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PROPOSED MINUTES FOR THE ACRS SUBCOMMITTEE MEETING ON RELIABILITY AND PROBABILISTIC ASSESSMENT

FEBRUARY 9, 1983 - WASHINGTON, D.C.

Purpose: The purpose of the meeting was to review the methodology of NUREG/CR-2497 Precursors to Potential Severe Core Damage Accidents: A Status Report. The meeting was one of two subcommittee meetings to discuss the report.

Principal Attendees of the meeting were:

ACRS	NRC	OTHER
D. Okrent, Chairman	M. Ernst, RES	W. Cottrell, ORNL
M. Bender, Member	F. Manning, RES	J. Minarick, SAI
J. Ebersole, Member	T. Spiess, NRR	
W. Kerr, Member	A. Thadani, RES	
C. Siess, Member		
W. Lipinski, Consultant		
C. Mueller, Consultant		
S. Guarro, Fellow		
R. Savio, Staff*		
L. Wainer, Staff		

A complete set of attendees and a copy of the meeting agenda are attached to the office copy of the minutes in the ACRS files.

Discussion

M. Ernst and F. Manning of the RES Staff described the background of the Precursor Program and the status of RES action on the Status Report NUREG/CR-2497. The program is performed by ORNL and SAI under a contract to the NRC, and uses

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operational data (LERs) from an extended time period to assess nuclear power plant risk. F. Manning said that the report on the evaluation of LER for the period 1969 through 1979 NUREG/CR-2497 had generated controversy over its methodology and in view of the controversy, the NRC had asked that the methodology be peer reviewed by the American Statistical Association, INPO, ACRS, and other peer review groups. An EPRI workshop is planned to form a consensus about the precursor methodology. M. Ernst also mentioned that RES had not formed an opinion on the methodology and indicated that the Staff would be interested in ACRS comments on the methodology.

F. Manning stated that they plan to present the status of the precursor program to date, in particular the 1980 - 1981 period study and information obtained at the February 1983 EPRI workshop at the March 9, 1983 Subcommittee meeting.

W. Cottrell of ORNL summarized the precursor program. The program was initiated after the Lewis committee recommendation that real data be used to obtain a better estimate of the probability of severe core damage. The intent of the program is to use LERs to identify potential accident precursors and estimate an overall severe core damage frequency for U.S. reactors. The program is complimentary the plant PRAs currently performed by Industry and the NRC.

Other work in the precursor program than the 1969-1979 LER Precursor study include documentation studies on diesel generator failures at Crystal River, electrical power systems at Big Rock Point, and LERs concerned with severe cooldown transients: PTS events and precursors.

He explained that the report NUREG/CR-2497 has been reviewed extensively by the NRC, INPO, NCAC, ORNL and SAI in its preliminary, draft, draft and final form. He also noted that comments about the report also were made by Duke Power and Dairyland Power on the Oconee and LaCrosse LERs. Currently, the report is undergoing review by the NRC, a peer review panel assembled by ORNL, and SAI, and NSAC.

He described the future plans for the precursor program. A draft of the 1980-1981 LER study to provided by July 1, 1983. He noted that the failure estimates for the 1980-1981 LER study would be independent of the 1969-1979 study. Computer models of event trees for sensitivity analyses and other modifications as generated from comments from peer review will be incorporated into the methodology of the precursor program. SAI and ORNL plan to complete a top-down logic for the precursor program by September 1983. SAI/ORNL also plan to obtain and incorporate more plant specific data into the study.

J. Minarick discussed the methodology and the major conclusions of NUREG/CR-2497. The study involved the identification of potential precursors to severe core damage, the identification of significant precursors, a calculation of significant precursors, a calculation of an overall severe core damage from precursor LERs, and the examination of precursors and their associated characteristics for trends.

He mentioned that the identification of precursors involved the selection from 19,400 LERs abstracts generated in the 1969-1979 period of 529 LERs for further detailed review, based on the criteria in Attachment A. The detailed review process of the 529 LERs for potential precursors was based on the criteria that the precursor (1) resulted in a failure of a function required to mitigate an off normal event or accident, (2) resulted in degradation of $\geq$ 20 functions, or (3) included unusual initiating events, such as, loss of offsite power. The detailed review process identified 169 potential precursors.

He next described the quantification of the 169 precursors and the identification of significant precursors. The 169 precursors were mapped onto event trees, categorized and documented, to provide a database for the calculation of failure probabilities and initiating event frequencies.

The failure probabilities and initiating event frequencies were developed in the following fashion:

1) The number of failures associated with a function or system was derived by identifying the events within the 169 database in which the failure occurred, and then assigning a severe (by rating the failure to rectify in the short term probabilities) to each event. The severity ratings for a function were then summed and then divided by the demand estimate derived from precursor data, to generate a failure probability.

2) In cases where initiating events and failure probabilities could not be estimated from the precursor, other estimate, such as, WASH-1400 were sequence of events trees were developed for each precursor event, using failure probabilities and iniating frequencies. The probability of severe core damage for each precursor was estimated. 52 precursor events were determined to be significant (SCD frequency $\geq 1x10^{-3}$).

He next discussed the trends analyses conducted in the study. The precursor events examined in the trends analyses were those significant event found at plants that went critical after January 1, 1969. The trends analyses included the comparison of initiating events frequencies and failure probabilities estimated in the study with those in WASH-1400. The analyses indicated that the precursor studies estimates were within a factor of 10 of the WASH-1400 estimates. He noted that SAI and ORNL considered the study estimates to be consistent with WASH 1400 estimates, except for the failure probabilities of the auxiliary feedwater system, PWR long term cooling and BWR ADS.

In analysing the selected precursors to see if the number of significant precursors varied with plant age, J. Minarick noted that SAI/ORNL could not conclude from the data that a there was decrease or an increase in the number of precursors with plant age. Also, the study failed to find any difference in the number of precursors with plant type, vendors, A/Es, or plant power rating.

SAI and ORNL. In the report also examined the number of precursors associated with human error. (Operator mistakes or maintenance errors which resulted in the failure of a system). The analyses indicated that 38% of the significant precursors appeared to involve human error, 26% of all precursor identified involved human error, and 29% of LERs reviewed involved human error.

The study also estimated the diesel generator failure probability during LOOP, using precursor information. The estimated probability was 0.064.

Total time on test plots trends were included in the trends analyses to determine the variation of instantaneous failure rates as a function of plant age. The failure rate in most initiating events or demand failures were either constant or decreasing except for the BWR ADS failure.

J. Minarick next described the calculation of the overall severe core damage frequency for the 1969-1979 period. The frequency was calculated using an approach similar to Apostalokis and Mosleh. The severe core damage frequencies contribution for 52 significant precursor events were summed and normalized over the total years of operating reactors in the study (432 years). The total severe core damage frequency was estimated to be 4.5×10^{-3}/reactor year. He noted that, further work on the estimate indicated that the severe core damage frequency could be a factor of 3 lower than what was originally estimated. This was because the original estimate assumed all the failures in the event trees to be independent rather than accounted for dependencies between the failures.

W. Cottrell next discussed the problem and issues derived from comments reviewed during the pre publication review of NUREG/CR-2497. He mentioned that a few of the misconceptions associated with the study were that the overall core damage probability was percieved as a future core damage probability, the study should have estimated the changes made in responses to the TMI-2 event, and drafts of the report were not peer reviewed. He also noted that the study did not include failures which occurred during shutdown which could not have occurred during plant operation, and failure immediately found by required testing after maintenance.

He explained that the precursor study defined severe core damage as the logical consequence of event trees. He said that the study made no effort to assess the situation beyond core damage.

He described the problems associated with the selection criteria. He indicated that inadequacies in the LER reporting system, lack of details in the LER abstracts, and inherent biases in the selection process were among the problems encountered in the study. He noted that in a 10% resample of the 19,400 LER database, SAI and ORNL identified four more potential precursors, and subsequently they found another potential precursor - a core uncovery event at LaCrosse. He said that in light of the 10% resample of the database and on comments received ORNL/SAI feel that there are more potential precursor to be identified in additional checking of the database.

In regards to the calculation of the demand failure probabilities and initiating eent frequencies, W. Cottrell indicated that the study's estimate were different from the estimates obtained in INPO's reassessment of the study. He said that the difference between INPOs and the precursor's estimate were attributed to the differences in the rectifiction assignments to each failure. He noted that the study's rectification factor of 1 became INPO's factor of 0.1, 0.01, or 0.5, and that the differences between INPOs and the precursor studies assignments were due to the differences in engineering judgement credit given to operators, and factors associated with the ascessibility and repairability of equipment.

W. Kerr asked W. Cottrell as to what he felt was the significance of the precursor study. W. Cottrell replied that the precursor study provides a measure on the meaningfulness of NRC and industry PRA estimates that are used in relation to the proposed safety goals. He also stated that the study indicated the probability of core damage in the 1969-1979 period was higher than what is expected at present.

A. Buslik of the NRR Staff discussed NRR's observations on the methodology of the precursor study. He noted that the general observations were presented in memo's dated December 16, 1982 and August 24, 1982 from H. Denton to R. Bernero and R. Minogue. He explained that the problem of overcounting in the precursor study could be eliminated by identifying events as either complimentary or precursor. He also indicated that the precursor study's approach to the weighting of 1 to the TMI-2 event produced a biased estimate of the SCD frequency.

He noted that if the approach he suggested was used to calculate a probability for the TMI-2 event, then the study's estimate of SCD would have not been biased.

Another observation he discussed were that the precursor study could assign low conditional SCD Probabilities to events of high potential safety significance; specifically he noted that the study assigned a low probability to the Davis Besse high pressure injection runback event which was similar to the TMI-2 event. J. Minarick in response to the observation, said that the precursor study failed to identify the Davis Besse event as significant because the LER provided inadequate information or the HPI runback.

On the systems availability calculations, A. Buslik mentioned that the study was arbitrary in its approach to degraded systems, and that system recovery should have incorporated more details in its approach. He also indicated that the concept of minimum and maximum unavailability was not precise.

He mentioned that the benefits from rectification were not fully accounted for by the precursor study. He gave as examples the failure of the precursor study to identify the Davis Besse event as a near "miss" of the TMI-2 event, and the loss of non nuclear instrumentation at Crystal River to be a near miss of the Rancho Seco event.

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NOTE: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 1717-H Street, N.W., Washington, D.C., or can be purchased from Alderson Reporting Company, Inc., 400 Virginia Avenue, S.W., Washington, D.C. 20024, 202/554-2345.

ACCIDENT SEQUENCE PRECURSOR

PROGRAM METHODOLOGY

- LER SYSTEM INCLUDES MANY REPORTS OF INDIVIDUAL COMPONENT INOPERABILITIES WHICH RESULTED IN LOSS OF SAFETY SYSTEM REDUNDANCY, BUT WHICH DID NOT PREVENT THE SAFETY SYSTEM FROM PERFORMING ITS FUNCTION.

- LER SYSTEM ALSO INCLUDES OCCASIONAL REPORTS OF TOTAL LOSSES OF SAFETY SYSTEM OPERABILITY OR LOSSES OF REDUNDANCY IN MORE THAN ONE SYSTEM.

- LER SYSTEM ALSO INCLUDES OCCASIONAL REPORTS OF INITIATING EVENTS WHICH REQUIRED THE USE OF PLANT SAFETY SYSTEMS TO MITIGATE THE CONSEQUENCES OF THE INITIATING EVENT.

- APPROXIMATELY 19,400 LERs (OR THEIR EARLIER EQUIVALENTS) ARE ON FILE FOR THE 1969 - 1979 TIME PERIOD.
- SINCE IT WAS NOT PRACTICAL TO REVIEW EACH LER IN DETAIL, CRITERIA WERE DEVELOPED TO SELECT FROM THE LER DATA BASE THOSE CONSIDERED WORTHY OF SUBSEQUENT DETAILED REVIEW FOR PRECURSORS.

CRITERIA FOR SELECTION OF LERs FOR DETAILED REVIEW AS PRECURSORS

1. ANY FAILURE TO FUNCTION OF A SYSTEM THAT SHOULD HAVE FUNCTIONED AS A CONSEQUENCE OF AN OFF-NORMAL EVENT OR ACCIDENT,
2. ANY INSTANCE WHERE TWO OR MORE FAILURES OCCURRED,
3. ALL EVENTS THAT RESULTED IN OR REQUIRED INITIATION OF SAFETY-RELATED EQUIPMENT (EXCEPT EVENTS THAT ONLY REQUIRED TRIP AND WHEN TRIP WAS SUCCESSFUL),
4. ALL COMPLETE LOSSES OF OFFSITE POWER AND ANY LESS FREQUENT OFF-NORMAL INITIATING EVENTS OR ACCIDENTS,
5. ANY EVENT OR OPERATING CONDITION THAT WAS NOT ENVELOPED BY OR PROCEEDED DIFFERENTLY FROM THE PLANT DESIGN BASES, AND
6. ANY OTHER EVENT THAT, BASED ON THE REVIEWER'S EXPERIENCE, COULD HAVE RESULTED IN OR SIGNIFICANTLY AFFECTED A CHAIN OF EVENTS LEADING TO POTENTIAL SEVERE CORE DAMAGE.

- INITIAL READING OF LER ABSTRACTS TO CHOOSE THOSE LERs WHICH DESERVE DETAILED REVIEWS FOR POTENTIAL PRECURSORS.

- DETAILED REVIEW OF SELECTED LERs CONSIDERING:

  - SPECIFICS OF ACTUAL EVENT

  - IMPACT OF EVENT ON REACTOR PLANT SYSTEMS AT THE PLANT AT WHICH THE EVENT OCCURRED.

  - NEED FOR SPECIFIC SYSTEMS/PORTIONS OF SYSTEMS IN THE PLANT AT WHICH THE EVENT OCCURRED FOR MITIGATING OFF-NORMAL EVENTS AND ACCIDENTS.

- SELECTION OF EVENTS AS POTENTIAL PRECURSORS IF THEY:

  - RESULTED IN THE FAILURE OF A FUNCTION REQUIRED TO MITIGATE AN OFF-NORMAL EVENT OR ACCIDENT.

  - RESULTED IN THE DEGRADATION OF TWO OR MORE FUNCTIONS.

  - INCLUDED UNUSUAL INITIATING EVENTS (LOOPS AND LESS FREQUENT EVENTS).

- A 10% SAMPLE OF THE LER ABSTRACT DATA BASE WAS REVIEWED TO IDENTIFY ANY EVENTS WHICH SHOULD HAVE BEEN SELECTED AS POTENTIAL PRECURSORS BUT WHICH WERE PREVIOUSLY MISSED.
- FOUR ADDITIONAL EVENTS WERE FOUND. THIS RESULTS IN AN ESTIMATE OF 83% OF ALL POTENTIAL PRECURSORS FOUND.

- 169 EVENTS WERE SELECTED AS PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE.

- THESE WERE DOCUMENTED, CATEGORIZED, AND "MAPPED" ONTO EVENT TREES WHICH DESCRIBED THE SEQUENCE OF ACTIONS REQUIRED TO MITIGATE A TRANSIENT OR ACCIDENT. THE EVENT TREE CHOSEN FOR EACH PRECURSOR WAS BASED ON THE MOST LIKELY INITIATING EVENT OR TRANSIENT WHICH COULD HAVE BEEN AFFECTED BY THE REPORTED FAILURES.

- THE EVENT TREES CONSIDERED THE INITIATING EVENT AND OPERABILITY OF FUNCTIONS REQUIRED TO MITIGATE THE INITIATING EVENT. UNAVAILABILITY OF REQUIRED MITIGATING SYSTEMS IN COMBINATION WITH AN INITIATING EVENT WAS CONSIDERED TO RESULT IN SEVERE CORE DAMAGE.

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: **78418**

Date: **February 2, 1973**

Title: **Failure of Charging Pump Valves to Open After an Ice Storm at Maine Yankee**

The failure sequence was:

1. During ECCS valve surveillance testing, the suction valve from the RWST to charging pump 14A failed to open due to ice on the valve stem, a result of an ice storm earlier in the day.
2. The suction valve from the RWST to charging pump 14B failed to open completely, also because of ice on the valve stem.

Corrective action:

The ice was removed from the valve stems and protective covers fabricated for the valves.

Design purpose of failed system or component:

The charging pump suction valves admit water from the RWST to the charging pumps during safeguards actuation, when pump suction flow is automatically switched from the volume control tank to the RWST.

Unavailability of system per WASH 1400:* High Pressure Injection System (Two of Three Pumps Required on Surry): 1.2×10^{-2}

Unavailability of component per WASH 1400:* Motor Operated Valve, failure to operate: 1×10^{-3}/D

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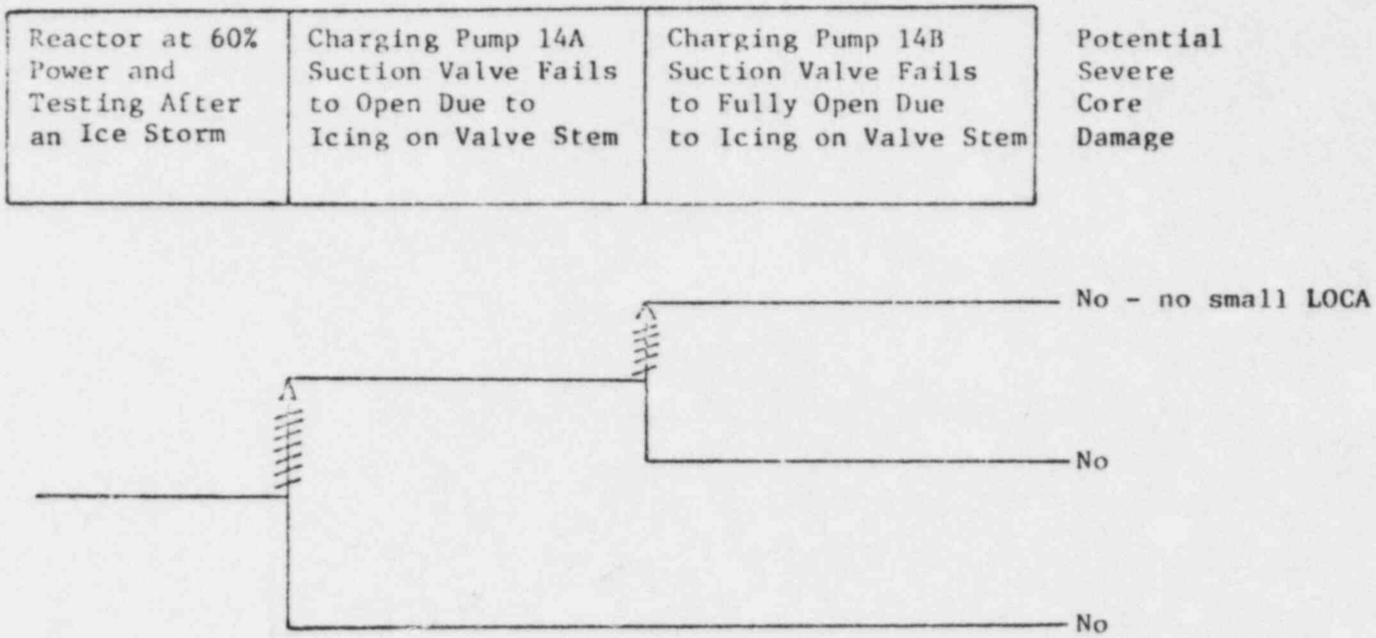
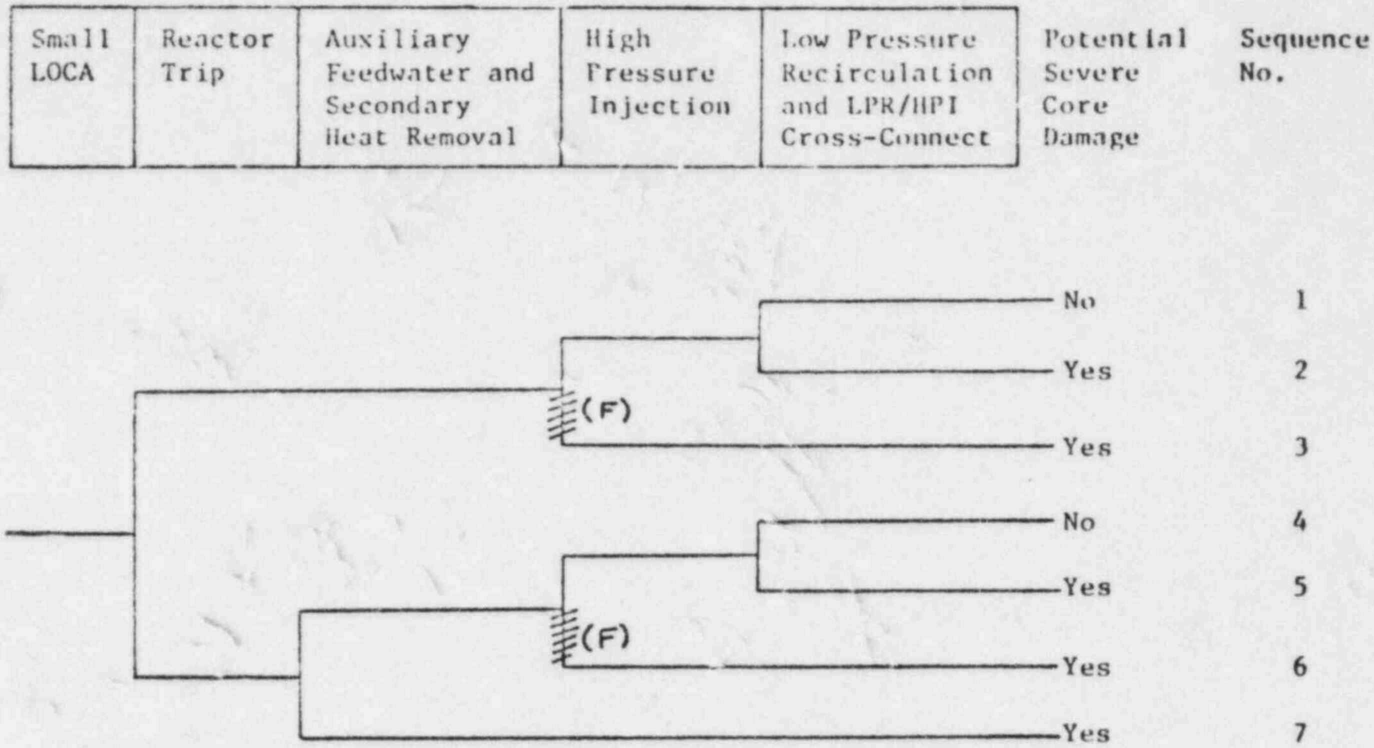
*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.



NSIC 48418 — Actual Occurrence for Failure of Charging Pump Valves to Open After an Ice Storm at Maine Yankee



NSIC 78418 — Sequence of Interest for Failure of Charging Pump Valves to Open After an Ice Storm at Maine Yankee

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 78418

DATE OF LER: February 7, 1973

DATE OF EVENT: February 2, 1973

SYSTEM INVOLVED: High Pressure Injection System

COMPONENT INVOLVED: Charging pump suction valves

CAUSE: Ice on valve stems

SEQUENCE OF INTEREST: small LOCA

ACTUAL OCCURRENCE: valve failure to open during testing

REACTOR NAME: Maine Yankee

DOCKET NUMBER: 50-309

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 825 MWe

REACTOR AGE: .3 yr

VENDOR: Combustion Engineering

ARCHITECT-ENGINEERS: Stone and Webster

OPERATORS: Maine Yankee Atomic Power Co.

LOCATION: 10 miles north of Bath, Maine

DURATION: 360(a) hours

PLANT OPERATING CONDITION: at 60% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; [✓ (c)] made inoperable; (d) ____________________

DISCOVERY METHOD: during testing

COMMENT: —

- INITIATING EVENT FREQUENCIES AND BRANCH FAILURE PROBABILITIES WERE DEVELOPED WHERE POSSIBLE FROM THE INFORMATION CONTAINED IN THE PRECURSORS. THIS WAS DONE TO:
  - GAIN AN UNDERSTANDING OF WHAT FAILURE VALUES WERE INDICATED BY THE LER DATA BASE.
  - PROVIDE FAILURE VALUES FOR SUBSEQUENT RANKING OF THE PRECURSORS.

- THE NUMBER OF FAILURES WAS ESTIMATED BY ASSIGNING SEVERITY RATINGS (FAILURE-TO-RECTIFY-IN-THE-SHORT-TERM PROBABILITIES) TO EACH SYSTEM FAILURE:
  - IF THE FAILURE WAS CONSIDERED EASILY RECTIFIABLE IN THE SHORT TERM FROM THE CONTROL ROOM, A SEVERITY RATING OF 0.1 WAS ASSIGNED.
  - IF THE FAILURE WAS CONSIDERED RECTIFIABLE IN THE SHORT TERM AT THE COMPONENT, A SEVERITY RATING OF 0.5 WAS ASSIGNED.
  - IF THE FAILURE WAS NOT CONSIDERED RECTIFIABLE IN THE SHORT TERM, A SEVERITY RATING OF 1.0 WAS ASSIGNED.
- THE NUMBER OF DEMANDS WAS ESTIMATED BASED ON TESTING REQUIREMENTS AND ACTUAL TRANSIENT DEMANDS.

Table C.1. Initiating event frequency and function failure probability estimates

Initiating event/function under consideration	Event description						Total number of events × severity	Total number of events an. potential events × severity	Observation period or demands assumptions	Initiating event frequency estimate (per year)	Demand failure probability estimate (per demand)		Value used in ASP calculations
	NSIC accession number	Event date	Plant	Plant age (d)	Description	Severity					Minimum function	Maximum function	
	PWR loss of main feedwater												
Initiating event	Initiating event frequency for a PWR LOFW was determined based on a review of main-feedwater-related events reported in NUREG-0611 (pp. 11-15 through 11-27). Eighty LOFW initiating events occurred over a three-calendar-year period. Approximately 50% appeared to be rectifiable in the short term. Westinghouse LOFW events were used for this estimate because the majority of PWR operating hours are contributed by Westinghouse plants.						40		72 post-1960 plant operating years	5.6 × 10^{-1}			5.6 × 10^{-1}/year
Reactor trip failure	Demand failure probability for failure to trip was assumed to be equal to value calculated in WASH-1400 (p. II-97).												3.6 × 10^{-5}/D
AFW and secondary heat removal failure given trip success	81523	June 10, 1973	Turkey Point 4	7	Failure of pumps to auto-start due to failure to install fuses	0.1	6.1	6.1	12 per plant year due to testing plus 1 per shutdown of <48 h plus 2 per shutdown >48 h. Based on 1979 operational data from NUREG/CR-1496, an average of 7.9 outages of <48 h and 3.7 outages of >48 h occurred per plant. This results in 3624 expected demands.		1.1 × 10^{-3}	1.1 × 10^{-3}	1.1 × 10^{-3}/D
	93421	Apr. 7, 1974	Point Beach 1	1252	Failure to deliver flow due to clogged suction strainers	0							
	91676	May 8, 1974	Turkey Point 3	555	Failure of pumps to start due to overtightened packings and controller malfunction	1.0							
	108078	Nov. 5, 1975	Kewaunee	603	Failure to deliver flow due to clogged suction strainers	1.0							
	133706	Dec. 11, 1977	Davis-Besse 1	121	Loss of AFW pump control due to mechanical binding and blown control power fuses	1.0							
	137305	Mar. 25, 1978	Farley	228	Failure of turbine-driven pumps to start plus open bypass valves	0.5							
	138830	Mar. 20, 1978	Rancho Seco	1731	Failure of AFW to deliver flow due to NNI failure	1.0							
	153164	Mar. 28, 1979	TMI-2	365	Failure of AFW to deliver flow due to closed valves	0.5							
AFW and secondary heat removal failure given failure to trip	Demand failure probability for failure of AFW and secondary heat removal given failure to trip assumed to be 17 times failure probability calculated for AFW and secondary heat removal given trip.												1.1 × 10^{-2}/D
PORV not demanded	Probability that the PORV would not be demanded following a reactor trip with successful AFW initiation was assumed to be 0.9.												0.9/D
Failure of open PORV to close and failure of operator to detect failure and close isolation valve	Probability that the PORV would stick open once it was open was assumed equal to 0.01/D. Probability of the operator failing to isolate the open valve was based on precursor information: 7 events involving a failed-open PORV with 2 of these events involving a failure to isolate the valve. (These events are detailed in the discussion of small LOCA initiating events herein.) This results in a combined failure probability of 0.0029.												2.9 × 10^{-3}/D
Failure of HPI given failure of PORV or PORV isolation valve to close	78418	Feb. 2, 1973	Maine Yankee	102	Failure of charging pump valves to open due to ice accumulation	1.0	2.0	5.0	12 per plant year due to testing, which result in 2472 total demands.		8.1 × 10^{-4}	2.0 × 10^{-3}	1.3 × 10^{-3}/D
	125106	May 6, 1977	Salem 1	146	Failure to clear pumps from overpressure protection during plant heatup (potential failure on demand)	1.0							

FUNCTION FAILURE PROBABILITY ESTIMATES
(EXAMPLE FROM TABLE C.1)
AFW AND SECONDARY HEAT REMOVAL FAILURE

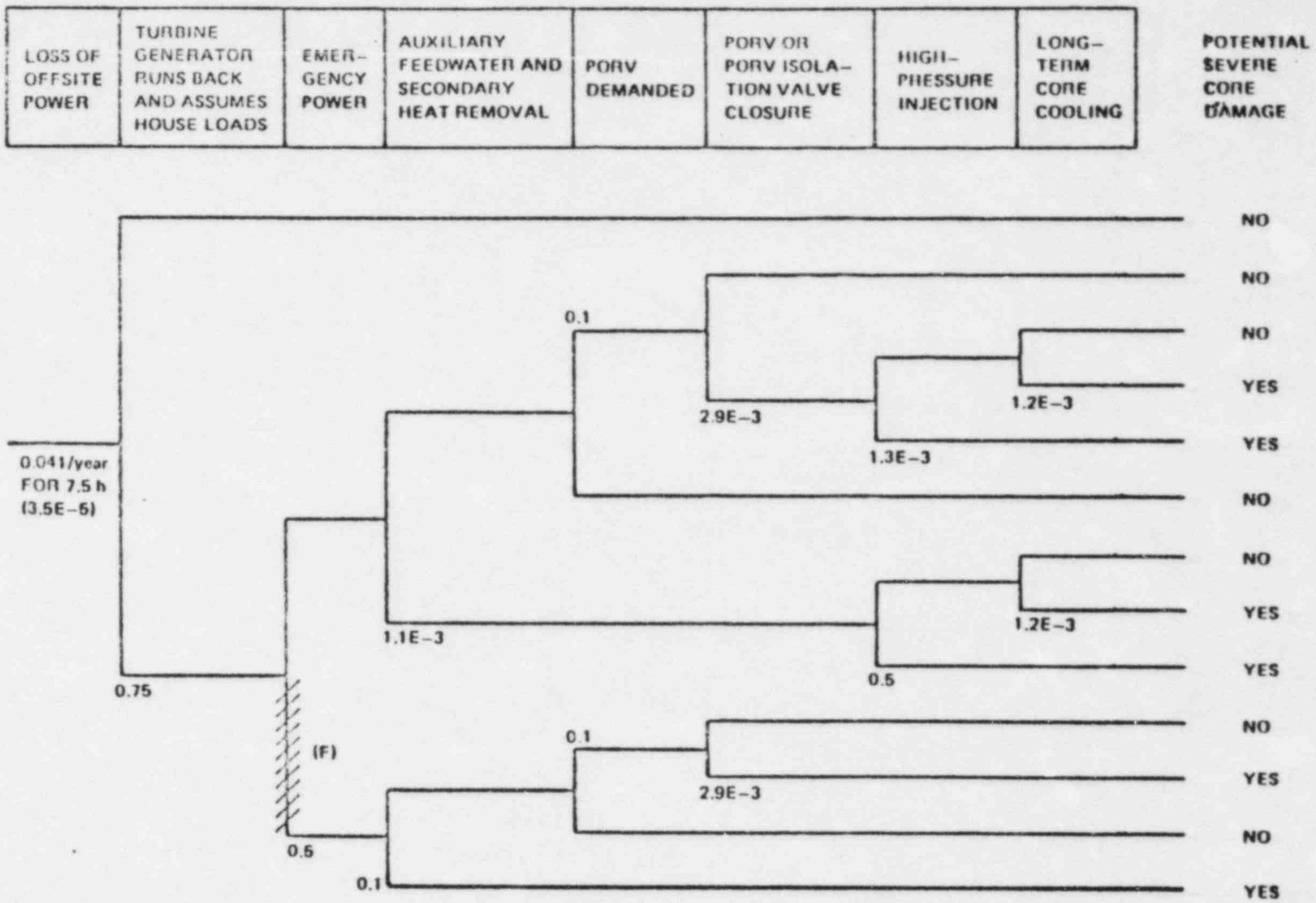
EVENTS	SEVERITY	SEVERITY SUM	DEMAND ASSUMPTIONS	FAILURE PROBABILITY
TP4-18JUNE73	0.1	6.1	5624	$1.1 \cdot 10^{-3}$/D
PB1-4APR74	1.0			
TP3-8MAY74	1.0			
KEW-5NOV75	1.0			
DB1-11DEC77	1.0			
FAR-25MAR78	0.5			
RS-20MAR78	1.0			
TMI2-28MAR79	0.5			

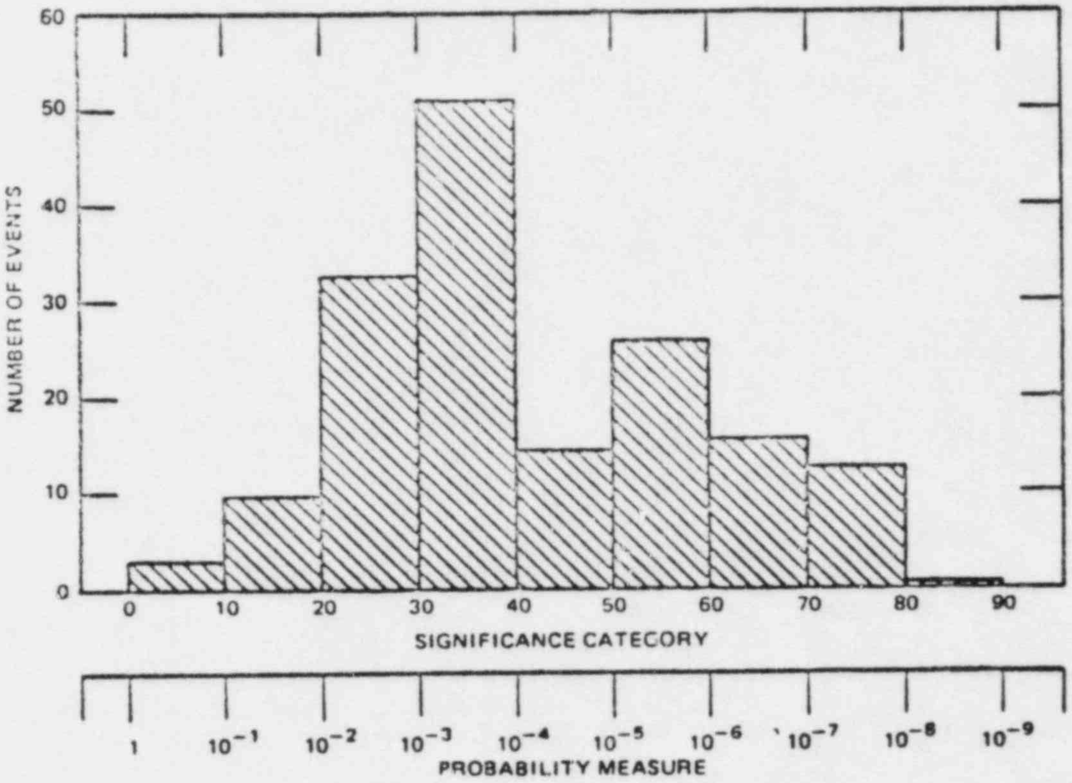
INITIATING EVENT FREQUENCIES AND DEMAND FAILURE PROBABILITIES DETERMINED USING PRECURSOR INFORMATION

EVENT	FREQUENCY OR FAILURE PROBABILITY
COMBINED PWR AND BWR LOSS OF OFFSITE POWER (≥30 MIN), PER YEAR	0.041
PWR LOSS OF OFFSITE POWER (≥30 MIN), PER YEAR	0.048
BWR LOSS OF OFFSITE POWER (≥30 MIN), PER YEAR	0.030
PWR SMALL LOCA, PER YEAR	8.3×10^{-3}
BWR SMALL LOCA, PER YEAR	2.1×10^{-2}
PWR AFW FAILURE, PER DEMAND	1.1×10^{-3}
PWR HPI FAILURE, PER DEMAND	1.3×10^{-3}
PWR LONG-TERM CORE COOLING (SUMP RECIRCULATION) FAILURE,PER DEMAND	1.2×10^{-3}
PWR EMERGENCY POWER FAILURE, PER DEMAND	1.8×10^{-3}
PWR STEAM GENERATOR ISOLATION FAILURE, PER DEMAND	1.2×10^{-3}
PWR HPI FOR STEAM LINE BREAK MITIGATION (CONCENTRATED BORIC ACID INJECTION) FAILURE, PER DEMAND	2.8×10^{-3}
BWR RCIC AND HPCI FAILURE, PER DEMAND	3.9×10^{-3}
BWR ADS FAILURE, PER DEMAND	2.7×10^{-2}
BWR EMERGENCY POWER FAILURE, PER DEMAND	5.0×10^{-3}
BWR HPCI FAILURE, PER DEMAND	5.7×10^{-2}
BWR REACTOR VESSEL ISOLATION FAILURE, PER DEMAND	3.0×10^{-3}

- CERTAIN INITIATING EVENT FREQUENCIES AND FAILURE PROBABILITIES COULD NOT BE ESTIMATED FROM THE PRECURSOR DATA.
- IN THESE CASES, PREVIOUS ESTIMATES (SUCH AS THOSE CONTAINED IN WASH-1400) WERE UTILIZED.

- BASED ON THE CALCULATED FAILURE PROBABILITIES AND FAILED AND DEGRADED STATES WHICH EXISTED DURING EACH PRECURSOR EVENT, THE PROBABILITY OF SUBSEQUENT SEVERE CORE DAMAGE GIVEN THE PRECURSOR CONDITIONS WAS DETERMINED USING THE SEQUENCE OF INTEREST EVENT TREES DEVELOPED FOR EACH EVENT.


LOSS OF OFFSITE POWER
TURBINE GENERATOR RUNS BACK AND ASSUMES HOUSE LOADS
EMER- GENCY POWER
AUXILIARY FEEDWATER AND SECONDARY HEAT REMOVAL
PORV DEMANDED
PORV OR PORV ISOLA- TION VALVE CLOSURE
HIGH- PRESSURE INJECTION
LONG- TERM CORE COOLING
POTENTIAL SEVERE CORE DAMAGE
0.041/year
FOR 7.5 h
(3.5E-6)
0.75
0.1
2.9E-3
1.2E-3
1.3E-3
1.1E-3
1.2E-3
0.5
(F)
0.5
0.1
2.9E-3
0.1
NO
NO
NO
YES
YES
NO
NO
YES
YES
NO
YES
NO
YES


NUMBER OF EVENTS
60
50
40
30
20
10
0
0
10
20
30
40
50
60
70
80
90
SIGNIFICANCE CATECORY
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
PROBABILITY MEASURE

- THE FIFTY-TWO EVENTS WITH A CONTRIBUTION TO SEVERE CORE DAMAGE EQUAL TO OR GREATER THAN 10^{-3} WERE SELECTED AS SIGNIFICANT PRECURSORS. FORTY-SEVEN OF THESE EVENTS OCCURRED AT PLANTS WHICH WENT CRITICAL AFTER JANUARY, 1969.

- THESE FORTY-SEVEN EVENTS WERE USED AS A BASIS FOR DETERMINING WHETHER SIGNIFICANT TRENDS WERE DISCERNIBLE IN THE PRECURSORS

TRENDS ANALYSIS

- COMPARISON OF CALCULATED INITIATING EVENT FREQUENCIES AND FUNCTION FAILURE PROBABILITIES WITH PREVIOUS ESTIMATES
- DETERMINATION OF TRENDS IN INSTANTANEOUS FAILURE RATES AS A FUNCTION OF PLANT AGE
- DEVELOPMENT OF TIME LINES TO VISUALLY INDICATE WHERE AND WHEN IN PLANT LIFE THESE EVENTS OCCURRED
- CONSIDERATION OF VARIATION IN NUMBER OF SIGNIFICANT EVENTS PER PLANT AS A FUNCTION OF PLANT AGE
- CONSIDERATION OF POTENTIAL DIFFERENCES BETWEEN PLANT TYPES AND AMONG VENDORS, A/E's, AND PLANT POWER RATINGS BASED ON THE NUMBER OF SIGNIFICANT EVENTS
- IDENTIFICATION OF DEGRADED FUNCTION EVENTS THAT OCCURRED WITHIN ONE MONTH OF EACH SIGNIFICANT PRECURSOR
- DETERMINATION OF PERCENTAGES OF PRECURSORS INVOLVING HUMAN ERROR
- ESTIMATION OF PROBABILITY OF A DIESEL GENERATOR FAILING TO START, GIVEN A NON-TESTING LOSS-OF-OFFSITE POWER DEMAND

Table 5.1. Initiating event frequencies and demand failure probabilities determined using precursor information compared with values determined in the *Reactor Safety Study*

Event	Frequency or failure probability	
	ASP value	*Reactor Safety Study* value
Loss of offsite power (combined PWR and BWR) ($\geq$30 min), per year	0.041	0.04[a]
PWR loss of offsite power ($\geq$30 min), per year	0.048	
BWR loss of offsite power ($\geq$30 min), per year	0.030	
PWR small LOCA, per year	8.3×10^{-3}	10^{-3}[b]
BWR small LOCA, per year	2.1×10^{-2}	10^{-3}[c]
PWR AFW failure, per demand	1.1×10^{-3}	3.7×10^{-5} (7×10^{-6} to 3×10^{-4})[d]
PWR HPI failure, per demand	1.3×10^{-3}	8.6×10^{-3} (4.4×10^{-3} to 2.7×10^{-2})[e]
PWR long-term core cooling (sump recirculation) failure, per demand	1.2×10^{-3}	1.3×10^{-2} (4.4×10^{-3} to 3.1×10^{-2})[f]
PWR emergency power failure, per demand	1.8×10^{-3}	1×10^{-5}[g]
PWR steam generator isolation failure, per demand	1.2×10^{-3}	
PWR HPI for steam line break mitigation (concentrated boric acid injection) failure, per demand	2.8×10^{-3}	
BWR RCIC and HPCI failure, per demand	3.9×10^{-3}	7.8×10^{-3}[h]
BWR ADS failure, per demand	2.7×10^{-2}	5×10^{-3} (3.3×10^{-3} to 7.5×10^{-3})[i]
BWR emergency power failure, per demand	5.0×10^{-3}	1×10^{-5}[j]
BWR HPCI failure, per demand	5.7×10^{-2}	9.8×10^{-2} (6.8×10^{-2} to 1.4×10^{-1})[k]
BWR reactor vessel isolation failure, per demand	3.0×10^{-3}	

[a] Ref. 1, p. I-85/86, footnote 3.

[b] Ref. 1, p. 63.

[c] Ref. 1, Sect. 5.3.4.1, p. 64.

[d] Ref. 1, Table II 5-8.

[e] Ref. 1, p. II-144.

[f] Ref. 1, p. II-176.

[g] Ref. 1, p. II-90.

[h] Ref. 1, p. 56.

[i] Ref. 1, p. II-405.

[j] Ref. 1, p. II-355.

[k] The *Reactor Safety Study* failure probabilities include a test and maintenance contribution that would not be included in numbers derived from testing. The nontest and maintenance failure probability is 1.3×10^{-2}/D (median) (Ref. 1, p. II-395).

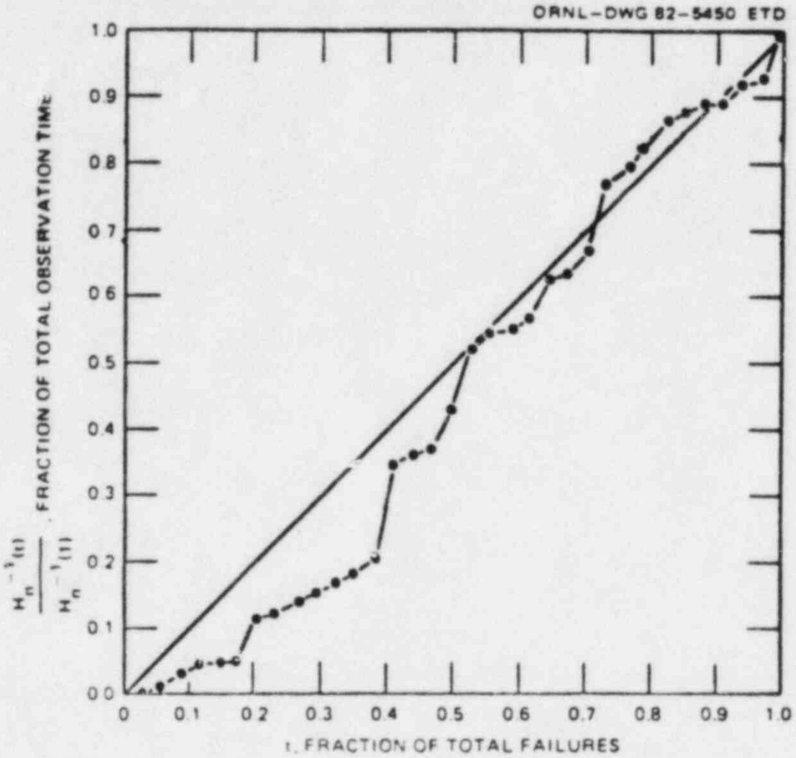


Fig. D.2. Time on test plot for PWR and BWR loss of offsite power.

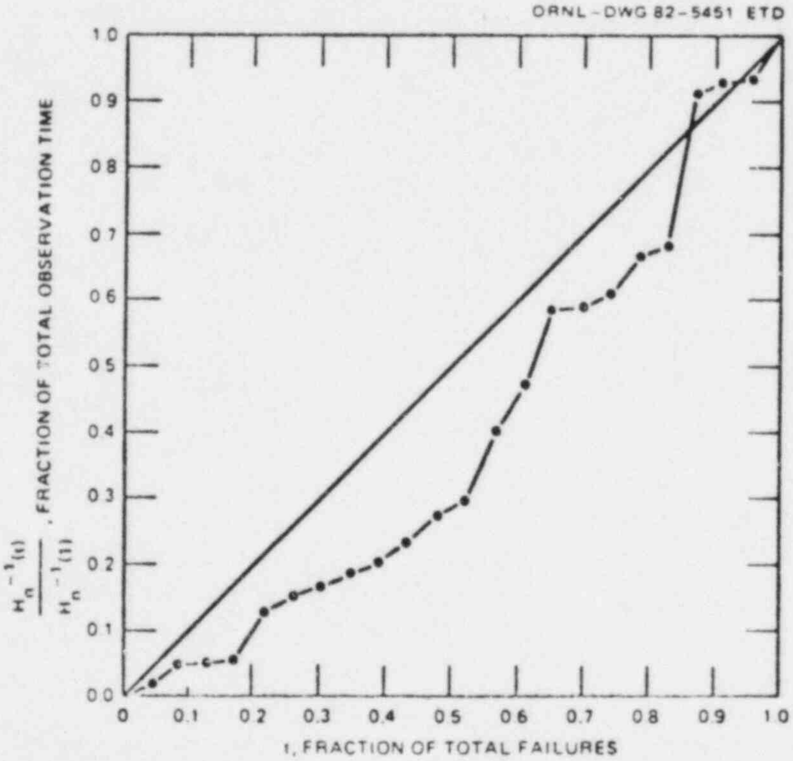


Fig. D.3. Time on test plot for PWR loss of offsite power.

Table 5.2. Total time on test plot trend indications

Initiating event or demand failure	Failure rate trend[a]
PWR and BWR loss of offsite power	Decreasing
PWR loss of offsite power	Decreasing
BWR loss of offsite power	Constant (perhaps increasing)
PWR small LOCA	Constant (perhaps decreasing)[b]
BWR small LOCA	Decreasing[b]
PWR AFW demand failure	Decreasing
PWR HPI demand failure	Decreasing[b]
PWR long-term core cooling (sump recirculation) demand failure	Constant (perhaps increasing)[b]
PWR emergency power demand failure	Decreasing
PWR steam generator isolation demand failure	Constant[b]
PWR HPI for steam line break mitigation demand failure	Decreasing
BWR HPCI and RCIC demand failure	Decreasing
BWR ADS demand failure	Increasing[b]
BWR emergency power demand failure	Constant (perhaps increasing)
BWR reactor vessel isolation demand failure	Decreasing[b]

[a] See Appendix D for cautions in interpreting these trends.

[b] This conclusion was based on a small number of observed events.

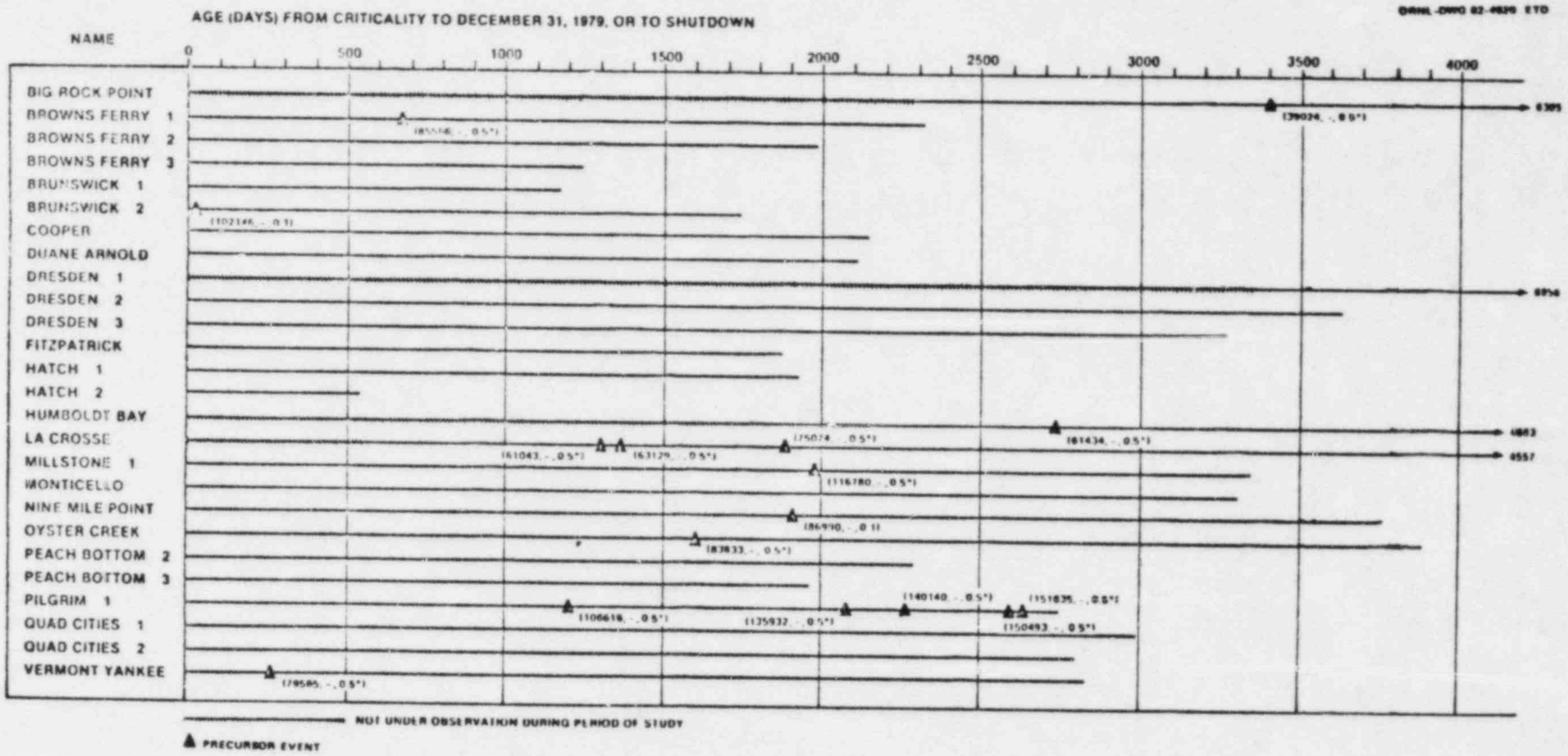


Fig. B.2. Time line for BWR loss of offsite power.

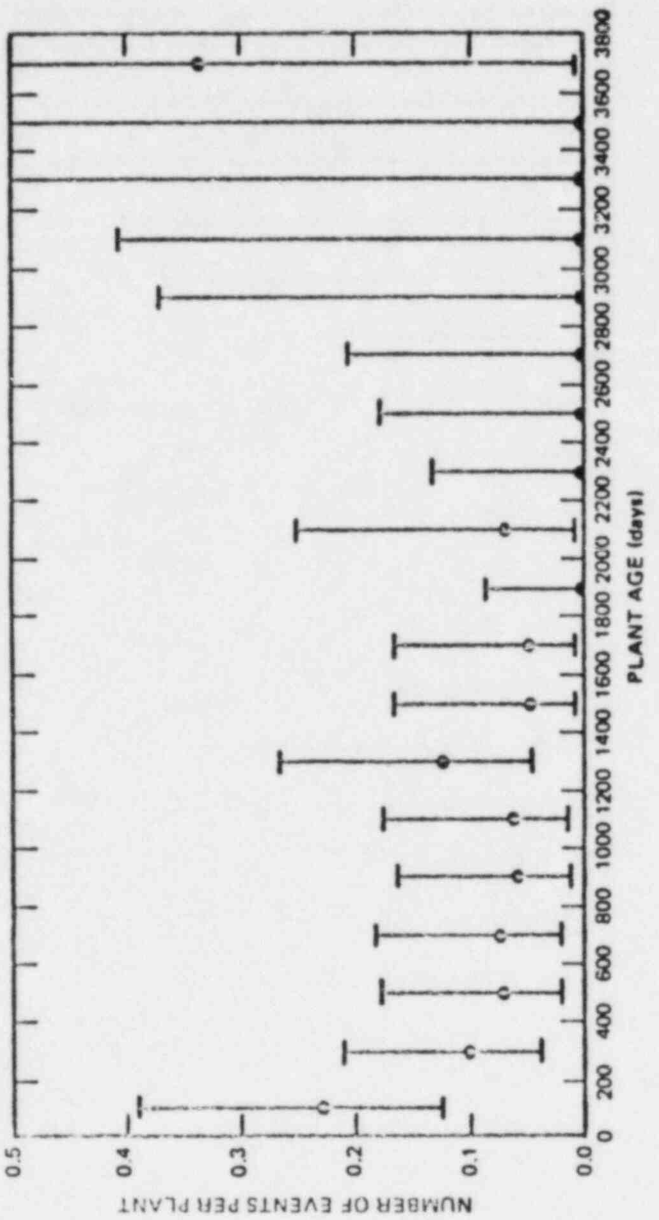

NUMBER OF EVENTS PER PLANT
0.0
0.1
0.2
0.3
0.4
0.5
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
3200
3400
3600
3800
PLANT AGE (days)

Table 5.3. Chi-square tests for differences between plant types and among reactor vendors, architect-engineers, and power levels

	Number of events		Calculated χ^2 values[a]	95% cumulative χ^2 distribution
	Observed	Expected		
Plant type				
PWR	31	27.36	1.15 (0.86)	3.84
BWR	16	19.63		
Reactor vendors				
PWR				
Babcock & Wilcox	10	6.14	3.02 (2.18)	5.99
Combustion Engineering	5	5.94		
Westinghouse	16	18.92		
All				
Babcock & Wilcox	10	5.42	4.58 (3.58)	7.81
Combustion Engineering	5	5.25		
General Electric	16	19.63		
Westinghouse	16	16.70		
Major architect-engineers				
Bechtel	16	15.63	8.90 (5.94)	11.10
Burns and Roe	5	2.58		
Ebasco	7	4.40		
Sargent and Lundy	2	6.96		
Stone and Webster	3	4.71		
United Engineers and Constructors	3	1.73		
Plant power ratings, MWe				
400–600	8	9.76	2.21 (1.15)	7.81
600–800	14	14.49		
800–1000	20	15.61		
1000–1200	5	7.14		

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[a] Values given without Yates (-0.5) correction appear first; values with Yates (-0.5) correction are in parentheses.

Table 5.4 (continued)

Significant precursor events				Additional events of interest within 1 month of significant precursor		
NSIC accession No.	Date	Plant	Description	NSIC accession No.	Date	Description
120293	Nov. 11, 1976	Hatch 1	While the plant was operating at power, all service water strainers were discovered plugged. The backflush motors had tripped. A standing order was given to backflush the strainers twice per shift until the motors could be returned to service.	120290	Nov. 2, 1976	While testing the 1C DG, an overspeed alarm began flashing erratically. The output breaker tripped automatically, and the DG was manually tripped.
				120689	Nov. 29, 1976	The RCIC inverter tripped with the reactor at 84% power. A capacitor in the output transformer had shorted.
123150	Mar. 2, 1977	Crystal River 3	Vital bus B failed because of the failure of inverter B output diode. This caused a reactor trip, turbine trip, and the 30% opening of the atmospheric dump valves. The cooldown rate was 164°F in 15 min.	123781	Mar. 7, 1977	Turbine driven AFW pump 3B tripped on overspeed when started from the control board with main steam as the supply because of a lack of governor response.
				124250	Mar. 22, 1977	Following a reactor trip, manual HPI was initiated on HPI train A. One of the two HPI isolation valves (MUV-24) failed to open because of improper torque switch setting.
128569	July 15, 1977	Brunswick 2	Following a turbine/reactor trip, a relief valve was being used to control reactor pressure. The valve failed to reset because of a ground in the solenoid operator.	120791	June 22, 1977	While HPCI was out of service for repairs and calibration, the 2A RHR pump failed to start during operability test required by Technical Specifications.
130788	Sept. 24, 1977	Davis-Besse 1	A half-trip of the SFRCS due to unknown causes initiated closure of startup feedwater valve FWSP7A. This resulted in reduced water level in SG 2 and subsequent RCS tempera- and pressure rise. The pressurizer PORV lifted 9 times and then stuck open. The valve stuck open because of rapid cycling caused by the omission of a close relay in its control circuit.	130943	Oct. 16, 1977	The governor for AFW pump 1-2 was found closed during testing. The possible cause was excessive vibration from the startup feed pump motor.

- PRECURSORS INVOLVING HUMAN ERROR

— SIGNIFICANT PRECURSORS	38%
— ALL PRECURSORS	36%
— 1979 SAFETY-RELATED LERs	29%

- EFFECTIVE NUMBER OF DIESEL GENERATOR FAILURES DURING ACTUAL LOOPS = 4.2.
- ACTUAL NUMBER OF DIESEL DEMANDS = 66
- DIESEL GENERATOR FAILURE PROBABILITY ESTIMATE = $\frac{4.2}{66} \approx .064$

- THE PROBABILITIES OF SUBSEQUENT SEVERE AND CORE DAMAGE DETERMINED FOR PRECURSORS ASSOCIATED WITH INITIATING EVENTS WERE USED TO ESTIMATE THE FREQUENCY OF SEVERE CORE DAMAGE <u>DURING THE 1969-1979 PERIOD</u>.

- THE ESTIMATE WAS BASED ON AN APPROACH SIMILAR TO THAT DESCRIBED BY APOSTOLAKIS AND MOSLEH IN WHICH THE AVERAGE FREQUENCY IS ESTIMATED AS

$$\langle \lambda \rangle = \frac{\sum_i P_i}{T}$$

WHERE P_i IS THE CONDITIONAL PROBABILITY OF SEVERE CORE DAMAGE FOR THE ith EVENT AND T IS THE CUMULATIVE NUMBER OF REACTOR YEARS.

- THIS ESTIMATE IS 4.5×10^{-3}
- SINCE THE METHOD ASSIGNS A NON-ZERO PROBABILITY-OF-FAILURE VALUE IN CASES WHERE SUCCESS ACTUALLY OCCURRED, IT CAN OVERESTIMATE $\langle \lambda \rangle$.
- INITIAL WORK INDICATES THIS OVERESTIMATION COULD BE A FACTOR OF THREE WITH THE EVENT TREES USED IN THE STUDY IF ALL OBSERVED COMBINATIONS OF FAILURES WERE ACTUALLY INDEPENDENT FAILURES.

ACCIDENT SEQUENCE PRECURSOR STUDY HIGHLIGHTS

PERIOD COVERED	1969-1979
TOTAL NUMBER OF LERs SEARCHED	19,400
NUMBER SELECTED FOR DETAILED REVIEW	529
NUMBER SELECTED AS PRECURSORS	169
NUMBER OF SIGNIFICANT EVENTS	52

A POINT ESTIMATE OF THE FREQUENCY OF SEVERE CORE DAMAGE CALCULATED FROM PRECURSOR INFORMATION FOR THE YEARS 1969-1979 LIES BETWEEN 1.7 x 10^{-3} AND 4.5 x 10^{-3} PER REACTOR YEAR.

REASONABLE AGREEMENT EXISTS BETWEEN ASP AND REACTOR SAFETY STUDY INITIATING EVENT FREQUENCIES AND FUNCTION FAILURE PROBABILITIES.

NO VARIATION WITH PLANT AGE CAN BE DEMONSTRATED IN THE NUMBER OF SIGNIFICANT EVENTS.

NO APPARENT DIFFERENCES EXIST BETWEEN PLANT TYPES AND AMONG VENDORS, ARCHITECT-ENGINEERS, AND PLANT POWER RATINGS.