BIG ROCK POINT PRA EXECUTIVE SUMMARY

- MOTIVATION FOR THE STUDY

- STUDY OBJECTIVES

- RESULTS

- CPCo CONCLUSIONS AND PLAN OF ACTION

8104100315

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MOTIVATION FOR STUDY

- CPCo Is Concerned About Projected Capital Expenditures

- Big Rock Point Has Performed Well In Terms Of Safety And Availability Over the Years

- A Basis For A Corporate Decision On Future Operation Of Big Rock Point Is Needed

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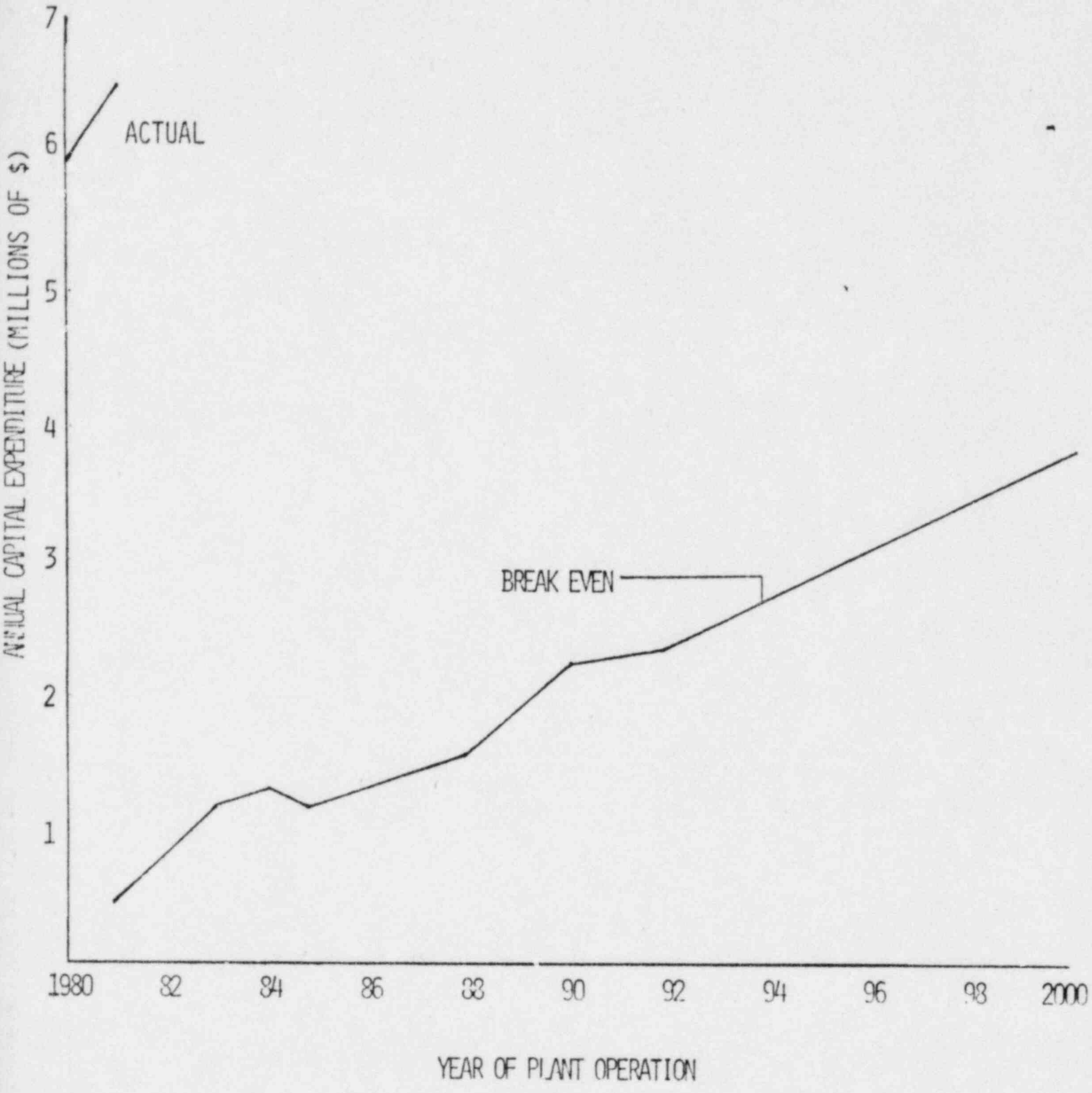

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STUDY OBJECTIVES

- Perform A Comprehensive And Systematic Evaluation Of Plant Safety
- Maintain Technical Objectivity In This Effort
- Explore Means Of Reducing Plant Risk
- Evaluate Effectiveness of Regulatory Issues Using PRA
- Develop A Cost Effective Risk Management Program For Big Rock Point

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PROBABILISTIC RISK ASSESSMENT
OF
BIG ROCK POINT

OBJECTIVE:
To Achieve
Cost Effective
Risk Reduction

Develop BRP PRA Model → Identify Risk Outliers → Select Plant Modifications → Short Term Plan → Long Term Plan

__Short Term Plan__

- Perform Detailed Design Work ~~Implement Mods~~
- Address Reg Issues
- Define CRMP

__Long Term Plan__

- Implement CRMP
- Implement Mods

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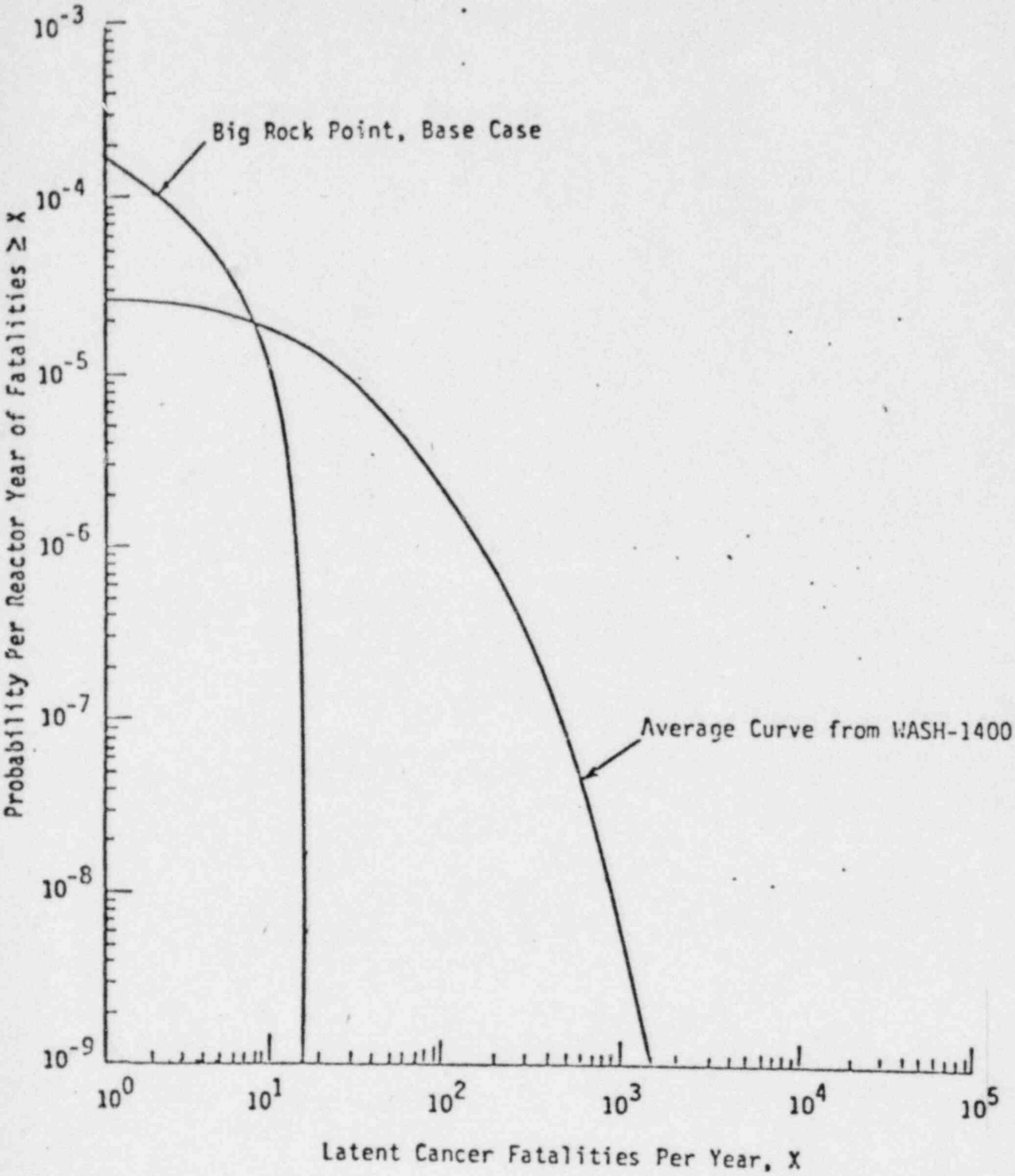

COMPARISON OF CCDF FOR LATENT FATALITIES PER YEAR
BETWEEN BIG ROCK POINT (BASE CASE)
AND THE AVERAGE CURVE FROM WASH-1400

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SUMMARY OF DOMINANT SEQUENCES

Sequence Class (No. of Sequences)	Core Damage Frequency (yr^{-1})	
Turbine Trip (3)	8.5×10^{-7}	8.7×10
Loss of Feedwater (1)	4×10^{-7}	
Loss of Main Condenser (6)	3.7×10^{-6}	
Loss of Offsite Power (15)	4.5×10^{-5}	4.7
LOCA (5)	4.3×10^{-5}	4.5
Steam Line Break Inside Containment (3)	1.1×10^{-4}	
Loss of Instrument Air (6)	3.3×10^{-5}	3.5
Spurious Closure of MSIV (4)	3.2×10^{-6}	
Spurious Opening of Turbine Bypass Valve (5)	7.1×10^{-5}	
ATWS (18)	4.7×10^{-5}	revised? 2.7×10^{-5}
Spurious Opening of RDS Isolation Valve (2)	1.7×10^{-5}	
High Energy Line Break (2)	1.5×10^{-6}	
Interfacing LOCA (2)	8.7×10^{-5}	9.1
Fire (6)	2.3×10^{-4}	
Stuck Open Safety Valve (8)	2.9×10^{-4}	
TOTAL (86 sequences)	9.82×10^{-4}	

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RISK REDUCTION ACTION PLAN

- PLANT MODIFICATIONS

- CONTINUING RISK MANAGEMENT PROGRAM

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SHORT TERM PLAN

1. CPCo will submit PRA Document 3/31

2. CPCo will submit PRA-Based Documentation for deferral items on 3/31

3. CPCo will initiate detailed design activities on PRA - Recommended Modifications as soon as the NRC issues a provisional acceptance of the CPCo Position stated in Item 2 above. This provisional acceptance involves deferral of implementation dates for these items pending completion of detailed review

4. CPCo will complete implementaton of the PRA-Recommended Modifications subsequent to completion of NRC review given NRC acceptance of PRA as an evaluation basis for future regulatory issues

CONCLUSIONS

- BRP CAN BE OPERATED AT AN ACCEPTABLY LOW LEVEL OF RISK TODAY.

- PRA INDICATES THAT THERE ARE SOME HIGHLY COST EFFECTIVE ACTIONS THAT CAN BE TAKEN TO REDUCE RISK.

- BASED UPON PROJECTED CAPITAL EXPENDITURES CPCo NEEDS SOME ASSURANCES THAT PRA WILL SERVE AS AN EFFECTIVE TOOL IN DECIDING UPON PLANT MODIFICATIONS.

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SCOPE OF STUDY

- COMPLETE BASELINE PRA
- SEQUENCE DEVELOPMENT AND PROBABILISTIC QUANTIFICATION
- IN-PLANT AND EX-PLANT CONSEQUENCES ANALYZED
- THOROUGH CONSIDERATION OF POTENTIAL PLANT MODIFICATIONS
- ON-GOING DEFINITION OF RISK MINIMIZATION PROGRAM

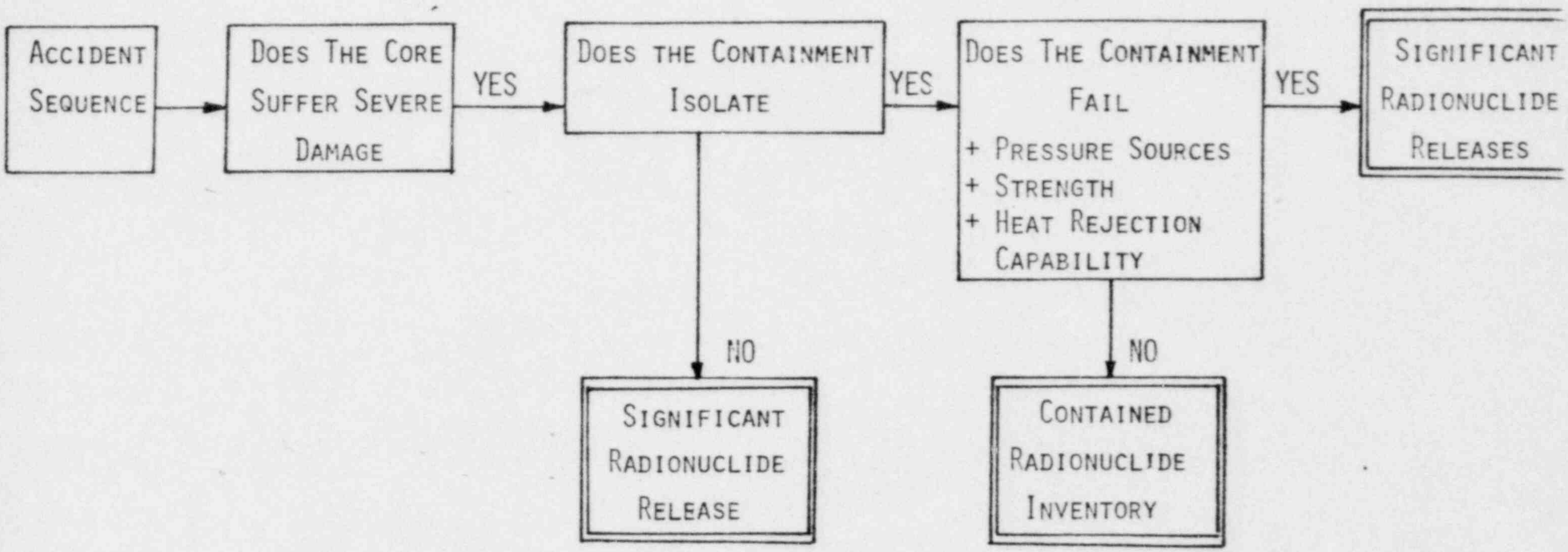
STUDY PURPOSE

EMPLOY THE TECHNIQUES OF PROBABILISTIC RISK ASSESSMENT (PRA) TO SUPPORT THE CONTINUED SAFE OPERATION OF THE BIG ROCK POINT NUCLEAR PLANT

APPROACH EMPLOYED

- o COMPLETE BASELINE PRA
  - + INITIATOR SPECIFIC TO PLANT
  - + ACCIDENT SEQUENCES (EVENT TREES AND FAULT TREES)
  - + PLANT SPECIFIC DATA
  - + IN-PLANT AND EX-PLANT CONSEQUENCES
- o DIFFICULT ISSUES TREATED DIRECTLY
  - + COMMON CAUSE FAILURES
  - + INTERNAL EVENTS (E.G., FIRES AND HIGH ENERGY LINE BREAKS)
  - + EXTERNAL EVENTS (E.G., SEISMIC AND WIND LOADINGS)
  - + EQUIPMENT ENVIRONMENTAL QUALIFICATION
- o INCLUDED IN SCOPE
  - + UNIQUE APPROACHES TO ASSURING COMPLETENESS
  - + FORMULATION AND INVESTIGATION OF EFFECT OF VARIOUS PLANT MODIFICATIONS
  - + SIGNIFICANT CPCo PARTICIPATION
- o EXCLUDED FROM SCOPE
  - + SABOTAGE
  - + DETAILED QUANTIFICATION OF PROBABILITY OF FAILURE TO SCRAM

CORE DAMAGE AND RADIONUCLIDE RELEASE EVALUATION



COMPARISONS OF CONTAINMENT PHYSICAL PARAMETERS

	BIG ROCK POINT	SURRY
POWER LEVEL (MW_t)	240	2,400
CONTAINMENT VOLUME (CUBIC FEET)	9.4×10^5	1.8×10^6
CONTAINMENT DESIGN PRESSURE (PSIG)	27.0	45.0
ENERGY RELEASES IN LARGE LOCA		
INFORMATION REQUIRED		
o PRIMARY VOLUME (CUBIC FEET)	3,639 (2,718 CU. FT. OF WATER)	8,387
o PRIMARY TEMPERATURE(°F)	566°F	572°F
o PRIMARY PRESSURE (PSIA)	1,350	2,295
PRESSURE IN CONTAINMENT FOLLOWING A LARGE LOCA ASSUMING NO CONTAINMENT SPRAY (PSIG)	PEAK PRESS. 20 PSIG	39.3 PSIG
VOLUME OF STEAM PRODUCED AT ATMOSPHERIC PRESSURE BY 1% DECAY POWER (CUBIC FEET PER MINUTE)	3.82×10^3 (@ 100°C)	3.88×10^4 (@ 100°C)
DIVIDED BY CONT. VOLUME	4.1×10^{-3}	2.2×10^{-2}
MASS OF UO_2 (POUNDS)	27,500 LBS.	175,600 LBS.
MASS OF ZIRCONIUM IN CORE (POUNDS)	11,270 LBS.	36,300 LBS.
PERCENT OF H_2 IN CONTAINMENT (ASSUMING ALL ZIRCONIUM REACTS)	7 TO 8% H_2	12 TO 14% H_2
REMOVAL RATE CONSTANT (HR^{-1})		
o IODINE (NATURAL)	1.5	1.4
o PARTICULATE (NATURAL)	0.9	0.6
o IODINE (SPRAY)	0.1	3.0 (BORIC ACID), (IF HYDROXIDE IN SPRAY, 30)
o PARTICULATE (SPRAY)	0.6	20.0

Table 7.1

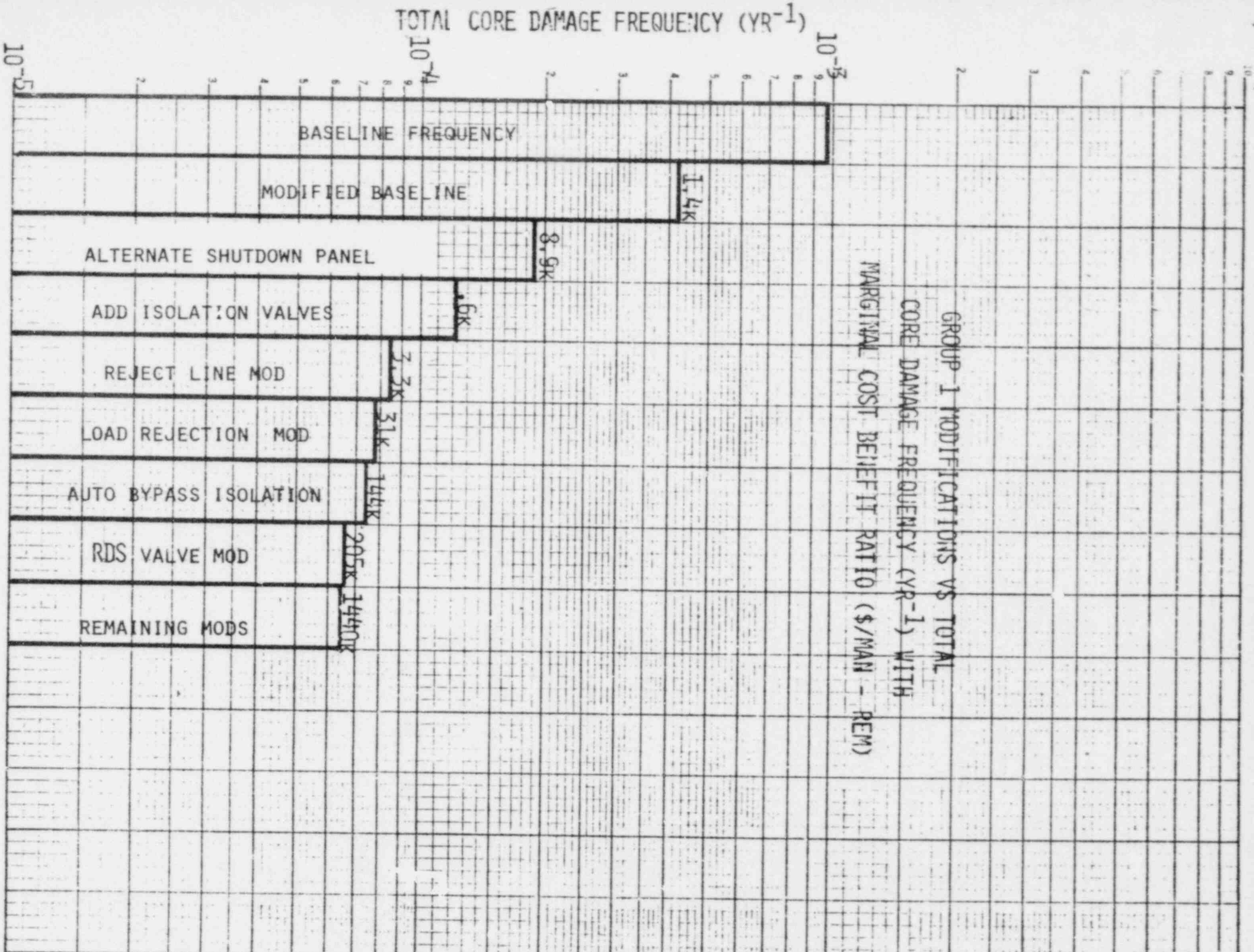
LIST OF RISK OUTLIERS AND SEQUENCE CLASSES AFFECTED

RISK OUTLIER / SEQUENCE CLASSES	Turbine Trip	Loss of Feedwater	Loss of Main Condenser	Loss of Offsite Power	LOCA	Steam Line Break Inside Containment	Loss of Instrument Air	Spurious Closure of MSIV	Spurious Opening of Turbine Bypass Valve	ATWS	Spurious Opening of RDS Isolation Valve	High Energy Line Break	Interfac'ng LOCA	Fire	Stuck Open Safety Valve
Emergency Condenser Makeup Failure	x		x	x			x	x	x						x
Environmental Qualification	x	x	x	x	x	x	x	x	x	x	x		x		x
Limited FW During ATWS										x					
MSIV Backup Failure				x											
Post Incident System Reliability	x	x	x	x	x	x	x	x	x		x		x		x
RDS/CS Reliability	x		x	x	x	x	x	x	x		x		x		x
Standby Diesel Reliability				x											
Instrument Air System Repair							x								
Leaking RDS Valves											x				
Single Valve Isolation of Primary System													x		
Proximity of Safety System Piping to High Energy Lines												x			
Concentration of Safety System Electrical Cables in Single Locations														x	
Late Automatic Isolation of Main Steam Line on Loss of Coolant									x						
Secondary System Instabilities		x		x						x	x				

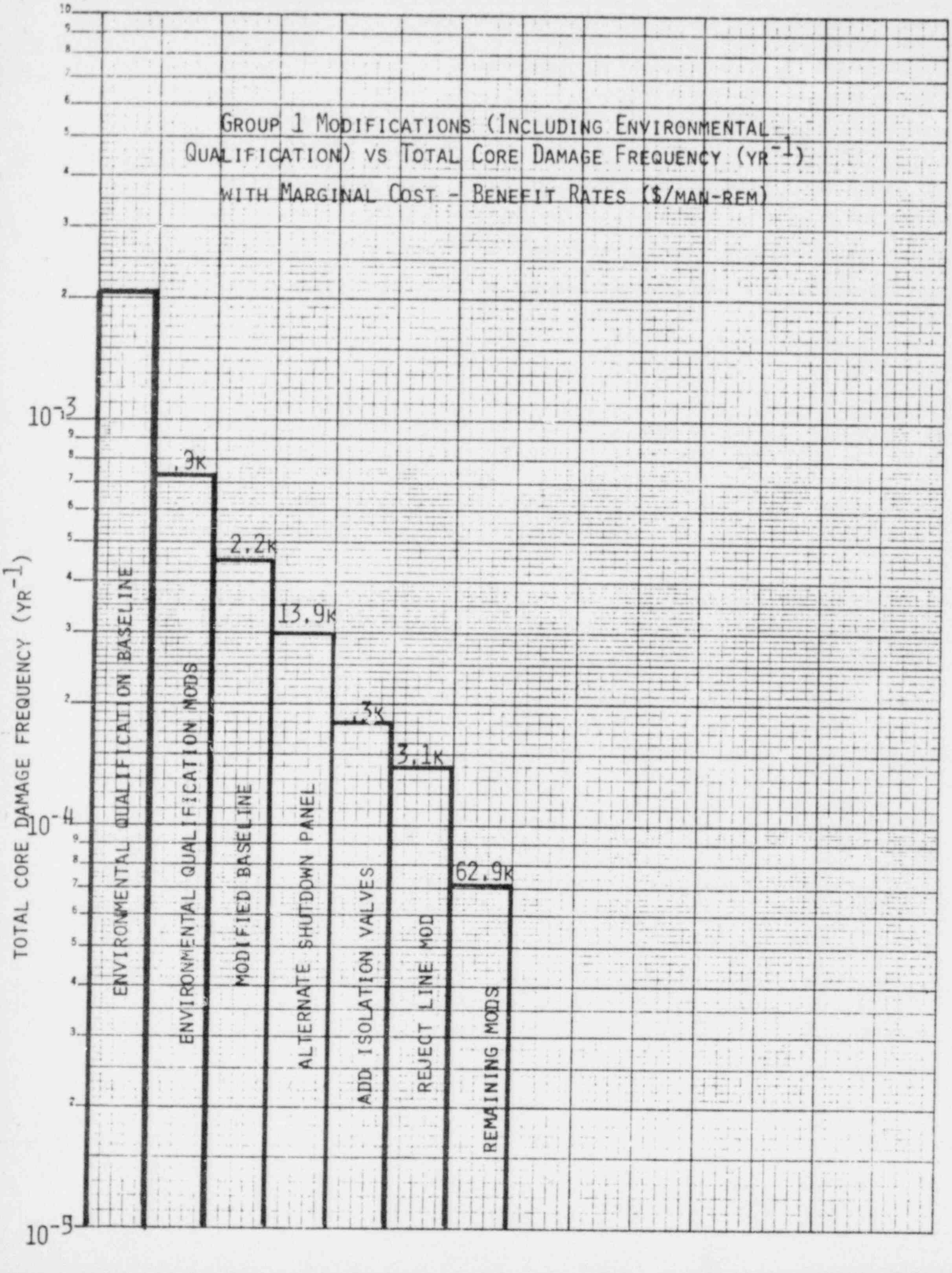
TABLE 1.1
SUMMARY OF DOMINANT SEQUENCES

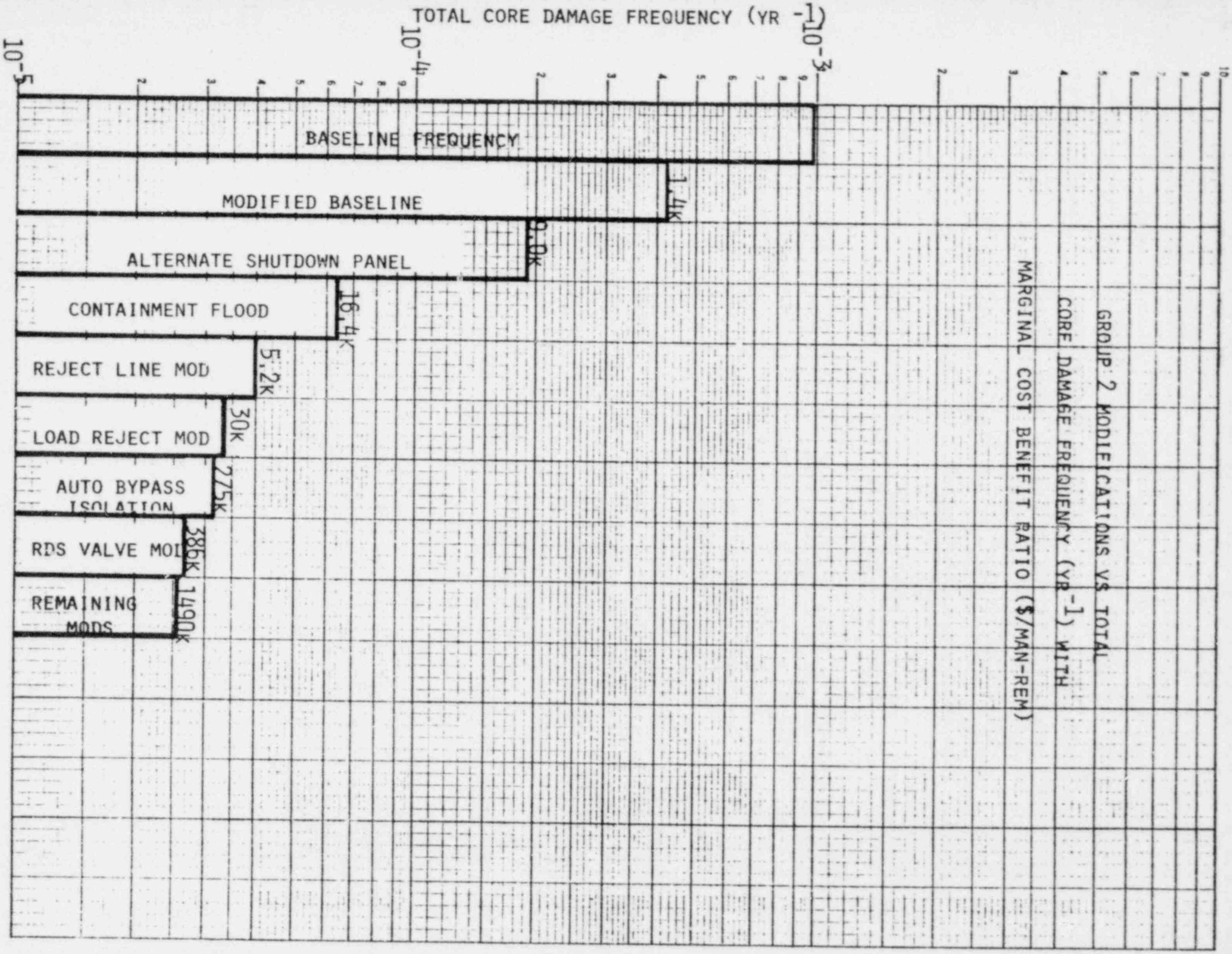
Sequence Class (No of Sequences)	Frequency (yr^{-1})	Uncertainty*
Turbine Trip (3)	8.7×10^{-7}	20
Loss of Feedwater (1)	4×10^{-7}	98
Loss of Main Condenser (6)	4.0×10^{-6}	35
Loss of Offsite Power (15)	4.7×10^{-5}	29
LOCA (5)	4.5×10^{-5}	19
Steam Line Break Inside Containment (3)	1.1×10^{-4}	49
Loss of Instrument Air (6)	3.5×10^{-5}	18
Spurious Closure of MSIV (4)	3.2×10^{-6}	40
Spurious Opening of Turbine Bypass Valve (5)	7.1×10^{-5}	22
ATWS (13)	2.7×10^{-5}	11
Spurious Opening of RDS Isolation Valve (2)	1.7×10^{-5}	9
High Energy Line Break (2)	1.5×10^{-6}	10
Interfacing LOCA (2)	9.1×10^{-5}	4
Fire (6)	2.3×10^{-4}	13
Stuck Open Safety Valve (8)	2.9×10^{-4}	45
TOTAL (81 sequences)	9.75×10^{-4}	15

* Uncertainty taken to be 95% confidence limit ÷ mean

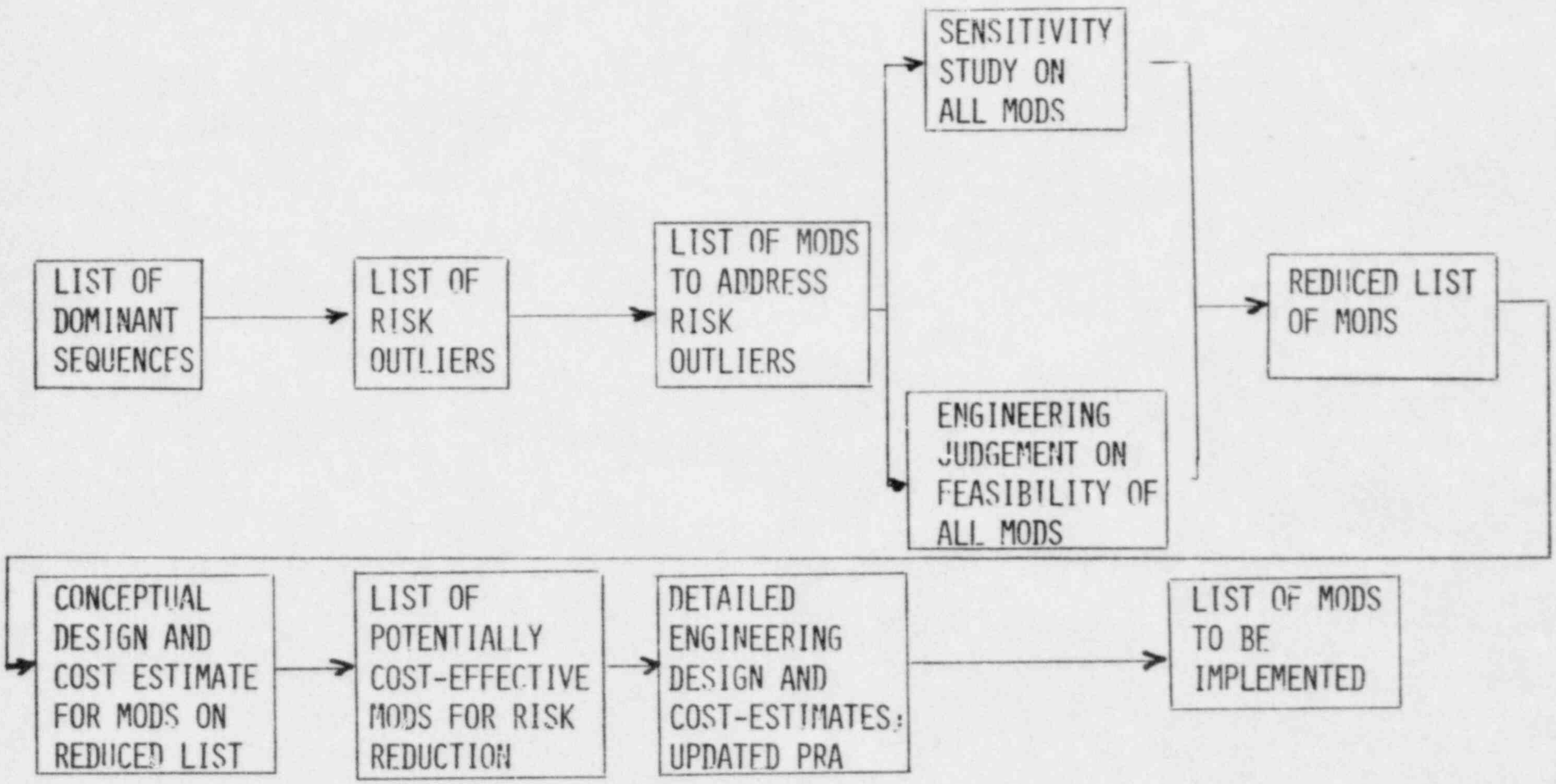


TOTAL CORE DAMAGE FREQUENCY (YR-1)
10-5
10-4
10-3
BASELINE FREQUENCY
MODIFIED BASELINE
1.4K
ALTERNATE SHUTDOWN PANEL
8.9K
ADD ISOLATION VALVES
16K
REJECT LINE MOD
3.3K
LOAD REJECTION MOD
31K
AUTO BYPASS ISOLATION
144K
RDS VALVE MOD
205K
REMAINING MODS
1440K
GROUP 1 MODIFICATIONS VS TOTAL
CORE DAMAGE FREQUENCY (YR-1) WITH
MARGINAL COST BENEFIT RATIO ($/MAN - REM)


GROUP 1 MODIFICATIONS (INCLUDING ENVIRONMENTAL QUALIFICATION) VS TOTAL CORE DAMAGE FREQUENCY (YR^{-1}) WITH MARGINAL COST - BENEFIT RATES ($/MAN-REM)
TOTAL CORE DAMAGE FREQUENCY (YR^{-1})
10^{-3}
10^{-4}
10^{-5}
ENVIRONMENTAL QUALIFICATION BASELINE
.9K
ENVIRONMENTAL QUALIFICATION MODS
2.2K
MODIFIED BASELINE
13.9K
ALTERNATE SHUTDOWN PANEL
.3K
ADD ISOLATION VALVES
3.1K
REJECT LINE MOD
62.9K
REMAINING MODS


TOTAL CORE DAMAGE FREQUENCY (YR -1)
10-5
10-4
10-3
BASELINE FREQUENCY
MODIFIED BASELINE
1.4K
ALTERNATE SHUTDOWN PANEL
9.0K
CONTAINMENT FLOOD
16.4K
REJECT LINE MOD
5.2K
LOAD REJECT MOD
30K
AUTO BYPASS ISOLATION
275K
RDS VALVE MOD
386K
REMAINING MODS
1490K
GROUP 2 MODIFICATIONS VS TOTAL CORE DAMAGE FREQUENCY (YR -1) WITH MARGINAL COST BENEFIT RATIO ($/MAN-REM)

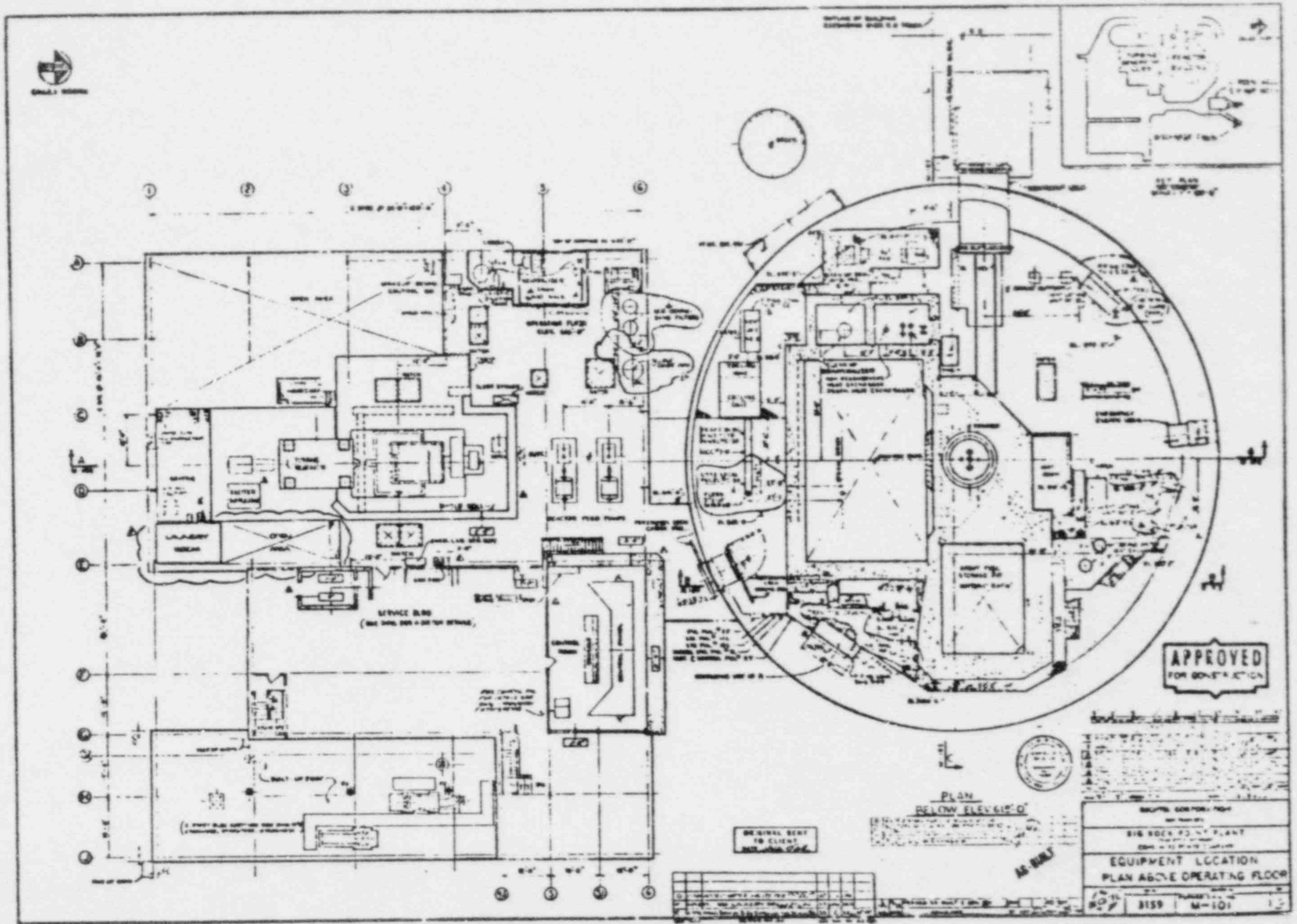
STEPS OF PROCESS FOR DEFINING
PLANT MODIFICATIONS RESULTING FROM PRA



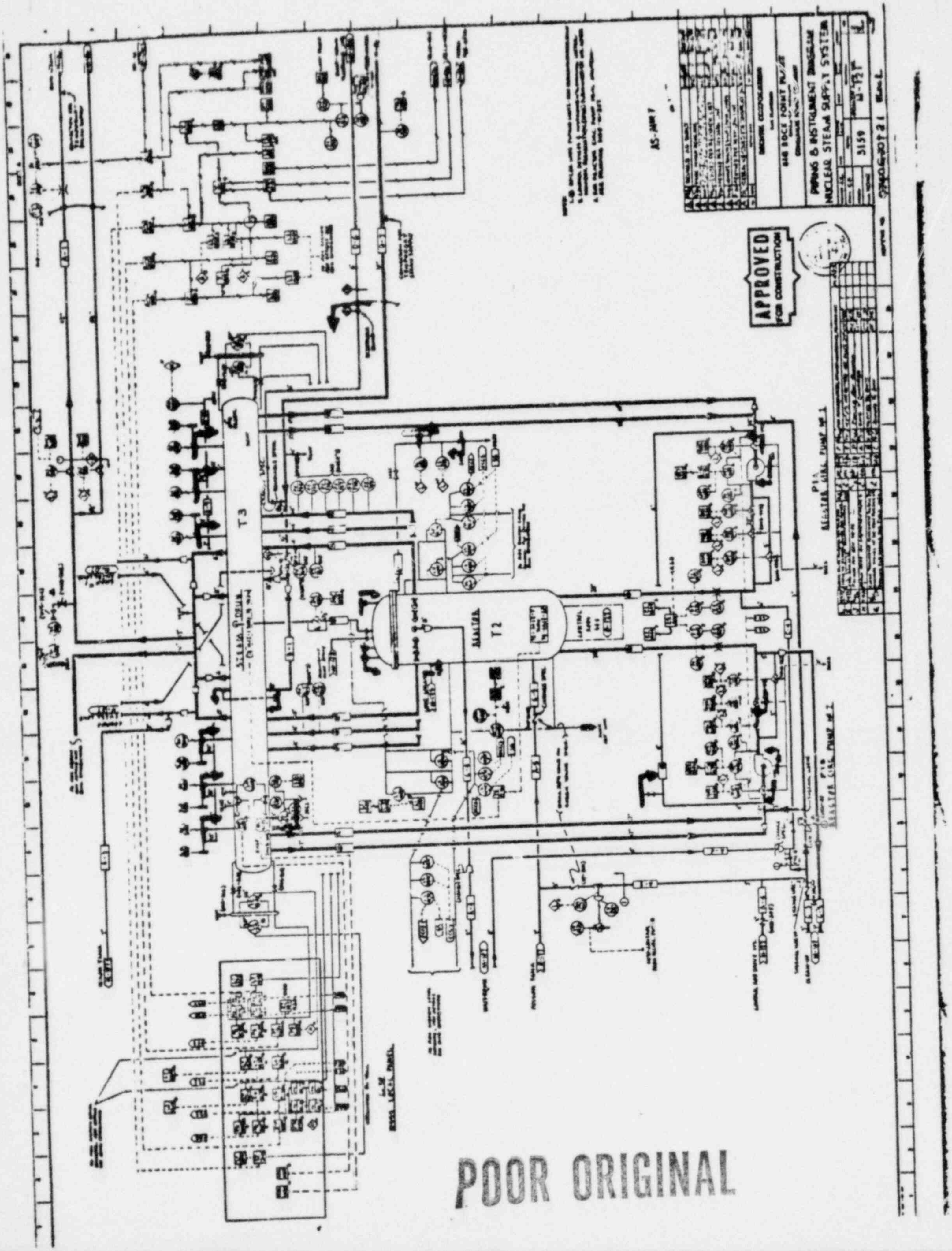
SAFE SHUTDOWN AREA RECOMMENDATIONS FOR FIRE PROTECTION

10CFR 50 Appendix R	Big Rock PRA
Equipment on Panel for Short Term Cooling	
MSIV control and power	MSIV control and power
Emergency condenser outlet valve control and power (1 loop)	Emergency condenser outlet valve control and power (2 loops)
Emergency Condenser makeup valve from fire system control and power	Emergency condenser makeup valve from fire system control and power
Manual Reactor Trip capability	
Instrumentation	
Drum Level	Drum Level
Emergency Condenser Level (annunciator)	Emergency condenser level (annunciator)
Emergency condenser valve position indication makeup & outlet valve	Emergency condenser valve position makeup & outlet valve
MSIV position indication	MSIV position

10CFR 50 Appendix R	Big Rock PRA
Long Term Cooling Equipment	
Shutdown Pump and motor operated valves Reactor cooling water pump Service Water pump & capability to power this equipment through shutdown area with emergency diesel	Emergency condenser makeup valve capable of remote and local hand operation
Other Miscellaneous Requirements	
Prevent emergency condenser inlet valves from closing on hot shorts	Prevent emergency condenser inlet valves from closing on hot shorts
	Provide power and control for a CRD pump from the safe shutdown area

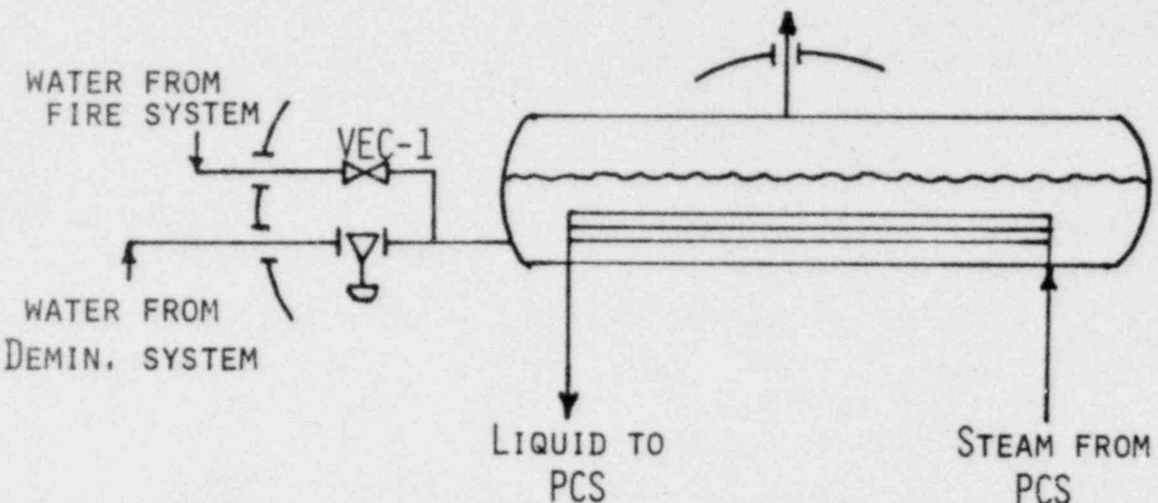


APPROVED
FOR CONSTRUCTION
SERVICE BLDG
PLAN
BELOW ELEV 615'-0"
ORIGINAL SENT TO CLIENT
AS-BUILT
EQUIPMENT LOCATION
PLAN ABOVE OPERATING FLOOR
3159
M-101



POOR ORIGINAL

EMERGENCY CONDENSER MAKEUP FAILURE MODIFICATIONS



MOD 1 REPLACE VEC-1 WITH REMOTE MANUAL D.C. POWERED VALVE

MOV FAILS TO OPEN	7.1E-3
OPERATOR FAILS TO ACTUATE	1.0E-3
	8.1E-3

MOD 2 REPLACE VEC-1 WITH AUTOMATIC D.C. POWERED VALVE

MOV FAILS TO OPEN	7.1E-3

MOD 3 PLACE DEMIN. PUMP AND AN AIR COMPRESSER ON THE EMERGENCY BUS

1.0E-2

CHECK VALVES FAIL TO OPEN	2.0E-3
DEMIN. PUMP OUT FOR MAINT.	7.3E-3
DEMIN. PUMP FAIL TO RUN	2.7E-4
LOSS OF AIR	2.7E-5
	1.0E-2

RESULTS OF EMERGENCY CONDENSER MAKEUP SENSITIVITY STUDY

MODIFICATION	LOSS OF OFFSITE POWER SEQUENCES W/O MOD	W. MOD	STUCK OPEN SAFETY VALVE SEQUENCES W/O MOD	W. MOD	ALL AFFECTED SEQUENCES W/O MOD	W. MOD	TOTAL CORE DAMAGE FREQUENCY W/O MOD	W. MOD
1. REPLACE VEC-1 WITH REMOTE MANUAL D.C. POWERED VALVE	4.7E-5	1.11E-5	2.9E-4	6.8E-5	4.5E-4	1.6E-4	9.75E-4	6.76E-4
2. REPLACE VEC-1 WITH AUTOMATIC D.C. POWERED VALVE	4.7E-5	1.10E-5	2.9E-4	6.7E-5	4.5E-4	1.59E-4	9.75E-4	6.75E-4
3. PLACE DEMIN. PUMP AND AN AIR COMPRESSER ON EMERGENCY BUS	4.7E-5	1.6E-5	2.9E-4	1.9E-4	4.5E-4	3.2E-4	9.75E-4	8.36E-4

THE NEED FOR SHIELDING AT BIG ROCK POINT

NEED MAY COME FROM REQUIREMENTS IN NUREG-0737

OR

COMPARISON BETWEEN THE PLACEMENT OF SHIELDING VS AFFECT ON CORE MELT PROBABILITY

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NUREG-0737 ACTION GUIDANCE

1. REVIEW VITAL AREAS WHICH MAY REQUIRE OCCUPANCY.
2. REVIEW SYSTEMS WHICH MAY CONTAIN LARGE QUANTITIES OF RADIO NUCLIDES.
3. DETERMINE WHICH OF THESE SYSTEMS AFFECT THE VITAL AREAS.
4. DETERMINE THE MAGNITUDE OF THE SOURCE AT THE TIME OF THE ACTION (NO SOURCE, GAP RELEASE, MELT RELEASE, ETC).
5. DETERMINE DOSE.

CRITERIA - 15 MR/HR AVERAGE OVER 30 DAYS FOR CONTINUOUS OCCUPANCY AREAS.

5 REM TOTAL EXPOSURE FOR ACCESS INTO AREAS INFREQUENTLY.

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METHODOLOGY AND ASSUMPTIONS

1. RADIOACTIVITY

   WORST CASE FOR DIRECT SHINE - LARGE LOCA

   WORST CASE FOR AIRBORNE - LOSS OF STATION POWER WITH FAILURE OF THE PCS TO ISOLATE RELEASES NOT AT T=0 BUT OVER TIME OBTAINED FROM INCOR CODES.

2. CALCULATIONAL METHODOLOGY

   SHINE - USE EQUATION CONCERNING DOSE TO A POINT FROM A FINITE SPHERICAL SOURCE.

   INHALATION - RISK ASSESSMENT METHODOLOGY PROVIDED RELEASE FRACTIONS VS TIME; X/Q CALCULATED FOR OUR CASE; VENTILATION CHARACTERISTICS USED; FACTOR OF 10^{-3} USED TO DETERMINE IODINE CONCENTRATIONS CHEMICALLY REDUCING ENVIRONMENT (ALWAYS WATER PRESENT AS STEAM OR WATER) NO OXIDIZING ENVIRONMENT (CONTAINMENT P > TB P).

3. DOSES

   25 REM FOR ACCIDENT CONDITIONS FROM NCRP REPORT 39 & DOE IN EP

   FACTOR OF 20 REDUCTION IN THYROID DOSE WHEN KI PILLS TAKEN, NCRP REPORT 55.

   5 REM WHOLE BODY, 30 REM THYROID FOR CONTINUOUS OCCUPIED AREAS.

4. SCENERIO

LARGE LOCA		LOSS OF POWER	
T=0	INITIATION	T=0	INITIATION
T=15M	MELT STARTS	T=15M	MELT STARTS
T=1HR	MELT PHASE ENDS (90% XE, KR)	T=1.5HR	MELT PHASE ENDS
		T=3HRS	VESSEL MELT THROUGH
T=4.4HR	REMAINING NUCLIDES RELEASED	T=5HRS	REMAINING NUCLIDES RELEASED

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Review Considered

<u>Vital Areas</u>

1. All operator actions specified by the procedure.
2. Fuel oil delivery, standby EDG set-up, Ingress and Egress.
3. Areas of possible maintenance (core spray, Hx room, screen house).
4. Inhabited areas (control room, technical support center, electric equipment room, sample analysis area).

<u>Sources</u>

1. Containment.
2. Post incident recirculation piping.

<u>Magnitudes of Radioactivity</u>

1. Normal operation levels.
2. Gap release (approximately 2% melt release).
3. Melt release.

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Operator Actions

Reviewed the procedures and determined the actions and where they take place.

The vast majority of the actions take place before RDS.

After melt actions -

- Occupy Control Room, TSC
- Get to Sample Analysis Area, Electric Equipment Room
- Fill fuel oil tanks (heating boiler, EDG, diesel fire pump, standby EDG)
- Ingress and egress
- Switch to recirculation (manual valves)

Other possible actions -

- Repair core spray system components in screenhouse.
- Repair post incident system components in core spray Heat Exchanger Room.
- Open manual valve in core spray Heat Exchanger Room should MOV fail.
- Switch strainer in BA[illegible]T strainers
- Switch loads should O[illegible] power be lost and connect standby EDG

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SUMMARY OF SHIELDING ISSUE

- NO ADDITIONAL SHIELDING IS REQUIRED TO SATISFY NUREG-0737
- RESULTS OF BASELINE PRA WILL NOT BE SIGNIFICANTLY AFFECTED BY ADDTION OF SHIELD WALL OR LOCAL SHIELDING
- POTENTIAL LOCAL SHIELDING REQUIRED FOR PLANT MODIFICATION WILL BE ADDRESSED IN THE DETAILED MODIFICATION DESIGN PACKAGE
- POTENTIAL FOR FURTHER REDUCTION IN THE DOSE OBTAINED FROM INGRESS OR EGRESS WILL BE CONSIDERED

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Results

All areas continuously occupied; < 5 rem.

All area infrequently occupied
(includes short and long action); < 25 rem.

Inhalation doses in continuously occupied
area; < 0.1 rem to the thyroid
(potassium iodine pills taken, realistic iodine
release-a factor of 1000 less than noble gas release)

Fill oil tanks; < 25 rem

Ingress and egress; <12 rem worst case - 0-7 rem typical

Involve only long term maintenance activities and are not
required to be performed.

Screenhouse - to fix components of core spray system (fire
protection system) prior to RDS if the need arises;
dose >25 rem (more details further back)

Core Spray Heat Exchanger Room - to fix components of the
post incident system prior to recirculation, if the need
arises; dose >25 rem (more details further back)

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Action

Reviewed Risk Assessment Dominent Sequences

Reviewed time to core melt vs core spray failure
and PIS failure

Core srpay failures - cannot significantly affect
system failure probability prior to core melt

PIS failure - can significantly affect system failure
probability prior to core melt for non-break scenerios
(17 hours until core melt from initial LTC action)

However Dominent Sequences with PIS failure make up only
2.4E-4 of the 9.8E-4 melt probability

so fixing PIS would not significantly affect total core
melt probability

As a result shielding will not be placed in these areas
unless as a result of the review other issues are raised

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TYPICAL ACCIDENT SCENERIO

LOSS OF OFFSITE POWER

SCRAM

EMERGENCY POWER ACTIVATION

MSIV'S CLOSE (ON LOSS OF AC)

EMERGENCY CONDENSER TRIED TO REMOVE DECAY HEAT BUT IT FAILS RESULTS IN RAPID PRESSURIZATION.

PRESSURE CONTINUES TO RISE UNTIL SAFETY VALVE SETPOINT IS REACHED.

SAFETIES OPEN PLANT COOLING WITH FEED AND BLEED BUT SPHERE IS FILLING WITH WATER.

MUST USE CORE SPRAY/RDS, IF THIS FAILS, NO COOLING AND CORE MELTS.

IF RDS/CORE SPRAY GOES, NEED LONG TERM COOLING.

IF LONG TERM COOLING FAILS, CORE WILL MELT LATER.

	RDS/CORE SPRAY FAILURE	LTC FAILURE
CORE GAP RELEASED (RDS)	5.9 HRS	5.9 HRS
CORE MELT STARTED	6.3 HRS	40 HRS
MELT 80%	11.5 HRS	46 HRS

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CALCULATIONS

CONTROL ROOM, TSC ELECTRICAL EQUIPMENT ROOM (720 HRS CONTINUOUS)
DAMPER CLOSED BY PROCEDURE ON HIGH AREA MONITOR ALARM AT 0.09 MR/HR
(5CFM LEAK)

SHINE FROM CONTAINMENT < 5 REM FOR ISOLATED CONTAINMENT CASE.

SHINE FROM AIRBORNE CLOUD < 1 REM FOR NONISOLATED CASE.

SAMPLE ANALYSIS AREA

SHINE FROM	T=1 HR	27 R/HR
CONTAINMENT	5HR	6.3 R/HR
(ISOLATED)	24 HR	0.7 R/HR

SHINE FROM AIRBORNE RELEASE < 5 REM OVER 720 HOURS

FILL OIL TANKS, START STANDBY EDG < 25 REM

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Ingress & Egress

(worst case - large LOCA)

Time (From Initiation	Dose Rate R/hr	Dose Received (Using 36 second entry time)
0	0	0
15 min	30 R/hr	0.3 rem
1 hr	1275 R/hr	12.75 rem
2 hr	1087 R/hr	10.87 rem
8 hr	435 R/hr	4.35 rem
24 hr	120 R/hr	1.20 rem

Typical Case - Loss of Offsite Power - CS fails

0	0	0
1.75 hr	0	0
2.25 hr	11.25 R/hr	0.11 rem
4.24 hr	675 R/hr	6.75 rem
8 hr	405 R/hr	4.05 rem
24 hr	108 R/hr	1.08 rem

Typical Case - Loss of Offsite Power -PIS fails

0	0	0
5.9 hr	0	0
8 hr	4.05 R/hr	0.04 rem
24 hr	1.5 R/hr	0.015 rem
46 hr	67.5 R/hr	0.675 rem

RMMARUSICH
3/17/81

Problem Areas

Screen House

Time After Trip, Hr	Melt Release
T=1	2834 R/hr
5	667
24	77.5
720	16.2

Core Spray Heat Exchanger Room

	Shine from Containment		Inside Room
Time (After Trip, hr)	Room Dose Rate R/hr	Near Pump Dose Rate R/hr	General Field R/hr
T=1	426	4260	1239
5	46.8	468	771
24	4.9	48.8	161
720	1.8	18.1	6.2

R.M. Marusich
3/17/81

NUREG 0737 ITEMS

AFFECTED BY PRA

DIRECT AFFECT

II.B.2	7/1/82	DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST-ACCIDENT OPERATIONS
II.B.3	7/1/82	POST-ACCIDENT SAMPLING CAPABILITY
II.F.2	7/1/82	INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING
II.K.3.14	7/1/82	ISOLATION OF ISOLAITON CONDENSERS ON HIGH RADIATION
II.K.3.19	7/1/81	INTERLOCK ON RECIRCULATION PUMPS LOOPS
III.A.1.2	TBD	UPGRADE EMERGENCY SUPPORT FACILITIES (NUREG-0696)
III.D.3.4	7/1/82	CONTROL ROOM HABITABILITY

POTENTIAL AFFECT

I.A.1.3		SHIFT MANNING
I.C.1	7/1/82	GUIDE FOR EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS
I.D.1	TBD	CONTROL ROOM DESIGN REVIEWS
I.D.2	TBD	SAFETY PARAMETER DISPLAY
II.B.1	7/1/82	REACTOR COOLANT SYSTEM VENTS
II.D.1	7/1/81	PERFORMANCE TESTING OF BWR AND PWR PELIEF AND SAFETY VALVES
II.F.1 ATT.5	7/1/82	CONTAINMENT WATER LEVEL MONITOR
II.K.3.20	7/1/81	LOSS OF SERVICE WATER FOR BIG ROCK POINT
II.K.3.25	7/1/82	EFFECT OF LOSS OF ALTERNATING CURRENT POWER ON PUMP SEALS
II.K.3.44	7/1/81	EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO FUEL FAILURE
III.A.2		IMPROVING LICENSEE EMERGENCY PREPAREDNESS LONG TERM (NUREG-0654/FEMA-REP-1)

OTHER ISSUES
AFFECTED BY PRA

I. SYSTEMATIC EVALUATION PROGRAM (SEP)- SEISMIC AND SYSTEMS INTERACTION

II. ELECTRICAL EQUIPMENT QUALIFICATION

III. FIRE PROTECTION

IV. GENERIC ISSUES

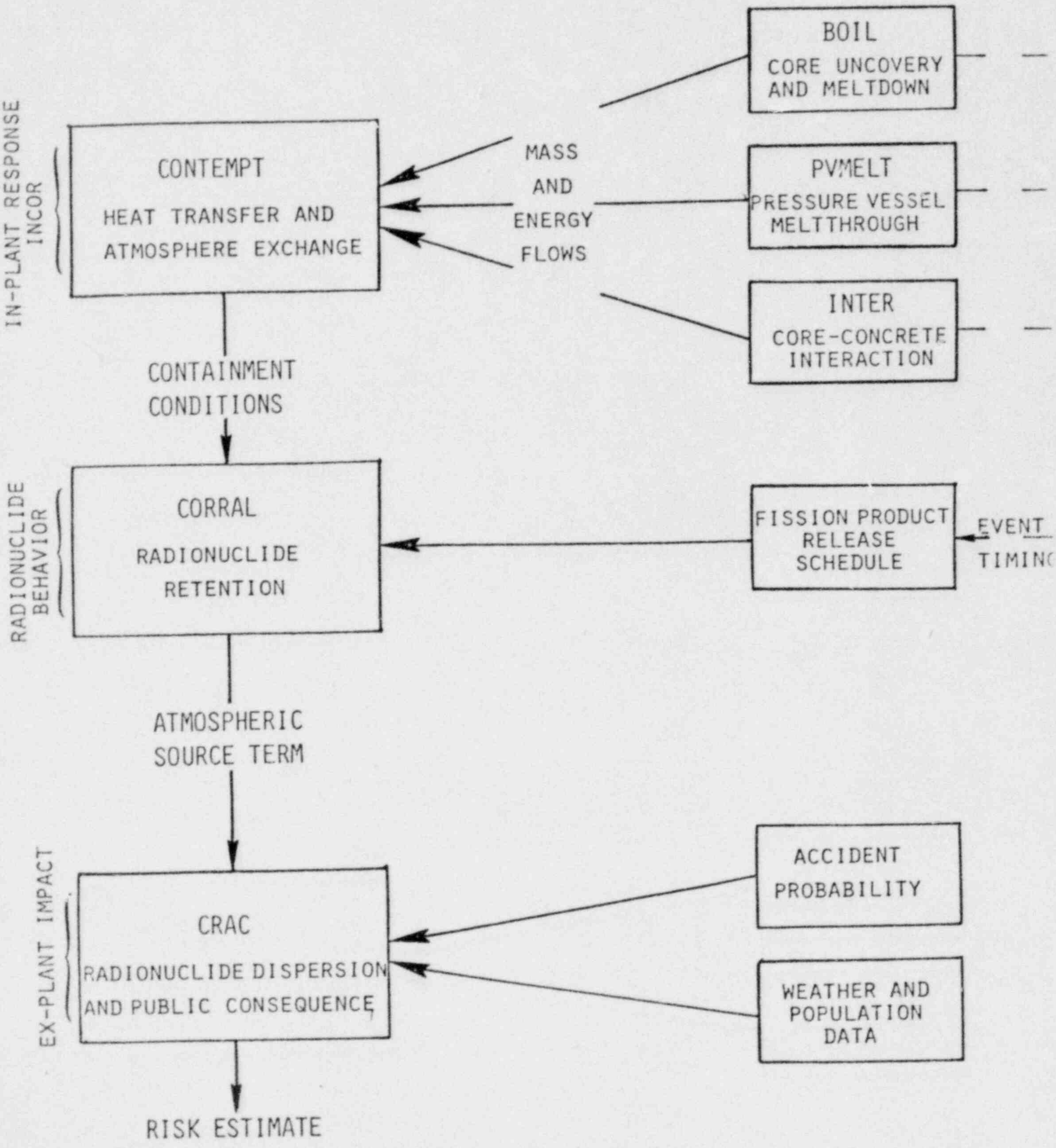
EXAMPLES: Onsite and Offsite Power Sources
Control of Heavy Loads (NUREG-0612)
Containment Integrity
ATWS

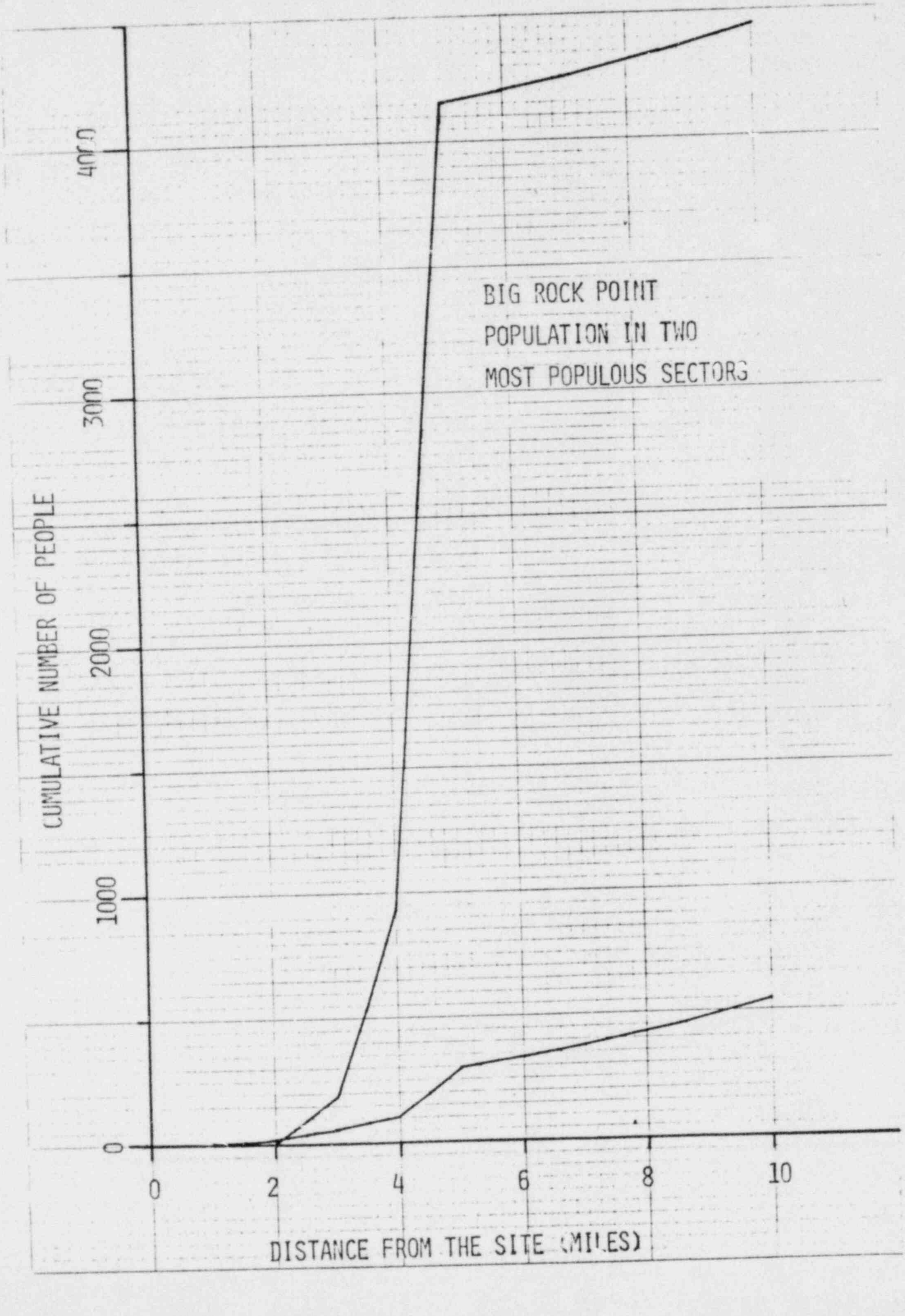
V. IE BULLETINS

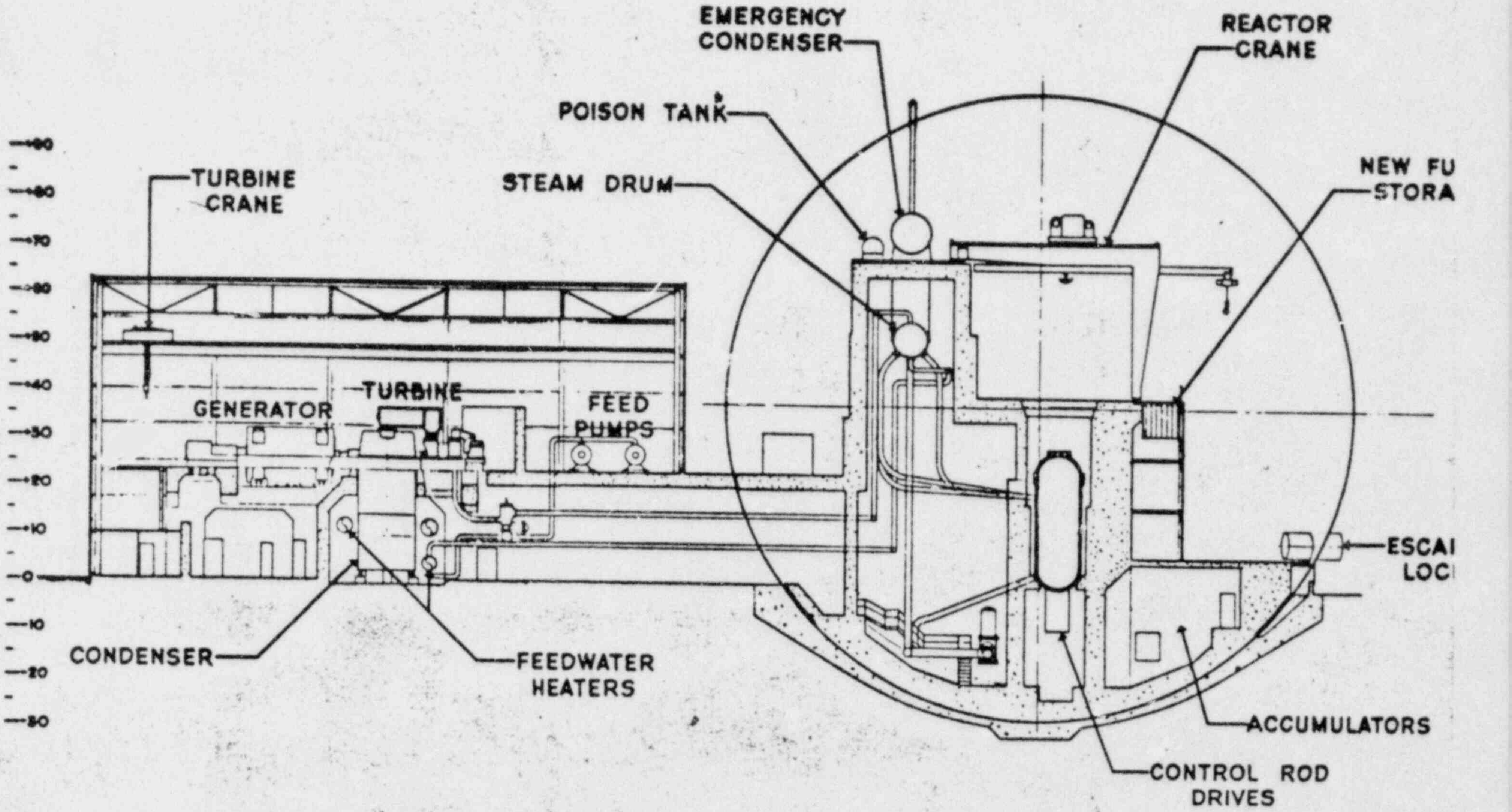
EXAMPLES: IE Bulletin 80-17 - Browns Ferry Failure To Scram
IE Bulletin 81-01 - Mechanical Snubber Reliability

ACCIDENT CONSEQUENCE ANALYSIS

RACAP CODE NETWORK

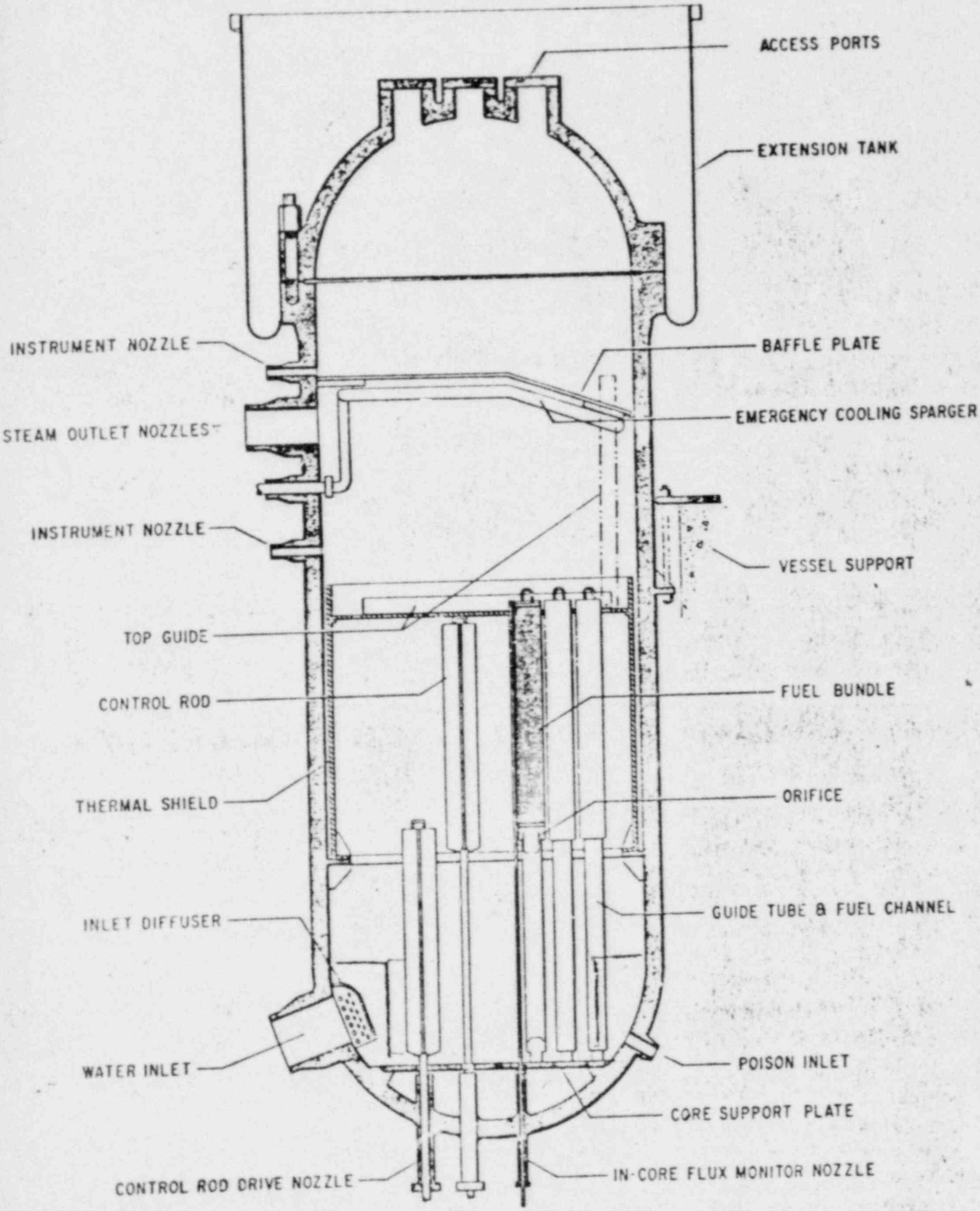



BIG ROCK POINT
POPULATION IN TWO
MOST POPULOUS SECTORS
CUMULATIVE NUMBER OF PEOPLE
4000
3000
2000
1000
0
0
2
4
6
8
10
DISTANCE FROM THE SITE (MILES)



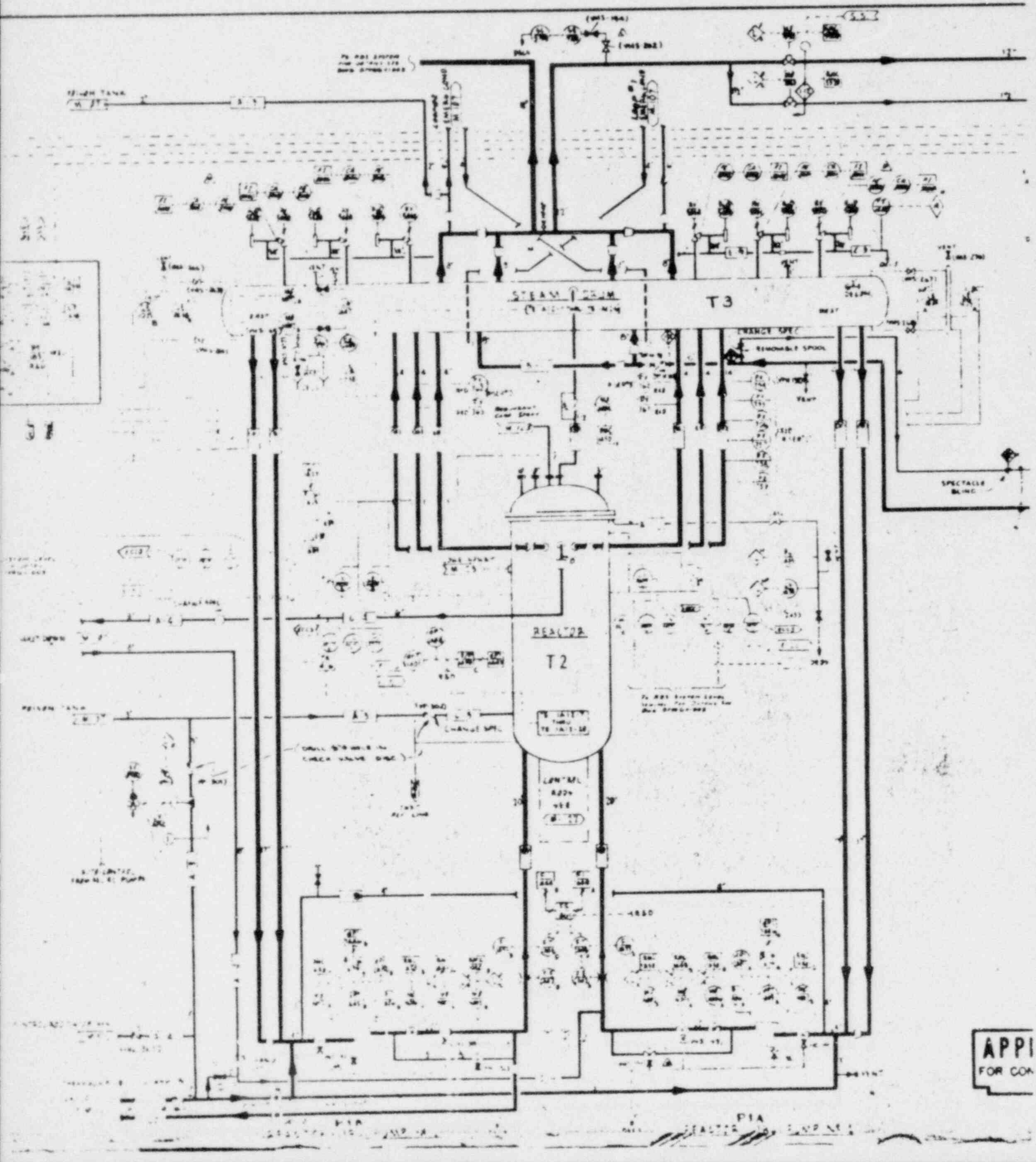
TURBINE BUILDING

REACTOR BUILDING



REACTOR VESSEL SCHEMATIC

POOR ORIGINAL



EMERGENCY CONDENSER SYSTEM

POOR ORIGINAL

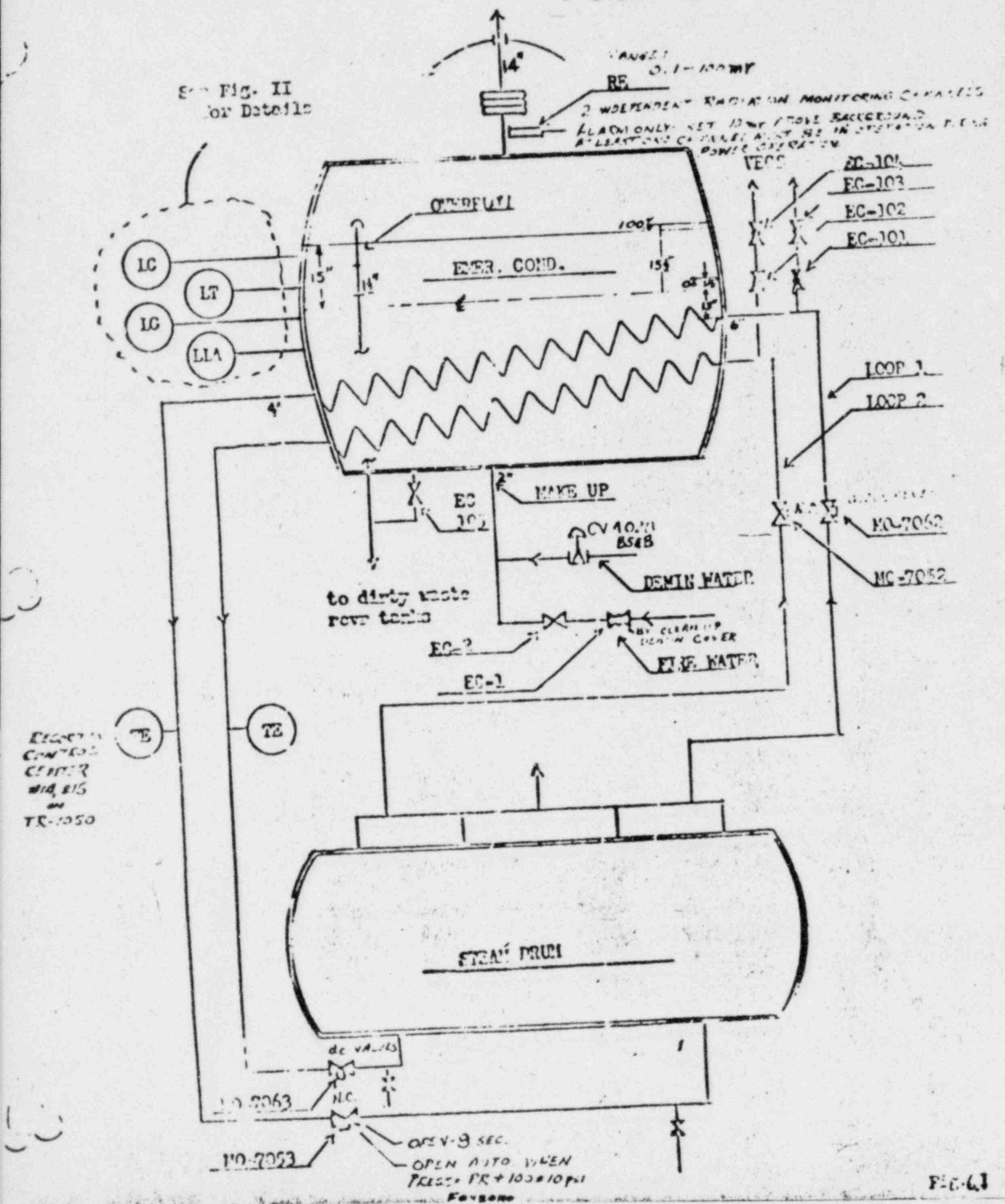
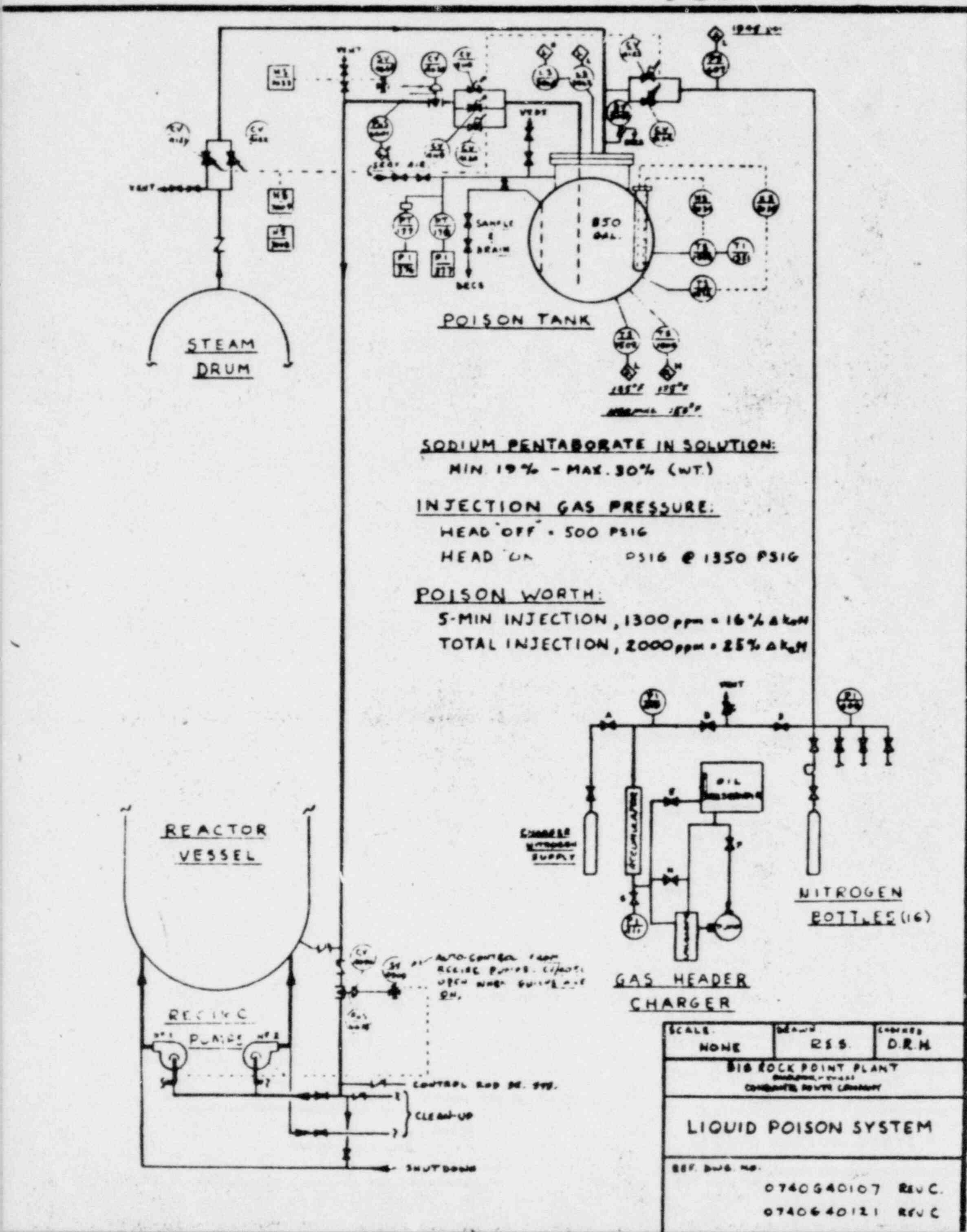
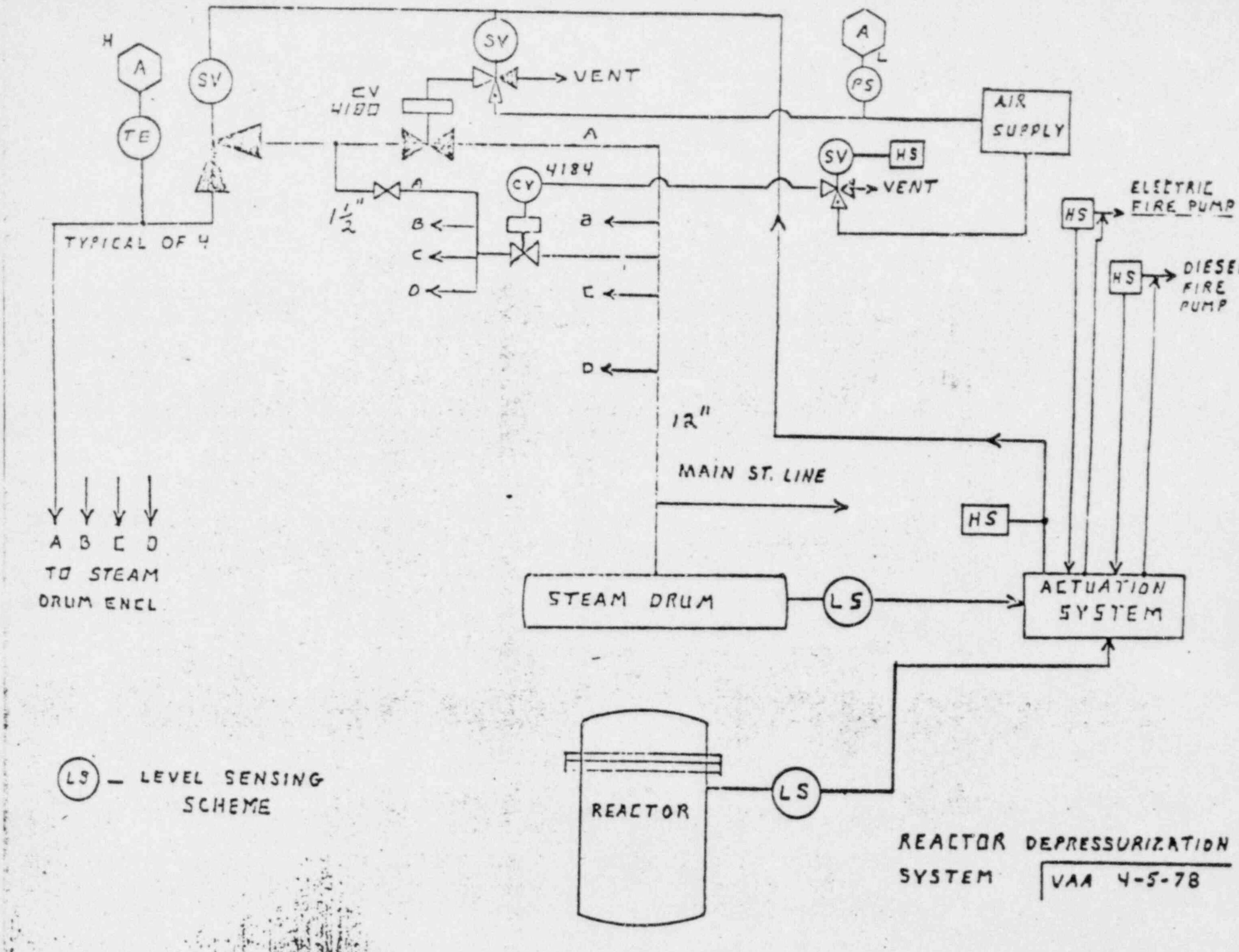


Fig. 6.1

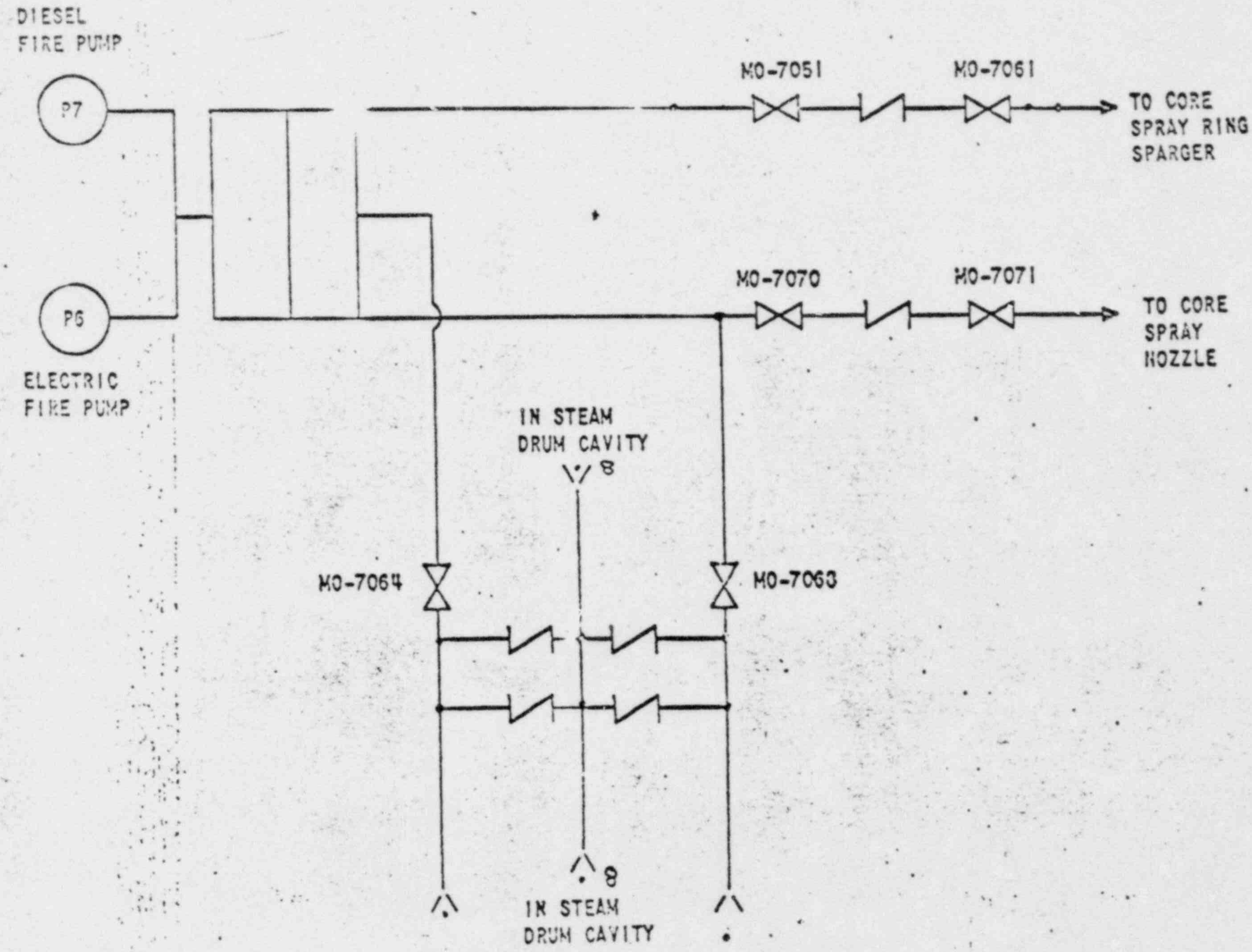
POOR ORIGINAL





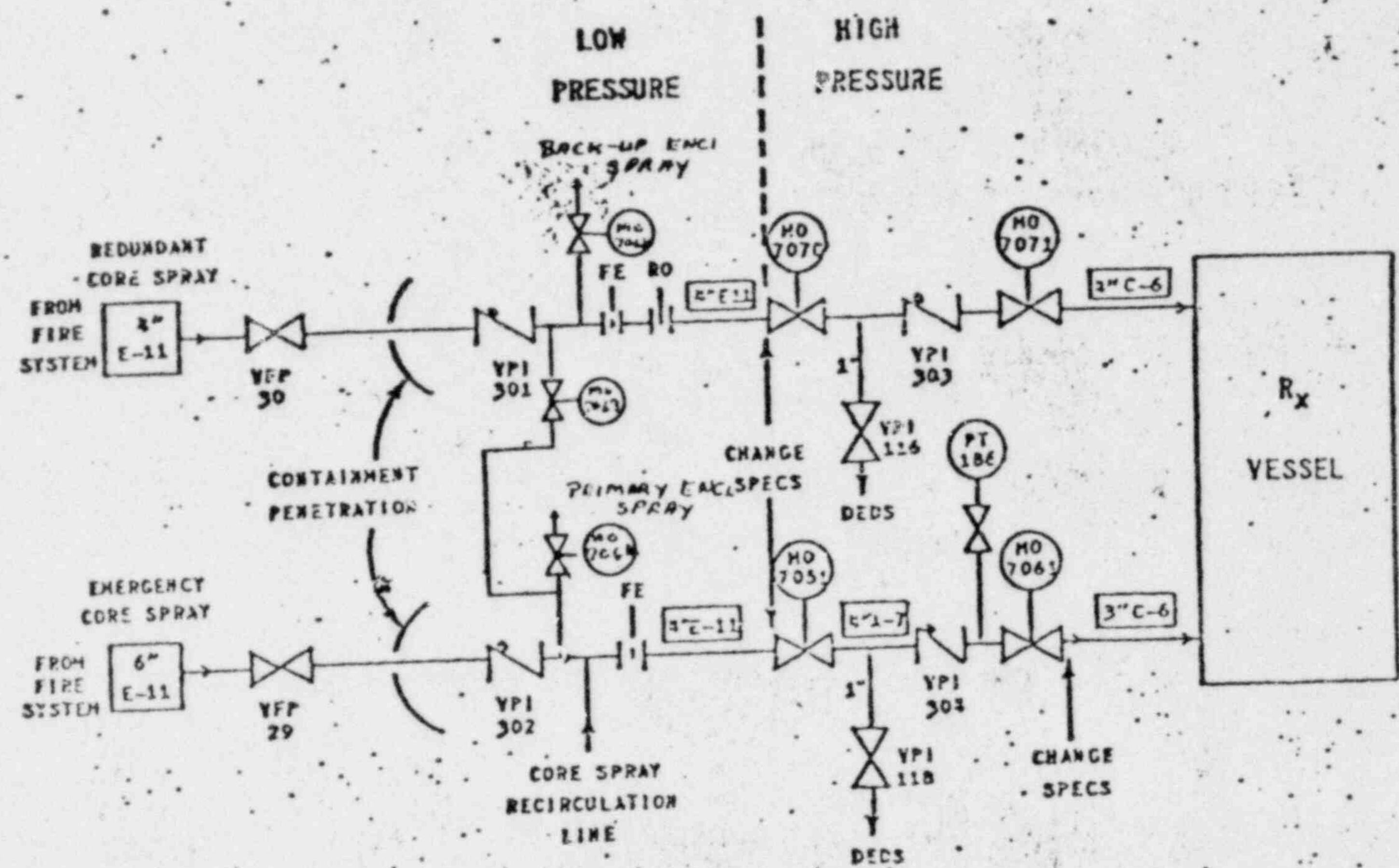
REACTOR DEPRESSURIZATION SYSTEM

VAA 4-5-78


DIESEL FIRE PUMP
P7
P6
ELECTRIC FIRE PUMP
MO-7051
MO-7061
TO CORE SPRAY RING SPARGER
MO-7070
MO-7071
TO CORE SPRAY NOZZLE
IN STEAM DRUM CAVITY
8
MO-7064
MO-7063
8
IN STEAM DRUM CAVITY

POOR ORIGINAL

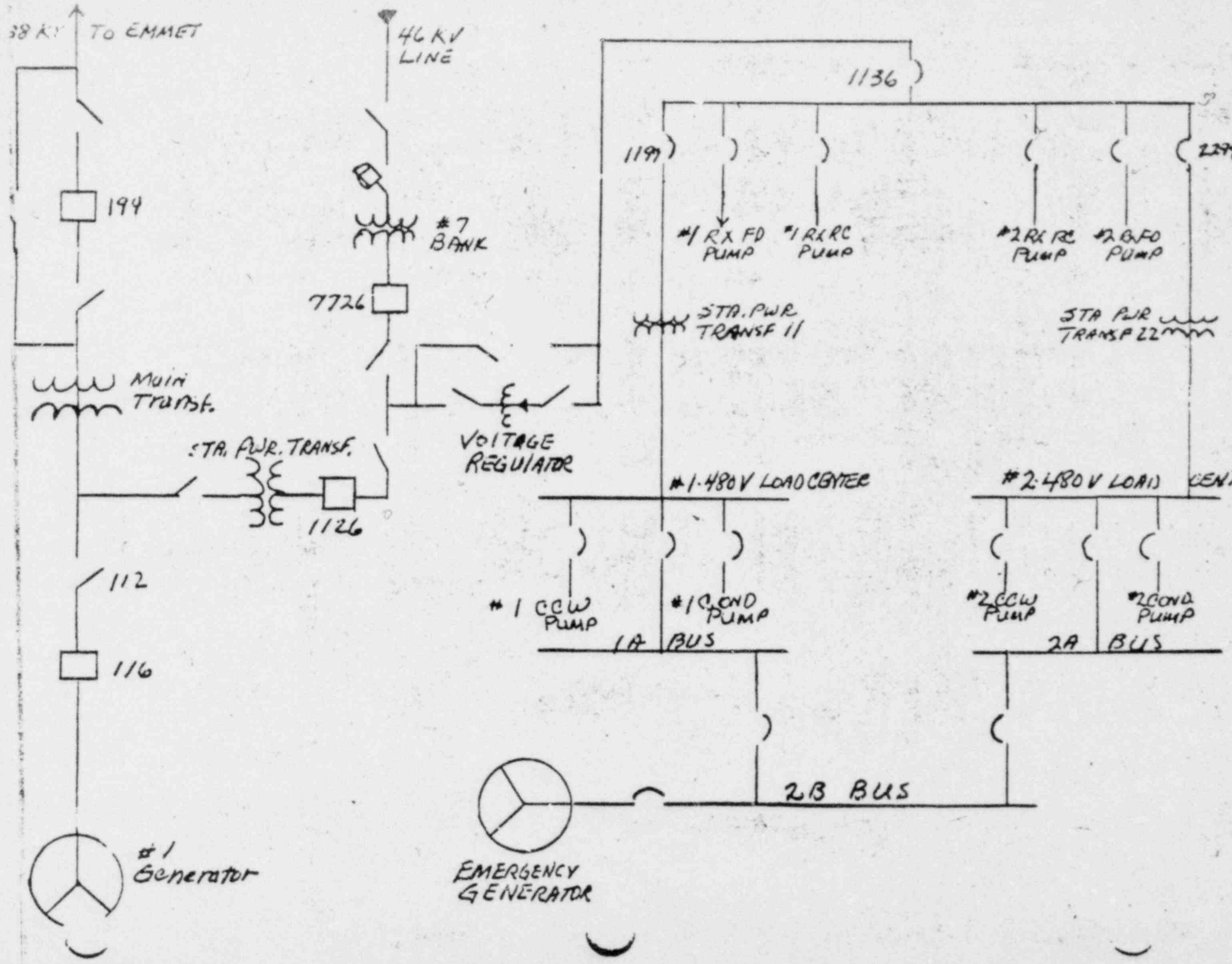
EMERGENCY CORE SPRAY AND REDUNDANT CORE SPRAY SYSTEMS



C-6: MATERIAL SS 304
PRIMARY RATING 1500 psig @ 1000° F

E-11: MATERIAL CARBON STEEL
PRIMARY RATING 150 psig @ 500° F

A-7: MATERIAL CARBON STEEL
PRIMARY RATING 1500 psig @ 850° F



138 KV TO EMMET
46 KV LINE
1136
1191
221
#1 RX FD PUMP
#1 RCRC PUMP
#2 RCRC PUMP
#2 RXFD PUMP
194
#7 BANK
7726
STA. PWR TRANSF 11
STA PWR TRANSF 22
Main Transf.
Voltage Regulator
STA. PWR. TRANSF.
1126
#1-480V LOAD CENTER
#2-480V LOAD CENTER
112
#1 CCW PUMP
#1 COND PUMP
#2 CCW PUMP
#2 COND PUMP
1A BUS
2A BUS
116
2B BUS
#1 Generator
EMERGENCY GENERATOR

FIGURE QUALITATIVE COMPARISON OF BIG ROCK POINT RISK WITH DECISION RULES PROPOSED IN NUREG-0739

LIMITS ON OCCURRENCE OF HAZARD STATE

HAZARD STATE	DECISION RULE ON MEAN FREQUENCY GOAL LEVEL	UPPER LIMIT	BIG ROCK POINT PRE-MOD. STATUS	BIG ROCK POINT POTENTIAL POST MOD. STATUS
SIGNIFICANT CORE DAMAGE	$<3x10^{-4}$/RY	$<1x10^{-3}$/RY	~~BELOW GOAL~~ 1×10^{-3}/YR AT LIMIT	~~BELOW GOAL~~ 8.4×10^{-5} BELOW GOAL
LARGE SCALE FUEL MELT (LSFM)	$<1x10^{-4}$/RY	$<5x10^{-4}$/RY	ABOVE ~~ABOVE~~ LIMIT 1×10^{-3}/YR	BELOW GOAL 8.4×10^{-5}
LARGE SCALE UNCONTROLLED RELEASE FROM CONTAINMENT [GIVEN LSFM] (1)	<0.01	<0.1	ABOVE LIMIT ~0.4	~~BETWEEN GOAL AND LIMIT FOR MOST SEQUENCES (1)~~ ABOVE LIMIT ~0.3 (INCLUDING LEAKAGE)

(1) THIS DECISION SEEMS TO BE ARBITRARY, OPEN TO INTERPRETATION, AND LIKELY UNACCEPTABLE BECAUSE IT IS SO STRONGLY RELATED TO THE SEQUENCE CHARACTERISTICS AND INDEPENDENT OF THE SEQUENCE PROBABILITY.

FIGURE QUALITATIVE COMPARISON OF BIG ROCK POINT RISK WITH DECISION RULES PROPOSED IN NUREG-0739

LIMITS ON RISK TO MOST EXPOSED INDIVIDUAL (1)

PROBABILITY GOAL	DECISION RULE ON MEAN FREQUENCY PER SITE-YEAR: GOAL LEVEL	DECISION RULE ON MEAN FREQUENCY PER SITE-YEAR: UPPER LIMIT	BIG ROCK POINT PRE-MOD. STATUS	BIG ROCK POINT POTENTIAL POST-MOD. STATUS
INDIVIDUAL PROBABILITY OF DELAYED CANCER DEATH (MOST EXPOSED PERSON)	$<5x10^{-6}$/SITE-YEAR	$<2.5x10^{-5}$/SITE-YEAR		
PROBABILITY OF EARLY DEATH (MOST EXPOSED PERSON)	$<1x10^{-6}$/SITE-YEAR	$<5x10^{-6}$/SITE-YEAR	BELOW GOAL	BELOW GOAL

(1) DECISION RULES ON MEAN FREQUENCY PER LARGE SCALE FUEL MELT HAVE NOT YET BEEN ESTIMATED.

FIGURE QUALITATIVE COMPARISON OF BIG ROCK POINT RISK WITH DECISION RULES PROPOSED IN NUREG-0739

SOCIETAL HEALTH RISK LIMITS				
	DECISION RULES			
MEASURE OF RISK	GOAL LEVEL	UPPER LIMIT	BIG ROCK POINT PRE-MOD. STATUS	BIG ROCK POINT POTENTIAL POST-MOD. STATUS
EXPECTED VALUE OF DELAYED CANCER DEATHS	<2 PER 10^{10} KWh	<10 PER 10^{10} KWh	BELOW GOAL	BELOW GOAL
EXPECTED FREQUENCY OF EARLY DEATHS (RAISED TO THE 1.2 POWER)	<0.4 PER 10^{10} KWh	<2 PER 10^{10} KWh	BELOW GOAL	BELOW GOAL

DATA COLLECTION FOR THE
BIG ROCK POINT NUCLEAR PLANT RISK ASSESSMENT

- PLANT SPECIFIC DATA COLLECTED FOR USE IN BIG ROCK POINT FAULT TREE QUANTIFICATION

- DATA USED TO DETERMINE COMPONENT FAILURE RATES, MAINTENANCE AND TEST UNAVAILABILITIES, REPAIR AND RESTORATION TIMES, AND OPERATOR ERROR PROBABILITIES

- SOURCES OF PLANT SPECIFIC DATA

  (1) Plant Maintenance Orders - plant MO's provided information on the cause of component failure and the time to accomplish repairs. The data was used to compute failure rates, failure modes and repair times.

  (2) Control Room Log Books - CRLB's are the daily operating history of the plant. They provide a history of the operation of equipment and actions taken by the operator during plant operations. From this source, data on equipment operation and outages, failure modes and failure rates can be obtained

  (3) Surveillance Tests- SRVT's are procedures by which safety related components and instrumentation can be tested against standards of normal operation. Information on failure rates, component operating history and test and maintenance unavailabilities can be obtained.

  (4) Licensee Event Reports, Event Reports, Deviation Reports, Plant Review Committee Meeting Minutes and NRC Correspondence - from these documents additional component failure information was obtained. They also provided additional insight on failures found in other sources which did not elaborate on the causes.

- DATA GATHERED FOR TEN-YEAR PERIOD 1970-1979

DETERMINATION OF COMPONENT FAILURE RATES

- COMPONENT FAILURE MODES SUMMARIZED FROM DATA
- COMPONENT DEMANDS AND OPERATING TIMES DETERMINED FROM DATA SOURCES
- ABOVE INFORMATION COMBINED TO DETERMINE COMPONENT FAILURE RATES

TABLE III-1

SUMMARY OF BIG ROCK POINT COMPONENT FAILURES AND OUTAGES

EMERGENCY DIESEL GENERATOR

DATE	COMPONENT	DESCRIPTION	SOURCE	FAILURE MODE
11/15/74	Emergency diesel generator	Tagged out; 1 hour	CRLB155	
11/22/74	Emergency diesel generator	Replaced fuel transfer pump; outage 2 hours	MØ74EPS527 MØ74EPS32413 CRLB151	
12/5/74	Emergency diesel generator	Selector switch to off; 15 minutes	CRLB156	
4/10/75	Emergency diesel generator	Failed to start; started after hand priming; approximately 30 minutes	AØ-9-75	[1]
7/17/75	Emergency diesel generator	Tagged out; 1 hour	CRLB163	
3/24/76	Emergency diesel generator	Tripped on high temperature; inlet screen plugged; estimated outage 9 hours	BRP1-050155-032476	FTØ
5/13/76	Emergency diesel generator	Tripped after running 1 hour and 15 minutes on high temperature; cleaned suction screen; outage 9 hours	BRP1-050155-051676 MØ76EPS12806	FTØ
5/17/76	Emergency diesel generator	No indication of cooling water flow after start; diesel shutdown; repacked coolant pump; cleaned suction screen; outage 12.5 hours	CRLB173 BRP1-050155-057877 MØ76EPS13806 MØ76EPS13901	FTØ
6/5/76	Emergency diesel generator	No voltage indication; outage 6.5 hours	CRLB173 MØ76EPS1009	FTS
8/5/76	Emergency diesel generator	Did not start within 15 seconds; not retested to see if it would start	ER-B-76-22	FTS
8/12/76	Emergency diesel generator	Failed to start; battery cable burned off; outage 1 hour	ER-B-76-22 MØ76EPS22508	FTS

SUMMARY OF BIG ROCK POINT COMPONENT FAILURES AND OUTAGES

PUMPS

DATE	COMPONENT	DESCRIPTION	SOURCE	FAILURE MODE
7/6/73	#1 Control rod drive pump	Replaced relief valve; outage 1 hour	MØ73CRD408 CRLB137	
9/21/73	#1 Control rod drive pump	Tagged out; reason unknown; outage 4 hours	CRLB140	
9/24/73	#1 Control rod drive pump	Tagged out; 6.5 hours	CRLB140	
9/25/73	#1 Control rod drive pump	Tagged out; 11.5 hours	CRLB140	
9/27/73	#1 Control rod drive pump	Tagged out; 16.5 hours	CRLB140	
11/5/73	#1 Control rod drive pump	Relief valve lifting; valve replaced; outage 1 hour	CRLB414 MØ73CRD692	FTØ
1/14/75	#1 Control rod drive pump	Leak in discharge elbow; outage 25 hours	MØ74CRD029 CRLB144	FTØ
1/26/74	#1 Control rod drive pump	Repacked pistons; outage 4 hours	MØ74CRD058 CRLB144	
5/31/74	#1 Control rod drive pump	Pump packing blown; est. outage 1 hour	MØ74CRD33006	FTØ
7/31/74	#1 Control rod drive pump	Packing leak; outage 3 hours	MØ74CRD419 CRLB152	
8/15/74	#1 Control rod drive pump	Broken valve spring replaced; outage 2.5 hours	MØ74CRD447	FTØ

SUMMARY OF BIG ROCK POINT COMPONENT FAILURES AND OUTAGES

CONTROL VALVES

DATE	COMPONENT	DESCRIPTION	SOURCE	FAILURE MODE
[illegible]/17/77	CV4094	Failed to close during test	CRLB190	FTC
[illegible]/17/77	CV4095	Failed to close during test	CRLB190	FTC
10/10/79	CV4094 & CV4095	Failed leak rate test	Surv. test T180-01 part A	IL
[illegible]/31/75	CV4096	Failed to seat properly; packing misadjusted	AØ -8-75	IL
6/8/77	CV4096	Replaced limit switch	MØ77CIS15802	
[illegible]/5/78	CV4096 & CV4097	Failed pressure test	CRLB202	IL
[illegible]/26/74	CV4097	Flange leak; bolts tightened	BRP1-050155-042674	XL
[illegible]/21/75	CV4097	Failed to seat during test	MØ75CIS07901	IL
5/26/75	CV4097	Failed pressure test	AØ-14-75	IL
[illegible]/17/75	CV4097	Failed leak rate test	BRP1-050155-04177	IL
[illegible]/19/76	CV4097	Failed leak rate test	BRP1-050155-06197	IL
[illegible]/20/78	CV4097	Failed leak rate test	BRP1-050155-012078	IL
[illegible]/21/78	CV4097	Failed to open on first attempt; opened on second	MØ78CIS02303 D-BRP-78-09	FTØ
[illegible]/25/78	CV4097	Spurious closure; failed to reopen	MØ78CIS02506	FTRØ

SUMMARY OF BIG ROCK POINT COMPONENT FAILURES AND OUTAGES

MOTOR OPERATED VALVES

DATE	COMPONENT	DESCRIPTION	SOURCE	FAILURE MODE
10/30/78	MØV7062 (Emergency Condenser Inlet Valve)	Failed to close by remote manual controller; valve stem cleaned	MØ78ECS30303	FTC
12/10/79	MØ7062	Valve failed to open after hand tightened against backseat; estimated outage 93 hours	E-BRP-79-41	FTØ
11/11/73	MØ7053 (Emergency Condenser Outlet Valve)	Emergency condenser outlet valve would not close	CRLB141	FTC
11/14/73	MØ7053	Would not close	CRLB141	FTC
4/5/73	MØ7053	Failed to open; adjusted packing	AØ-8-73	FTØ
6/5/78	MØ7053	Was inoperable after hand tightened against backseat; estimated outage 2184 hours	BRP1-050155-060578	
1/25/72	MØ7063 (Emergency Condenser outlet valve)	Failed to close; motor burned out after improperly set torque switch, motor replaced	March 3, 1972 letter to AEC CRLB117	FTC
10/23/75	MØ7063	Control pulled to stop for work on RE07B; 15 minutes	CRLB165	
11/11/75	MØ7056, 7057, 7058, 7059 (Shutdown cooling system primary and secondary valves)	Shutdown inlet and outlet valves control placed in pull to stop position; 3 hours	CRLB166	
8/27/77	MØ7051 (Core Spray Valve)	Failed to close	D-BRP-77-104	FTC

TABLE III-3

FAILURE RATES COMPUTED FROM BIG ROCK POINT SPECIFIC DATA

COMPONENT	FAILURE MODE	TOTAL FAILURES	TOTAL DEMANDS	TOTAL OPERATION	FAILURE RATE
1. Emergency Diesel Generator	failure to start	12	669		1.79×10^{-2}/d
	failure to run	7		355	1.97×10^{-2}/hr
2. CRD Pump	failure to run	13		67894	1.91×10^{-4}/hr
3. Feedwater Pump	failure to start	4	297		1.34×10^{-2}/d
	failure to run	8		119520	6.69×10^{-5}/hr
4. Service Water Pump				87648	
5. Condenser Circulating Water Pump	failure to start	3	209		1.43×10^{-2}/d
	failure to run	3		119520	2.5×10^{-5}/hr
6. Demineralized Water Pump	failure to run	3		44820	6.69×10^{-5}/hr
7. Reactor Cleanup Pump	failure to run	18		59760	3.01×10^{-4}/hr
8. Shutdown Cooling Pump	failure to run	1		27888	3.58×10^{-5}/hr
9. Condensate Pump	failure to start	1	462		2.16×10^{-3}/d
	failure to run	2		119520	1.67×10^{-5}/hr
10. Reactor Cooling Water Pump	failure to run	1		87648	1.14×10^{-5}/hr
11. Fuel Pit Pump	failure to start	1	572		1.74×10^{-3}/d
	failure to run	11		87648	1.25×10^{-4}/hr
12. Electric Fire Pump	failure to start	2	355	399	5.63×10^{-3}/d
13. Diesel Fire Pump	failure to start	1	326	146	3.06×10^{-3}/d

TABLE III-3

FAILURE RATES COMPUTED FROM BIG ROCK POINT SPECIFIC DATA

COMPONENT	FAILURE MODE	TOTAL FAILURES	TOTAL DEMANDS	TOTAL OPERATION	FAILURE RATE
14. MSIV (MØ7050)	failure to close	2	52		3.84×10^{-2}/d
15. CV4014 (Turbine Bypass Valve)	failure to open	4	28		1.42×10^{-1}/d
16. Control Valves	failure to open	16	750		2.13×10^{-2}/d
	failure to close	16	621		2.57×10^{-2}/d
	failure to remain open	4		66 @ 59760	1.01×10^{-6}/hr
	failure to remain closed	3		66 @ 59760	7.6×10^{-7}/hr
17. Motor Operated Valves	failure to open	7	989	21 @ 59760	7.07×10^{-3}/d
	failure to close	10	639		1.56×10^{-2}/d
	failure to remain closed	1		1254970	8.83×10^{-7}/hr
18. Core Spray Valves (MØ7061, 7062, 7070, & 7071)	failure to open	3	230	4 @ 59760 / 158832	1.3×10^{-2}/d
19. Emergency Condenser Valves	failure to open	2	125		1.6×10^{-2}/d
20. [illegible] Isolation Valves (CV4025, 4027, 4031, 4091, 4092, 4093, 4094, 4095, 4096, 4097, 4102, 4103, 4117, 4200, MØ7050, 7065, 7067, VFW9, VRW304)	failure to isolate	19	227		8.37 x 10[illegible]
	failure of leak test	16	160		.10/d
21. RDS Isolation Valve	failure to open	1	215	15240	4.65×10^{-3}/d

GENERIC DATA USED IN BIG ROCK POINT RISK ASSESSMENT

- GENERIC DATA WAS USED WHERE PLANT SPECIFIC DATA WAS EITHER UNAVAILABLE OR CONSIDERED INAPPROPRIATE
- SOURCES OF GENERIC DATA

(1) WASH-1400, REACTOR SAFETY STUDY, AUGUST 1974
(2) GE RECOMMENDED FAILURE RATES, GE-22A2589, MAY 1974
(3) IEEE-500, COMPONENT RELIABILITY DATA, 1977
(4) NUREG/CR-1363, DATA SUMMARIES OF LER's, JUNE 1980
(5) NUREG/CR-1205, DATA SUMMARIES OF LER's OF PUMPS, JANUARY 1980

- ENGINEERING JUDGEMENT USED TO DETERMINE RECOMMENDED VALUE FOR A PARTICULAR COMPONENT

Table III-4a

GENERIC DATA - VALVES

EVENT IDENTIFICATION	SOURCES									
	1. WASH 1400		2. GE	3. IEEE-500[1]	4. CRNL-704[2]	5. AI[3]	6. EGG		*Recommended	
VALVES	λ[4]	Q[5]	λ	λ	λ	λ	λ	Q	λ	Q
Motor Operated(MOV)										
FTO	-	1.0	1.5	2.5			1.0	-	1.5	1.0
FTC	-	1.0	1.6	2.5	[L=0.5		0.8	-	1.6	1.0
FTRO	-	0.1	0.15	0.124	U=30]	[3.4]	-	-	0.15	0.1
FTRC	-	0.1	0.16	0.124			-	-	0.16	0.1
Safety										
FTO	-	0.01	1.1	-	[L=1		-	8/d[6]	1.1	8
FTC	-	-	2.7	-	U=10]	[11.4]	3.0	-	2.7	1.2[7]
FTRC	10	-	0.42	-			-	3/d	0.42	-
Check										
FTO	-	0.1	0.16	-	[L=0.7		0.08	0.1	0.08	0.1
FTC	-	-	-	-	U=10]		0.3	0.5	0.3	0.5
Reverse Leak	0.3	-	1.6	-		[2.3]	0.5	-	0.5	-
Diaphragm										
FTO	-	0.3	2.1	-	-		1.0	-	2.1	0.3
FTC	-	0.3	2.1	-	-		0.8	-	2.1	0.3
FTRO	-	0.1	-	-	-	[2.0]	-	-	(0.15)	0.1
FTRC	-	0.1	-	-	-		-	-	(0.16)	0.1
Solenoid(SOV)										
FTO	-	1.0	-	3.9		-	1.0	-	3.9	1.0
FTC	-	1.0	-	3.9	[L=1	-	0.8	-	3.9	1.0
FTRO	-	-	0.15	0.14	U=40]	-	-	-	0.15	(0.1)
FTRC	-	-	0.15	0.14		-	-	-	0.15	(0.1)

(1) IEEE-500 data applies to actuation only
(2) CRNL-704 data includes upper(U) and lower(L) bounds; brackets indicate rate is for all failure modes
(3) AI data rates are for all failure modes (hence brackets)
(4) λ is hourly failure rate, number indicates failures per million hours
(5) Q is demand failure rate, number indicates failures per thousand demands
(6) Value for BWR relief valves
(7) From CE data

* Recommended *if* the data were to be applied to BRP; but, in most cases, actual BRP data were used.

MAINTENANCE AND TEST UNAVAILABILITIES USED IN BIG ROCK POINT RISK ASSESSMENT

- MAINTENANCE PHILOSOPHY AT BIG ROCK POINT IS TO REPAIR PROBLEMS AS THEY OCCUR
- ONLY MAINTENANCE OUTAGES WHICH OCCURRED WHILE THE PLANT WAS AT FULL POWER WERE CONSIDERED
- NO TEST UNAVAILABILITIES OF SAFETY SYSTEMS DURING FULL POWER OPERATION

TABLE III-5

MAINTENANCE UNAVAILABILITIES

COMPUTED FROM THE BIG ROCK POINT SPECIFIC DATA

COMPONENT	NUMBER OF COMPONENTS	TOTAL OUTAGE	MAINTENANCE UNAVAILABILITY
1. Emergency Diesel Generator	1	180	3.01×10^{-3}
2. CRD Pump	2	380	3.17×10^{-3}
3. Feedwater Pump	2	127	1.06×10^{-3}
4. Service Water Pump	2	24	2.0×10^{-4}
5. Condenser Circulating Water Pump	2	285	2.4×10^{-3}
6. Demineralized Water Pump	1	434	7.26×10^{-3}
7. Reactor Cleanup Pump	1	117	1.95×10^{-3}
8. Shutdown Cooling Pump	2	61	5.1×10^{-4}
9. Condensate Pump	2	58	4.85×10^{-4}
10. Reactor Cooling Water Pump	2	192	1.60×10^{-3}
11. Fuel Pit Pump	2	4.5	3.76×10^{-5}
12. Electric Fire Pump	1	48	8.03×10^{-4}
13. Diesel Fire Pump	1	8	1.33×10^{-4}
14. CV4014 (Turbine Bypass Valve)	1	151	2.52×10^{-3}
15. Motor Operated Valves	21	85	7.49×10^{-5}
16. Core Spray Valves (MØ70[illegible], 7061, 7070 & 7071)	4	10	6.2×10^{-5}/valve
17. Emergency Condenser Valves	4	2277	9.5×10^{-3}

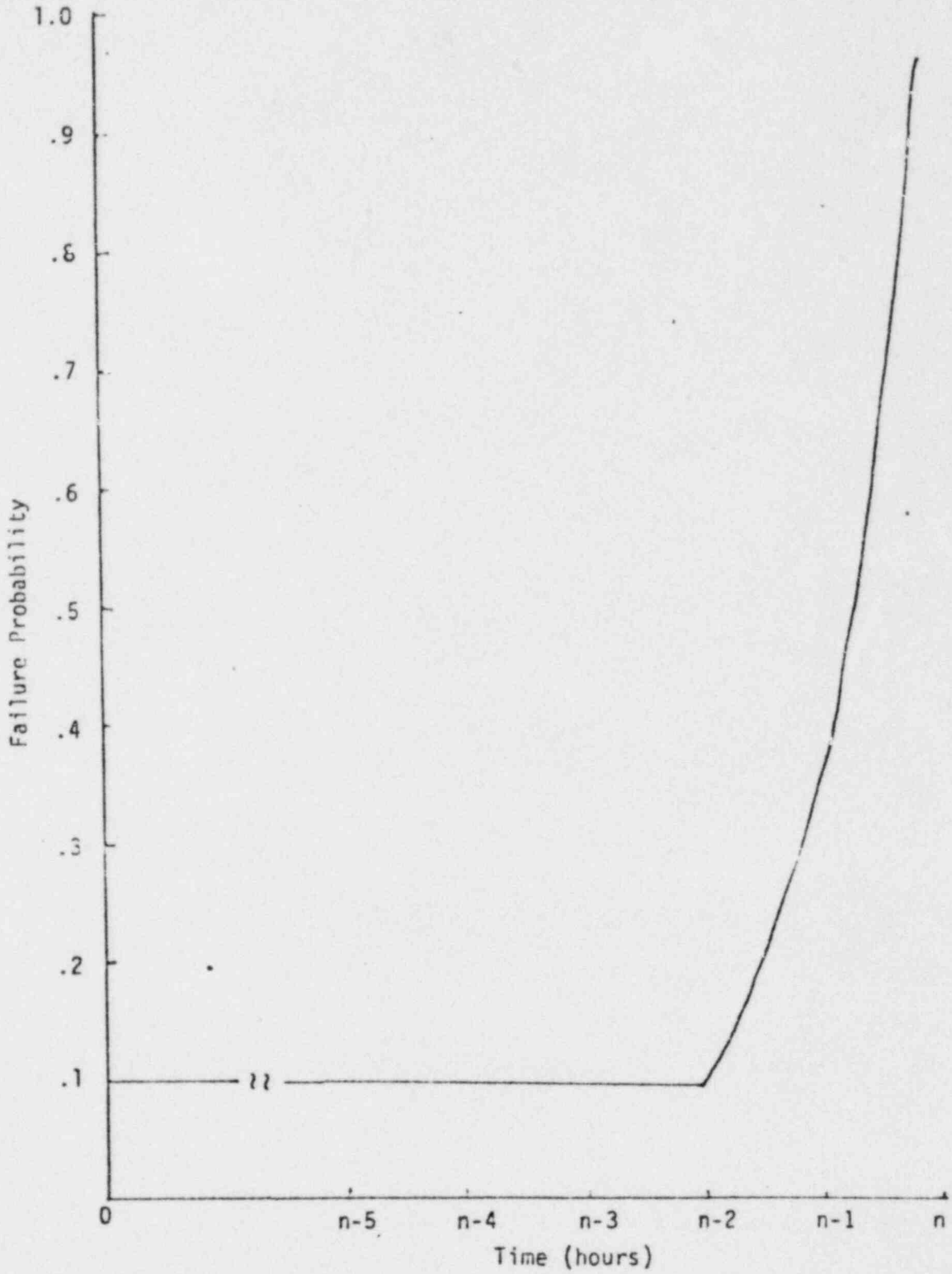
HUMAN ERROR PROBABILITIES
USED IN BIG ROCK POINT RISK ASSESSMENT

- MANY BACKUP SYSTEMS REQUIRE OPERATOR ACTION TO FUNCTION
- USED SWAIN AND GUTTMANN'S "HANDBOOK OF HUMAN RELIABILITY WITH EMPHASIS ON NUCLEAR POWER PLANT APPLICATIONS" AS FOUNDATION
- FACTORS WHICH DETERMINE HUMAN ERROR PROBABILITIES

  (1) EXPERIENCE
  (2) TRAINING
  (3) ADEQUATE PROCEDURES
  (4) STRESS
- MANY HUMAN ERROR PROBABILITIES USED IN BRP RISK ASSESSMENT HAD TO BE EVALUATED AS A FUNCTION OF TIME

Figure III. 2

Probability of failure to enter containment to open valve VEC-1 vs. time prior to safety valve actuation (in hours) at which possible need to open VEC-1 is recognized



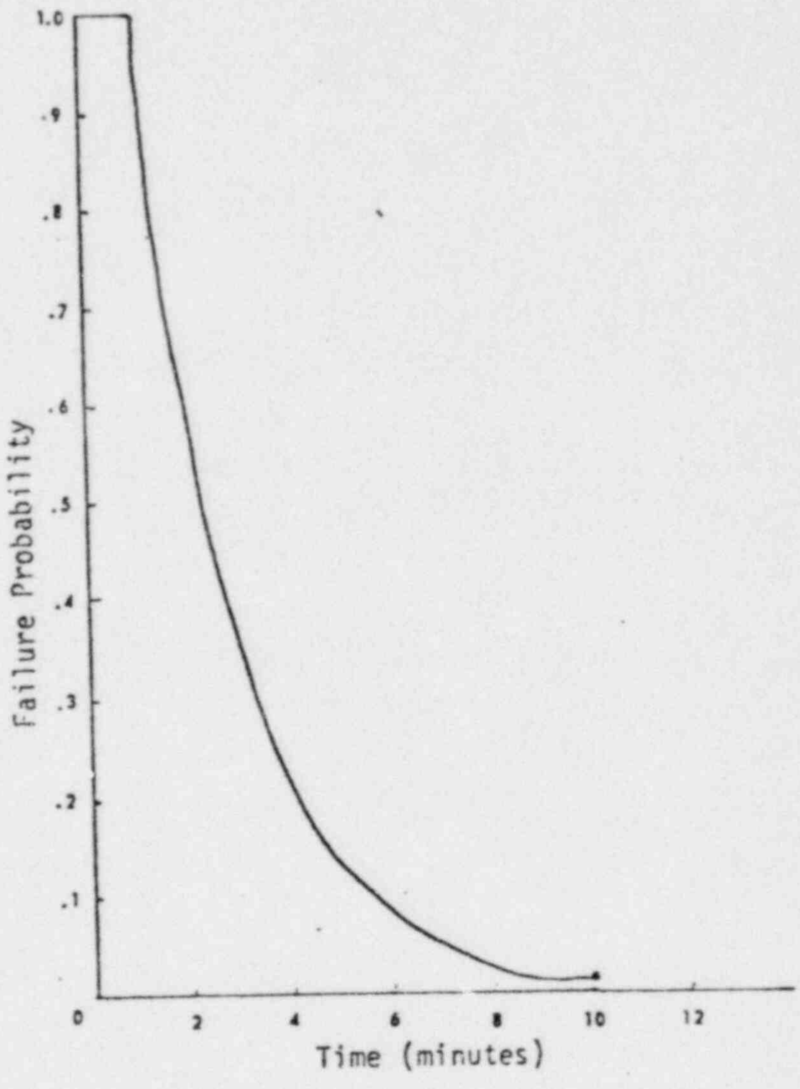


Figure III-3 Estimation of Failure to Initiate Liquid Poison Given an ATWS

TABLE III-7

SUMMARY OF HUMAN ERROR PROBABILITIES FOR BIG ROCK POINT

1. Operator performs an action for which there is no reason, i.e., closing a normally open valve necessary for flow or opening a normally closed circuit breaker necessary for power

   1×10^{-4}

2. Operator fails to open an MOV or start a pump when this action is necessary for successful system operation, a procedure is available, and location is familiar

   1×10^{-3} (low stress)

   2×10^{-3} (moderate stress)

3. Operator fails to return valves or components to service following maintenance or test

   1×10^{-3}

4. Operator fails to open an MOV, start a pump or close a circuit breaker when location is unfamiliar or not frequently used

   3×10^{-3} (low stress)

   6×10^{-3} (moderate stress)

5. Operator fails to close circuit breaker after LOSP when procedure is available but vague but sufficient indicators are available to alert him to action

   1×10^{-2}

6. Operator fails to shed loads and place demin. . .mp in operation when procedure is vague following LOSP.

   0.25

7. Operator fails to enter containment and open fire water valve for emergency condenser makeup

   See Figure III.2

8. Operator fails to place standby diesel generator in service before RDS operation takes place following a failure of the emergency diesel generator

   See Figure III.5

9. Operator fails to shed loads and place control rod drive pump on diesel generator with defined procedures following LOSP

   .1

REPAIR AND RESTORATION TIMES
USED IN BIG ROCK POINT RISK ASSESSMENT

- CUMULATIVE REPAIR DISTRIBUTIONS FORMULATED FROM BRP MAINTENANCE DATA
- REPAIR DATA FOR EMERGENCY DIESEL GENERATOR USED IN EVENT TREE QUANTIFICATION
- RESTORATION OF OFFSITE POWER USED IN EVENT TREE QUANTIFICATION

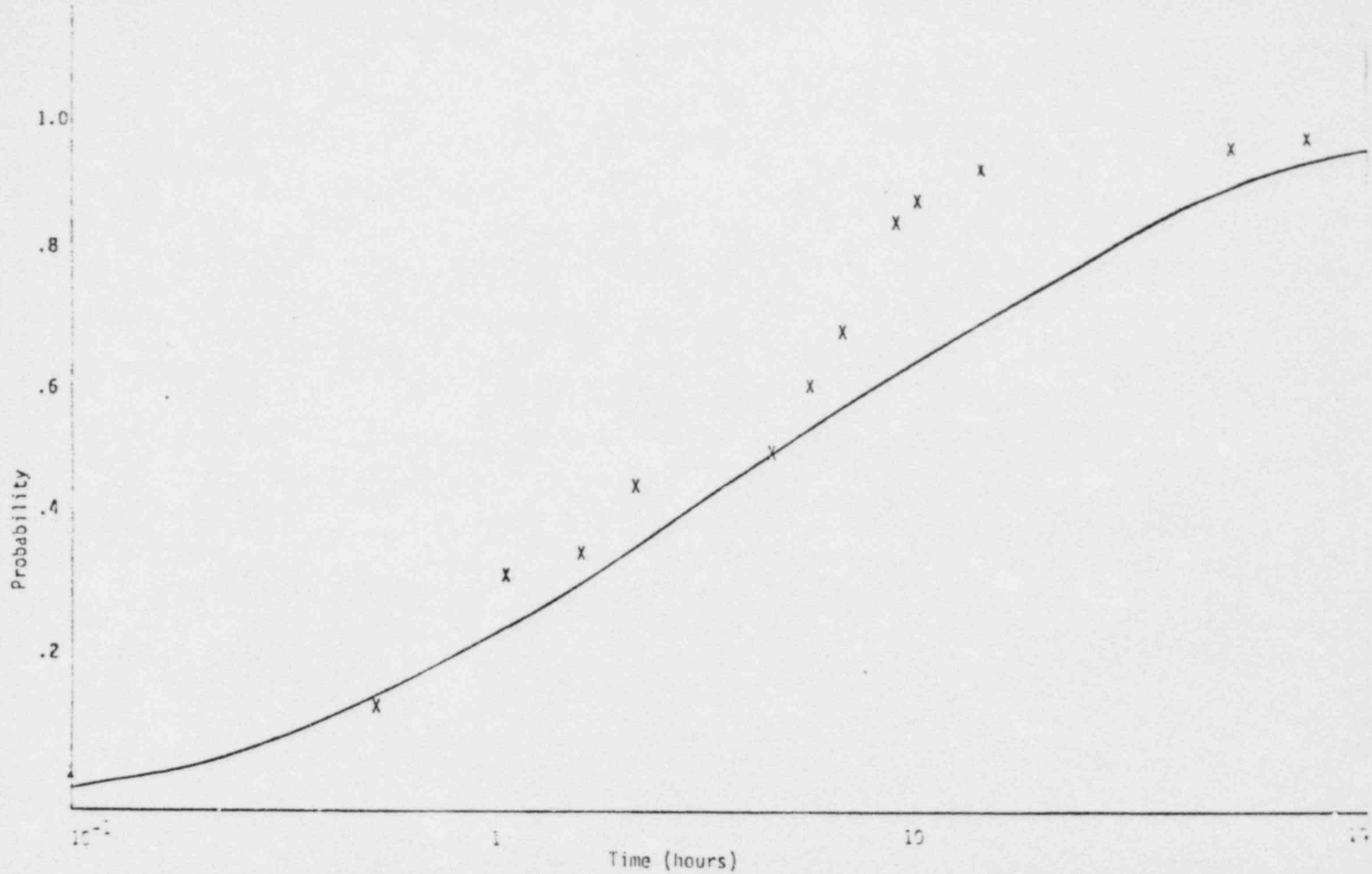


Figure III-6. Log Normal Distribution Fit to the Emergency Diesel Generator Data.

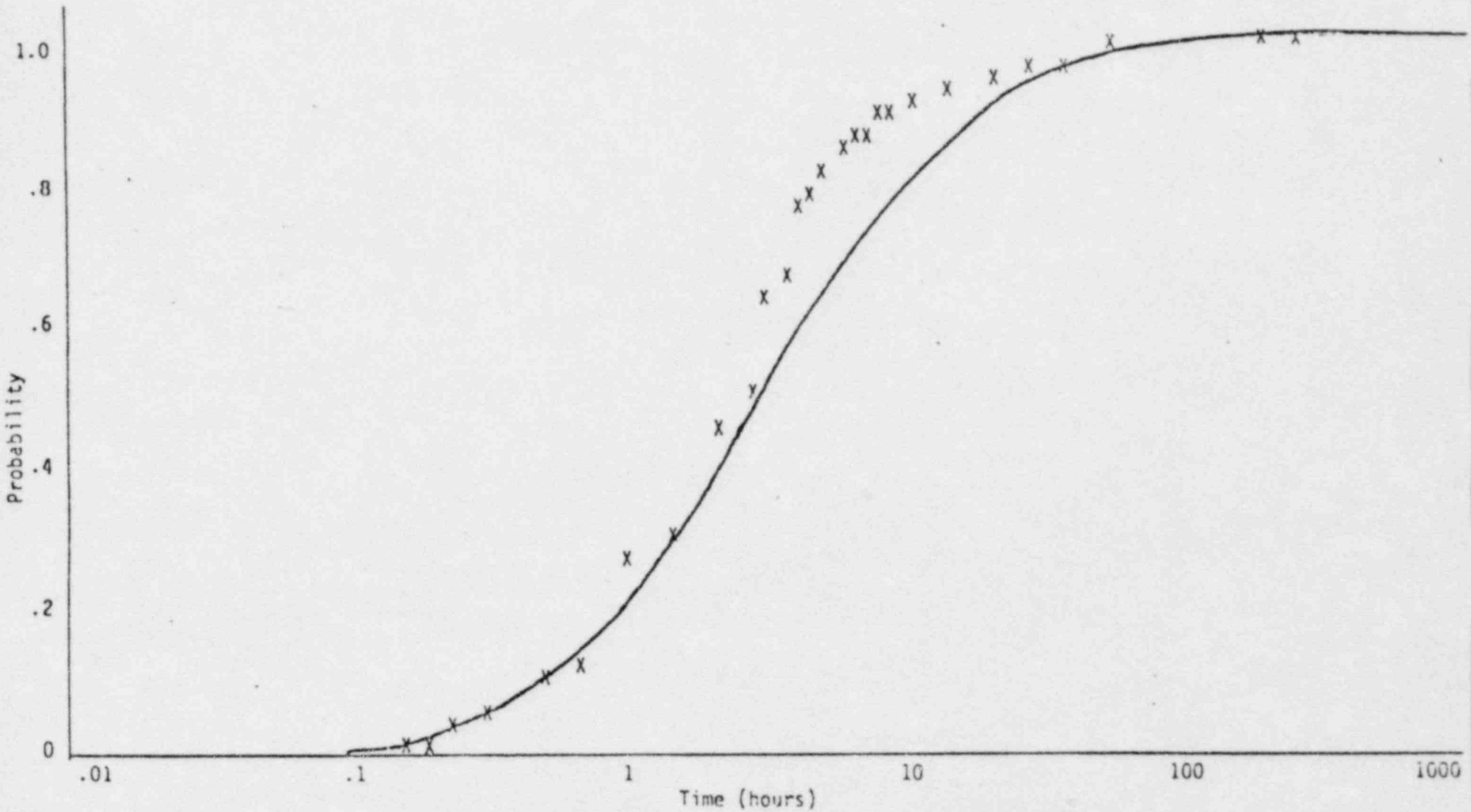


Figure III-7. Log Normal Distribution Fit to the Pump Repair Data

Table III-11

RAW DATA FOR LOSS OF OFFSITE POWER AT

THE BIG ROCK POINT SITE

DATE	DURATION (MINUTES)	
	46 KV LINE	138 KV LINE
5/2/69	< 5	
6/26/69	< 5	
8/4/69	< 5	
10/19/69	619	
7/8/70	< 5	
9/11/70	< 5	
10/2/70	< 5	
11/19/70	< 5	
12/3/70		(1)
8/22/71	187	
9/28/71	< 5	< 5
1/25/72	119	20
6/1/72	61	
11/3/72	< 5	
11/7/72		< 5
5/21/74	< 5	
7/22/75	< 5	
12/8/75	< 5	
2/2/76	596	
3/12/76	10	
5/5/76		< 5
10/21/76	< 5	
8/31/77		201
4/6/78		63
4/17/78		142
5/31/78		8
4/6/79		1329
4/12/79	75	
6/6/79	817	
9/26/79		< 5
10/10/79	94	

(1) Nine power interruptions occurred on this date.

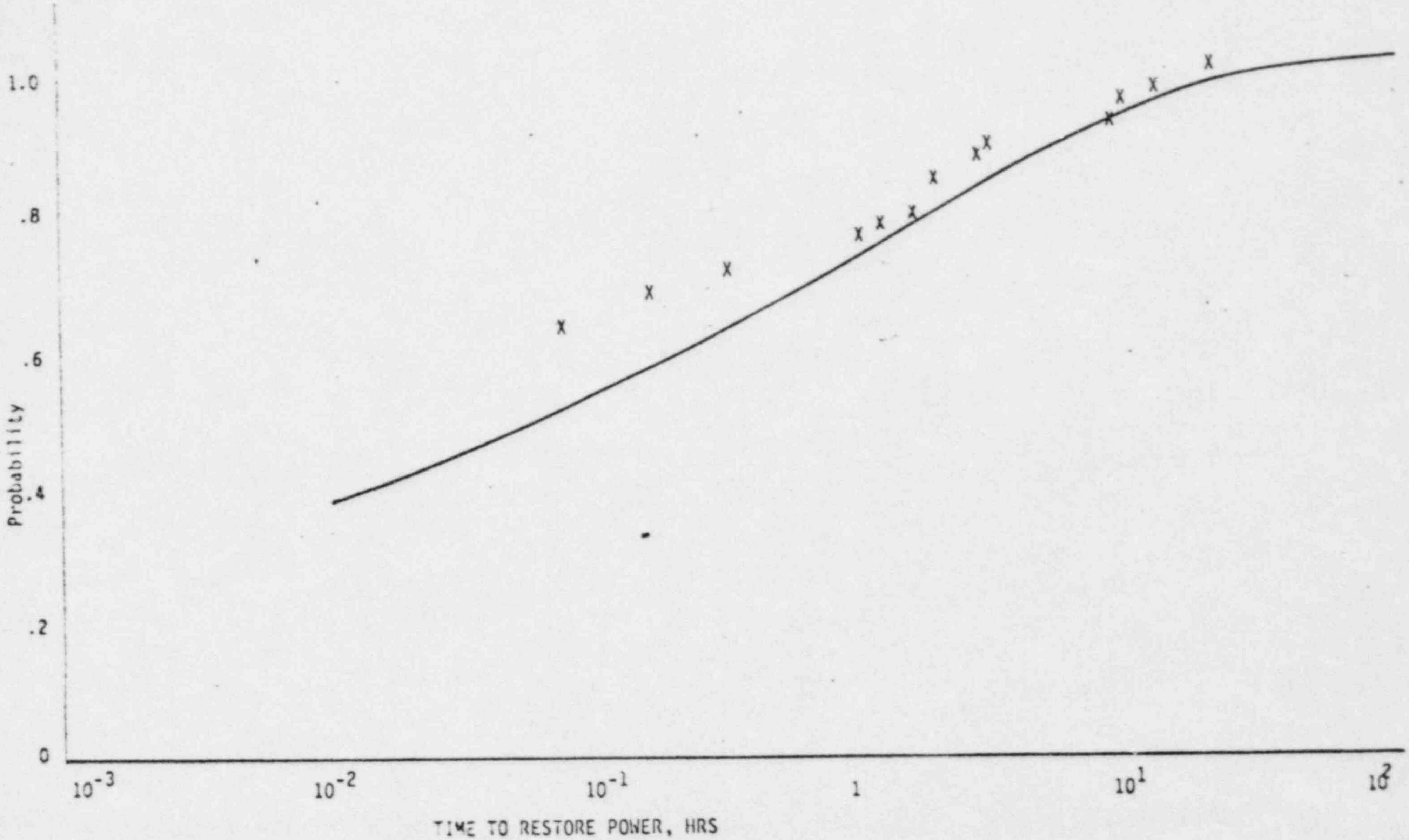


Figure III-8. Gamma Distribution Fit to the Restoration of Offsite Power Data.

COMMON CAUSE FAILURE ANALYSIS: METHODOLOGY

I. IDENTIFICATION OF COMMON MODE MECHANISM

A. "LIKE" VS. "UNLIKE" COMPONENTS

B. SUSCEPTIBILITY TO COMMON CAUSE

C. OPPORTUNITY FOR COMMON CAUSE FAILURE

II. QUANTIFICATION

A. "LIKE" COMPONENTS

1. CONSERVATIVE FILTER - 10% COUPLING FOR ALL REDUNDANT COMPONENTS, SUBSYSTEMS, OR SYSTEMS
[NUCLEAR TECHNOLOGY, VOL. 46, DECEMBER 1979]

2. EXTERNAL EVENTS

B. "UNLIKE" COMPONENTS

1. EXTERNAL EVENTS

III. INTEGRATION INTO FAULT TREE AND EVENT TREE MODELS

EXAMPLES OF

COMMON CAUSE FAILURE MECHANISMS

- INTERNAL - MANUFACTURER

  LOCATION

  TEST

  MAINTENANCE

- EXTERNAL - FIRE

  EARTHQUAKE

  TORNADO

  FLOOD

  HIGH ENERGY LINE BREAK

  CONTROL ROOM HABITABILITY (SMOKE, RADIATION)

  AIRPLANE CRASH

  HUMAN ERROR

DOMINANT COMMON MODE FAILURE MECHANISMS

- INTERNAL - MANUFACTURER

  LOCATION

- EXTERNAL - FIRE

  HIGH ENERGY LINE BREAK

UNCERTAINTY ANALYSIS

I. METHODOLOGY

A. Dominant Sequences Analyzed

1. Determine all components or events included in Dominant Accident Sequences
2. Develop a dominant accident sequence "fault tree"

B. Assume Probability Density Functions (PDFs) for All Events

C. Propagate Event Uncertainties to Arrive at Sequence (and CDF) Uncertainty

D. Perform PDF Sensitivity Study

II. PDFs

A. Engineering Judgement, based upon Data Source (i.e. WASH-1400, EG&G, or Plant), and/or Data Population Size (several trials vs. few trials)

B. PDFs Chosen For:

1. Components - Generic (log-normal, poisson)
   - Plant Specific (binomial, poisson, gamma, F)
2. Human Errors or Actions - (normal, log-normal)
3. Initiators (log-normal, poisson, gamma)

III. PROPAGATION

A. Use Method of Moment Propagation

1. First and Second Moments about the Origin
2. Faster Computer Running Time Than Monte Carlo Method, with small accuracy loss
3. 95% Chebychev Confidence Limit is Conservative

B. Results are: Mean, Standard Deviation, 95% Chebychev Confidence Limit (For Normal Distribution)

IV. SENSITIVITY STUDY

A. Choose Different PDFs For

1. Components or Events
2. Initiators

B. Results Indicate Sensitivity to Initiator PDFs

PDF EXAMPLES

INITIATORS:

LOCA	LOG-NORMAL, EF = 20
LOSS OF FEEDWATER	LOG-NORMAL, EF = 10
FIRES	GAMMA

HUMAN ERRORS:

FAILURE TO TRANSFER DEMIN. PUMP TO 2B BUS	LOG-NORMAL, EF = 3
INADVERTANT OPENING/ CLOSING OF VALVE	LOG-NORMAL, EF = 10
FAILURE TO ENTER CONTAINMENT FOLLOWING DEMIN. PUMP FAILURE	NORMAL

COMPONENTS:

GENERIC - MOVs	LOG-NORMAL, EF = 3
Solenoids	LOG-NORMAL, EF = 3
Check Valves	Poisson

PDF EXAMPLES (CONTINUED)

COMPONENTS: (CONTINUED)

PLANT SPECIFIC -	PUMP FAILS TO START	BINOMIAL
	PUMP FAILS TO RUN	POISSON
	VALVE FAILS TO REMAIN OPEN	GAMMA
	MSIV FAILS TO CLOSE	F

RESULTS OF BRP PRA UNCERTAINTY ANALYSIS

CASE*	MEAN	STANDARD DEVIATION	95% (CHEBYCHEV) CONFIDENCE LIMIT	EF($\frac{95\%}{MEAN}$)	
A	9.85×10^{-4}/YR	4.4×10^{-3}	1.5×10^{-2}	15	BASE CASE
B	"	3.1×10^{-3}	1.1×10^{-2}	11	
C	"	1.8×10^{-3}	6.6×10^{-3}	6	
D	"	2.2×10^{-3}	7.9×10^{-3}	8	
E	"	1.5×10^{-2}	4.9×10^{-2}	49	
F	"	1.6×10^{-2}	5.3×10^{-2}	53	

*CASE A = LOG-NORMAL INITIATING FREQUENCIES (IFs) EXCEPT FIRE (GAMMA), EFs = 10, 20.
MIXED SPECTRUM FOR COMPONENT DISTRIBUTIONS

B = CASE A IFs, LOG-NORMAL COMPONENTS

C = LOG-NORMAL IFs (EXCEPT FIRE), EFs = 3, 10. CASE B COMPONENTS

D = LOG-NORMAL IFs (EXCEPT FIRE), LOCA EFs = 20, OTHERS ESTIMATED WITH $\chi^2_{0.95}$
CASE B COMPONENTS

E = χ^2 IFs (EXCEPT FIRE), CASE B COMPONENTS

F = CASE E IFs, CASE A COMPONENTS