AFF. H SNPS

KMC

DR DONALD F KNUTH
President

December 2, 1982

Dr. Charles C. Graves
Reactor Systems Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Graves:

The purpose of this letter is to provide the information you requested regarding hand calculations performed by KMC with respect to postulated ATWS events in boiling water reactors.

As we have discussed, these calculations were meant to generally model the response of a General Electric BWR 4 reactor with a Mark I vapor suppression containment to postulated ATWS events initiated by turbine trip or main steam isolation valve closure. Some sensitivity calculations were also performed for BWR5/Mark II and BWR6/Mark III systems. The purpose of the calculations was to show the approximate maximum containment pool temperatures which could be expected as a result of these transients. In addition, sensitivity studies were performed to show the effect of operator delay in carrying out the emergency procedures. As you requested, we have also provided containment temperature estimates if the boron delivery rate of the standby liquid control system was increased to an equivalent of 86 gpm.

It should be noted that this work served as an addition to our probabilistic study, "Technical Support for the Utility Group on ATWS" (December 31, 1981) which was submitted to the NRC on April 23, 1982, and was performed by Science Applications, Inc. This PRA utilized the references in enclosure A. In particular, it used deterministic results from a General Electric Company study, "Assessment of BWR Mitigation of ATWS, Volume II," NEDE 24222, December 1979.

On May 12, 1982, the utility group supplemented this study with an update of the PRA which took into consideration the effects of improving operator procedures in handling ATWS events and a reconsideration of the failure criteria used in the original PRA. The original PRA had assumed that any time suppression pool temperature exceeded 200°F that the event would be considered to lead to unacceptable conditions. This was based upon the NRC's maximum allowable pool temperature for design basis

If boiling temp of the pool water ↑ → steam bubble won't collapse → steam goes to air space → cnt ↑↑

analyses. Upon re-examination of this simplistic failure criterion, we decided that the more realistic and more likely scenario during the MSIV closure ATWS would involve containment pressurization as the pool and containment atmosphere heated up. Since General Electric company has performed tests with quencher devices to show stable steam condensation with subcooling as little as 10°F at the quencher device, we felt that if adequate subcooling was available at the quenchers, the suppression pool could survive pool temperatures well in excess of 200°F. In fact, we requested General Electric company's opinion on this issue and they said:

> "As the pool temperature increases the wetwell air space pressure above the pool will increase as a result of higher partial pressure from water vapor and noncondensable gases... The increased pressure will raise the boiling temperature of the water in the suppression pool...
>
> If the water in contact with the steam reaches saturation conditions, no significant pool boundary loads will be encountered because the steam bubbles will not collapse. Under such conditions, the steam would rise to the pool surface and be released to the containment air space, thus increasing the containment pressure and preventing pool-wide boiling and/or condense on the containment surfaces which act as heat sinks. The real limit under such conditions would be associated with exceeding the containment capability as a result of overpressure."

(See enclosure B for GE's report "Quencher Discharge Considerations for ATWS Events.")

Our calculations showed, given the height of water and the partial pressure in the airspace above the quencher, that ten degrees of subcooling can be maintained up to 285°F, which corresponds to 62 psig, the design pressure of many MK I containments. Therefore we concluded that if the boron injection system could shutdown the reactor before the pool temperature exceeded 285°, adequate core cooling could be maintained. For Mark III containments, the design pressure of the containment is 15 psig. However, as reported in the Safety Evaluation Report for the Grand Gulf Nuclear Station, Supplement 3 (NUREG 0831), Mississippi Power & Light and Sandia National Laboratories studies show that the containment is capable of maintaining integrity up to 56 psig.

These two factors (procedures and successful vapor suppression) reduced our calculation of probabilities of unacceptable consequences from ATWS in a generic BWR4/Mark I from $4.1 \times 10^{-5}$ to $1.5 \times 10^{-5}$. This figure was then used in our value impact analysis as submitted to the NRC.

Having reviewed the previously submitted information, let me now turn to our most recent calculations. General Electric, in support of the Emergency Procedure Guidelines for BWR's, performed some calculations of natural circulation reactor core flow and reactor power level as a function of reactor water level. These calculations were provided to the NRC in meetings between NRC and the utilities developing the EPG's. They are also provided in graph form herein as enclosures C and D. Subsequent to this meeting we learned from GE that the reactor power level when the water is at the top of the active fuel should have been 8% rather than 15%. This lower power was based on comparisons which showed that the GE calculations over-predicted the power at low water levels.

KMC used these curves, the analyses of NEDE 24222, and the parameters listed in enclosure E for a typical BWR4/MKI to calculate reactor power level versus time given an MSIV closure ATWS, assuming no boron injection and automatic initiation of HPCI and RCIC at vessel level 2. (Enclosure F) This would be the case in which no operator actions were taken. Note that the power would level out at about 31% based on the combined flow rate of HPCI and RCIC. Dashed curves show the effect on reactor power of purposefully lowering the water level without delay in the ATWS event. We should also note that these curves were smoothed and do not contain peaks and valleys due to relief valves opening and closing.

In our most recent analyses as discussed with the staff and the ACRS, we recomputed our PRA results based on the new EPG's and GE's power versus level information. For this we assumed the MSIV events would always lead to unacceptable consequences; however, certain turbine trip events could be mitigated by operator actions. Originally, we had assumed that the operator would fail 99% of the time in handling ATWS events, which resulted in a "core melt" probability of $4.1 \times 10^{-5}$. Nearly 70% of this core melt probability is made up of turbine trip events in which the main condenser remains available as a heat sink for the reactor. Enclosure G shows the event sequence. The power level at which the reactor would steady out without operator action depends upon the reactor power history and rod pattern, but after recirculation pump trip the reactor power would be between 25 and 40%, of which a steam flow corresponding to 25% full power would go to the main condenser via the turbine bypass valves. Note that this asumed power level is higher than our curve in enclosure F because the main feedwater system would maintain normal water level. The remaining steam, corresponding to 0 to 15% of full power, would be discharged to the suppression pool. Based on our typical pool size, this would result in a pool heatup rate of about 3.75°F/minute. The EPG's require the operator to initiate boron injection and lower water level when pool temperature exceeds 110°F. This will lower core power until at 25% reactor power the main condenser would absorb all the steam generated and the relief valves would close resulting in no more heatup of the pool. In fact, with the RHR system recirculating the pool water, the pool temperature would, at this time, begin to decrease at about 0.5°F/minute. It should also be noted that

the injected boron will be thoroughly mixing in the core at this power/flow condition, so that shutdown will be proceeding as the boron injects. At twenty-eight minutes, it was assumed hot shutdown will be achieved with the pool about 110°F.

It should be noted that for this scenario to be effective, the plant must have its MSIV isolation setpoint changed from level 2 to level 1, or the MSIV isolation blocked so that the isolation valves do not trip while level is being lowered, thereby changing this to a main steam line isolation event (see graph enclosure C and D). Many plants have done this already. It is a standard feature on the BWR6. Also it was suggested by the BWROG to the NRC as a means of minimizing challenges to SRV's in a letter from D. B. Waters to D.G. Eisenhut dated March 31, 1981, entitled "BWR Owners' Group Evaluations of NUREG 0737 Requirements II.K.3.16 and II.K.3.18." (For those early plants with greater bypass capacity, this modification may not be necessary, because the water level need not be dropped so low to reduce power within the capacity of the bypass valves.) *

Enclosure G points out that during the turbine trip ATWS, should the operator delay ten minutes in taking any action, the pool would rise in temperature to no more than 149°, which is below the depressurization requirement of the EPG's and over fifty degrees below the staff's 200°F limit. When one applies this success with a ten minute delay in the PRA, the human error probability drops from virtual failure (.99) to .16. (The curve used by SAI to compute HEP compares favorably with the NRC's operator recovery curve in its IREP study of Arkansas Power & Light's ANO-1, even though different methodologies were used.) Enclosure H shows the effect of this likely success in handling turbine trips on the ATWS core melt probability. It should be noted that the ATWS risk with these assumptions is $1.6 \times 10^{-5}$, which is virtually identical to the $1.5 \times 10^{-5}$. Enclosure I shows the probability numbers used in the utility submittals and as discussed with the ACRS subcommittee on ATWS on October 22. (The May 12, 1982 core melt figure of $1.5 \times 10^{-5}$ was actually used in our value impact calculations.) Once again, it should be noted that this number, $1.6 \times 10^{-5}$, does not include any credit for success in handling the MSIV closure ATWS event, although our analysis shows that that would be the case.

Enclosure J shows the MSIV closure event based on the operator following the EPG's. It uses the power versus time curve of enclosure F and the parameters described in enclosure E. Note that reactor power reduces to 8% at about 10 minutes into the event. When sufficient boron has been injected for hot shutdown, the procedure calls for the water level to be raised and the flow increased such that the boron readily mixes in the core and hot shutdown is achieved at an assumed time of 29 minutes and the pool temperature at 221°F. Also please note that the EPG had required a controlled depressurization of the reactor vessel which added about 43°F to the pool final temperature. From this point onward in the postulated event, additional boron would be injected and a slow controlled cooldown would be performed.

Enclosure K shows the effect of the operator delaying 0, 2 or 10 minutes in initiating his actions. It also shows similar calculations for a BWR 5 or 6. The pool temperatures are somewhat less in a BWR 5 or 6 because the steady state power level with no operator action is lower since the HPCS pump's output varies with reactor system pressure and is much less than HPCI flow.

Our conclusion, based on these calculations, is that if operators response within reasonable times, boiling water reactors can successfully withstand the effects of the postulated worst case MSIV closure ATWS. It should be noted, however, that we took no credit in our final probabilistic calculation for this success path.

KMC and its clients of the ATWS utility group continue to believe that the fixes proposed in our petition for rulemaking (i.e., SDV improvements, RPT, ARI, and improved emergency procedures) are sufficient to cope with ATWS events and to reduce the likelihood of unacceptable consequences from ATWS to an acceptably low level. We support the recommendations of the ATWS task force as described by R.L. Bernero before the ACRS subcommittee on ATWS on October 22, 1982, with the exception of doubling the effective boron injection rate, which we feel is unnecessary. I hope this letter has provided you with an adequate explanation of our analysis and hand calculations. If you need more information, please contact me.

Sincerely,

Donald F. Knuth

Donald F. Knuth
KMC, Inc.

Enclosure A

## REFERENCES FOR SAI PRA


Assessment of BWR Mitigation of ATWS, Volume II, (NUREG-0460, Alt 3), NEDE-24222, December 1979.

Probabilistic Analysis of the Reliability of BWR/4 Systems for Small LOCA Events, Science Applications, Inc., General Electric, NEDE/24809, April 1980.

ATWS: A Reappraisal:
Part I. EPRI NP-251, August 1976
Part II. EPRI NP-265, Volume 1,2,3, August 1976
Part III. EPRI NP-801, July 1978

SECY-80-409, Proposed Rulemaking to Amend 10 CFR Part 50 Concerning ATWS Events, NRC, September 4, 1980.

ALO-78, SAI-154-79-PA, Component Failures that Lead to Reactor Scrams, SAI, April 1980.

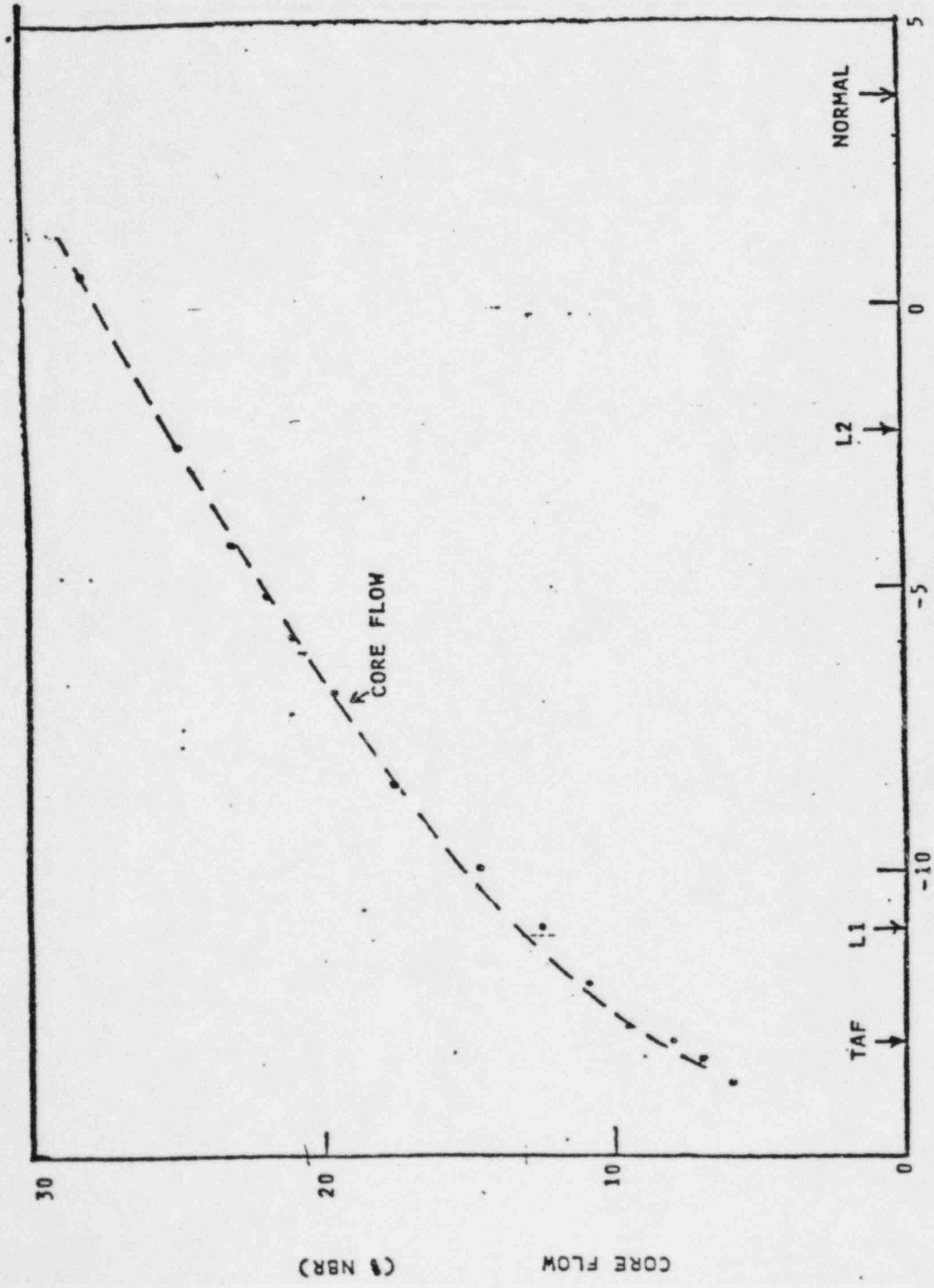
NUREG-0460, Anticipated Transients Without Scram for Light Water Reactors, NRC, Volumes 1,2, April 1978, and Volume 3, December 1978.

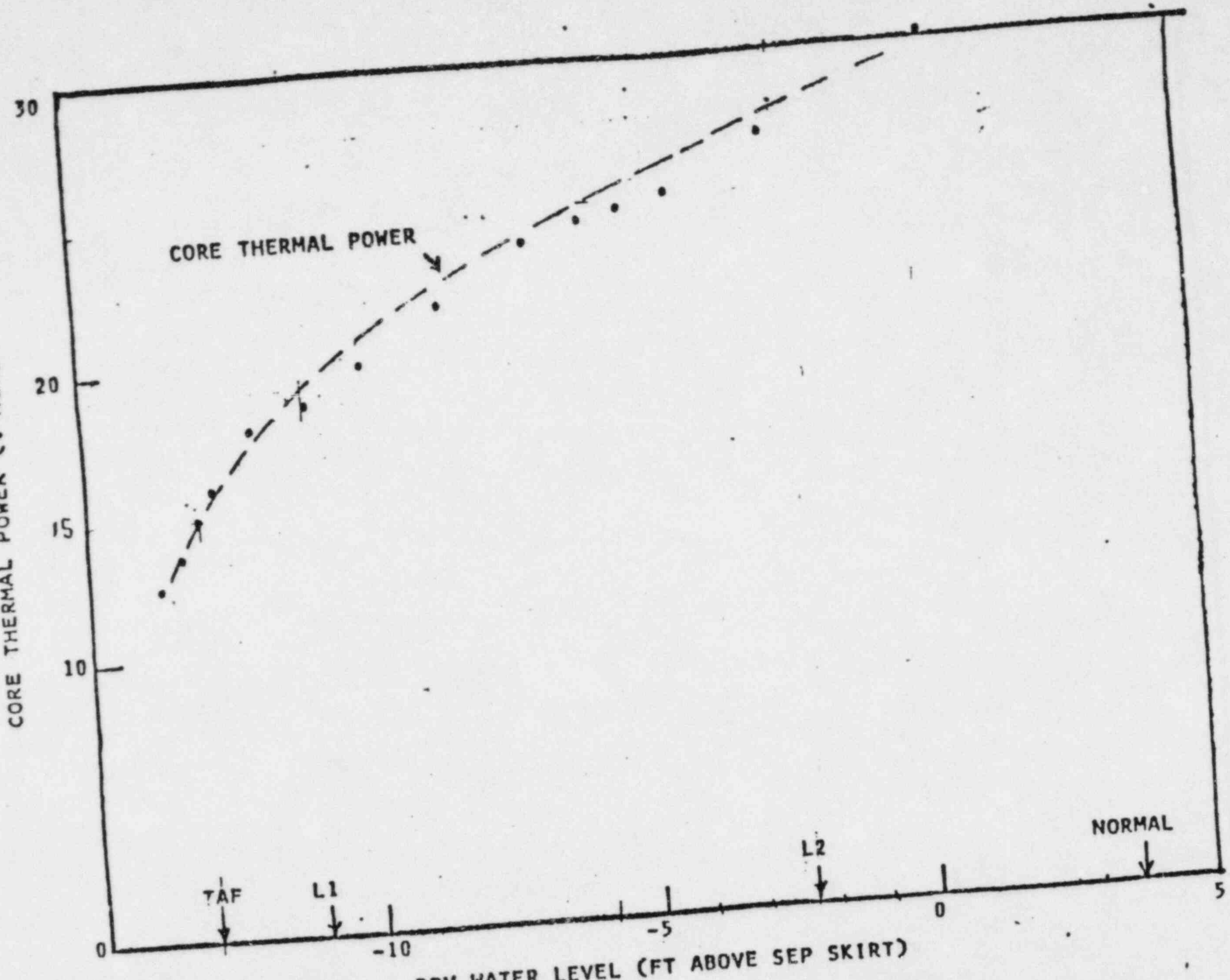
WASH-1270, "Technical Report on ATWS for Water-Cooled Power Reactors," NRC, Septemb er 1973.

Probabilistic Risk Assessment of the Limerick Generating Station, Philadelphia Electric Company, March 1981.

WASH-1400, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, NRC, October 1975.

Enclosure C


CORE FLOW
NORMAL
L2
L1
TAF
5
0
-5
-10
RPV WATER LEVEL (FT ABOVE SEP SKIRT)
30
20
10
0
CORE FLOW (% NBR)




REACTOR CORE THERMAL POWER VS RPV WATER LEVEL - REDY ESTIMATE

Enclosure D

Enclosure E

# KEY PARAMETERS USED IN CALCULATING CONTAINMENT TEMPERATURES

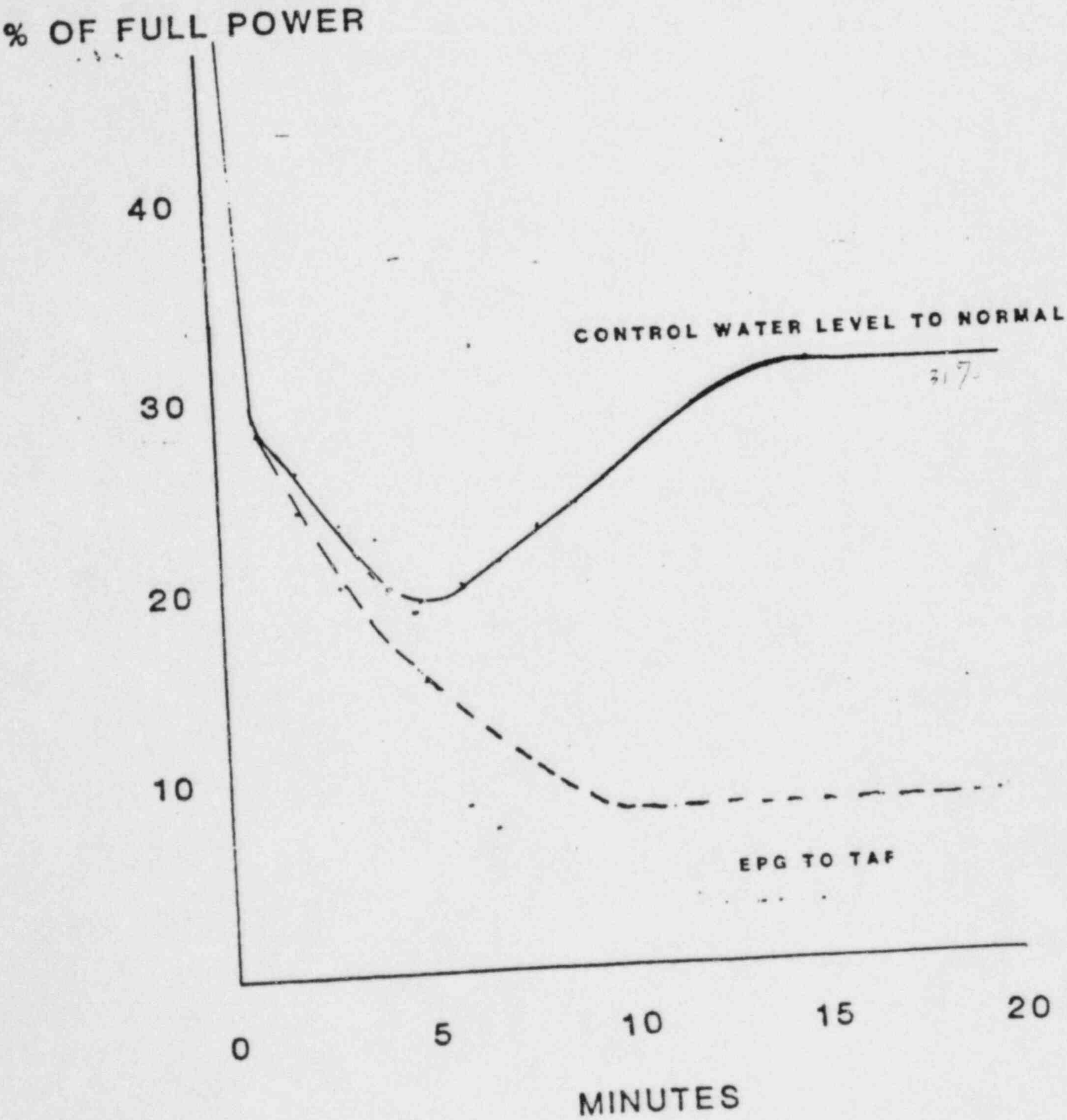
| | |
|---|---|
| Power | 3393 MW(t) |
| Wet Well Volume (liquid) | 130,000 $FT^3$ |
| (free) | 150,000 $FT^3$ |
| Drywell Volume (free) | 250,000 $FT^3$ |
| HPCI Flow | 5600 GPM |
| RCIC Flow | 625 gpm |

Encl 1

# REACTOR POWER ESTIMATE

## BWR-4 NO BORON {auto initiation of HPCI & RCIC at level 2}

MSIV closure ATWS



Enclosure G

Elevation rate 4-6°F/min
[illegible]

## TURBINE TRIP ATWS

(70% of TRANSIENTS)

| TIME/POOL TEMP. | CONDITION | ACTION |
| --- | --- | --- |
| 0/90 | 100% POWER TURBINE TRIP<br>POWER, PRESSURE INCREASE,<br>RPT, POWER DECREASE TO 25-40%<br>TURBINE BYPASS OPEN TO 25%<br>SRV'S OPEN 0-15% POWER TO POOL. | MODE SW TO S/D |
| 1/104 | (ASSUME ARI UNSUCCESSFUL) | ATTEMPT SCRAM<br>MONITOR POOL |
| 2/108 | POOL TEMP. INCREASE 3.75°/MIN<br>W/O POOL COOLING | |
| 3/112 | | START SLCS<br>THROTTLE FW<br>ALIGN RHR TO POOL |
| 4/115 | RPV LEVEL DROPPING TO L2<br>POWER DECREASING | TERMINATE HPCI/RC[illegible] |
| 5/119 | RPV LEVEL ABOUT -72"<br>POWER ≤ 25% (ALL TO CONDENSER)<br>BORON MIXING IN CORE. | MONITOR CONDITION[illegible] |
| 10/119 | POWER SLOWLY DECREASING<br>(ASSUME POOL COOLING BEGINS HERE<br>TEMP. DROPS .5°/MIN.) | |
| 28/110 | HOT SHUTDOWN ACHIEVED | CONTINUE BORON<br>INJECTION |

IF OPERATOR ACTION WERE DELAYED 10 MINUTES, THE MAXIMUM POOL TEMPERATURE WOULD BE 149°

MSIV closure on
Level 1

# BWR ATWS PRA

ORIGINAL RESULT $4.1 \times 10^{-5}$

.. DOMINATED BY TURBINE TRIP (70%)

.. HEP = .99

WITH EPG $1.6 \times 10^{-5}$

.. OPERATOR CAN DELAY 10 MINUTES ON TURBINE TRIP EVENT AND STILL BE SUCCESSFUL, (Pool temp < 200°F)

.. HEP ≤ .16

single action HEP may be optimistic

# BWR SENSITIVITY TO HUMAN ERROR
# IN
# INITIATION OF PROCEDURE

| | PROBABILITY OF UNACCEPTABLE CONSEQUENCES |
|---|---|
| NRC BASE CASE (PRE-UTILITY PROPOSAL) | 2 E-4 |
| RECALCULATED BASE LINE | 1.3 E-4 |
| ORIGINAL RULE (NO CREDIT FOR OPERATOR ACTION) | 4.1 E-5 |
| AMEND RULE (May 12, 1982) (Credits 285°F Pool Temp) | 1.5 E-5 |
| FINAL CALCULATION USING EPG'S (Credits EPG for Turbine Trip, No Credit for Exceeding 200° Pool) | 1.6 E-5 |

Enclosure J

439gpm

## MSIV CLOSURE ATWS

| TIME/POOL TEMP. | CONDITION | ACTION |
|---|---|---|
| 0/90 | 100% POWER. MSIV's TRIP<br>POWER, PRESSURE INCREASE<br>RPT POWER DECREASE TO 25-40<br>FW TO ZERO (STEAM DRIVEN)<br>RPV LEVEL AND POWER DROP. | MODE SW TO S/D |
| 1/106 | RPV LEVEL AT L2, POWER 28%<br>HPCI/RCIC INITIATE<br>(ARI FAILS) | ATTEMPT SCRAM<br>MONITOR POOL |
| 2/112 | RPV LEVELS DROP AS SRV FLOW<br>EXCEEDS HPCI/RCIC | INITIATE SLCS<br>STOP HPCI/RCIC<br>STOP FW (ELEC. PUMPS)<br>ALIGN RHR TO POOL |
| 10/146 | RPV LEVEL NEAR TAF<br>POWER 8%<br>Tpool AT HCTL | CONTROL LEVEL<br>BEGIN RPV DEPRESS W/S |
| 27/217 | HOT S/D BORON IN CORE | RAISE WATER LEVEL<br>TO MIX BORON |
| 29/221 | HOT S/D ACHIEVED | CONTINUE BORON<br>INJECTION. |

but [illegible]

MSIV CLOSURE EVENT

CONTAINMENT TEMPERATURE SENSITIVITY

SLCS FLOW OF 86 GPM

TAF WATER LEVEL = 8% POWER

| | TIME DELAY IN INITIATION | | |
|---|---|---|---|
| BWR-4 WITH HPCI, RCIC | 0 | 2 | 10 |
| MAINTAIN NORMAL LEVEL | 171 | 186 | 234 |
| EPG | 186 | 197 | 239 |
| EPG NO BLOWDOWN | 143 | 154 | 196 |
| BWR-5 OR 6 WITH HPCS, RCIC | | | |
| MAINTAIN NORMAL LEVEL | 162 | 167 | 189 |
| EPG | 148 | 184 | 218 |
| EPG NO BLOWDOWN | 135 | 141 | 175 |

# MSIV CLOSURE EVENT

## CONTAINMENT TEMPERATURE SENSITIVITY

## SLCS FLOW OF 43 GPM

## TAF WATER LEVEL = 8% POWER

OLD MBAP on PEAK at 190°F 8/83

| | | | | |
|---|---|---|---|---|
| | TIME DELAY IN INITIATION | | | |
| BWR-4 WITH HPCI, RCIC | 0 | 1 min | 2 | 10 |
| MAINTAIN NORMAL LEVEL | 220 | | 235 | 283 |
| EPG | 211 | | 221 | 264 |
| EPG NO BLOWDOWN | 168 | | 176 | 221 SHOREHAM |
| | | 220+ | 220+ 5-6°F/min* | 260-270+* |
| BWR-5 OR 6 WITH HPCS, RCIC | | | | |
| MAINTAIN NORMAL LEVEL | 202 | | 207 | 229 |
| EPG | 202 | | 209 | 243 |
| EPG NO BLOWDOWN | 159 | | 166 | 200 |

+ Lowell

* NEDE 24222

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

APR 13 1983

MEMORANDUM FOR: Distribution

FROM: Jim Martin
Reactor Risk Branch
Division of Risk Analysis, RES

SUBJECT: A PERSPECTIVE ON EMERGENCY PLANNING, RISK AND THE SOURCE TERM ISSUE

Attached for your information is a set of CCDFs which show the potential benefits of following the emergency response guidance in NUREG-0654/FEMA-REP-1, Rev. 1. Succinctly, this guidance says that for a core melt accident evacuate early, within about two miles, preferably before a major release; everybody else take shelter; send monitoring teams out to find hot spots (e.g. >1R/hr); and evacuate the hot spots expeditiously if an actual release occurs.

To illustrate the potential benefits of this action plan a set of CRAC2 runs was made for the Shoreham site. Comparisons to the summary evacuation CCDFs in the SNL siting study were made for the SSTI release scenario. Early evacuation areas within 1,2,3 and 5 miles of the site were assumed-everybody else was exposed to the 2 hr. puff release plus four (vs 24) hours of ground contamination before leaving. Cases with rain and no rain were invesitgated. People in the early evacuation zones were presumed to start to leave at the beginning of the release and traveled at 10 mph. If they were to leave earlier an incremental additional benefit would accrue.

The results are displayed in the attached CCDFs, which are self explanatory for the most part. In essence, ZERO early fatalities were calculated for the three mile early evacuation cases and only two weather sequences contributed to early fatalities for the two mile early evacuation assumption. Little or no differences are observed when the rain cases are switched off (washout coefficient set equal to zero).

The last set of CCDFs were generated with the release fractions for SST1 cut in half, except 100 percent of the noble gases was released. Here, no fatalities were calculated for any weather sequences when an early evacuation of the two mile area was assumed.

~~8304260292~~ 13

A/4

Early injury CCDFs are also attached. Here, the three mile early evacuations had a pronounced benefit and for the five mile early evacuation scenario no early injuries were calculated for the SST1/2 cases.

The final graph shows the width of plumes vs "downwind" distance. This shows that people normally would have to travel only short distances to get away from hot spots.

I'm writing a NUREG which will discuss these perspectives in more detail, but I believe the results are important enough to present the gist of them now. I should have a draft in a month.

Thanks to Dan Alpert and Jay Johnson of SNL for doing the runs for me.

Jim Martin

Jim Martin
Reactor Risk Branch
Division of Risk Analysis, RES

Enclosure:
As stated

Original

Distribution:
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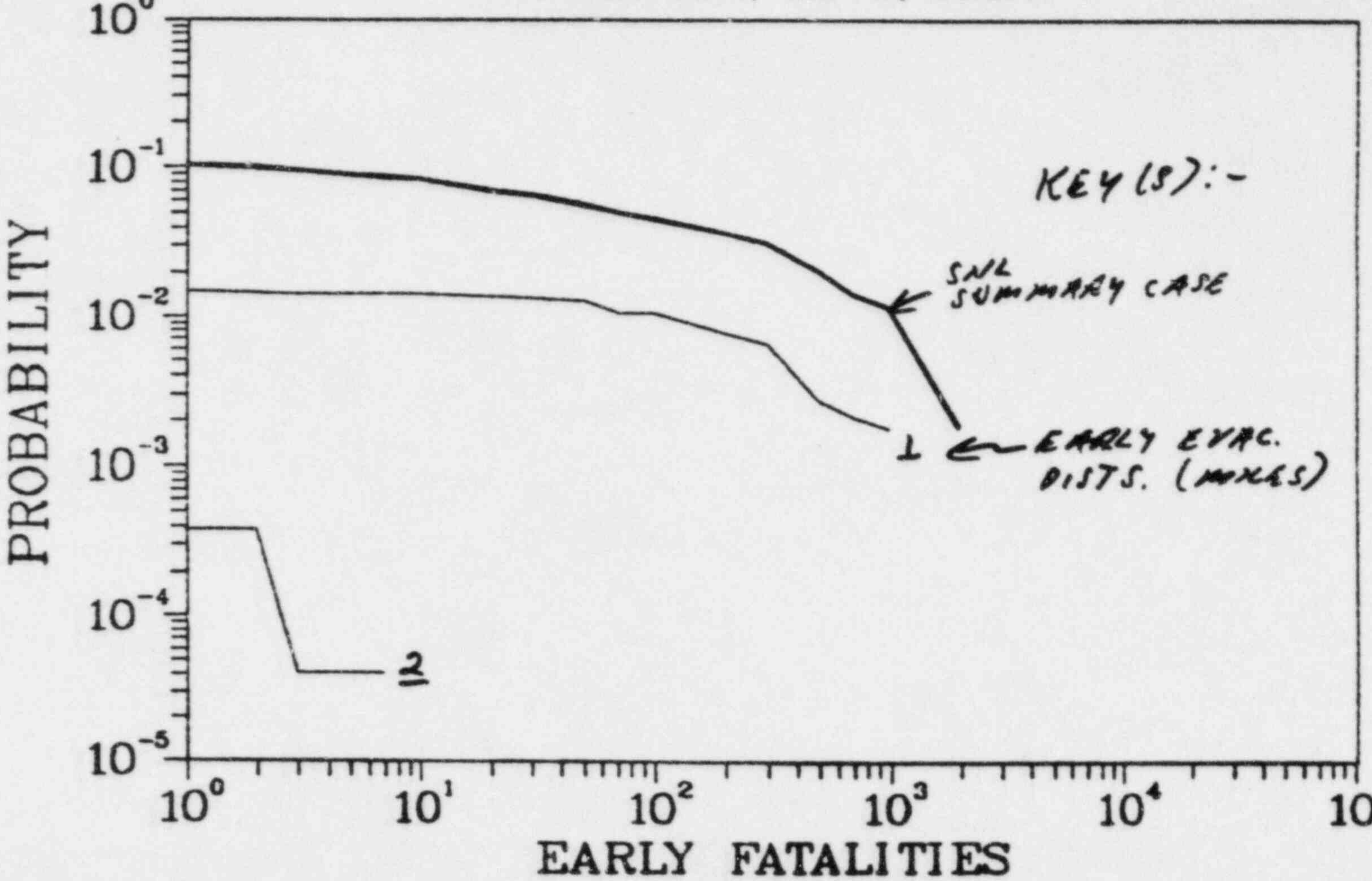
E. Jordan

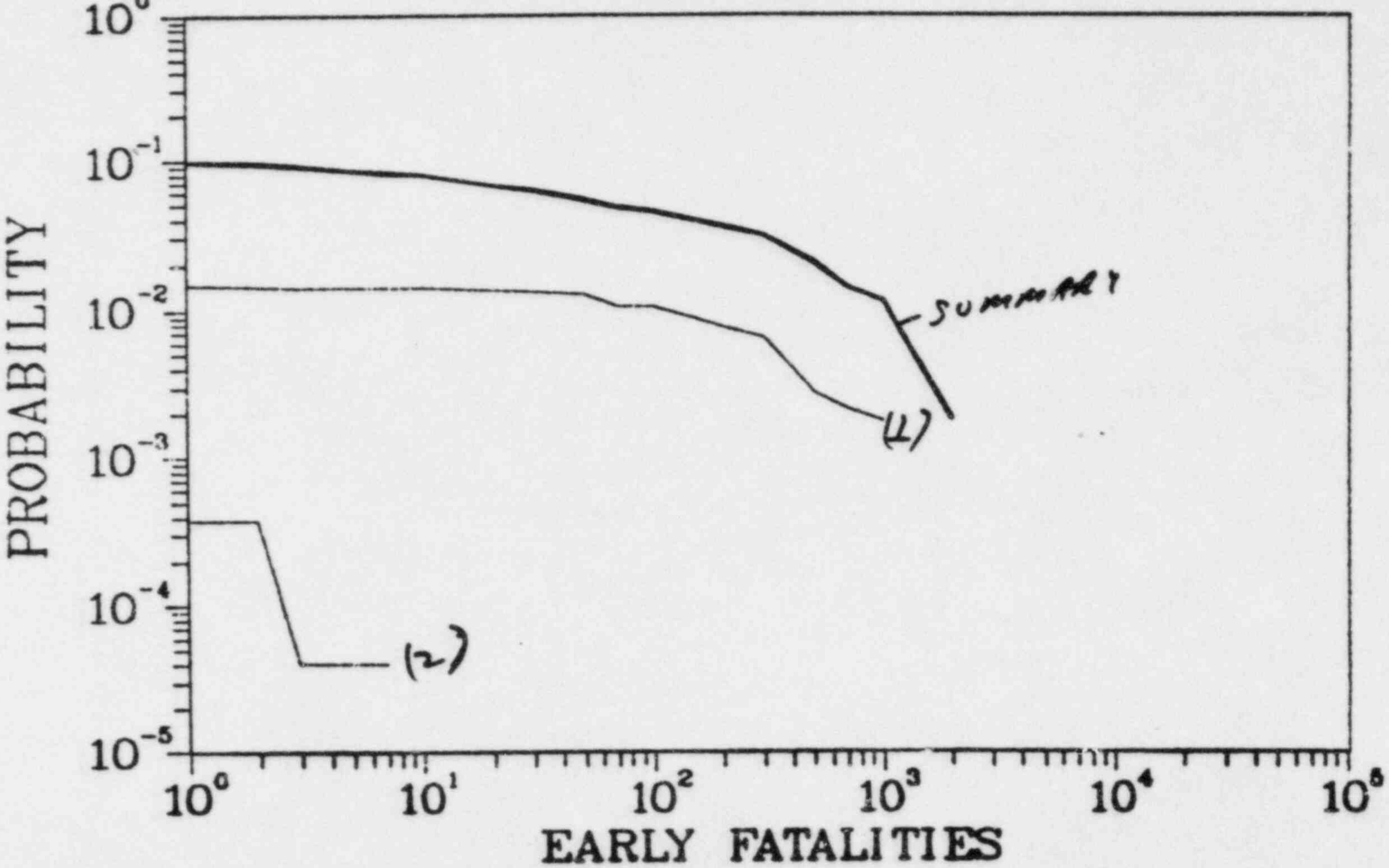
- SUMMARY (10 MI EVAC, 1,3,3hr. DELAY)
- 1 MI. EVAC., 4 hr. RELOC.
- 2 MI. EVAC., 4 hr. RELOC.

INCL.
SHOREHAM, SST1, RAIN



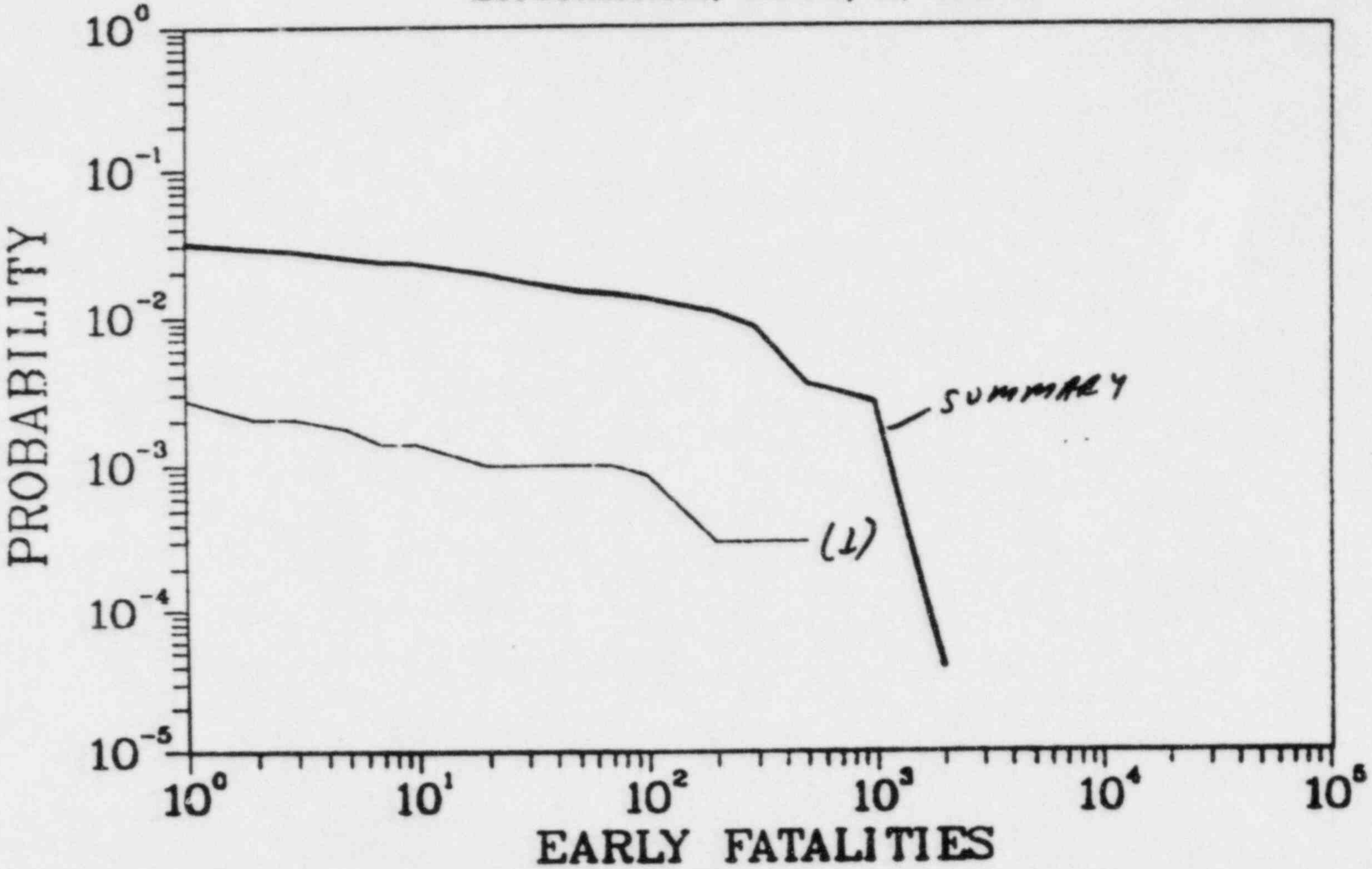
- SUMMARY (10MI EVAC., 1,3,5-hr. RELOC)
- 1MI. EVAC, 4hr. RELOC.
- 2MI. EVAC., 4hr. RELOC

SHOREHAM, SST1, NO RAIN

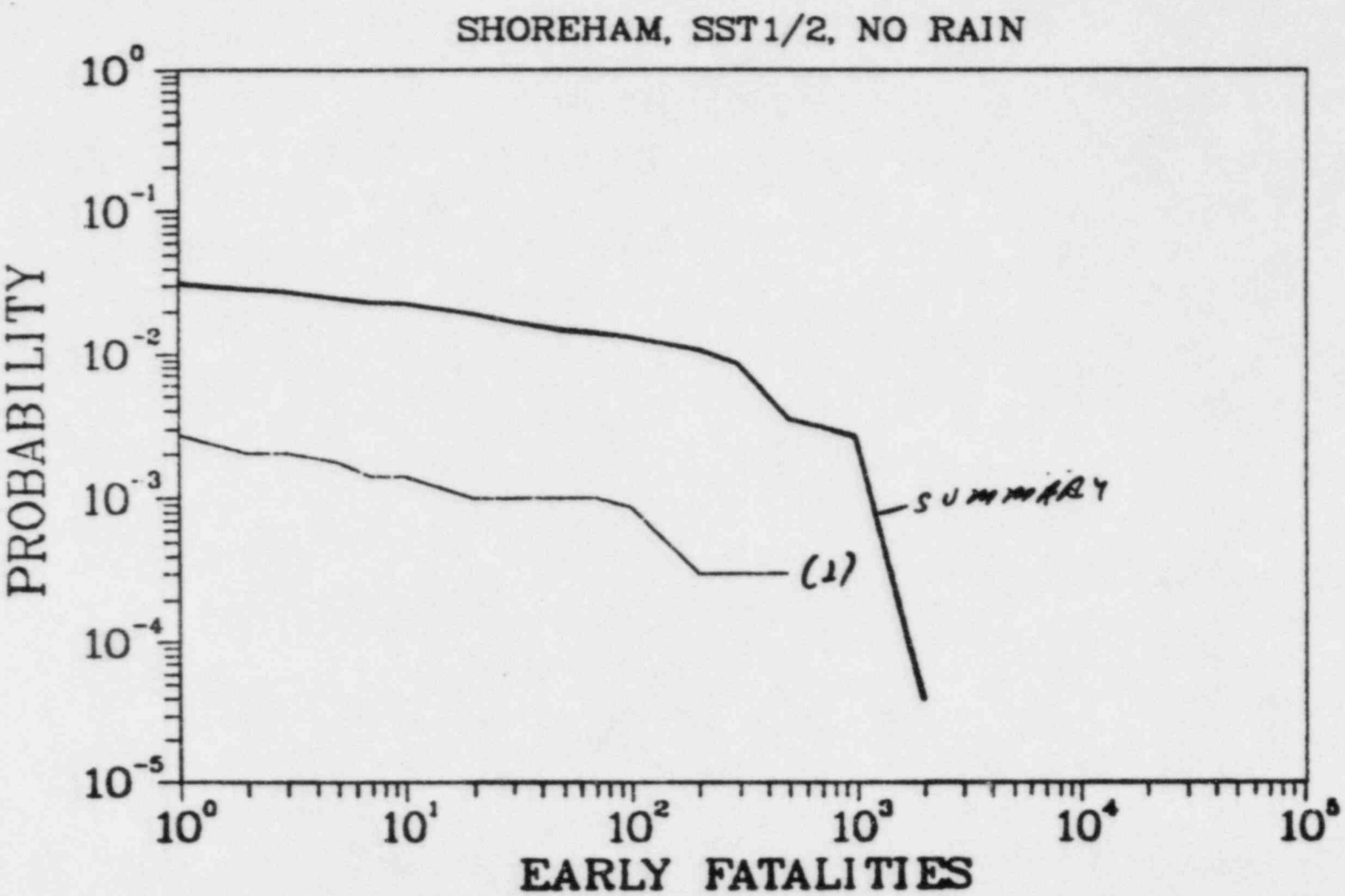


- SUMMARY [illegible]
- 1MI. EVAC (4 hr REZ?)

INCL.
SHOREHAM, SST1/2, RAIN

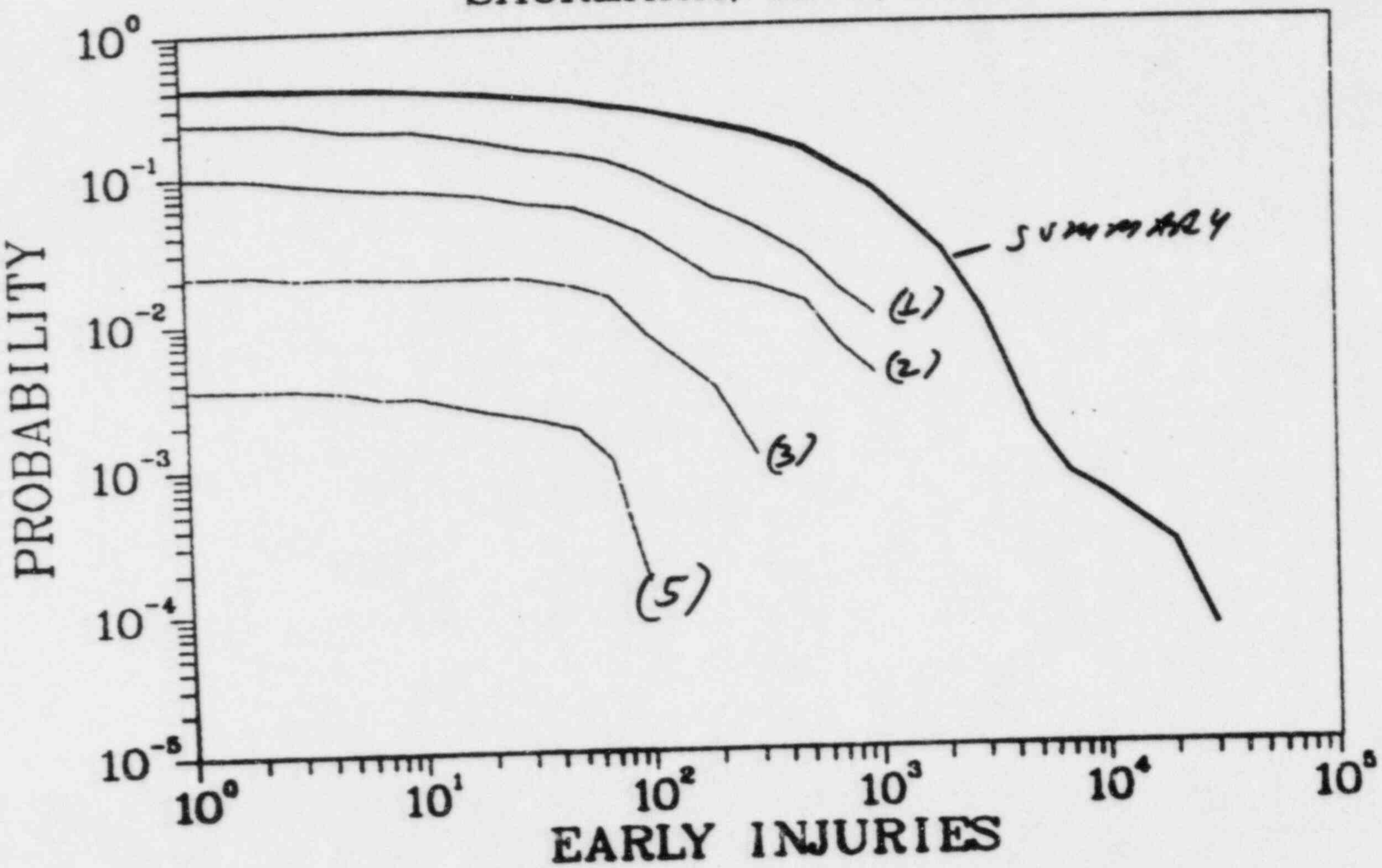


– SUMMARY (10 MI. EVAC, 1,3,5-hr. [illegible])
– 1 MI. EVAC, 7 h. RELOC

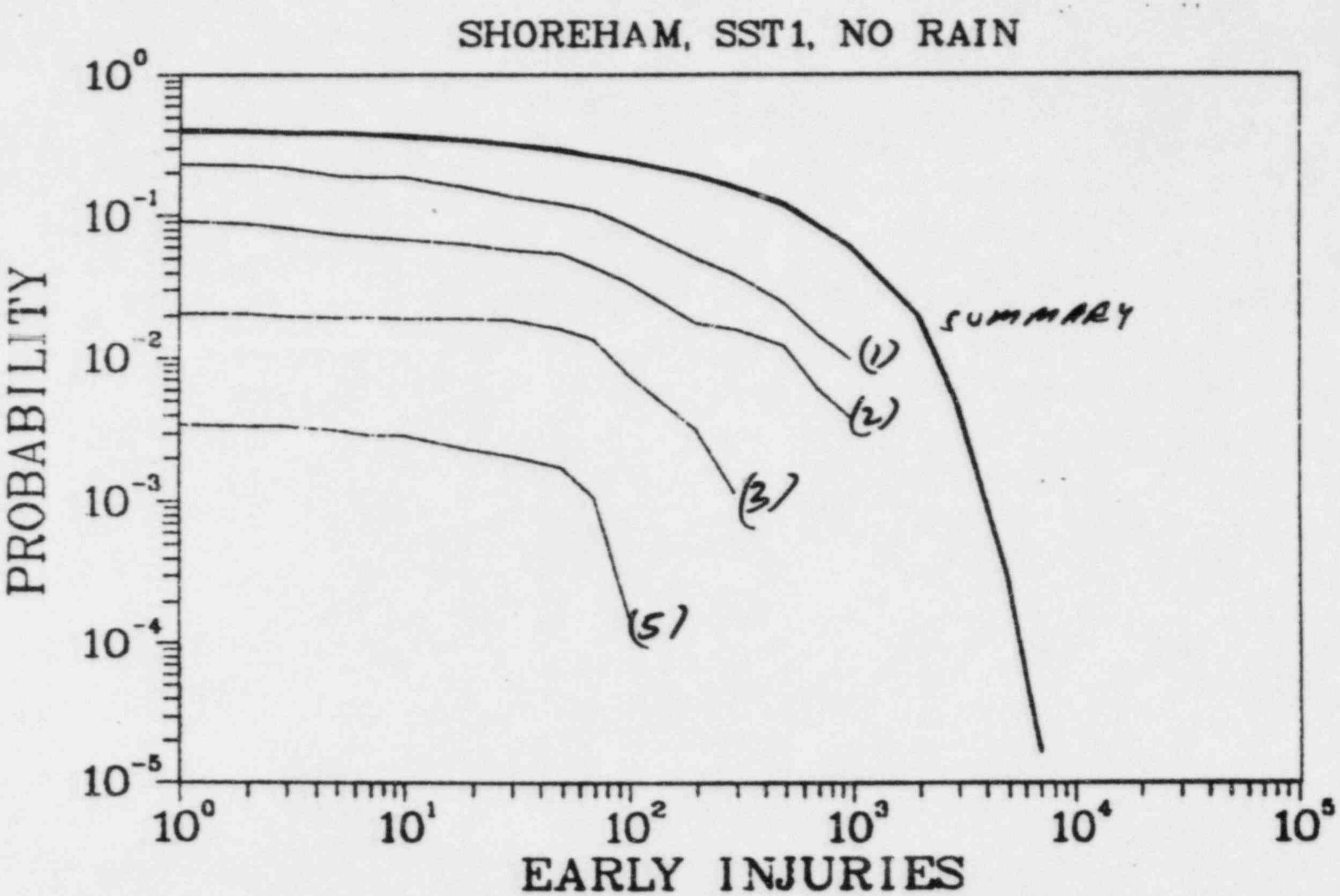


- SUMMARY (10 MI. EVAC., 1, 3, 5-hr DELAY)
- 1 MI. EVAC; 4 hr RELOC.
- 2 MI. EVAC, "
- 3 MI. EVAC, "
- 5 MI. EVAC, "

INCL.
SHOREHAM, SST1, RAIN

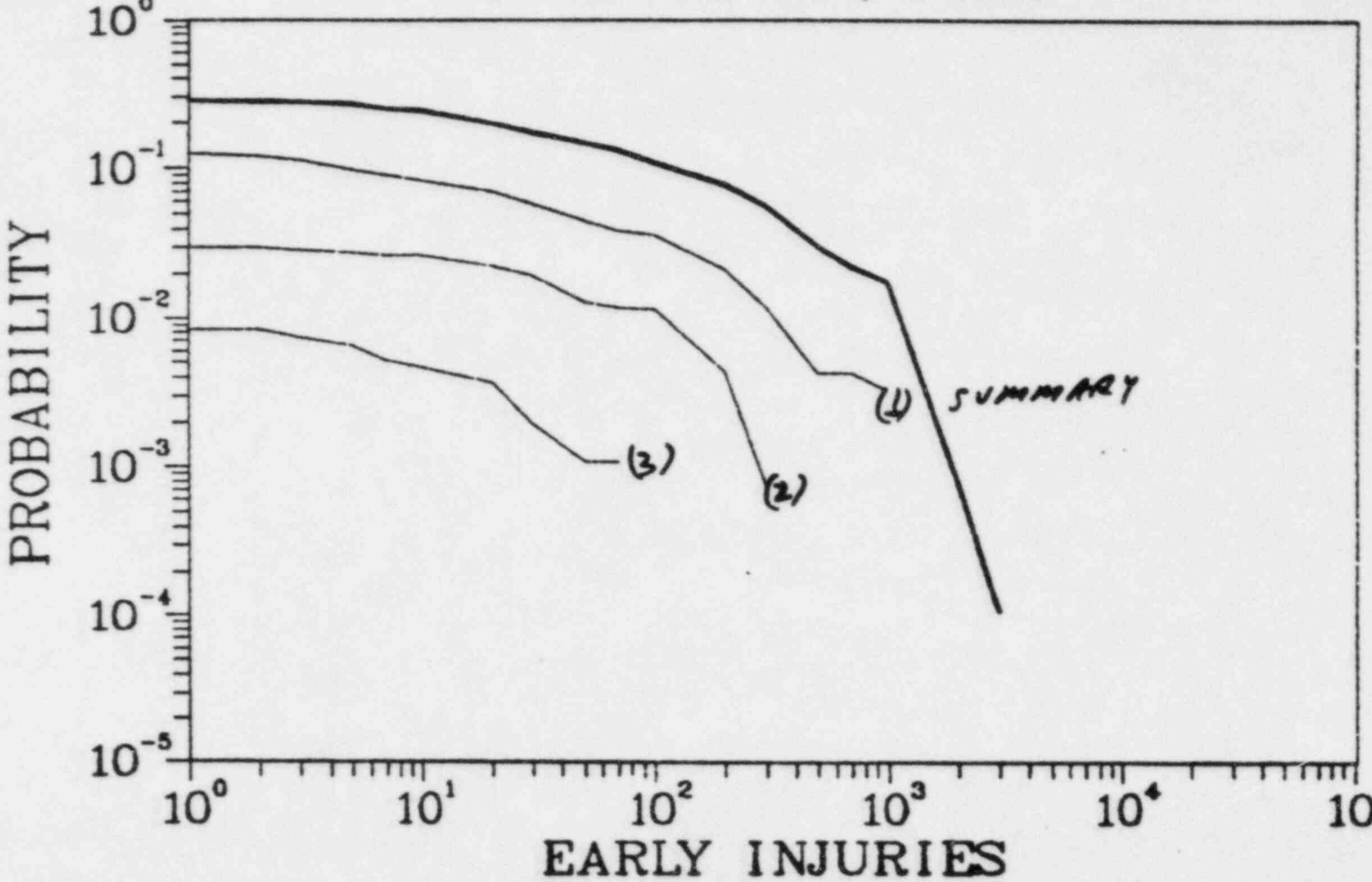


- SUMMARY (10 MI. EVAC, 1, 3, 5-hr DELAY)
- 1 MI. EVAC., 4 hr. RELOC
- 2 MI. EVAC., "
- 3 MI. EVAC., "
- 5 MI. EVAC. "



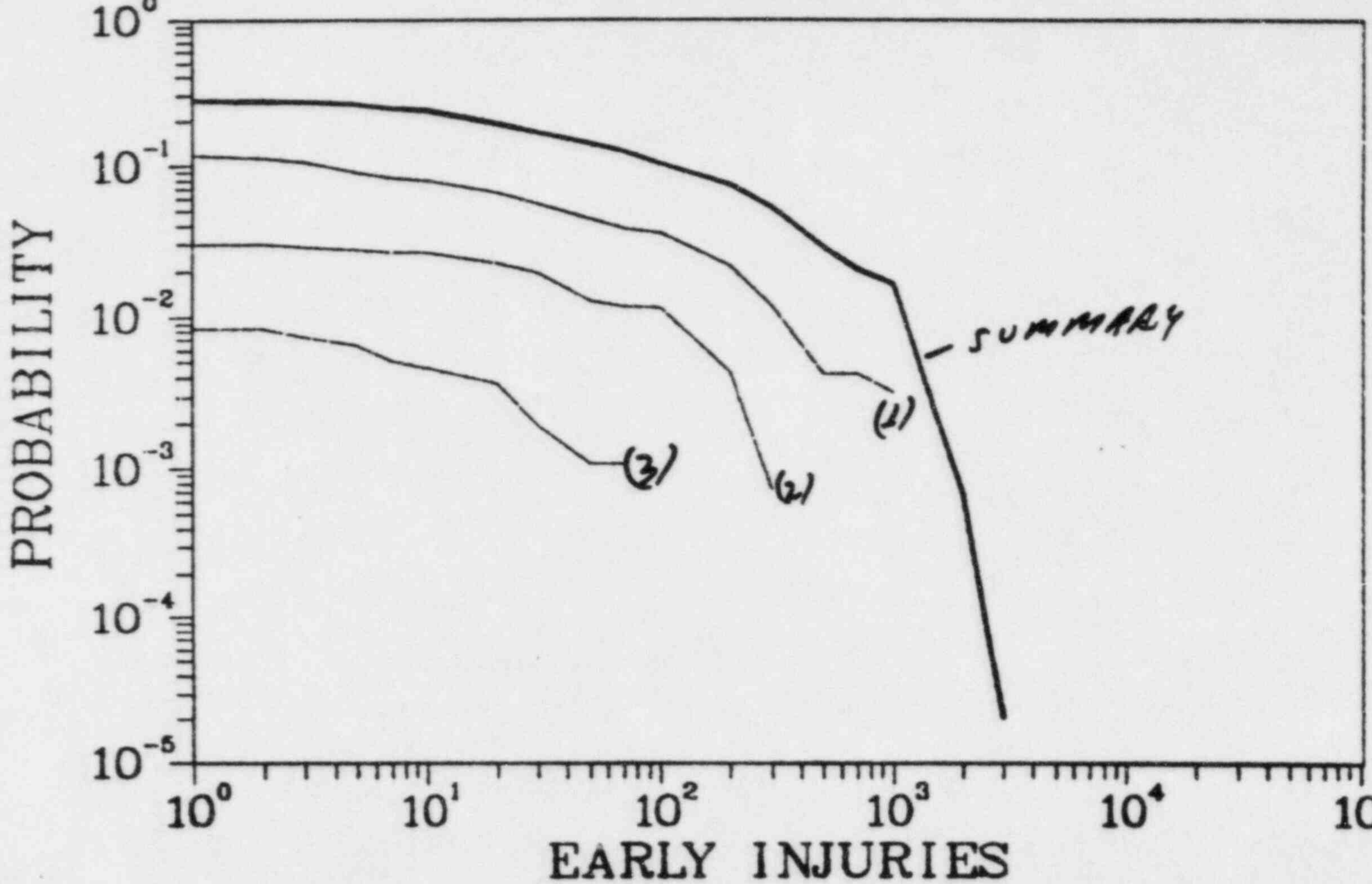
- SUMMARY (10 MI. EVAC, 1,2,3 MI. RELOC.)
- 1 MI. EVAC, 7 h. RELOC
- 2 MI. EVAC, 7 h. RELOC
- 3 MI. EVAC, 7 h. RELOC
- 5 MI. EVAC = 0)

INCL.
SHOREHAM, SST1/2, RAIN



- SUMMARY (10 MI EVAC, 1,2,3 HR DELAY)
- 1 MI. EVAC, 4 HR DELAY
- 2 MI. EVAC, "
- 3 MI. EVAC, "
(- 5 MI EVAC. = 0)

SHOREHAM, SST1/2, NO RAIN



# SHOREHAM. [99%ile = ESTIMATED]

| RAIN (λ=1.E-4) | | SST1 | | | SST 1/2 | | |
|---|---|---|---|---|---|---|---|
| | | MEAN | 99%ile | PEAK | MEAN | 99%ile | PEAK |
| EVAC (SUMMARY) 30%, 40%, 30% (1,3,5-hr Delays) | E.F. | 33.4 | 1500. | 2160. | 8.64 | 200. | 2100. |
| 10 MI EVAC 24 hr. RELOC | E.I. | 232. | 3000. | 37,300. | 59.2 | 1400. | 3660. |
| EVAC 4 | E.F. | 6.64 | 120. | 1930. | 0.260 | [<1.0] | 556. |
| 1 MI EVAC 4 hr. RELOC | E.I. | 41.9 | 1000. | 1800. | 14.7 | 300. | 1830. |
| EVAC 5 | E.F. | 1.23E-3 | [<1.0] | 7.74 | 0. | 0. | 0. |
| 2 MI EVAC 4 hr. RELOC | E.I. | 15.9 | 570. | 1200. | 2.80 | 100. | 373. |
| EVAC 6 | E.F. | 0. | 0. | 0. | 0. | 0. | 0. |
| 3 MI EVAC 4 hr. RELOC | E.I. | 2.24 | 85. | 432. | 0.192 | [<1.0] | 98.0 |
| EVAC 7 | E.F. | 0. | 0. | 0. | 0. | 0. | 0. |
| 5 MI. EVAC 4 hr. RELOC | E.I. | 0.156 | [<1.0] | 150. | 0. | 0. | 0. |

| NO RAIN | | SST1 | | | SST 1/2 | | |
|---|---|---|---|---|---|---|---|
| | | MEAN | 99%ile | PEAK | MEAN | 99%ile | PEAK |
| EVAC SUM | E.F. | 32.8 | 1100. | 2160. | 8.64 | 200. | 2100. |
| | E.I. | 191. | 2700. | 8090. | 55.8 | 1700. | 3280. |
| EVAC 4 | E.F. | 6.64 | 100. | 1930. | 0.260 | [<1.0] | 556. |
| | E.I. | 40.1 | 1000. | 1800. | 14.4 | 330. | 1830. |
| EVAC 5 | E.F. | 1.23E-3 | [<1.0] | 7.74 | 0. | 0. | 0. |
| | E.I. | 15.5 | 570. | 1200. | 2.80 | 120. | 373. |
| EVAC 6 | E.F. | 0. | 0. | 0. | 0. | 0. | 0. |
| | E.I. | 2.24 | 85. | 432. | 0.192 | [<1.0] | 98. |
| EVAC 7 | E.F. | 0. | 0. | 0. | 0. | 0. | 0. |
| | E.I. | 0.155 | [<1.0] | 150. | 0. | 0. | 0. |

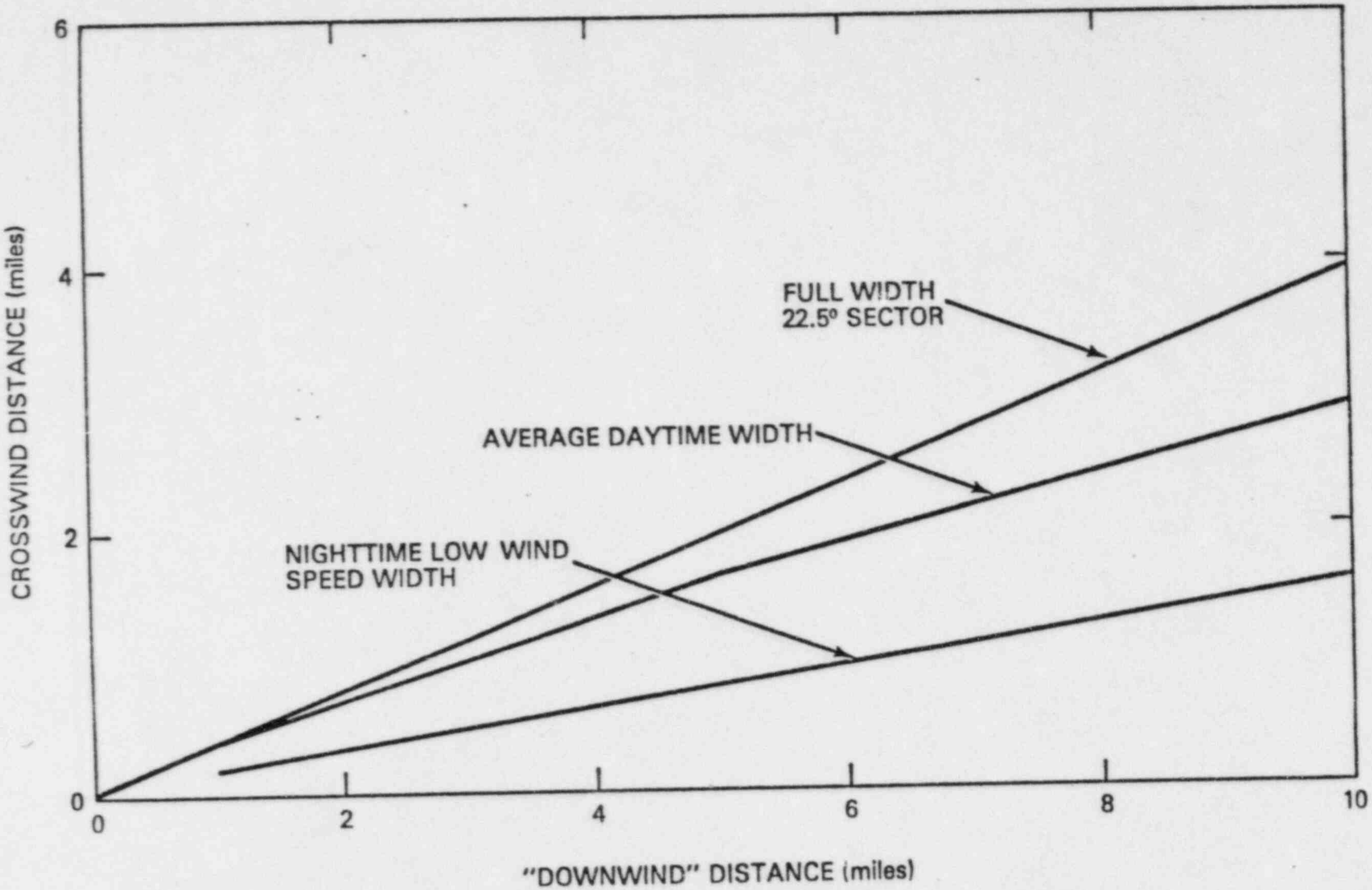



**Figure 4. Theoretical full widths of puffs at one percent of maximum concentration vs distance along track of puff.**

James A. Martin, Jr

- 98 -

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C.

SEP 2 0 1983

MEMORANDUM FOR: Ashok C. Thadani, Chief
Reliability and Risk Assessment Branch
Division of Safety Technology

FROM: L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration

SUBJECT: WAPR, SHOREHAM, MIDLAND AND SEABROOK PRA REVIEWS

As requested in your memo of September 12, 1983, the assigned AEB reviewers for the seven PRAs are listed below:

| | |
|---|---|
| Limerick | S. Acharya, X28375 |
| Millstone | P. Easley, X28341 |
| GESSAR | J. Read, X28301 |
| ~~Shoreham~~ | ~~P. Easley, X28341~~ |
| Midland | S. Acharya, X28375 |
| Seabrook | M. Wohl, X27065 |
| WAPR | J. Read, X28301 |

Based upon our experience, we consider any PRA without external events to be incomplete. This experience indicates that risk inferences suggesting regulatory action based solely on internal events may very well be incorrect. For example, the Indian Point, Zion and Limerick PRA's appear to indicate that cost/benefit balancing based upon estimated risks and risk reductions from considerations of only internal events can be grossly overshadowed by the incremental risks from external events. We suggest that LILCO be urged to include external events in their PRA.

Two plants (Midland and Seabrook) have had hearing testimony given by AEB with regard to public risks based upon "mini-PRAs". For these plants, publication of staff PRA reviews could constitute revision of staff FES testimony. AEB intends to consider board notifications in these cases upon receipt of PRA's.

While agreeing to participate in Westinghouse's proposed presentations, we would object to any scheduled review activity based upon a piecemeal "modular" submittal such as used by Westinghouse in the WAPR SAR submittal. We have found such drawn-out and disjointed reviews to be very inefficient in use of reviewer's time.

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We suggest that PRA review schedules be established at the earliest possible date to assure that resources are provided in a timely and effective manner. Included in such a schedule should be provisions for AEB to provide comparisons of environmental risks with what has been presented in FES's for Seabrook and Midland.

L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration

cc: R. Mattson
R. Vollmer
T. Speis
W. Houston
F. Rowsome
B. Sheron
W. Butler
J. Rosenthal
R. Frahm
A. Buslik
G. Kelly
S. Newberry
E. Chan
S. Davis