears to make more sense to use a mean value as the parameter representing central tendency.

Median

The median value has properties which also make it desirable, as noted in the matched quotations below:

the median often is an appropriate measure of central tendency for random variables that are not symmetrically distributed (G-1).

...particularly if it is desired to eliminate the effect of extreme values (G-2).

Despite the equal usefulness of both the mean and median when the distribution is known, criticisms are still made against one or the other. It is argued by Kendall (G-6) that if only the median is used (dropping the context of the log-normal distribution and the 90% and 10% points), nuclear reactors would have the appearance of being safer than they really are. This is true, since the "best estimate" sequence probability estimates for each category are calculated as a median; if the mean is used to represent the sequence probabilities, the point estimate will appear higher (see Figure G.2). However, this discussion ignores a point that is repeated several times in WASH-1400: "One cannot generally use point values and treat them as being exact, since there will always be variabilities and uncertainties" (Ref. G-7). The method of analysis used requires that some form of distribution or spread be stated. Any statement of the result is incomplete if the associated uncertainty is not specified, i.e., as a variance.

It is believed that the use of either a mean or median for display purposes is technically correct and can be justified. An estimate of the medians is provided to display the results for consistency with WASH-1400. (Since the results will be compared with WASH-1400, it is felt that the Limerick results should be available in the same form as those in WASH-1400.) However, the median values are only estimated, based upon calculations using the mean values, and assuming a distribution for the final calculations using the mean values, and assuming a distribution for the final sequence values. It cannot be overemphasized that the real importance of any comparison of sequence probabilities lies in the comparison of the total uncertainty range established in WASH-1400 versus the range established in the Limerick study, and not in a comparison of the central tendency or best estimate values.

G.2 PROPAGATION OF MEAN VALUES THROUGH A BOOLEAN ALGEBRAIC EXPRESSION (i.e., Fault Tree)

The following section describes the mathematical basis for the propagation of mean values through a fault tree, assuming that all basic

input components are independent. Many computer codes, such as WAMBAM, will propagate any set of pointwise input values to generate a point estimate of the top gate. Here it is shown that if these input values are means, then the output for the top event will be a mean value.

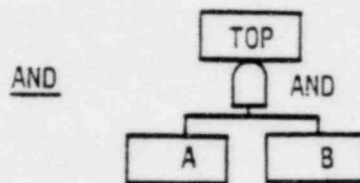
In the following discussion it will be useful to adopt the following notation (G-8):

1. $P(A)$ = the probability of event A occurring. This has a value between zero and one. $P(A)$ will be considered as an "uncertainty variable" (see Appendix F) which has the same properties as a random variable.
2. For convenience let $X = P(A)$ and $Y = P(B)$. X and Y will be treated as though they are random variables with $0 \leq x, y \leq 1$.
3. X and Y have probability density functions $g(x)$ and $h(y)$ respectively with the following properties:
 - a) $g(x) \geq 0, 0 \leq x \leq 1$ a') $h(y) \geq 0, 0 \leq y \leq 1$
 - b) $\int_0^1 g(x)dx = 1$ b') $\int_0^1 h(y)dy = 1$
 - c) $P(a < X < b) = \int_a^b g(x)dx$ c') $P(a < Y < b) = \int_a^b h(y)dy$.
4. X and Y have a joint probability density function $f(x,y)$ such that:
 - a) $f(x,y) \geq 0, 0 \leq x,y \leq 1$
 - b) $\int_0^1 \int_0^1 f(x,y)dxdy = 1$
 - c) $P(X,Y \in S) = \int_S \int f(x,y)dxdy$.
5. Two events are independent if, and only if, $P(A \text{ and } B) = P(A) \cdot P(B)$. Two random variables are independent if, and only if, $f(x,y) = g(x)h(y)$ where:

$g(x) = \int_0^1 f(x,y)dy$ marginal distribution of X

$h(y) = \int_0^1 f(x,y)dx$ marginal distribution of Y .
6. The mean of a random variable X is defined as $\int_0^1 x g(x)dx$. The mean of a function of two random variables $A(X,Y)$ is defined as $\int_0^1 \int_0^1 A(x,y)f(x,y)dxdy$.

The propagation of mean values through fault trees will now be considered for several simple examples with independent or mutually exclusive inputs. Only AND, OR, and NOT gates need to be considered since all Boolean statements (i.e., fault trees) can be formed using only these gates (G-9).

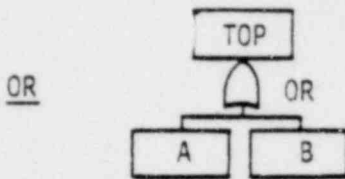


By the definition of independent events, it is known that the Boolean representation of an AND gate is $P(A \text{ AND } B) = P(A)P(B)$. $P(A)$ and $P(B)$ are treated as though they are random variables, so $P(A \text{ AND } B)$ is also a random variable, which can be written as $P(A \text{ AND } B) = X \cdot Y$. This equality implies that the mean value of the random variable $P(A \text{ AND } B)$ is equal to the mean value of the random variable $(X \cdot Y)$.

The mean of $(X \cdot Y)$ is, by definition, equal to $\int_0^1 \int_0^1 xyf(x,y)dx dy$. Since A and B are independent events, $P(A)$ and $P(B)$ are independent random variables. Therefore, $X(=P(A))$ and $Y(=P(B))$ are independent random variables, and it can be seen that:

$$\begin{aligned} \text{mean of } X \cdot Y &= \int_0^1 \int_0^1 xyf(x,y)dx dy = \int_0^1 \int_0^1 xyg(x)h(y)dx dy \\ &= (\int_0^1 xg(x)dx)(\int_0^1 yh(y)dy) = (\text{mean } X) (\text{mean } Y). \end{aligned}$$

In other words, if the mean value of $P(A)$ and the mean value of $P(B)$ are input to a simple AND gate (A and B independent), the use of the Boolean formula $P(A \text{ AND } B) = P(A)P(B)$ gives the mean value of $P(A \text{ AND } B)$.



The Boolean algebra calculation for a simple OR gate makes use of the well-known (G-8) formula that $P(A \text{ OR } B) = P(A) + P(B) - P(A \text{ AND } B)$. A and B are assumed to be independent events, so this can be written as $P(A \text{ OR } B) = P(A) + P(B) - P(A)P(B)$. In addition, $P(A)$ and $P(B)$ are treated as though they are random variables. Therefore, $P(A \text{ OR } B)$ is also treated as a random variable and can be written $P(A \text{ OR } B) = X + Y - X \cdot Y$. This equality implies that the mean of the random variable $P(A \text{ OR } B)$ is equal to the mean value of the random variable $(X + Y - X \cdot Y)$.

The mean value of $(X + Y - X \cdot Y)$ is, by definition, equal to:

$$\int_0^1 \int_0^1 (x + y - xy)f(x,y)dx dy = \int_0^1 \int_0^1 xf(x,y)dx dy + \int_0^1 \int_0^1 yf(x,y)dx dy - \int_0^1 \int_0^1 xyf(x,y)dx dy;$$

by the definition of independence:

$$= \int_0^1 x \left(\int_0^1 f(x,y)dy \right) dx + \int_0^1 y \left(\int_0^1 f(x,y)dx \right) dy - \int_0^1 \int_0^1 xyg(x)h(y)dx dy$$

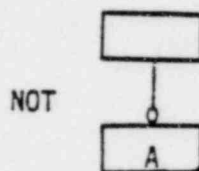
by the definition of marginal distributions:

$$= \int_0^1 xg(x)dx + \int_0^1 yh(y)dy - \left(\int_0^1 xg(x)dx \right) \left(\int_0^1 yh(y)dy \right)$$

$$= (\text{mean } X) + (\text{mean } Y) - (\text{mean } X)(\text{mean } Y).$$

In other words, if the mean value of $P(A)$ and the mean value of $P(B)$ are input to a simple OR gate (A and B independent), the use of the formula $P(A \text{ OR } B) = P(A) + P(B) - P(A)P(B)$ yields the mean value of $P(A \text{ OR } B)$.

Now consider the case when A and B are mutually exclusive, and thus dependent. Then $P(A \text{ AND } B) = 0$, and $P(A \text{ OR } B) = P(A) + P(B) - P(A \text{ AND } B) = P(A) + P(B)$. This equality would then imply that the mean value of the random variable $P(A \text{ OR } B)$ is equal to the mean value of the random variable $P(A) + P(B)$, i.e., $X + Y$. Therefore, for a simple OR gate, the propagation of mean values would hold with the formula $P(A \text{ OR } B) = P(A) + P(B)$, where A and B are dependent and mutually exclusive.



It is also known that $P(\text{NOT } A) = 1 - P(A)$. Again $P(A)$ is treated as a random variable, so $P(\text{NOT } A)$ is treated as a random variable that can be written as $P(\text{NOT } A) = 1 - X$.

Therefore, the mean of $P(\text{NOT } A)$ is equal to the mean of $(1-X)$, which is by definition:

$\text{mean}(1-X) = \int_0^1 (1-x)g(x)dx = \int_0^1 g(x)dx - \int_0^1 xg(x)dx$, since the area under a probability distribution sums to 1:

$$= 1 - (\text{mean } X).$$

In other words, if the mean value of $P(A)$ is input into a NOT gate, the use of the formula $P(\text{NOT } A) = 1 - P(A)$ results in the mean value of $P(\text{NOT } A)$.

Therefore, it has been shown that if the inputs to a gate are assumed independent, then when the data is input as means of the distributions, the propagated value will be the mean value of the gate.

However, it is not necessarily true that all inputs to higher level gates of a fault tree are independent, even if the components are independent. Take, for example, the following tree with independent components A, B, and C:

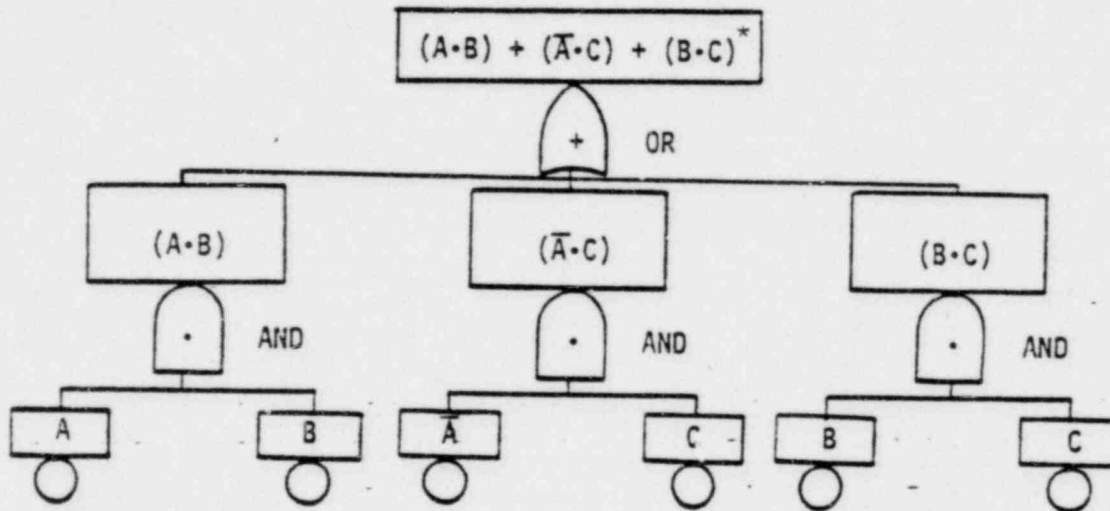


Figure G.3 Example Fault Tree

In this case, the gates immediately above the component level have independent inputs (since the components are assumed to be independent), but the top gate does not have independent inputs. For example, $(A \cdot B)$ and $(B \cdot C)$ are not independent since component B appears in both inputs. Therefore, the previous discussion about simple gates must be extended to apply to this tree.

All Boolean expressions (i.e., fault trees) can be transformed into equivalent and unique (principal disjunctive normal) form also known as the "sum of products canonical form" (see G-9). In this form the propagation of means is valid. A simple method of creating this form is by using a truth table.

For example, the following truth table may be formed from the fault tree in Figure G.3, showing all possible success or failures of the component, A, B, and C.

* + = OR
 · = AND
 \bar{A} = NOT A

Table G.1 TRUTH TABLE OF EXAMPLE FAULT TREE
(See Figure G.3)

A	B	C	A·B	$\bar{A}\cdot C$	B·C	$(A\cdot B) + (\bar{A}\cdot C) + (B\cdot C)$
1*	1	1	1	0	1	1
1	1	0	1	0	0	1
1	0	1	0	0	0	0
1	0	0	0	0	0	0
0*	1	1	0	1	1	1
0	1	0	0	0	0	0
0	0	1	0	1	0	1
0	0	0	0	0	0	0

Hence, it can be seen that the top event, $(A\cdot B) + (\bar{A}\cdot C) + (B\cdot C)$, occurs only when A AND B AND C occurs, or when A AND B AND NOT C occurs, or when NOT A AND B AND C occurs, or when NOT A AND NOT B AND C occurs. Another way of writing this is:

$$(A\cdot B) + (\bar{A}\cdot C) + (B\cdot C) \Leftrightarrow (A\cdot B\cdot C) + (A\cdot B\cdot \bar{C}) + (\bar{A}\cdot B\cdot C) + (\bar{A}\cdot \bar{B}\cdot C)$$

(principal disjunctive normal form)

Notice that a principal disjunctive normal form contains a series of unique terms, each term containing every component exactly once.

Assuming that each component is independent from the others, then all components in each term are independent, since each component occurs only once. Furthermore, all the terms in the principal disjunctive normal form are mutually exclusive. The discussion above of a simple OR gate, with two mutually exclusive input events, can be easily extended to a case with more than two mutually exclusive input events.

*1 = the component fails
0 = the component does not fail

In this manner, a fault tree may be changed to a Boolean logic form, i.e., principal disjunctive normal form of a series of unique terms. Therefore, when distinct component inputs are independent, the previous discussion applies, and mean value inputs to a fault tree lead to the mean value of the top event.

component reliability. Similarly, component reliability is very much affected by the environment in which it operates. For example, valve reliability is strongly affected by the moisture in the environment, the operating temperature, and the fluid with which it operates.

The major limitations in the component failure rate characterization are:

- Assumption of similar environment for all components of the same type
- No modeling of the age dependence of failure rates
- Treatment of all components of similar types as part of the identically same population
- Only a portion of all failures have been reported, processed, and finally appear in one of the available data sources, such as:
 - NRC LER file
 - WASH-1400
 - GE component information retrieval (CIR) system
 - NPRDS (nuclear plant reliability data system).

In addition to the above items, there is a potential concern arising from the implementation of component failure rate data. The data reported is generally taken and treated as random independent failures. The inclusion of dependent failures, known as common-cause or common-mode failures, plays a very important role in the evaluation of risk.

The term common-mode refers to two or more items failing as a result of the same cause or failure mode. Concern for this type of problem arises since there is a limit to the attainable reliability with a single component or subsystem; but, if components fail independently of each other, it is possible to use several so that the failure of one or more is circumvented by one or more operational units.

It is apparent that common-mode failures are primarily significant for redundant systems. For example, suppose a subsystem has two redundant components each with a probability of failure of 0.01; the probability that both have failed is $0.01 \times 0.01 = 0.0001$. But if the components fail at the same time due to a common cause, the probability of the redundant system failure is that of a single component: 0.01. The change between coupled and uncoupled failures is even larger for more redundant systems. Common-mode or common-cause failures are very important, **but** there are aspects that assist in their identification and elimination. These are noted as follows:

1. Common-mode effects have safety significance only in redundant systems, thus all systems need not be examined for these effects. This selection must be done with care, however, because some backup arrangements are not always obvious.
2. Common-mode effects are minimized by design, manufacture, and procedural diversity.
3. Isolation barriers are used to minimize such common-mode effects as pipe whip, fire, and missiles.

WASH-1400 used a mathematical artifice when common-mode effects could not be identified. This was severely criticized by the Lewis Committee and has been mentioned previously. WASH-1400 employed the rationale that the true failure rate of a redundant system is bounded at the low end by assuming completely independent subsystems and at the high end by assuming completely dependent subsystems keyed to the failure of the highest failure rate subsystem. Having established these bounds, the RSS chose the expected value as being the geometric mean of these extremes. It is obvious that there is no hardware basis for this selection and it is avoided in the "realistic" analyses presented in the LGS risk assessment.

There is another type of common-mode coupling that has received less attention than the type discussed above. This is the fact that the consequences from one type of failure can modify the operating

environment of other components and result in both accelerated failure as well as possibly immediate failure. This type of common-mode is primarily addressed through the Code of Federal Regulations and through the Regulatory Guides by requiring equipment certification as to operability in a degraded environment. In the LGS analysis, the environmental changes that result as an accident progresses through its sequences is continually examined. When the environment exceeds operability requirements for a component, it is assumed to fail. This is a conservative assumption, but more realistic treatment cannot be justified in the absence of test data extending beyond design specifications (see also Section I.3).

I.4.2 Meteorological Data

The consequence analysis carried out using the CRAC computer code makes use of several simplifying assumptions to model the transport of fission products from the site and release to the environment. The principal components of the modeling are:

- Wind direction at various elevations
- Wind speed at various elevations
- Atmospheric/plume dispersion
- Rainfall.

The wind direction and speed could be modeled continuously at all heights and distances from the site. However, the available data limits the model. The site meteorological data is collected from one tower with some backup data available from satellite towers. However, the existing meteorological data does not provide a continuous plot of wind direction and speed in all directions at all distances. The CRAC model used in the consequence analysis employs wind direction and speed during the course of the accident.

The CRAC model also incorporates some dispersion of the plume arising from the turbulence of the air and roughness of the terrain. The plume dispersion can have an effect on the prediction of early fatalities at large distances from the plant, since early fatalities are a threshold effect.

In a review paper, Van der Hoven (I-1) summarizes data on atmospheric dispersion experiments performed on various terrains in Washington, Idaho, Louisiana, Pennsylvania, and Tennessee. The Washington experiment used an ^{85}Kr tracer and the other measurements were made with an SF_6 (nonradioactive) tracer. These measurements were performed for Pasquill stability classes E, F, and G, in wind speeds less than 2 m/sec. The terrains were classified into the following types:

- Type I - Smooth desert-like (Washington and Idaho)
- Type II - Wooded flat terrain (Louisiana and Pennsylvania)
- Type III - Wooded hilly terrain (Tennessee).

The author concludes that for flat forested surfaces, the diffusion model (CRAC code) will overpredict the peak concentration by 20 to 40, whereas for hilly forested terrain, the overprediction is 50 to 500.

I.4.3 Population Data

The population at various distances and directions from the plant can be determined with sufficient accuracy to characterize the health effects resulting from the postulated accidents. Since population is grouped by sector, small uncertainty is introduced into the analysis. However, the modeling of the evacuation, sheltering, and breathing rates of the population during an accident sequence plays more important roles in determining consequences to the population in an accident.

- Rapid evacuation could result in very few or no fatalities even under the most adverse accident conditions.
- Adequate sheltering could minimize the population dose.
- Accurate modeling of the breathing rates can also drastically change the early fatality estimate.

I.4.4 Accident Consequences

There are very few benchmarks which can be used to establish the accuracy of the consequence models used in the nuclear power plant analysis. Data on the behavior of reactor systems are being gathered by many large-scale (e.g., LOFT and Semiscale) as well as laboratory-scale controlled experiments on core release functions, plate-out factors, and other factors that attenuate the release of radioactivity. Since it is impractical to perform full scale replications of accident sequences, the fragmentary data from accidents such as TMI-2, Windscale, and SL-1 must be used in system models.

There are many barriers that are designed to confine or disperse the radioactivity in the event of an accident. In general terms, these barriers are: fuel matrix, coolant, reactor system cooling boundary, primary containment, secondary containment, and the atmospheric dispersion of material before it reaches the public. Each barrier has varying abilities to confine or dissipate the materials depending on the barrier structure, geometry, and environmental chemistry and physics. The amount of material retained by or on the various barriers may be calculated using simplifying assumptions, but many inaccuracies are involved in using laboratory data for modeling the amount of release in a damaged nuclear power plant. The reason for these uncertainties are problems in applying small sample laboratory data to a model of a complex power plant and extrapolating the data to accident conditions. Because of the uncertainties, the WASH-1400 analysis tended to be conservative, i.e., predict higher than expected releases so that the consequences would not be underestimated.

Comparisons can be made of these predictions and the amount of material released in severe reactor accidents. Recently, there has been an attempt (I-2) to calculate the SL-1 accident using updated versions of the CORRAL and CRAC codes used in WASH-1400. Even though the models were more detailed than those of WASH-1400, and the SL-1 geometry is simpler, the results overestimated the release by four times the amount actually observed.

The recent accident at TMI-2 provides useful data on actual releases from damaged cores (I-3) as shown in Table I-1.

Table I.1

CORE RELEASE FRACTIONS OF TOTAL INVENTORY FROM TMI-2

Material	RCS	Release to		Auxiliary Building	Environment
		Reactor Building			
		Gaseous	Liquid		
Noble Gas	0.6	0.6		0.05	0.05
Iodine	0.3	0.006	0.2	0.03	2×10^{-7}
Ce	0.5	<0.01	0.4	0.03	
Be, Sr	0.02		0.01		