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DESIGNATED ORIGINAL

Certified By

ENCLOSURE 3

LIMERICK GENERATING STATION

PROBABILISTIC RISK ASSESSMENT

SUMMARY OF CHRONOLOGY

- MAY 6, 1980 - NRC LETTER
- MAY 21, 1980 - NRC MEETING - SCOPE OF STUDY
- OCTOBER 2, 1980 - NRC MEETING - INTERIM PROGRESS REPORT
- DECEMBER 9, 1980 - NRC MEETING - RESULTS
- MARCH 17, 1981 - SUBMITTAL OF REPORT

NRC LETTER
MAY 6, 1980

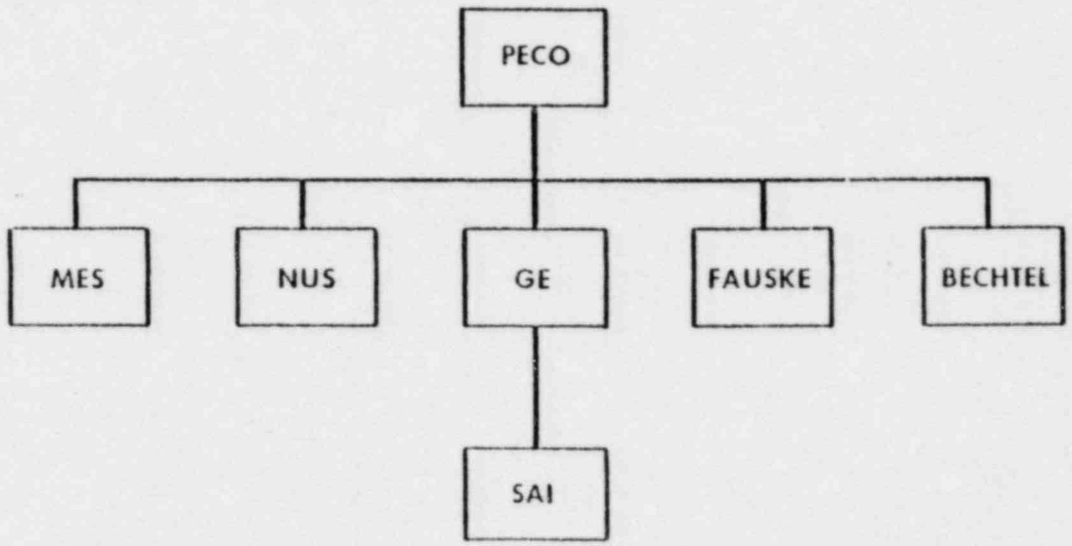
- CONDUCT A PRELIMINARY RISK ASSESSMENT
- RECOGNIZE WASH 1400 CRITICISMS
- COMPLETE EVALUATION IN 120 DAYS

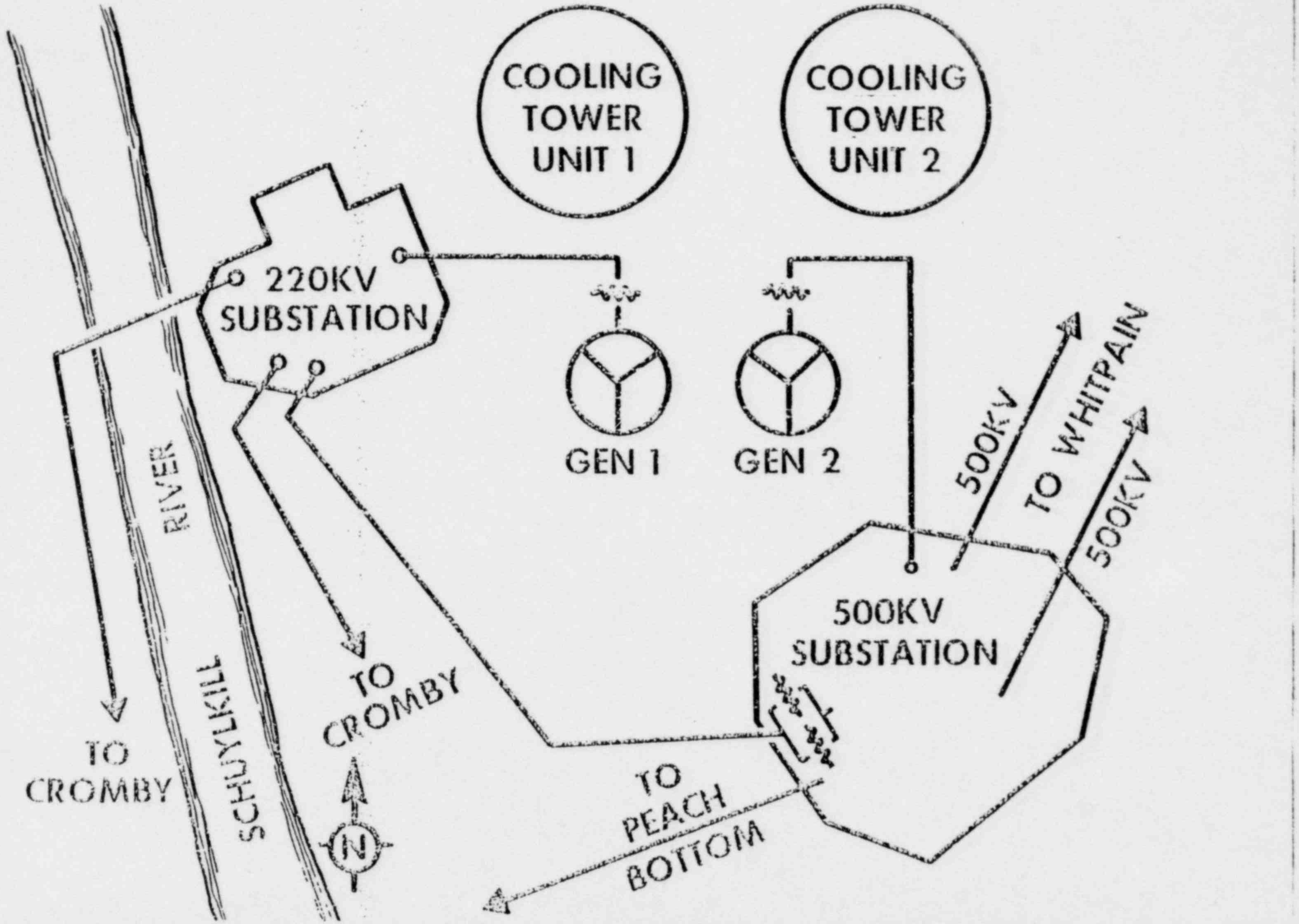
LGS PRA SCOPE

- EARLY FATALITIES FOR ALL BUT EXTERNAL EVENTS
- 1970 POPULATION
- SITE DIFFERENCES
- DESIGN DIFFERENCES
- DATA/METHODOLOGY DIFFERENCES

ij

LGS PRA




COOLING TOWER UNIT 1
COOLING TOWER UNIT 2
220KV SUBSTATION
GEN 1
GEN 2
500KV
TO WHITPAIN
500KV
500KV SUBSTATION
RIVER
SCHUYLKILL
TO CROMBY
TO CROMBY
N
TO PEACH BOTTOM

CASES STUDIED

- WASH 1400 at Limerick Site
- Limerick Plant at Limerick Site

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WASH 1400 PLANT AT LIMERICK SITE

- WASH 1400 Probabilities
- WASH 1400 Release Fractions
- WASH 1400 Methods
- Limerick Site Population
- Limerick Site Meteorology

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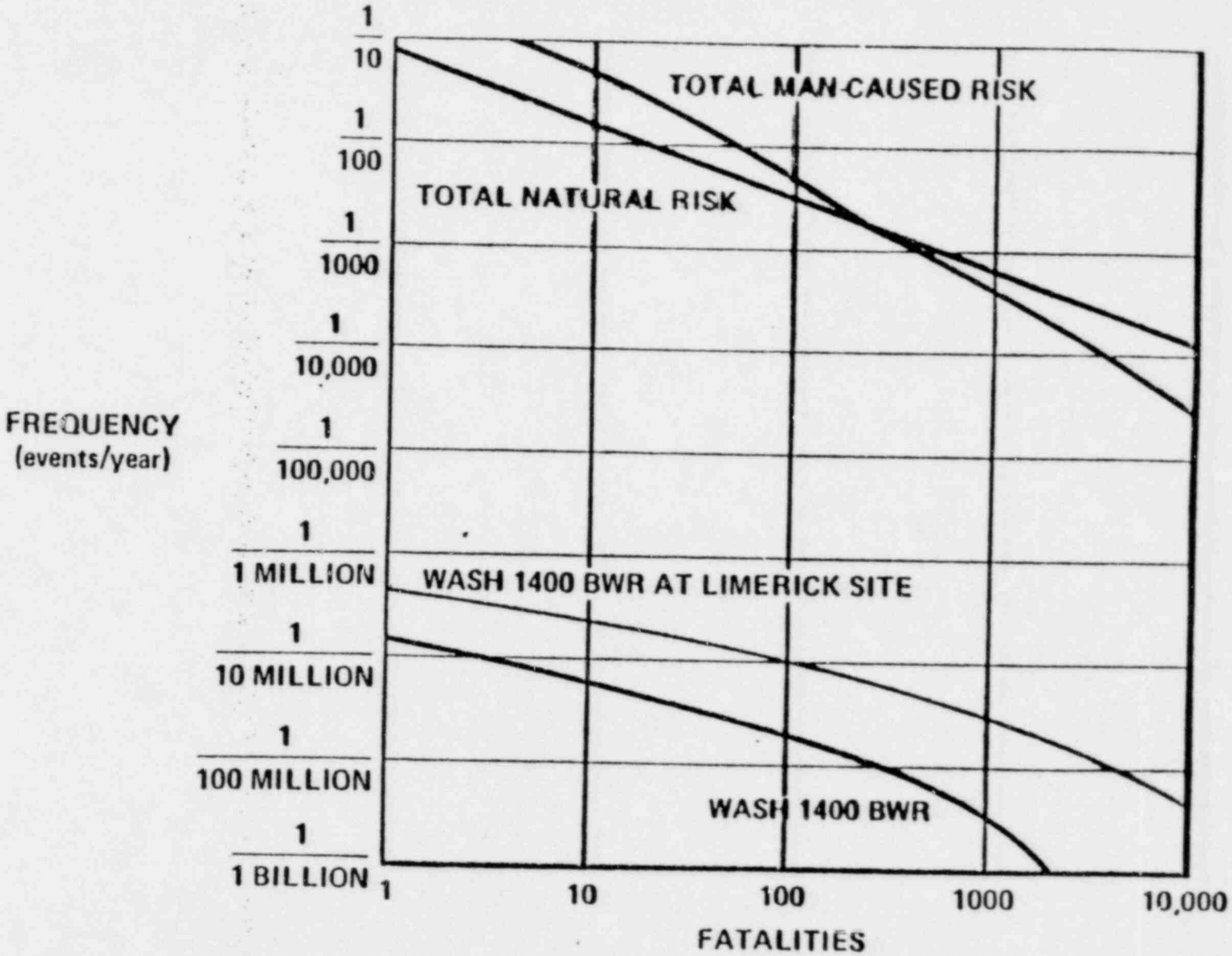
Limerick Preliminary Risk Assessment

Site Comparison



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LIMERICK PLANT AT LIMERICK SITE

- Limerick Plant Features
- Limerick Site Features
- Updated Data and Methods

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LIMERICK ANALYSIS

- Systems -- Limerick (BWR/4)
- Procedures -- Limerick Projection
- Operating Experience Data -- Philadelphia Electric Company Where Applicable
- Containment -- Limerick Mark II
- Sequences -- Limerick Specific
- Containment Analysis -- Limerick Specific
- Consequences -- Limerick Site-Specific

Risk -- Limerick Specific

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LIMERICK PLANT AT LIMERICK SITE

METHODOLOGY

ACCIDENT INITIATORS

- Plant Initially Operating at Power
- Plant Safety Challenge Occurs
  - Normal Transient
  - Small Loss-of-Coolant Accident
  - Large Loss-of-Coolant Accident
  - Transient Without SCRAM

Allows Specific Capability to be Analyzed

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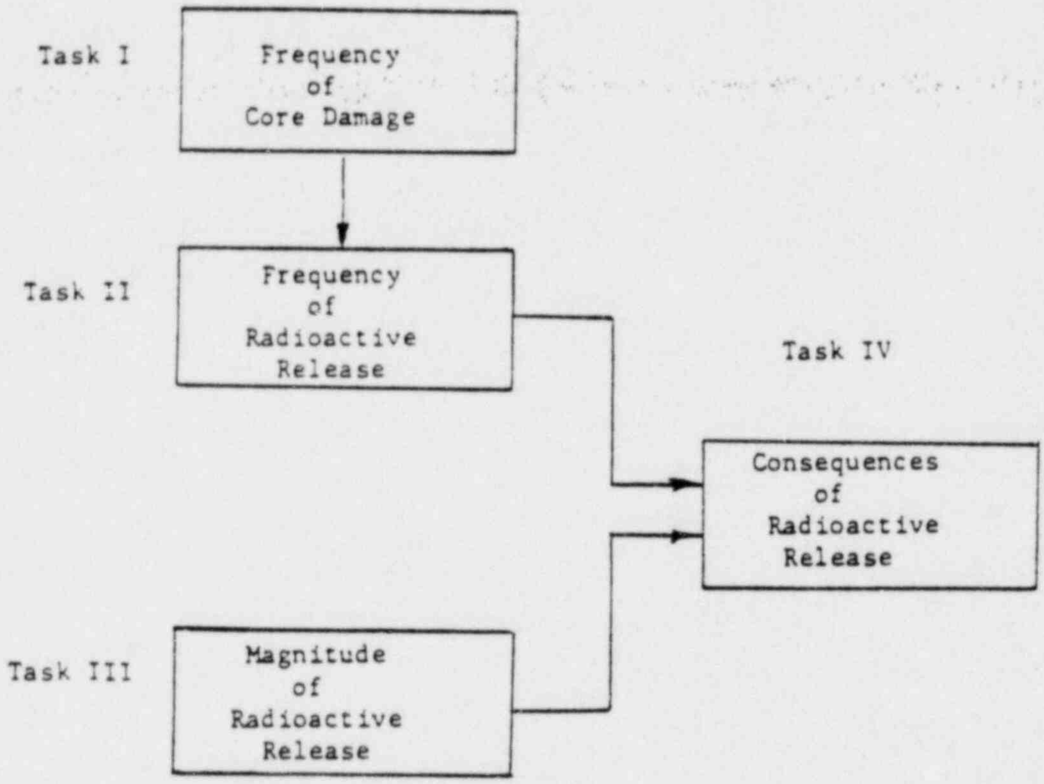
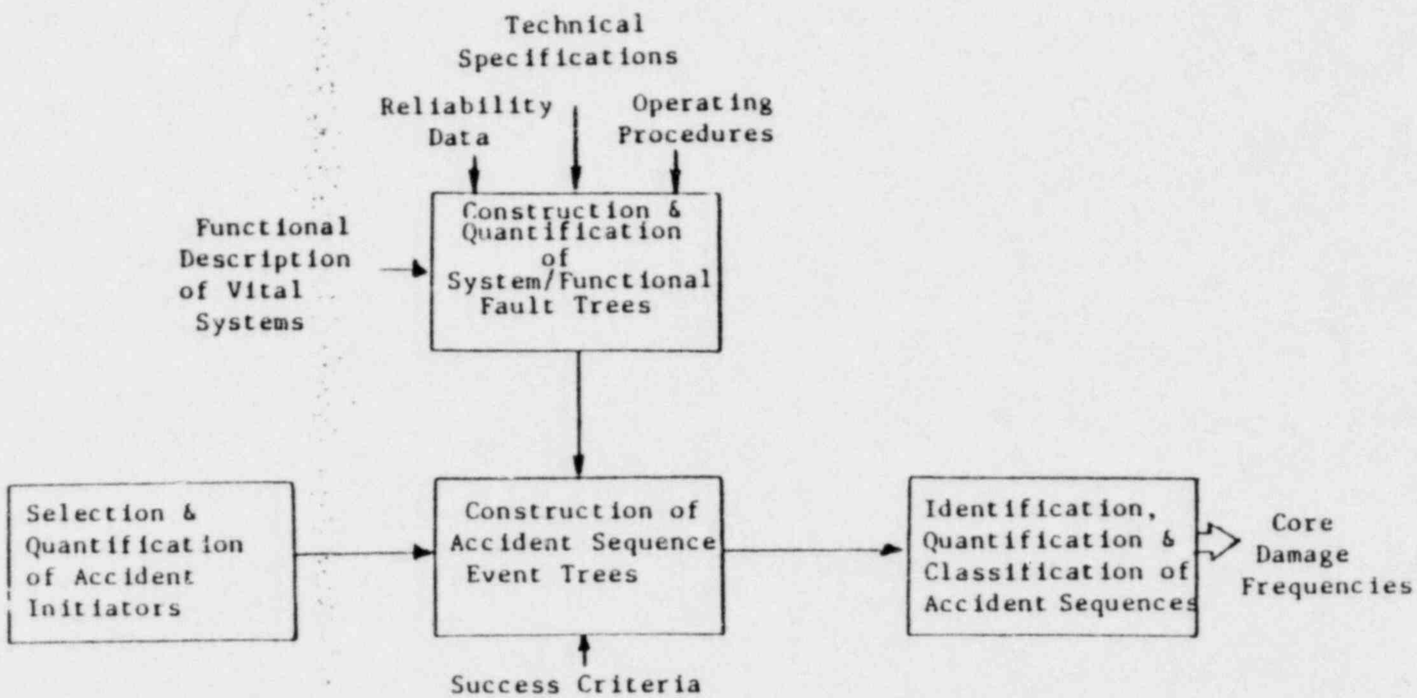


Major Tasks of the Analysis



Determination of Core Damage Frequency (Task I)

Accident Initiators

- Transients
- Transients Without Scram
- Loss of Coolant Accidents

Turbine Trip
MSIV Closure
Loss of Feedwater
IORV
Loss of Offsite Power

Large
Small
Very Small

Indepth Analysis for Transients Assured

8547 54

System Level Fault Trees

- CRD INJECTION
- FEEDWATER
- CONDENSATE
- AUTOMATIC DEPRESSURIZATION SYSTEM
- RESIDUAL HEAT REMOVAL
  - LOW PRESSURE COOLANT INJECTION
  - CONTAINMENT SPRAY
  - RHR SERVICE WATER
- HIGH PRESSURE COOLANT INJECTION
- REACTOR CORE ISOLATION COOLING
- LOW PRESSURE CORE SPRAY
- EMERGENCY SERVICE WATER
- POWER: AC (NORMAL), AC (EMERGENCY), DC
- DIESELS
- STANDBY LIQUID CONTROL

FAULT TREES ARE LOGIC MODELS FOR FUNCTIONS

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GENERIC ACCIDENT SEQUENCES

- Melting Before Containment Failure
- Melting After Containment Failure
- Transient Without SCRAM
- Transient Without SCRAM (Heat Removal Failure Case)

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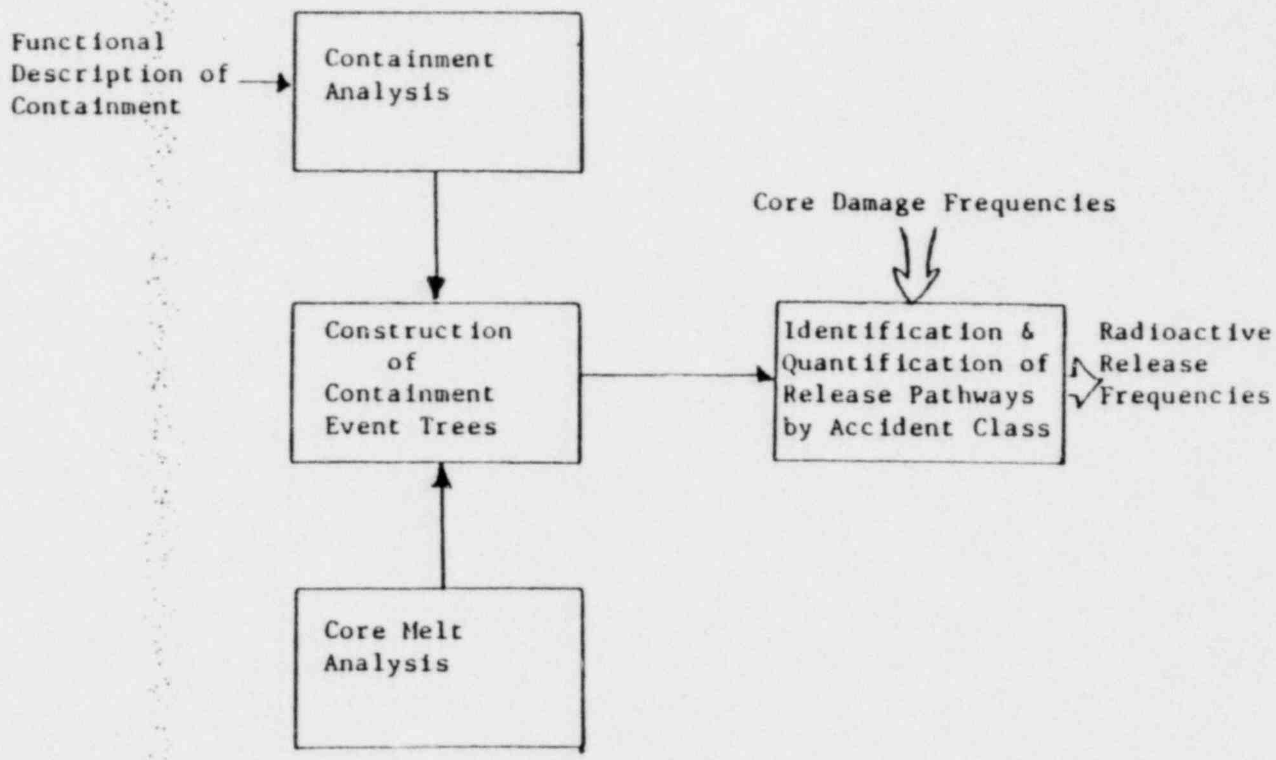
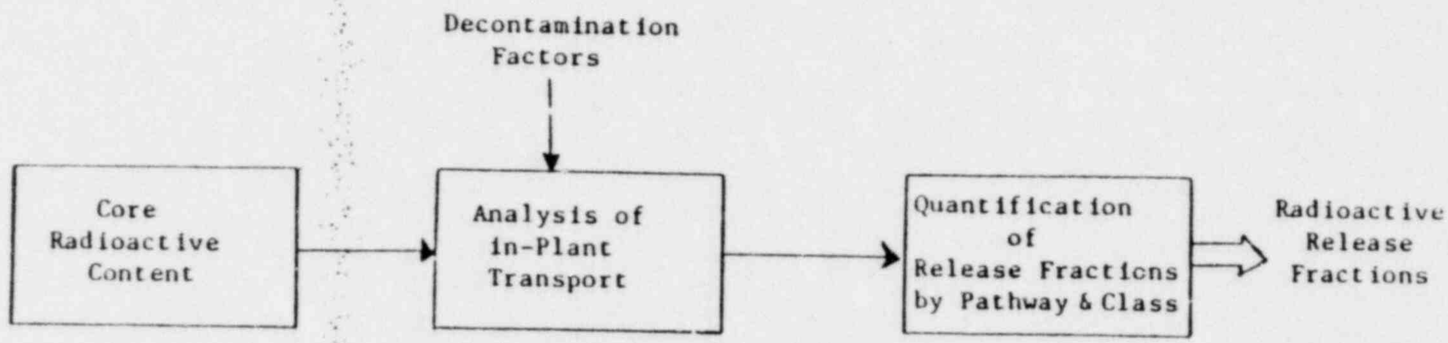
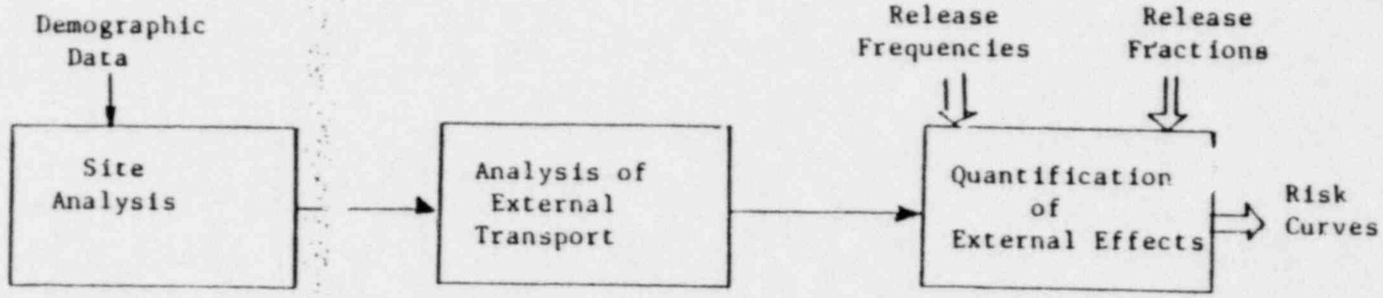


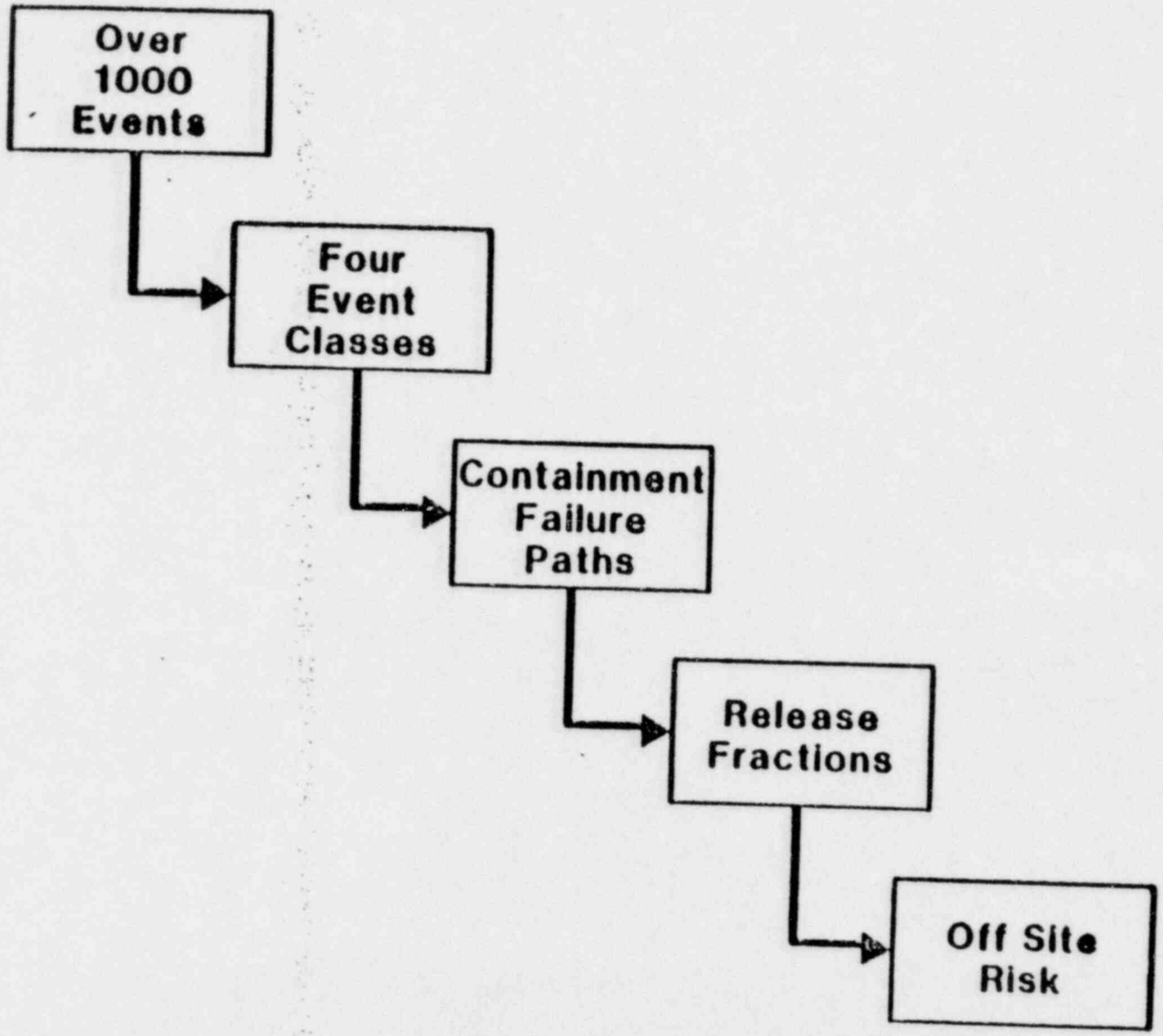
Determination of Radioactive Release Frquency (Task II)



Determination of Magnitudes
of Radioactive Release (Task III)



Determination of External Effects (Task IV)


Over 1000 Events
Four Event Classes
Containment Failure Paths
Release Fractions
Off Site Risk

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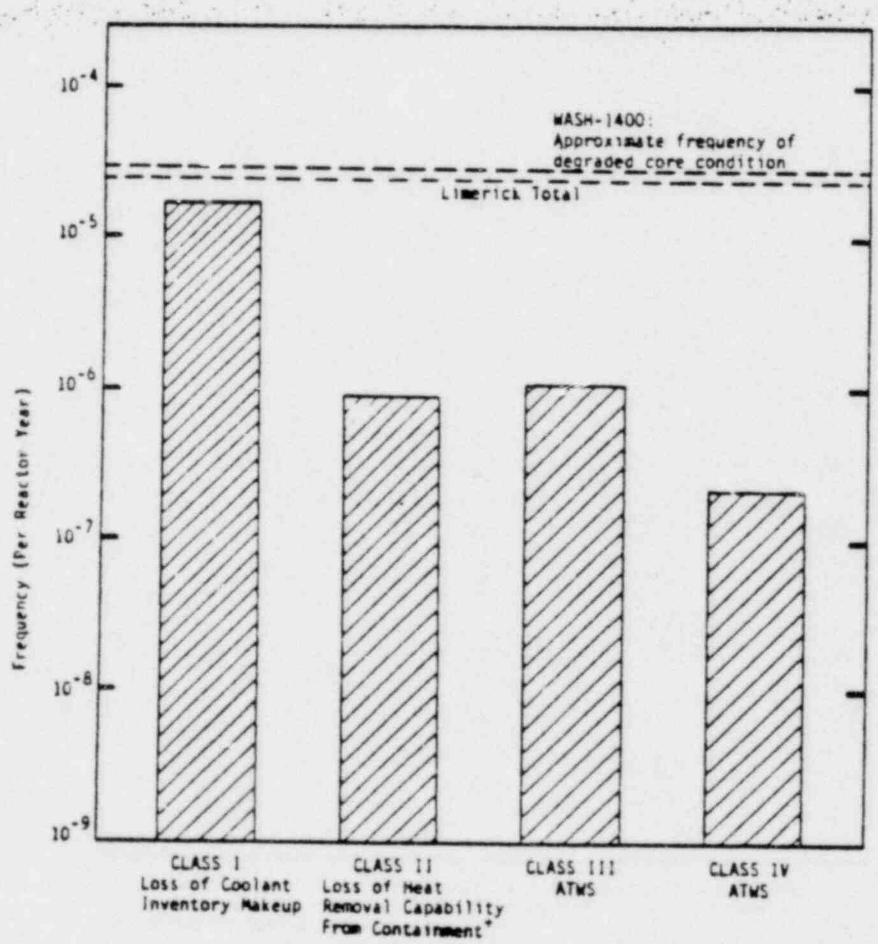
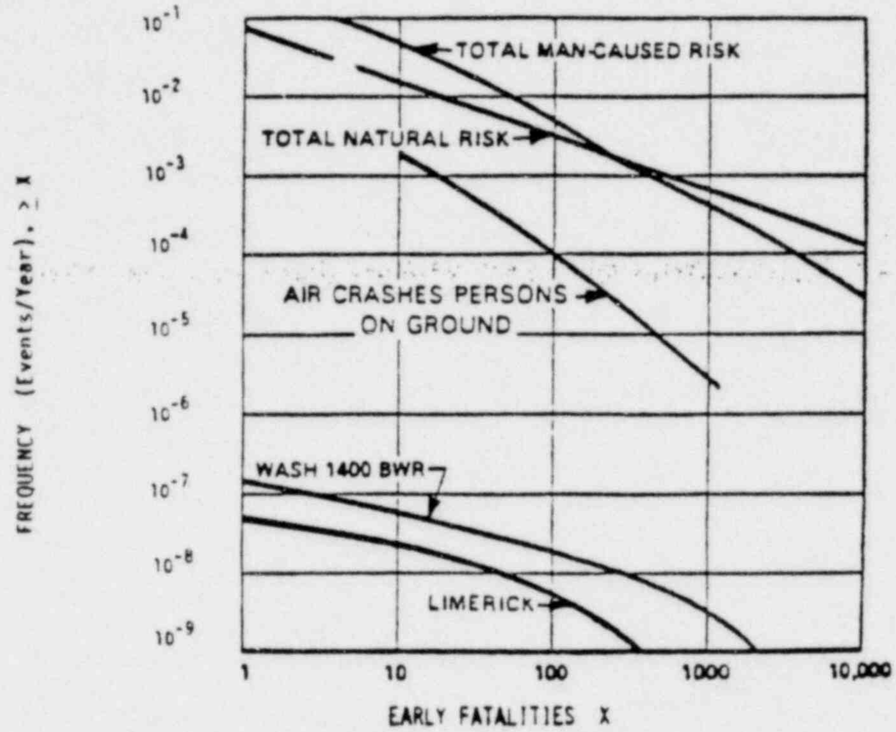
LIMERICK PLANT AT LIMERICK SITE

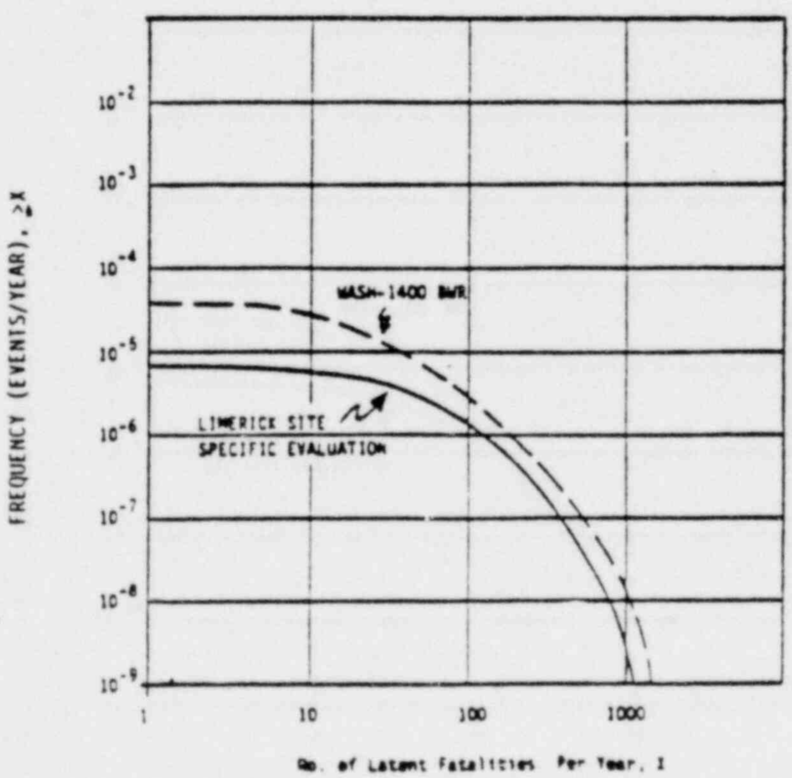
RESULTS

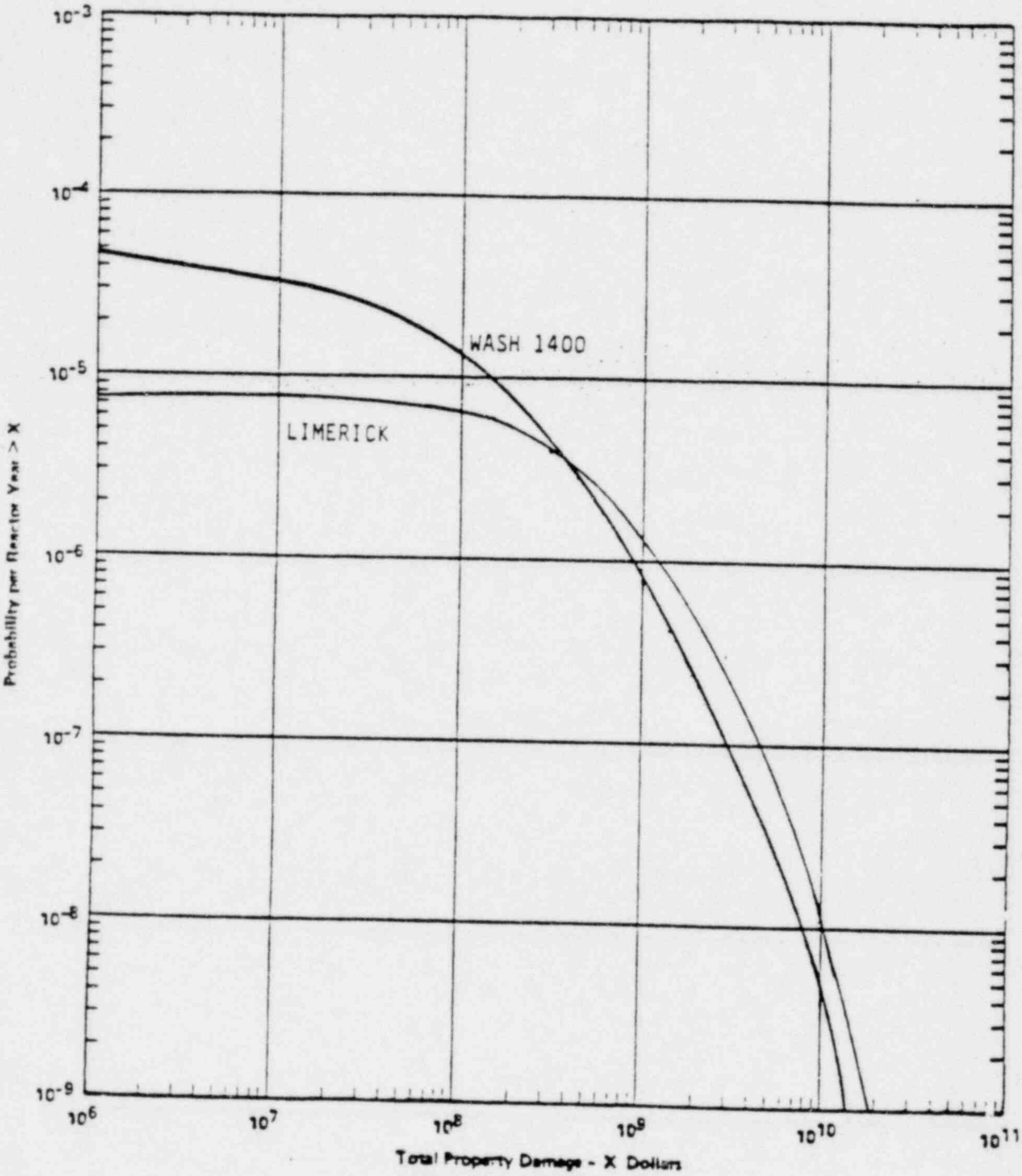
Summary of the Accident Sequence Frequencies Leading to Degraded Core Conditions Summed Over All Accident Sequences within a Class.



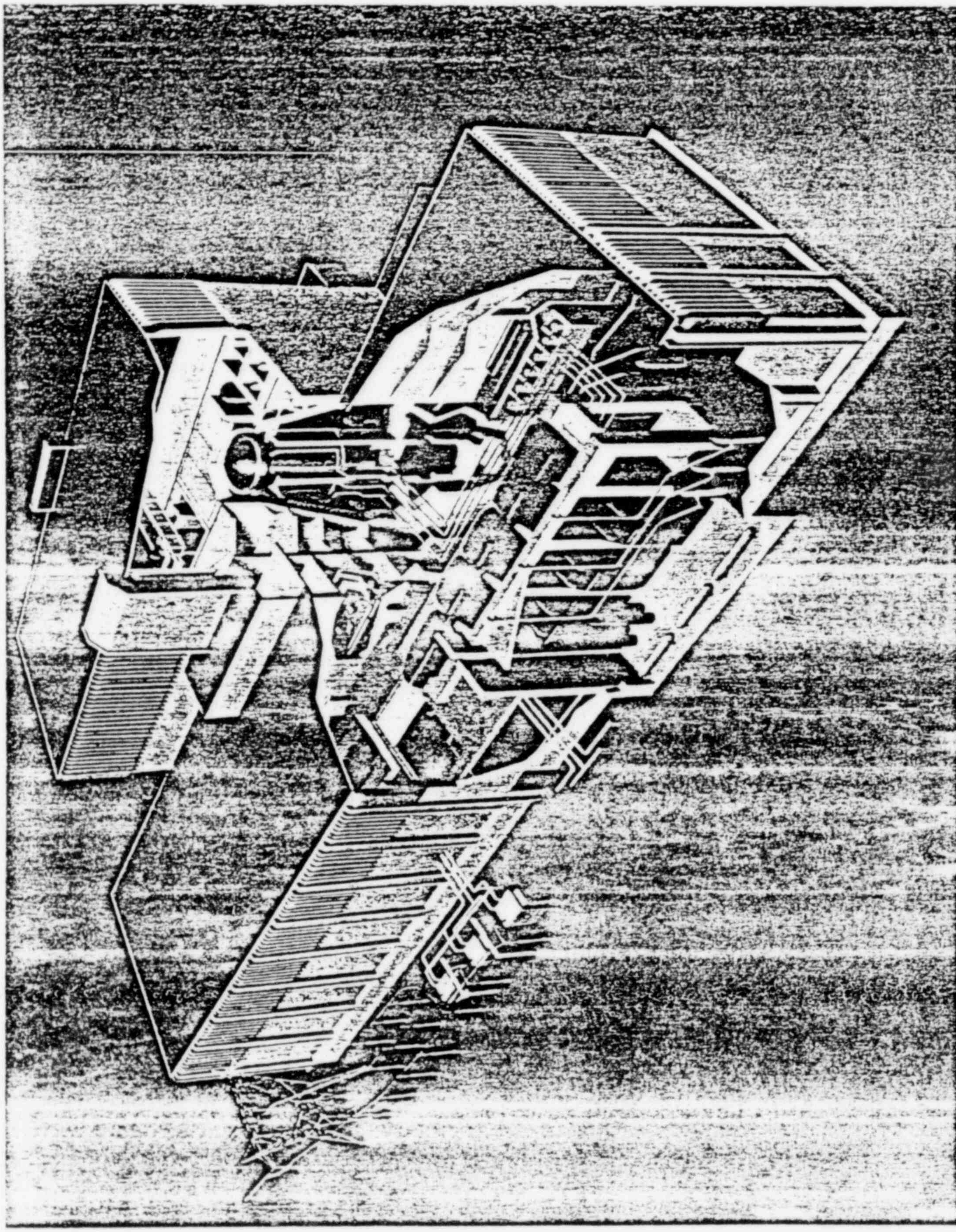
Limerick/WASH-1400 Risk Comparison

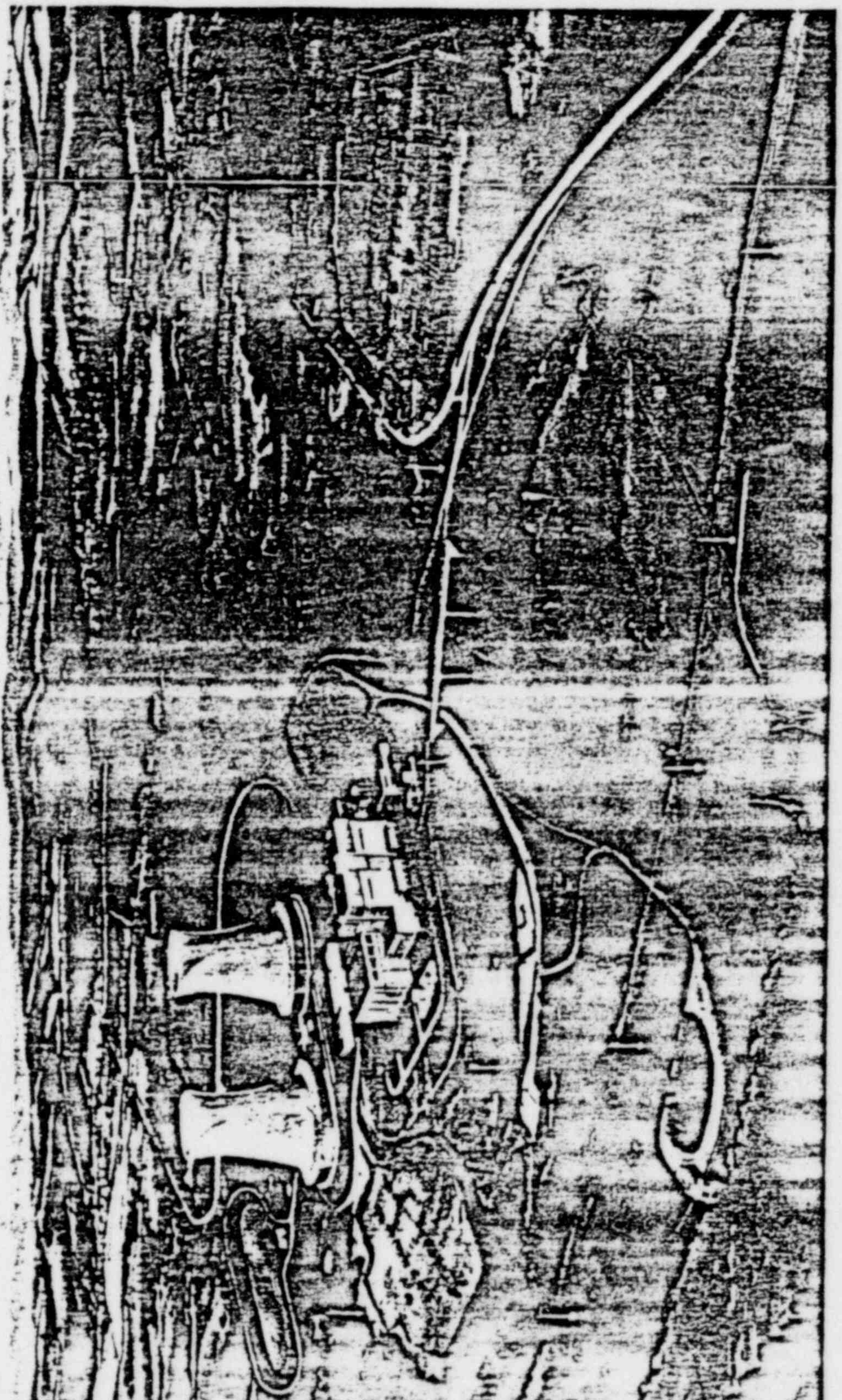


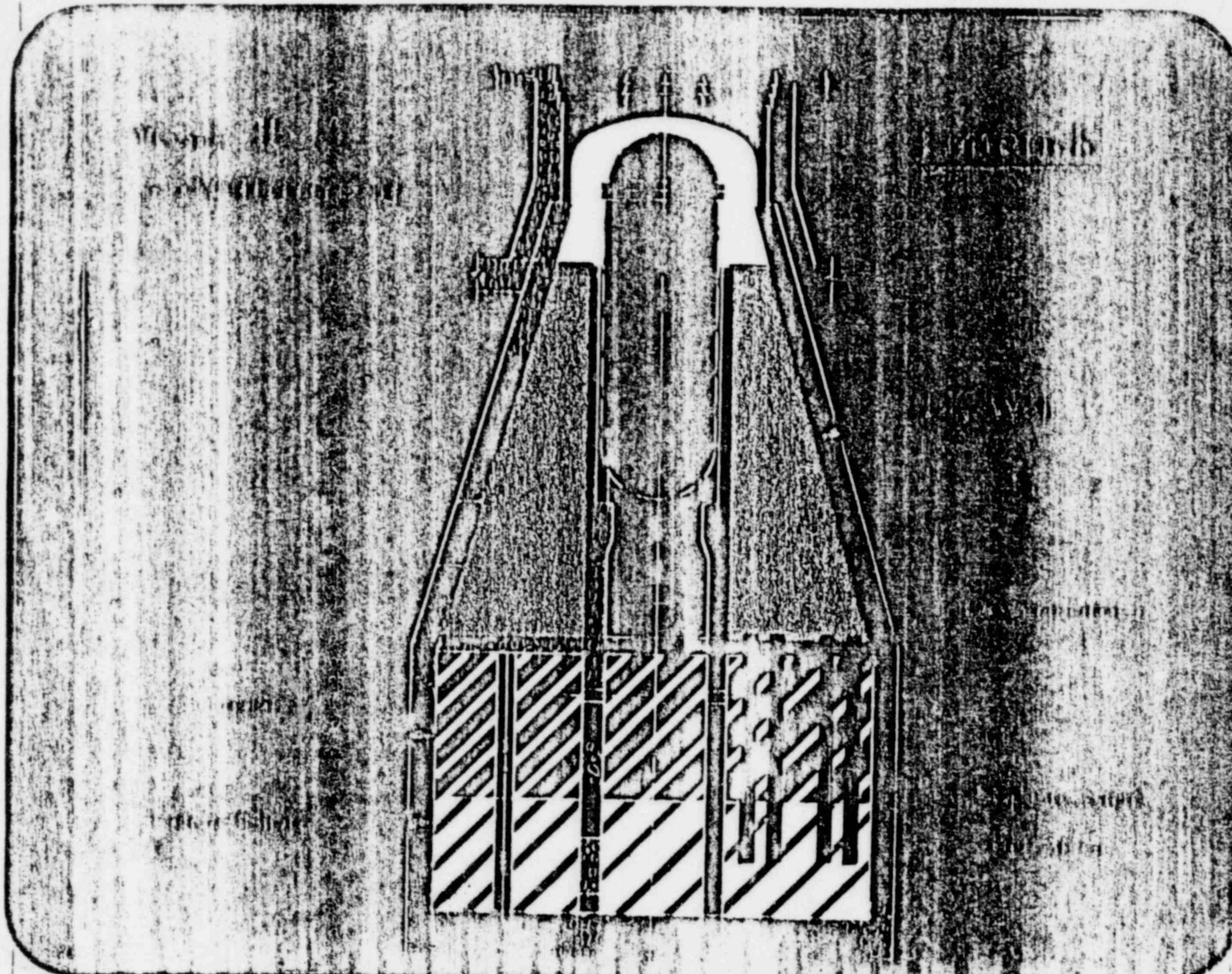




Comparison of WASH-1400 and Limerick







GEMINI

PURPOSE

PROVIDE A BRIEF OVERVIEW OF SELECTED SAFETY RELATED SYSTEMS.

DISCUSSION TOPICS

- MAJOR SYSTEM COMPONENTS
- FLOW RATES
- FLOW PATHS
- LOGIC

SYSTEMS TO BE DISCUSSED

- ECCS SYSTEMS
  - HIGH PRESSURE COOLANT INJECTION (HPCI)
  - AUTOMATIC DEPRESSURIZATION (ADS)
  - CORE SPRAY (CS)
  - LOW PRESSURE COOLANT INJECTION (LPCI)
- REACTOR CORE ISOLATION COOLING (RCIC)
- RESIDUAL HEAT REMOVAL (RHR)
  - SUPPRESSION POOL COOLING
  - CONTAINMENT SPRAY
  - STEAM CONDENSING
- RHR SERVICE WATER
- STANDBY LIQUID CONTROL

HPCI

- NUMBER OF PUMPS - 1 (TURBINE DRIVEN)
- CAPACITY - 5,600 GPM
- WATER SOURCE - CONDENSATE STORAGE/SUPPRESSION POOL
- INITIATION SIGNAL - REACTOR LOW WATER LEVEL (L2) OR HIGH DRYWELL PRESSURE
- PERMISSIVES - NONE
- TRIP SIGNALS - ISOLATION SIGNALS

  1. STEAM SUPPLY BREAK
     A. HIGH TEMPERATURE IN AREA OF STEAM LINE
     B. HIGH STEAM FLOW
  2. REACTOR LOW PRESSURE
  3. HIGH TURBINE EXHAUST DIAPHRAGM PRESSURE

  TURBINE TRIPS

  1. REACTOR HIGH WATER LEVEL (L8)
  2. HIGH TURBINE EXHAUST PRESSURE
  3. LOW PUMP SUCTION PRESSURE
  4. OVERSPEED

- POWER SUPPLY - DC

ADS

- TOTAL NO. OF SRV'S - 14-ALL PIPED TO SUPRESSION POOL.
- OPERATION - PRESSURE/MANUAL
- ADS FUNCTION - 5 OF 14 VALVES
- OPERATION - PRESSURE/MANUAL/AUTOMATIC
- AVERAGE CAPACITY - 1,035,000 LBS/HR/VALVE
- INITIATION SIGNAL - REACTOR LOW WATER LEVEL (L1)
  AND HIGH DRYWELL PRESSURE
- PERMISSIVES - 1 CS OR 1 RHR PUMP RUNNING
  2 MINUTE TIME DELAY
- POWER SUPPLY - DC

CS

- NUMBER OF PUMPS - 4
- NUMBER OF LOOPS - 2
- CAPACITY - 3,175 GPM/PUMP
- WATER SOURCE - SUPPRESSION POOL/CONDENSATE STORAGE
- INITIATION SIGNAL - REACTOR LOW WATER LEVEL (L1)
  OR HIGH DRYWELL PRESSURE
- PERMISSIVES - REACTOR LOW PRESSURE
- TRIP SIGNAL - MANUAL
- POWER SUPPLY - AC

LPCI

- NUMBER OF PUMPS - 4
- NUMBER OF LOOPS - 4
- CAPACITY - 10,000 GPM/PUMP
- WATER SOURCE - SUPPRESSION POOL
- INITIATION SIGNAL - REACTOR LOW WATER LEVEL (L1) OR HIGH DRYWELL PRESSURE
- PERMISSIVES - REACTOR LOW PRESSURE
- TRIP SIGNAL - MANUAL
- POWER SUPPLY - AC

RCIC

- NUMBER OF PUMPS - 1 (TURBINE DRIVEN)
- CAPACITY - 600 GPM
- WATER SOURCE - CONDENSATE STORAGE/SUPPRESSION POOL
- INITIATION SIGNAL - REACTOR LOW WATER LEVEL (L2)
- PERMISSIVES - NONE
- TRIP SIGNALS - ISOLATION SIGNALS

  1. STEAM SUPPLY PIPE BREAK
     A. HIGH TEMPERATURE IN AREA OF STEAM LINE
     B. HIGH STEAM FLOW
  2. REACTOR LOW PRESSURE
  3. HIGH TURBINE EXHAUST DIAPHRAGM PRESSURE

  TURBINE TRIPS

  1. REACTOR HIGH WATER LEVEL (L8)
  2. HIGH TURBINE EXHAUST PRESSURE
  3. OVERSPEED
  4. LOW PUMP SUCTION PRESSURE

- POWER SUPPLY - DC

SUPPRESSION POOL COOLING

- NUMBER OF HEAT EXCHANGER LOOPS AVAILABLE - 2
- NUMBER OF RHR PUMPS AVAILABLE - 4(2 PER LOOP)
- FLOW PATHS
  - RHR SIDE: SUPPRESSION POOL TO SUPPRESSION POOL
  - RHR SERVICE WATER SIDE: COOLING TOWER/SPRAY POND TO COOLING TOWER/SPRAY POND
- INITIATION SIGNAL - MANUAL
- TRIP SIGNAL - MANUAL
- POWER SUPPLY - AC

CONTAINMENT SPRAY

- NUMBER OF SPRAY LOOPS AVAILABLE
  - DRYWELL - 2
  - WETWELL - 2
- NUMBER OF RHR PUMPS AVAILABLE - 4
- WATER SOURCE - SUPPRESSION POOL
- INITIATION SIGNAL - MANUAL
- PERMISSIVES
  - DRYWELL SPRAY
    DRYWELL PRESSURE, PLUS
    LPCI INITIATION SIGNAL PRESENT, PLUS
    LPCI INJECTION VALVE CLOSED
  - WETWELL SPRAY
    NO LPCI INITIATION, OR
    LPCI INJECTION VALVE CLOSED
- TRIP SIGNAL - MANUAL
- POWER SUPPLY - AC

STEAM CONDENSING

- NUMBER OF HEAT EXCHANGER LOOPS AVAILABLE - 2
- NUMBER OF RHRSW PUMPS AVAILABLE - 4 (2 PER LOOP)
- FLOW PATH
  - RHR SIDE: STEAM FROM MAIN STEAM VIA HPCI STEAM SUPPLY LINE
  - RHR SERVICE WATER: SUCTION FROM AND RETURN TO COOLING TOWER/SUPPRESSION POOL
  - CONDENSATE RETURN TO SUPPRESSION POOL/RCIC PUMP SUCTION
- INITIATION SIGNAL - MANUAL
- TRIP SIGNAL - MANUAL
- POWER SUPPLY - AC

RHR SERVICE WATER

- SHARED SYSTEM
- TWO LOOPS, EACH CONSISTING OF:
  - 1 HEAT EXCHANGER FROM EACH UNIT
  - 2 PUMPS
- CAPACITY - 9,000 GPM/PUMP
- WATER SOURCE
  - SPRAY POND
  - COOLING TOWER
- INITIATION SIGNAL - MANUAL
- PERMISSIVES - NONE
- TRIP SIGNAL
  - MANUAL
  - HIGH RADIATION
- POWER SUPPLY - AC

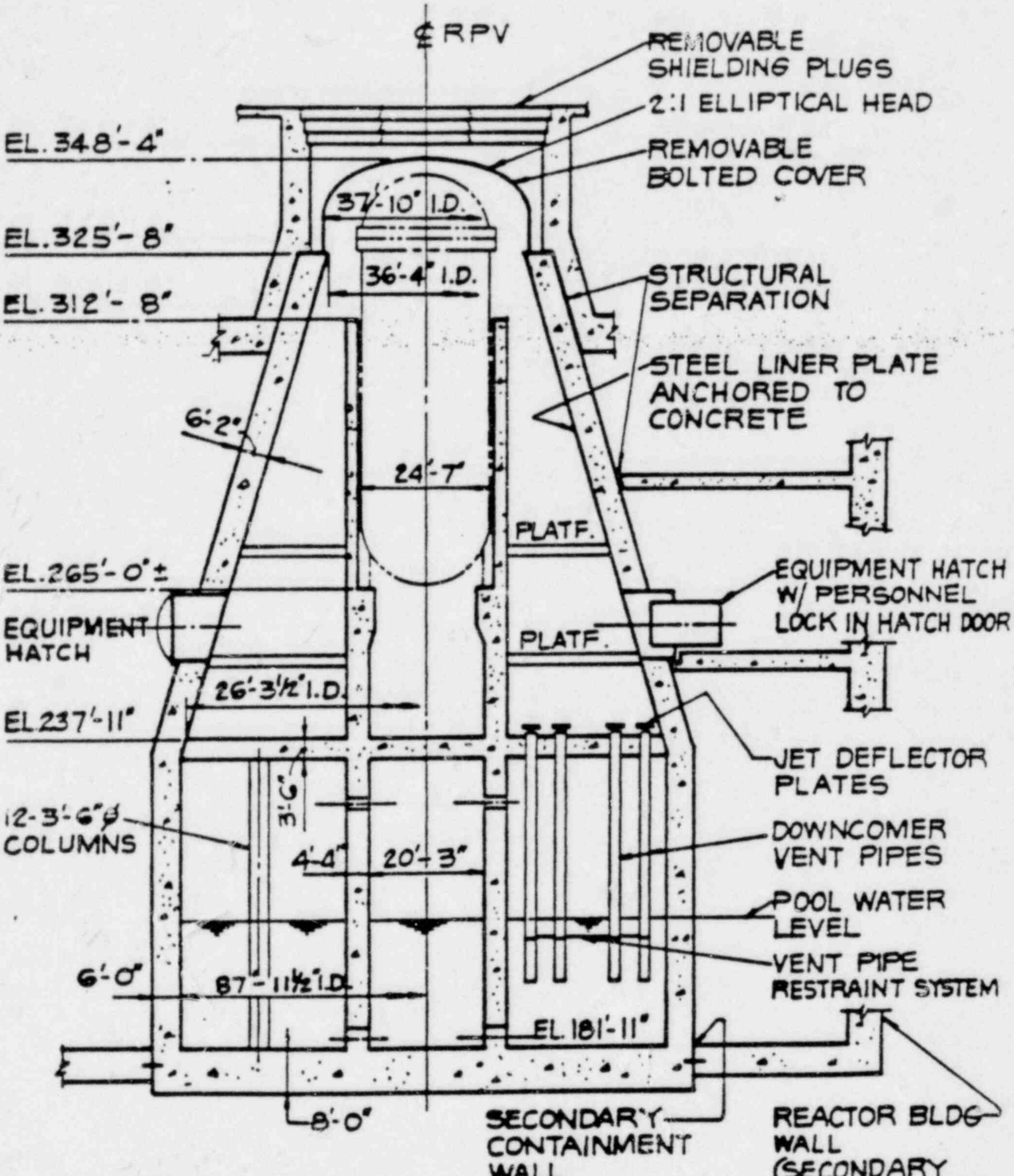
STANDBY LIQUID CONTROL

- NUMBER OF PUMPS - 3
- CAPACITY - 43 GPM/PUMP
- INITIATION SIGNAL
  - REACTOR HIGH PRESSURE OR REACTOR LOW WATER LEVEL (L2)
- PERMISSIVES
  - APRM NOT DOWN SCALE
  - TWO MINUTE TIME DELAY
- TRIP SIGNAL - STORAGE TANK LOW LEVEL
- POWER SUPPLY - AC

LIMERICK PRIMARY CONTAINMENT
ULTIMATE PRESSURE CAPACITY INVESTIGATION

SUBTASKS

1. CONTAINMENT RESPONSE TO PRESSURE BEYOND DESIGN PRESSURE
   - A. SEMI-INFINITE CYLINDER CALCULATIONS
   - B. FINITE ELEMENT ANALYSIS
2. REFUELING HEAD AND HATCH PRESSURIZATION RESPONSE
   - A. REFUELING HEAD
   - B. HATCHES
3. PRIMARY BOUNDARY VALVES PRESSURE RESPONSE
   - A. BUTTERFLY VALVES
   - B. GATE, GLOBE & CHECK VALVES
4. WRITE REPORT



VERTICAL SECTION

CONTAINMENT GENERAL ARRANGEMENT

REVISIONS			
LTR	DESCRIPTION	DATE	APPROVED

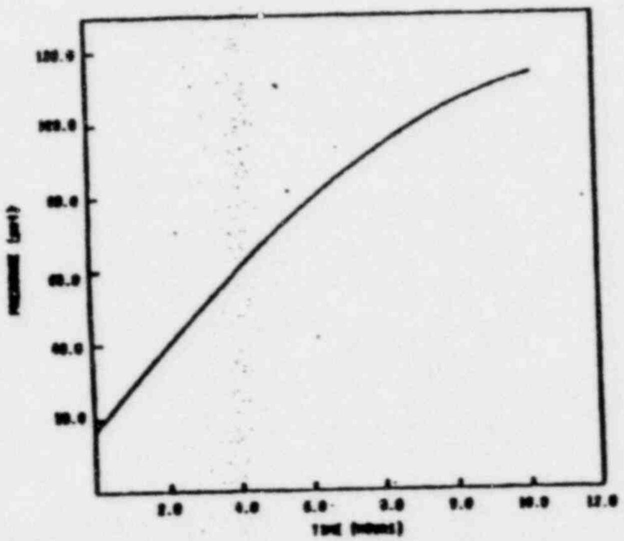


Figure 1.A Containment Overpressure (TV), Pressure versus Time.

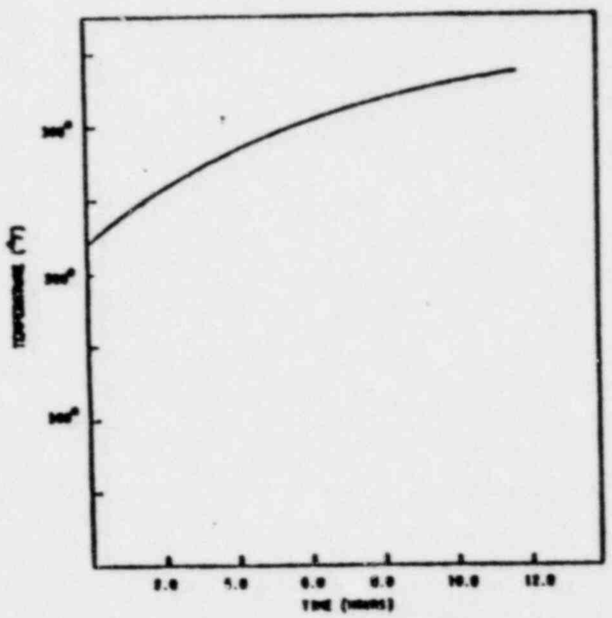


Figure 1.B Containment Overpressure (TV), Temperature versus Time.

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SCIENCE APPLICATIONS, INC.

SCALE		SHEET	

PRESSURE AND TEMPERATURE TIME HISTORY

REVISIONS

LTR	DESCRIPTION	DATE	APPROVED

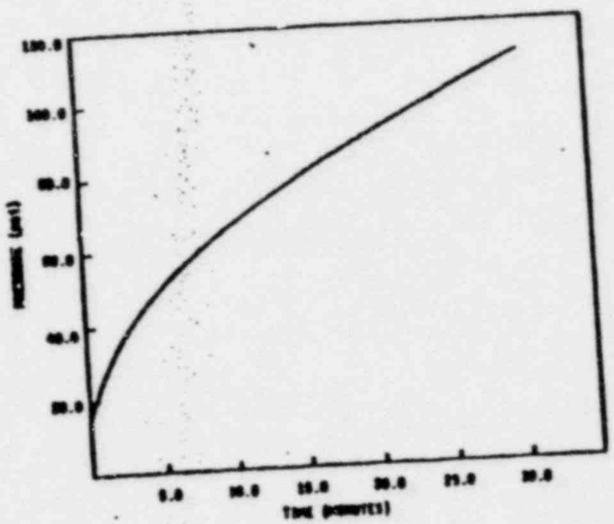

Figure [illegible] [illegible], Pressure versus Time.

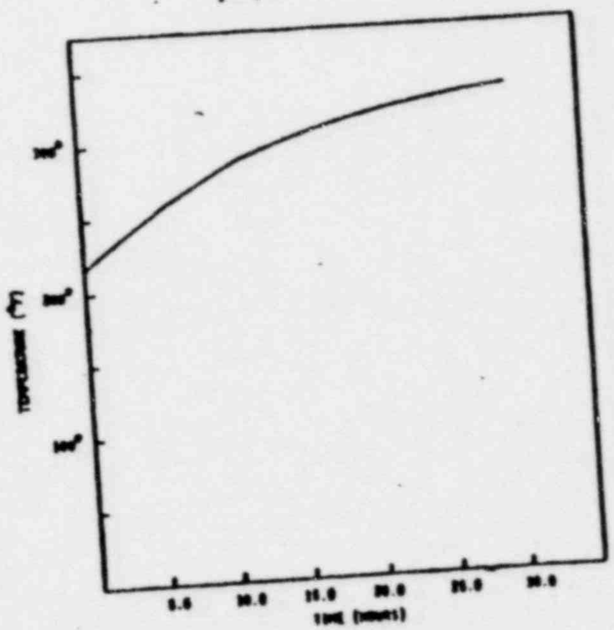

Figure [illegible] [illegible], Temperature versus Time.

SCIENCE APPLICATIONS, INC.

SCALE

SHEET

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PRESSURE AND TEMPERATURE TIME HISTORY

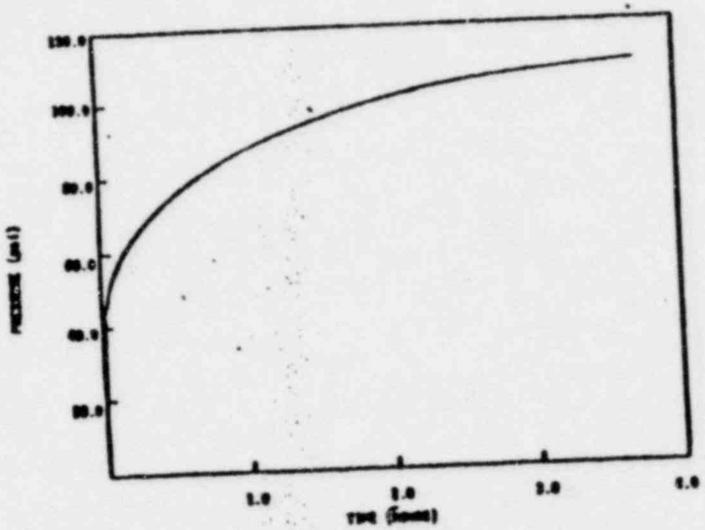

Figure 3.8 Large LOCA (Ar), Pressure versus Tim.

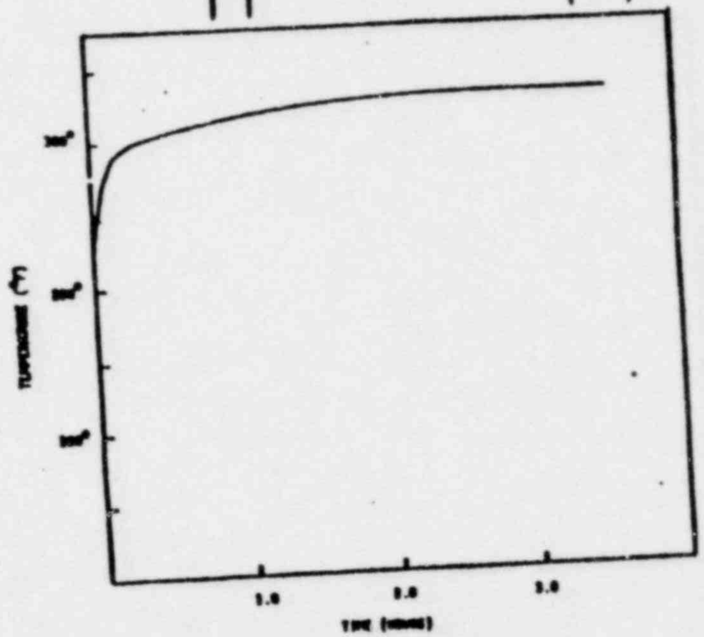

Figure 3.8 Large LOCA (Ar), Temperature versus Time

REVISIONS

SCIENCE APPLICATIONS, INC.

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PRESSURE AND TEMPERATURE TIME HISTORY

REVISIONS

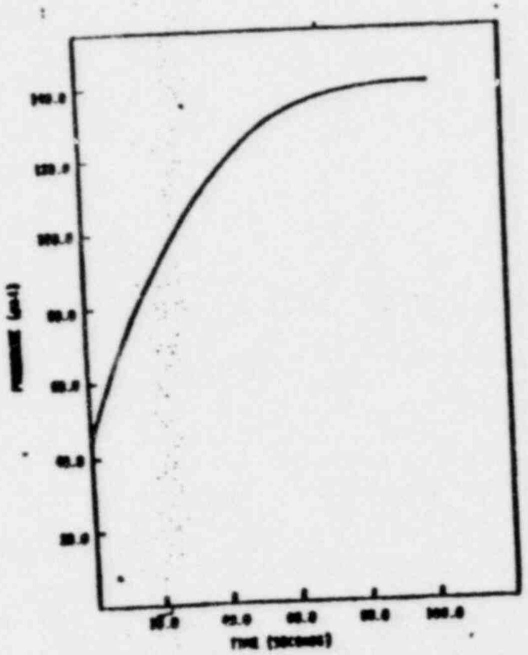

Figure [illegible] Rapid Steam Generation, Pressure versus Time.

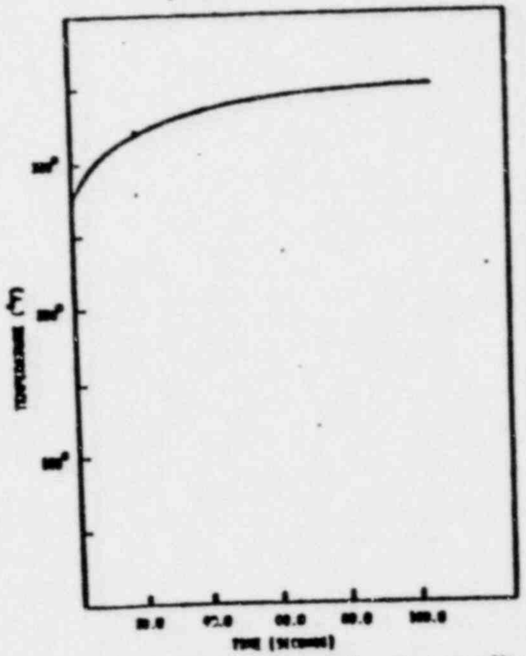

Figure [illegible] Rapid Steam Generation, Temperature versus Time.

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PRESSURE AND TEMPERATURE TIME HISTORY

TABLE I

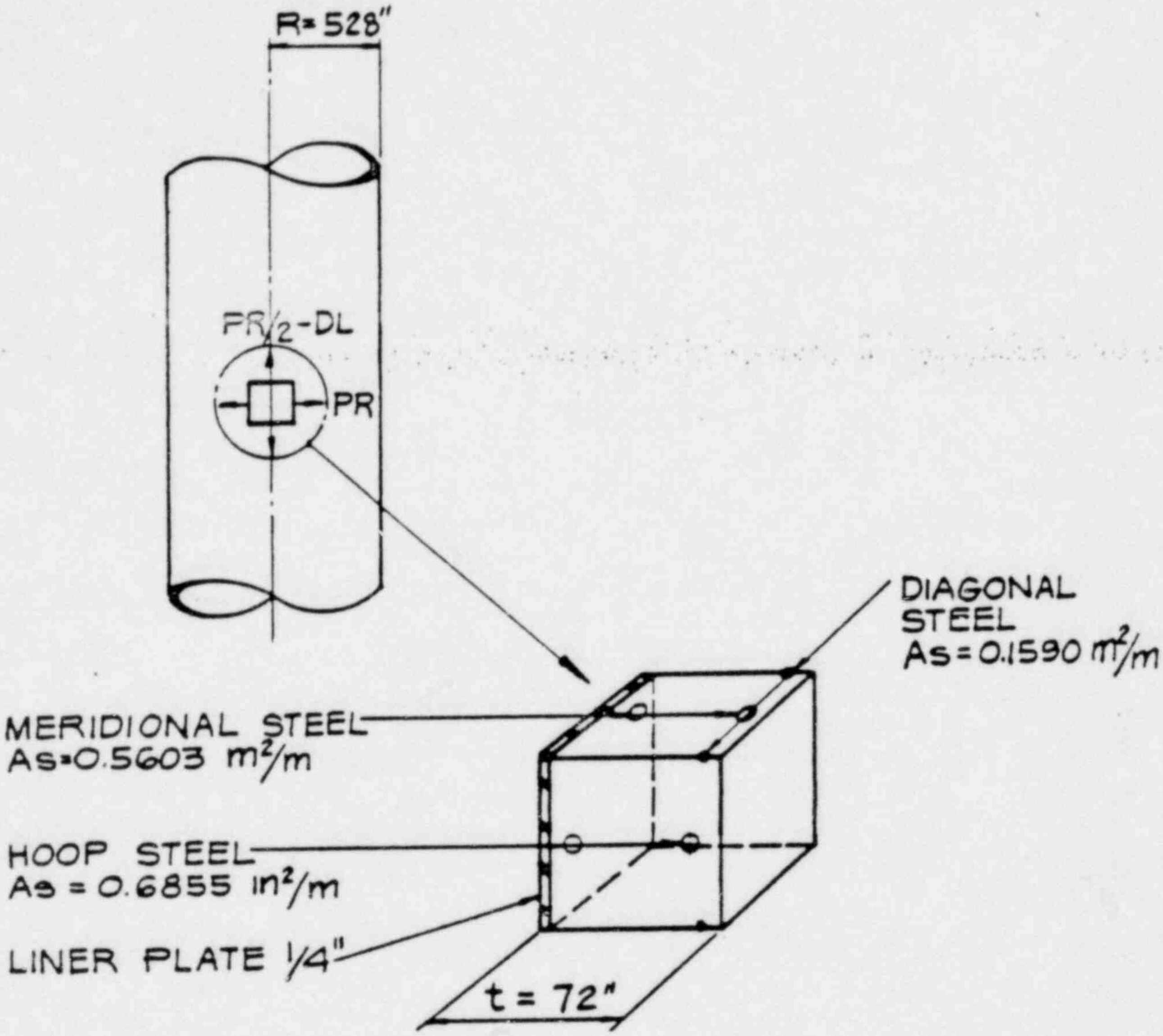
MATERIAL PROPERTIES

MATERIAL	COMPRESSIVE STRESS	YIELD STRESS	ULTIMATE STRESS	MODULUS OF ELASTICITY	POISSON RATIO	REMARK
CONCRETE	4000.0 psi	—	—	3605.0 KSI	0.17	—
REINFORCEMENT	—	68000.0 psi	100000.0 psi	29000.0 KSI	0.30	ASTM A618 GRADE
LINER PLATE	—	24000.0 psi	—	29000.0 KSI	0.30	ASTM A-285 GRADE A FIREBOX QUALITY

CONTAINMENT ANALYSIS FOR EXTREME PRESSURE

SEMI-INFINITE CYLINDER ANALYSIS

1. NEGLECT RESTRAINT OF BASE SLAB AND DIAPHRAGM SLAB
2. USE "CECAP" COMPUTER PROGRAM - LINEAR ELASTIC ANALYSIS
3. MEMBRANE LOAD CALCULATED BY $\frac{PR}{T}$ & $\frac{PR}{2T}$
4. MATERIAL PROPERTIES FROM AS-BUILT RECORDS AT MID HEIGHT OF WETWELL
5. INCLUDE REBAR AND LINER PLATE
6. FAILURE CRITERIA - YIELD STRESS OF ALL STEEL COMPONENTS



SEMI INFINITE ANALYTICAL MODEL

TABLE 2

(A) RESULT - SEMI-INFINITE CYLINDER ANALYSIS

PRESSURE	STRESS KSI			STRAIN INCH / INCH $\times 10^{-6}$		
	HOOP	MER.	DIAG.	HOOP	MER.	DIA
120 psi	68.00	11.60	69.63	4173	401	2287

(B) RESULT — FINITE ELEMENT ANALYSIS

PRESSURE	LOCATION	STRESS KSI f_y = 68 KSI			STRAIN INCH/INCH $\times 10^{-6}$ $\epsilon_y = 2345 \times 10^{-6}$ in/in		
		HOOP	MER.	DIAG/SHEAR	HOOP	MER	DIAG/SHEAR
120 psi	BASE SLAB & WALL	24.5	36.67	53.13	879	1264	1832
	MID-HEIGHT	45.64	38.64	—	1574	1332	—
	DIAPHRAGM SLAB & WALL	47.30	41.10	22.55	1631	1591	820
150 psi	BASE SLAB & WALL	36.755	52.08	64.55	1398	1796	2226
	MID-HEIGHT	67.24	58.33	—	2319	2011	—
	DIAPHRAGM SLAB & WALL	55.00	53.09	29.61	1902	1831	1020

CONTAINMENT ANALYSIS FOR EXTREME PRESSURE

FINITE ELEMENT CONTAINMENT ANALYSIS

1. IDEALIZE ENTIRE CONTAINMENT

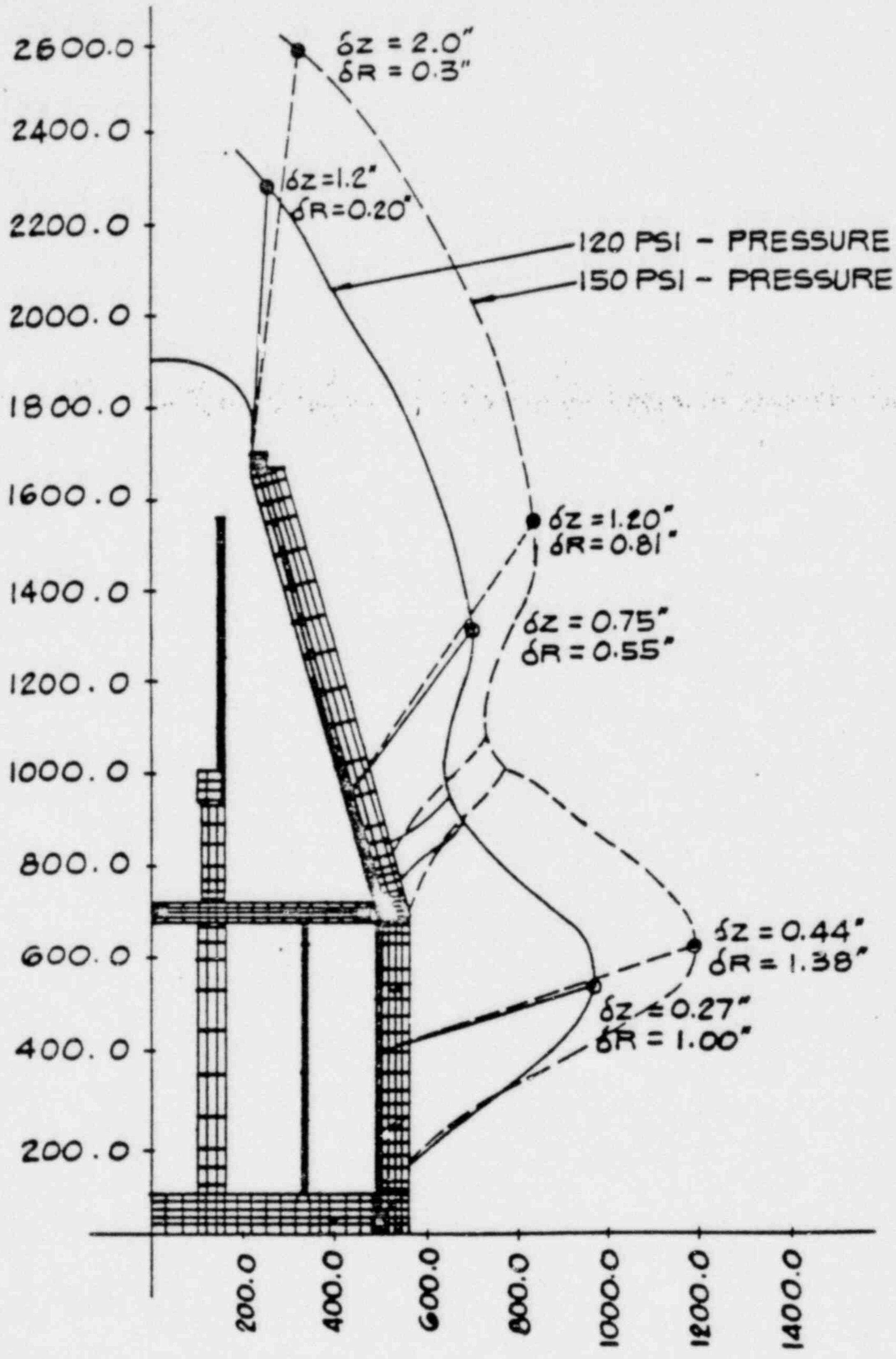
2. USE "FINEL" COMPUTER PROGRAM WITH IDENTICAL MODEL AS USED IN INITIAL DESIGN

3. APPLY UNIFORM INTERNAL PRESSURE IN WETWELL AND DRYWELL

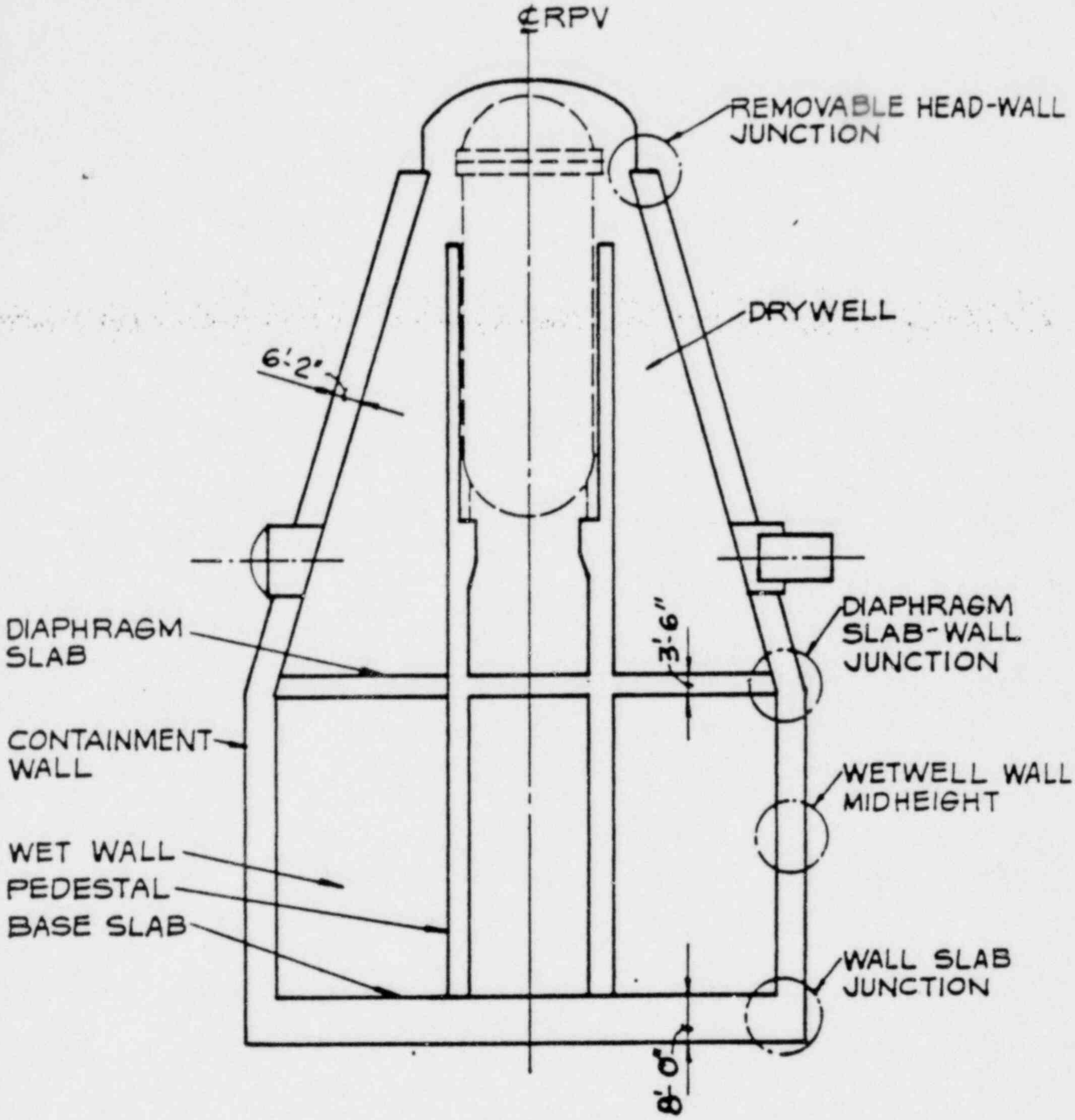
4. USE AS-BUILT MATERIAL PROPERTIES

5. "FINEL" COMPUTER PROGRAM IS CAPABLE OF NON-LINEAR ANALYSIS - CONCRETE CRACKING AND STEEL YIELDING ARE INCLUDED IN CODE

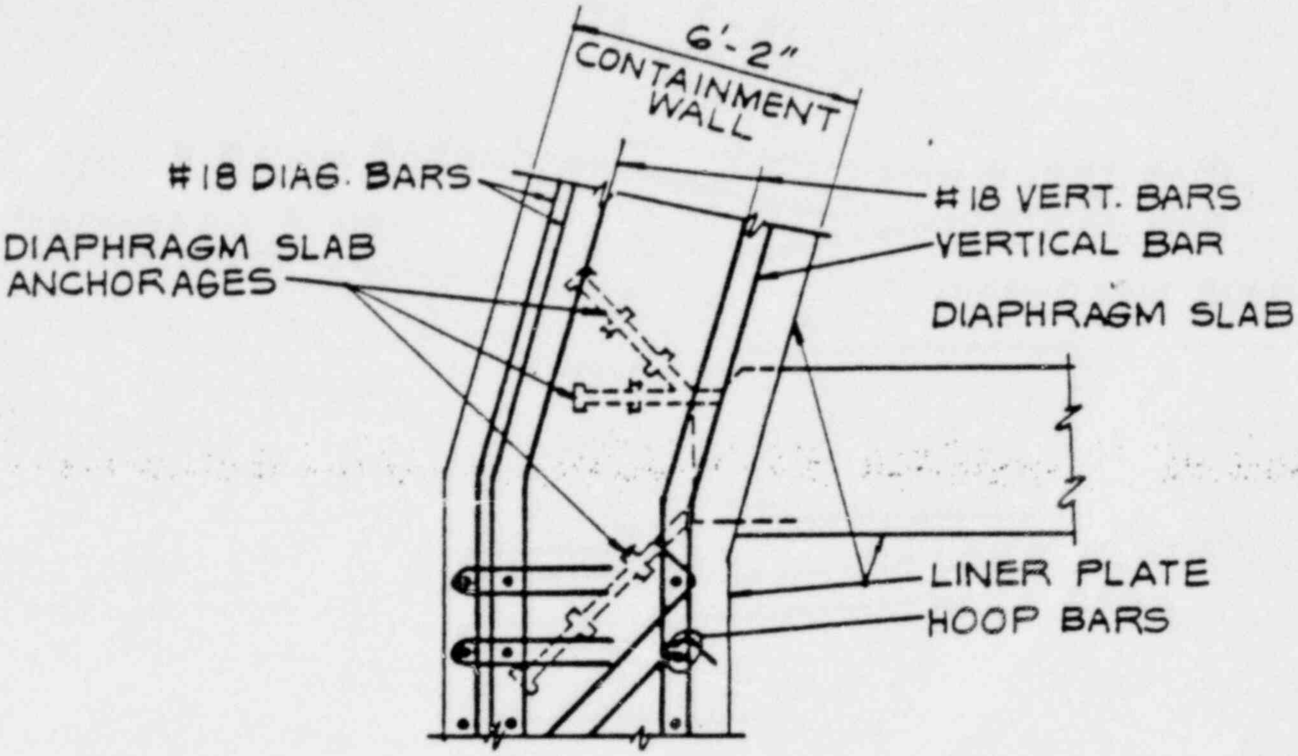
6. FAILURE CRITERIA - YIELD STRESS OF ALL STEEL COMPONENTS AT A SECTION



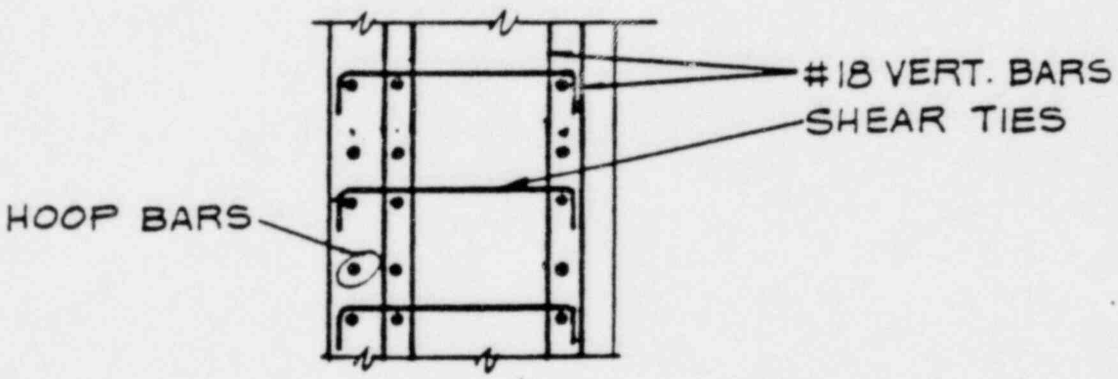
FINITE ELEMENT MODEL WITH
DISPLACEMENTS AT 120 & 150 PSIG



LOCATION OF CRITICAL SECTION

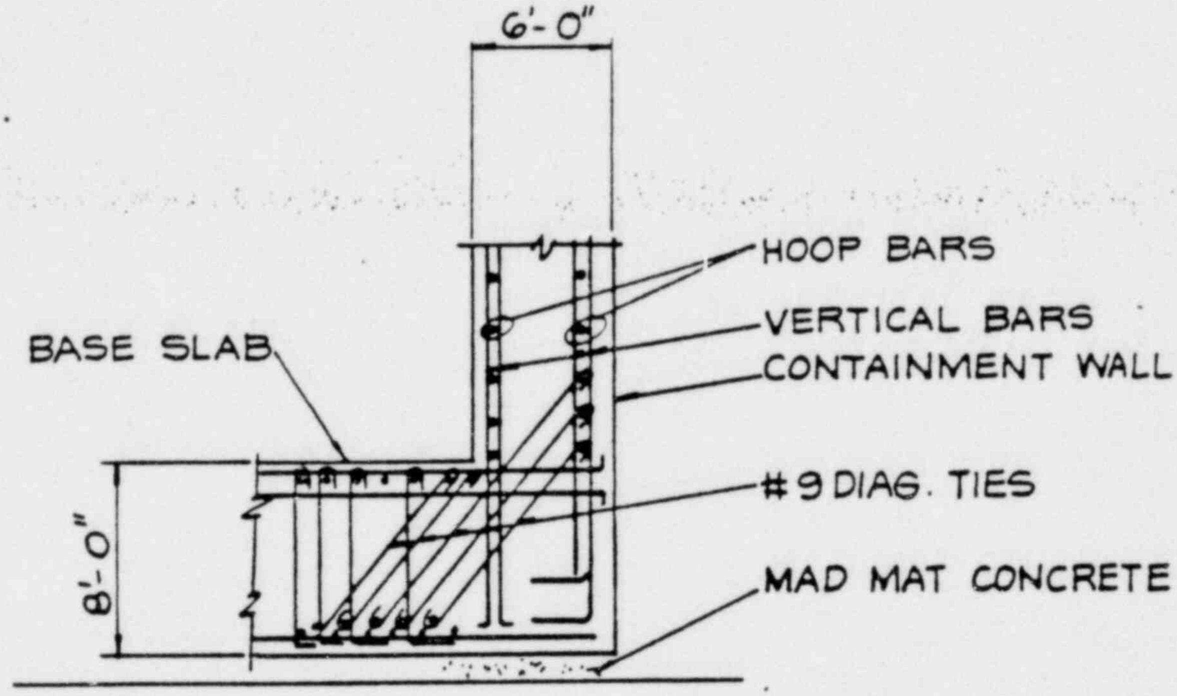


DIAPHRAGM SLAB-WALL JUNCTION



WALL-MIDHEIGHT

DETAIL OF CRITICAL SECTION



WALL SLAB JUNCTION

DETAIL OF CRITICAL SECTION

PRIMARY BOUNDARY VALVE PRESSURE CAPACITY

INVESTIGATION

1. IDENTIFY POTENTIAL VENT PATHS

2. IDENTIFY VALVE TYPES IN VENT PATHS

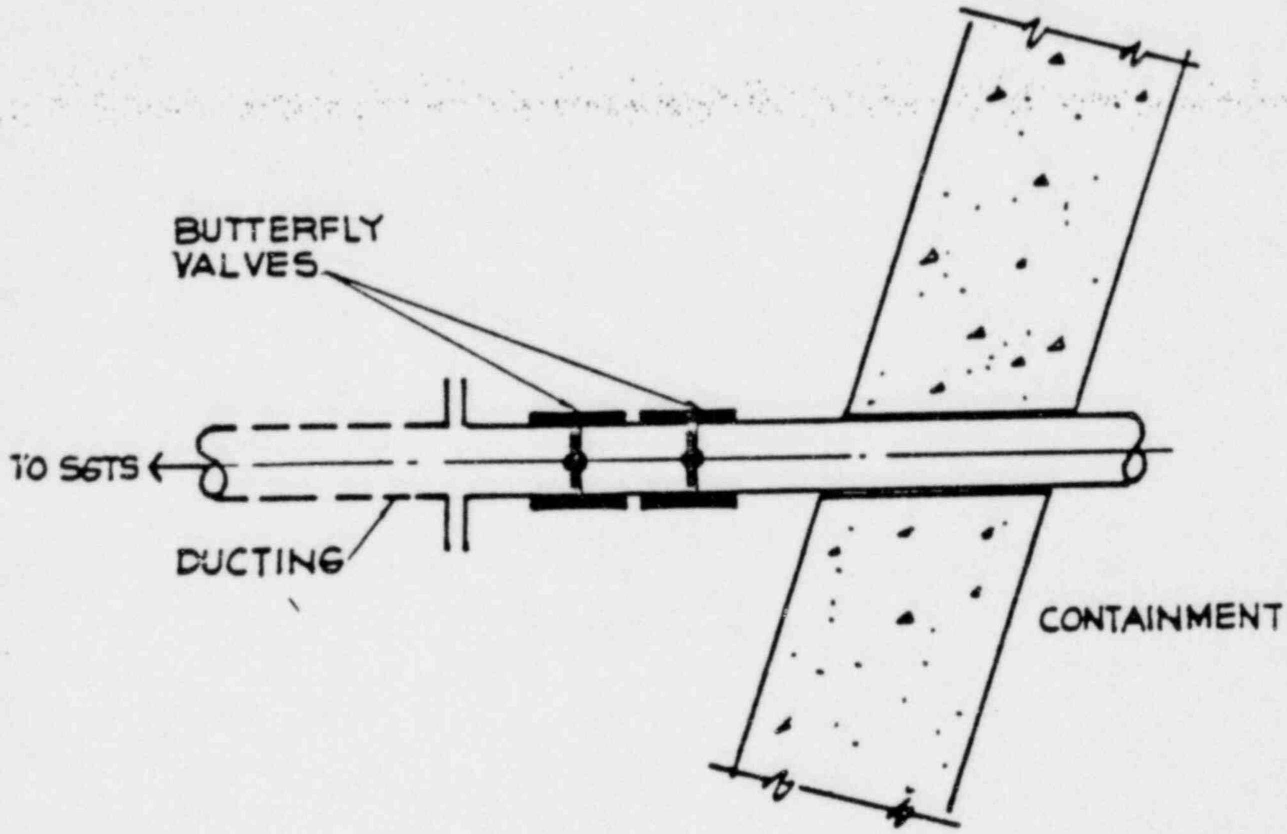
   A. BUTTERFLY VALVES
   B. GATE VALVES
   C. GLOBE VALVES
   D. CHECK VALVES

3. DETERMINE CAPACITY OF VALVE TYPES

CONCLUSIONS

1. NO VALVE FAILURES BELOW 300 PSIG

2. LEAKAGE NOT PREDICTED BELOW PRESSURE OF 140 PSIG.



TYPICAL POTENTIAL VALVE LEAKAGE PATH

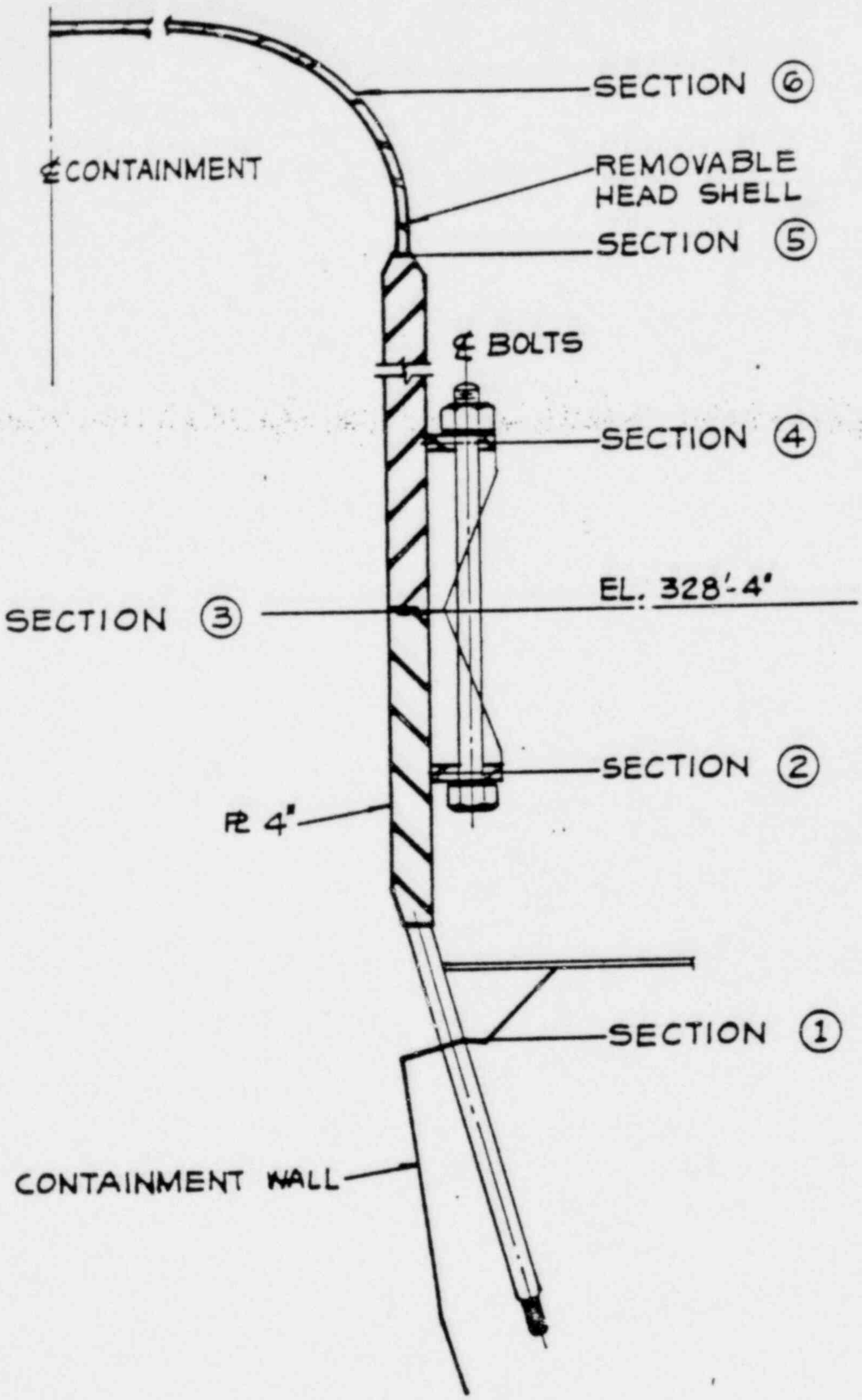
REFUELING HEAD AND HATCHES PRESSURE CAPACITY
(CHICAGO BRIDGE & IRON)

REFUELING HEAD & HATCHES

1. LINEAR ELASTIC SHELL ANALYSIS
2. PRESSURE AND TEMPERATURE LOADS INCLUDED
3. BOUNDARY CONDITIONS FROM BECHTEL "FINEL" ANALYSIS
4. ANALYSIS PRESSURES AS GIVEN BY BECHTEL "FINEL" RESULTS

CONCLUSIONS

1. GENERAL MEMBRANE STRESSES BELOW YEILD AT 120 PSIG
2. NO FAILURES PREDICTED



REMOVABLE HEAD WALL JUNCTION

TABLE 3

PRESSURE	SECTION	MAXIMUM STRESS INTENSITY KSI TEMPERATURE 340°F	TEMPERATURE 70°F
120 psi	1	30.0	23.8
	2	5.3	7.9
	3	7.5	9.3
	4	6.3	4.7
	5	9.2	9.1
	6	18.0	18.0
160 psi	1	14.0	41.70
	2	3.7	11.2
	3	0.8	16.0
	4	6.3	6.9
	5	9.1	12.0
	6	24.0	24.0

STRESS AT CRITICAL SECTIONS SHOWN ON FIG. 9

REFERENCE: CB&I ENCLOSURE

CONCLUSIONS

ULTIMATE PRESSURE FOR STRUCTURAL INTEGRITY

140 PSIG

FAILURE

VERTICAL CRACK AT WETWELL MIDHEIGHT

DISCUSSION OF RESULTS

- Site Differences
- Plant Design Differences
- Data Differences
- Methodology Differences

8547 45

ACCIDENT SEQUENCE DEFINITION

- <u>Bases for Analyses</u>
  - Scope/Ground Rules (Section 1.5)
  - Definition of Terms
  - Success Criteria

- <u>Event Tree Development</u>
  - Sources of Radioactivity
  - Accident Initiators
  - Classes of Accident Sequences

- <u>Results Summary</u>
  - Core Melt Frequency

SCOPE

LIMERICK GENERATING STATION PRA

- PERFORM A WASH-1400 TYPE ANALYSIS TO CALCULATE THE RISK TO THE PUBLIC ASSOCIATED WITH THE OPERATION OF THE LIMERICK GENERATING STATION.

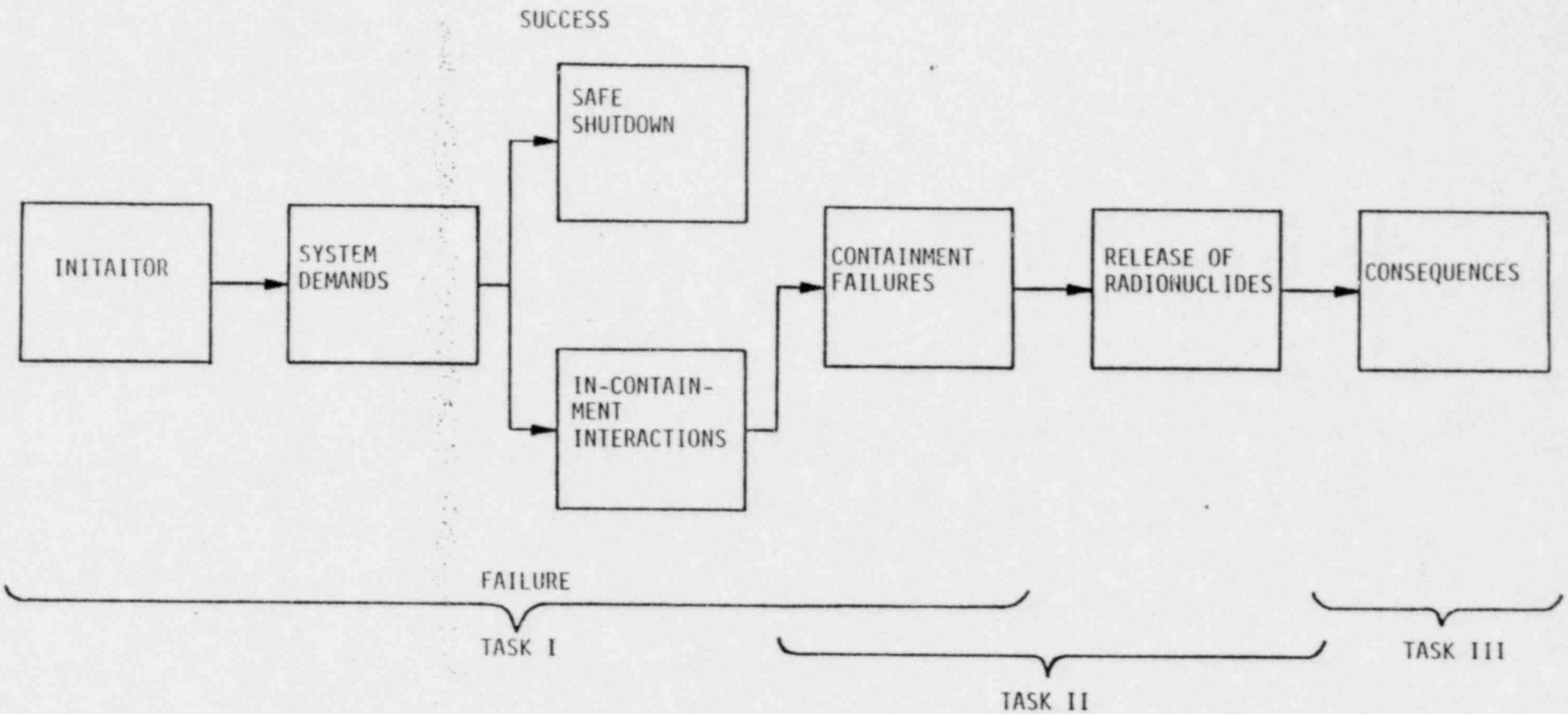
- ASSESS THE MAJOR CONTRIBUTORS TO POSTULATED CORE MELT IDENTIFIED IN WASH-1400 PLUS OTHER RELATED SEQUENCES.

- DO NOT EVALUATE POSSIBLE CONTRIBUTORS WHICH WERE DISMISSED IN WASH-1400 AS LOW PROBABILITY EVENTS, I.E., EXTERNAL EFFECTS, SABOTAGE, FIRES, NON-CORE RELATED EVENTS.

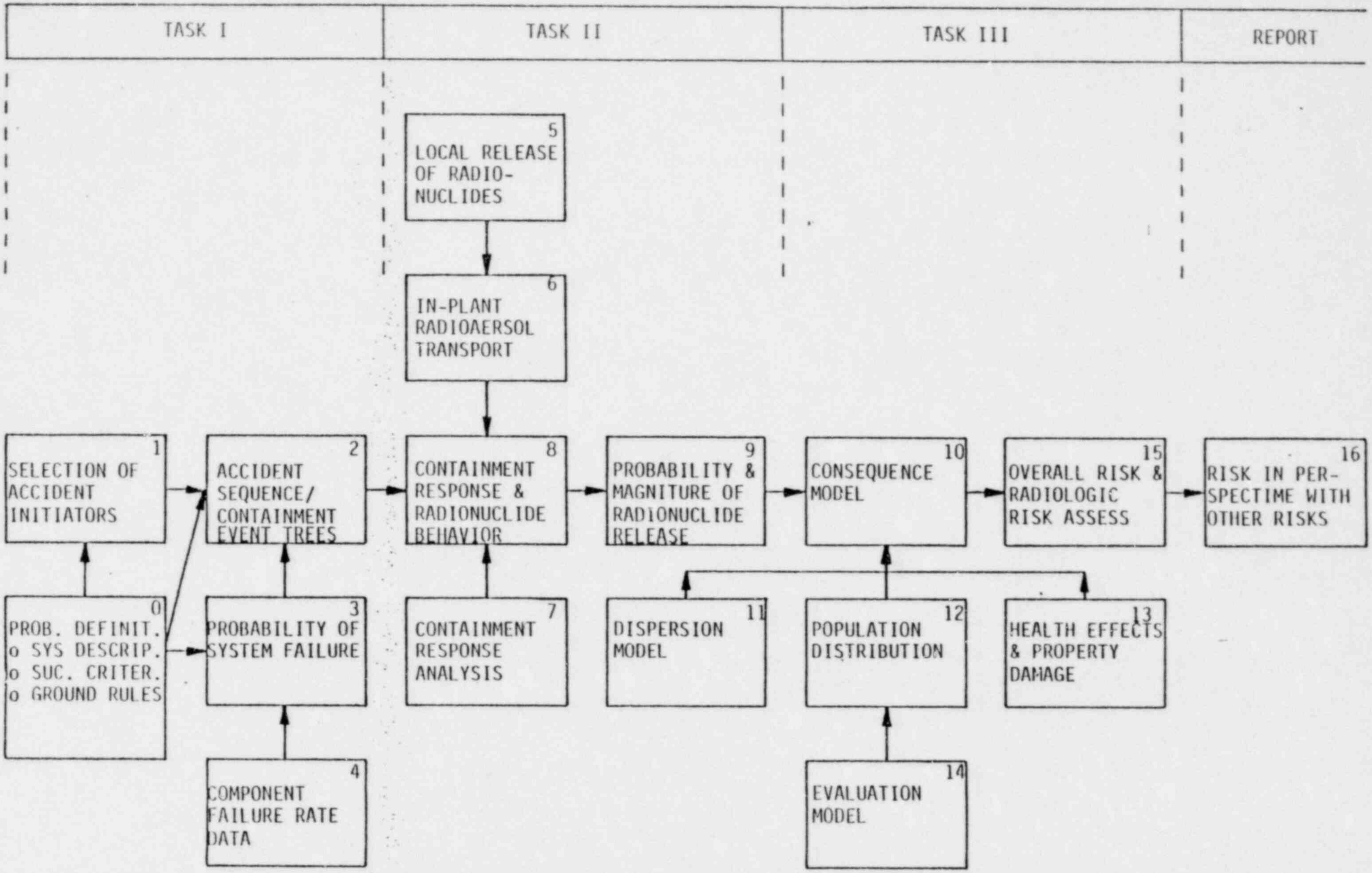
- USE PLANT SPECIFIC SYSTEM CONFIGURATIONS AND DATA WHERE AVAILABLE.

- USE SITE SPECIFIC POPULATION AND METEOROLOGY FOR CONSEQUENCE CALCULATIONS.

- PERFORM WITHIN 120 DAYS



Flow of Information for Final Quantitative Evaluation



Simplified Task Diagram for Probabilistic Risk Analysis. (This diagram May Also be Used as a Map to Direct The Reviewer to The Appropriate PRA Report Section for Detailed Discussion.)

OVERVIEW OF ACCIDENT SEQUENCE PROBABILITIES

- ATTEMPT TO MAINTAIN THE SAME CRITERIA FOR SUCCESSFUL SYSTEM OPERATION AS WE PERCEIVE WAS USED IN WASH-1400
    - NO HEROIC ACTIONS
    - LITTLE CREDIT GIVEN FOR OPERATOR ACTION WITHIN 30 MINUTES

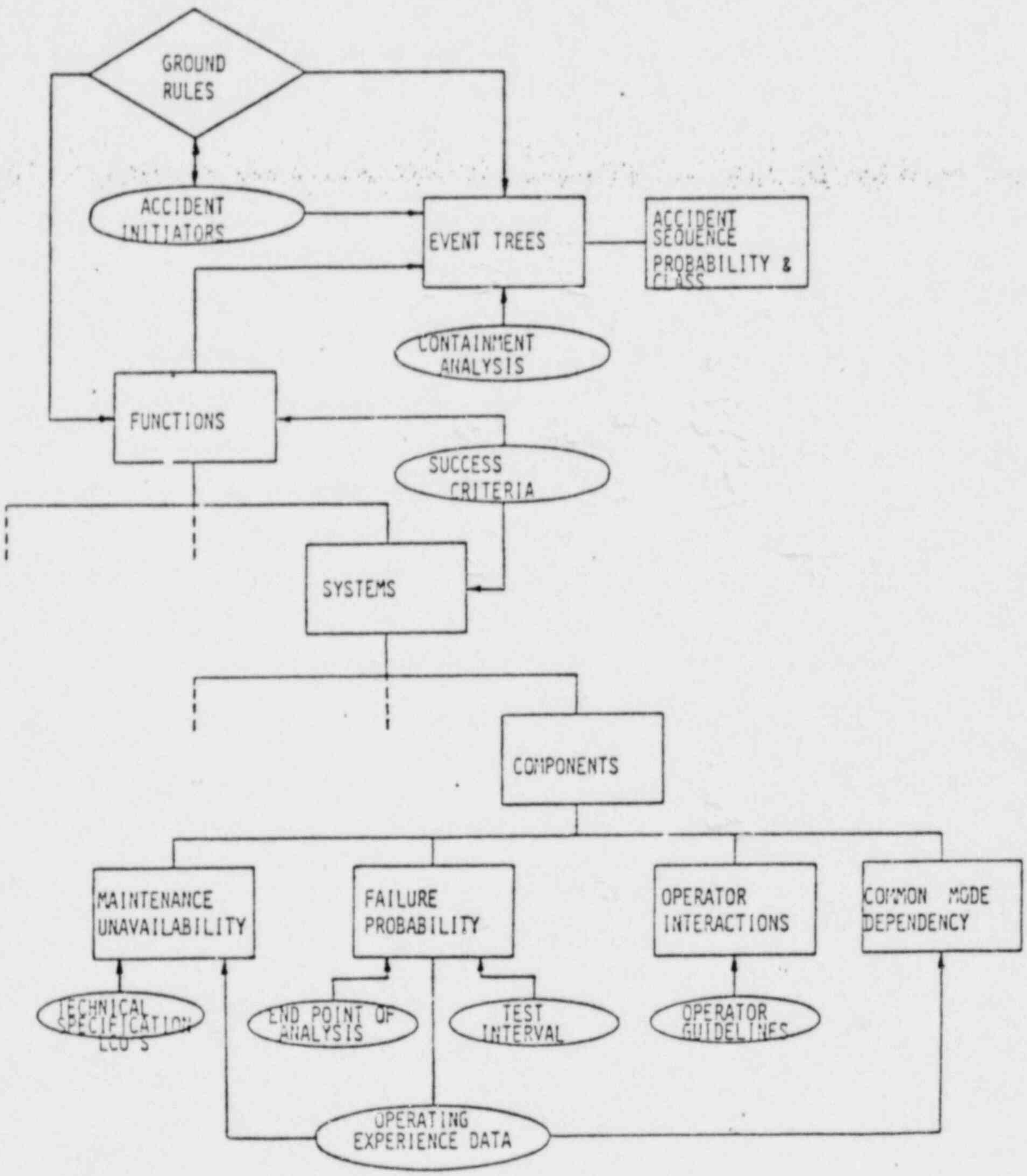
- CHANGES ONLY IN THE FOLLOWING
    - DIFFERENTIATION IN THE TYPES OF ACCIDENT INITIATORS
    - SYSTEM SUCCESS CRITERIA
    - NEW FEATURES (I.E. ATWS ALTERNATIVE 3A)
    - NEW DATA (COMPONENTS, MAINTENANCE, DIESELS, OFFSITE POWER)

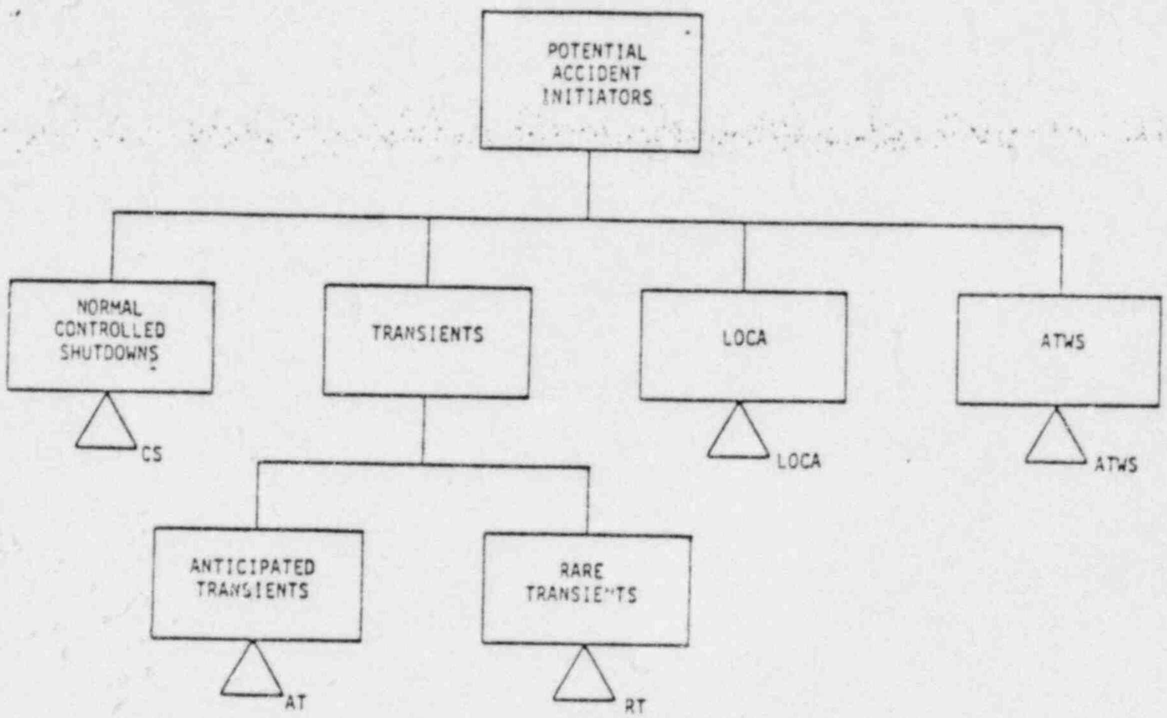
- USE SAME TECHNIQUE AS WASH-1400:
    - EVENT/FAULT TREE METHODOLOGY

GROUNDRULES: INITIAL CONDITIONS AND ASSUMPTIONS

- ACCIDENT INITIATORS TO BE CONSIDERED: TRANSIENTS; LOCAS; ATWS
- END POINT OF THE ANALYSIS: 20 HOURS -- HOT SHUTDOWN
- METHODOLOGY TO BE USED: SAME AS WASH-1400; ADDRESS CRITICISMS POINTED OUT BY LEWIS COMMITTEE
- PLANT CONFIGURATION: LIMERICK
- SYSTEM INCLUDED: ALL SYSTEMS
- OPERATOR INTERACTION: NO OPERATOR INTERVENTION TO PREVENT PROPER SYSTEM OPERATION
- SYSTEMATIC FAILURE CAUSES
- COMPONENT FAILURE RATE DATA: LATEST AVAILABLE
- MAINTENANCE AND TEST DATA: GENERAL ELECTRIC
- SUCCESS CRITERIA: GENERAL ELECTRIC
- UNCERTAINTIES

FLOW OF INFORMATION





Simplified Hierarchy of Possible Initiating Events
for a Typical PRA

SYSTEMS TABULATED AS A FUNCTION OF ACCIDENT INITIATORS

ACCIDENT INITIATOR	SUCCESS CRITERIA*	
	Coolant Injection	Containment Heat Removal
Large LOCA: Steam Break $\geq .2ft^2$ Liquid Break $\geq .1ft^2$	1 of 4 Core Spray Pumps OR 1 Core Spray Subsystem (2 pumps)	Normal Heat Removal OR 1 RHR OR Containment Overpressure Relief
Small LOCA: Steam Break .016 to .08 ft^2 Liquid Break .004 to .1 ft^2	HPCI OR Large LOCA Items Above	Same
Small-Small LOCA	HPCI OR RCIC OR Feedwater OR ADS AND { 2 of 4 CSIS OR 1 of 4 LPCI OR Condensate }	Same
Transient**	Same	Normal Heat Removal OR 1 RHR OR Containment Overpressure Relief
IORV	Same	Normal Heat Removal OR 1 RHR OR Containment Overpressure Relief
Transient + SORV	Same	Same

*Success Criteria are the minimum number of systems required to maintain the fuel temperature below 2200°F.

**Includes all the observed transients from operating experience data except IORV

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

SOURCES OF RADIONUCLIDE RELEASE

ESTIMATED TO BE SMALL CONTRIBUTION TO RISK IN WASH-1400:

- NORMAL EMISSIONS DURING PLANT OPERATIONS
- UNUSUAL OCCURRENCES
- SPENT FUEL STORAGE POOL ACCIDENTS
- TRANSPORATION ACCIDENTS
- OFF GAS SYSTEM ACCIDENTS
- REFUELING ACCIDENT

- REACTOR CORE DURING/OR IMMEDIATELY FOLLOWING POWER OPERATION

DEFINITION OF ACCIDENT SEQUENCE CLASSES

- DECOUPLE THE PROBABILISTIC EVALUATION OF ACCIDENT SEQUENCES FROM THE DETERMINISTIC IN-CONTAINMENT CALCULATIONS OF RELEASE FRACTIONS.

- ELIMINATE SMOOTHING AMONG RELEASE CATEGORIES (SEE LEWIS COMMITTEE REPORT)

GENERIC ACCIDENT SEQUENCE TYPES

GENERIC ACCIDENT SEQUENCE DESIGNATOR	CORE CONDITION	CONTAINMENT CONDITION	EXAMPLE SEQUENCE
CLASS I	• CONTROL RODS INSERTED • POWER LEVEL AT 30 MIN. POST SCRAM LEVEL	• LOW PRESSURE AT INITIATION OF DEGRADED CORE CONDITION	T_FQUV
CLASS II	• CONTROL RODS INSERTED • POWER AT 20-30 HOUR POST SCRAM LEVEL	• CONTAINMENT FAILED AT INITIATION OF DEGRADED CORE CONDITION	TM
CLASS III	• ATWS • POWER AT APPROX. 30% LEVEL AT BEGINNING OF CORE DEGRADED CONDITION	• CONTAINMENT AT ELEVATED PRESSURE PRIOR TO CORE DEGRADED CONDITION	T_FC_MU $T_FC_MC_2$ (Mode 1/3)
CLASS IV	• ATWS • POWER AT APPROX. 30% LEVEL AT BEGINNING OF CORE DEGRADED CONDITION	• CONTAINMENT FAILED AT INITIATION OF DEGRADED CORE CONDITION	$T_FC_MC_2$ (Mode 1)

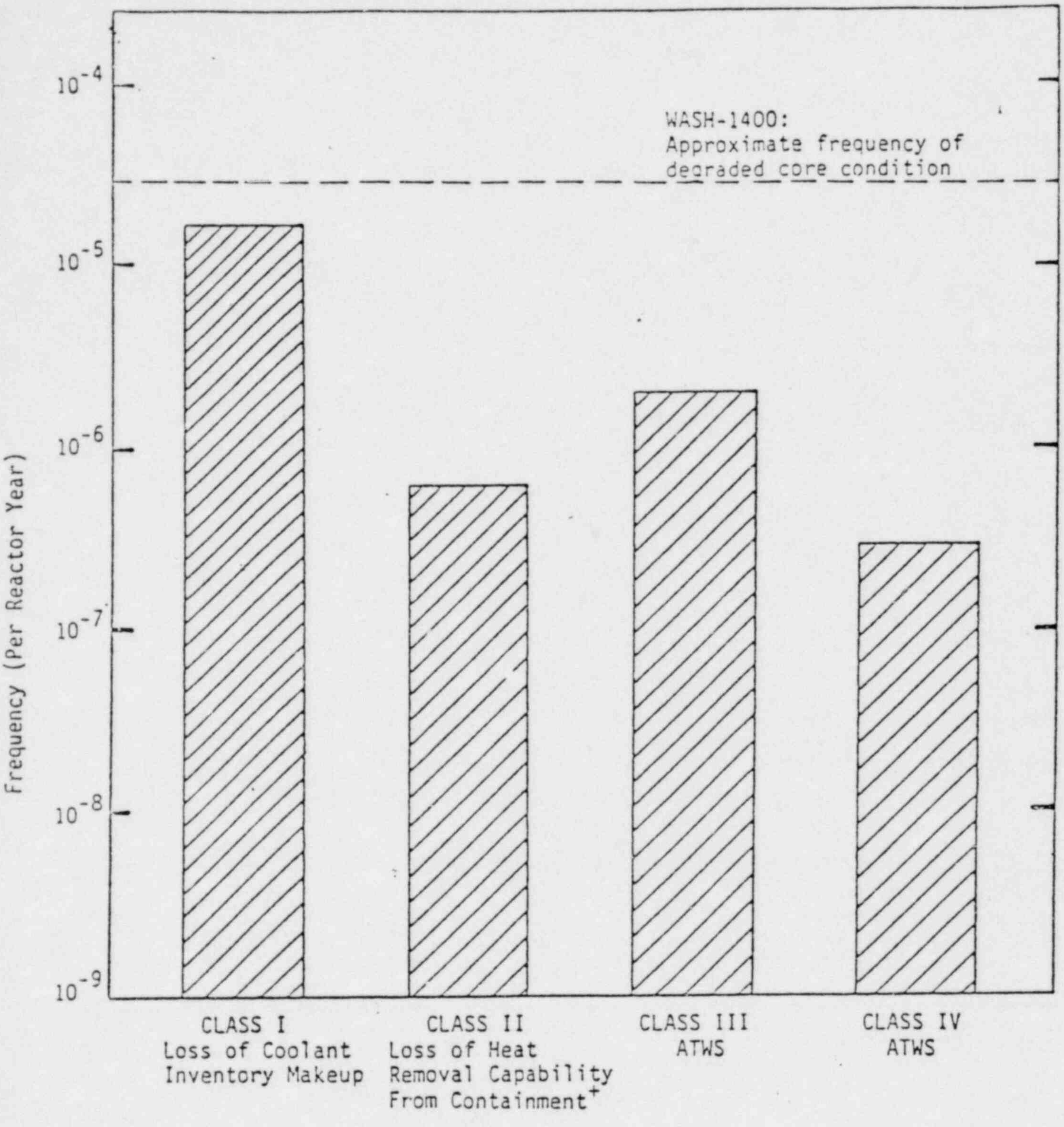
INCLUDED IN THE LIMERICK PRA ARE AREAS OF POTENTIALLY HIGHER RISK THAN ASSESSED IN WASH-1400

1. HIGHER PROBABILITY FOR ATWS

2. HGIHER RADIONUCLIDE RELEASE FRACTIONS FOR ATWS COMPARED WITH OTHER ACCIDENT SCENARIOS.

3. LOSS OF OFFSITE POWER MAY BE A SIGNIFICANT CONTRIBUTOR TO CORE MELT FREQUENCY

4. FAILURE TO PROVIDE COOLANT INJECTION MAY BE A SIGNIFICANT CONTRIBUTOR TO CORE MELT FREQUENCY



+Makes use of containment overpressure relief

QUANTITATIVE TREATMENT OF DEPENDENCY

- DETAILED FAULT TREES INCLUDE SYSTEM INTERFACES AND INTERDEPENDENCIES SUCH AS COMMON LOGIC, SENSORS, AND ELECTRIC POWER

- ENVIRONMENTAL DEGRADATION OF EQUIPMENT FOR IDENTIFIED CASES OF EXCEEDING DESIGN SPECIFICATIONS, FOR EXAMPLE:

  1. CONTAINMENT LEAKAGE INTO REACTOR BUILDING (E.G., TW) (CONSERVATIVE, SAME AS WASH-1400)

  2. FAILURE OF ROOM COOLING

  3. ADVERSE IN-CONTAINMENT ENVIRONMENT

- DIESEL DEPENDENCY MODEL BASED UPON OPERATING EXPERIENCE W/DIESELS

- MAINTENANCE DEPENDENCY MODEL

- COMMON-MODE OPERATOR ERROR INCLUDED FOR

  1. MISCALIBRATION OF INSTRUMENTATION

  2. MANUAL SYSTEM INITIATION

SYSTEM INTERFACES INCORPORATED INCLUDE:

- ELECTRIC POWER - AC & DC EMERGENCY BUSES
- ROOM COOLING (EMERGENCY SERVICE WATER)
- INITIATION LOGIC - COMMON SENSORS AND LOGIC (RX LEVEL OR HI DRYWELL) AMONG SYSTEMS
- COMMON SUCTION AND DISCHARGE LINES
- COMMON WATER SOURCES

BOOLEAN COMBINATION OF SYSTEMS REQUIRED TO ENSURE THAT COMMON INTERFACES ARE PROPERLY ACCOUNTED FOR IN QUANTIFICATION

TECHNICAL SPECIFICATIONS

- DEPENDENCIES OF SYSTEMS DURING SYSTEM MAINTENANCE OUTAGES WHILE THE PLANT IS OPERATING

- USE PEACH BOTTOM TECHNICAL SPECIFICATIONS

- EXAMPLE:
  RCIC IN MAINTENANCE
  HPCI MUST BE ON-LINE

LIMITING CONDITIONS OF OPERATION FOR THE PLANT USED IN THIS ANALYSIS

System/Component Out of Service	Systems Required to be Operable	Redundant Core Spray Subsystems	LPCI	Both Core Spray Subsystems	Remaining LPCI Subsystems	Remaining Containment Cooling Subsystem Components	Diesel for Operable Loop	RCIC	ADS	HPCI	Remaining Diesel Generators	Containment Cooling Subsystems	Remaining Relief Valves	Remaining Recirc. Loop	Allowable Out of Service Time (days)	Shutdown
One Core Spray Subsystem		X	X												7	
Both Core Spray Subsystems															3	X
One LPCI Pump			X	X											7	
One LPCI Subsystem			X	X											7	
Both LPCI Subsystems															3	X
Two HPSW Pumps						X									30	
Three HPSW Pumps						X									15	
Three Containment Cooling Subsystem Loops						X	X								7	
All Containment Cooling Subsystem Loops															1	X
HPCI			X	X				X	X						3	
RCIC										X					7	
One ADS Valve										X					7	
More Than One ADS Valve															1	X
One Diesel Generator			X	X							X	X			7	
More Than One Diesel															1	X
Safety Valve Function of One Relief Valve													X		30	
Safety Valve Function of Two Relief Valves													X		3	
Safety Valve Function of More Than Two Relief Valves															1	X
One Recirculation Loop														X	1	
More Than One Inoperable Control Rod in a 5x5 Array															1	X
Withdrawn Control Rod Which Can't be Moved Because of Possible Collet Housing Failure															3	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.

COMMON-MODE TREATMENT IN WASH-1400

1. DURING EVENT TREE CONSTRUCTION

   - INCORPORATION OF FUNCTIONAL DEPENDENCIES BETWEEN SYSTEMS IN THE ACCIDENT SEQUENCES.

   - DEVELOPMENT OF ACCIDENT SEQUENCES INCLUDING CONTAINMENT FAILURE MODE DEFINITIONS WHICH INCORPORATE SYSTEM AND ACCIDENT INTERDEPENDENCIES.

COMMON-MODE TREATMENT IN WASH-1400

2. DURING FAULT TREE CONSTRUCTION

- RESOLUTION OF FAILURES TO A LEVEL SUCH THAT COMMON SYSTEM HARDWARE WILL BE IDENTIFIED.

- FAULT TREE CONSTRUCTION WHICH IDENTIFY HUMAN INTERFACES, TEST AND MAINTENANCE INTERFACES, AND OTHER INTERFACES OF POTENTIAL DEPENDENCY.

COMMON-MODE TREATMENT IN WASH-1400

3. DURING FAULT TREE QUANTIFICATION

- PRACTICAL DATA UTILIZATION, WHICH INCORPORATES UNCERTAINTIES AND VARIATIONS.

- QUANTIFICATION FORMULAS WHICH INCORPORATE DEPENDENCIES AND CONTRIBUTIONS DUE TO HUMAN ERROR, TEST AND MAINTENANCE, AND ACCIDENT RELATED ENVIRONMENTS.

- MATHEMATICAL TECHNIQUES INVOLVING BOUNDING CALCULATIONS AND ERROR PROPAGATION CALCULATIONS, WHICH SERVE TO DETERMINE THE SIGNIFICANCE OF POSSIBLE DEPENDENCIES AND SERVE TO INCORPORATE RESULTING UNCERTAINTIES.

COMMON-MODE TREATMENT IN WASH-1400

4. DURING EVENT TREE QUANTIFICATION

- IDENTIFICATION OF COMPONENTS COMMON TO MORE THAN ONE SYSTEM BY BOOLEAN ALGEBRA TECHNIQUES.

- QUANTIFICATION FORMULAS WHICH INCORPORATE COUPLINGS AND DEPENDENCIES ACROSS SYSTEMS DUE TO HUMAN ERROR, T & M, AND ACCIDENT ENVIRONMENTS.

- GROUPING OF ACCIDENT SEQUENCES OF SIMILAR OUTCOME AND IDENTIFICATION OF THE DOMINANT ACCIDENT SEQUENCES USING DISCRIMINATION AND BOUNDING TECHNIQUES.

COMMON-MODE TREATMENT IN WASH-1400

5. SPECIAL ENGINEERING INVESTIGATIONS

- INVESTIGATION OF SPECIAL, SUSCEPTIBLE ACCIDENT SEQUENCES TO DETERMINE ANY REMAINING POSSIBLE COMMON MODES INCLUDING THOSE DUE TO EXTERNAL EVENTS AND COMMON COMPONENT SENSITIVITIES.

- A SPECIAL DESIGN ADEQUACY TASK TO INVESTIGATE COMMON MODE FAILURES RESULTING FROM EARTHQUAKES, OTHER EXTERNAL FORCES, AND POST ACCIDENT ENVIRONMENTS.

- FINAL CHECKS ON THE FAULT TREE AND EVENT TREE MODELS FOR MODEL ACCURACY AND CONSISTENCY.

LEVEL OF DETAIL IN FAULT TREES

- FAULT TREE INPUT LEVEL ITEMS
- GENERIC FAULT TREES
- EXAMPLE OF COMPONENT LEVEL INPUT

- RESOLUTION OF USEFUL OPERATING EXPERIENCE DATA APPEARS TO BE AT THE COMPONENT LEVEL

- COMPONENT LEVEL DATA USED FOR THE LIMERICK QUANTIFICATION INCLUDE:

  - MANUAL VALVES
  - MOTOR OPERATED VALVES
  - PUMPS
  - TURBINES
  - INSTRUMENTATION SENSORS/SWITCHES
  - DIESELS*
  - OFFSITE POWER (INITIATOR)*
  - MAINTENANCE*
  - OPERATOR ERROR (IN GENERAL)*

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* SPECIAL TOPICS

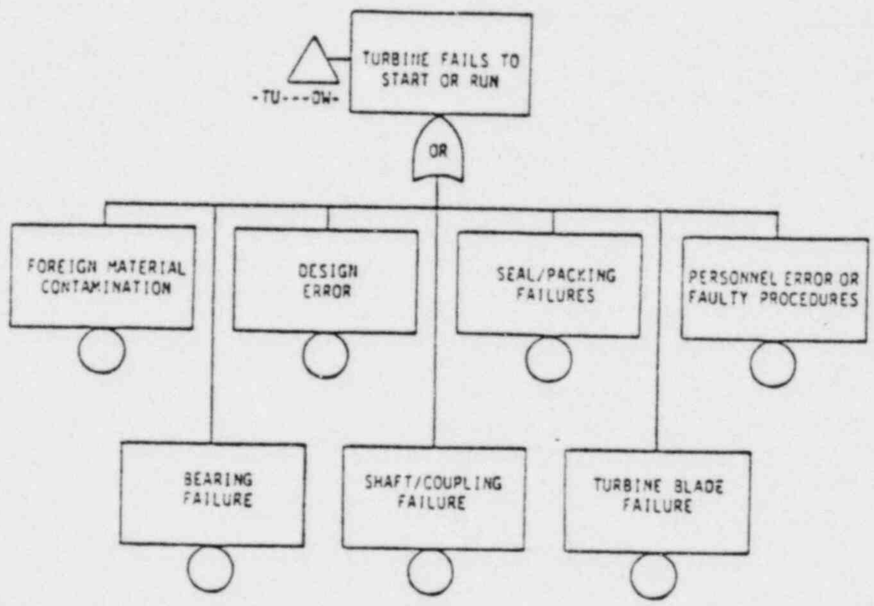
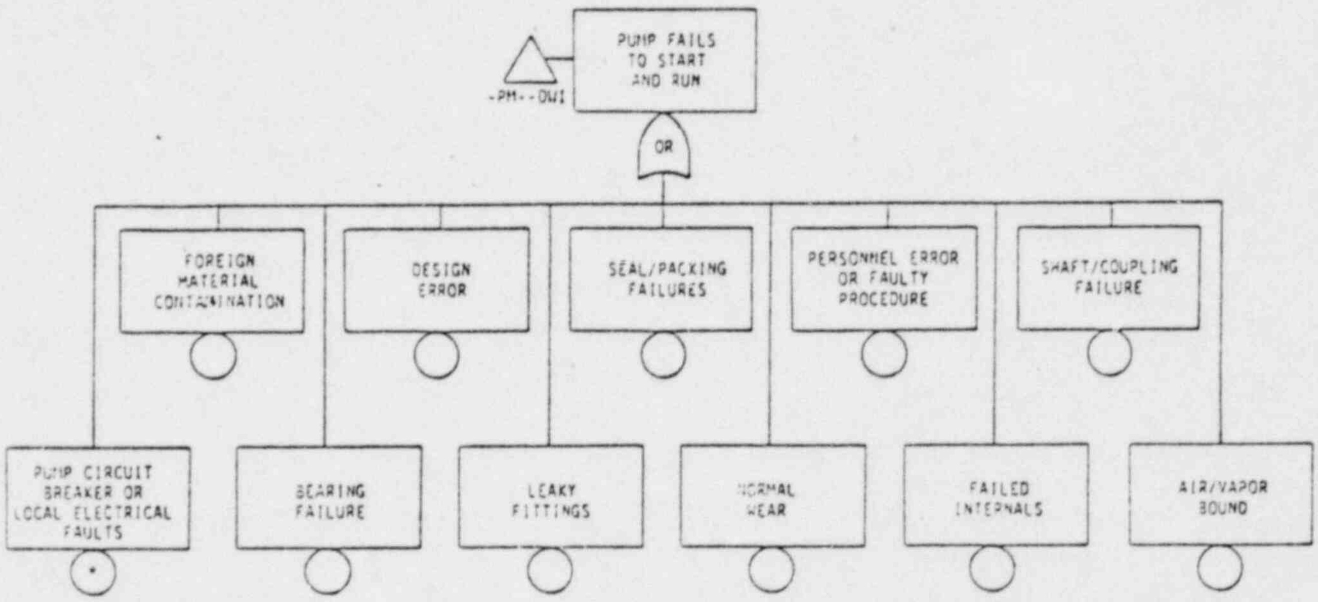


Figure 3.16. Generic Fault Tree Model Displaying the Potential Failure Modes of the Turbine (RCIC or HPCI)



*Estimated not based on data.

Figure 3.15. Generic Fault Tree for Pumps, Including the Dominant Failure Modes of Pump Failure Observed in Operation

Figure B.1.16

HPCI

Sheet 1 of 21

HPCI

SHEET 2 of 21

HPCI

SHEET 8 of 21

H.P.C.I.

SHEET 9 of 21

HPCI

SHEET 10 of 21

- DISCUSSION OF SELECTED ACCIDENT SEQUENCES
  - $T_T QUX$ CLASS I
  - $T_F QW(Q)$ CLASS II
  - $T_F C_M C_2$ CLASS IV
- QUANTIFICATION OVERVIEW
- RESULTS

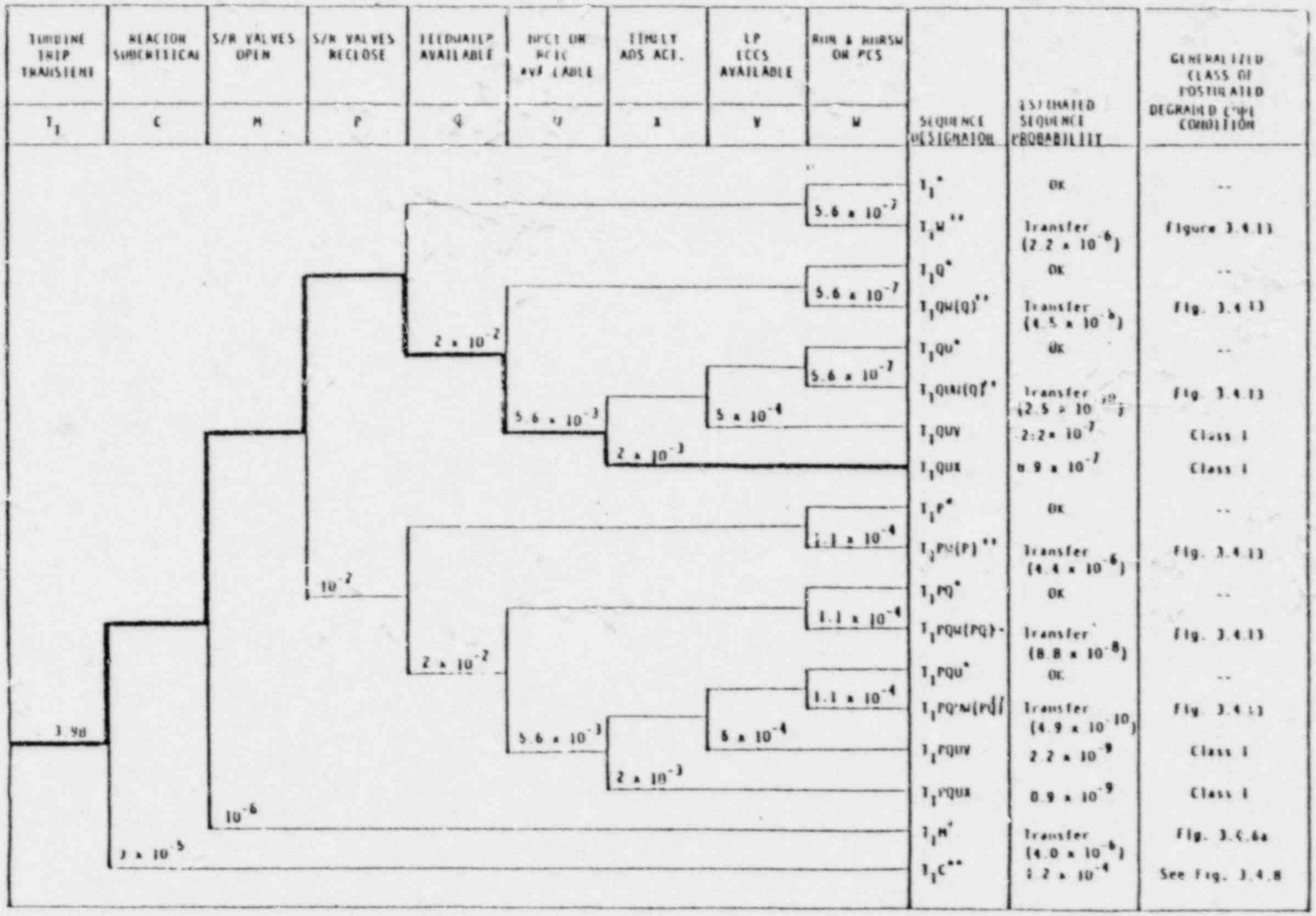
CLASS I

$T_T QUX$

TURBINE TRIP INITIATOR WITH SUBSEQUENT FAILURE TO PROVIDE ADEQUATE COOLANT MAKEUP CAPABILITY

A CONTRIBUTING SEQUENCE TO CLASS I

- INITIATOR: TURBINE TRIP (T_T)
- LOSS OF HIGH PRESSURE COOLANT MAKEUP
  - FEEDWATER UNAVAILABLE (Q)
  - HPCI AND RC' VAILABLE (U)
- INABILITY TO DEPRESSURIZE RAPIDLY
  - MANUAL DEPRESSURIZATION (X)



*Hot 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

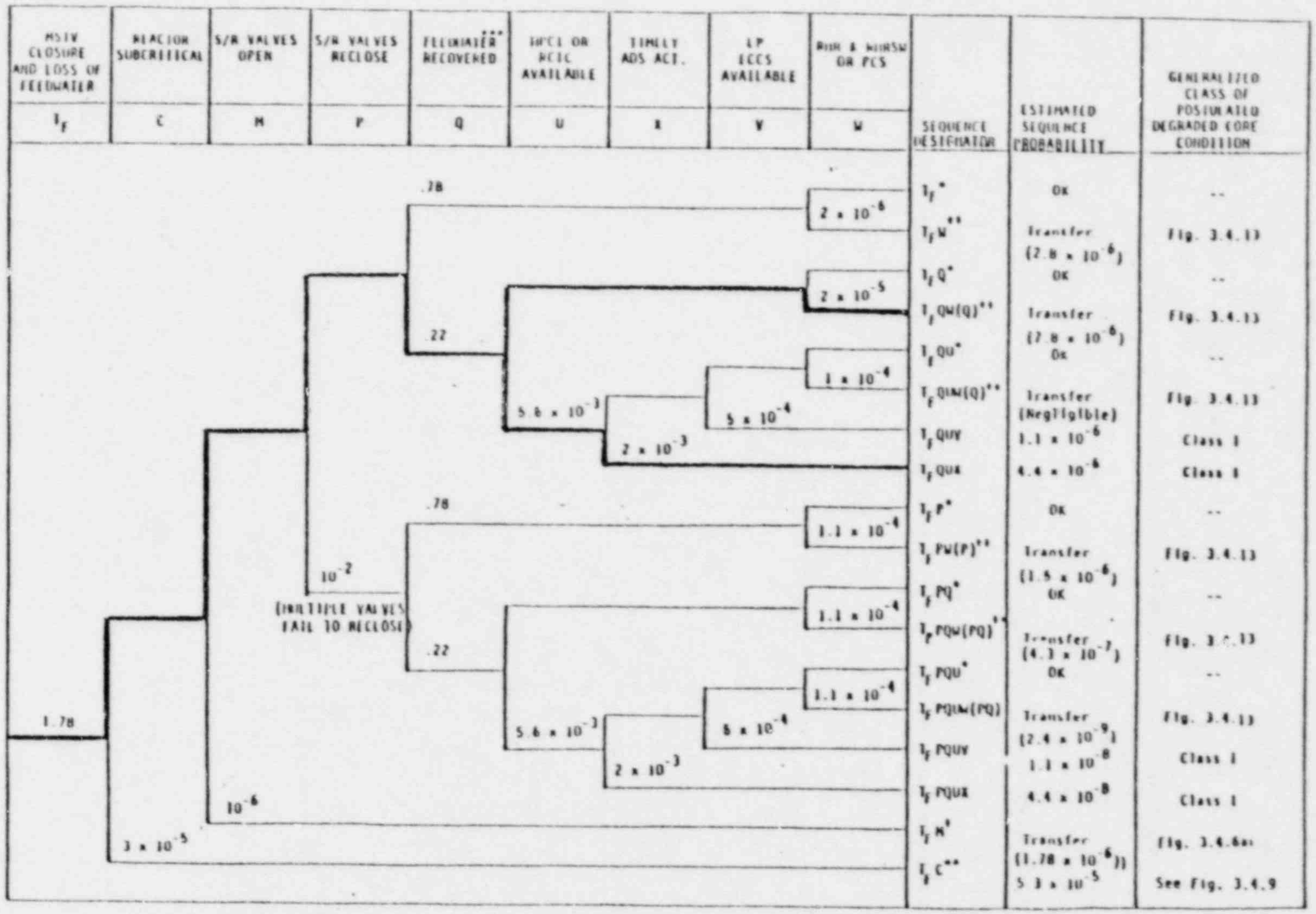
CLASS II

$T_F QW(Q)$

MSIV CLOSURE INITIATOR COUPLED WITH A LONG-TERM PROBLEM WITH THE POWER CONVERSION SYSTEM.

A CONTRIBUTING SEQUENCE TO CLASS II

- INITIATOR: MSIV CLOSURE (T_F)
- FEEDWATER UNAVAILABLE SHORT TERM (Q)
- CONTAINMENT HEAT TREMOVAL INADEQUATE (W)
  - RHR
  - RCIC IN THE STEAM CONDENSING MODE
  - POWER CONVERSION SYSTEM



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

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

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

CLASS IV

$T_F^2 C_M C_2$

MSIV CLOSURE INITIATOR WITH SUBSEQUENT FAILURE TO INSERT CONTROL RODS ON DEMAND

DOMINANT SEQUENCE FOR CLASS IV*

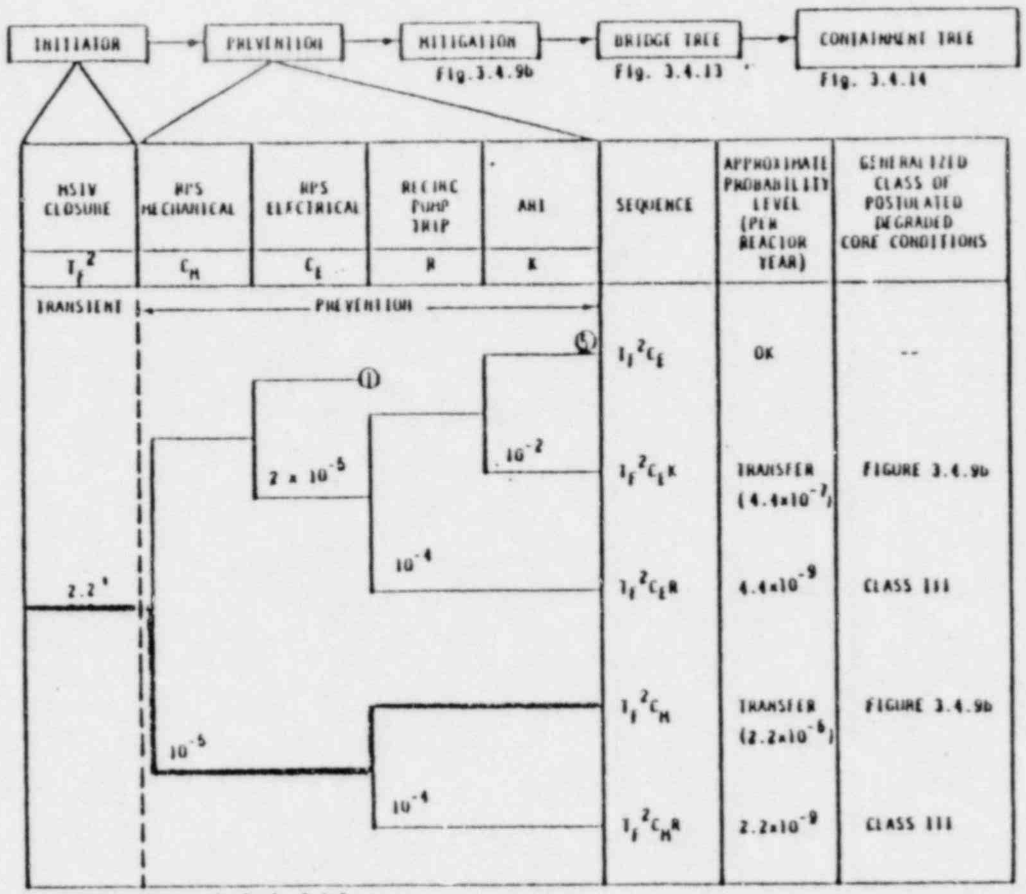
- INITIATOR - MSIV CLOSURE OR TURBINE TRIPS LEADING TO ISOLATION EVENTS (T_F^2)
- MECHANICAL COMMON-MODE FAILURE OF ALL CONTROL RODS TO INSERT (C_M)
- FAILURE OF BOTH SLC PUMPS TO PROVIDE BORON INJECTION (C_2)
- HPCI REMAINS ON THROUGHOUT DURATION OF TRANSIENT ($\overline{U}$)

---

* COMPARABLE WASH-1400 SEQUENCE IS TC

TENTATIVE REQUIREMENTS TO BE IMPOSED BY NRC

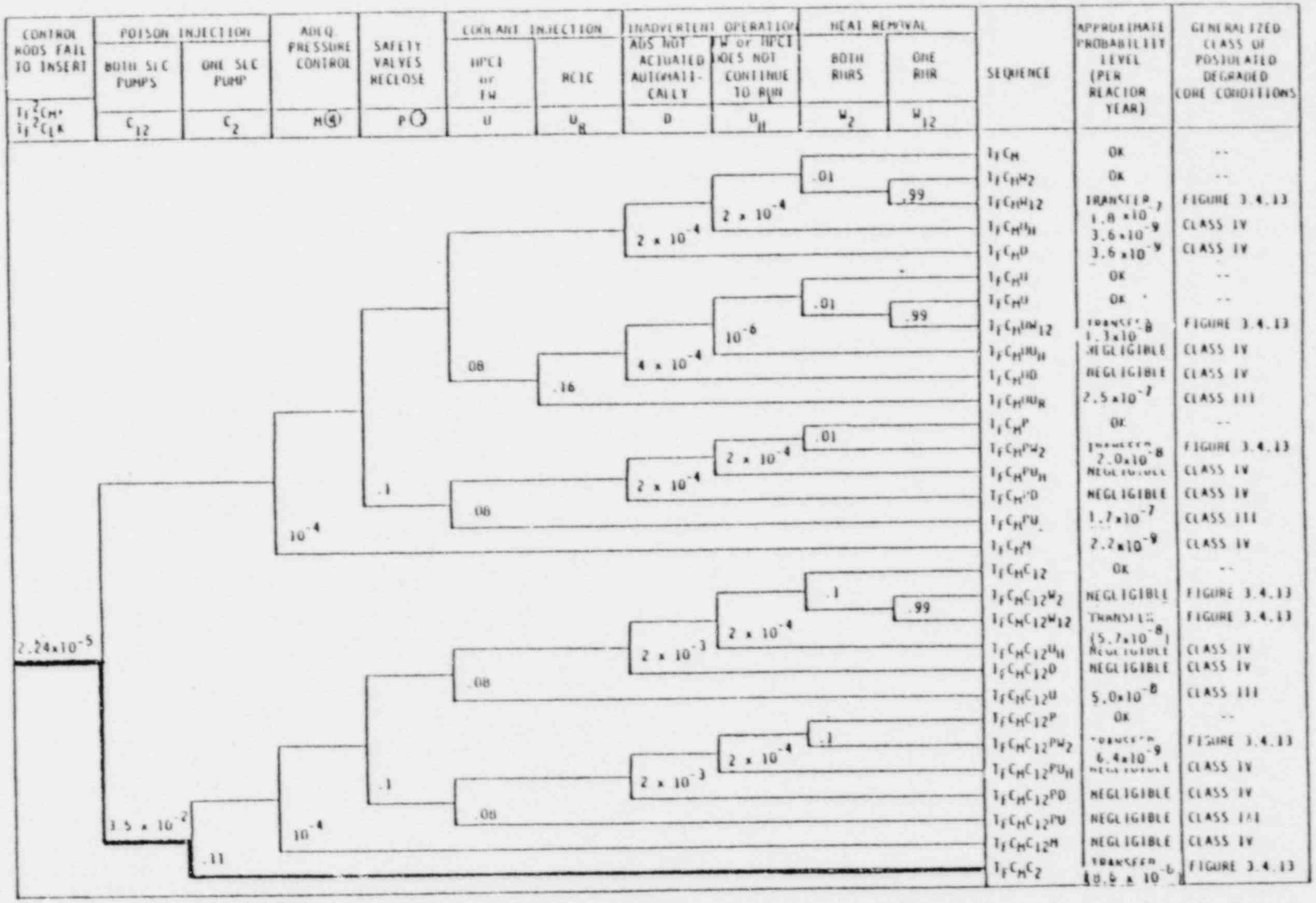
ALTERNATE 2A	ALTERNATE 3A	ALTERNATE 4A
ARI SCRAM DISCHARGE VOLUME MODIFICATION RPT REDUCED VESSEL ISOLATION PERMIT FW RUNBACK	ARI SCRAM DISCHARGE VOLUME MODIFICATION RPT REDUCED VESSEL ISOLATION PERMIT FW RUNBACK CLOSE CONTAINMENT ISOLATION VALVES OR FUEL FAILURES AUTOMATIC SLC 86 GPM	ARI SCRAM DISCHARGE VOLUME MODIFICATION RPT REDUCED VESSEL ISOLATION PERMIT FW RUNBACK CLOSE CONTAINMENT ISOLATION VALVES OR FUEL FAILURES AUTOMATIC SLC 300 - 400 GPM OPTIMIZATION STUDY IF 4A IS NOT PRACTICAL



All notes are in Table 3.4.2a.

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

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

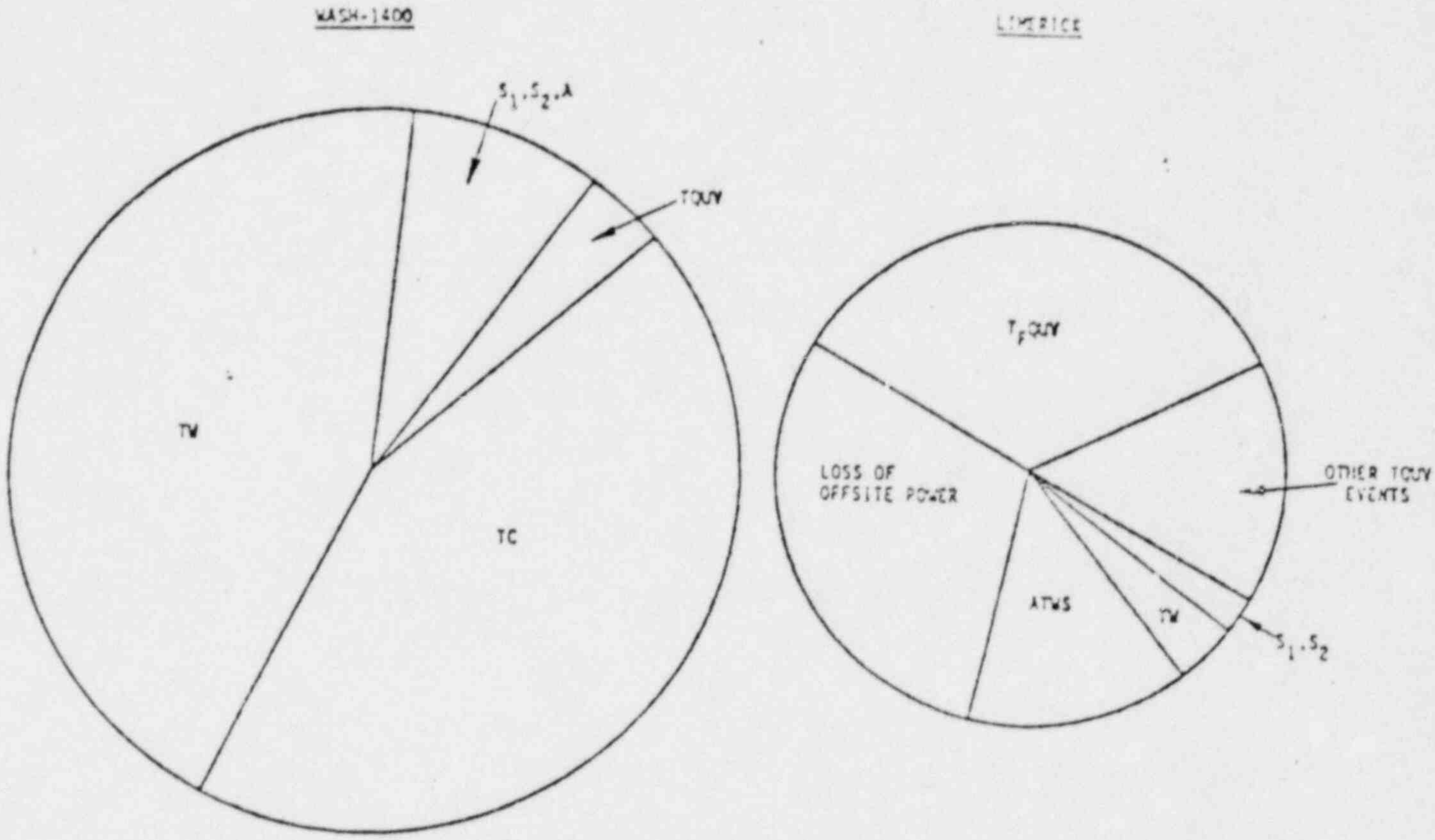


All notes are in Table 3.4.2a.

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

RESULTS

- Comparison of Limerick vs. WASH-1400 for point estimate of core melt frequency
- Summary of contributing sequences by class
- Overview of regime of investigation. (area of forecasting)



CORE MELT FREQUENCY (PER REACTOR YEAR)

Figure 3.5.4 Comparison of the Contributing Accident Sequences to the Calculated Frequency of Core Melt from WASH-1400 and the Limerick Analysis.

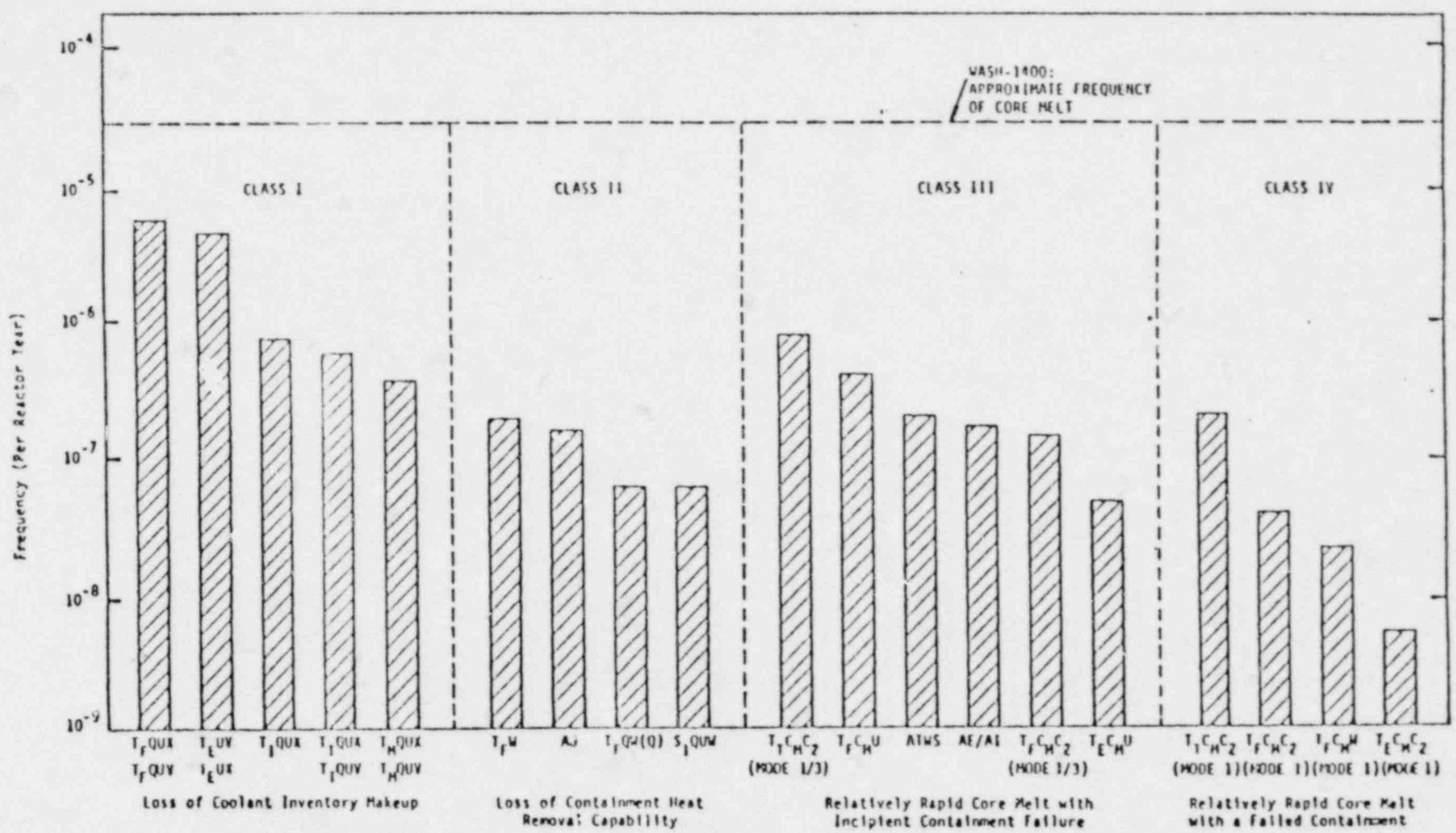
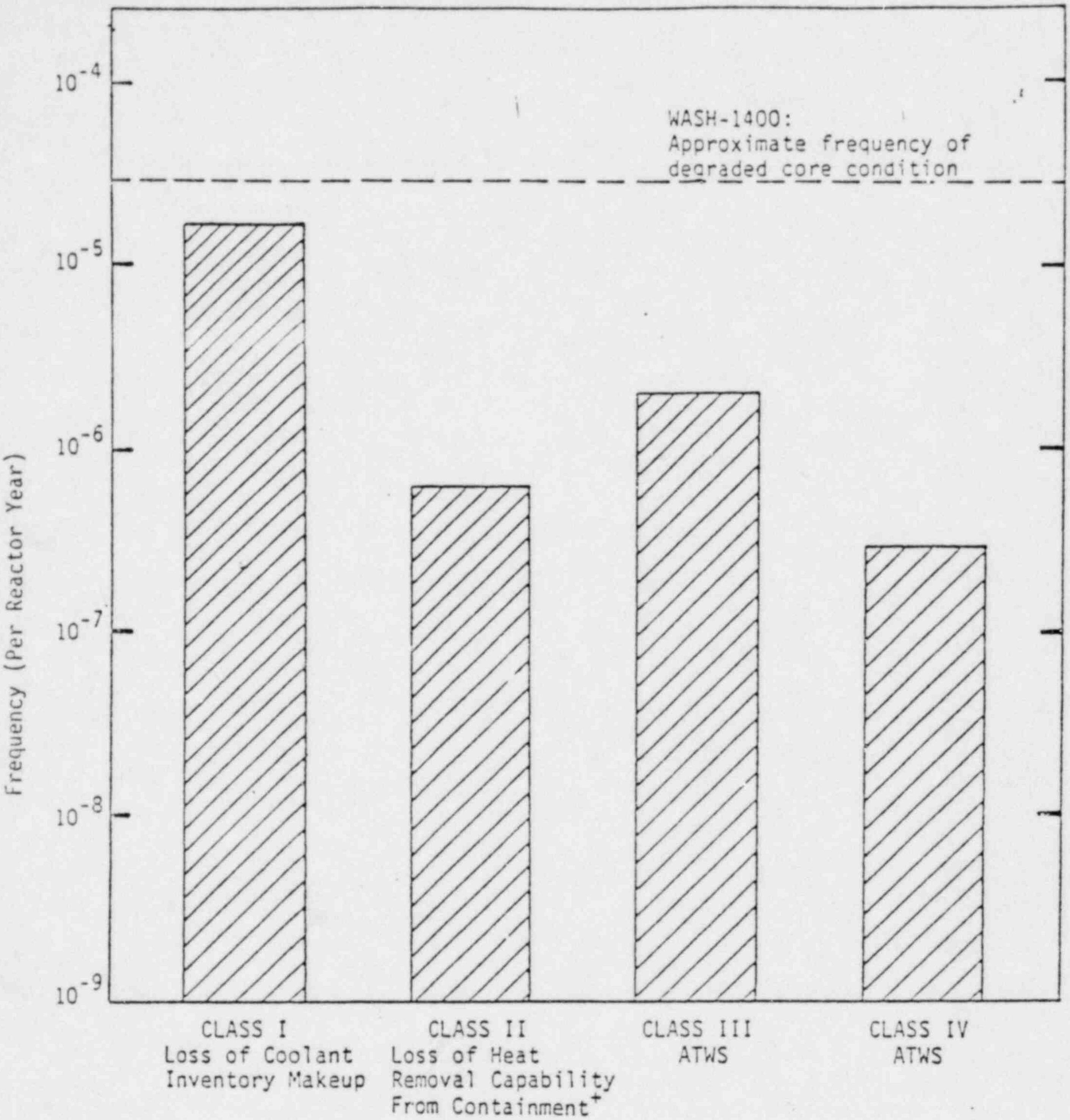
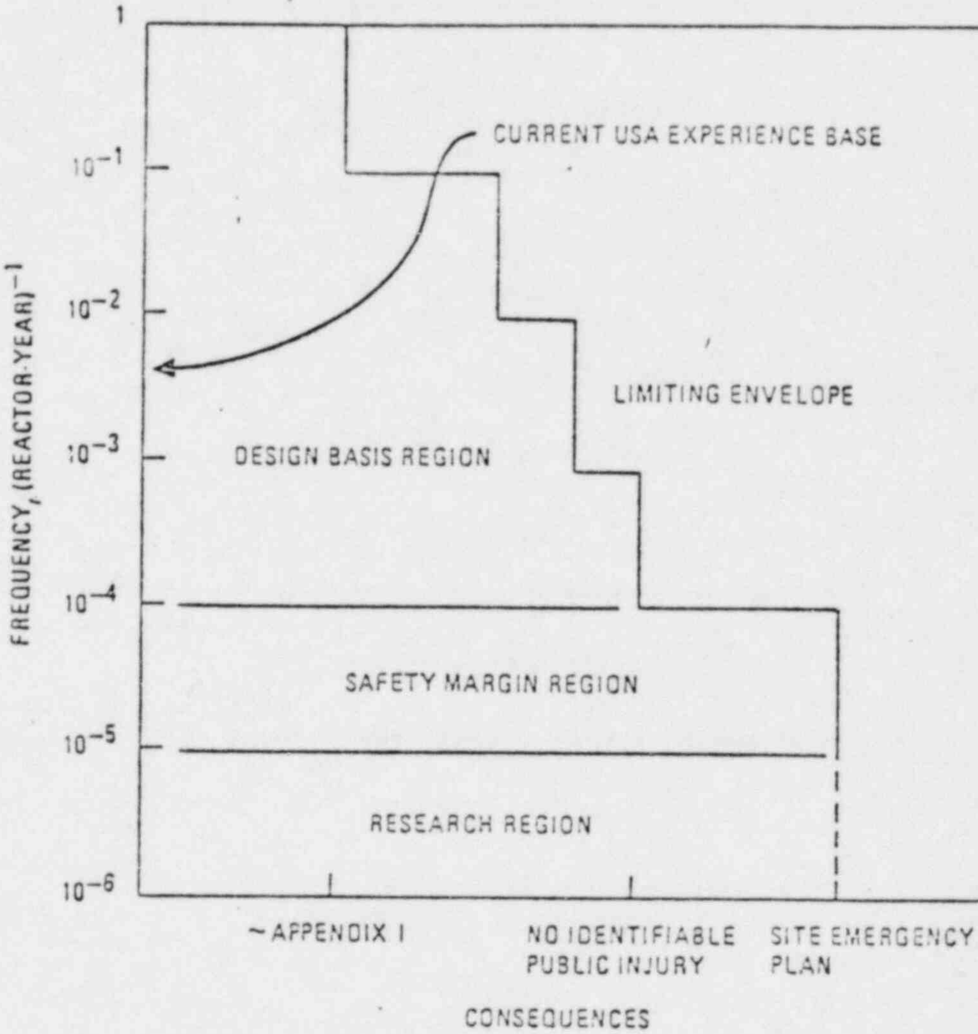


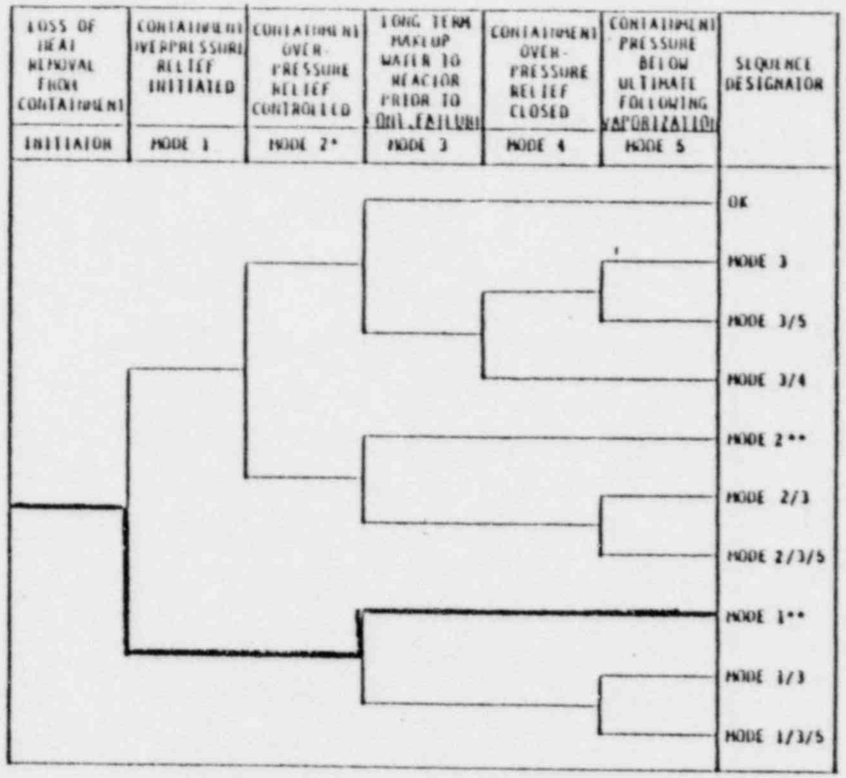
Figure 3.5.3 Summary of Dominant Accident Sequences Presented by Class



+Makes use of containment overpressure relief



Interim map of quantified safety regions



* 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).

BRIDGE TREE EVENT SEQUENCES IMPACT

SEQUENCE	FAILURE MODE	IMPACT	TIME FRAME
	NONE	OK	NA
MODE 1	COR FAILS	DELAY CORE MELT	27 HOURS
MODE 2	COR FAILS	DELAYED CORE MELT	27 HOURS
MODE 3	COOLANT MAKEUP	CORE MELT (SIMILAR TO TQUV)	2-10 HOURS
MODE 3/4*	COR FAILS OPEN	CORE MELT (DIRECT RELEASE)	2-10 HOURS
MODE 5	LONG TERM MAKE-UP FAILS AND CONTAINMENT INTEGRITY FAILS	POTENTIAL DIRECT RELEASE FROM CONTAINMENT FOLLOWING CORE MELT	2-10 HOURS

* MODE 4 IS TREATED THE SAME AS MODE 3

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	$T_T PW$	6.6×10^{-6}	MODE 1[+] (2×10^{-2})	—	1.4×10^{-7}	—	—
			MODE 1/3 (2×10^{-5})	—	negligible	—	—
			MODE 3 (1.9×10^{-4})	1.1×10^{-9}	—	—	—
			MODE 3/4 (1×10^{-5})	—	—	—	6.6×10^{-11}
	$T_M W$	1.3×10^{-6}	MODE 1 (2×10^{-2})	—	3.5×10^{-8}	—	—
			MODE 1/3 (2×10^{-5})	—	negligible	—	—
			MODE 3 (1.9×10^{-4})	3.4×10^{-10}	—	—	—
			MODE 3/4 (1×10^{-5})	—	—	—	1.8×10^{-11}
	$T_T PW$ $T_P PW$	1.5×10^{-5}	MODE 1 (2×10^{-2})	—	3.2×10^{-7}	—	—
			MODE 1/3 (2×10^{-5})	—	negligible	—	—
			MODE 3 (1.9×10^{-4})	2.9×10^{-9}	—	—	—
			MODE 3/4 (1×10^{-5})	—	—	—	1.5×10^{-10}
	$T_E PW_4$	6.5×10^{-7}	MODE 1 (2×10^{-2})	—	1.2×10^{-8}	—	—
			MODE 1/3 (2×10^{-5})	—	negligible	—	—
			MODE 3 (1.9×10^{-4})	1.2×10^{-10}	—	—	—
			MODE 3/4 (1×10^{-5})	—	—	—	6.5×10^{-12}
	$T_I C'W$ $T_I W$	8.3×10^{-6}	MODE 1 (2×10^{-2})	—	1.7×10^{-7}	—	—
			MODE 1/3 (2×10^{-5})	—	negligible	—	—
			MODE 3 (1.9×10^{-4})	1.6×10^{-9}	—	—	—
			MODE 3/4 (1×10^{-5})	—	—	—	8.3×10^{-11}
TW	TOTAL			6.2×10^{-9}	6.6×10^{-7}	—	3.3×10^{-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 CSR 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.

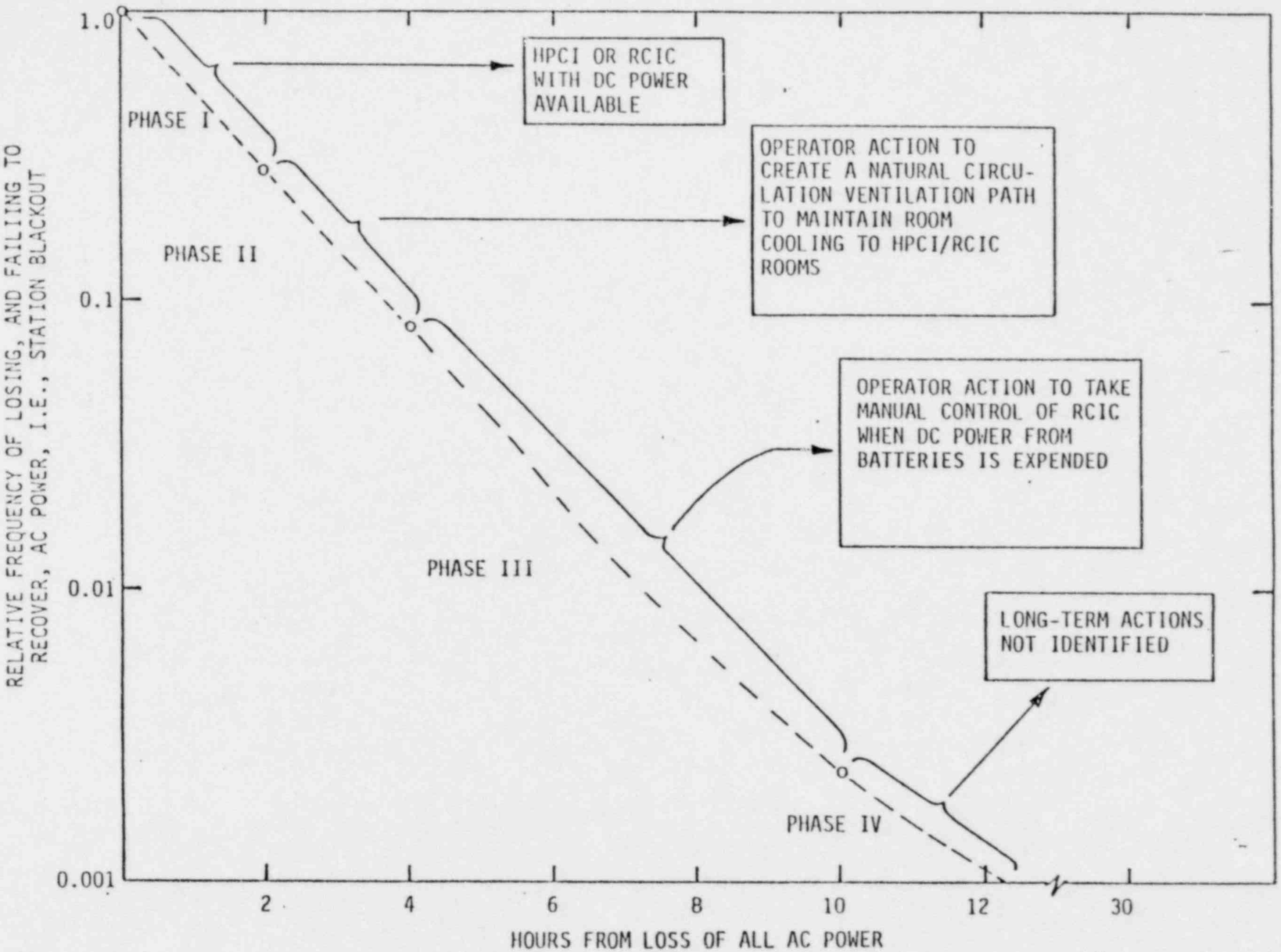
** The mode designators given in this table are the sequence designators from Figure 3.5.5 and are formed by the product of the event probabilities associated with the sequence designators. For example, the probability of sequences designated mode 1/3 is calculated as the product of the probabilities of (mode 1) x (mode 3 given mode 1) x (not mode 5) each of which is taken from Table 3.5.3; 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.

LOSS OF OFFSITE POWER

- LOSS OF OFFSITE POWER INITIATOR (T_E)
- FAILURE TO RECOVER OFFSITE POWER FOR 4 HOURS
- LOSS OF ALL DIESELS
- FAILURE TO RECOVER DIESELS FOR 4 HOURS
- DEPLETION OF STATION BATTERIES

SUMMARY OF POSSIBLE SCENARIOS CONNECTED WITH STATION BLACKOUT



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

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

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

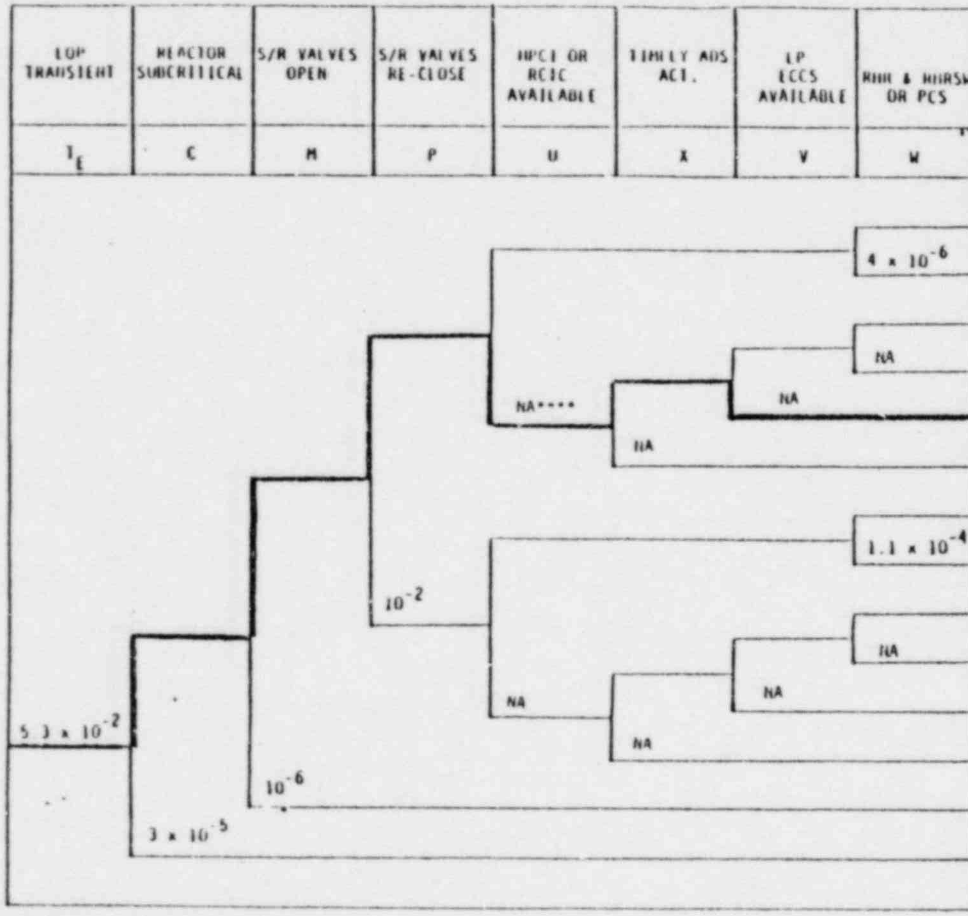
** Conditional probability of successful operation of RCIC using manual control with no power (DC or AC) for times greater than 4 hours.

† Because of the redundancy of the available low pressure pumps the dominant contributor to the 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.

††† 45 Min. used as the discrete assessed time of recovery.

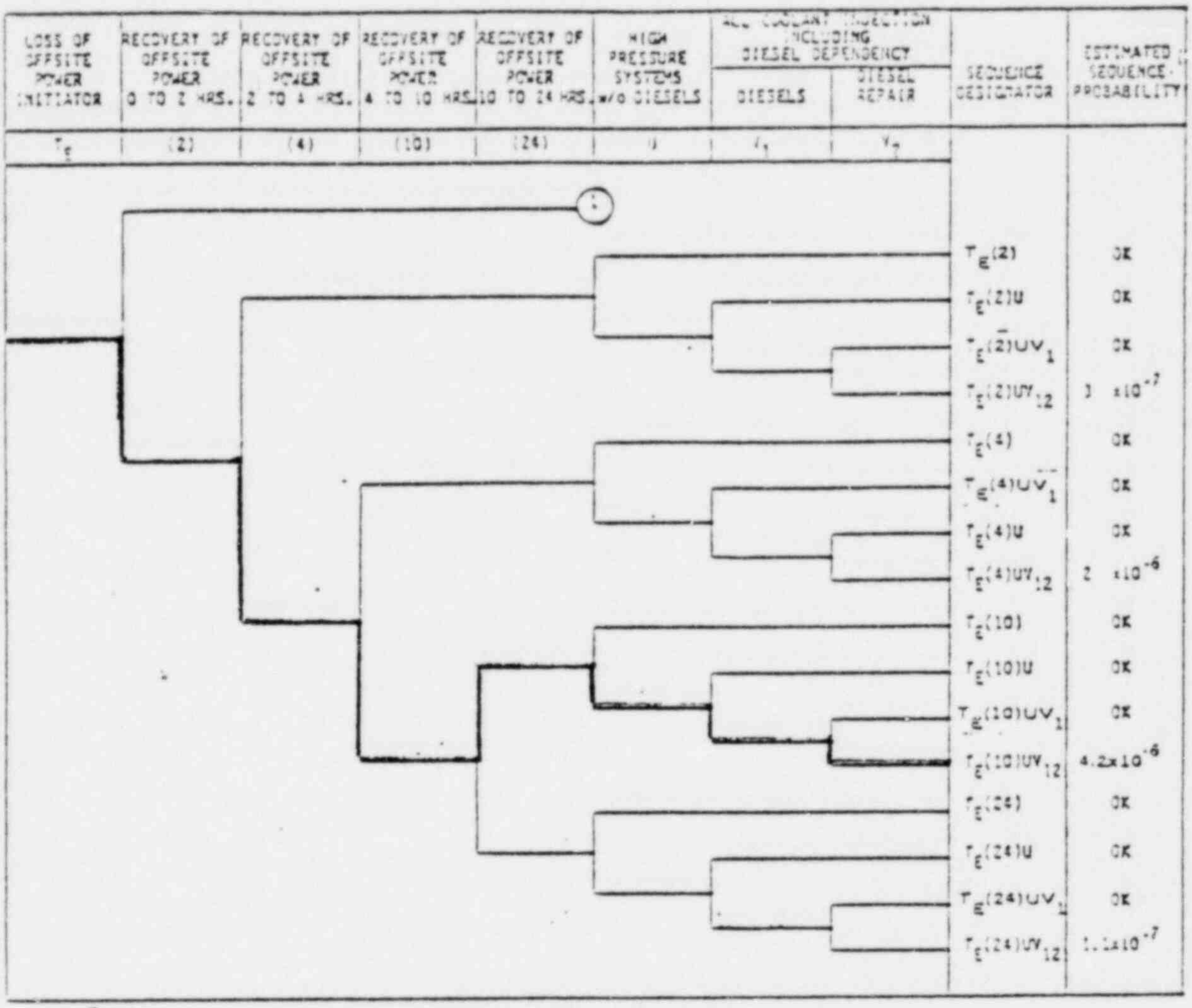


SEQUENCE DESIGNATOR	ESTIMATED SEQUENCE PROBABILITY	GENERALIZED CLASS OF POSTULATED DEGRADED CORE CONDITION
T_E*	OK	--
T_EW ++	Transfer (2.1×10^{-2})	Fig. 3.4.13
T_EU*	OK	--
T_EUW ++	Transfer (Negligible.).	Fig. 3.4.13
T_EUV	4.0×10^{-6}	Class I
T_EUX	8.4×10^{-7}	Class I
T_EP*	OK	--
$T_EPW(P)$++	Transfer (5.8×10^{-8})	Fig. 3.4.13
T_EPV	OK	--
$T_EPVW(P)$++	Transfer	Fig. 3.4.13
T_EPUV	1.2×10^{-8}	Class I
T_EPUX	2.4×10^{-9}	Class I
T_EM+	Transfer (5.3×10^{-8})	Fig. 3.4.6a
T_EC**	1.6×10^{-6}	See Fig. 3.4.10

*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 = $1.08 \times 10^{-[illegible]}$) 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



(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.

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

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

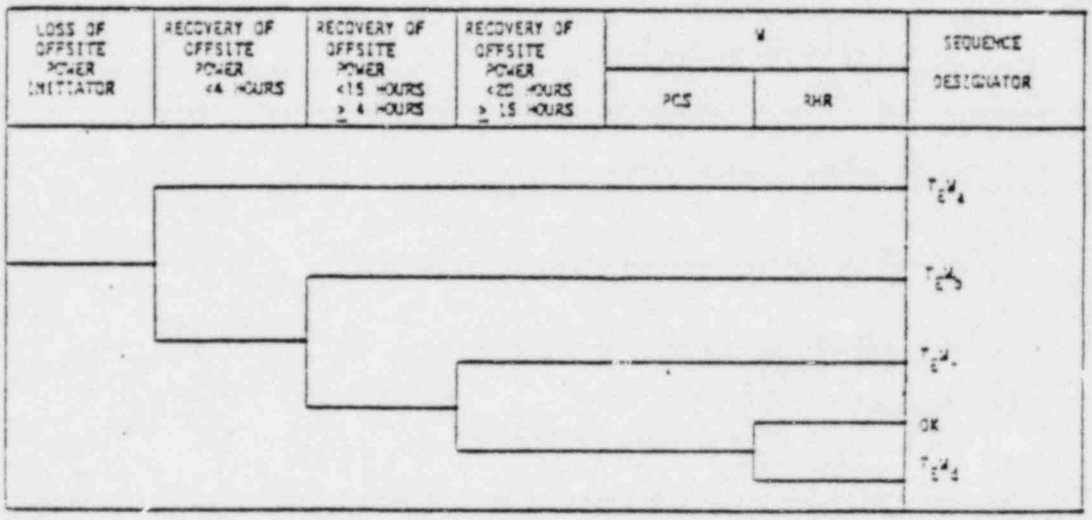
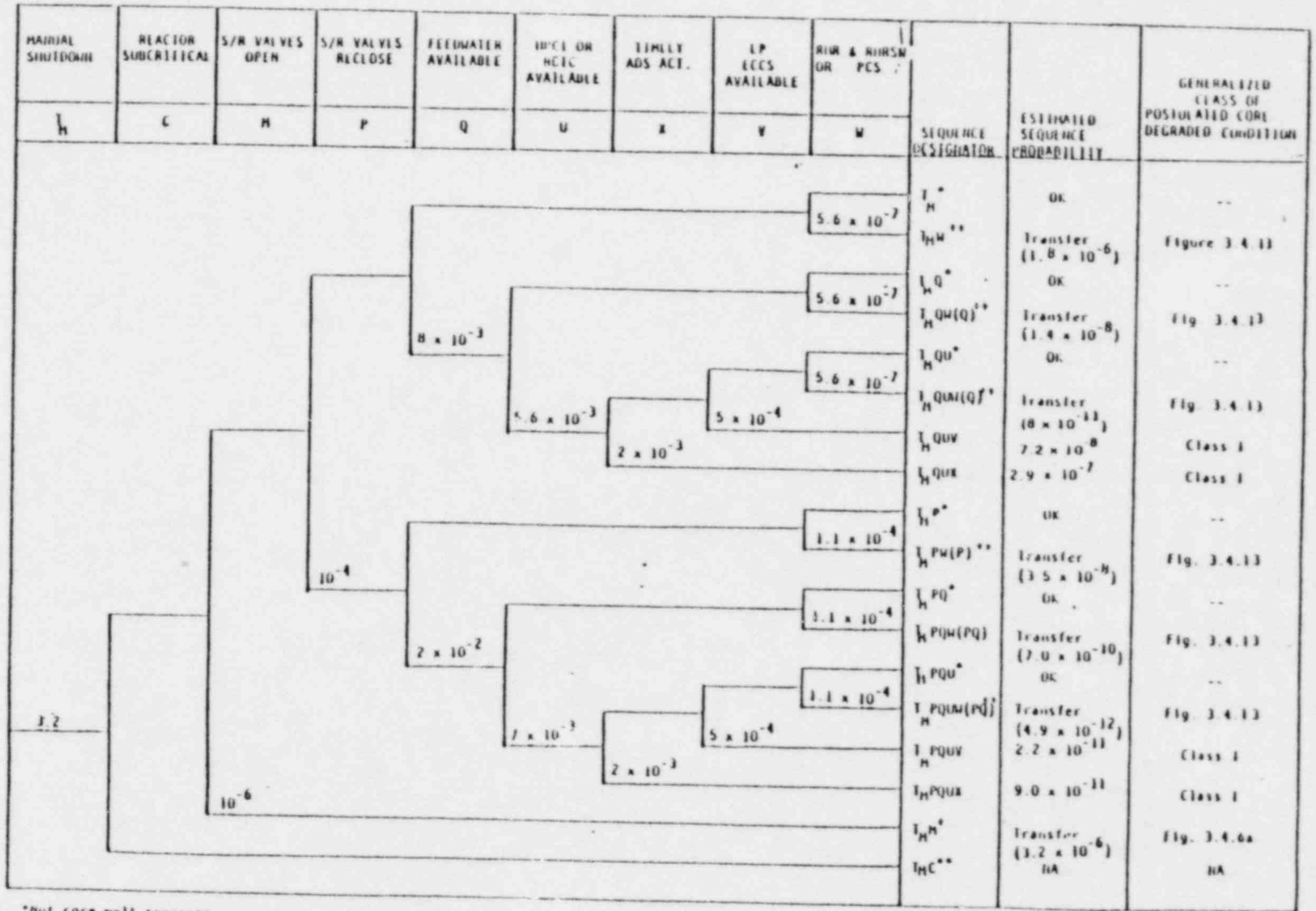


Figure 3.4.4c Time Phased Event Tree for Calculating Containment Heat Removal Capability Following a Loss of Offsite Power

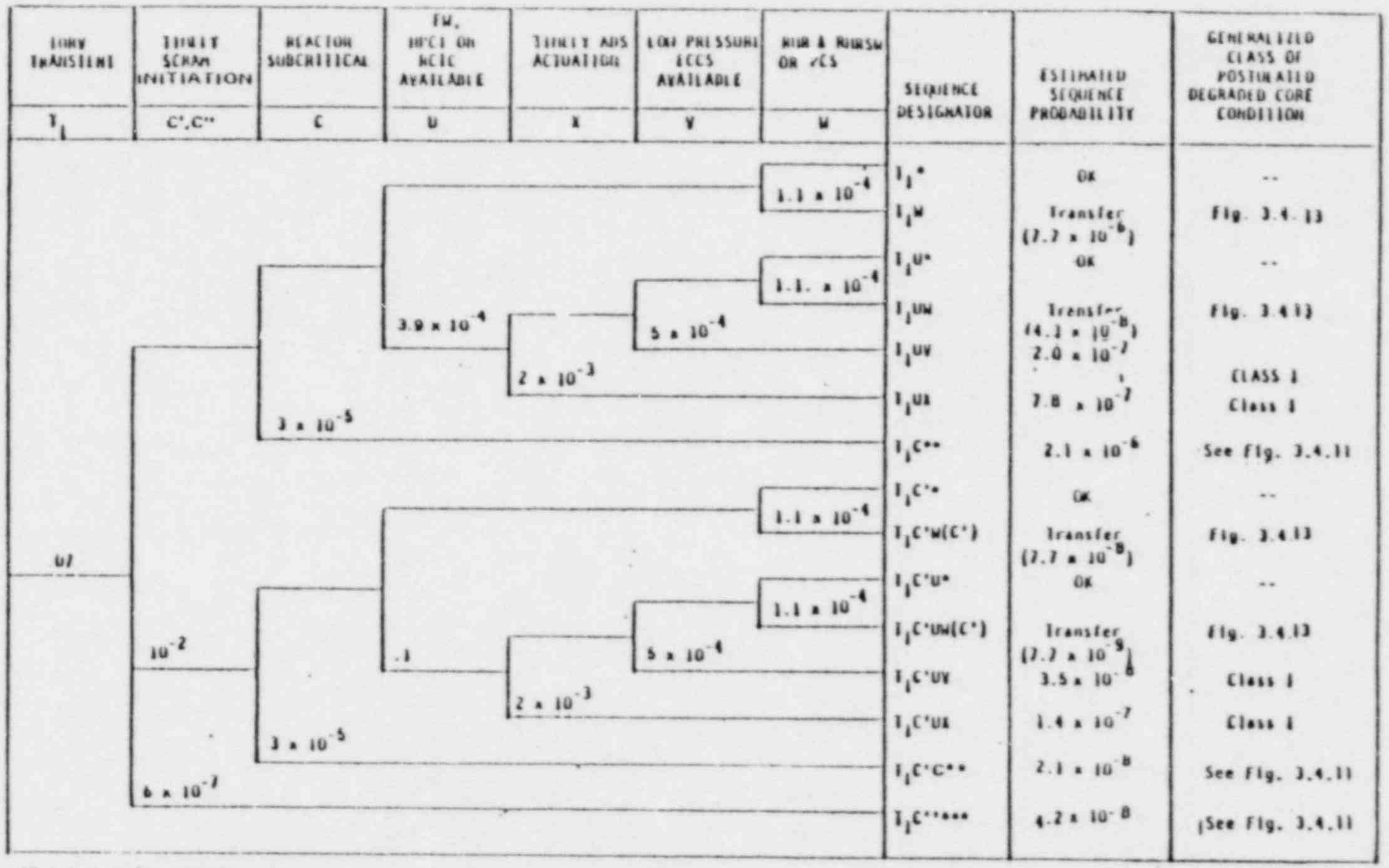
3.4.1.5 Inadvertent Open S/R Valve Transient (See Figure 3.4.5)

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



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

Figure 3.4.2 Manual Shutdown Event Tree

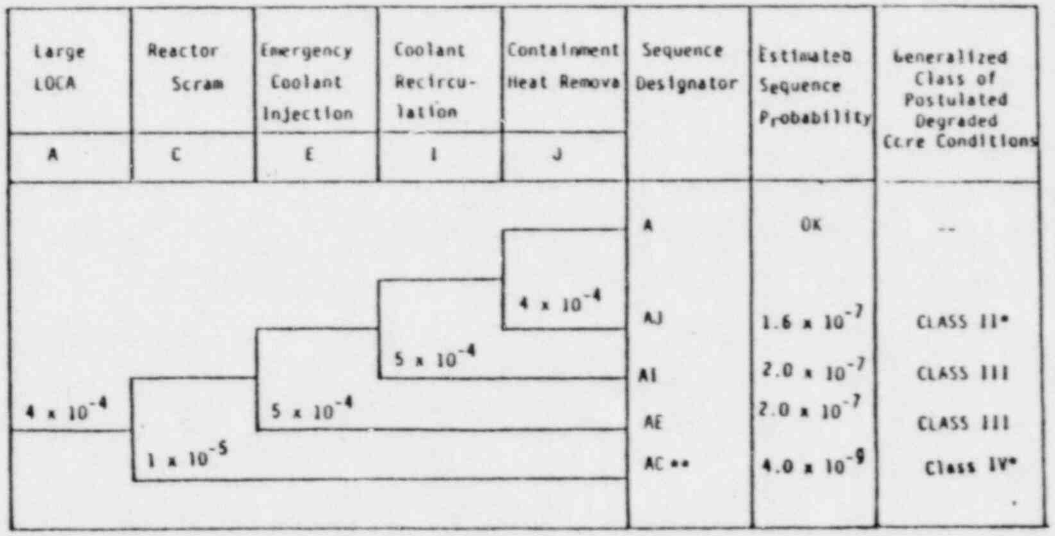


*Not core melt sequence

**ATWS initiators are treated in a separate event tree

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

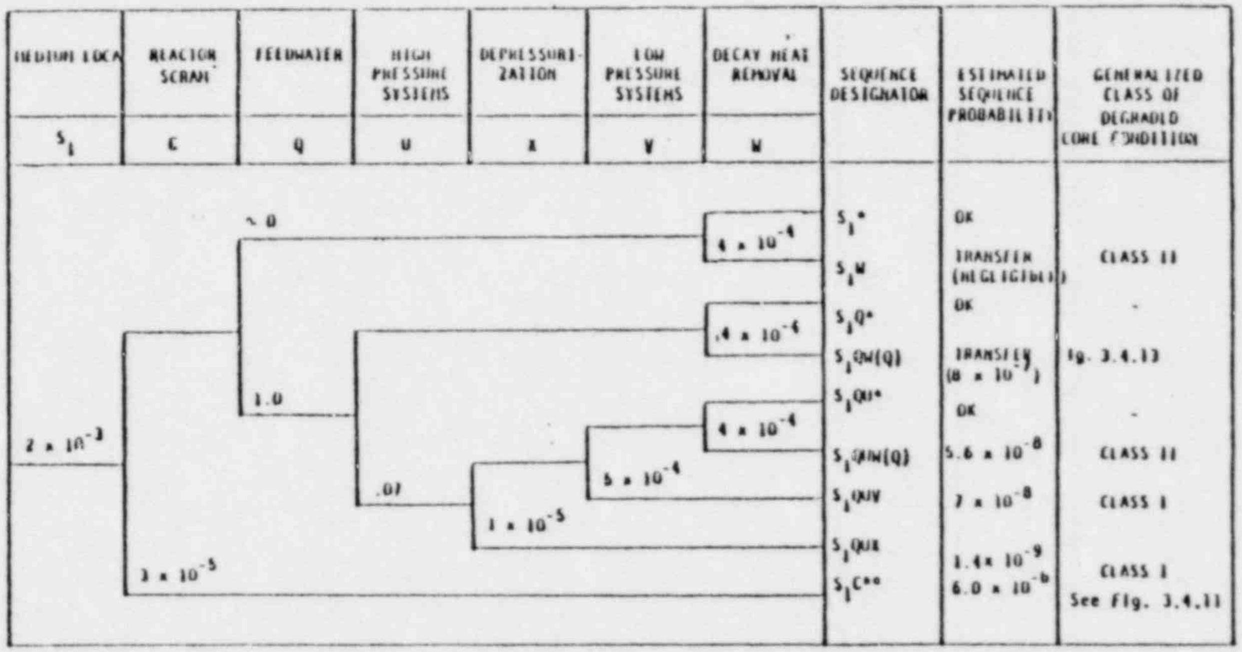
Figure 3.4.5 Inadvertent Open Safety Relief Valve Transient Event Tree



*Transfer to Bridge Tree inappropriate in this case

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

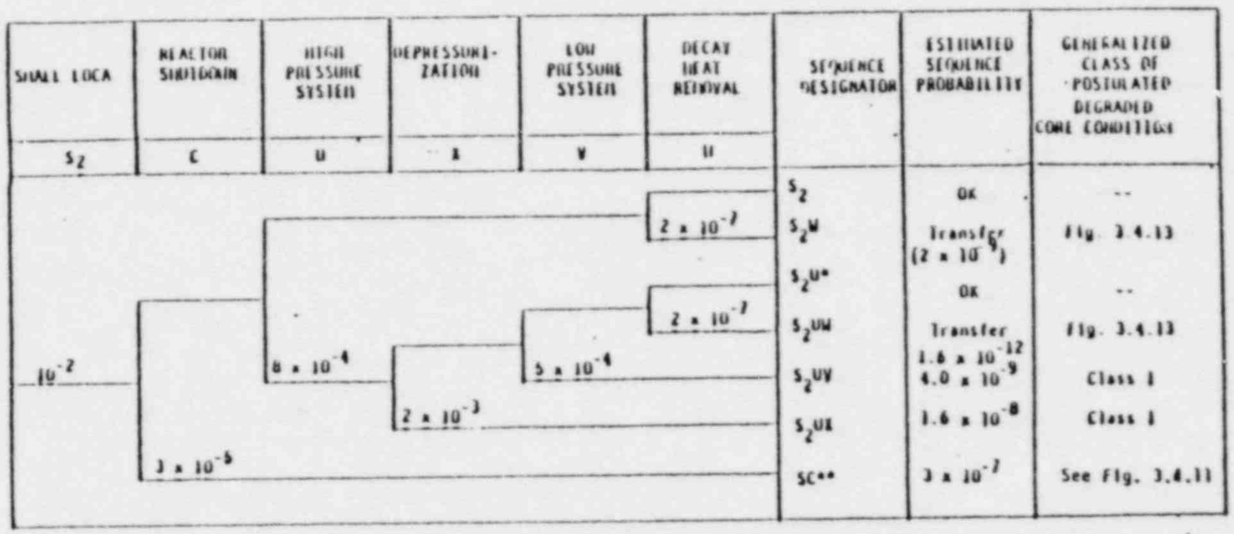
Figure 3.4.6a Limerick Large LOCA Event Tree



*Not a Core Melt Sequence

**Treated in ATWS Trees

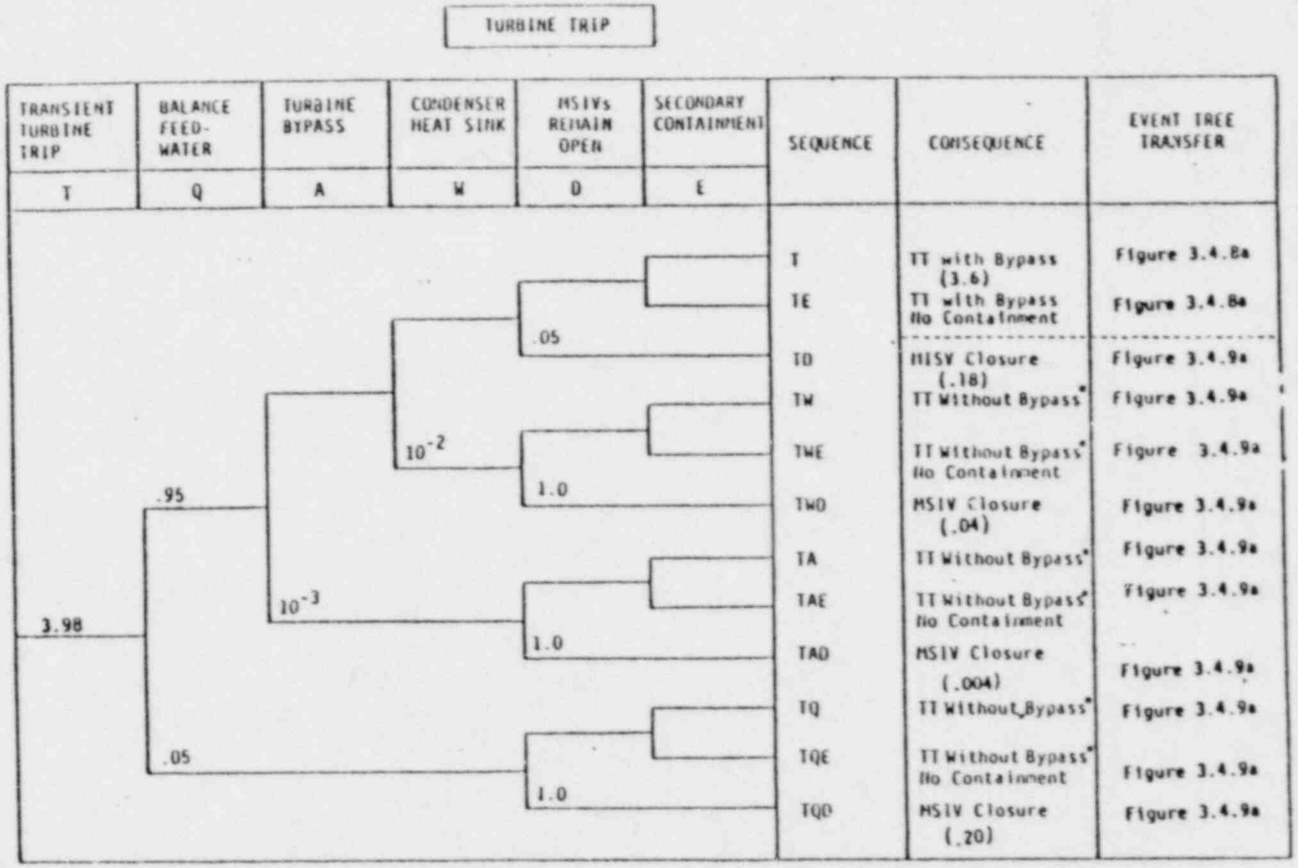
Figure 3.4.6b Limerick Medium LOCA Event Tree (S_1)



*Not a core melt sequence

**Treated in ATWS trees

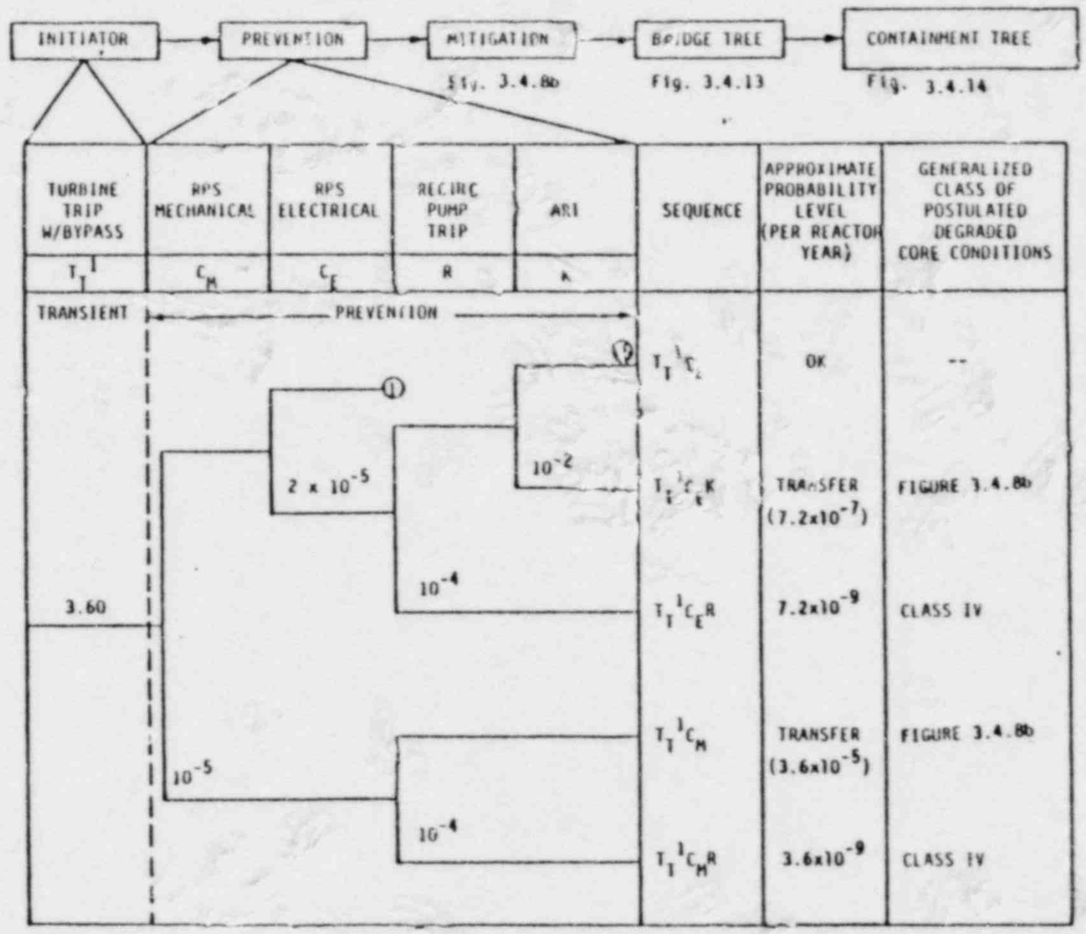
Figure 3.4.6c. Limerick Small LOCA Event Tree (S_2)



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

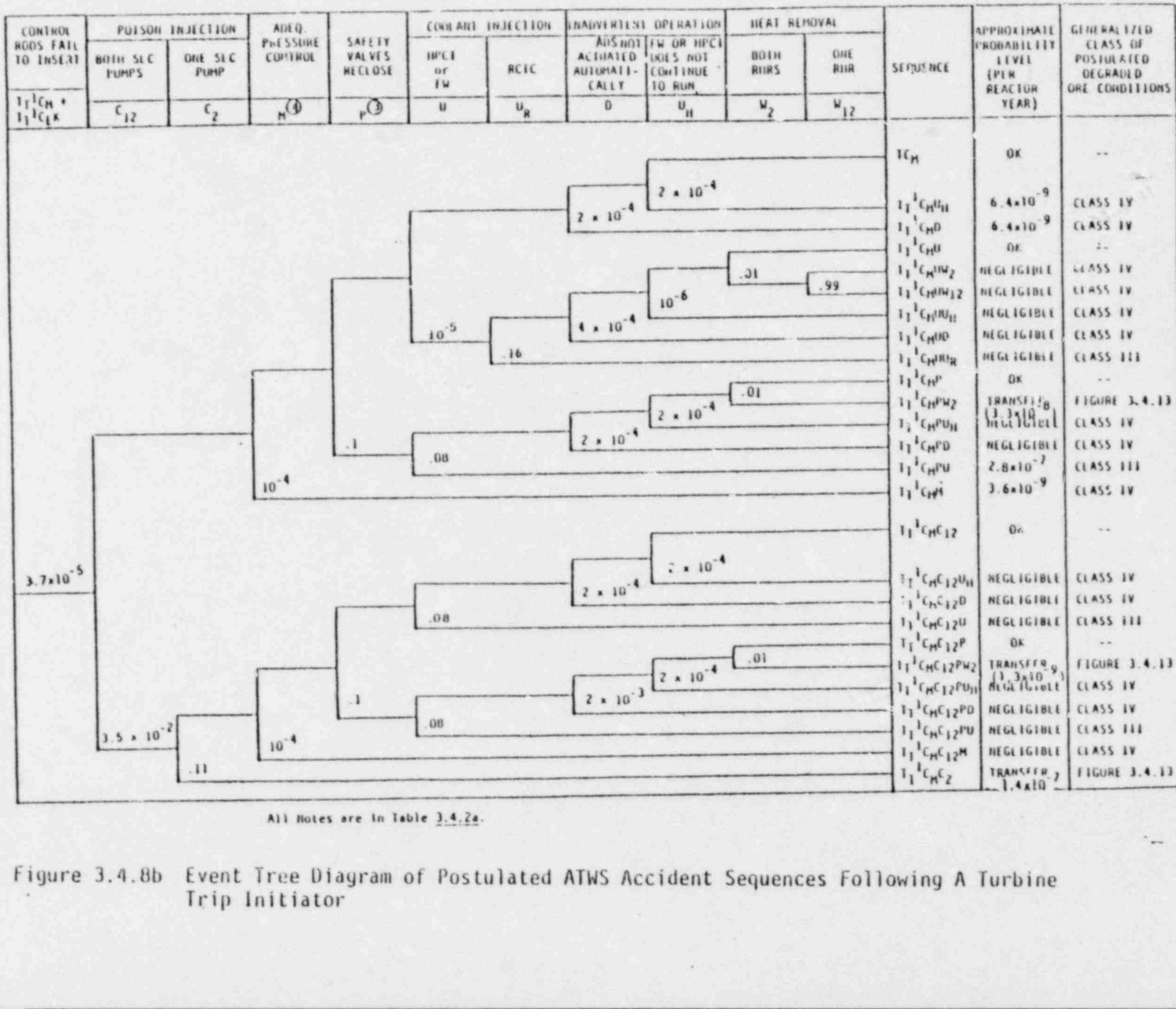
Figure 3.4.7 Event Tree Diagram of Accident Sequences Following a Turbine Trip Initiator.

NOTE: This event tree is evaluated assuming that a turbine trip followed by a failure to scram is in progress. The use of the tree is to discriminate between events leading to isolation and those for which the condenser remains availab.e



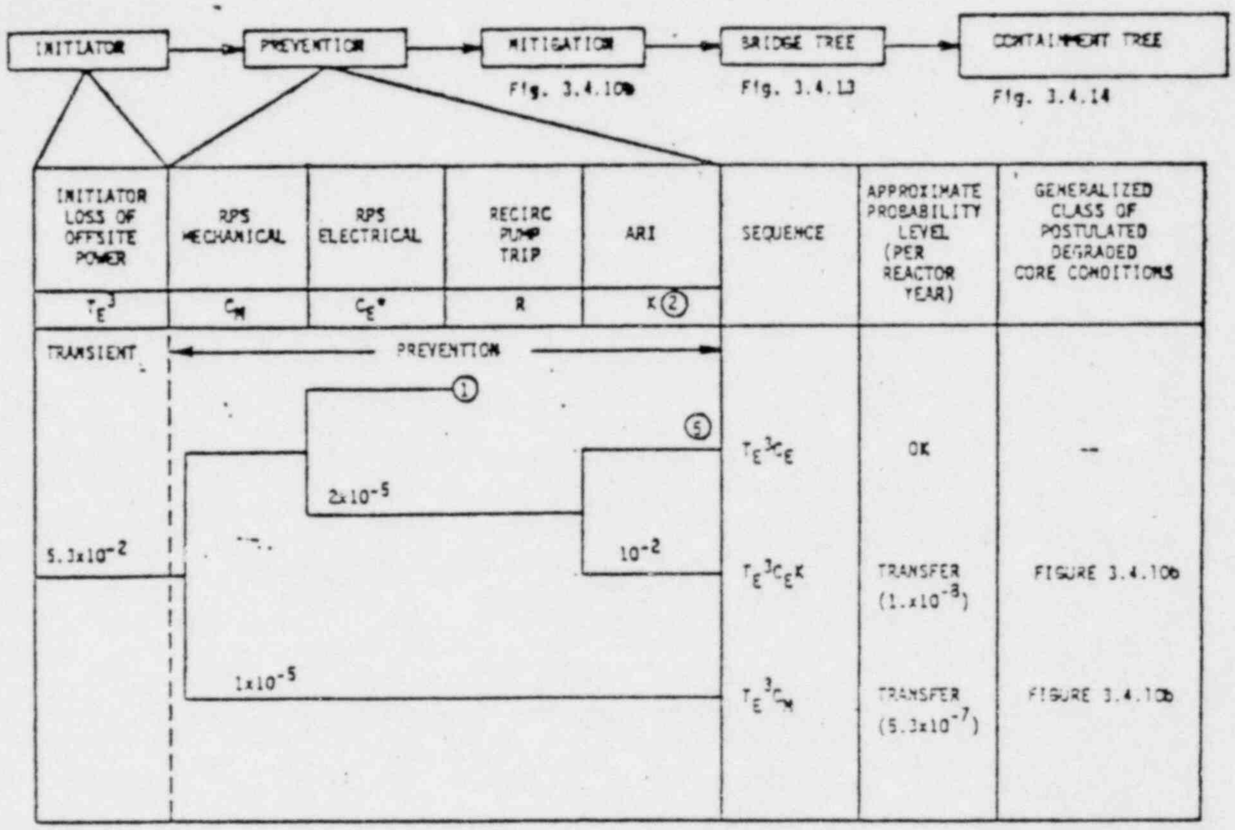
All Notes are explained in Table 3.4.2a.

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



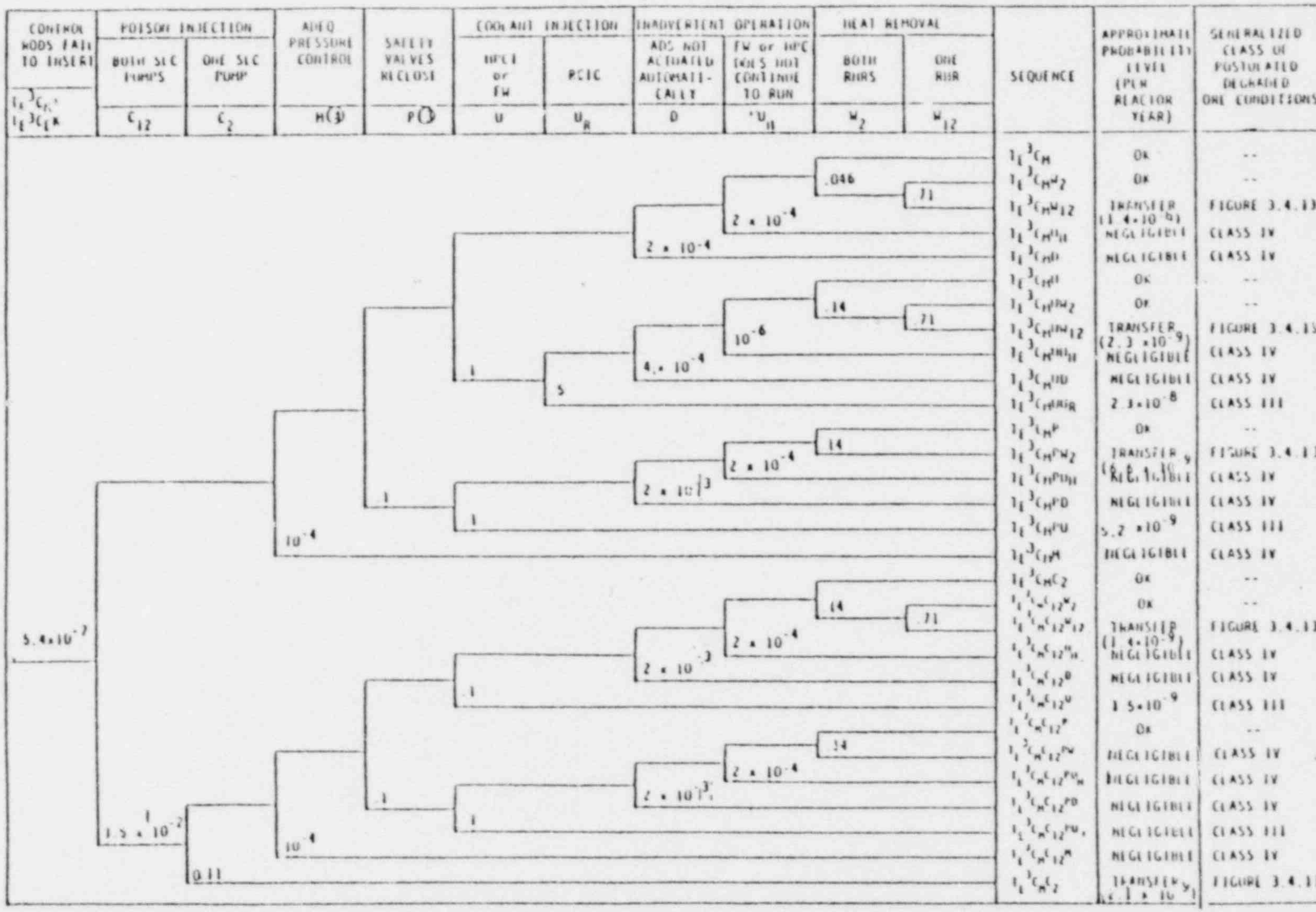
All Notes are in Table 3.4.2a.

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



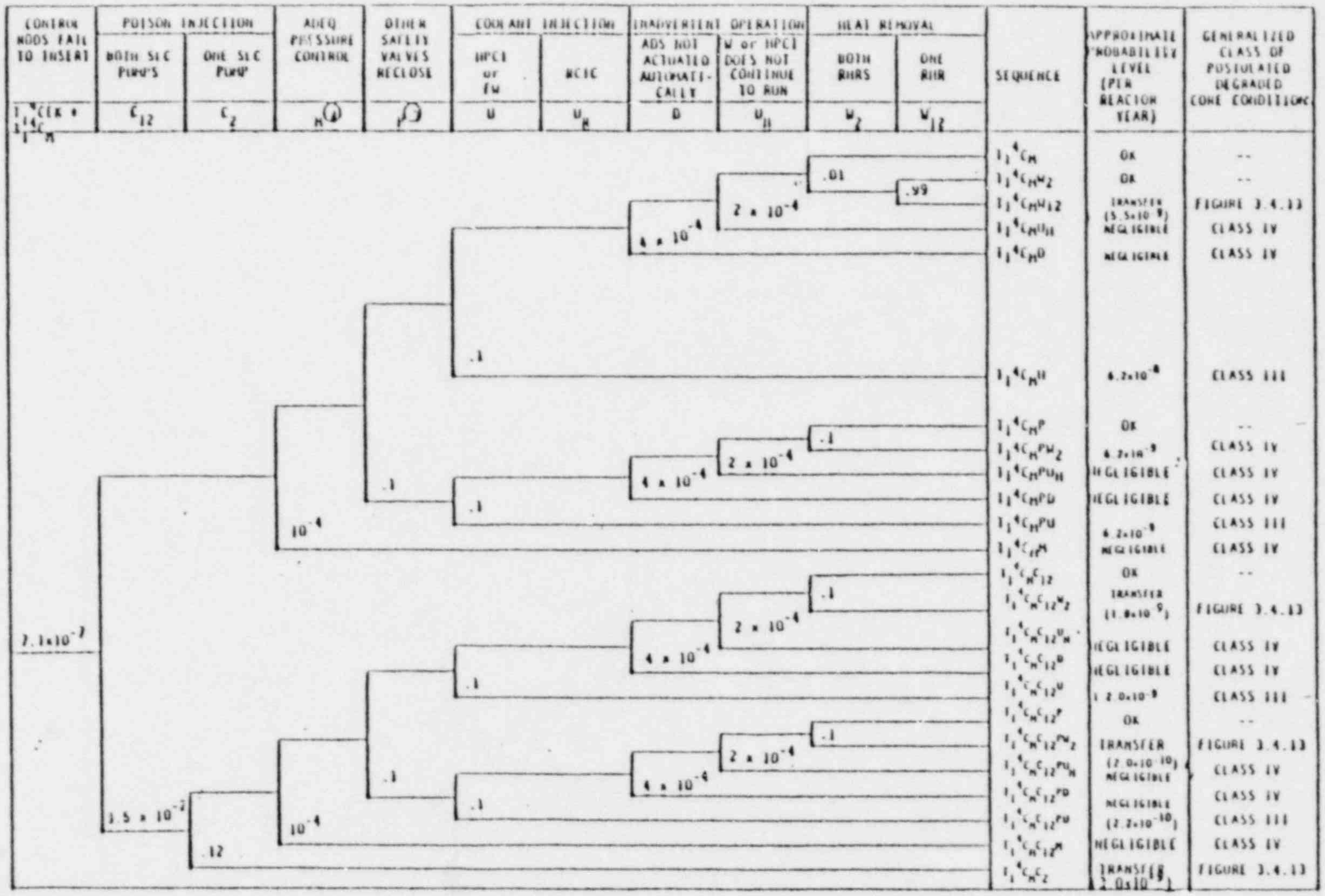
* For the loss of offsite power initiator, electrical faults leading to a failure to scram may be virtually zero for loss of offsite power incidents. However, since no detailed evaluation has been performed to verify this assertion, electrical RPS failures are included here for completeness. They are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

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



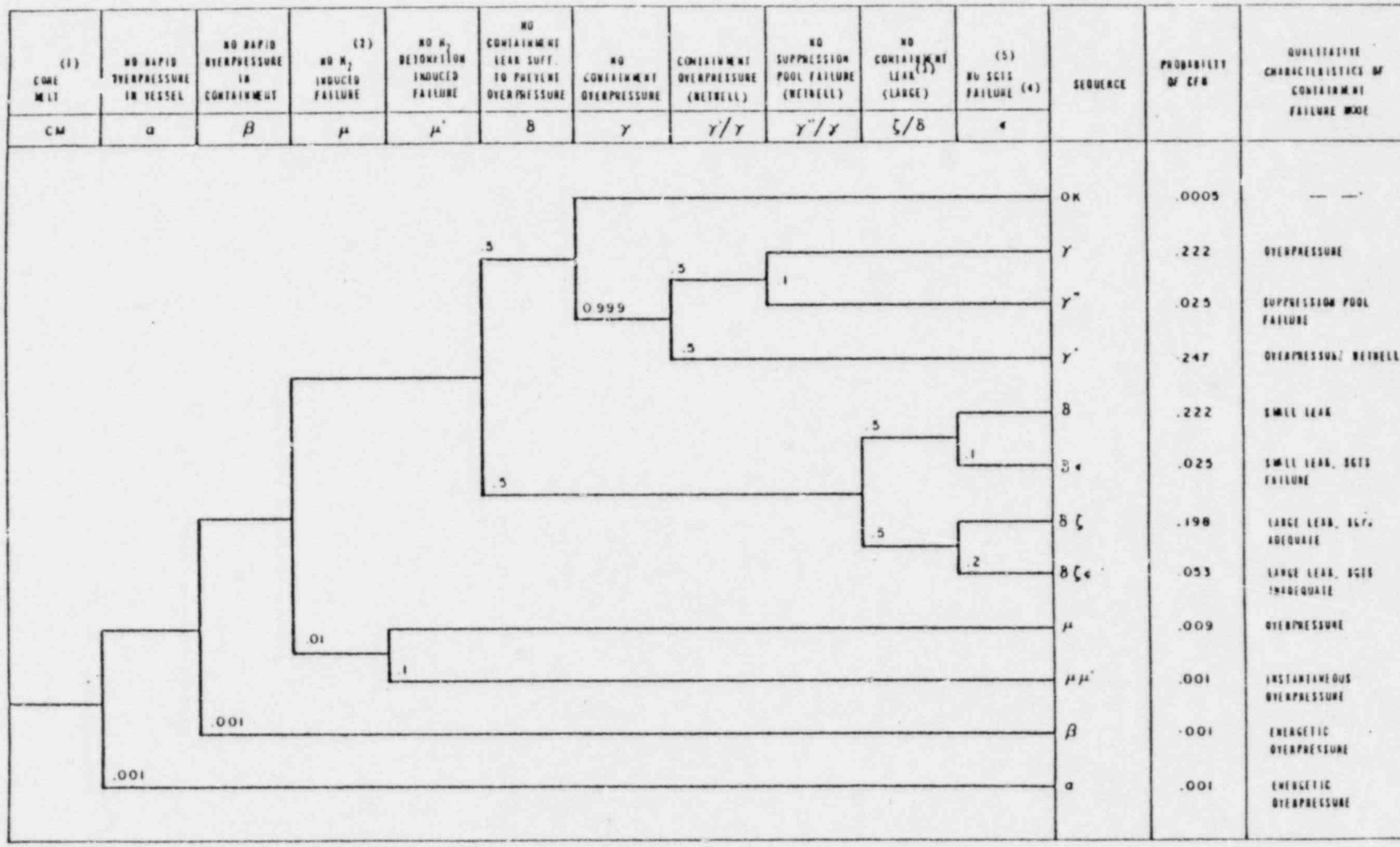
All notes are in Table 3.4.2a.

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



All notes are in Table 3.4.2a

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



(1) CONTAINMENT FAILURE MAY HAVE OCCURRED PRIOR TO CORE MELT. IN THOSE CASES (CLASS II AND CLASS IV), THE CONTAINMENT FAILURE MODES ARE ONLY USED AS MECHANISMS FOR RELEASE FRACTION DETERMINATION.

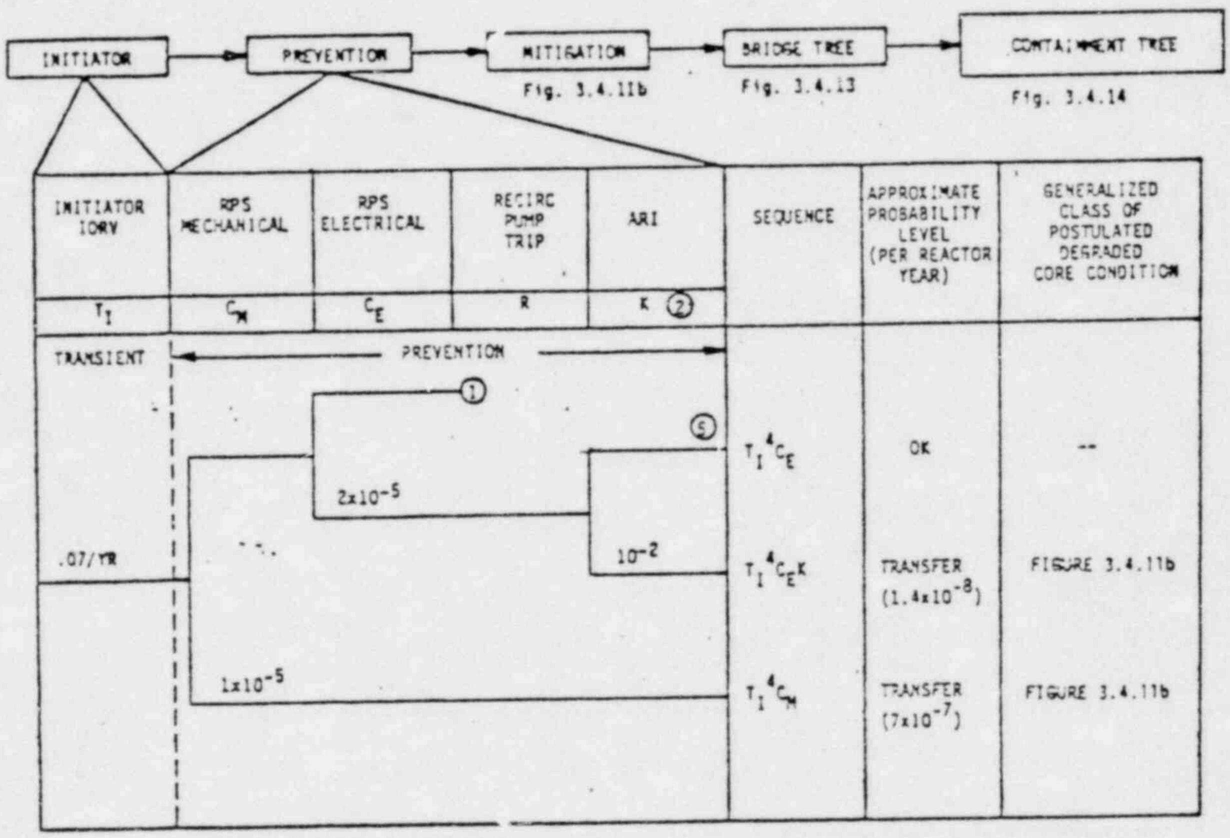
(2) ASSUMES THAT H_2 EXPLOSION IN CONTAINMENT CAUSES OVERPRESSURE FAILURE WITH DIRECT PATHWAY TO OUTSIDE ATMOSPHERE.

(3) LEAKAGE AT 2400 VOLUME PERCENT/DAY.

(4) FAILURE STANDBY GAS TREATMENT SYSTEM.

(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



All notes are in Table 3.4.2a.

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

SYSTEM UNAVAILABILITY

- FAULT TREE/EVENT TREE DATA INPUT
- METHODS OF EVALUATING THE FAULT TREE
- TABLE OF COMPARISON WITH WASH-1400
- DISCUSSION OF HPCI FAULT TREE
  - SUBSEQUENT STARTES
  - MAINTENANCE UNAVAILABILITY
  - ELECTRIC POWER
- FAULT TREE VERIFICATION: COMPARISON OF FAULT TREE RESULTS VERSUS OPERATING EXPERIENCE DATA

FAULT TREE INPUT DATA

THE NEXT SET OF DATA TO BE DISCUSSED IS THE FAULT TREE INPUT DATA. THESE DATA TAKE ON A WIDE VARIETY OF FORMS AND INCLUDE THOSE DISCUSSED BELOW.

- TRANSIENT INITIATORS
- COMPONENT FAILURE RATE
- HUMAN ERROR RATE
- MAINTENANCE OUTAGE OF SAFETY SYSTEMS
- OFFSITE POWER UNAVAILABILITY
- LOCA INITIATORS

SOURCES OF COMPONENT FAILURE RATE DATA

- PLANT- OR COMPONENT-SPECIFIC DATA (E.G., LOSS OF OFFSITE POWER),
- NRC GENERIC DATA (PUMPS (16), VALVES (17), DIESELS (18), AND HUMAN ERRORS (19),
- IEEE-500 (ELECTRONIC COMPONENTS, (20), (NOT USED),
- VENDOR DATA,
- THE RACTOR SAFETY STUDY (21), AND
- NPRDS (NOT USED).

MODEL INPUT DATA

ITEM	SOURCE OF INPUT DATA				
	GE	NRC/EG&G	WASH-1400	SWAIN GUTTMAN	PMJ OR PEACH BOTTOM OPERATING EXPERIENCE
<u>Components</u>					
Pumps		X			
Turbines			X		
Valves		X			
Instrumentation	X		X		
<u>Systems</u>					
Diesels		X	X		X
HPCI	X				
<u>Maintenance</u>	X				X
<u>Offsite Power</u>					X
<u>Human Error Probabilities</u>			X	X	

TRANSIENT	Frequency (Per Reactor Year)			
	EPRI Survey of 12 BWRs			BWR OP. EXP.
	All Years	Exclude Year 1	Exclude Year 1 &<25% Power	GE Assessment
MSIV Closure	1.34	.57	.35	1.08
Closure of all MSIVs (4)	0.67	0.19	0.13	1.00
Turbine Trip Without Bypass (5)	0.00	0.00	0.00	0.01
Loss of Condenser (8)	0.67	0.38	0.22	0.067
Turbine Trip	7.62	4.23	2.95	3.98
Partial Closure of MSIVs (6,7)	0.12	0.14	0.12	0.20
Turbine Trip with Bypass (3,13,30,33,34,35,36,37)	3.88	1.98	1.21	1.33
Startup of Idle Recirculation Loop	0.38	0.08	0.09	0.25
Pressure Regulator Failure (9,10)	0.43	0.35	0.31	0.67
Inadvertent Opening of Bypass (12)	0.04	0.05	0.00	0.00
Rod Withdrawal (27,28,29)	0.14	0.14	0.06	0.10
Disturbance of Feedwater (20,21,23,24,25,26)	1.39	0.65	0.53	0.68
Electric Load Rejection (1,2)	1.04	0.70	0.63	0.75
Loss of Offsite Power (31,32)	.16	.11	.14	.38
Inadvertent Open Relief Valve (11)	.20	.08	.03*	.06
Loss of Feedwater (22)	.27	.16	.06	.70
TOTAL	9.43	5.04	3.51	6.2

*Modifies to .07 based upon NUREG-0626.

SUMMARY OF TRANSIENT INITIATOR FREQUENCIES

Source	Total Transient Initiator Frequency
WASH-1270	1
WASH-1400	10
NUREG-0460	6 to 8
EPRI	3.5(ATWS) - 5.04(BWR)
GE Evaluation	6.2

FAILURE RATE DATA COMPARISON

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

COMPARISON OF PRINCIPAL COMPONENT FAILURE
PROBABILITIES FROM THREE DATA SOURCES

(As Used for Input to the Fault Tree Model.
These Probabilities of Component Failure are
Assessed for the Duration of the Event Analyzed)

Component	WASH-1400	GE	NRC
Pump	2×10^{-3}	2.8×10^{-3}	2×10^{-3}
Valves			
NCFC*	1.25×10^{-3}	0.6×10^{-3}	3×10^{-3}
NCFO*	1.25×10^{-4}	0.6×10^{-4}	1×10^{-3}
Diesels	3×10^{-2}	--	6×10^{-2}
ADS Relief Valve	1.25×10^{-5}	3×10^{-2}**	5×10^{-3}

*NCFC - Normally Closed Fails Closed
NCFO - Normally Closed Fails Open

**Data Taken in 1968 during a time period in which a generic relief valve problem existed. The problem was subsequently corrected.

COMPONENT DATA VARIATIONS NOT INCLUDED

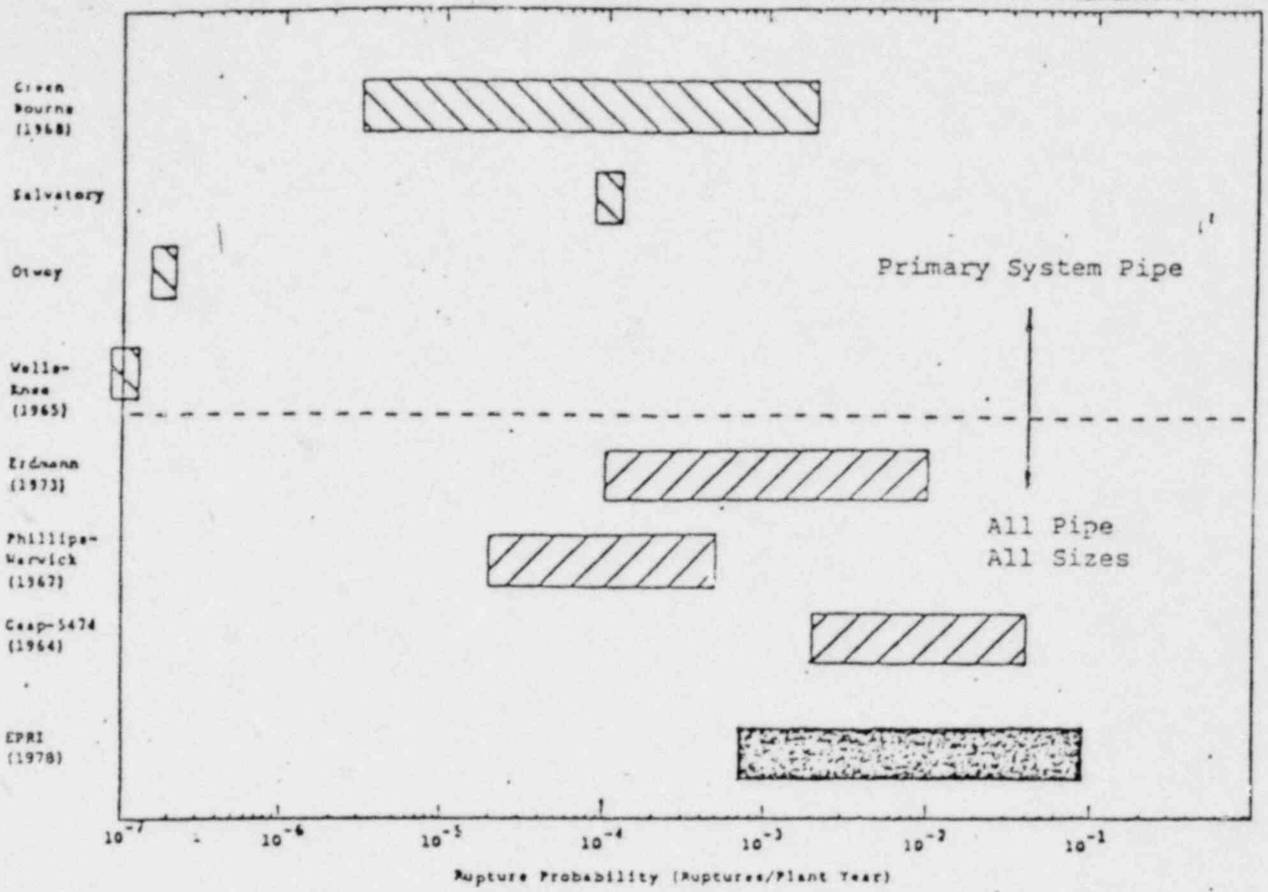
- VARIABILITY AMONG COMPONENTS, E.G., MOTOR-OPERATED VALVES (SIZE, APPLICATION, QUALIFICATION REQUIREMENTS, ENVIRONMENT),
- VARIABILITY AMONG MANUFACTURERS (CONSIDERED IN SOME CASES),
- VARIABILITY AMONG COMPONENT MODELS (CONSIDERED IN SOME CASES),
- VARIABILITY IN METHOD OF INSTALLATION, AND
- VARIABILITY IN COMPONENT AGE.

MAINTENANCE UNAVAILABILITY

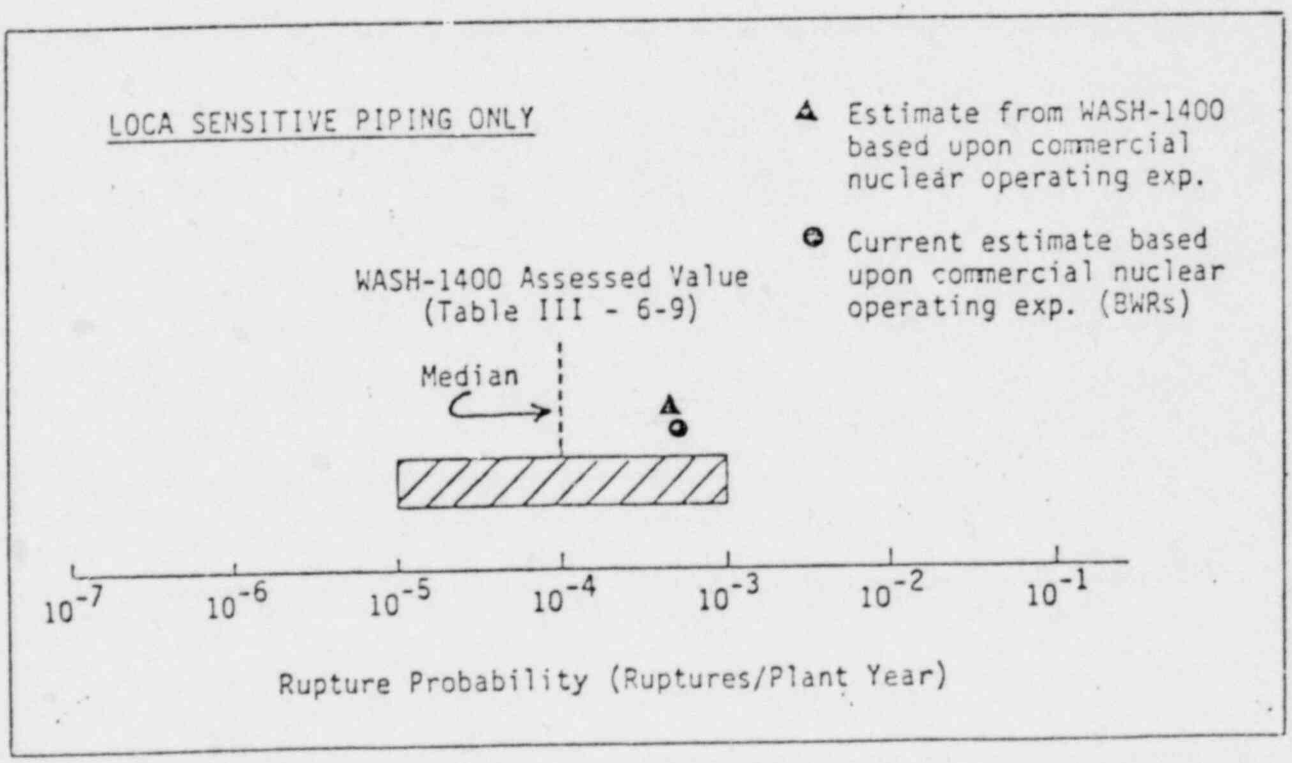
- USE GE ASSESSMENT BASED UPON BWR GENERAL OPERATING EXPERIENCE AND PEACH BOTTOM SITE-SPECIFIC DATA
- COMBINE IN PROPER LOGIC

NOTE THERE IS A SIGNIFICANT DIFFERENCE IN SYSTEM LEVEL MAINTENANCE UNAVAILABILITY BETWEEN CURRENT OPERATING EXPERIENCE AND THAT ASSUMED IN WASH-1400.

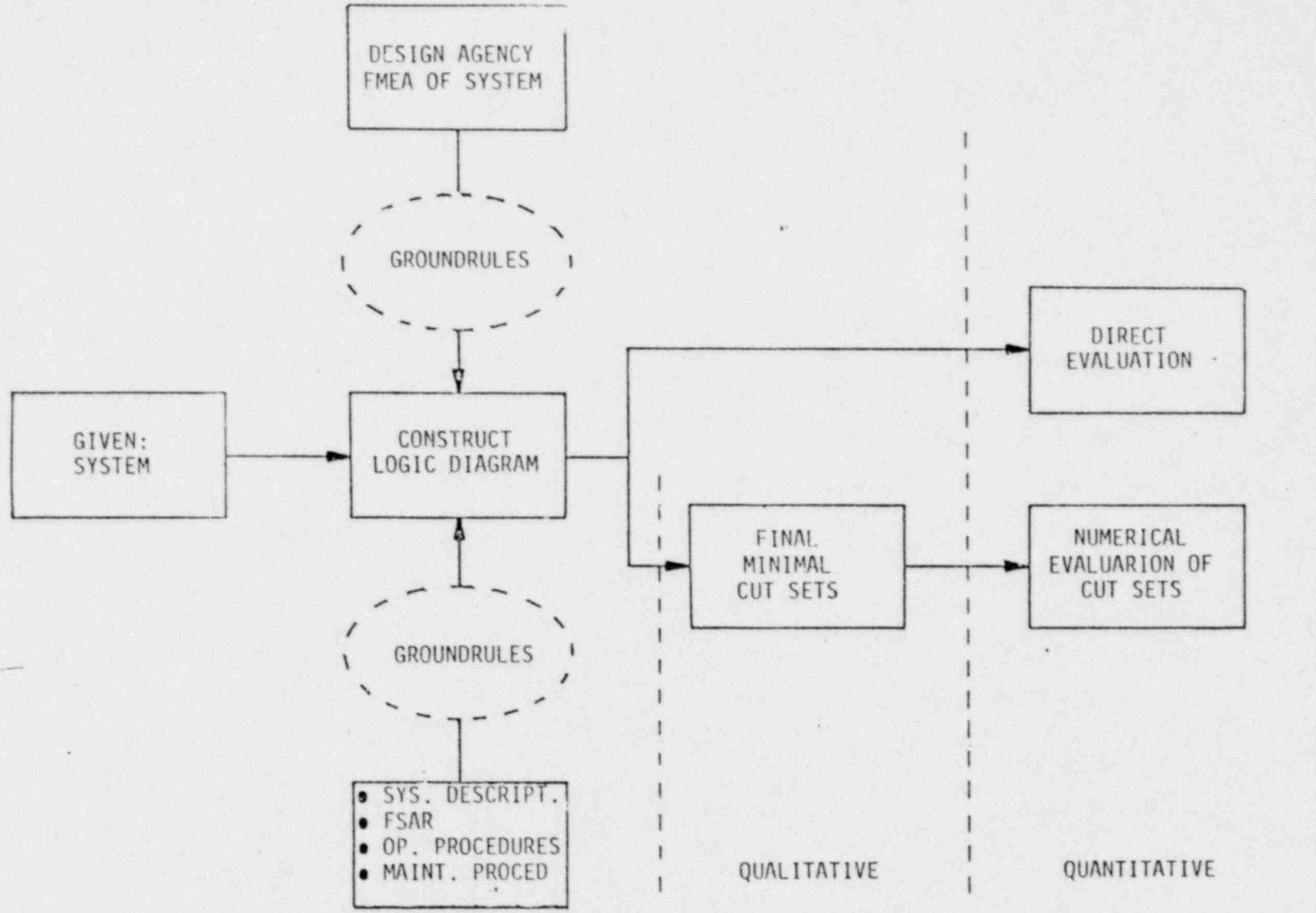
HOWEVER, BECAUSE THE LCOs REQUIRE A MINIMUM COMPLEMENT OF SYSTEMS AVAILABLE AT ALL TIMES THE NET IMPACT OF THE CHANGE ON CALCULATED CORE MELT FREQUENCY IS SMALL.



Comparison of Evaluated Rupture Probabilities for Pipe to Estimate Nuclear Power Plant Pipe Rupture Probabilities



Estimates of LOCA Initiated by A Large Pipe Break



Fault Tree Evaluation: Quantitative or Qualitative

Comparison of System Level Failure Probabilities (Point Estimate)

	Limerick (Mean)	Peach Bottom (Median)
Scram System	$3x10^{-5}$	$1x10^{-5}$
High Pressure Injection		
HPCI	$8x10^{-2}$	$9.8x10^{-2}$
RCIC	$7x10^{-2}$	$8x10^{-2}$
FW	$2x10^{-2}$ - 0.3*	$2x10^{-2}$ - .2*
Depressurization	$2.5x10^{-3}$	$5x10^{-3}$
Low Pressure Injection		
LPCI	$1.7x10^{-3}$ } $9x10^{-5}$	$1.5x10^{-2}$ } $2x10^{-4}$
CS	$6.2x10^{-3}$ }	—
Condensate	1.0	1.0
Standby Liquid Control	$4x10^{-3}$ (automated)	0.1
Emergency Service Water	$2x10^{-4}$	$1x10^{-4}$
Service Water	$2.3x10^{-4}$	$4x10^{-4}$
Containment Heat Removal		
RHR	$4.0x10^{-4}$	$2.3x10^{-4}$
PCS	$5x10^{-3}$†	$7x10^{-3}$**
RCIC in the Steam Cond. Mode	.1	NA

* Sequence dependent

** Estimated based upon operating experience information

† Taken from GE operating experience information

BENCHMARK COMPARISON
OF BWR/4 FAULT TREE MODEL
AGAINST AVAILABLE DATA
(RCIC AND HPCI ONLY)

SUMMARY OF HPCI/RCIC FIELD DATA REPORTED
TO GENERAL ELECTRIC AS OF 10/24/79

PLANT	HPCI		RCIC	
	ATTEMPTS TO START	FAILURES TO START	ATTEMPTS TO START	FAILURES TO START
A	99	1	96	0
B	96	1	88	3
C	100	11	100	4
D	149	26	125	17
D	160	23	270	31
E	105	4	92	7
E	106	4	100	10
F	151	1	184	2
TOTAL	966	71	1055	74

SUMMARY OF FAULT TREE CALCULATED FAILURE PROBABILITIES
FOR START AND RUN COMPARED WITH AVAILABLE DATA

SYSTEM	FAULT TREE MODEL CALCULATION	SURVEY
HPCI	4.6×10^{-2}	7.3×10^{-2}
RCIC	4.5×10^{-2}	7.0×10^{-2}

ESTIMATE THE COMBINED HIGH PRESSURE SYSTEM PERFORMANCE BASED UPON THE FAILURE PROBABILITY

- High Pressure System Unreliability (HPCI & RCIC + CRD) ~ $\boxed{2.3\times10^{-3}\text{/demand}}$

- Number of Demands on High Pressure Systems

+ Loss of FW	=	.27/yr
+ Loss of Offsite PWR	=	.16/yr
+ Loss of Aux. PWR	=	.04/yr
+ MSIV Closure	=	.79/yr
+ Loss of Condenser	=	.67/yr

= $\boxed{1.93\text{ d/year}}$

- Number of BWR Years of Experience ~ $\boxed{189\text{ years}}$

⟹ $\boxed{\text{Probability of a single occurrence of RCIC and HPCI being unavailable simultaneously upon demand}}$ ~ $\boxed{.84}$

HUMAN ERROR PROBABILITY

OVERVIEW:

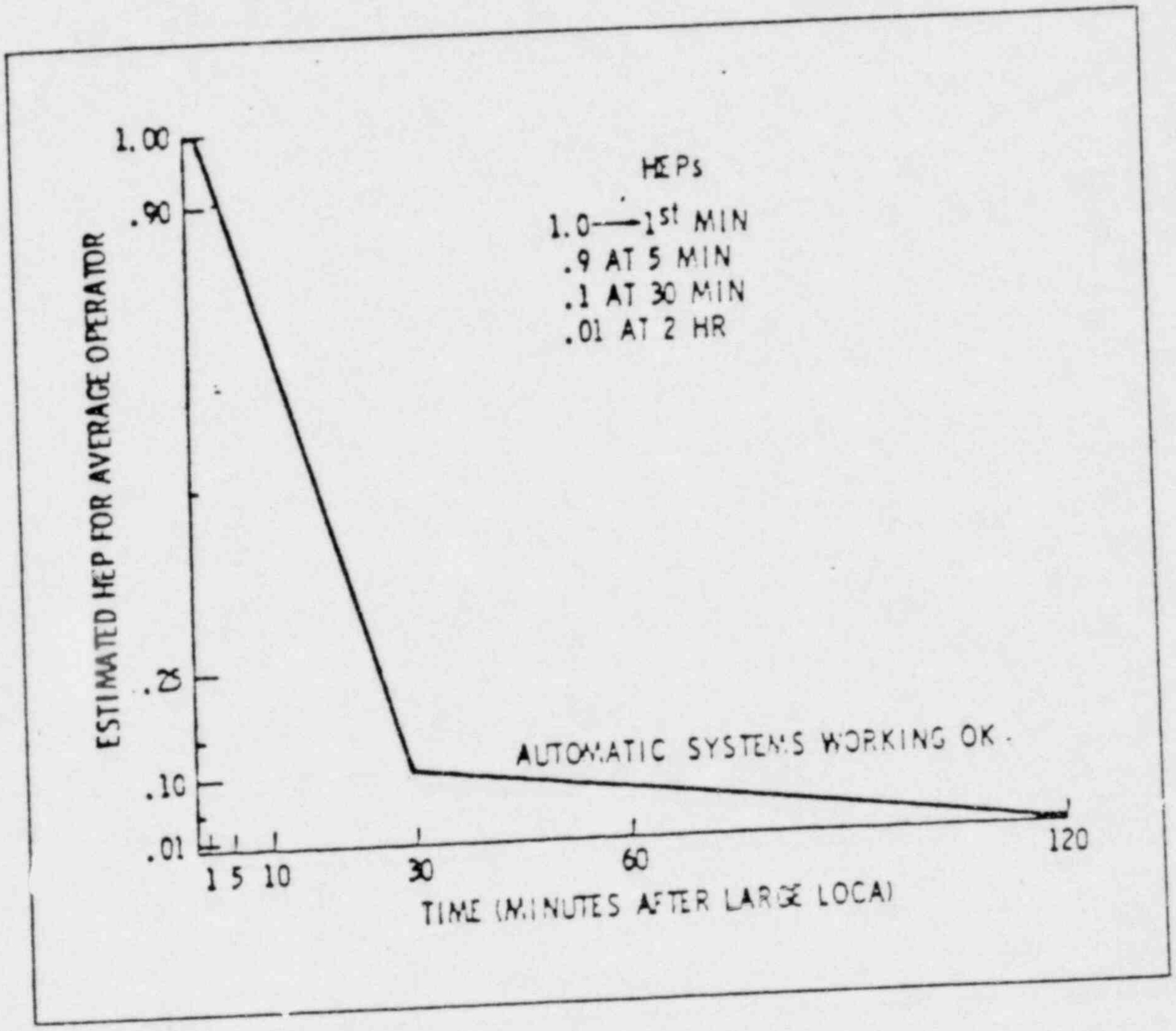
1. PLANT SPECIFIC OPERATING AND MAINTENANCE PROCEDURES DID NOT EXIST FOR LIMERICK DURING PRA PREPARATION

2. GE/BWR EMERGENCY GUIDELINES WERE USED FOR MANUAL OPERATOR RESPONSE DURING ACCIDENT CONDITIONS

3. NEED FOR OPERATOR RESPONSE FOR AUTOMATIC SYSTEM INITIATION IS TAKEN TO BE CONTINGENT UPON AUTOMATIC SYSTEM FAILURES (CONSERVATIVE ASSUMPTION).

4. SUMMARY OF HEPs USED IN THE LGS PRA:

REQUIRED ACTION	HEP	REF
OPENING REMOTE MANUAL VALVES	0.9	EST.
AUTO. SAFETY SYSTEM BACKUP INITIATION (30 MIN. HPCI, RCIC, LPCI, CS)	0.1	WASH-1400
RHR INITIATION (15 MIN.) (ATWS SEQUENCES)	0.01	SWAIN
DEPRESSURIZATION (30 MIN.)	0.002	SWAIN
VALVE ALIGNMENT DURING MAINT.	0.0001	SWAIN
RHR INITIATION (20 HRS.)	6×10^{-5}	NUS



Estimated Human Performance After a
Large LOCA

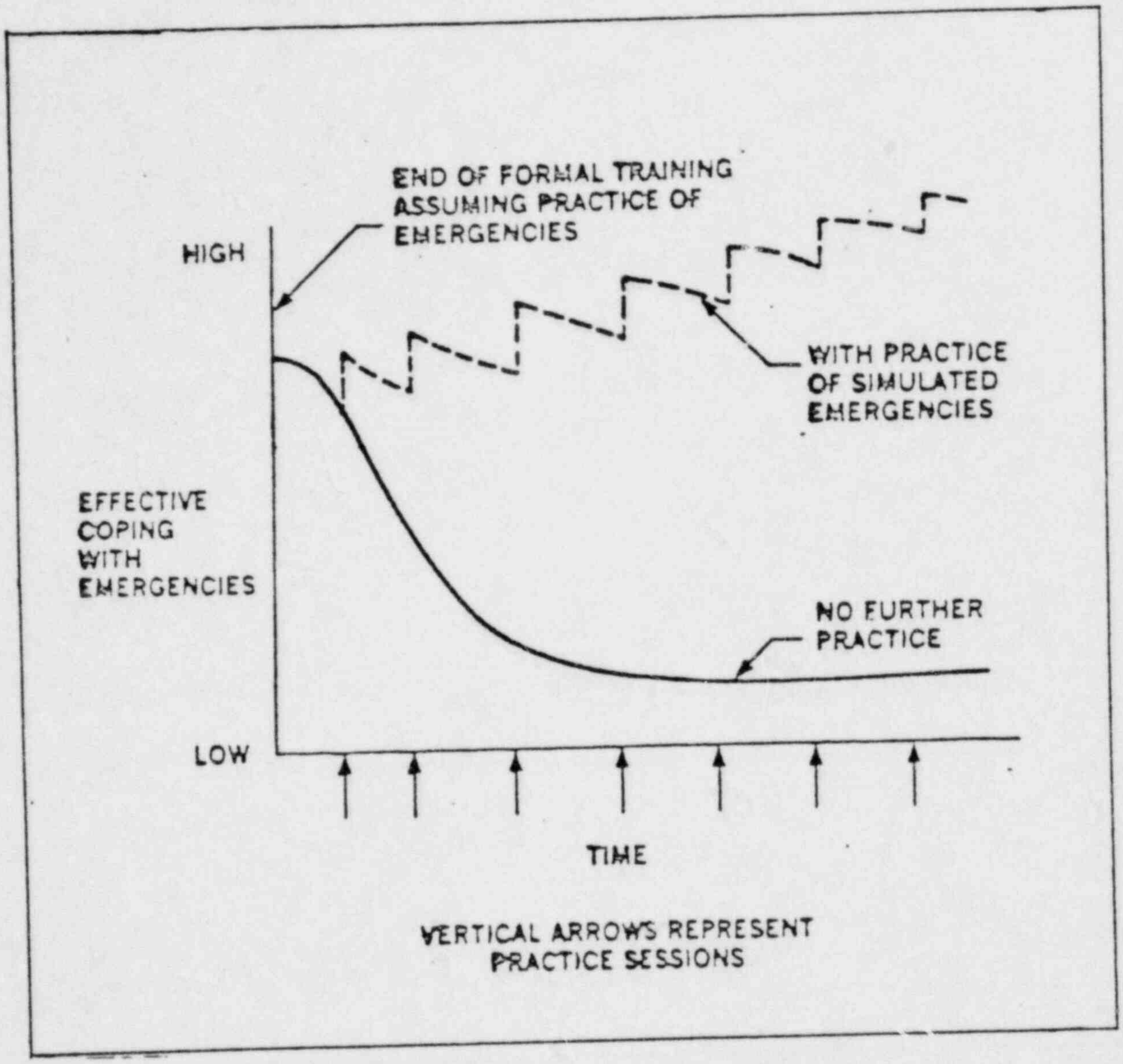
ESTIMATED PROBABILITIES OF FAILURE TO INITIATE AFWS*

At End Of X Minute	Situation without Dedicated Operator		
	Regular Operator	Shift Supervisor HEPs (Conditional)	Total Failure HEPs
5	.05	----	.05
15	.01	1.0 (HD)**	.01
30	.005	.4 (MD)***	.002
60	No change	No change	No change

*Lower and upper uncertainty bounds of a factor of 10 are assigned to each estimate in the "total" columns (C-1). All HEPs are rounded.

**High dependence assumed; Swain value of .5 of modified upwards to 1.0.

***High dependence assumed; Swain value of .15 of modified upwards to .002.



Qualitative Effects of Practice and No Practice on Maintenance of Emergency Skills (taken from Swain & Guttman)

UNCERTAINTIES: THE MAJOR OBJECTIVE OF THIS RISK ASSESSMENT IS TO DEVELOP A BEST ESTIMATE COMPLEMENTARY CUMULATIVE DISTRIBUTION FUNCTION (CCDF) FOR EARLY AND LATENT FATALITIES FOR THE LIMERICK PLANT. IN ADDITION, THE FOLLOWING CHARACTERIZATION OF UNCERTAINTIES IS PERFORMED:

1. UNCERTAINTIES FOR SELECTED DOMINANT SEQUENCE PROBABILITIES ARE GENERATED, USING A MONTE CARLO SIMULATION OF THE SYSTEM MODELS, AND THE INDIVIDUAL COMPONENT UNCERTAINTY DISTRIBUTION.

2. SUBJECTIVE CHARACTERIZATION OF CCDF UNCERTAINTY, INCLUDING THE UNCERTAINTIES IN:

   - SEQUENCE EVALUATIONS
   - IN-CORE RADIOACTIVE RELEASE PROCESSES
   - EX-PLANT CONSEQUENCE CALCULATIONS.

UNCERTAINTY ANALYSIS

- WASH-1400
- EPRI (NP-1130)
- LIMERICK (ESTIMATE)
- SUMMARY OF KEY AREAS OF UNCERTAINTY
- SEQUENCE PROBABILITY MONTE CARLO SIMULATION

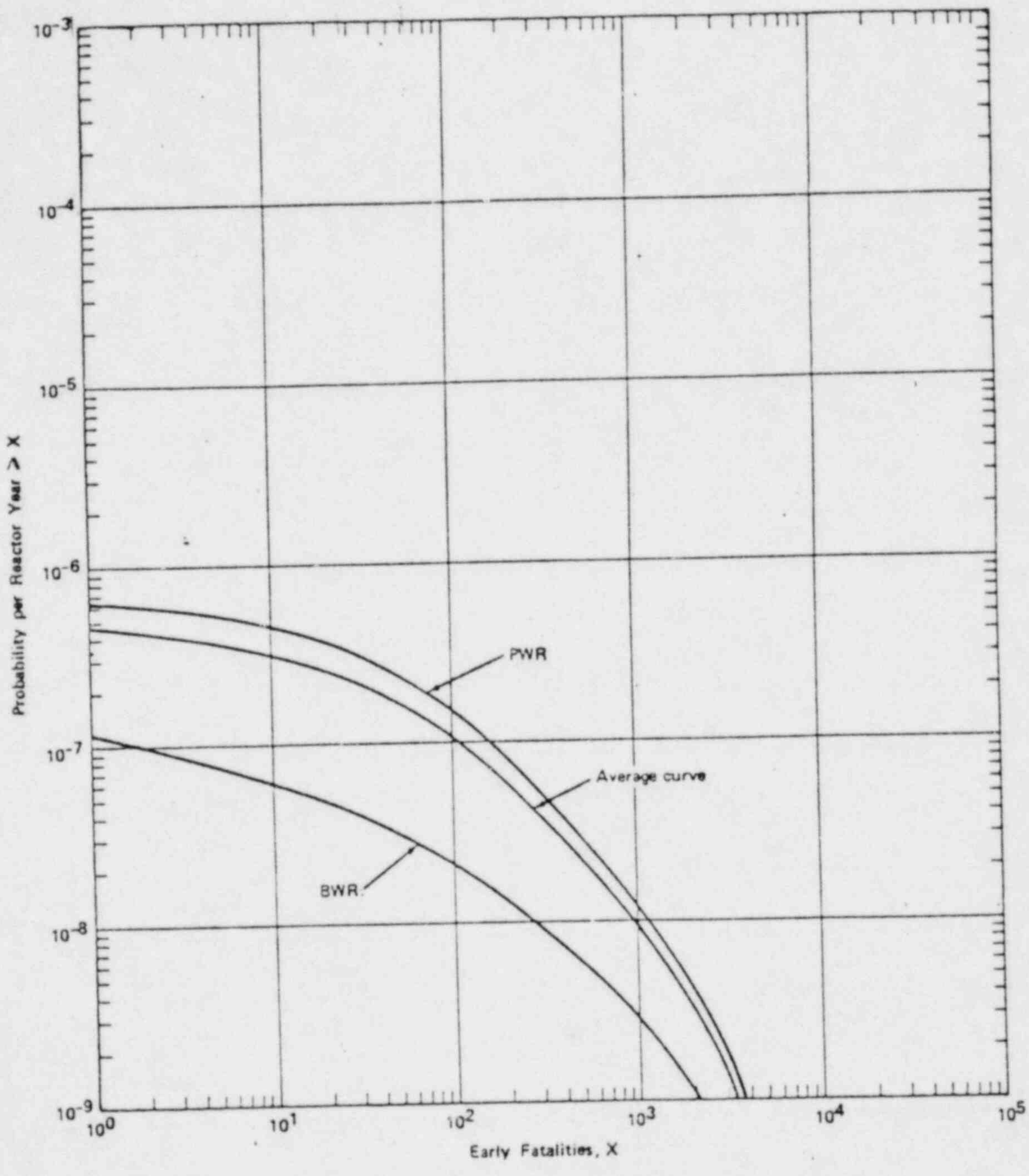


FIGURE 5-3 Probability Distribution for Early Fatalities per Reactor Year

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

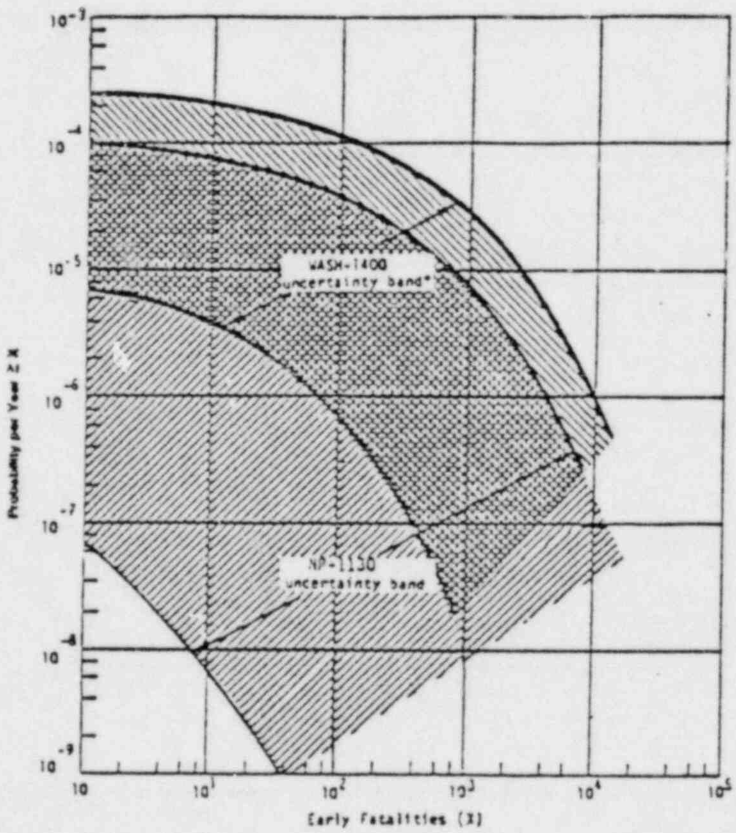
FIGURE 5-4 Probability Distribution for Early Illness per Reactor Year

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

Table 3.8.1

SHIFTS IN ACCIDENT FREQUENCY AND CONSEQUENCES
FROM EPRI NP-1130

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



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

Figure 3.8.1 Probability Distribution for Early Fatalities Per Year for 100 Reactors (From EPRI NP-1130)

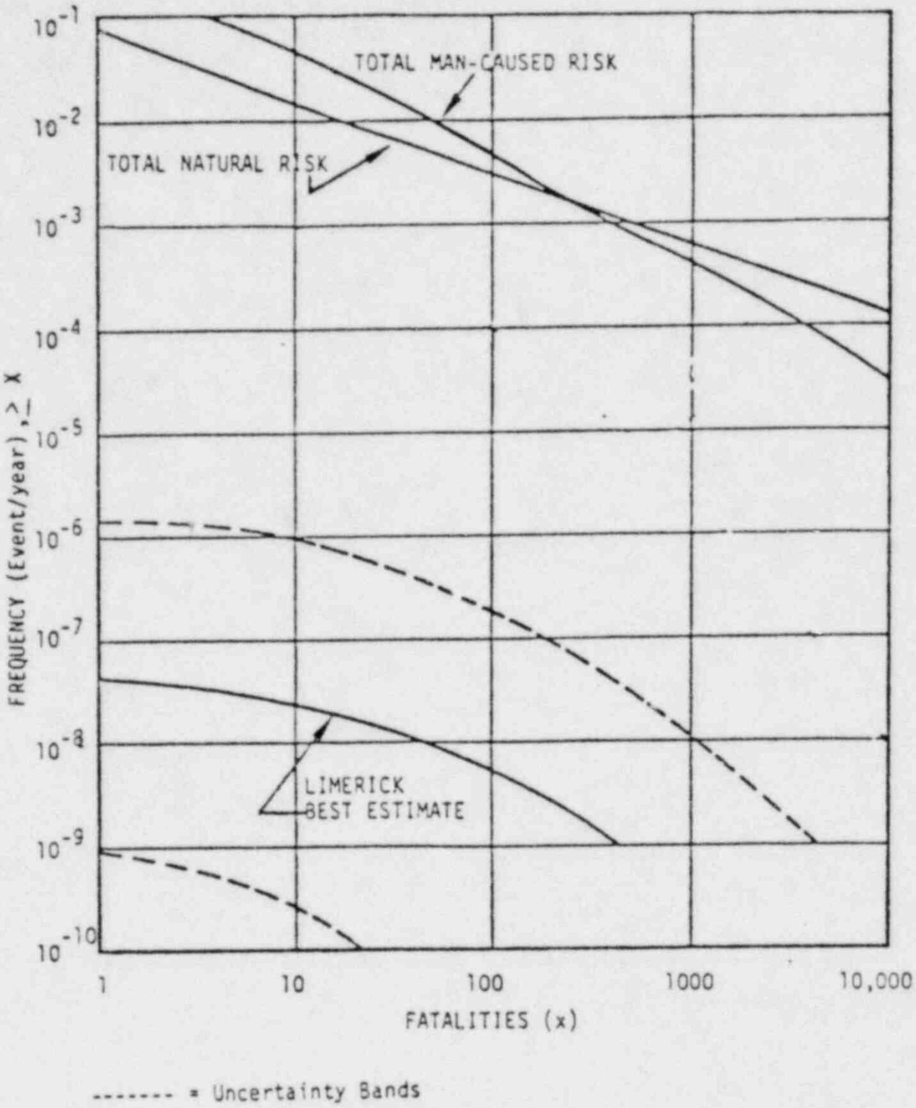


Figure 3.8.3 Summary of Risks Assumed by the Population Surrounding the LGS for Early Fatalities with the Estimated Uncertainty Band.

UNCERTAINTIES

- THE CHARCTERIZATIONS OF COMPONENT LEVEL INPUT DATA UNCERTAINTIES (INCLUDING HUMAN ERROR) ARE NOT WELL-DEFINED. THE PROBABILITY DISTRIBUTION FUNCTION IS NOT IN GENERAL KNOWN, ALTHOUGH LOG NORMAL SHAPES HAVE TYPICALLY BEEN ASSIGNED IN PAST PRAs WHERE UNCERTAINTY ESTIMATES HAVE BEEN ATTEMPTED.

- THE UNCERTAINTIES IN THE CONSEQUENCE CALCULATIONS ARE EVEN LESS WELL UNDERSTOOD. THERE DOES NOT APPEAR TO BE A COMPLETE EVALUATION OF THE EXTENT OF POTENTIAL UNCERTAINTIES IN THIS AREA.

- PRESENTLY, NO FORMAL MECHANISM FOR PROPAGATING UNCERTAINTIES THROUGH THE ENTIRE PRA (I.E., PROBABILITIES AND CONSEQUENCES) IS IN USE. THEREFORE, CHARACTERIZATION OF UNCERTAINTIES ON THE CALCULATED CCDFs HAS THUS FAR BEEN SUBJECTIVE.

Table 3-2

PARTIAL PRA CHECKLIST FOR COMPLETENESS AND APPLICABILITY

SUBJECT	DISCUSSION
METHODOLOGY	
Elimination of Smoothing of Release Categories	The Reactor Safety Study used a discrete set of release categories and then assigned probabilities to adjacent categories based upon uncertainty in sequence categorization. The use of a larger number of accident release categories provides greater specificity in consequence definition. Based on more precise definition in accident sequence releases, the need for smoothing may be unnecessary.
No Repair of Failed Systems	In the Reactor Safety Study very little or no credit is given to the operator for restoring a system to service if it is failed or in maintenance.
Time Phasing	Systems required for operation are considered unavailable if they fail probabilistically. However, in some cases these systems can be restored within a rather short time. Repair of systems is generally not included; however, for specific systems for which explicit operator procedures exist and adverse environment is not considered a problem, repair may be included in the modeling and quantification.
Duration of System Operation Required	The length of time required for system operation is set by the definition of successful end state used in the analysis, i.e., stable conditions, hot shutdown, or cold shutdown. The time required to reach these specified conditions directly affects the evaluated conditional input probabilities to the logic model.

Table 3-2 (Cont'd)

SUBJECT	DISCUSSION
<u>METHODOLOGY (cont'd)</u>	
Completeness	Finite Number of Sequences Examined: There are an infinite number of possible accident sequences. However, due to human and code limitations, only a finite number of these can be examined. Since this set of sequences represents an approximation to the spectrum of conceivable accident sequences, sequences are often grouped to conservatively incorporate the events and consequences which may develop. This and other systematic procedures help to make sure that all significant risk contributors are included in the analysis.
Common Cause	Methods to Ensure Common Cause Events are Included: Because common cause failures entail a wide spectrum of possibilities and enter into all areas of modeling and analysis, common cause failures cannot be isolated as a separate study, but instead must be considered throughout all the modeling and quantification steps. Possible common cause events are handled both at the event tree level, allowing for the failure of one system to cause the failure of another, and at the fault tree level, accounting for similar components to fail in a dependent manner. Common cause methods used in PRA methodology include: - Functional dependencies among systems in event trees, - System fault tree construction extended to include the common hardware dependencies among systems, and - Common human interfaces, test and maintenance among similar components

Table 3-2 (Cont'd)

SUBJECT	DESCRIPTION
METHODOLOGY (cont'd)	
Binary Nature of Fault Trees	Fault trees, as used in PRAs, are inherently binary. The trees model only the success or failure of components or events; no partial successes are usually considered. Because of this, assumptions must be made as to what the "success" of a component refers to. Often, it is conservatively assumed that any partial failure is a complete failure.
Hardware	Under the limitations of time and money, assumptions are sometimes made regarding the operability of certain systems. Specifically, non-safety-related systems may be given little credit for their accident mitigating potential. These types of assumptions must be clearly noted.
DATA	
Data Available	The quantification of event tree/fault tree models should be based upon the best available data base applicable to each specific component or system. However, the availability of data appropriate for such quantification is limited on both a generic basis and a plant-specific basis. Generally, it is useful to combine several data sources; however, variabilities that are usually not in the input data may include: - component size, application, or environment, - manufacturers, and - component ages. Newly constructed plants will not have plant specific data; therefore, the use of some generic data source may be required.

Table 3-2 (Cont'd)

SUBJECT	DESCRIPTION
DATA (cont'd)	
Plant/Component Age	Data for plants with a long operating history are not available. Therefore component failure rate data is in general an average of failure rates over the initial 5 to 10 years of plant operation. This average is anticipated to be representative even during the wearout phase of plant life.
ATWS Frequency	Because operating experience is insufficient to adequately characterize the potential for ATWS, there has been a great deal of speculation. Currently the frequency used is that derived from Reference (3). The frequency could be approximately 10 times higher or 10 times lower depending upon one's assumptions concerning precursors and rectification.
Diesel Failure Rates	A single diesel failure rate to start and run for relatively short periods of time are well documented. Failure of two or three diesels is subject to larger uncertainty but can be estimated using available data. Failure of four diesels is highly uncertain.
PCS Availability	A key system used in normal plant shutdown is the Power Conversion System. However, very little data exists to characterize the PCS reliability under a wide variety of the conditions under which it may be required.
Constant failure rate assumption, e.g. pipe failures, instrument failures	The failure rate is generally 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, and main stream isolation valves.

Table 3-2 (Cont'd)

SUBJECT	DESCRIPTION
DATA (cont'd)	
Use of log-normal to describe the frequency distribution of failure rates for certain components	Log-normal distributions are generally assumed to describe component failure probability distributions. However, sufficient data does not exist to justify this assumption.
Human error probabilities	Data cited in the Reactor Safety Study (1) and the Human Reliability Handbook (4) are generally used; however, very little actual data exists to support these evaluations.
Radionuclide Decontaminations Factors (DF)	Decontamination factors have a significant impact on the calculated economic consequences of radionuclide releases to the environment. There is a significant amount of data on obtainable decontamination factors from small scale tests; however, very little well-documented data exists on large scale decontamination efforts. Therefore, the justification of DFs used in the analysis is necessary.
Meteorological Data	One of the key input parameters in calculating the distribution of radionuclide in the environment is the meteorological data. Consequence assessment codes such as CRAC or CRACIT (see Section 7) make use of sampling schemes for weather patterns from a single year of data. The meteorological data input and sampling scheme coupled together can affect the calculated risk. Therefore, it is useful to have a discussion of the effects anticipated if different years of meteorological data are used.
CONTAINMENT	
Core Melt (General)	Assumptions regarding the timing of events and the physical proceses, e.g. particle fragmentation, core slumping, reactor pressure vessel melt through, concrete interaction involved in the postulated core melt are modeled in a simplistic fashion. The implication of these simplifications should be identified.

Table 3-2 (Cont'd)

SUBJECT	DESCRIPTION
<u>CONTAINMENT (cont'd)</u>	
Reactor Pressure Vessel (RPV) Failure	The manner in which the RPV fails during a postulated core melt is highly uncertain. One method assumes RPV failure from creep rupture - 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 failure modes assume failure from melting, but the manner of melting is also uncertain. If convection currents are assumed, then hot spots would tend to occur on the edges of the vessel and the entire bottom head would fail. If there are not any convection currents, heat transfer would mainly be from conduction and hot spots would occur at the bottom and the entire bottom head would not fail. These last two methods both consider pressure relief during RPV melt by assuming melt through the bottom head pentrations.
Molten Core Reaction	An area of large uncertainty is the manner in which the molten core will act after it fails the RPV. It is uncertain whether the molten core will: • drop onto the floor below in one coherent mass, • fragment and disperse around containment from blowdown of RPV if a large blowdown force occurs, • stay localized below the vessel, or • react with water present causing steam explosion(s).
Molten Core	In some of the dominant sequences, the oxide layer may be predicted to freeze. The implication of this frozen layer is not well known. It is thought that once the oxide layer freezes the vaporization release stops. Radionuclides are released during vaporization; the gases (non-condensibles) generated at the core/concrete

Table 3-2 (Cont'd)

SUBJECT	DESCRIPTION
CONTAINMENT (cont'd)	
Molten Core (cont'd)	interface bubble up just through the metal layer and then through the oxide layer (the oxide layer being on top). The fission products are contained in the oxide layer and are carried away by the gases crossing through. If the oxide layer is frozen, the gases cannot bubble through and escape to containment at the edges of the core and the fission products do not get transported from the melt.
Hydrogen Generation	The amount and timing of hydrogen generation during the core melt process may have a significant effect on the quantity of radionuclides released. Therefore, the models, assumptions, and results should be discussed to indicate the impact on accident sequence release fractions.
Steam Explosion	The probability of a steam explosion (in-vessel or in-containment) is the subject of wide controversy. The probabilities used in the Reactor Safety Study are expected to be an upper bound. More recent PRA's tend to use lower conditional probabilities for steam explosion based upon Sandia experiments.
Hydrogen Explosion	It is considered possible that a hydrogen explosion of sufficient magnitude to result in radionuclide releases comparable to in-vessel steam explosion may occur if the explosive mixture of gases is brought together with an ignition source.
RELEASE FRACTION REACT/CORRAL MODEL:	
Radionuclide Release	The CORRAL code uses the Reactor Safety Study values for best estimate percent releases for each group of radionuclide. These values are uncertain and recent experimental data indicate the larger numbers are conservative and the lower estimates are too low. Reference (5) offers

Table 3-2 (Cont'd)

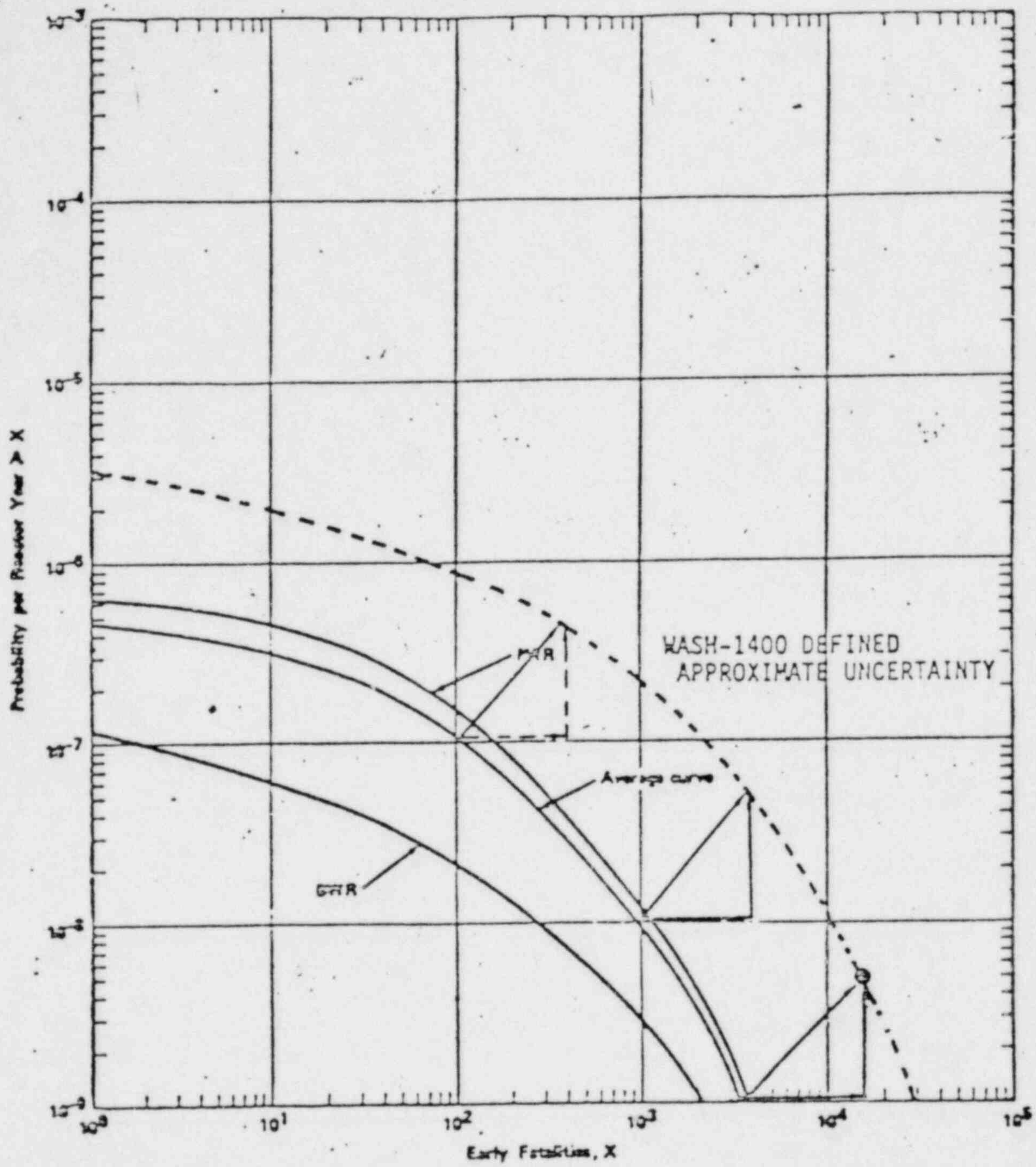
SUBJECT	DESCRIPTION
RELEASE FRACTION REACT/CORRAL MODEL: (cont'd)	
Radionuclide Release	some additional insight into this area where data is sparse and attempts at modeling are under development.
Containment Integrity at High Temperatures and Pressure	Containment integrity may be assumed to be maintained with the temperatures > 250 °F and at internal pressures above the design limits over extended periods of time. These conditions are beyond the design limits of containment, therefore current best estimates may be optimistic.
EX-PLANT EFFECTS	
Radionuclide transport and dispersion	The model used to define the plume of radioactive material as it traverses large distances (>20 miles) has not been verified experimentally.
Evacuation model	The calculation of early fatalities is very sensitive to the assumption on evacuation of the population. Many aspects of evacuation are untested.
Shielding effectiveness	There exists a significant variation in the type of structure available in the environs of a power plant for sheltering of the population and the shielding offered by these structures.
Dose-Mortality response curve	The applicability of given dose mortality response curve is strongly dependent upon the health of a person and the degree of medical attention he receives once exposed. The calculation of early fatalities is very sensitive to the assumption made in selecting a response curve.

Table 3-2 (Cont'd)

SUBJECT	DESCRIPTION
EX-PLANT EFFECTS (cont'd)	
Threshold effect on latent cancers	There has been a long-standing controversy on the existence of a dose threshold; that is, a threshold below which latent cancer risk to a person is zero.
Duration of radionuclide release	The release of radionuclides calculated by CORRAL to escape with each containment failure mode and accident sequence is assumed to occur over a 30 minute period. This permits modeling the release as a puff. If the release is of longer duration it is possible, because of wind direction changes, that concentrations of radionuclides will be lower than if a "puff" release is assumed.

UNCERTAINTY IN SEQUENCE FREQUENCY

- IDENTIFY THE DOMINANT SEQUENCE IN EACH RADIONUCLIDE RELEASE CLASS
- DETERMINE THE UNCERTAINTIES
  - INITIATOR: OPERATING EXPERIENCE DATA
  - COMPONENT INPUTS: WASH-1400
- MONTE CARLO SIMULATION OF DOMINANT SEQUENCES
- CALCULATE THE UNCERTAINTY IN THE FREQUENCY OF THE SEQUENCE.



PROBABILITY DISTRIBUTION FOR EARLY FATALITIES PER REACTOR YEAR

NOTE: APPROXIMATE UNCERTAINTIES ARE ESTIMATED TO BE REPRESENTED BY FACTORS OF 1/4 AND 4 ON CONSEQUENCE MAGNITUDES AND BY FACTORS OF 1/5 AND 5 ON PROBABILITIES.

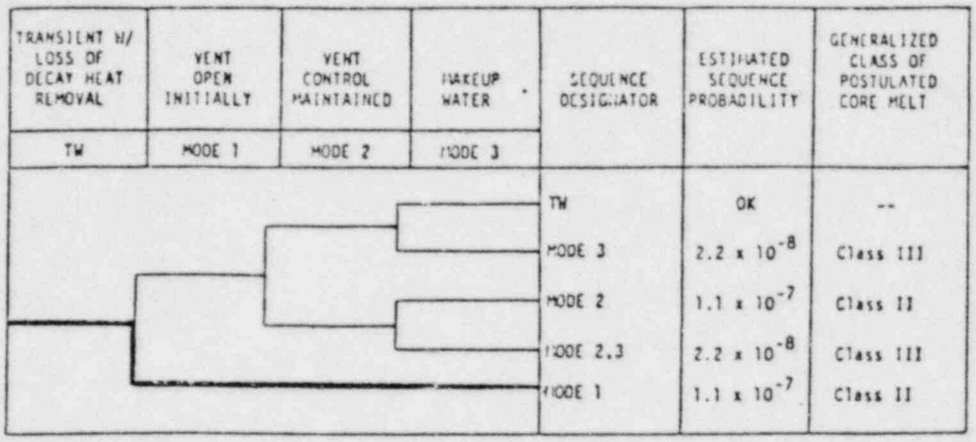


Figure 3.4.11. "Bridge" Event Tree Providing the Link Between Identified Accident Sequences which Result in Containment Overpressure and the Containment Event Sequences Following Core Melt.

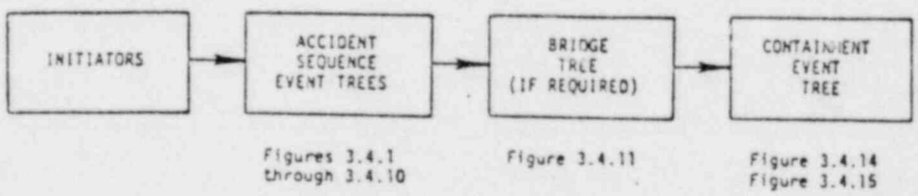


Figure 3.4.12. Flow Chart of the Event Trees Used to Define Accident Sequences.

LIMERICK SITE COMPARISON
(Relative to WASH 1400)

- Realistic Site
- Higher Population
- Different Weather
- Same Evacuation Model

8547 46

LIMERICK DESIGN COMPARISON
(Relative to WASH 1400 BWR)

- MK II Reinforced Concrete Steel-lined Containment
- Larger Standby Gas Treatment System
- Containment Overpressure Relief
- More and Improved Safety/Relief Valves
- Improved Piping
- Improved Shutdown System
- Spray Pond for Emergency Cooling Water
- Improved Emergency Pump Capability
- Four Dedicated Emergency Diesel Generators
- More Reliable Offsite Power

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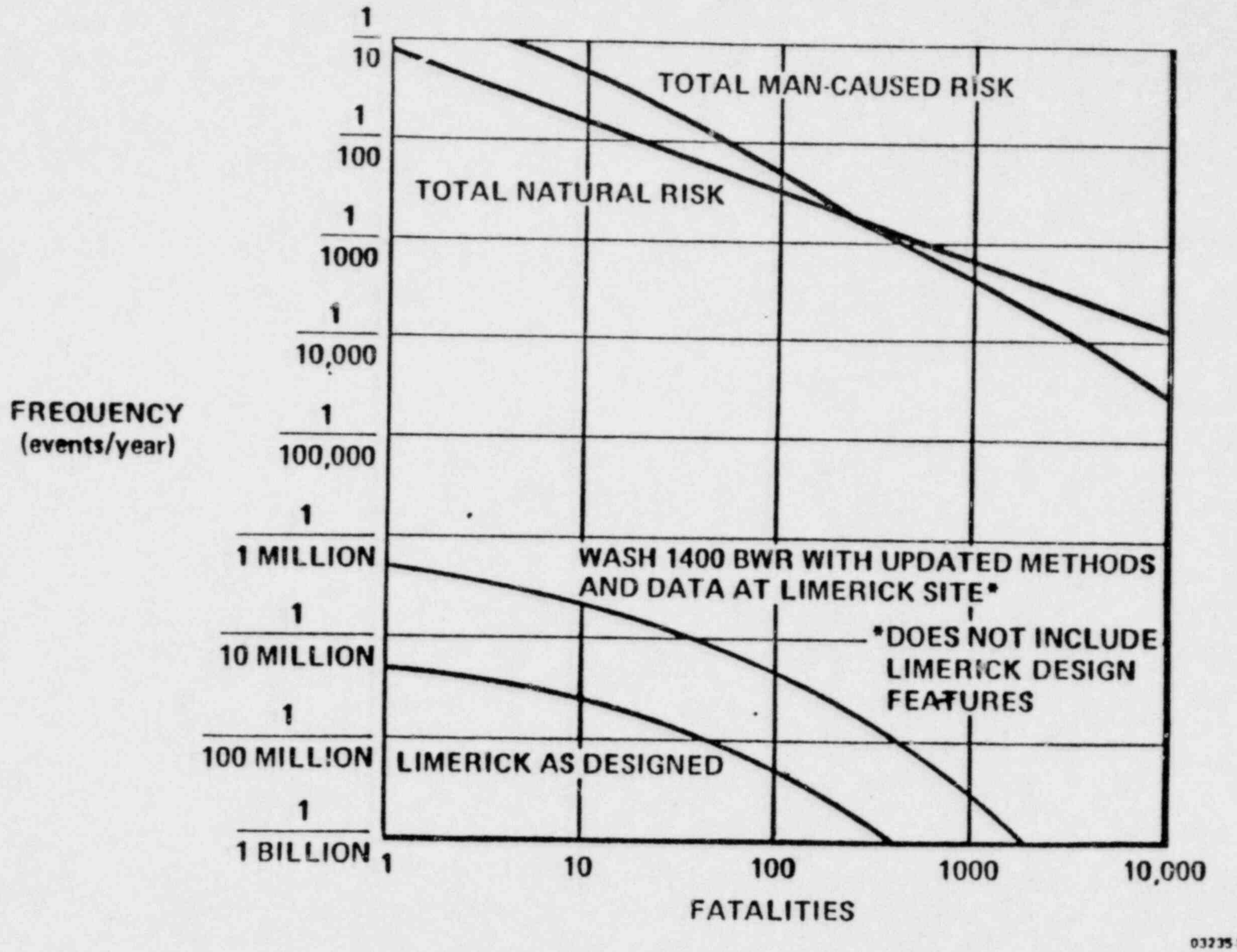
Limerick Preliminary Risk Assessment
Design Features Comparison



LIMERICK DATA COMPARISON
(Relative to WASH 1400)

- Larger Data Base
- Initiating Frequencies
- Equipment Reliability
- Maintenance Times

8547 47

LIMERICK METHODOLOGY COMPARISON
(Relative to WASH 1400)

- Improved Computer Models
- More Comprehensive Treatment of Transients
- Updated Decontamination Factors
- Updated Treatment of Hydrogen/Steam Physics
- Use of Emergency Operator Guidelines

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SUMMARY OF DIFFERENCES
(Relative to WASH 1400)

- Site Effect - Up
- Methods and Data Effect - Mixed
- Limerick Design Effect - Down
- Net Effect - Down

8547.50

RESULTS

- Limerick Site

  Higher Population

- Limerick Plant

  Better Than WASH 1400 Plant

- Limerick Plant-Site Together

  Lower Risk Than WASH 1400 BWR

- Limerick Generating Station

  Presents Minimal Risk to Public

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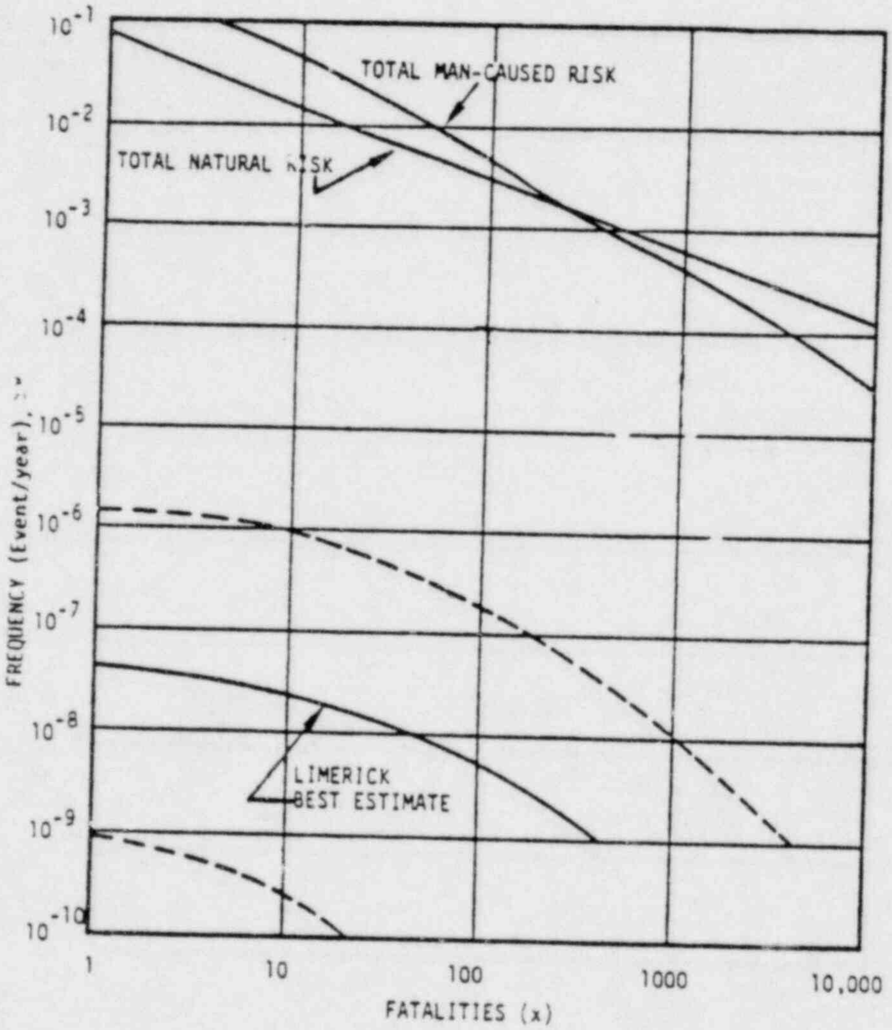
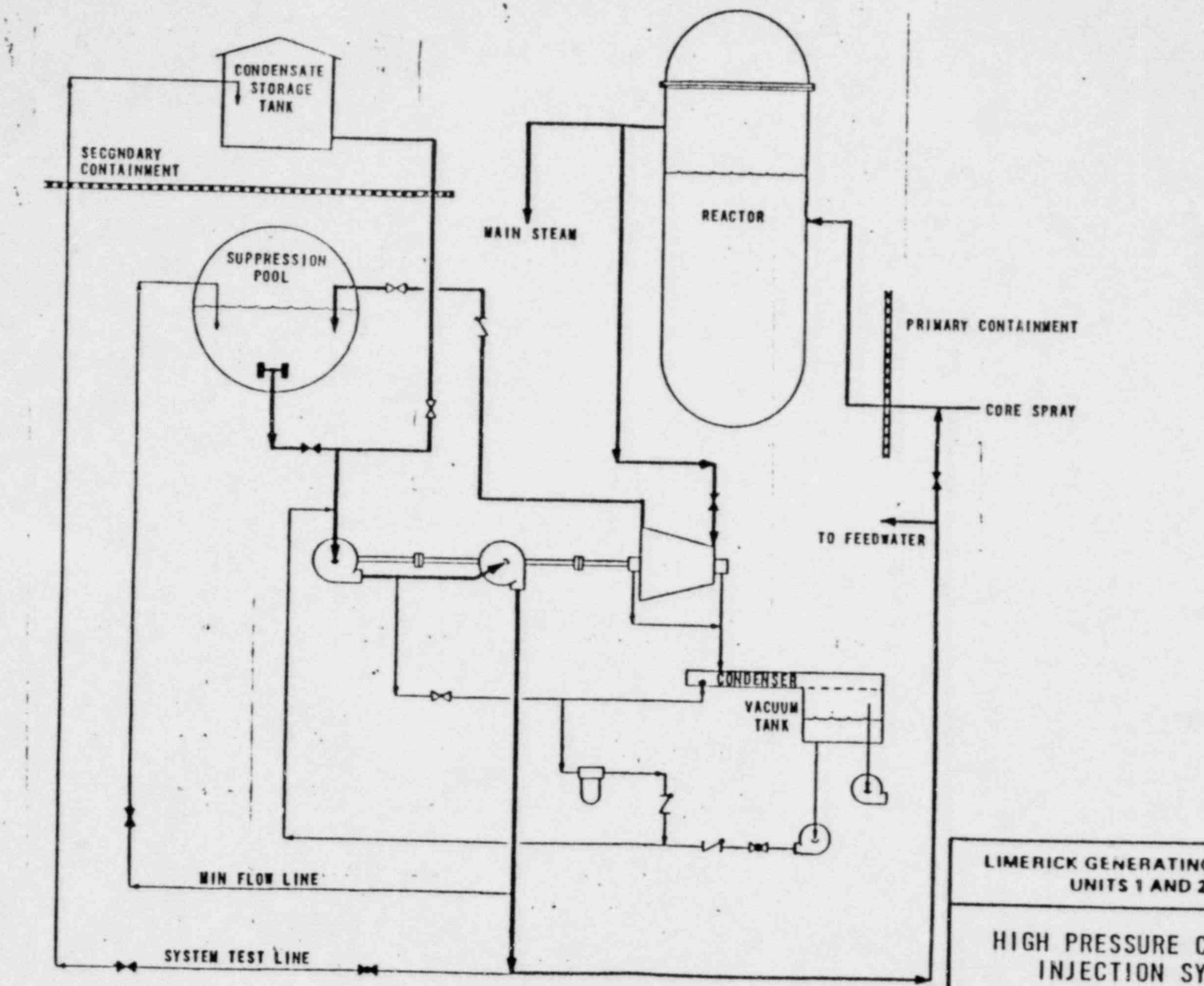
Uncertainty Bands on Limerick
Best Estimate CCDF

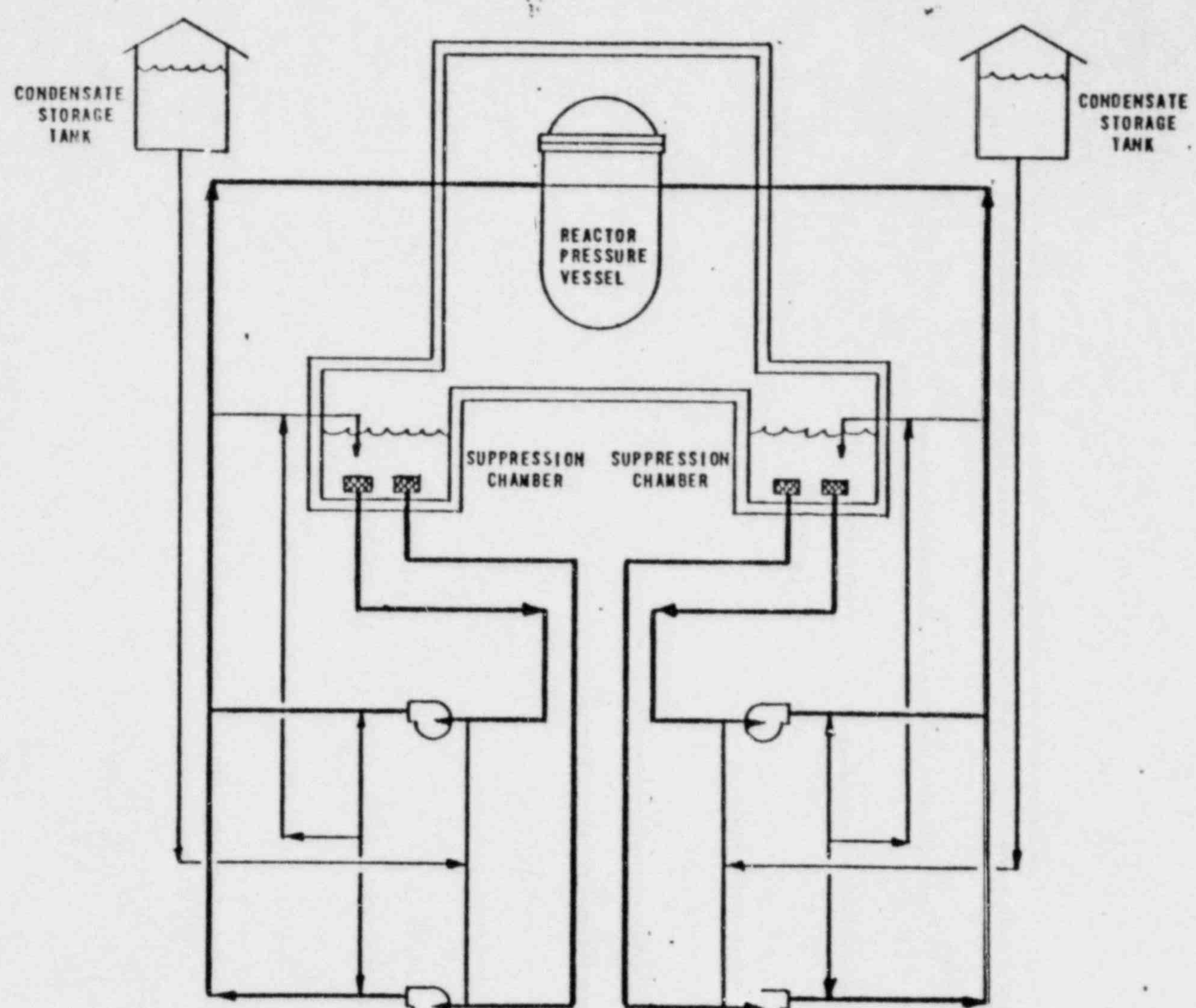


------- = Uncertainty Bands



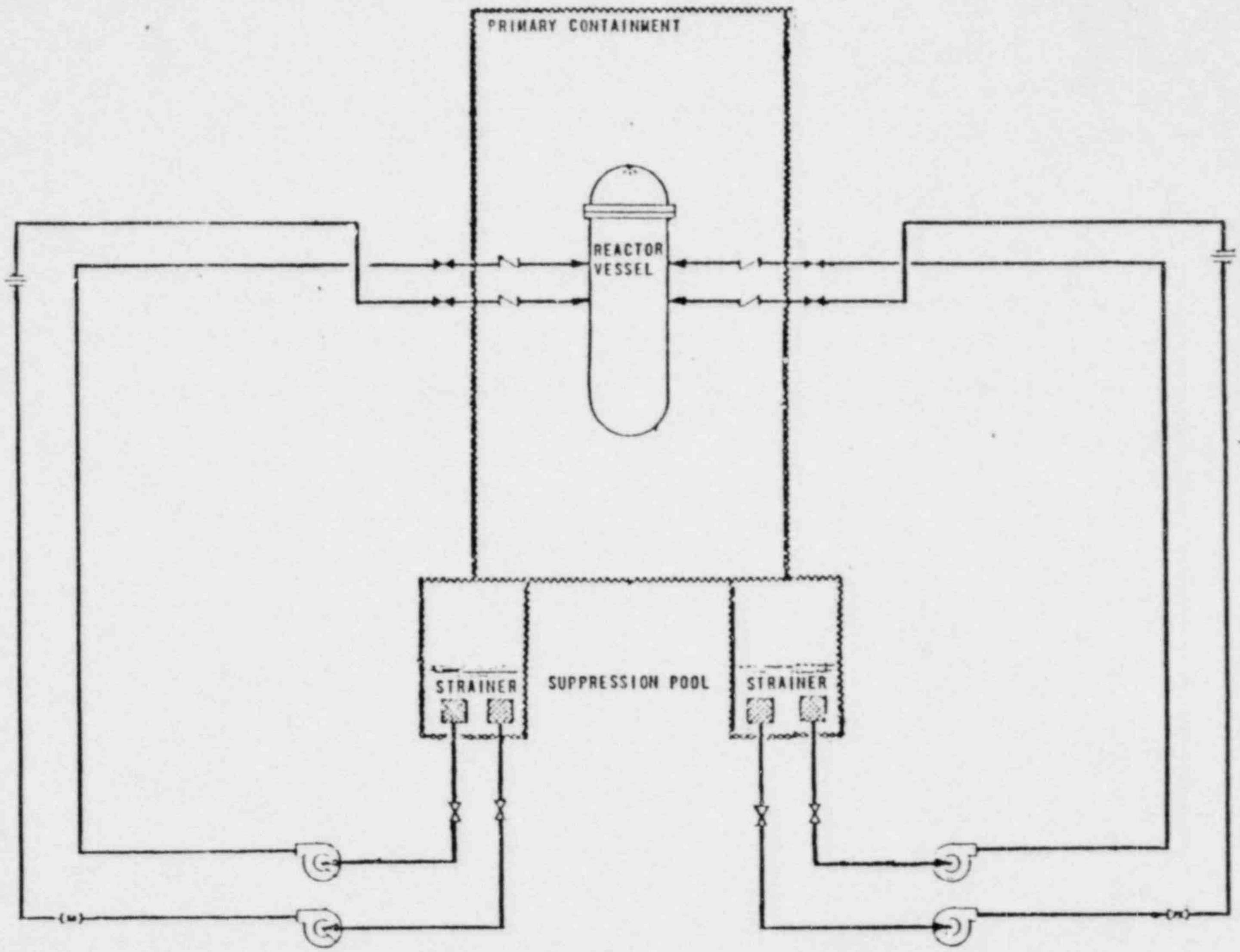
LIMERICK GENERATING STATION
UNITS 1 AND 2

HIGH PRESSURE COOLANT INJECTION SYSTEM



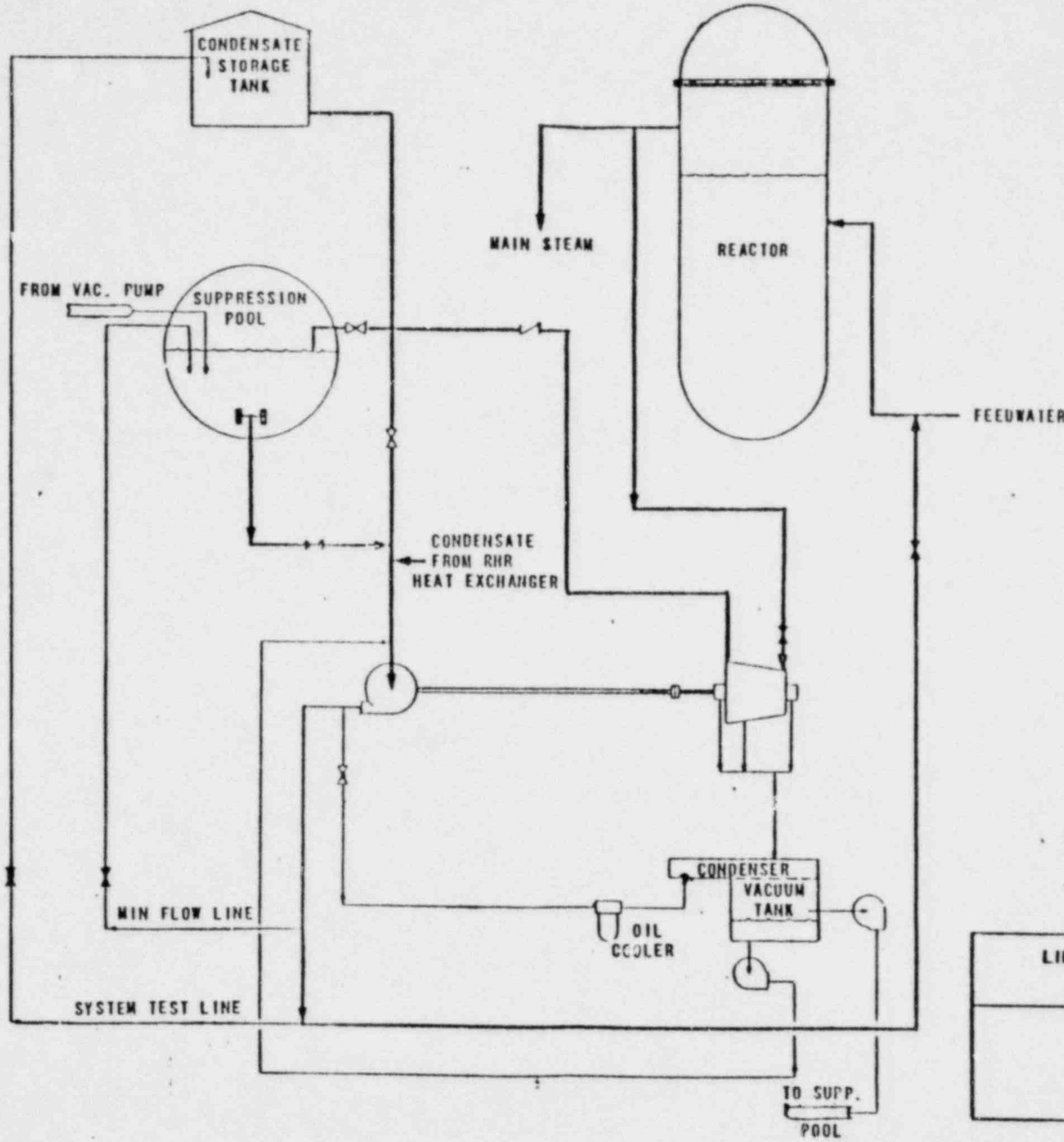
LIMERICK GENERATING STATION
UNITS 1 AND 2

CORE SPRAY SYSTEM



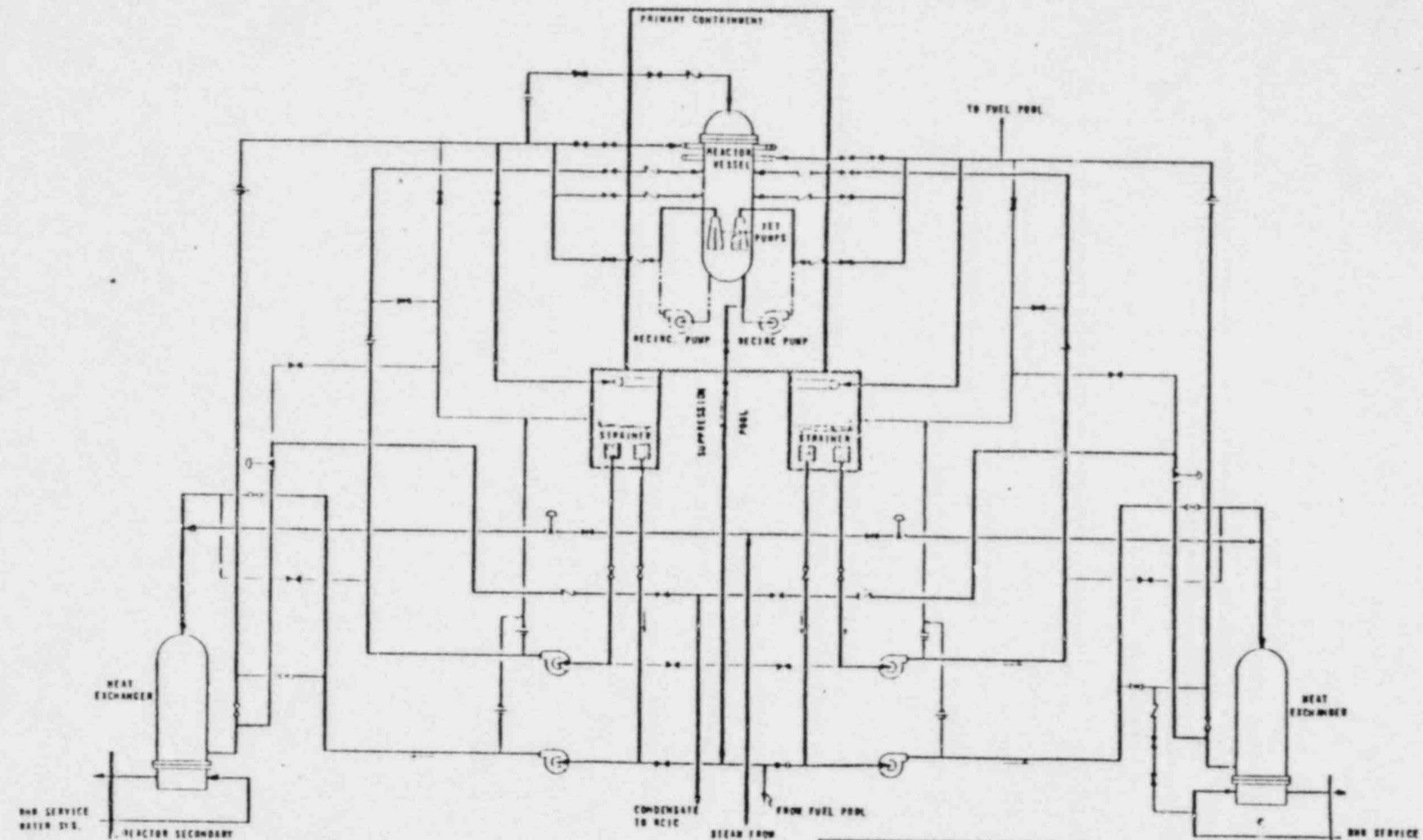
LIMERICK GENERATING STATION
UNITS 1 AND 2

RESIDUAL HEAT REMOVAL
SYSTEM (LPCI)



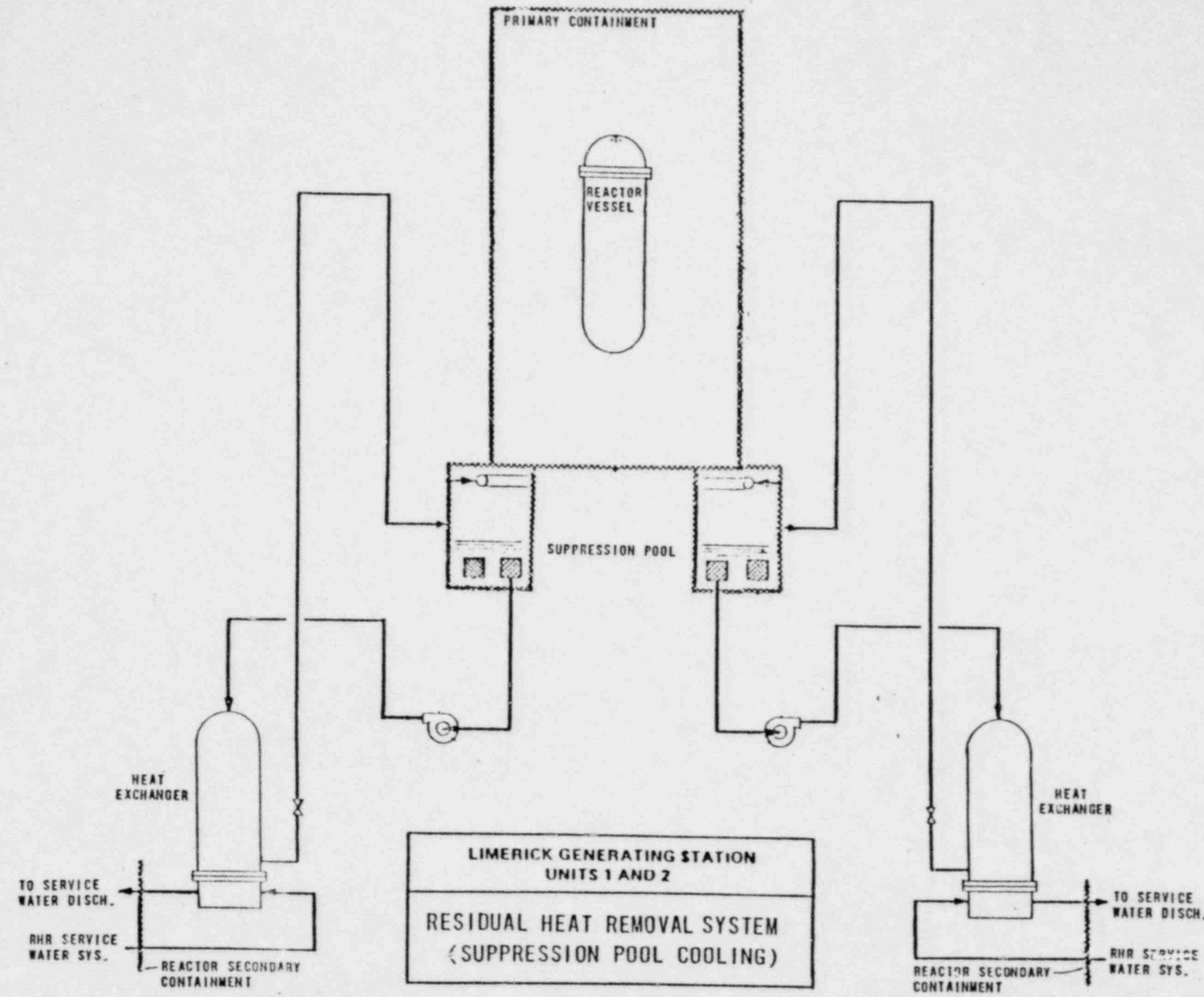
LIMERICK GENERATING STATION
UNITS 1 AND 2

RCIC DIAGRAM



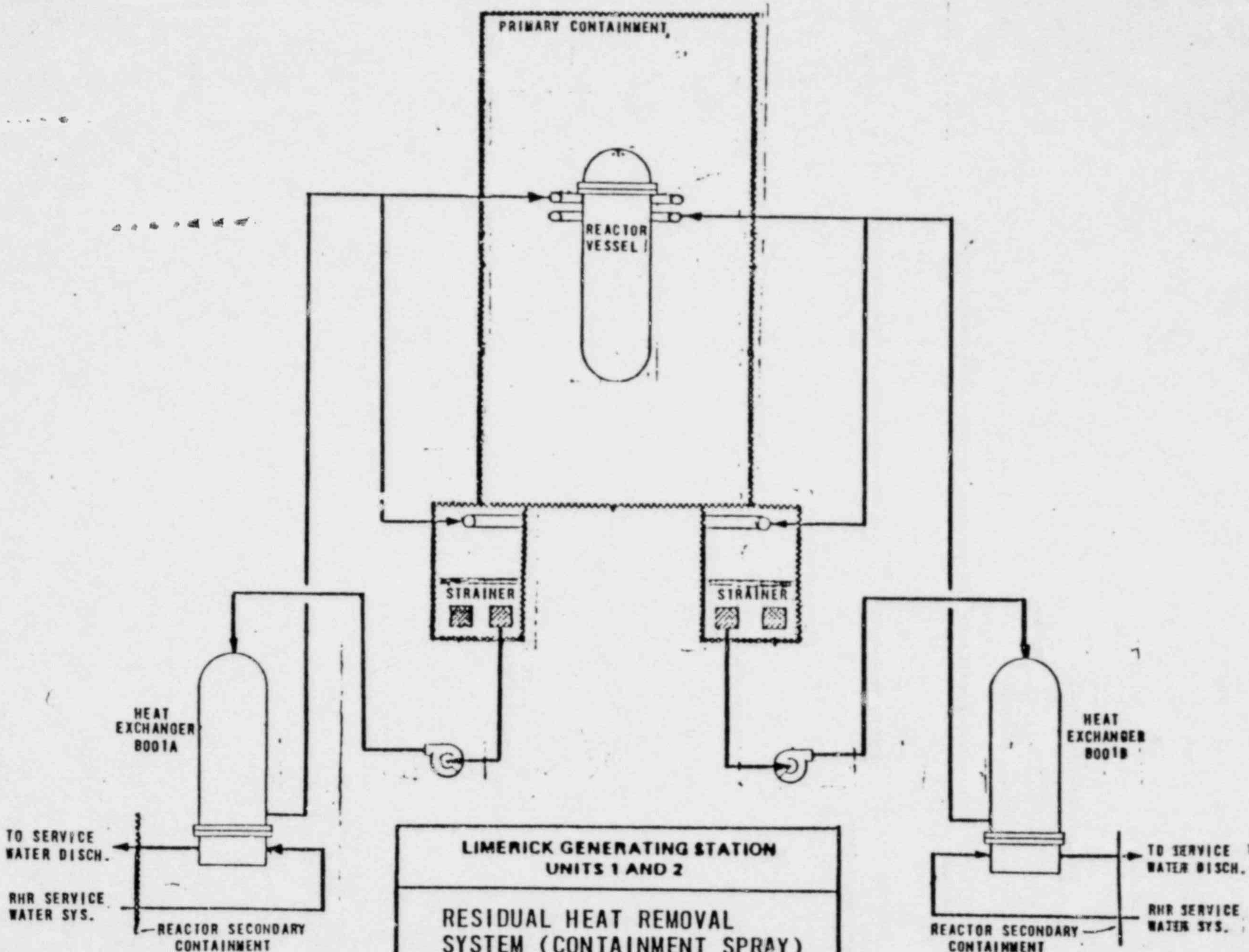
LIMERICK GENERATING STATION
UNITS 1 AND 2

RESIDUAL HEAT REMOVAL SYSTEM



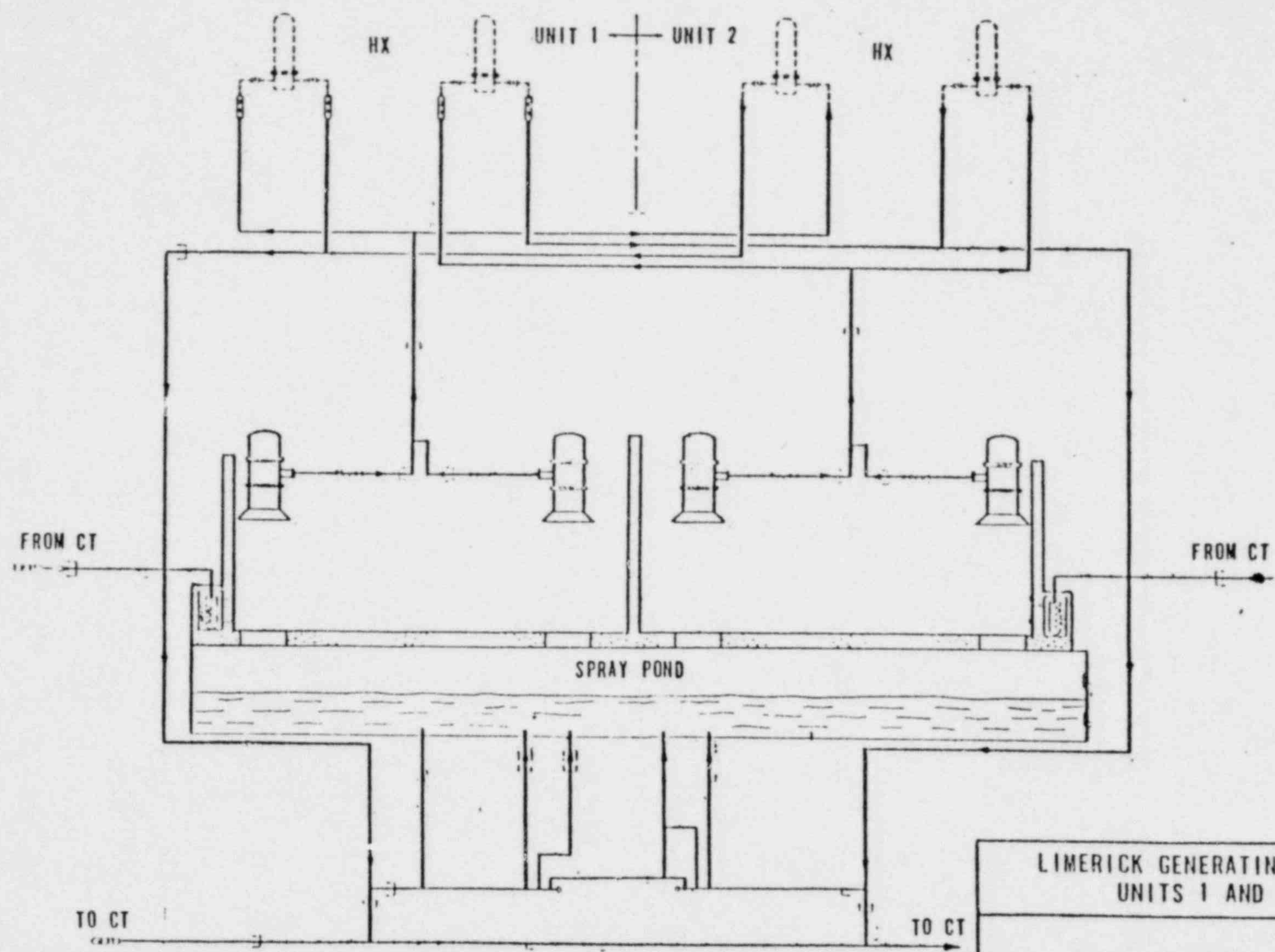
LIMERICK GENERATING STATION
UNITS 1 AND 2

RESIDUAL HEAT REMOVAL SYSTEM
(SUPPRESSION POOL COOLING)



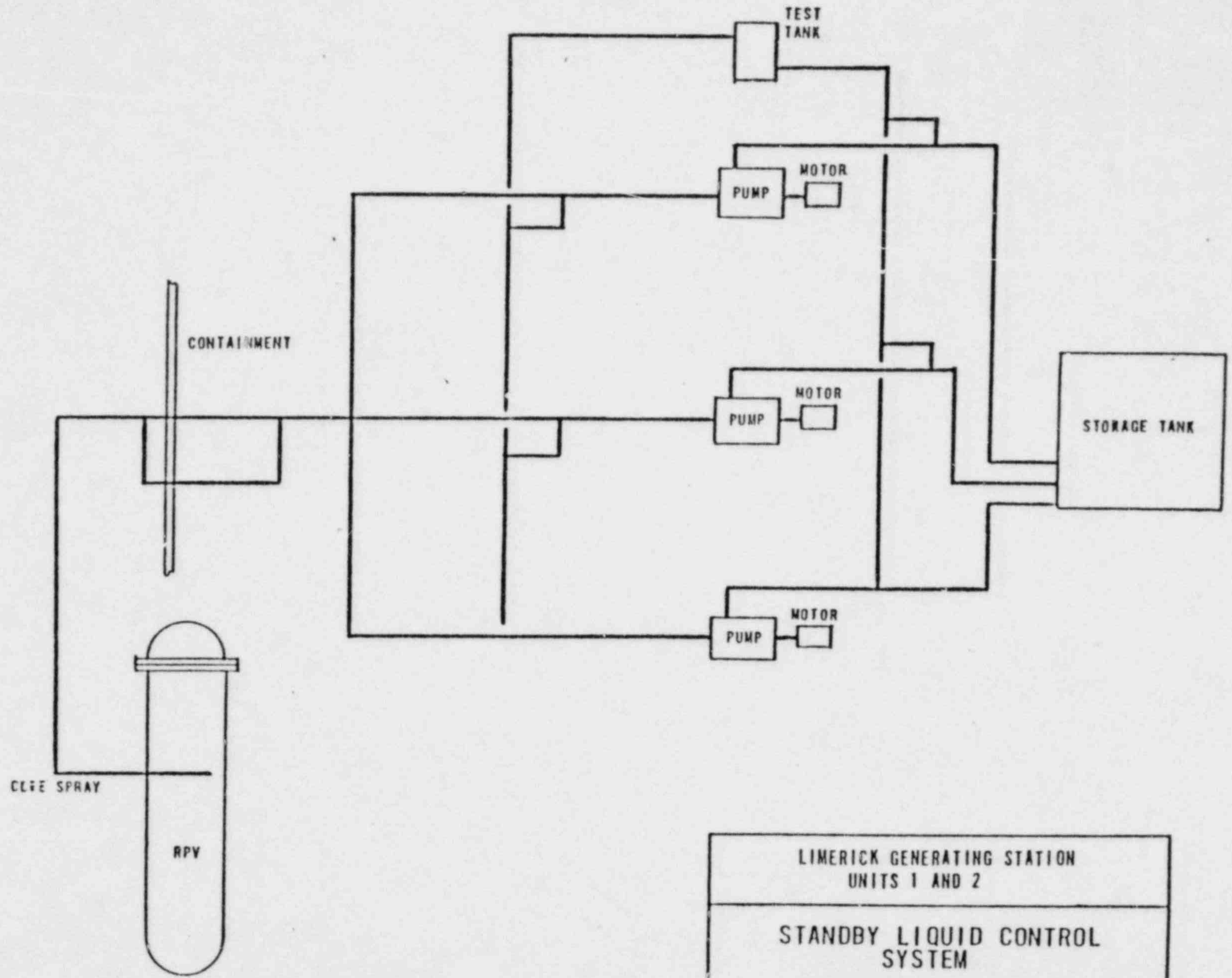
LIMERICK GENERATING STATION
UNITS 1 AND 2

RESIDUAL HEAT REMOVAL
SYSTEM (CONTAINMENT SPRAY)



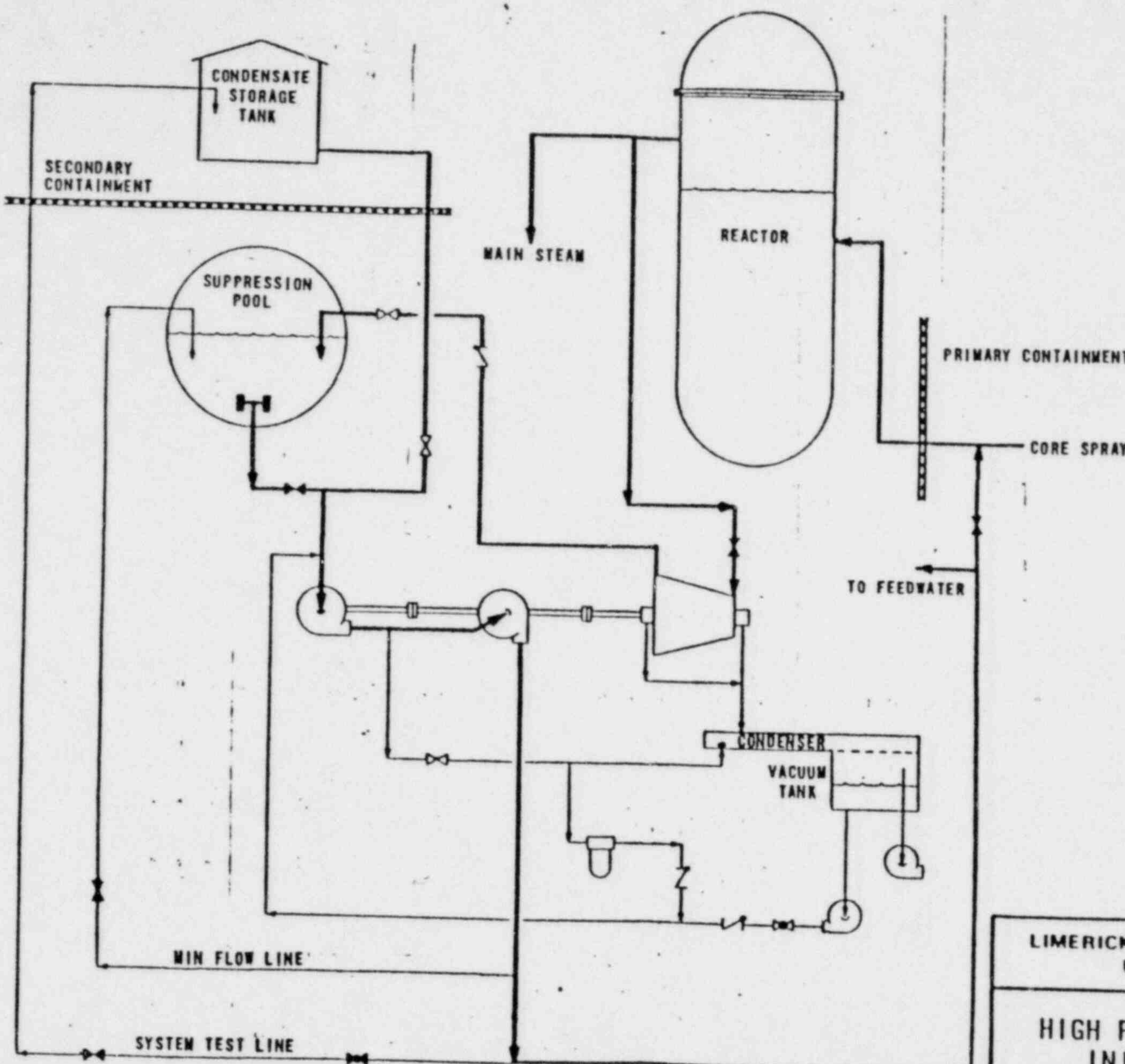
LIMERICK GENERATING STATION
UNITS 1 AND 2

RHR SERVICE WATER



LIMERICK GENERATING STATION
UNITS 1 AND 2

STANDBY LIQUID CONTROL SYSTEM



LIMERICK GENERATING STATION
UNITS 1 AND 2

HIGH PRESSURE COOLANT INJECTION SYSTEM

ENCLOSURE 4

COMMENTS OF MR. ALAN NOGEE

The following are comments received by NRC from Mr. Alan Nogee during a meeting recess period on February 11, 1982.

Mr. Nogee noted that 32 contentions related to the PRA were recently posed in the Limerick proceeding and that many of these contentions pertained to the basis of comparison between the PRA and the WASH-1400 study. He was of the opinion that selective use was made of updated methodology and data in preparing the PRA, leading to biased results and a questionable basis for comparison. Mr. Nogee also took issue with the use of 1970 population figures for both Limerick and WASH-1400. He stated that this represented data compiled 5 years prior to completion of Peach Bottom (WASH-1400) but 15 years prior to the projected completion of Limerick. Mr. Nogee also questioned why a 25 mile radius evacuation zone was assumed in the PRA and WASH-1400 study while the Limerick Emergency Plan assumes a 10 mile radius.

Mr. Nogee also took issue with the fact that GE participated in the preparation of the PRA. He felt that this amounted to GE evaluating the risks of its own design. He noted that the Indian Point and Zion PRAs were prepared by independent groups. Also, according to Mr. Nogee, the minimum time required to perform a PRA should be 18 months, and he noted that PECO completed its PRA in only 10 months. Mr. Nogee also stated that Keystone Alliance and Limerick Ecology Action were interested in assuring that the NRC staff performed a thorough, independent review of the PRA and offered to work closely with the staff in its review.

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Certified By Harry Hahl