ATWS DISTRIBUTION

A. Thadani
T. Su
L. Ruth
K. Parczewski
M. Srinivasan
H. Vander Molen
M. Tokar
D. Thatcher
R. Kendall
F. Akstulewicz
F. Cherny
M. Aycock
T. Novak
R. Tedesco
R. Denise
R. Mattson
K. Kniel
T. Speis
P. Check
D. Eisenhut
B. Grimes
R. Bosnak
D. Muller
F. Schroeder
J. Norberg
E. Jakel
H. Richings

ACRS (21)
PDR
GIB R/F
NRR R/F
Central/Docket Files
Subject File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 2 0 1980

MEMORANDUM FOR: Karl Kniel, Chief
Generic Issues Branch, DST

FROM: Ashok C. Thadani
Generic Issues Branch, DST

SUBJECT: NRC-EPRI ATWS MEETING SUMMARY

The staff met with the Electric Power Research Institute (EPRI) on May 5, 1980 to discuss the EPRI as well as the NRC considerations of the significant transients, the frequencies of these transients, and the testing frequencies of the electrical portions of the scram systems.

I. EPRI Presentation on Frequency of Anticipated Transients

The EPRI analyses (Enclosure 2) concludes that:

- the total frequency of anticipated transients is 10.59 per reactor year for PWRs and 9.37 per reactor year for BWRs.
- the transients important for ATWS consideration have frequencies of 3.74/RY and 4.7/RY for PWRs and BWRs respectively.
- the ATWS events below 25% rated power level do not result in severe consequences and thus the frequencies of transients of significance is further reduced to 1.96/RY and 3.52/RY for PWRs and BWRs respectively.
- the extropolation of two transients using the learning curve (first year frequency + 39 x average frequency of years 2 through 8) /40 and individual plant design considerations would further reduce the significant transient frequencies to

  1.45/RY for B&W designed plants
  1.65/RY for CE designed plants
  1.18/RY for W designed plants
  3.52/RY for GE designed plants

Staff Comments:

The following Staff Comments were provided to EPRI concerning the frequency of significant transients in PWRs and BWRs.

PWRs

- The list of significant transients considered by EPRI was incomplete. The list should have as a mimimum included events of pressurizer relief or safety valve stuck open, safety injection actuation, feedwater flow instability, loss of circulation water and loss of power to the necessary plant system. Further, additional events which result in steam generator isolation (e.g., low steam generator pressure) and/or tripping of the main-feedwater system should also be included because these events would result in mismatch between power generation and heat removal capability.

- Exclusion of events below 25% power may be inappropriate because of 1) unavailability of auxiliary feedwater system (which may not be automatically actuated due to Common Mode Failures (CMF) in the scram system), 2) more severe value of the moderator temperature coefficient (MTC) and 3) the calculated consequences from rod withdrawal at subcritical conditions are severe.

- The significant transient data should be averaged over the first five years experience since the experience beyond five years is small. The data should not be extrapolated to the projected forty year plant lifetime.

- Using the EPRI data, the NRC staff estimates that the significant transient frequency for ATWS considerations is approximately five per reactor year. This conclusion is further supported by the experience with B&W plants as discussed in the draft NUREG-0667 report.

For the reasons enumerated above the staff did not agree with the EPRI assessment that the transient frequency for ATWS consideration is between one and two per reactor year for PWRs.

BWRs

- The staff noted that the EPRI significant transient list was incomplete. The list should have included inadvertent opening of safety or relief valves, turbine bypass problems, trip of main steam isolation valve (MSIV), loss of feedwater heating and any other events that result in reactor vessel isolation.

- Most of the isolation type events (Note - with any fuel failure, the condenser would remain isolated) at about 25% power level are not significantly different than those events at higher initial power levels (the staff referenced NEDO-10349, a GE ATWS report) because of concerns with the energy deposited in the suppression pool and the potential for flux oscillations.

- As in the case of the PWRs the data for transients should not be extended to 40 year projected plant lifetime and because of limited experience beyond five years for any plant, the data should be averaged over the first five years of operation.

Thus, on the basis of the EPRI data as well as other sources of data the staff concludes that the frequency of significant transients is approximately 8/RY and not 3.52/RY as claimed by EPRI.

II. RPS Testing Frequency

In its presentation, EPRI proposed (Enclosure 3) that the correct testing frequency for RPS electrical of the reactor protection system portions is approximately 100-200 per reactor year for BWRs and approximately 24 to 100 per reactor year for PWRs. EPRI also noted that the breakers do not dominate the RPS unavailability.

The staff responded that:

- each channel test is not an appropriate scram system test since the concern is with common mode failures.
- Full Scram system tests are completed once per month (as required by technical specifications) although subsystem tests are staggered through the month. Some limited portions of BWR scram system are tested more frequently (4 times/month) at some plants.
- Tests may not detect all CMFs. For example, if ten percent of CMFs are undetectable by tests, then increasing the frequency of testing significantly will have little impact on the overall reliability of the scram system.
- Extensive testing could introduce CMFs.
- The assumption of independence in assessing the contribution of the breakers to the scram unreliability may be invalid.
- The staff noted that increasing the frequency of testing would not have an appreciable influence on overall scram unreliability since some consideration was given to higher testing frequency in the final scram unreliability estimates given in NUREG-0460, Vol. 3 and 4. The following table summarizes the influence of different testing frequency.

Assumptions: χ^2 Distribution
900 Reactor Years Experience
(Updated as per EPRI estimate)
One Scram Failure.

Test Frequency	Electrical Portion Unreliability 95% Conf.	50% Conf.
12	2.2×10^{-4}	8×10^{-5}
24	1.1×10^{-4}	4×10^{-5}
50	5.3×10^{-5}	1.9×10^{-5}

NUREG-0460, Vol, 3,4 Value

Electrical Portion	1.5×10^{-5}
Hydraulic/Mechanical	1.5×10^{-5}

In conclusion the staff noted that the ATWS record is substantial in terms of data analysis and any further studies are unlikely to appreciably change the conclusion.

The list of attendees is given in Enclosure 1.

A. Thadani

A. Thadani
Generic Issues Branch
Division of Safety Technology

Enclosures:
As stated

ENCLOSURE 1

EPRI - MEETING ON ATWS
MAY 5, 1980

A. Thadani	NRC/DST
Lee Abramson	NRC/ASB
G. S. Lellouche	EPRR
I. B. Wall	EPRI
J. W. Cleveland	SAI/Palo Alto
G. B. Peeler	NUS
Armand Lakner	NRC
Rick Kendall	NRC/ICSB
Dale Thatcher	NRC/ICSB
Stephen Maloney	Boston Edison/AIF
M. Srinivasan	NRC/ICSB
F. Schroeder	NRC/DST
K. Kniel	NRC/DST

ENCLOSURE 2

ENCLOSURE 2

Limiting Transients for ATWS*

I. Babcock & Wilcox

A. Loss of offsite power (LOOP)

B. Total loss of feedwater (LOF)

C. Transients leading to LOF (LOL)

II. Combustion Engineering

A. 2560 MWt Core

1. Uncontrolled rod withdrawal (CEA)
2. Partial loss of feedwater (PLOF)
3. Loss of load (LOL)
4. Total loss of feedwater (LOF)

B. 3800 MWt Core

1. Uncontrolled rod withdrawal
2. Partial loss of primary coolant flow (PPCF)
3. Loss of load
4. Total loss of feedwater

III. Westinghosue (No transient yields results of significance but the most limiting transients are the following)

A. Loss of load

B. Total loss of feedwater

IV. General Electric

Any transient leading to excessive pool temperatures (GE)

---

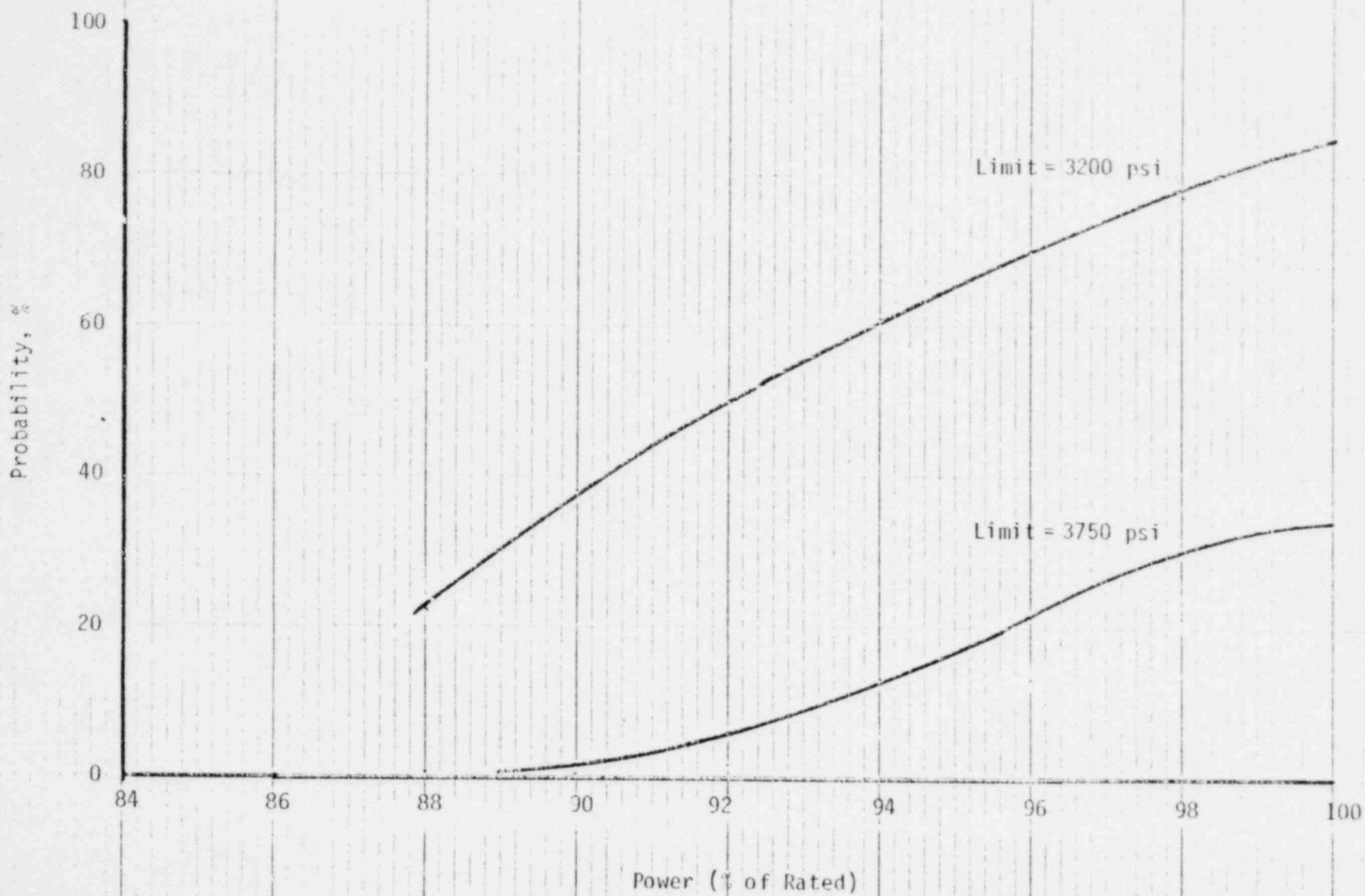
* These transients have been specified by NRC in WASH 1270 and the Status Reports as being those which lead to excessive pressures.

Correspondence Between Significant ATWS
Transients and Plant Transient Data

ATWS Transient	Plant Transient	
PWR		
PPCF	# 1*	Loss of RCS (1 Loop)
CEA	# 2	Uncontrolled Rod Withdrawal
PLOF	#15	Loss or Reduction in Feedwater Flow (1 Loop)
LOF	#16	Total Loss of Feedwater Flow (All Loops)
LOL	#18	Closure of All MSIV
	#24	Loss of Condensate Pumps (All Loops)
	#25	Loss of Condensor Vacuum (LCV(
	#33	Turbine Trip (TT)
	#34	Generator Trip (GT)
LOOP	#35	Loss of Station Power
BWR	# 1	Load Rejection
	# 3	Turbine Trip
	# 5	MSIV (All Loops)
	# 8	Loss of Condenser Vacuum
	# 9	Pressure Regulator Fails Open
	#10	Pressure Regulator Fails Closed
	#20	Feedwater, Increasing Flow at Power
	#24	Feedwater, Low Flow
	#31	Loss of Offsite Power
	#32	Loss of Auxiliary Power

* This number refers to the detailed transient frequencies presented in EPRI NP 801

Probability of Exceeding Pressure Limit
For a B & W Loss of Feedwater



Effect of Power Level on Transient Frequency

PWR's	P ≥ 0	P ≥ 25%	P ≥ 50%
All transients	10.59	5.26	3.4
ATWS*	3.74	1.96	1.6

BWR's	P ≥ 0	P ≥ 25%	P ≥ 50%
All Transients	9.37	6.72	5.6
ATWS	4.7	3.52	3.38

* For PWR's the ATWS numbers are for all ATWS transients without discriminating as to NSSS vendors; Westinghouse still would be zero.

Estimators of the Mean Occurrence Rate in BWRs for Power > 25% of Full Power[1]

Transient # (See Table IV)	Point Value (event/year)[3]	Estimator 5%	Estimator 50%	Estimator 95%
BWR				
#1	0.66[2]	0.44	0.65	0.92
#3	0.65	0.46	0.65	0.94
#5	0.49	0.32	0.50	0.70
#8	0.53	0.38	0.53	0.76
#9	0.21	0.13	0.22	0.37
#10	0.14	0.08	0.15	0.28
#20	0.28	0.18	0.29	0.45
#24	0.33	0.22	0.34	0.51
#32	0.02	0.01	0.04	0.11
#31 Loss of Offsite Power	0.14	0.08	0.15	0.28

Estimators for the Mean Occurrence Rate in PWRs for Power > 25% of Full Power

Transient # (See Table IV)	Point Value (event/year)[3]	Estimator 5%	Estimator 50%	Estimator 95%
PWR				
#1	0.12	0.06	0.13	0.22
#2	0.01	0.00	0.02	0.06
#15	0.45	0.33	0.45	0.61
#16	0.07	0.03	0.07	0.14
#18	0.07	0.03	0.07	0.14
#24	0.0	0.00	0.01	0.05
#25	0.08	0.04	0.08	0.16
#33	0.68	0.53	0.68	0.86
#34	0.21	0.13	0.21	0.32
#35	0.27	0.18	0.27	0.40

1 These tables are taken from EPRI NP801
2 This value (0.66) should read 0.65
3 Reactor Year

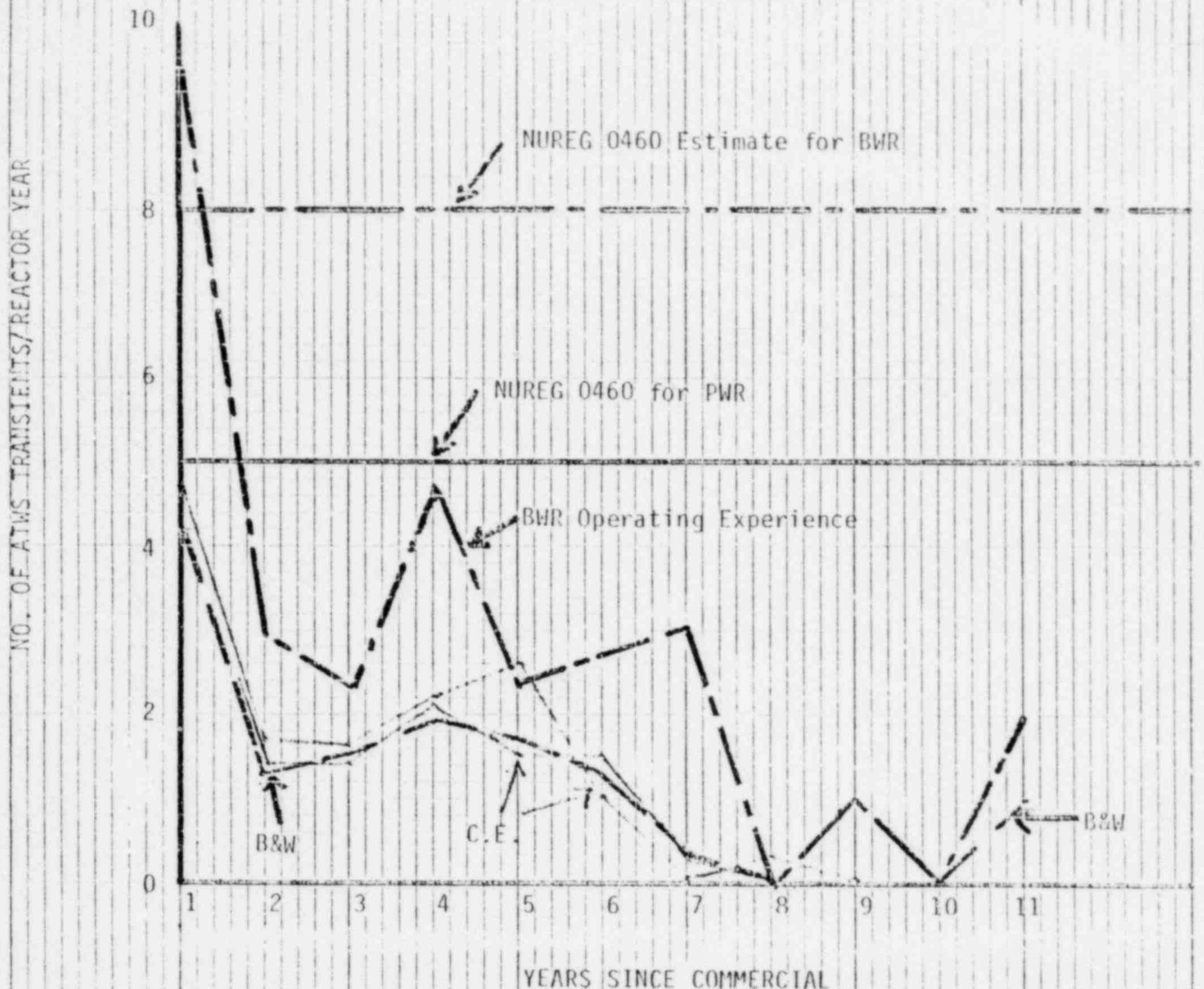
Effect of Bypass Capability on

ATWS Transient Freuqency

For Power Levels > 25 % of Rated

	Bypass Capacity	Event/Reactor Year
B & W	100%	0.5
C.E.	100%	0.7 (2560 MWt Core) 0.4 (3800 MWt Core)
W	--	-0-
G.E.	> 30%	1.22

COMPARING STAFF ESTIMATES OF AT FREQUENCIES WITH PLANT OPERATING EXPERIENCE



SUMMARY

1. Hypothesis testing indicates the 1st year of Turbine and Generator Trip transients is substantially different from subsequent years at the 95% level

2. Event frequencies are conservatively estimated for power levels > 25% of full power.

3. Events are per reactor calendar year.

4. Event frequencies relate to an average plant availability of aoubt 65%. To reach 80% the frequencies would have to be increased by 25%.

Reactor Median Transient Initiation

Frequencies Relevant for ATWS

		Events/Year
I.	Babcock & Wilcox	
	1) LOOP	0.27
	2) LOF	0.07
	3) LOL	1.11
		Sum =1.45
II.	Combustion Engineering	
	a) 2560 MWt Core	
	1) CEA	0.02
	2) PLOF	0.45
	3) LOL	1.11
	4) LOF	0.07
		Sum =1.65
	b) 3800 MWt Core	
	1) CEA	0.02
	2) PPCF	0.13
	3) LOL	1.11
	4) LOF	0.07
		Sum =1.23
III.	Westinghouse (none of significance, but those most limiting are)	
	1) LOL	1.11
	2) LOF	0.07
		Sum =1.18
IV.	General Electric	Sum =3.52

ENCLOSURE 2

RPS Failure Frequency

$$\lambda(\text{failures/year}) < \frac{\chi^2_{2(1-\alpha),\,2r+2}}{2T}$$

r = no. of failures of RPS

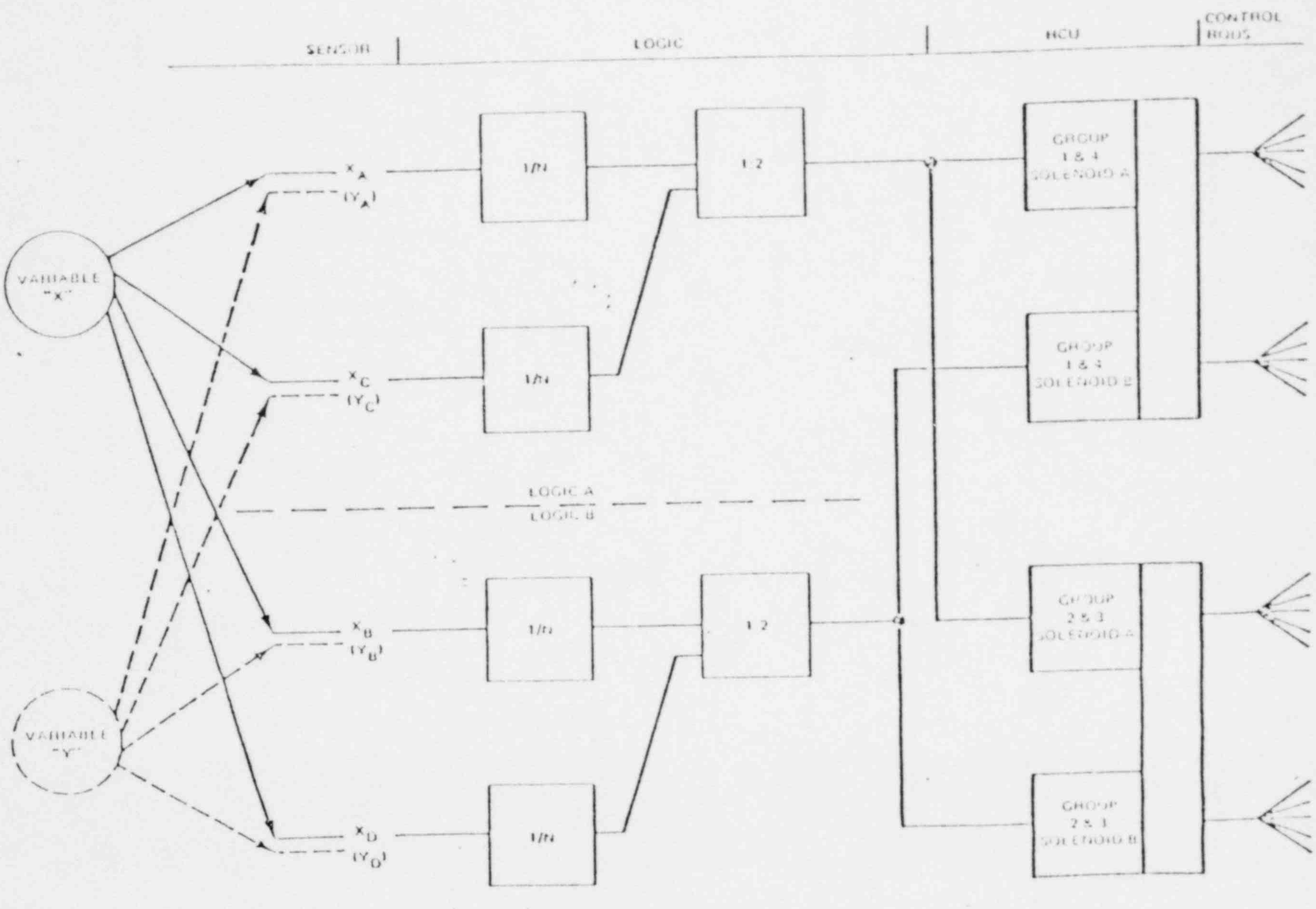
T = no. of years of reactor operation

α = confidence level

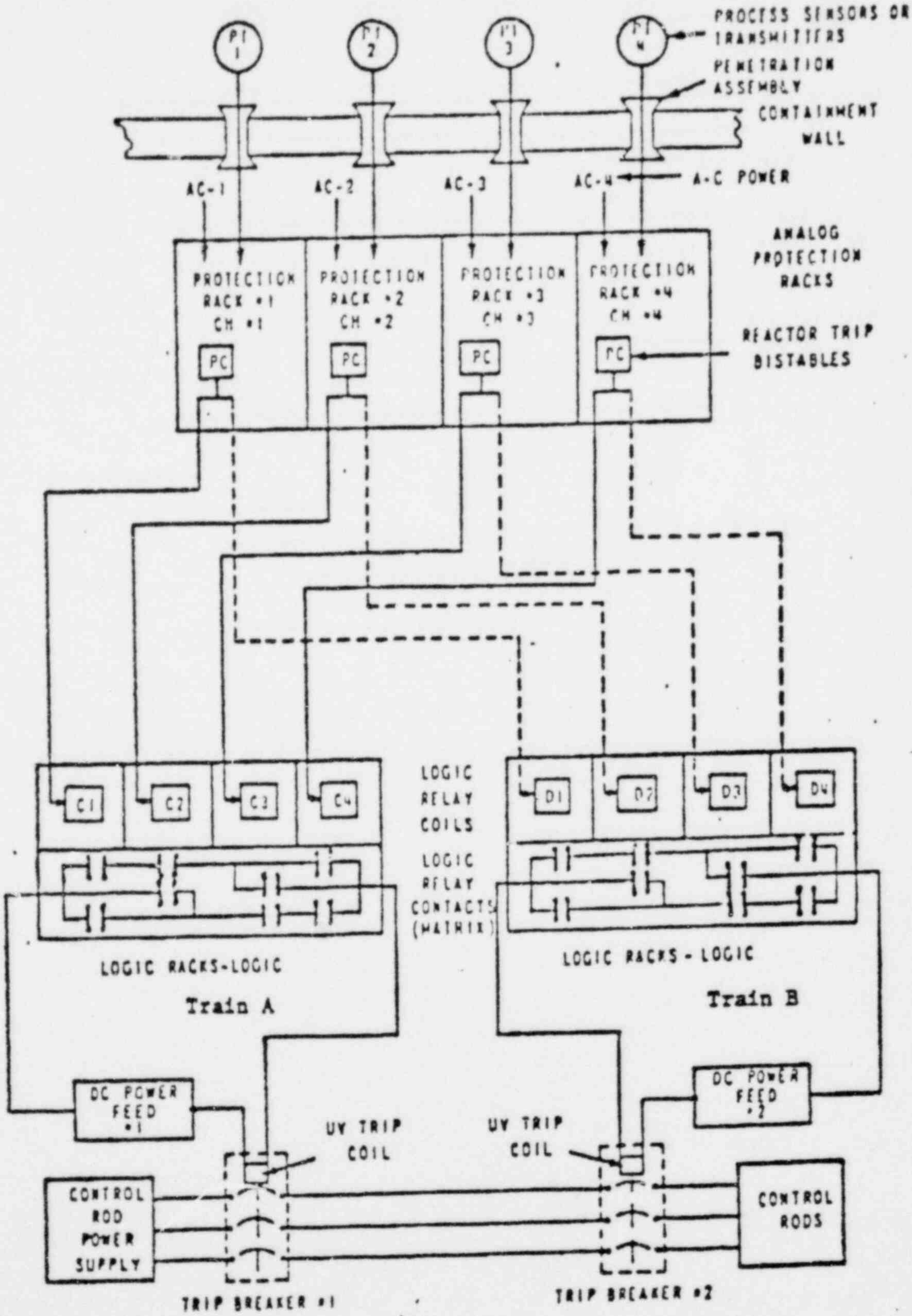
RPS Unavailability

$U = \lambda/2N$

N = No. of tests of the electrical system



RPS Relay Logic Configuration



DESIGN TO ACHIEVE ISOLATION BETWEEN CHANNELS

FIGURE 7.2-3

TESTING OF THE ELECTRICAL PORTION OF THE RPS

BWR's

Scram Signals	No. of Channels	Test Frequency
APRM Highflux	4	Weekly
High Main Steamline Radiation	?	Weekly
High Pressure in Vessel	4	30 days
High drywell pressure	4	30 days
MSIV	?	30 days
Turbine Control Valve	4	30 days
Turbine Stop Valve	?	30 days
Others		

AVERAGES ABOUT 5/week

TESTING OF THE ELECTRICAL PORTION OF THE RPS

Westinghouse (senser to Bistable)

Scram Signals	No. of Channels	Test Frequency
High Flux	4	Each 28 days
Overtemperature	4	Each 28 days
Overpower ΔT	4	Each 28 days
Low reactor Coolant flow	3/loop	Each 28 days
Low Pressurizer Pressure	4	Each 28 days
High Pressurizer Pressure	4	Each 28 days
High Pressurizer Level	3	Each 28 days
	Average $\sim$	6/week

Bistable to Actuator	6 (2/4)	Each 28 days
Breakers	2 (1/2)	Each 28 days

TESTING OF THE ELECTRICAL PORTION OF THE RPS

B & W (Sensor to Bistable)

Scram Signals	No. of Channels		Test Frequency
Power range high flux	4		Each 30 days
Pressure Temperature	4		Each 30 days
Reactor Coolant Temperature	4		Each 30 days
High reactor pressure	4		Each 30 days
Low reactor pressure	4		Each 30 days
Others		Average	6/week

Bistable to Breaker 4 (2/4)

TESTING OF THE ELECTRICAL PORTION OF THE RPS

C.E. (Sensor to Bistable)

Scram Signals	No. of Channels	Test Frequency
High flux	4	Each 30 days
R.C. Flow	4	Each 30 days
Low pressurizer pressure	4	Each 30 days
High pressurizer Pressure	4	Each 30 days
Steam Generator Level	4	Each 30 days
Steam Generator Pressure	4	Each 30 days
Others		

Averages ~ 6/week

Logic 40

Logic trip relays 24 (includes breakers in pairs) each 30 days

Trip Breakers (in pairs, any 1/2 any 2/4) 8 each 30 days

TRIP LEVELS REACHED DURING W ATWS TRANSIENTS

Transient	RPS Trip Due To
Loss of Load	Turbine trip High Pressurizer Pressure Over temerpature ΔT
Loss of Feedwater	Turbine Trip Over temperature ΔT High Pressurizer Pressure
Loss of Offsite Power	Undervoltage Underfrquency Over temperature ΔT Over power ΔT Others
Rod Withdrawal	High Flux Over temperature ΔT Over power ΔT Pressurizer high level

SUMMARY OF TESTING RATES
FOR EACH REACTOR

BWR's	
Depending on Transient	100-200/year
PWR's	
Sensors to Bistable	
Depending on transient	100-200/year
Bistable to Actuator	
W	78 /year
B & W	48/year
C.E.	480/year
Breakers	
W	24/year
B & W	48/year
C. E. (Direct test)	96/year
C. E. (with Logic Trip Relays)	288/years

CALCULATION OF FAILURE RATE PER YEAR

Based on 900 years of LWR experience

	50%	95%
With KAHL	1.9×10^{-3}	5.3×10^{-3}
Without KAHL	7.7×10^{-4}	3.3×10^{-3}

Unavailability Per Demand

Based on 100 channel test/year

	50%	95%
With KAHL	9.5×10^{-6}	2.7×10^{-5}
Without KAHL	3.8×10^{-6}	1.7×10^{-5}

ASSUMMING BREAKERS DOMINATE SCRAM FAILURE FOR PWR's

No. of Breaker Failures $\approx$ 20

Reactor Years of Experience $\approx$ 300

		50%	95%
Failure Rate/year		6.7×10^{-2}	9.7×10^{-2}
Single Breaker Unavailability/Demand			
24 tests/year		1.5×10^{-3}	2×10^{-3}
48 tests/year		7.5×10^{-4}	1×10^{-3}
Unavailability of all Breakers/Demand			
24 tests/year	1/2	2.2×10^{-6}	3.9×10^{-6}
	2/4	$<< 10^{-6}$	$<< 10^{-6}$
48 tests/year	1/2	5.6×10^{-7}	1.1×10^{-6}
	2/4	$<< 10^{-6}$	$<< 10^{-6}$

Conclusion is that Breakers do not dominate RPS unavailability.