GENERAL ELECTRIC PRESENTATION TO

NRC OPERATING BRANCH ON

ATWS

SEPTEMBER 13, 1978

MH/1653

JMW
9/13/78

GE PRESENTATION TO THE NRC OPERATING BRANCH

I. INTRODUCTION

II. RELIABILITY OF SCRAM SYSTEM

III. RECIRCULATION PUMP TRIP

IV. ASSESSMENT OF SYSTEM CAPABILITY

V. REALISTIC VALUE/IMPACT ASSESSMENT

VI. RISK TO PUBLIC

VII. CONCLUSIONS

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3/13/78

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I. INTRODUCTION

NRC/AEC	GE RESPONSE
1969-STUDY ⟶	MARCH 1971 - NEDO 10349
STUDY OF COMMON MODE FAILURE IN ELECTRICAL SYSTEMS	
1973-WASH 1270 ⟶	JUNE 1975 - NEDO 20626
MITIGATION REQUIRED	
1975 STATUS REPORT ⟶	JUNE 30, 1976 - CONSEQUENCE DESCRIPTION
10^{-7} SAFETY GOAL	
DETERMINISTIC EVENTS	SEPT. 30, 1976 - RELIABILITY ASSESSMENT
1978 - NUREG 0460 ⟶	AUGUST 1, 1978-ACRS ATWS SUBCOMMITTEE PRESENTATION
ATWS IS DESIGN BASIS ACCIDENT	
SAFETY GRADE OR 10^{-3} MITIGATION REQUIRED	
NO CREDIT FOR SCRAM SYSTEM IMPROVEMENTS	
DETERMINISTIC METHODS TO MEET PROBABILISTIC GOALS	

EDF:BP/1234
9/13/78

ATWS
OVERVIEW

1969	ACRS REQUESTS STUDY	
	- STUDY CMF -	
1971	NEDO 10349 - RPT	
	- RESOLUTION SOUGHT -	
1973	WASH 1270	
	- MITIGATION REQUIRED -	
1974	NEDO-20626 - AUTO-BORON	
	- RESOLUTION SOUGHT -	
1975	STATUS REPORT	
	- 10^{-7} MITIGATION -	
1976	MITIGATION	RELIABILITY
	- UNWARRANTED -	- ARSS -
1977	HANAUER TASK FORCE	
	- RECONSIDERATION -	
1978	NRC STAFF POSITION - NUREG-0460	
	- 10^{-6} MITIGATION WITH RELIABILITY	

JMW:PAT/1057
9/13/78

II. RELIABILITY OF CURRENT SCRAM SYSTEM

EDF:cc/72
9/13/78

GE RELIABILITY STUDY

COMPREHENSIVE BWR SCRAM SYSTEM ANALYSIS PERFORMED

- SUBMITTED BWR SCRAM SYSTEM RELIABILITY ANALYSIS REPORT TO NRC - SEPT. 1976

- ANALYZED CURRENT GENERIC SCRAM SYSTEM DESIGN

  REACTOR PROTECTION SYSTEM LOGIC-RELAY AND SOLID STATE ELECTRICAL SYSTEMS

  CONTROL ROD DRIVE MECHANICAL SYSTEMS

- INCLUDED COMPLETE STUDY OF BWR SCRAM SYSTEM WITH PARTICULAR ATTENTION TO COMMON MODE FAILURES

EDF:BP/1312
9/13/78

COMPREHENSIVE RELIABILITY ASSESSMENT

- o CONSUMED 8 MAN-YEARS
- o 440 PAGES
- o SYSTEMS ANALYZED:
  - REACTOR PROTECTION SYSTEM RELAY LOGIC
  - REACTOR PROTECTION SYSTEM SOLID STATE LOGIC
  - CONTROL ROD DRIVE MECHANICAL COMPONENTS
  - HYDRAULIC CONTROL UNITS
  - SCRAM AIR HEADER
  - SCRAM DISCHARGE VOLUME
- o ANALYSIS CONSISTED OF HYPOTHESIZING:
  - 659 FAILURE MODES
  - 72 COMMON CAUSE FAILURES
  - 484 FAILURE MODES IN FAULT TREES
- o ANALYSIS INCLUDED INVESTIGATION OF 455 REPORTED INDIVIDUAL COMPONENT ABNORMALITIES
- o RESULTS CONFIRMED BY OTHER BWR STUDIES

EDF:cc/146
9/13/78

GE RELIABILITY STUDY SUMMARY

- EXISTING SCRAM SYSTEM UNRELIABILITY IS 0.8×10^{-6}/DEMAND
- MAJOR CONTRIBUTOR TO SCRAM UNRELIABILITY IS SCRAM LOGIC/SENSORS
- COMMON CAUSE FAILURE POTENTIAL IN MECHANICAL CONTROL ROD DRIVE IS LOWER THAN 10^{-7}/YEAR

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III. RECIRCULATION PUMP TRIP

JMW:mks/680
9/13/78

RPT HISTORY

1971	ALTERNATE PATH TO SAFE SHUTDOWN NO REDUNDANCY OR PEDIGREE
1973	STATUS LESS CLEAR STILL NOT SAFETY GRADE
1976	MONTICELLO RPT NOT SAFETY GRADE
1976	LETTER, RUSCHE TO WARD SAFETY GRADE NOT REQUIRED
1978	NUREG-0460

RPT CRITERIA
TOUGHENED

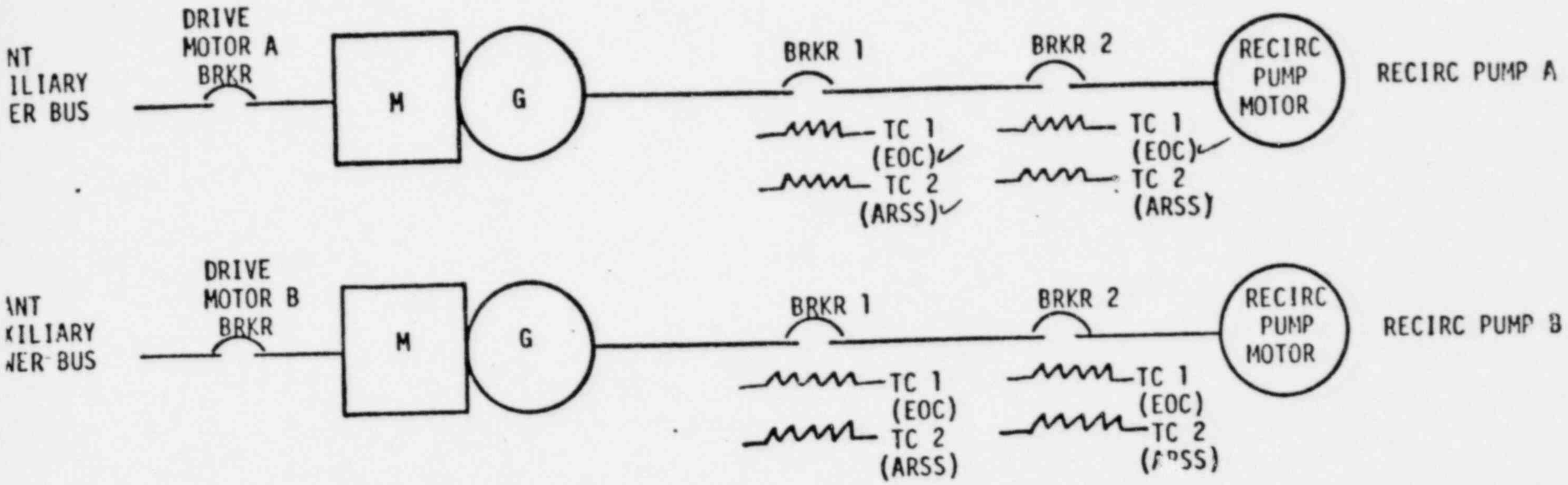
EFFECT ON OTHER SYSTEMS
CRITERIA EVEN WORSE

JMW:mks/681

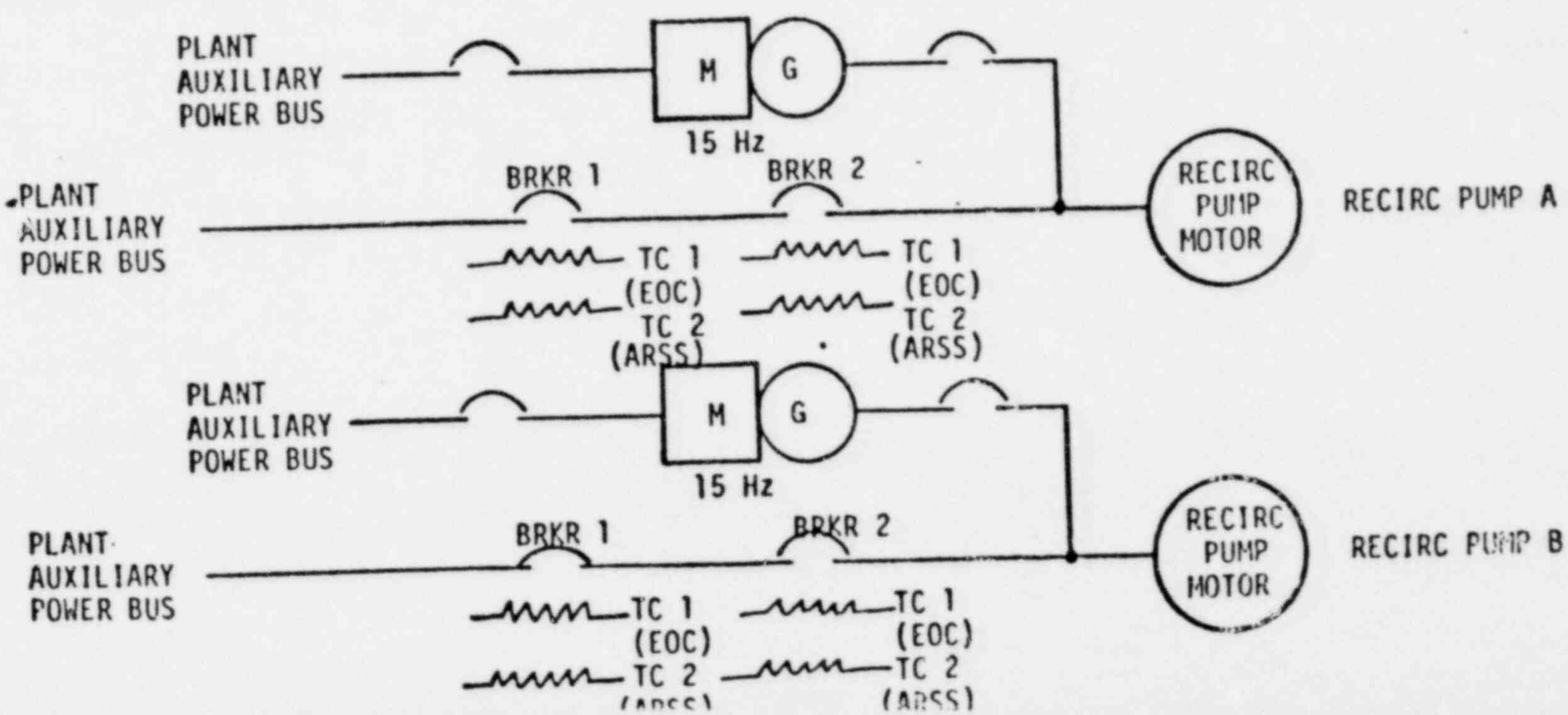
RPT

	ATWS PUMP TRIP	EOC RPT
INITIATING SIGNALS	HIGH REACTOR PRESSURE	TURBINE STOP VALVE INITIATION
	LOW WATER LEVEL	TURBINE CONTROL VALVE INITIATION
SAFETY GRADE	NO	YES
FAST ACTING	NO	YES
TRIP FUNCTION	DRIVE MOTOR OR PUMP BREAKER	PUMP MOTOR

JMW:mks/682
9/13/78



BWR /4 RECIRC PUMPS



IV. ASSESSMENT OF SYSTEM CAPABILITY

JMW:mks/685
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CURRENT BWR SYSTEM CAPABILITY

OVER PRESSURE PROTECTION PROVIDED BY:

- RECIRCULATION PUMP TRIP TO REDUCE POWER
- SAFETY/RELIEF VALVES TO RELIEVE PRESSURE

NUCLEAR SHUTDOWN PROVIDED BY:

- RECIRCULATION PUMP TRIP TO REDUCE POWER
- MANUAL INITIATION OF STANDBY LIQUID CONTROL SYSTEM

CORE COOLING PROVIDED BY:

- RECIRCULATION PUMP TRIP TO REDUCE POWER
- NORMAL FEEDWATER AS MAKE-UP INVENTORY
- ECCS INVENTORY SUPPLY
- REPLENISHMENT OF CONDENSATE STORAGE TANK BY ALTERNATE WATER SOURCES

RECIRCULATION PUMP TRIP SIGNIFICANTLY REDUCES ATWS RISK

EDF:cc/151
9/13/78

ATWS EVENT SEQUENCE

NON-ISOLATION EVENTS - TURBINE TRIP WITH BYPASS
LOAD REJECTION WITH BYPASS

- TURBINE/GENERATOR TRIP INITIATED

- TURBINE STOP VALVES CLOSED/BYPASS VALVES OPENED ALLOWING CLOSED CYCLE HEAT REMOVAL - FEEDWATER AVAILABLE FOR CORE COVERAGE

- RPT OCCURS TO REDUCE POWER

- SLC INJECTS BORON TO ACHIEVE NUCLEAR SHUTDOWN

- SUCCESSFUL SHUTDOWN

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ATWS EVENT SEQUENCE

ISOLATION EVENTS - MSIV CLOSURE
FW CONTROLLER FAILURE
RECIRC CONTROLLER FAILURE
LOSS OF FW FLOW
PRESSURE REGULATORY FAILURE
LOSS OF AUX POWER
LOSS OF CONDENSOR VACUUM

- MSIV CLOSURE INITIATED
- S/RV'S OPEN TO RELIEVE PRESSURE
- RPT OCCURS TO REDUCE POWER
- HPCI/RCIC ON AT LEVEL 2 (DRAWING FROM CONDENSATE STORAGE) TO MAINTAIN WATER LEVEL
- SLC INJECTS BORON TO ACHIEVE NUCLEAR SHUTDOWN
- MSIV'S REOPENED AFTER PRESSURE EQUILIBRATES
- CST INVENTORY MAINTAINED VIA CST TRANSFER PUMP
- SHUTDOWN COOLING WITH RHR
- SUCCESSFUL SHUTDOWN

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ATWS EVENT SEQUENCE

INADVERTENT SORV

- o SORV OCCURS
- o RPT MANUALLY INITIATED TO REDUCE POWER
- o RHR PLACED IN POOL COOLING MODE
- o SLC INJECTS BORON TO ACHIEVE NUCLEAR SHUTDOWN
- o DEPRESSURIZE USING ADDITIONAL S/RV'S
- o HPCI/RCIC ON (DRAWING FROM CONDENSATE STORAGE) TO MAINTAIN WATER LEVEL
- o SUCCESSFUL SHUTDOWN

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POSSIBLE CHANGES TO SCRAM SYSTEM

- o RELIABILITY STUDY INDICATES THAT RELIABILITY OF SCRAM SYSTEM IS LIMITED BY ELECTRICAL PORTION OF THE SCRAM SYSTEM NOT THE MECHANICAL PORTION

- o INCREASE SCRAM SYSTEM RELIABILITY BY THE FOLLOWING POSSIBLE CHANGES:

  - DIVERSE SET OF SENSORS AND LOGIC FOR INITIATION OF BACK-UP SCRAM SIGNAL

  - TRIP RECIRCULATION PUMPS TO REDUCE POWER USING DIVERSE SENSORS AND LOGIC

  - ADD TWO SCRAM AIR HEADER EXHAUST VALVES THAT WILL BE REDUNDANT WITH THE EXISTING BACK-UP SCRAM EXHAUST VALVES

CONCLUSION:

MODIFICATIONS WOULD REDUCE UNRELIABILITY OF THE SCRAM SYSTEM BY GREATER THAN FACTOR OF 100.

EDF:cc/155
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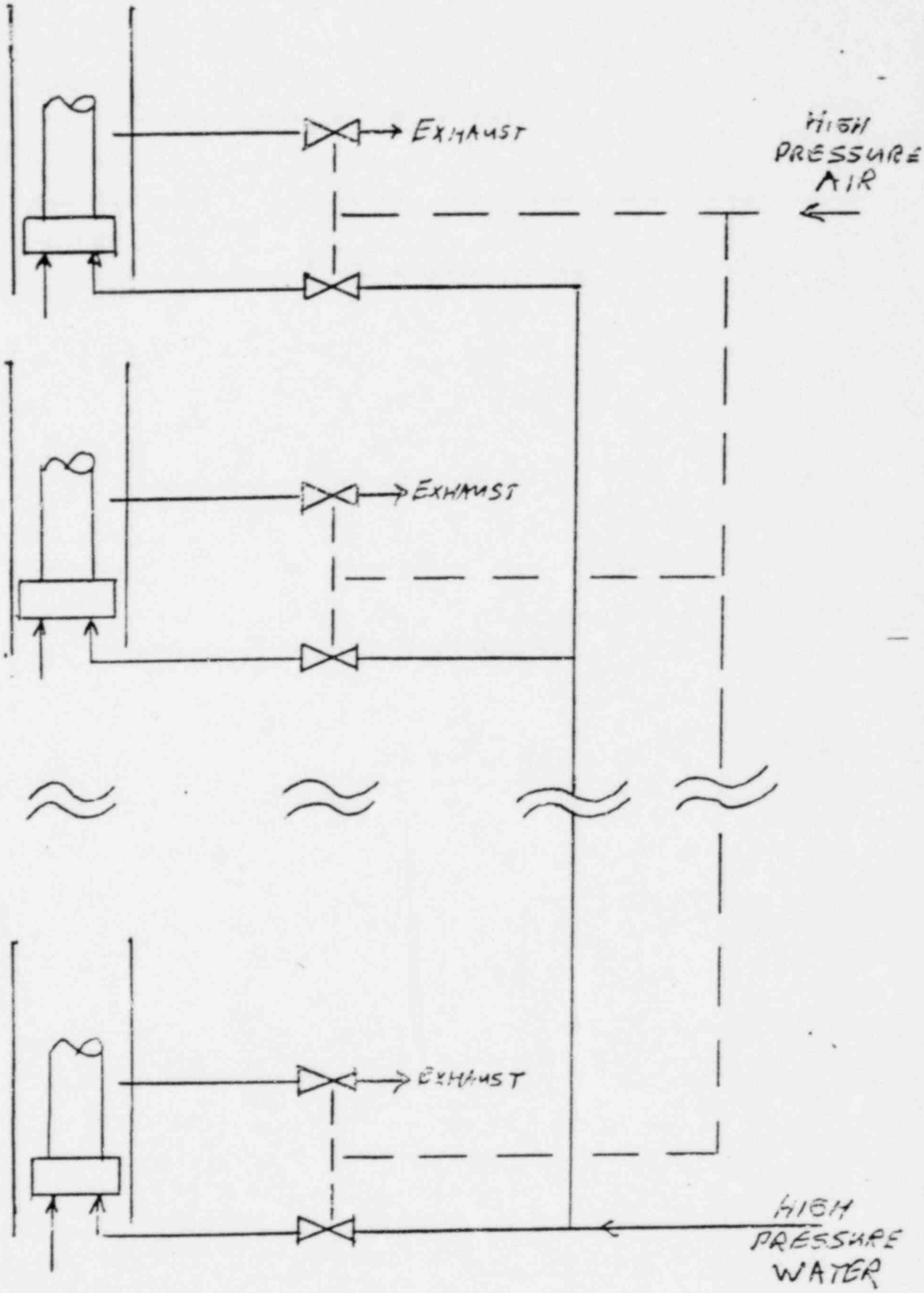
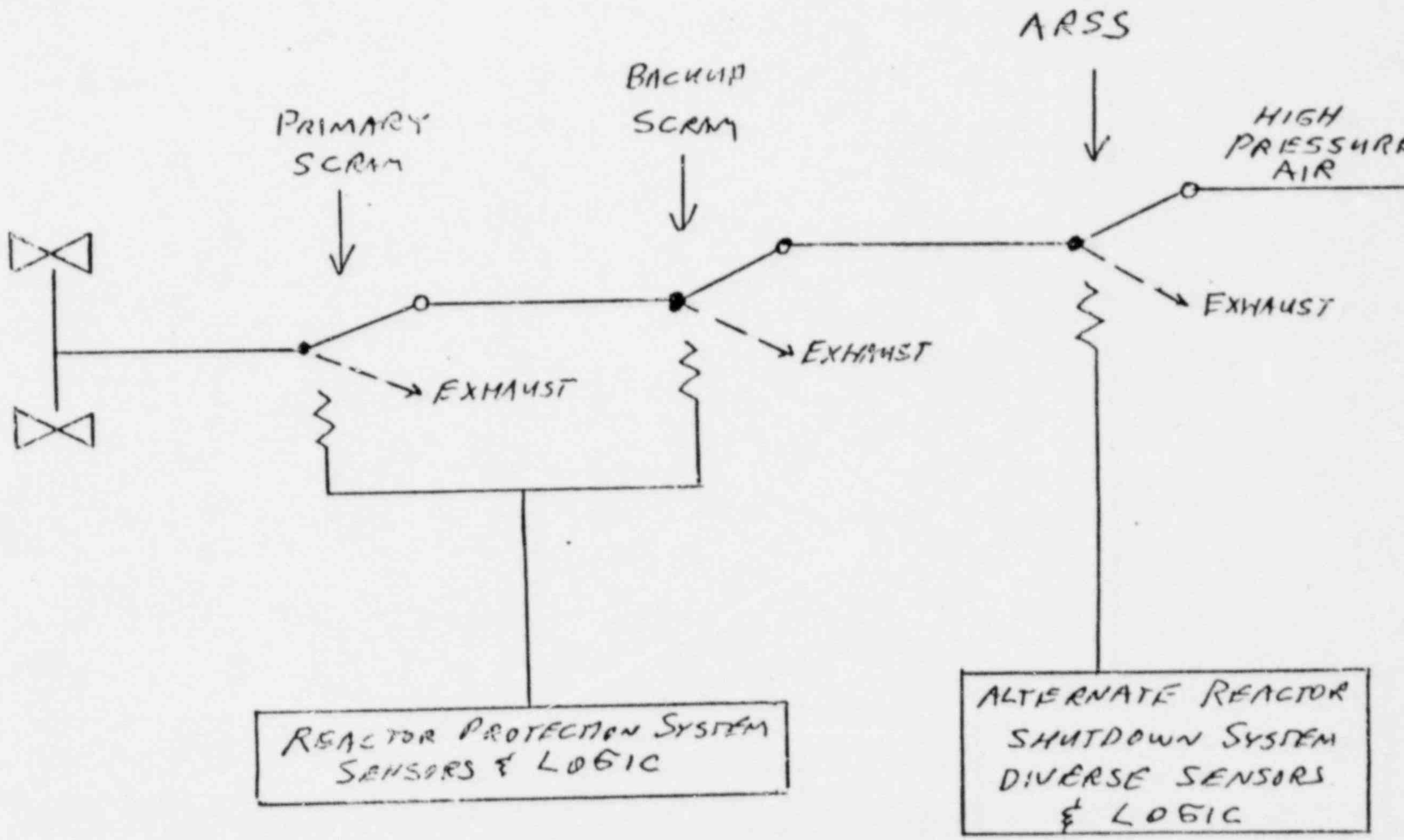
REACTOR SCRAM SYSTEM



SCRAM SYSTEM AIR SUPPLY



CHANGES TO SCRAM SYSTEM
(FREQUENCY COMPARISON)

	NUREG-0460	GENERAL ELECTRIC
o TRANSIENT FREQUENCY	6/YEAR	3.5/YEAR
o SCRAM SYSTEM UNRELIABILITY	$3x10^{-5}$/DEMAND	$0.8x10^{-6}$/DEMAND
o SCRAM SYSTEM WITH MODIFICATIONS (GE ASSESSMENT)	($3x10^{-7}$/DEMAND)	$< 10^{-7}$/DEMAND
o MITIGATED ATWS PROBABILITY	($2x10^{-6}$/YEAR)	$< 10^{-7}$/YEAR

RECALL:

- o NUREG 0460 GOAL 10^{-6}/YEAR
- o REASONABLE GOAL 10^{-5}/YEAR

NO CHANGE WARRANTED

EDF:PAT/1130
9/13/78

CONCLUSIONS

		ATWS FREQUENCY
o	NUREG 0460 SAFETY GOAL	10^{-6}/YEAR
o	EXISTING SCRAM SYSTEM	3×10^{-6}/YEAR

- EXISTING SCRAM SYSTEM MEETS REASONABLE SAFETY GOAL

o RECIRCULATION PUMP TRIP SIGNIFICANTLY REDUCES ATWS CONSEQUENCES

o INCLUSION OF ARSS WITH EXISTING SYSTEM SUCCESSFULLY MITIGATES ALL EVENTS

JMW:MKS/683
9/13/78

V. REASONABLE VALUE IMPACT ASSESSMENT

EDF:PAT/1114
9/13/78

VALUE ASSESSMENT BASED ON KEY ASSUMPTIONS

	NUREG-0460	REALISTIC ASSESSMENT
P(ATWS)	2×10^{-4}/REACTOR-YEAR	3×10^{-6}/REACTOR-YEAR
OUTAGE TIME	12 MONTHS	0
BWR RADIOLOGICAL RISK OF ATWS CORE MELT	1000-3000 MAN-REM/ REACTOR-YEAR	ASSUME UNCHANGED FOR THIS ASSESSMENT
DOLLAR VALUE OF RADIOLOGICAL EXPOSURE	$1000/ MAN-REM	
BWR OFFSITE PROPERTY DAMAGE RISK OF ATWS CORE MELT	$50,000-$200,000/REACTOR-YEAR	
DIRECT VALUE	$19 - 47 MILLION	$0.3 - 0.7 MILLION
INDIRECT VALUE	$23 MILLION	0

TOTAL VALUE	$42 - 70 MILLION	$0.3 - 0.7 MILLION

EDF:sj/355
9/13/78

IMPACT OF
SPURIOUS BORON INITIATION
AUTOMATIC STANDBY LIQUID CONTROL SYSTEM

- o 1 SPURIOUS EVENT/10 YEARS/PLANT CAUSED BY:
  - SURVEILLANCE TESTING REQUIREMENTS
  - EQUIPMENT FAILURES
  - OPERATOR ERROR
- o CLEAN-UP TIME

  1 MONTH ASSUMING 86 GPM SYSTEM WITH 10-MINUTE OPERATION
- o IMPACT/PLANT

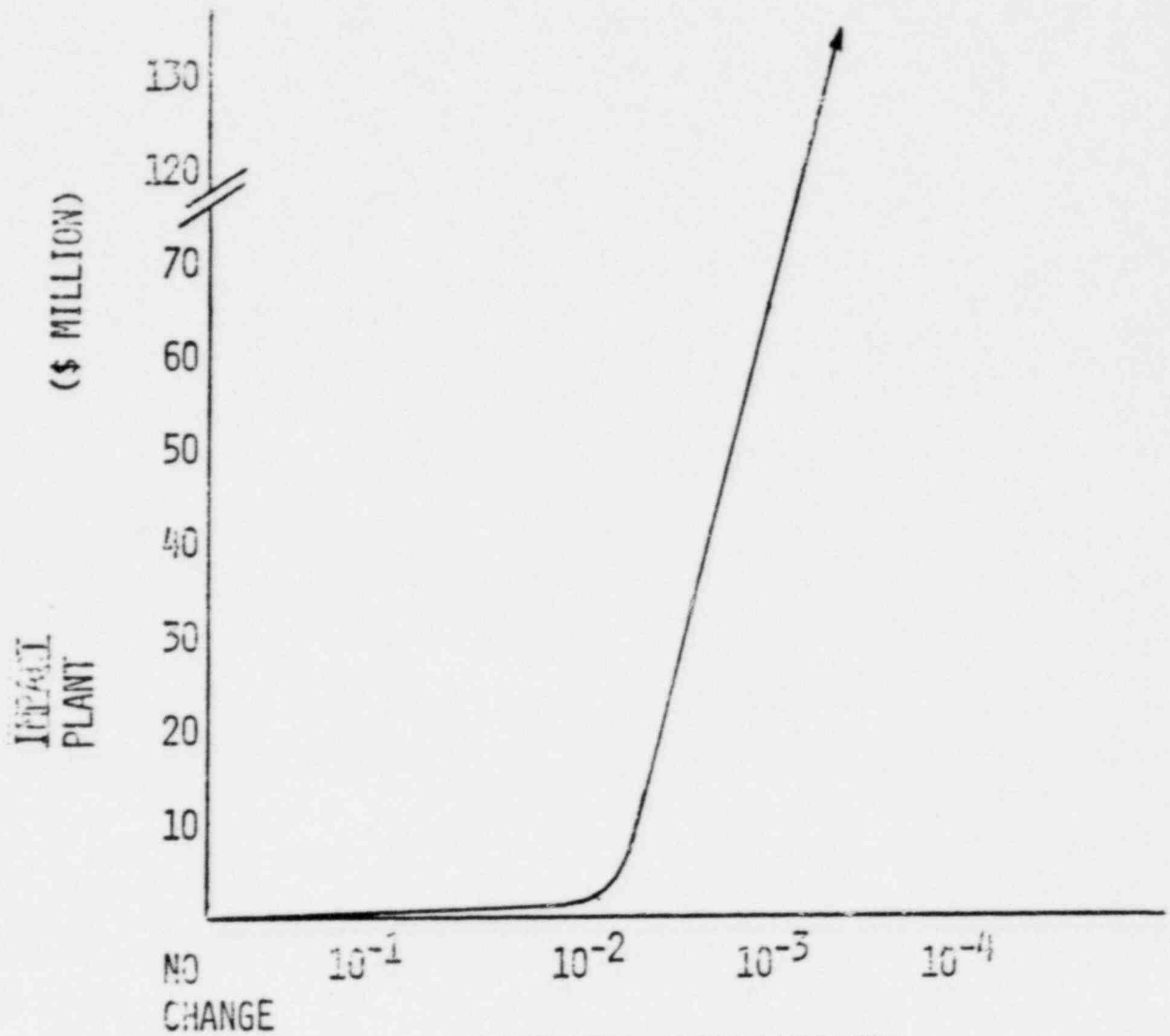
  (4 EVENTS / PLANT LIFETIME) X (1 MONTH/EVENT) X ($15 MILLION/MONTH) =

  $60 MILLION FOR 86 GPM SYSTEM/PLANT

EDF:PAT/1128
9/13/78

IMPACT SENSITIVITY TO

INCREMENTAL UNRELIABILITY



INCREMENTAL UNRELIABILITY

NOTES:

10^{-2} - SCRAM SYSTEM CHANGES

10^{-3} - SCRAM SYSTEM CHANGES, AUTOMATED STANDBY LIQUID CONTROL SYSTEM

EDF:sj/354
9/13/78

VI. ATWS RISK TO THE PUBLIC

(BASED ON WASH-1400)

TOTAL AVG LWR RISK	$5x10^{-5}$/REACTOR-YR
TOTAL BWR RISK	$3x10^{-5}$/REACTOR-YR
ATWS BWR RISK	$1x10^{-5}$/REACTOR-YR
NON-ATWS BWR RISK	$2x10^{-5}$/REACTOR-YR

24 DOMESTIC OPERATING BWRs PRESENTLY

CONSERVATIVELY ASSUME 500 REACTORS BY 2000

THEN ATWS RISK FROM OPERATING BWRs = $\frac{(24)\ (1x10^{-5})}{(500)\ (5x10^{-5})}$

$= 0.96\%$

SO ATWS REPRESENTS 1% OF RISK TO PUBLIC FROM OPERATING PLANTS

JMW:MKS/684
9/13/78

CONCLUSIONS

o	CURRENT BWR DESIGNS ARE HIGHLY RELIABLE AND REPRESENT SMALL RISK TO THE PUBLIC
o	NO CHANGES TO CURRENT DESIGNS ARE NECESSARY
o	IF CURRENT NUREG-0460 REQUIREMENTS WERE IMPLEMENTED IT WOULD RESULT IN AN UNJUSTIFIED FINANCIAL BURDEN TO THE PUBLIC
o	NUREG-0460 SHOULD BE REVISED

EDF:cc/157
9/13/78

Stu

MEETING SUMMARY DISTRIBUTION

*E. Case
*V. Stello
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*R. Fraley, ACRS (16)
T. B. Abernathy, DTIE
J. Miller
H. Thornburg, IE
K. Seyfrit, IE
*Docket Files/Central Files
*NRR Reading
*RSB Reading
Attendees
L. Gifford-GE Bethesda
R. Woods
*J. Norberg
*A. Thadani
*W. Minners
S. Weiss

*Denotes person to receive a copy of slides

Ashok

Here is my input, consisting of a revised list of initiating events, a revised list of initial conditions, and a paragraph concerning PWR initial MTC, and a revised list of systems and equipment. The most drastic cuts were in BWR initiating events, and I've attached a page explaining why I made some of these cuts.

As a general comment, I think we're all going to have to re-review this document several times, so consider the enclosed a "first cut," still open to discussion.

Finally, I would like to get a PWR systems expert to look this over. Most of my experience is with BWRs, and you'll notice that the BWR lists were revised the most.

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A. Initiating Events

The ATWS evaluation shall include the following transients, which are expected to occur one or more times during the life of the nuclear power unit, unless it is demonstrated that a particular transient is never limiting.

1. Pressurized water reactors

[use existing list]

2. Boiling Water Reactors

a. Limiting Pressure Increase. These transients include loss of load (load rejection without bypass, turbine trip without bypass, and loss of condenser vacuum), main steam line isolation valve closure, and pressure regulator failure.

b. Limiting Reactor Water Inventory Decrease. These transients include loss of feedwater, pressure regulator failure, and inadvertant opening of condenser bypass valves.

c. Limiting Reactor Coolant Flow Increase. These transients include failure of the recirculation flow controllers and the startup of an idle recirculation loop.

d. Limiting Reactor Water Temperature Decrease. These transients include ~~malfunction of the~~ feedwater controller failure (maximum flow), loss of the maximum credible ~~number of fee~~ amount of feedwater heating, and inadvertant initiation of ~~emergency core~~ high pressure cold water injection systems.

e. Loss of Normal Electrical Power. This event covers the simultaneous loss of power from the unit generator and from the offsite grid, leaving the reactor with the onsite emergency diesel generator sets functioning as the only source of a-c power.

e. Stuck open safety/relief valve. This analyses should be performed considering the effect of failing to reclose an amount of relief valves equivalent to 10% of rated steam flow.

comments on revised list of initiating events

~~The~~

The preceding list is based on two modifications. First, I have tried to allow some flexibility for eliminating analyses of non-limiting transients. Second, I have eliminated some transients from the BWR list. The reason is that recirc pump trips and inadvertant rod withdrawals do not cause a reactor scram, and the analyses presently on record do not take credit for a scram.

In addition, I should point out that one of the Dresden units managed to lose a whole string of feedwater heaters. DOR intends to require all BWR licensees to analyze loss of a string of heaters, unless the plant can demonstrate that the Dresden event is not credible for its particular installation.

Finally, although I left the BWR IORV event in the list for historical reasons, I think we should consider leaving it out. This event also does not lead to a scram (at least not for quite a long interval of time) and might better be analysed as part of the TMI lessons learned program. I say this in the context that we have 17 analyses already (in the GE submittal) and therefore have some basis for believing that IORV will never be limiting. It might be tactically advantageous for us to tell the industry, "GE gave us a report and it resulted in our scaling back our requirements a little."

d. Moderator Temperature Coefficient

The initial moderator temperature coefficient value used must be less negative than that experienced during 99% of the time the reactor is at relevant power levels (see Section I.C.a.1 above). Because future changes in fuel design, cycle length, and other changes can affect the moderator temperature coefficient, a statement of intent must be submitted, which commits the licensee to maintaining the moderator temperature coefficient more negative than the value used for ATWS analyses for 99% of the plant's operational life. This statement must include:

1) a determination of the design aim for the MTC at steady state full power BOC equilibrium xenon conditions with normal control configuration, including reloads, and the resulting probability function (at relevant levels) for the steady state condition.

2) Confirmation from startup experiments (from similar reactors for plants not yet in operation) that the MTC (at the required state condition) has been as expected. This should include analysis of extrapolations if measurements are done at zero power.

3) an augmentation factor which accounts for the effect of power change operational transients on the steady state MTC probability function. If this augmentation factor is less than 2×10^{-5} Δk/°F, detailed justification must be provided.

Initial Conditions (PWRs)

Core Power
Moderator Coefficient
Doppler Coefficient
Primary System Pressure (peak location)
Safety and Relief Valve Capacities and Setpoints
Core Inlet Temperature
Auxiliary Feedwater flow rate
High Pressure Injection System Flow Rate
Service Water Temperature & Flow
Containment Volume
Steam Generator Inventory
Core Flow
Boron Concentration
Fuel Element Gap Size
Heat Exchanger Capability

Initial Conditions(BWRs)

Core Power
Void Coefficient
Doppler Coefficient
Reactor Pressure (peak location)
Safety and Safety/Relief Valve Capacities and Setpoints
High Pressure Coolant Injection Flow Rate
Reactor Core Isolation Cooling System Flow Rate
Vessel collapsed Water Level
Suppression Pool Volume and Temperature
Core Flow
Core Active Void Fraction

Systems:

For systems relied upon, ~~to/brin~~ provide the signal used to actuate, the setpoint, and bases for assumed actuation time. This should include valve opening and closing times.

D. Systems and Equipment

The following systems are relied upon to mitigate the ATWS event, bring the reactor to a cold shutdown, and maintain it in that condition:

1. Pressurized Water Reactors

   Automatic Initiation of Turbine Trip
   Integrated Control System
   Pressurizer Safety and Relief Valves
   Auxiliary Feedwater
   Secondary Safety Valves
   Residual Heat Removal System
   Containment and Containment Pressure Suppression System
   High Pressure Safety Injection System
   Low Pressure Safety Injection System
   Steam/feedwater Isolation System
   Chemical and Volume Control System

2. Boiling Water Reactors

   Automatic Recirculation Pump Trip System
   High Pressure Coolant Injection or Core Spray
   Reactor Core Isolation Cooling System
   Residual Heat Removal System
   Safety/Relief Valves
   Standby Liquid Control System
   Feedwater Pump Trip

ATWS

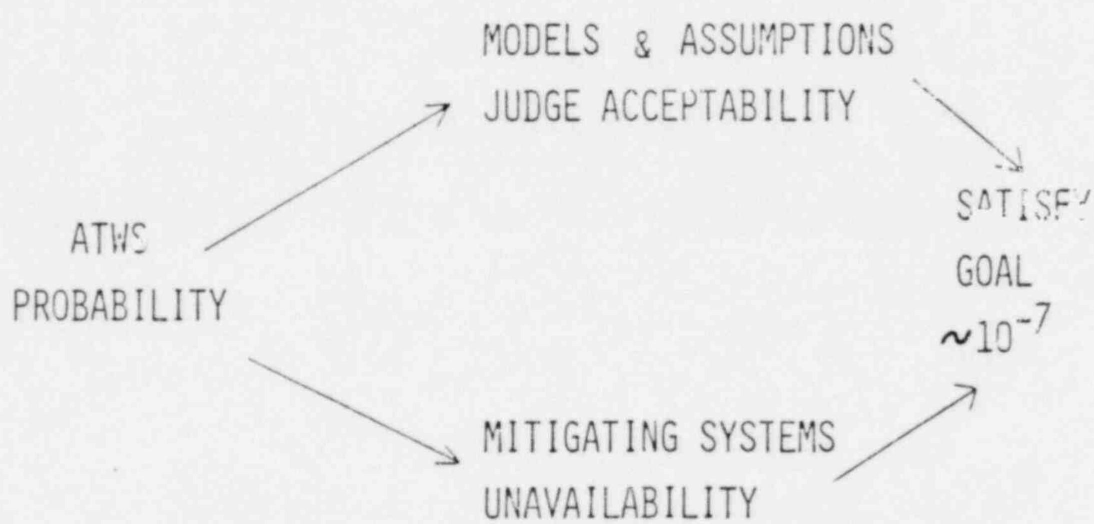
ANTICIPATED TRANSIENT FOLLOWED BY FAILURE TO SCRAM

WASH-1270

GOAL $\sim 10^{-7}$

SPECIFIED ACCEPTANCE LIMITS

STAFF APPROACH



INDUSTRY RESPONSE

$P(ATWS) \ll$ STAFF ESTIMATE

STAFF REQUIREMENTS EXCESSIVE

TABLE I

TYPICAL MATRIX FOR PRIMARY SYSTEM PEAK PRESSURE CALCULATIONS

		Power* (%)	Δ T Avg* (°F)	MTC* (PCM/°F)	ΔDoppler* (%)	GAP* (mils)	ΔS/G MASS* (%)	AUX Feed* Time (Seconds)	ΔRCS Volume (%)	ΔPressure Level (%)
	0	100	0	-11.1	0	3.0	0	20	0	0
FACTORIAL MATRIX	1	104	4	-7	+25	4.4	-5	40	0	0
FACTORIAL MATRIX	2	104	4	-7	-25	1.[illegible]	5	15	0	0
FACTORIAL MATRIX	3	104	-4	-15	+25	4.4	5	15	0	0
FACTORIAL MATRIX	4	104	-4	-15	-25	1.6	[illegible]	40	0	0
FACTORIAL MATRIX	5	96	4	-15	+25	1.6	-5	15	0	0
FACTORIAL MATRIX	6	96	4	-15	-25	4.4	5	40	0	0
FACTORIAL MATRIX	7	96	-4	-7	+25	1.6	5	40	0	0
FACTORIAL MATRIX	8	96	-4	-7	-25	4.4	-5	15	0	0
	9	100	0	-11.1	0	3.0	0	25	5	0
	10	100	0	-11.1	0	3.0	0	25	0	4
Base Case	11	100	0	-11.1	0	3.0	0	25	0	0

*Indicates reactor operating parameters which were in each vendor's matrix

TABLE II

PROBABILITY DISTRIBUTION FOR REACTOR OPERATING PARAMETERS

Parameter	Distribution* B&W	CE	W
Power (%)	N(100,2)	N(100,2)	N(100,2)
T_{avg} (^oF)	U(-4,4)	U(-4,4)	U(-4,4)
Error in Doppler Estimate (%)	N(0,12.5)	N(0,12.5)	N(0,12.5)
Moderator Temp. Coefficient (PCM/^oF)	N(-19,5)	N(-13,2.9)	N(-19,5.2)
Reactor Cooling System Volume (difference from nominal (%))	N(0,2.5)	N(0,2.5)	N(0,2.5)
Steam Generator Level (difference from nominal (%))	N(0,4)		N(0,4)
GAP B&W Mils CE % difference from nominal W Units of UA BTU/hr ^oF	N(3,1)	N(0,25)	N(4000,1000)
Pressurizer Level (difference from nominal (%))	N(0,2.5)		

* N(μ,σ) ≡ Normal distribution; μ ≡ mean; σ ≡ standard deviation.

U(a,b) ≡ Uniform distribution in the interval a to b.

TABLE III

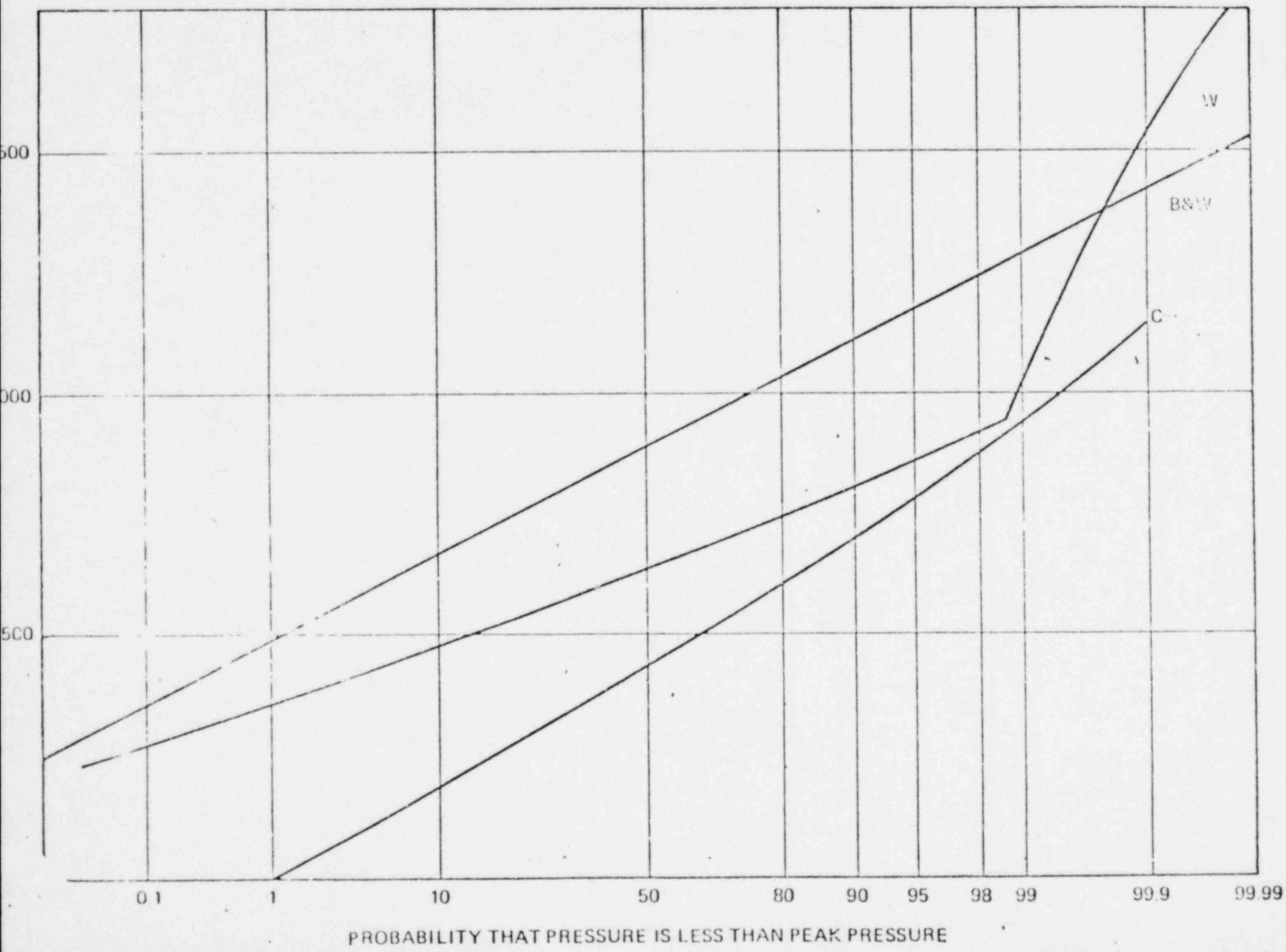
EQUIPMENT FAILURES+

Vendor	Event	Prob. of Occurrence of Event	Pressure Increase Due to Event
B&W	Relief valve fails to open	.02**	200 psi*
	1 train of Aux. Feedwater is lost	.04	150 psi*
CE	1 train of Aux. Feedwater is lost	.04	175 psi*
W	Relief valve fails to open	.04**	150 psi
	1 train of Aux. Feedwater is lost	.04	140 psi

+Where two failure modes are given, they were both included independently as discussed in Section 8 of the text.

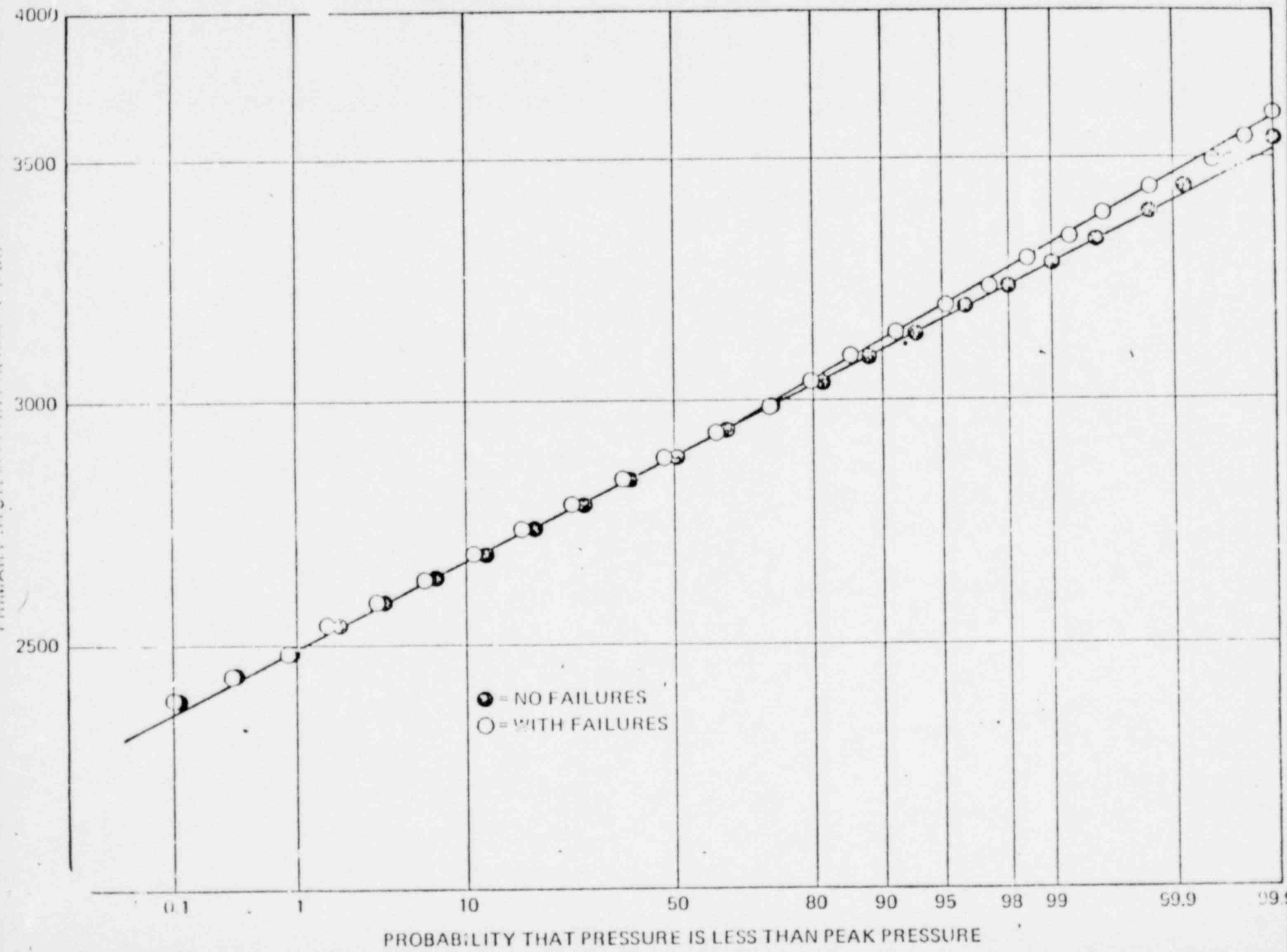
*Estimated sensitivity based upon an ATWS primary system peak pressure of about 3200 psi.

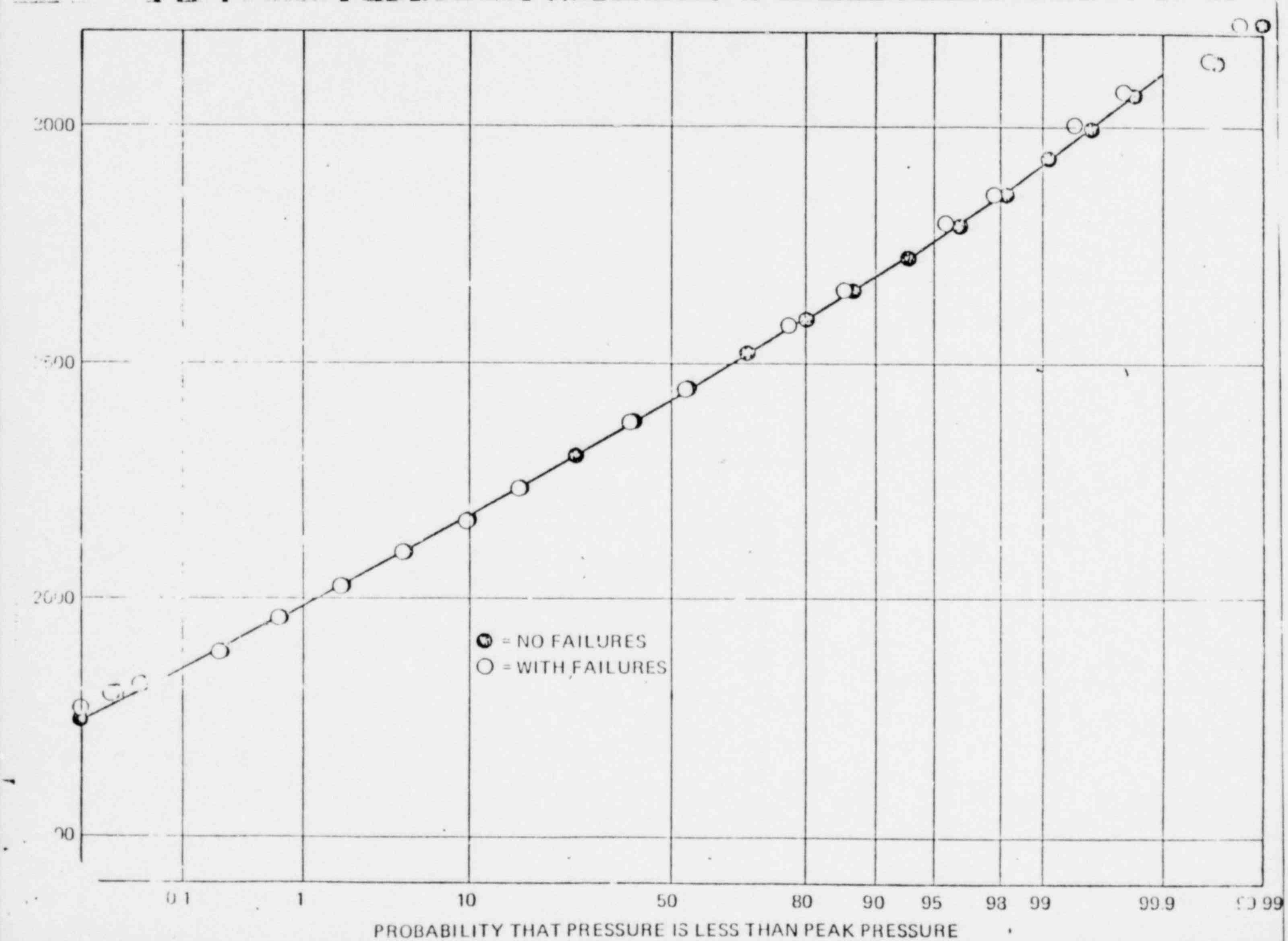
**These probability valves are different because of the different number of valves.



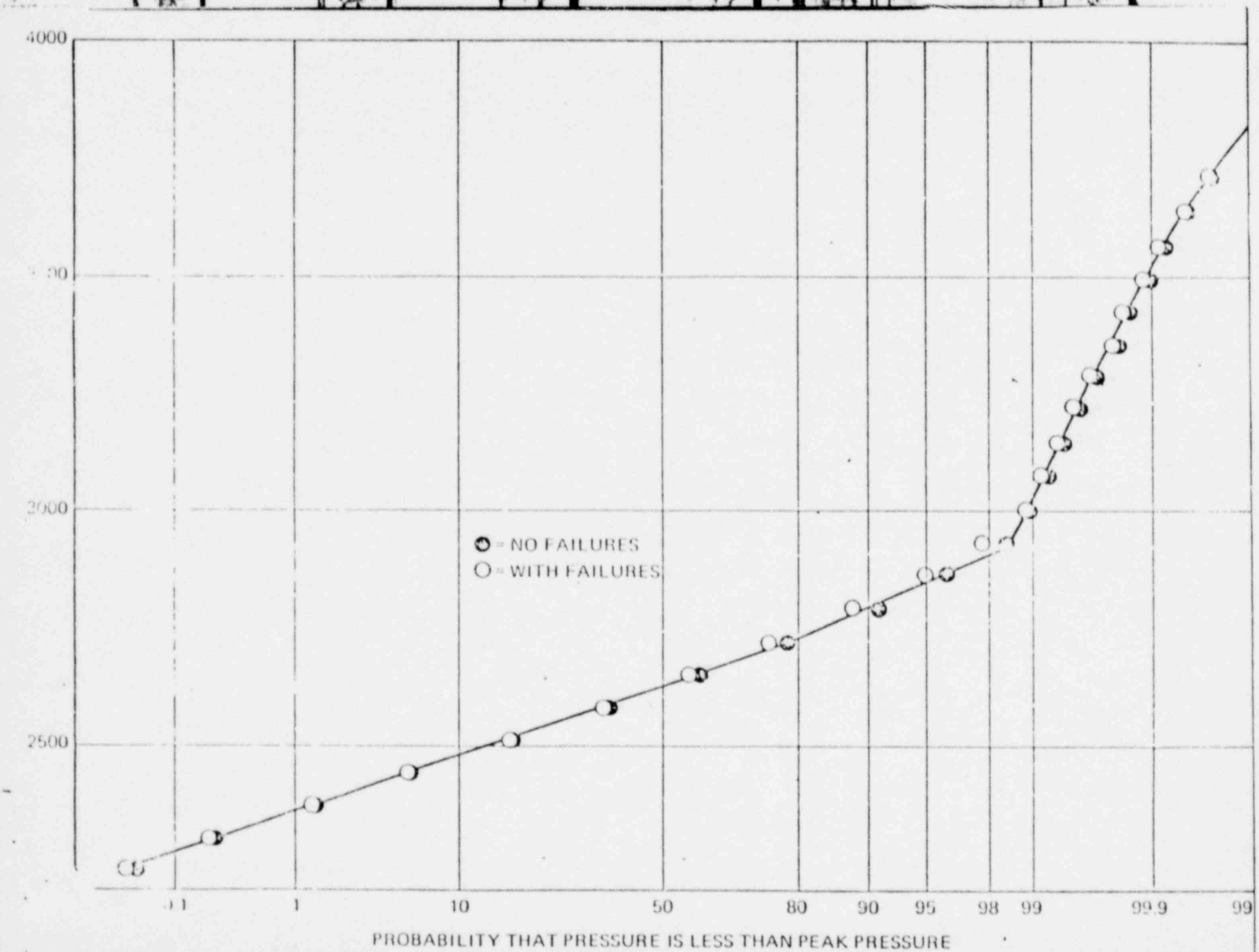
500
000
5C0
0.1
1
10
50
80
90
95
98
99
99.9
99.99
W
B&W
C
PROBABILITY THAT PRESSURE IS LESS THAN PEAK PRESSURE

Figure 2 B & W PEAK PRESSURE DISTRIBUTIONS





3000
2000
= NO FAILURES
= WITH FAILURES
0.1
1
10
50
80
90
95
98
99
99.9
PROBABILITY THAT PRESSURE IS LESS THAN PEAK PRESSURE



4000
3000
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● = NO FAILURES
○ = WITH FAILURES
.1
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99.9
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PROBABILITY THAT PRESSURE IS LESS THAN PEAK PRESSURE

CAUTIONS & CONCLUSIONS

1. CAREFUL SELECTION OF ALL IMPORTANT PARAMETERS NECESSARY.

   WE PICKED A FEW PARAMETERS.

2. LINEAR & INDEPENDENCE ASSUMPTION ON PARAMETERS NEED

   CAREFUL STUDY.

3. DATA BASE MAY BE INSUFFICIENT TO GET GOOD ESTIMATES

   ON PARAMETER DISTRIBUTIONS.

4. METHOD PROVIDES AN EASY ESTIMATION OF PARAMETER EFFECTS.

410. ROD POSITION INDICATION (10) (5 pages)

411. Handwritten Notes, Seismically Induced Turbine Trip with Bypass Failure and Failure to Scram, Abstract and Recommendation (26 pages)

412. Attachment III-3 (3 pages)

413. Question 222.22 (Attachment A) (13 pages)

414. July 27, 1979 Memorandum for S. H. Hanauer from A. Thadani Subject: NRC-INDUSTRY ATWS MEETING SUMMARY (8 pages)

415. September 8, 1976 Memo to V. Stello from Brian K. Grimes Subject: WESTINGHOUSE ATWS CATEGORY B - LOSS OF NORMAL AC POWER (10 pages)

416. Power Reactor Operating Experience (3 pages)

417. Westinghouse ATWS Loss of Normal AC Power (9 pages)

418. February 25, 1980 Memorandum for A. Thadani from Ralph O. Meyer Subject: CURRENT STATUS OF ATWS EARLY VERIFICATION FUELS ISSUES (9 pages)

419. February 25, 1980 Routing and Transmittal Slip to Thadani et al., from Steve Hanauer, attaching February 25, 1980 Note to S. Hanauer from D. Ross (3 pages

420. February 22, 1980 Memorandum for Ashok Thadani from R. Wayne Houston Subject: B&W ATWS REPORT (BAW-1610, JANUARY 1980) (5 pages)

421. February 21, 1980 letter to William Russell from Michael W. Golay, attaching Policy Recommendations for the Treatments of ATWS Events for Standardized Nuclear Power Plants, January 9, 1980 (41 pages)

422. December 5, 1979 Note to C. Stahle from Ashok Thadadi (5 pages)

423. November 25, 1977 Memorandum for Roger J. Mattson from Malcolm L. Ernst Subject: EXPANDED OUTLINE FOR ATWS I-V ANALYSIS (14 pages)

424. March 9, 1977 Letter to E. G. Case from Thomas G. McCreless Subject: COMMENTS ON ATWS, attaching November 25, 1976 letter to M. Plesset, October 28, 1974 letter to W. R. Stratton (13 pages)

425. August 16, 1976 Note to Benard Rusche from Ashok Thadani Subject: W - ATWS (2 pages)

426. March 15, 1977 Note to ATWS Distribution List from D. Ross (10 pages)

427. Draft notes on NUREG-0461 (2 pages)

428. April 20, 1978 Memorandum for H. Denton from W. Minners Subject: ATWS (3 pages)

429. June 2, 1975 to V. Stello R. R. Maccary Subject: B&W DRAFT REPORT - MECHANICAL COMMON MODE FAILURE ANALYSIS OF CONTROL ROD DRIVE MECHANISMS (4 pages)

430. February 26, 1980 Memorandum for Ashok Thadani from R. Wayne Houston Subject: EFFECT OF RAPID CONTAINMENT ISOLATION ON ATWS RADIOLOGICAL CONSEQUENCES (4 pages)

431. December 7, 1979 Memorandum for Stephen H. Hanauer from K. I. Parczewski Subject: ATWS MEETING WITH CE AND CE PLANT OWNERS GROUP (34 pages)

432. May 27, 1980 Memorandum for Karl Kniel from Ashok Thadani Subject: NRC- W OWNERS GROUP ATWS MEETING SUMMARY (12 pages)

433. September 15, 1978 Memorandum for Roger J. Mattson from Richard C. DeYoung Subject: REVIEW OF SUPPLEMENT 1 OF NUREG-0460, attaching November 23, 1977 Memorandum for Roger J. Mattson from Harold R. Denton(11 pages)

434. September 5, 1978 Draft Notes by Buhl (12 pages)

435. October 20, 1978 Routing and Transmittal Slip to T. Novak from Gordon Chipman (5 pages)

436. November 10, 1977 Memorandum for T. M. Novak from D. F. Bunch Subject: ATWS DRAFT PAPER (3 pages)

437. Accident Analysis, Probability and Risk Assessment: The Subjectivistic Viewpoint and Some Suggestions (Nuclear Safety, Vol. 19, No. 3, May-June, 1978) (11 pages)

438. November 2, 1977 Memorandum for Roger J. Mattson from Harold R. Denton Subject: COMMENTS ON ATWS REPORT (3 pages)

439. Routing and Transmittal Slip to A. Thadani from Ronald R. Bellamy, attaching Accident Analysis Branch Input to ATWS Letter, March 5, 1979 Note to F. Cherny et al., from Ashok C. Thadani (10 pages)

440. June 2, 1977 Memorandum for T. M. Novak from A. C. Thadani Subject: EPRI ATWS MEETINGS SUMMARY (36 pages)

441. November 10, 1977 Memorandum for T. M. Novak from D. F. Bunch Subject: ATWS DRAFT DRAFT PAPER (3 pages)

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442. April 20, 1978 Memorandum for H. Denton from W. Minners Subject: ATWS (3 pages)

443. March 14, 1977 Routing and Transmittal Slip to ATWS Distribution from Hal Ornstein (6 pages)

444. Undated Memorandum for Denton from Vollmer (7 pages)

445. Draft (2 pages)

446. July 24, 1978 Re-Evaluation of PWR Reactor Protection System (RPS) Fault Tree With Updated Data, Draft, SAI/SR-196-78-PA (6 pages)

447. ATWS: A Reappraisal, A Presentation to the Advisory Committee on Reactor Safeguards (27 pages)

448. May 9, 1977 To R. Blond et al., from Hal Ornstein (14 pages)

449. Undated Note to D. F. Bunch from H.E. P. Krug and P.S. Tam Subject: ESTIMATED THYROID DOSES FROM ATWS AS A FUNCTION OF STEAM GENERATOR TUBE LEAKAGE - PRELIMINARY STATUS REPORT (20 pages)

450. Anticipated Transients Without Scram for Westinghouse Plants December 1979 (227 pages)

451. January 1980 Analysis of B&W NSS Response to ATWS Events (81 pages)

DOCS 139 –
143

1
16 2
25
78
/265

Akstulewicz

Box 3

ROD POSITION INDICATION (10)

QUESTION:

WOULD LOSS OF ALL NORMAL AC POWER RESULT IN LOSS OF CONTROL ROD POSITION INDICATION ON ANY PLANTS? EXPLAIN HOW THIS WOULD IMPACT ACTUATION OF ANY ATWS MITIGATING EQUIPMENT.

ANSWER:

- ROD POSITION INDICATING SYSTEMS ARE ON DC POWER
- THEREFORE, NO IMPACT ON ATWS MITIGATING EQUIPMENT

JMW
8/8/79

MITIGATION EQUIPMENT (5)

QUESTION:

DESCRIBE HOW EACH ITEM OF ALTERNATE #3 IN NUREG 0460 VOLUME 3 IS ADDRESSED IN GE REPORT. CAN THESE CHANGES BE MADE ON ALL PLANTS?

ANSWER:

- RPT - LIMITS PRESSURE TO LESS THAN SERVICE LEVEL C

  ARI - INSERTS RODS TO PROVIDE MITIGATION

  SLCS - PROVIDES MITIGATION IF ARI FAILS

  LOGIC CHANGES - LOWERS ISOLATION LEVEL TO PREVENT TT FROM ISOLATING (Turb. Trip)

  FW RUNBACK - REDUCES POWER AND STEAM RELEASED TO POOL

  PROCEDURES & TRAINING - WILL BE DEVELOPED TO BE CONSISTENT WITH ANALYSIS

  SCRAM DISCHARGE VOLUME - NOT USED IN ANALYSIS

- ALL CHANGES SHOULD BE REFERRED TO THE UTILITIES FOR OPERATING PLANTS. FOR REQUISITION (Non Operating Plants), THOSE CHANGES APPEAR POSSIBLE.

JMW
8/8/79

OPERATOR ACTION (3)

QUESTION: IN LIGHT OF TMI WHY SHOULD CREDIT BE GIVEN FOR 10 MINUTE OPERATOR ACTION?

ANSWER:

- NOTHING AT TMI LEADS US TO CONCLUDE THAT WITH PROPER TRAINING AND INDICATIONS OPERATOR ACTION IN 10 MINUTES OR LESS IS PROPER.

- VERY FEW ACTIONS REQUIRED

POOL COOLING (25)

QUESTION: HOW WOULD OPERATOR DECIDE TO COOL POOL WHEN VESSEL LEVE'_ IS LOW?

ANSWER:

- THE OPERATOR WILL INITIATE POOL COOLING BY PROCEDURE. TWO LOW PRESSURE COOLING SYSTEMS ARE AVAILABLE FOR USE EVEN WITH RHR IN POOL COOLING.

BORON IMPACT (31)

QUESTION:

WHAT IS THE EFFECT OF BORON ON RCPB?

Materials Prob ?

ANSWER:

WE HAVE NOT IDENTIFIED ANY PROBLEMS

None

w/o fully examing details

JMW
8/8/79

<u>BORON IMPACT (31)</u>

<u>QUESTION:</u>

WHAT IS THE EFFECT OF BORON ON RCPB?

Materials Prob ?

<u>ANSWER:</u>

WE HAVE NOT IDENTIFIED ANY PROBLEMS

None

w/o fully examing details

JMW
8/8/79

NOTE TO: Turbine Trip File [illegible] I
FROM: H E P Krug, Nuclear Engineer, Section B, AAB, DSE
SUBJECT: Seismically Induced Turbine Trip with Bypass Failure and Failure to Scram.

ABSTRACT AND RECOMMENDATION

This note discusses an essentially incredible postulation. To wit: seismic input somehow generates a trip signal in the turbine trip system. The turbine trips but the turbine bypass valves do not open. In addition and regardless of turbine stop and throttle valve closure, the turbine does not generate a scram signal.

It appears that this specific postulation emerged for the first time as a result of ASLB questions raised during the Hartsville ASLB CP hearings. An adequate understanding of the evolution of this event requires assimilation of a considerable number of historical details. These have been relegated to Attachment I.

I, and a number of others, believe that the subject postulation should not be deemed credible. The primary basis is that properly implemented staff requirements on both turbine trip inputs to the RPS and the turbine buildings are sufficient to provide reasonable assurance that the postulated event is, essentially, incredible; considering the resistance to seismic motion inherent in turbine generator units. This perception is consistent with past staff practice.

III

It just so happens that staff requirements and philosophy associated with turbine buildings and the equipment they contain are difficult to firmly establish and justify in a few words. As a result, the reader is <u>urged</u> to read the written and verbal testimony of James P. Knight contained in Attachment <u>III</u>. <u>I know of no other single document where these bases are so clearly presented</u>.

The remainder of this note consists of a Summary section followed by seven attachments which are:

Attachment I – Historical Highlights

Attachment II – Scenario for BWR Turbine Trip Without Associated Scram and With Failure Of The Turbine Bypass System.

Attachment III – Testimony of James P. Knight and Harold Polk; Testimony of Harry E.P. Krug; and Testimony of James D. Thomas.

Attachment IV – Note to: Hartsville File by F. Kantor.

Attachment V – Turbine Generator Considerations Related to Seismic Input.

Attachment VI – Availability of Turbine Trip and Control Valve Fast Closure Trip Scrams.

Attachment VII – ASLB Findings.

HEPK 18 SEP 78 I

SUMMARY

As a result of ASLB requests during the Hartsville CP hearings an event, consisting of a siesmically induced turbine fast closure (trip) with failure to scram and with concurrent failure of turbine bypass values to open, was evaluated. Details of the postulated scenario are provided in Attachment II. As the Transcript shows, the genesis of this postulation derives from ASLB concerns about the capability of the turbine building to withstand the SSE without jeopardizing safety equipment inside the adjacent auxiliary building. Details of the ASLB actions are provided in Attachment I.

Primarily as a result of the testimony of James P. Knight, a copy of which is included in Attachment II the ASLB reversed its earlier position and concluded that the Hartsville turbine building could be built under an LWA-I authorization (10 CFR 50, Appendix B, "Quality Assurance" need not be applied). This finding is consistent with past practice. The ASLB explicitly indicated in its findings (see Attachment VII) in agreement with Attachment III, that a turbine building constructed in accordance with national building codes and designed to withstand the SSE and built by a competent constructor, provided adequate assurance against gross collapse.

It is clear from the ASLB transcript (Attachments III and VII) and ASLB findings that the ASLB accepted the staff and applicant position that the reactor protection system is adequately protected against malfunction of the turbine scram system. More importantly, the ASLB again endorsed the long standing staff policy that the level of protection required should be compatible with the magnitude of the associated risk (10CFR 50, Appendix A, Criterion 1). For example, ASLB considered it important that the scram sensors on the turbine were "fail safe" in that a broken wire would cause a scram. A scram occurs because the turbine scram circuits deenergize to scram. For additional details see Attachment 3.

The staff completed its Hartsville presentation by calculating the radiological consequences as it the remainder of the event was the same as the rod drop accident. This approach was intended to represent a reasonable bound upon consequences deemed credible for this seismically induced event. The ASLB accepted the staff's conclusions for purposes of permitting the construction of the turbine building, based upon the radiological consequences associated with the rod drop related scenario. Partly because of time constraints, the staff did not discuss in depth, and ASLB did not explicitly consider, the reliability of the turbine trip system when considering the scenario. However, that the ASLB considered the turbine trips to be of high quality and reliability appropriate for inclusion in the RPS circuitry is clear from the ASLB findings and transcript

Attachment IV is of interest in this context.

It is appropriate to emphasize certain considerations at this point.

(1) Appendix B, "Quality Assurance" has never been applied (See James P Knight in Attachment III) to a turbine building.

(2) The staff requires, that IEEE 279 - 1971 be applied to the turbine scram system (except for seismic requirements).

(3) One result of (2) is that the turbine scram system be "fail safe". In general, it must be considered a contradiction to postulate that a fail safe system does not fail safe (See, also, Attachment V).

(4) For all previously considered turbine trip events, i.e. those not "seismically

induced", the staff gives credit for turbine trip scram in accident scenarios.

Attachment 1 I

Historical Highlights

During the Hartsville Environmental Hearing, the ASLB made their Site Suitability and NEPA findings. As was typical, at least at that time, NRC ~~Projects~~ then routinely authorized, among other things, the construction of the turbine building under an LWA-I authorization (i.e., authorization of the construction of the turbine building without the imposition of the 18 criteria of 10 CFR Part 50, Appendix B, Quality Assurance).

The ASLB stopped ~~cancelled~~ the authorization for the construction of the turbine building (1) because it is adjacent to the auxiliary building, a safety structure, and (2) because the turbine building contains turbine trips which activate the reactor protection system causing a reactor scram.

At a later hearing, James P Knight ~~additional staff witnesses~~ explained the long standing staff position that turbine buildings need not be built in accordance with 10 CFR 50 Appendix B provided that the turbine building ~~would not endanger safety systems and structures if the turbine building suffered damage during a safe shutdown earthquake~~ was designed and constructed so that gross collapse would not occur. (The written and oral (Hartsville) testimony of James P. Knight is included in Attachment (3), along with a summary of the highlights of his testimony.)

The staff further explained that the Hartsville reactor protection trips were qualified to IEEE-279 (1971) except for seismic requirements. IEEE-279 requires that the scram sensors be fail safe (i.e., de-energize to fail) and that they be powered from essential ~~AC~~ electrical busses.

In light of this additional testimony, the ASLB found that the turbine building need not be constructed in accordance with 10 CFR 50 Appendix B, and that the turbine RPS sensors as designed provided adequate ~~protection~~ capability for their intended functions.

The basic theory accepted by the [redacted] ASLB, and very clearly presented to the ASLB by James P. Knight and which, is also clearly spelled out in the regulations, is that the level of protection required of plant systems should be directly proportional to the threat these systems pose to the health and safety of the public.

When the ASLB received additional information along these lines, it supported the original NRC authorization, i.e., that the turbine building need not be constructed in accordance with 10 CFR 50 Appendix B, and that the design [redacted] of the turbine S[redacted] inputs to the RPS ~~were~~ is acceptable.

The ASLB's [redacted] findings concerning these matters comprises Attachment VII.

Partly because of time constraints, the staff did not discuss in depth, and ASLB did not explicitely consider, the reliability of the turbine trip system when considering the scenario. However, that the ASLB considered the turbine trips to be of high quality and reliability, appropriate for inclusion in the RPS circuitry is clear from the ASLB findings and transcript.

HEPK 19 SEP 78 BWR

Attachment 2

Scenario For Turbine Trip Without Associated Scram And With Failure Of The Turbine Bypass System.

The worst turbine trip sequence identified proceeds as detailed in the following description. The seismic event activates a turbine trip sensor causing a turbine trip. It is vital to note that turbine stop or throttle valve closure (turbine trip) is not the direct result of seismic motion. In addition, safety grade equipment is designed not to activate as a result of seismic motion. Unless a manual scram is performed, the plant will either ride through the seismic event or scram initiation will be the result of the response of systems or components which are not safety grade.

After the turbine trip, the following essentially incredible pair of events then occurs (1) no turbine generated scram is initiated and (2) the turbine bypass valves fail to open, or at least stick closed long enough for a portion of the fuel to experience cladding failure.

The bypass valves are "fail safe" in that their normal position is closed. The bypass valves are designed to remain closed if vacuum is too low and the valves will automatically close if vacuum drops below a ~~set~~ trip point.

With the above preamble ~~following introduction~~, the ~~[illegible]~~ essentially incredible scenario proceeds as follows:

(1) Turbine stop valves close (no turbine island scram).

(2) Turbine throttle valves close (no turbine island scram)

(3) Turbine bypass valves are assumed to say closed.

(4) The reactor scrams on high flux. (about 1 to 2 seconds into the event)

(5) About 7% of the fuel may experience transition boiling. Clad temperatures remain below those usually associated with clad perforation (See Attachment ___)

(6) Safety relief valves discharge contaminated steam to the suppression pool.

(7) By current core performance branch ground rules, the fuel experiencing transition boiling perforates (melting of the fuel is not assumed because

of low maximum fuel pellet temperatures. Clad perforation is presumed to result from pellet/clad interaction (PCI); not so much because PCI induced failures are expected, but because the Core Performance Branch apparently believes that they cannot be ruled out for this event.

(B) Part of the released fission products are sensed by the radiation detectors in the main steam line. This activates the main steam line isolation valves to close in about five seconds. The fission products are released from the fuel in a time dependent manner. Similarly, the release of fission products to the steam from the reactor coolant will be time dependent. If little or no steam is flowing to the turbine building, the steam line radiation detectors would be activated by a "diffusing" radioactive flow. In this instance, little radioactivity would pass by the MSIVs before they are closed (~5 seconds closure time).

If the bypass valves stay shut long enough for the postulation of fuel failure, then they should be assumed to remain shut for the course of the event. Sustained steam flow for any reason could prevent fuel perturbation.

(9) At this stage, it would take something like a steam line break or main condenser failure to release substantial amounts of radioactivity to the environment.

(10) Radioactivity , inside closed MSIVs would pass through the safety relief valves and into the suppression pool. Doses released from the containment would be small.

Recd 15 Sep 77
NEP King

~~Exclude~~ in
~~mass~~
ATTACHMENT IV

[illegible] TO: Ha[illegible]ville File

FROM: F. [illegible]nton, S[illegible]e Analyst, Section B, Accident Analysis Branch, DSI

SUBJECT: TURBINE TRIP [illegible]SES THROUGH SUPPRESSION POOL PATHWAY FOR HARTSVILLE HEARING ON FEBRUARY 25, 1977

Analysis in testimony* assumes activity from failed fuel [instantaneously] transferred to condenser. This conservative and unrealistic assessment has shown acceptable consequences. In reality, the initiating event (valve closure) will occur before fuel failure and will prevent contaminated steam from reaching the condenser. The sequence of events is as follows:

1. seismic [illegible]
2. turbine stop valve closes (0.8s) with failure of trip sensors
3. reactor power and pressure excursion; 7% fuel experiences boiling transition (1-2s)
4. safety relief discharges contaminated steam to suppression pool
5. some activity and pressure (slight) buildup in containment
6. assume activity is leaked from containment at same rate fraction as for LOCA (conservative)
7. containment (purge lines) will isolate on scram signal which occurs at [illegible] power high flux signal

Radiological consequences:

- 7% of fuel rods experience boiling transition within [illegible] sec and these rods are assumed to fail releasing gap activity (10% I and NGs)
  PF = 1.5 assumed I
- 10% of [illegible] [illegible] to primary coolant water becomes volatile in [illegible] [illegible]

---

*Supplemental testimony in response to Board questions by [illegible] Thomas entered into record on February 25, 1977.

- all noble gases released to water reach steam dome
- the safety relief valves vent the activity to the suppression pool
- no credit for retention of iodines or noble gases in suppression pool
- the activity is released to the containment
- containment is assumed to pressurize and primary containment leaks at same rate as that assumed for LOCA (0.3% per day)
- the relative doses from turbine trip in comparison to LOCA (0-2 hr) doses are as follows:

	LOCA	Turbine Trip
Core fraction affected	1.0	.07
Amount iodine released	.5 x .5	.1 x .1 (1.5)
Amount noble gases released	1.0	.1 x (1.5)
Total iodine released to containment	.25	.00105
Total noble gases released to containment	1.0	.0105

Peaking Factor

Therefore, turbine trip doses can be ratioed in comparison to LOCA doses.

Thyroid dose for TT is $\frac{.00105}{.25}$ = .004

WB dose for TT is $\frac{.0105}{1}$ = .0105

Hartsville LOCA doses shown in Section 15.3 of SER are 8 rem thyroid and 5 rem WB (0-2 hr).

Therefore, TT doses are .032 rem thyroid and .052 rem whole body [illegible]

for Hartsville compared to 2.5 [illegible]

Note DF inside core

We have not specifically analyzed the situation where, after turbine stop valve closure and fuel failure, the non seismic portion of the steam lines fail. The overall probability is so remote that it need not be considered.

Required sequence of events is:

to open

1. Severe seismic event which causes simultaneous closure of turbine control and stop valves and failure of bypass system.
2. Simultaneous failure of all turbine building trip signals.
3. Steam line break at a time greater than 1.5 to 2 sec after turbine valve closures.
4. MSIV closure in broken line 5.0 sec after steam line break.

These are fail safe. i.e SCRAM should result.

F. Kantor, Site Analyst
Section B
Accident Analysis Branch
Division of Site Safety and
Environmental Analysis

Note: Analysis shown above developed by G. Chi[illegible] and [illegible] Coffe[illegible].
The analysis was not entered directly in [illegible] by [illegible] Thomas
and F. Kantor [illegible]
[illegible]

Note, Safety relief valves open on Hi pressure instantly.

HEPK 19 SEP 78 Attachment [illegible] I

Turbine Generator Considerations Related to Seismic Input.

~~[illegible]~~

Seismic motion is not a direct cause of a turbine trip. For a trip to occur, seismic excitation must cause a condition which generates a turbine trip signal.

No fuel damage will occur as long as there is steam flow through the main steam lines. Even partial flow can prevent fuel damage.

Nor is it necessarily true that a turbine will trip during the early part of the seismic motion. Typical turbine trip conditions are given in the attached Table 3.3-1.

(Based upon conversations with Westinghouse, Pacific Gas and Electric, and the Los Angeles Department of Water and Power, actual data on the effects of seismic input to any type of central power generation station are too sparse to be of value, even in California.

It appears that many operators disconnect the turbine/generator high vibration sensors because of concerns regarding spurious trips. Thus, with respect to shaft deflection, seismic input would most likely be detected by the thrust bearing wear indicator. For the other turbine trips to actuate, either a manual trip must be executed or damage to the turbine generator or its supporting systems, or loss of condenser vacuum must occur.

According to Westinghouse, depending upon the direction of ground motion, turbine generator sets can take high seismic inputs without tripping. Westinghouse and Bechtel further believe that turbine stop and throttle valve closure with failure to actuate the scram sensors is essentially incredible. Turbines are relatively massive devices. Because, the turbine stop valves are designed to close within about 0.2 seconds, as compared to about 5 seconds for MSIVs, some believe that the

<u>actual</u> quality (and reliability) of turbine stop valves is higher than that of MSIVs. In addition, there are usually one redundant set of limit switches on the stop valves and another redundant set on the throttle valves. A few turbines also have trip diversity in that, instead of limit switches, the throttle valve scram actuator is an oil pressure sensor.

Attachment (VI) consists of a General Electric realibility analysis performed to predict the probability of failure to scram, given a turbine trip or a generator trip. GE concluded that the probability of failing to scram the reactor following a turbine or generator trip is of the order of 10^{-6} per demand, considering random failures only.

ATTACHMENT III-3

Harry K

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

Tennessee Valley Authority

(Hartsville Nuclear Plant, Units A1, A2, B1 and B2)

Docket No.
STN 50-518
STN 50-519
STN 50-520
STN 50-521

SUPPLEMENTAL TESTIMONY OF NRC STAFF
IN RESPONSE TO BOARD QUESTIONS
By
James D. Thomas

This testimony is offered in response to the Board's questions concerning turbine trip expressed in the Board's Partial Initial Decision of December 10, 1976, which reads as follows:

"The Board adds the following issue to the health and safety phase of the hearing, sua sponte:

1. Should the turbine trips of the reactor protection system and the turbine bypass system be seismically qualified?
2. Should the turbine building be seismically qualified?
3. Should the reactor protection system receive shutdown signals from buildings outside the nuclear island?"

In the GESSAR-238 design, the reactor trip system (RTS) receives inputs from sensors located in the turbine building (generally a non-seismic Category I structure). These sensors monitor the open-closed status of the turbine stop and control valves and provide the RTS a scram signal should these valves begin to close during power operation (turbine stop or control valve fast closure is indicative of a pending turbine trip).

The singular function of these RTS turbine trip sensors is to protect the fuel during turbine trip transients by providing a scram signal in sufficient time to prevent the fuel from exceeding its design safety limit (a minimum critical power ratio of 1.07).

Since these sensors will be designed to IEEE-279-1971 requirements, (requirements applied to all other RTS sensors), there is a high probability that the design safety limits on the fuel will not be exceeded during anticipated operational occurrences (in accordance to GDC 20 and 29).

α 10⁻⁶/yr SSE

Primes

These RTS turbine trip sensors are not required to function during any design basis accident or during the safe shutdown earthquake and the General Electric Company (GE) does not take credit for their function in the GESSAR-238 accident analysis calculations.

Blds Only

However, due to the fact that these RTS turbine trip sensors are located in a non-seismic Category I area (turbine building), we required GE to analyze the consequences of the failure of the RTS trubine trip sensors and demonstrate that other RTS sensors are available to assure that a reactor scram could be achieved.

In their response, GE stated that the probability of failure is small - on the order of 10^{-5} per year. Further, GE calculated the percentage of the fuel rods in the core which could be subject to boiling transition as the result of the failure of the RTS turbine trip sensors when they are called upon to operate. The most limiting case identified was that for a turbine trip at full power with a failure of the turbine bypass system which results in 7 percent of the fuel rods being subject to boiling transition. For this event the reactor scram signal generated from the high neutron flux sensors to the RTS will terminate the transient and maintain reactor system pressure well below the ASME allowable limit.

We have reviewed the GE response and conclude that the GE analysis is sufficiently conservative and that there exists, based on our own calculations, a margin in excess of 1200 Fahrenheit between the temperature the fuel experiences during the event, and the predicted end-of-life fuel melting temperature.

We have analyzed the radiological consequences of this event using the following assumptions:

(1) The 7 percent of the fuel rods in the core which experience transition boiling will immediately perforate and release the activity contained in the fuel-clad gap to the reactor coolant.

(2) The containment and reactor vessel isolation control system does not operate until all the fission products released to the reactor coolant have escaped to the turbine condensor.

(3) The fission products are then released from the turbine condensor in accordance to the criteria specified in the Appendix to Standard Review Plan 15.4.7.

Utilizing these parameters, we calculate a 0-2 hour dose of 7.6 Rem to the thyroid and 0.7 Rem whole body assuming a relative concentration of 1×10^{-3} sec/m^3, and a course of event dose of 5.0 Rem to the thyroid and 0.2 Rem whole body assuming a relative concentration value of 1×10^{-4} sec/m^3.

(These doses were given in Supplement No. 2 to the Safety Evaluation Report for GESSAR-238). Due to more favorable meteorology, the 0-2 hour doses at the Hartsville site for this event would be 3.7 Rem to the thyroid and 0.4 Rem whole body and the course of the event dose would be 2.6 Rem to the thyroid and 0.1 Rem whole body.

Based on our review of the information submitted by GE and our own calculations, we have concluded the following:

(1) There is a high probability that the RTS turbine trip sensors will prevent the fuel from exceeding safety limits during anticipated turbine trip transients.

(2) The probability of failure of the RTS turbine trip sensors is low.

(3) In the event of RTS turbine trip sensor failure, other RTS sensors are available which can effect a reactor scram and maintain reactor system pressure below the ASME allowable limit.

(4) The radiological consequences of the RTS turbine trip failure event are small fractions of the 10 CFR Part 100 guideline exposures.

Based on the above conclusions, our specific responses to the Board's questions are as follows:

(1) The turbine trip sensors of the reactor protection system and the turbine bypass system are not required to be seismically qualified since neither is required to function during a safe shutdown earthquake.

(2) The turbine building is not required to be seismically qualified since it contains no components connected to the reactor protection system that are required to function during the safe shutdown earthquake.

(3) The reactor protection system in the GESSAR-238 design may receive signals from outside the nuclear island since these trip signals assure that the design safety margins for the fuel will be maintained during anticipated operational occurrences.

Attachment VI ~~Relevant to~~ ~~A~~ ~~Synthesis~~. KRUG

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QUESTION 222.22 (Attachment A)

Availability of turbine trip and control valve fast closure trip scrams.

ABSTRACT

A reliability analysis has been performed to predict the probability of failure to scram, given a turbine trip or a generator trip. A mathematical model of the system has been developed showing the relationship between component failure and system failure. The known or assumed failure rates, test intervals, repair rates, and logic for the components are of failure inputs to a computer program which performs the reliability/availability calculations.

The impact of "common mode failures" on the system is discussed.

CONCLUSIONS

The probability of failing to scram the reactor following a turbine or generator trip is of the order of 10^{-6} per demand, considering random failures only. There are no single point failures. The most dominant failure combinations involve the pressure permissive that bypasses these scrams below 30% power level. This bypass is common to both channels, turbine trip and generator trip scrams.

METHOD OF SOLUTION

A Reliability Block Diagram (RBD) is developed for each problem, one for the Turbine Trip Scram (Figure 1) and one for the Control Valve Fast Closure Trip Scram (Figure 2). The RBD is a "success" oriented diagram. The object is to represent the logic of the system in such a way that if a path can be traced from beginning to end through "good" components, the system is good. The multiple paths depict the redundant paths to success so that if a failed component is encountered in one path, success may still be achieved by utilizing another path with no failures.

Some components may be important to the success of more than one path. For example, component 105 in Figure 1 is a mounting bracket that holds two switches that are used in two different circuits. For this reason, component 105 must appear in two different paths. The solution to the problem must take into account that a failed component fails all the paths it appears in.

The solution to the RBD is programmed for GAMM (Ref. 1). GAMM accepts the input data on the failure rates, repair times, test intervals and logic, and computes a numerical availability for the system.

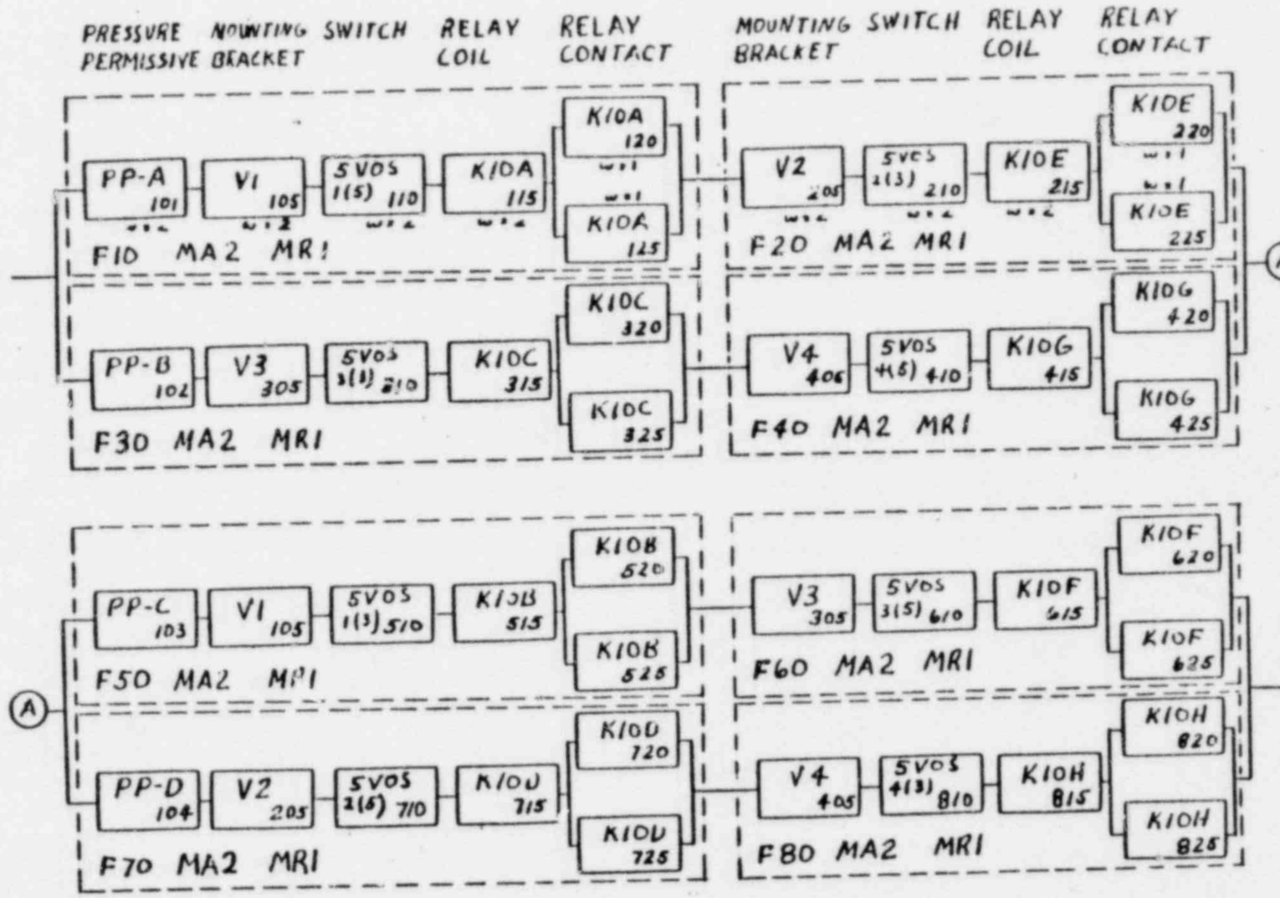

Figure 1. Turbine Trip Scram Availability Model

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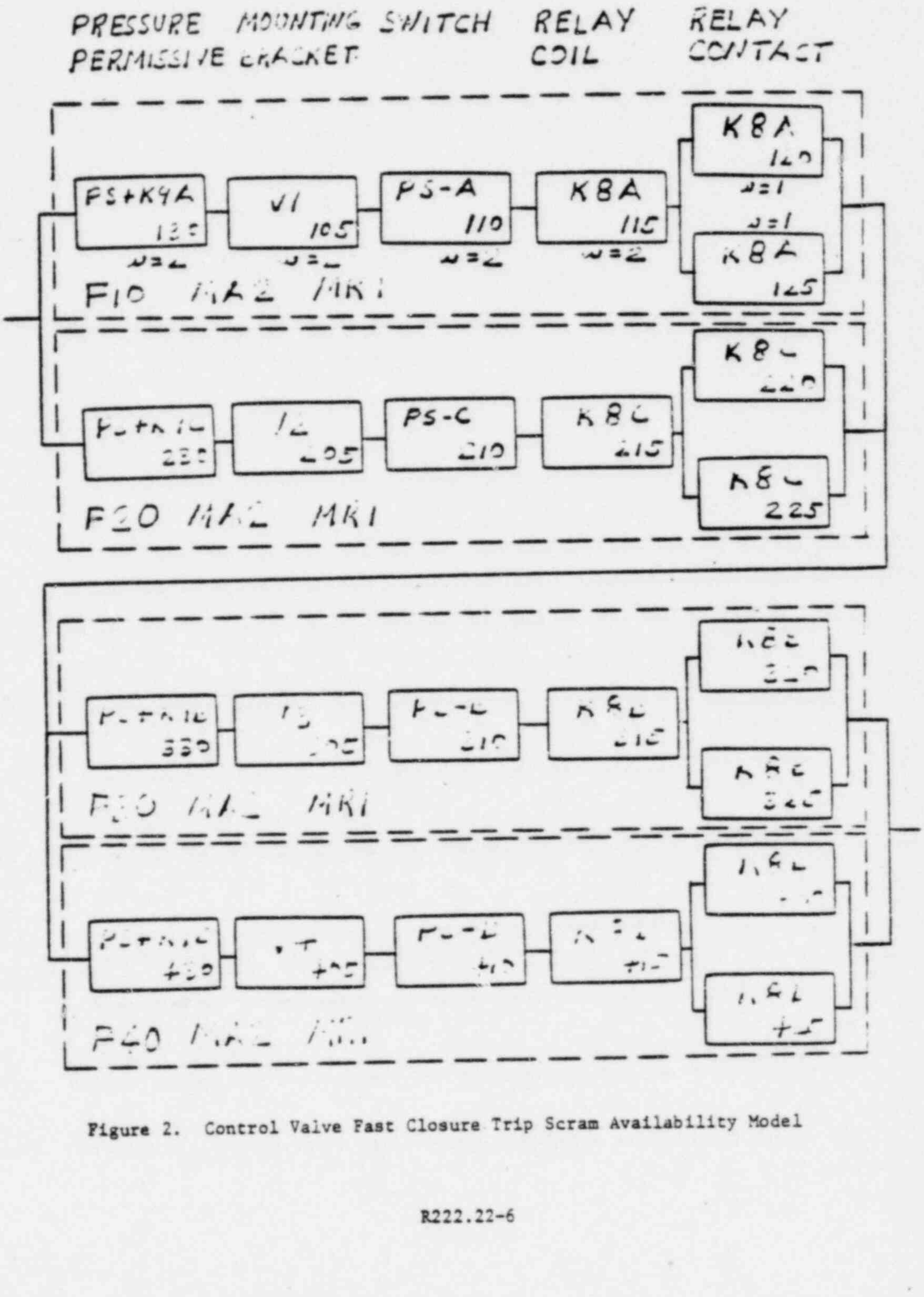

Figure 2. Control Valve Fast Closure Trip Scram Availability Model

ASSUMPTIONS

The following assumptions realistically bound the scope of these problems to make them tractable.

1. Given a turbine trip, all four turbine stop valves close.
2. Given a control valve fast closure trip, the pressure in the hydraulic system actuating the control valve exceeds the trip point on all four pressure switches.
3. The turbine stop valves are "exercised" daily, thus verifying the mounting bracket, the switch, and the relay (except the contacts). (Large Steam Turbine recommends daily testing.)
4. The control valve fast closure trip is exercised weekly, thus verifying the pressure switches and the relay (except the contacts). (Large Steam Turbine recommends weekly testing.)
5. The pressure permissive which bypasses the two scram functions below 30% equivalent power is effectively tested each time.
6. Redundant relay contacts are tested when the channels are functionally tested. According to Reference 2, such testing shall be performed at least semi-annually.
7. The failure rates for the components (except for the mounting bracket) are as follows:

Component	Failure/10^6 Hours
Pressure Switch	1.1
Limit Switch	1.1
Relay Coil (and armature)	0.4
Relay Contact (fail to open)	0.03
Mounting Bracket	0.01 (estimate)

8. The Reactor Protection System Logic beyond the channel inputs from turbine trip scram and control valve fast closure trip scram was not modeled because it was assumed to have a low probability of failure relative to the trip channels.
9. The numerical predictions of availability assume only random failures.

RESULTS, TURBINE TRIP SCRAM

The solution to the Trubine Trip Scram problem diagrammed in Figure 1 using the failure rates and test intervals from the assumptions was obtained by using the GAMM computer program (Reference 1). As indicated on Table 1, the unavailability, that is the probability that the turbine trip scram will fail on demand, is 9.9×10^{-7}.

Table 1 also shows the top 20 combinations that contribute to failure. Over 88 percent of the unavailability is charged to the two pairs of pressure permissives that bypass the scram signal below 30% equivalent power. These paired components are 101 and 102, and 103 and 104. In addition, each of these same pressure permissives are paired with other components to contribute an additional 10 percent unavailability. This makes for a total of approximately 98.5 percent of the unavailability of the system that involves the pressure permissives.

Table 2 shows the effect each component has on system unavailability. The sensitivity index is the delta change in system unavailability, given that the component failure rate is changed by 10 percent. Thus, the higher the sensitivity index figure, the more influence that component will have on system unavailability. As might be expected, the pressure permissives rank high, the first four. These are followed by the limit switches ranking 5 through 12, but with a sensitivity index of only 2.5% of that of the pressure permissives. The least sensitive are the contacts which are highly redundant and so have a minimal effect on the system unavailability.

GAMM calculates the system unavailability, given that a component is failed. Of particular interest here is the mounting bracket which holds two limit switches. Given that a mounting bracket (say component 105) is failed, the system unavailability is 1.6×10^{-3}. This is of interest because it simulates the condition that would occur if only three of the four turbine stop valves close. In other words, if only three turbine stop valves actually close, the probability of failure to scram increases from 9.9×10^{-7} to 1.6×10^{-3}, but the pressure transient associated with only three valves closing is not as severe as with four closing. (The effect on the turbine overspeed is not considered.) Tables 1 and 2 also show a calculated unreliability, but that calculation has no meaning that relates to the problem at hand and should be ignored.

Table 1

MAJOR CONTRIBUTORS TO SYSTEM UNAVAILABILITY

TURBINE TRIP SCRAM

Component Combinations		Unavailability	Percent
103	104	4.3909500E-07	44.2732
101	102	4.3909500E-07	44.2732
101	410	9.0672970E-09	0.9142
102	110	9.0672970E-09	0.9142
103	810	9.0672970E-09	0.9142
101	310	9.0672970E-09	0.9142
102	210	9.0672970E-09	0.9142
104	510	9.0672970E-09	0.9142
103	710	9.0672970E-09	0.9142
104	610	9.0672970E-09	0.9142
103	715	3.2971989E-09	0.3325
102	215	3.2971989E-09	0.3325
103	815	3.2971989E-09	0.3325
101	415	3.2971989E-09	0.3325
102	115	3.2971989E-09	0.3325
101	315	3.2971989E-09	0.3325
104	515	3.2971989E-09	0.3325
104	615	3.2971989E-09	0.3325
110	310	9.2927999E-10	0.0937
110	410	9.2927999E-10	0.0937

Table 2

COMPONENT EFFECTS ON SYSTEM UNAVAILABILITY

TURBINE TRIP SCRAM

Component		System Unavailability				System Unreliability			
		Sensitivity		With Comp	With Comp	Sensitivity		With Comp	With Comp
ID	Name	Index	Rank	Perfect	Failed	Index	Rank	Perfect	Failed
101	Press Permissive	4.643E-08	1	5.274E-07	8.014E-04	5.804E-06	1	2.857E-04	1.357E-02
102	Press Permissive	4.643E-08	2	5.274E-07	8.014E-04	5.804E-06	2	2.875E-04	1.357E-02
103	Press Permissive	4.641E-08	4	5.276E-07	8.010E-04	5.795E-06	3	2.858E-04	1.355E-02
104	Press Permissive	4.641E-08	3	5.276E-07	8.010E-04	5.795E-06	4	2.858E-04	1.355E-02
105	MT BKT	2.107E-11	21	9.915E-07	1.599E-03	7.534E-08	21	3.430E-04	2.614E-02
110	SW VLV 5VOS Z15<	1.160E-09	6	9.801E-07	8.010E-04	4.234E-06	5	3.014E-04	1.350E-02
115	K10A Coil	4.216E-10	14	9.875E-07	8.010E-04	1.536E-06	13	3.284E-04	1.349E-02
120	K10A Cont No. 1	1.439E-13	25	9.917E-07	1.106E-06	1.834E-09	25	3.438E-04	5.531E-04
125	K10A Cont No. 2	1.439E-13	26	9.917E-07	1.106E-06	1.834E-09	26	3.438E-04	5.531E-04
205	MT BKT	2.107E-11	22	9.915E-07	1.599E-03	7.528E-08	23	3.430E-04	2.612E-02
210	SW VLV 5VOS 2Z3<	1.159E-09	7	9.801E-07	8.010E-04	4.228E-06	7	3.015E-04	1.348E-02
215	K10E Coil	4.216E-10	15	9.875E-07	8.009E-04	1.534E-06	15	3.284E-04	1.347E-0.
220	K10E Cont No. 1	1.452E-13	29	9.917E-07	1.093E-06	1.666E-09	29	3.438E-04	5.340E-04
225	K10E Cont No. 2	1.432E-13	30	9.917E-07	1.093E-06	1.666E-09	30	3.438E-04	5.340E-04
305	MT BKT	2.107E-11	23	9.915E-07	1.599E-03	7.534E-08	22	3.430E-04	2.614E0-2
310	SW VLV 5VOS 3Z3<	1.160E-09	5	9.801E-07	8.010E-04	4.234E-06	6	3.014E-04	1.350E-02
315	K10C Coil	4.216E-10	13	9.875E-07	8.010E-04	1.536E-06	14	3.284E-04	1.349E-02
320	K10C Cont No. 1	1.439E-13	27	9.917E-07	1.106E-06	1.832E-09	27	3.438E-04	5.529E-04
325	K10C Cont No. 2	1.439E-13	28	9.917E-07	1.106E-06	1.832E-09	28	3.438E-04	5.529E-04
405	MT BKT	2.107E-11	24	9.915E-07	1.599E-03	7.517E-08	24	3.430E-04	2.608E-02

Table 2 (Continued)

Component		System Unavailability				System Unreliability			
		Sensitivity		With Comp	With Comp	Sensitivity		With Comp	With Comp
ID	Name	Index	Rank	Perfect	Failed	Index	Rank	Perfect	Failed
410	SW VLV 5VOS 4%5<	1.159E-09	8	9.801E-07	8.009E-04	4.215E-06	8	3.016E-04	1.345E-02
415	K10G Coil	4.215E-10	16	9.875E-07	8.009E-04	1.530E-06	16	3.285E-04	1.343E-02
420	K10G Cont No. 1	1.430E-13	31	9.917E-07	1.090E-06	1.330E-09	31	3.438E-04	4.956E-04
425	K10G Cont No. 2	1.430E-13	32	9.917E-07	1.090E-06	1.330E-09	32	3.438E-04	4.956E-04
510	SW VLV 5VOS 1%3<	1.159E-09	10	9.801E-07	8.006E-04	4.185E-06	9	3.019E-04	1.335E-02
515	K10B Coil	4.213E-10	18	9.875E-07	8.006E-04	1.519E-08	17	3.286E-04	1.334E-02
520	K10B Cont No. 1	1.407E-13	33	9.917E-07	1.049E-06	5.095E-10	33	3.438E-04	4.019E-04
525	K10B Cont No. 2	1.407E-13	34	9.917E-07	1.049E-06	5.095E-10	34	3.438E-04	4.019E-04
610	SW VLV 5VOS 3%5<	1.158E-09	11	9.801E-07	8.006E-04	4.179E-06	11	3.020E-04	1.333E-02
615	K10F Coil	4.212E-10	19	9.875E-07	8.005E-04	1.516E-06	19	3.286E-04	1.332E-02
620	K10F Cont No. 1	1.400E-13	37	9.917E-07	1.037E-06	3.415E-10	37	3.438E-04	3.828E-04
625	K10F Cont No. 2	1.400E-13	38	9.917E-07	1.037E-06	3.415E-10	38	3.438E-04	3.828E-04
710	SW VLV 5VOS 2%5<	1.159E-09	9	9.801E-07	8.007E-04	4.185E-06	10	3.019E-04	1.335E-02
715	K10D Coil	4.213E-10	17	9.875E-07	8.006E-04	1.519E-06	18	3.286E-04	1.334E-02
720	K10D Cont No. 1	1.407E-13	35	9.917E-07	1.049E-06	5.074E-10	35	3.438E-04	4.017E-04
725	K10D Cont No. 2	1.407E-13	36	9.917E-07	1.049E-06	5.074E-10	36	3.438E-04	4.017E-04
810	SW VLV 5VOS 4%3<	1.158E-09	12	9.801E-07	8.006E-04	4.167E-06	12	3.021E-04	1.330E-02
815	K10H Coil	4.212E-10	20	9.875E-07	8.005E-04	1.512E-06	20	3.287E-04	1.328E-02
820	K10H Cont No. 1	1.398E-13	39	9.917E-07	1.033E-06	9.927E-12	39	3.438E-04	3.449E-04
825	K10H Cont No. 2	1.398E-13	40	9.917E-07	1.033E-06	9.927E-12	40	3.438E-04	3.449E-04

RESULTS, CONTROL VALVE FAST CLOSURE TRIP SCRAM

The solution to the Control Valve Fast Closure Trip Scram problem diagrammed in Figure 2 was obtained by utilizing the GAMM computer program (Reference 1). The failure rates and test intervals are as indicated in the Assumptions Section. As indicated in Table 3, the unavailability, that is the probability that the control valve fast closure trip will fail on demand, is 1.3×10^{-6}.

Table 3 also shows the top 20 component combinations that contribute to failure. Nearly 69 percent of the unavailability is charged to the two pairs of pressure permissives that bypass the scram signal below 30% equivalent power. These paired components are 130 and 230, and 330 and 430. In addition, each of these same pressure permissives are paired with other components to contribute an additional 26 percent unavailability. This makes for a total of 95% of the unavailability of the system that involves the pressure permissives.

Table 4 shows the effect each component has on system unavailability. The sensitivity index is the delta change in system unavailability, given that the component failure rate is changed by 10 percent. Thus, the higher the sensitivity index figure, the more influence that component will have on system unavailability. The first four in sensitivity index rank are the pressure permissives, components 130, 230, 330, and [illegible] These are followed by the pressure switches ranking 5 through 8, the [illegible] coil (and armature) ranking 9 through 12, the mounting brackets ranking 13 through 15, and, finally, the relay contacts ranking 17 through 24.

DISCUSSION OF RESULTS

The failure rate of the mounting bracket was estimated at 0.01 failures per million hours. This is a relatively low failure rate which should be appropriate for a passive component such as a bracket. The sensitivity index shows that these mounting brackets are two to three orders of magnitude less influential than the pressure permissive, so a more precise estimate is not needed.

The most influential portion of the circuit is the pressure permissives associated with the 30% power level bypass. For example, if everything else in the circuit was perfect (could not fail), the unavailability of the turbine trip scram would only drop from 9.9×10^{-7} to 8.8×10^{-7}. Similarly, on the control valve fast closure trip scram, the unavailability would drop from 1.3×10^{-6} to 8.8×10^{-7}. The assumption that the pressure permissive will

Table 3

MAJOR CONTRIBUTORS TO SYSTEM UNAVAILABILITY

CONTROL VALVE FAST CLOSURE TRIP SCRAM

Component Combinations		Unavailability	Percent
130	230	4.3909500E-07	34.3942
330	430	4.3909500E-07	34.3942
130	210	6.1405356E-08	4.8099
110	230	6.1405356E-08	4.8099
310	430	6.1405356E-08	4.8099
330	410	6.1405356E-08	4.8099
315	430	2.2329220E-08	1.7490
130	215	2.2329220E-08	1.7490
115	230	2.2329220E-08	1.7490
330	415	2.2329220E-08	1.7490
110	210	1.6262400E-08	1.2738
310	410	1.6262400E-08	1.2738
110	215	5.9135999E-09	0.4632
310	415	5.9135999E-09	0.4632
115	210	5.9135999E-09	0.4632
315	410	5.9135999E-09	0.4632
115	215	2.1504000E-09	0.1684
315	415	2.1504000E-09	0.1684
105	230	5.5823051E-10	0.0437
305	430	5.5823051E-10	0.0437

Table 4

COMPONENT EFFECTS ON SYSTEM UNAVAILABILITY

CONTROL VALVE FAST CLOSURE TRIP SCRAM

Component		System Unavailability				System Unreliability			
ID	Name	Sensitivity Index	Sensitivity Rank	With Comp Perfect	With Comp Failed	Sensitivity Index	Sensitivity Rank	With Comp Perfect	With Comp Failed
105	MT BKT	7.614E-11	13	1.275E-06	8.206E-04	2.337E-07	13	1.352E-03	2.803E-02
110	SW Press	8.375E-09	5	1.192E-06	8.205E-04	2.559E-05	6	1.098E-03	2.778E-02
115	K8A Coil	3.045E-09	9	1.245E-06	8.205E-04	9.243E-06	9	1.262E-03	2.769E-02
120	K8A Contact No. 1	3.761E-12	19	1.276E-06	2.485E-06	2.654E-08	19	1.354E-03	2.364E-03
125	K8A Contact No. 1	3.761E-12	20	1.276E-06	2.485E-06	2.654E-08	20	1.354E-03	2.364E-03
130	Press Permissive A	5.236E-08	1	7.584E-07	8.199E-04	3.437E-05	1	1.010E-03	2.734E-02
205	MT BKT	7.614E-11	14	1.275E-06	8.206E-04	2.337E-0[illegible]	14	1.352E-03	2.803E-02
210	SW Press	8.375E-09	6	1.198E-06	8.205E-04	2.559E-05	5	1.098E-03	2.778E-02
215	K8C Coil	3.045E-09	10	1.245E-06	8.205E-04	9.243E-06	10	1.262E-03	2.769E-02
220	K8C Contact No. 1	3.761E-12	17	1.276E-06	2.485E-06	2.654E-08	17	1.354E-03	2.364E-03
225	K8C Contact No. 1	3.761E-12	18	1.276E-06	2.485E-06	2.654E-08	18	1.354E-03	2.364E-03
230	Press Permissive C	5.236E-08	2	7.524E-07	8.199E-04	3.437E-05	2	1.010E-03	2.734E-02
305	MT BKT	7.604E-11	15	1.275E-06	8.199E-04	2.278E-07	15	1.352E-03	2.736E-02
310	SW Press	8.364E-09	7	1.192E-06	8.198E-04	2.494E-05	7	1.105E-03	2.711E-02
315	K8B Coil	3.041E-09	11	1.295E-06	8.197E-04	9.008E-06	11	1.284E-03	2.702E-02
320	K8B Contact No. 1	1.887E-12	23	1.276E-06	1.847E-06	8.978E-09	23	1.354E-03	1.696E-03
325	K8B Contact No. 1	1.887E-12	24	1.276E-06	1.847E-06	8.978E-09	24	1.354E-03	1.696E-03
330	Press Permissive B	5.231E-08	4	7.529E-07	8.192E-04	3.350E-05	3	1.019E-03	2.668E-02

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Table 4 (Continued)

Component		System Unavailability				System Unreliability			
		Sensitivity		With Comp	With Comp	Sensitivity		With Comp	With Comp
ID	Name	Index	Rank	Perfect	Failed	Index	Rank	Perfect	Failed
405	MT BKT	7.604E-11	16	1.275E-06	8.199E-04	2.278E-07	16	1.352E-03	2.756E-02
410	SW Press	8.364E-09	8	1.192E-06	8.198E-04	2.494E-05	8	1.105E-03	2.711E-02
415	K8D Coil	3.041E-09	12	1.246E-06	8.197E-04	9.008E-06	12	1.264E-03	2.702E-02
420	K8D Contact No. 1	1.887E-12	21	1.276E-06	1.847E-06	8.978E-09	21	1.354E-03	1.696E-03
425	K8D Contact No. 1	1.887E-12	22	1.276E-06	1.847E-06	8.978E-09	22	1.354E-03	1.696E-03
430	Press Permis-sive D	5.231E-08	3	7.529E-07	8.192E-04	3.350E-05	4	1.019E-03	2.668E-02

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be tested every month is crucial to this result. If the reactor and turbine were to run for a very long time without reducing power below 30%, it would be necessary to test this pressure permissive with a pseudo signal.

Perhaps even more noteworthy is the fact that it is the <u>same</u> pressure permissive modeled into both systems; if it is failed on one it is failed on the other. The important thing to note is that the systems are not really independent. In particular, the control valve fast closure scram cannot effectively backup the turbine trip scram and vice versa. Stated another way, if the turbine trips off and the reactor fails to scram, it is unlikely that a subsequent fast closure of the control valve will cause a scram because the more probable cause of the original failure is the pressure permissive which is common to both systems.

The Reactor Protection System logic common to all channels and the Control Rod Drives were not modeled because as noted in Assumption 8 the probability of failure of that portion of the system was only 2×10^{-9} per demand. The results of this calculation justify that assumption; that the channel logic for both the turbine trip scram and control valve fast closure trip scram dominate the unavailability.

As stated in Assumption 9, the numerical results are based on the assumption of random failures only. All of the failures that contribute significantly to the overall probability of failure are combination failures to exactly two components. There are no single component failures that cause system failure and failures involving three or more failed components are of insignificantly low probability.

REFERENCES

1. GAMM - General Availability Mathematical Model, Version of 29 July 1971, General Electric Company.

2. Standard Technical Specifications, General Electric Boiling Water Reactors, Draft 1975.

3. 22A2689, Recommended Component Failure Rates for Use in Reliability/Availability Analysis.