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Testimony: Advisory Committee on Reactor Safeguards

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In reply to the Committee's request, I will briefly discuss the implications of the Reactor Safety Study (WASH-1400) for nuclear power plant licensing and will point out areas of research where further work would be desirable. Detailed background for this discussion is contained in my published review of WASH-1400, in correspondence between Commissioner E.A. Mason and myself, and also in an exchange of letters with Mr. Levine of the Office of Nuclear Regulatory Research.

I would like to take this occasion to thank Commissioner Mason for his excellent cooperation in assuring access to NRC documents and to the NRC Staff, and for providing me with the opportunity to exchange views on the reactor safety issue with him.

There are two major safety aspects of licensing decisions for commercial power reactors: first, determining the suitability of proposed sites; and second, assessing the adequacy of proposed plant designs. Analyses of the likelihood and consequences of catastrophic nuclear accidents --such as those presented in WASH-1400--in principle can provide valuable information for assessing both site suitability and adequacy of design. In particular, the meteorological and demographic characteristics of specific sites strongly influence the scale of potential accident *consequences*, and the *probability* of catastrophic reactor failure and subsequent atmospheric venting of radioactivity is dependent in important ways on the details of reactor and containment designs.

As presently written and interpreted, Title 10 of the Code of Federal Regulations--and Part 100 in particular--effectively excludes consideration of the probabilities and consequences of catastrophic core-melt accidents in the course of licensing proceedings. As a serious attempt to provide an evidentiary basis for possible amendments of existing NRC regulations--so as to deal appropriately with potential core-melt accidents--WASH-1400 represents an important step forward in the regulatory process.

In their present form, neither the probability nor the consequence calculations in WASH-1400 are suitable for use in determining reactor design or site selection criteria. However, though its ostensible purpose is more general, the WASH-1400 consequence methodology--which consists in large part of standard meteorological techniques--could be adapted for use with respect to specific sites. From the preliminary work we have seen, or have produced ourselves at M.I.T., there are very large risk differentials between existing sites, and such site-specific consequence calculations may therefore prove to be crucial in guiding future site selection, and in re-assessing the suitability of existing metropolitan area sites.

Absolute Probability Calculations

In contrast to the situation with regard to the consequence methodology, there appear to be very serious questions regarding the usefulness of the "fault-tree," absolute probability approach used in WASH-1400.

I will mention two outstanding issues in that connection: (i) the difficulties involved in making the inclusive listing of failure sequences which is essential to any absolute probability approach; and (ii) the necessity for estimating "common-mode," interactive failure probabilities.

There are a number of fundamental barriers to making a realistic inclusive listing of potential sub-system or component failure sequences leading to a catastrophic accident:

(1) For any reasonable choice of sub-systems, there are a very large number of potentially significant failure mechanisms. Those mechanisms involving man-machine relationships are particularly difficult to individually enumerate and analyze.

(2) The number of significant failure mechanisms would be expected to expand enormously as uncertainties, statistical and scientific, increase. The primary difficulty here is that one knows relatively little about very low probability events involving long chains of individual component and interactive failures. While the median failure probabilities associated with such sequences are low, there are in general expected to be very many such sequences, as a pure result of the combinatorics of a many-component system. The low median probabilities, the large number of such sequences, and the considerable uncertainties in the associated failure rates may very well combine to give important aggregate effects. Such effects were excluded from appearing in the WASH-1400 calculations by virtue of essentially arbitrary choices of small uncertainty ranges for common-mode failure rates, as I will discuss later.

(3) There is no plausible way to make an *a priori* check that a listing of failure sequences is inclusive. In the *Reactor Safety Study*, the listing was in fact incomplete, as dramatically demonstrated by the occurrence of the Browns Ferry, Alabama fire, which involved interactive failures of two reactor units during construction of a third. Though certain specific shared systems at the Surrey and Peach Bottom sites were considered in the course of the *Reactor Safety Study*, the WASH-1400 fault-tree analysis does not include general consideration of interactive failures of several reactor units, such as occurred at Browns Ferry.

(4) The results of calculations based on such an inclusive listing would be expected to depend sensitively on the precise nature of the division into sub-systems on which the analysis is based. In other words, the systemic behavior predicted by a stochastic model such as the one employed in WASH-1400 in general depends on the degree of descriptive detail incorporated in the model. For example, an important step in the WASH-1400 probability estimation process involves the use of "event trees" formed out of sub-systems containing many individual components. Event-tree logic is used to reduce the number of failure sequences analyzed in detail to manageable proportions. It is by no means clear that the same short list of dominant failure sequences as follows from the use of event trees would also result from an alternative, admittedly much more tedious, analysis entirely based on individual components.

(5) The basic assumption implicit in making an inclusive listing of failure sequences--that the dynamic failure characteristics of a man-machine system can be reasonably approximated in terms of a series of transitions between well-defined, previously enumerated, states--is at best implausible. The transition rates between states and the characteristics of individual components would be expected to change markedly during the course of an accident, and significant changes must also occur during an ordinary service cycle. The presence of such time-dependent effects greatly increases the degree of difficulty of an absolute probability calculation.

Common-Mode Failures

The major defect of the WASH-1400 treatment of common-mode failures is that it is based on the use of an arbitrary mathematical procedure for assessing common-mode failure rates. Without providing any justification for doing so, the WASH-1400 fault-tree calculation relates common-mode failure rates to statistically observed failure rates for individual components, in such a way as to minimize the resulting estimates of failure rates and also of the associated uncertainty ranges. The effects involved amount to several orders of magnitude, and they are very serious.

I will briefly illustrate the nature of the problem with a simplified example.

Suppose one is presented with a "machine" consisting of an explosive charge and a box containing two pennies. A cycle of the "machine" consists of first shaking the box, and then setting off the explosion if both pennies come up heads. Of course, assuming the pennies are independent, fair coins, the probability of a "failure" (explosion) is 1/4, multiplying two individual "component" failure probabilities of 1/2. On the other hand, if the two pennies are welded together, with their faces in parallel, then they act as one coin and the "totally dependent" failure probability is 1/2. Of course, if the two pennies are connected in an unspecified way, or if some unknown, external force intervenes, then the probability of failure is indeterminate. *Specification of this "common-mode" failure probability therefore requires knowledge regarding the physical failure mechanism linking the system's components.*

In the WASH-1400 procedure for calculating the common-mode failure rate of the analog to such a system, a mean probability for failure would be specified by taking the geometric mean, $\sqrt{\frac{1}{2}\cdot\frac{1}{4}}$, of the polar "totally dependent" and "totally independent" failure probabilities. However, no physical justification for such a construction is ever given; common-mode failure mechanisms remain unspecified.

In this simplified case the range of values of failure probabilities, from 1/4 to 1/2, is unimpressive. However, in the WASH-1400 analysis enormous ranges are involved. For example, in the most crucial failure sequence in BWR's -- involving control rod malfunction -- the spread between the "totally dependent" and "totally independent" probabilities of sub-system failure runs from 10^{-6} to 10^{-12}, and the geometric mean of $10^{-9} = \sqrt{10^{-12}\cdot 10^{-6}}$ is used without physical justification in the fault-tree calculations. There are further examples of such large spreads between the independent and dependent "bounds", mainly involving human errors.

This arbitrary procedure, which is used in WASH-1400 for estimating the failure probabilities of important reactor failure sequences, is neither an acceptable tool for statistical analysis nor does it provide an appropriate framework for judging the reliability of a reactor system.

Alternative Approaches

For the reasons I have discussed, the absolute probability approach does not appear to be feasible, and we are therefore forced to seek alternative methods for assuring and judging the adequacy of present provisions for light water reactor safety.

In this situation there are two general lines of attack which may prove fruitful: (i) employing passive means for limiting accident consequences; (ii) attempting to evaluate nuclear accident probabilities by comparing the reliability of analogous nuclear and non-nuclear safety systems.

A number of passive mechanisms for potential mitigation of potential accident consequences are available, including containment design for controlled overpressure failures, inert containment atmospheres, more remote or undergound siting of nuclear plants and the mandated use of siliceous rather than limestone aggregate in containment construction. Given the potential serious consequences of nuclear accidents, these possibilities deserve serious study and evaluation.

Use of the comparative approach would necessitate detailed comparisons between all similar safety-relevant procedures and equipment in the energy industry generally, including analyses of refinery operation and reliability, of the reliability of coal and oil based energy production equipment, of procedures (welding, testing, inspecting, repairing, etc.) used in constructing heavy equipment (boilers, oil tanker hulls, reactors, etc.), and of personnel training and related matters. One would supplement studies of that kind with in-depth comparisons of nuclear reactor safety features and practices now in use in Europe, Japan and the United States.

In the absence of such analyses, and in view of the inherent difficulties in the absolute probability approach, we are thrown back on using available commercial power reactor experience as a worst case bound for reactor reliability. The current experience bound of roughly one percent per reactor-year for the occurrence of core-melt accidents is not reassuring, and this again emphasizes that further efforts should be made to improve our understanding of accident likelihood.

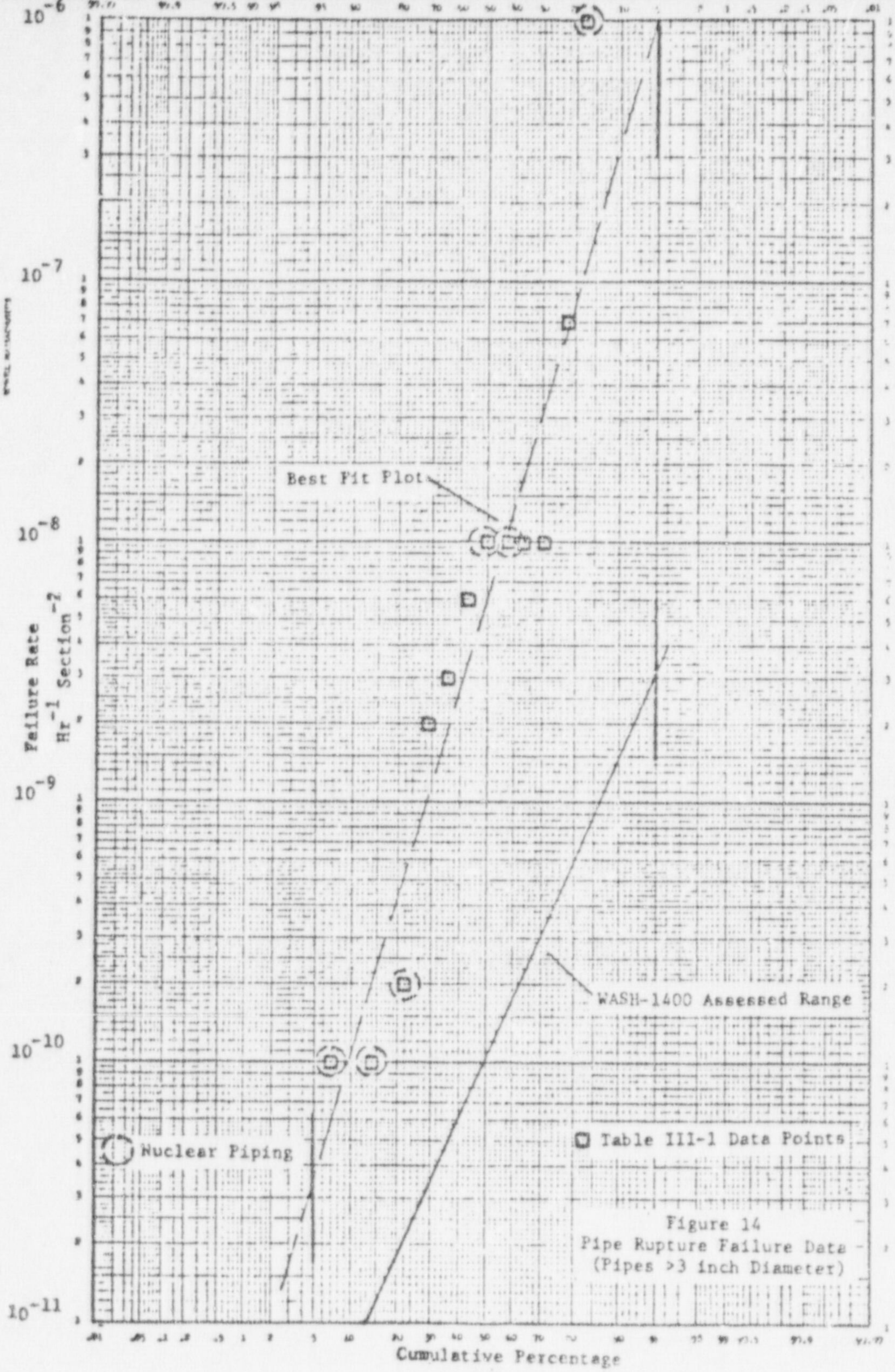

Figure 14
Pipe Rupture Failure Data
(Pipes >3 inch Diameter)

6x10^{-9}

10^{-8}

[6 x 10^{-11} =
10^{-6}]

ERA consultants Report

Carl Setter

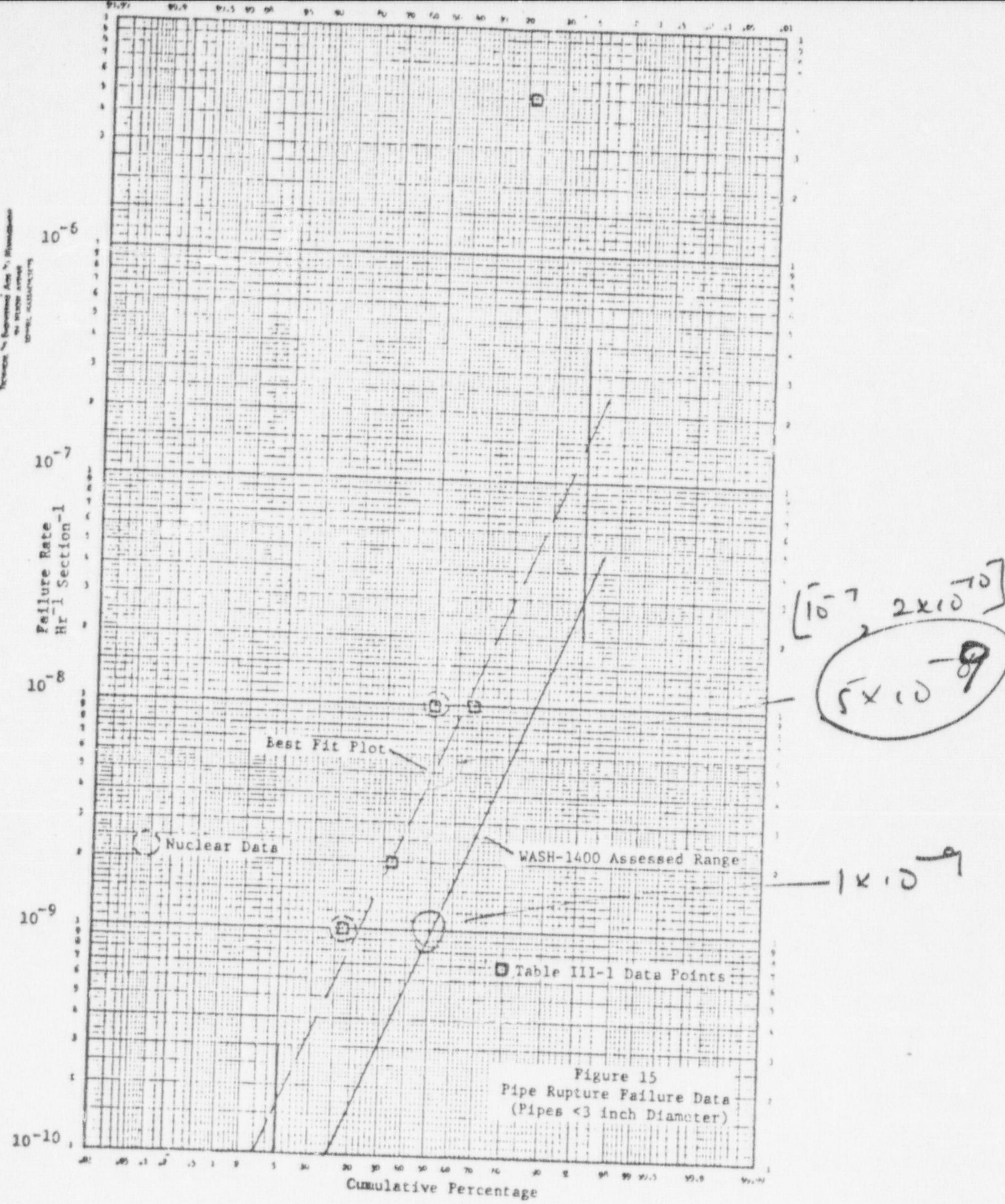

Figure 15
Pipe Rupture Failure Data
(Pipes <3 inch Diameter)

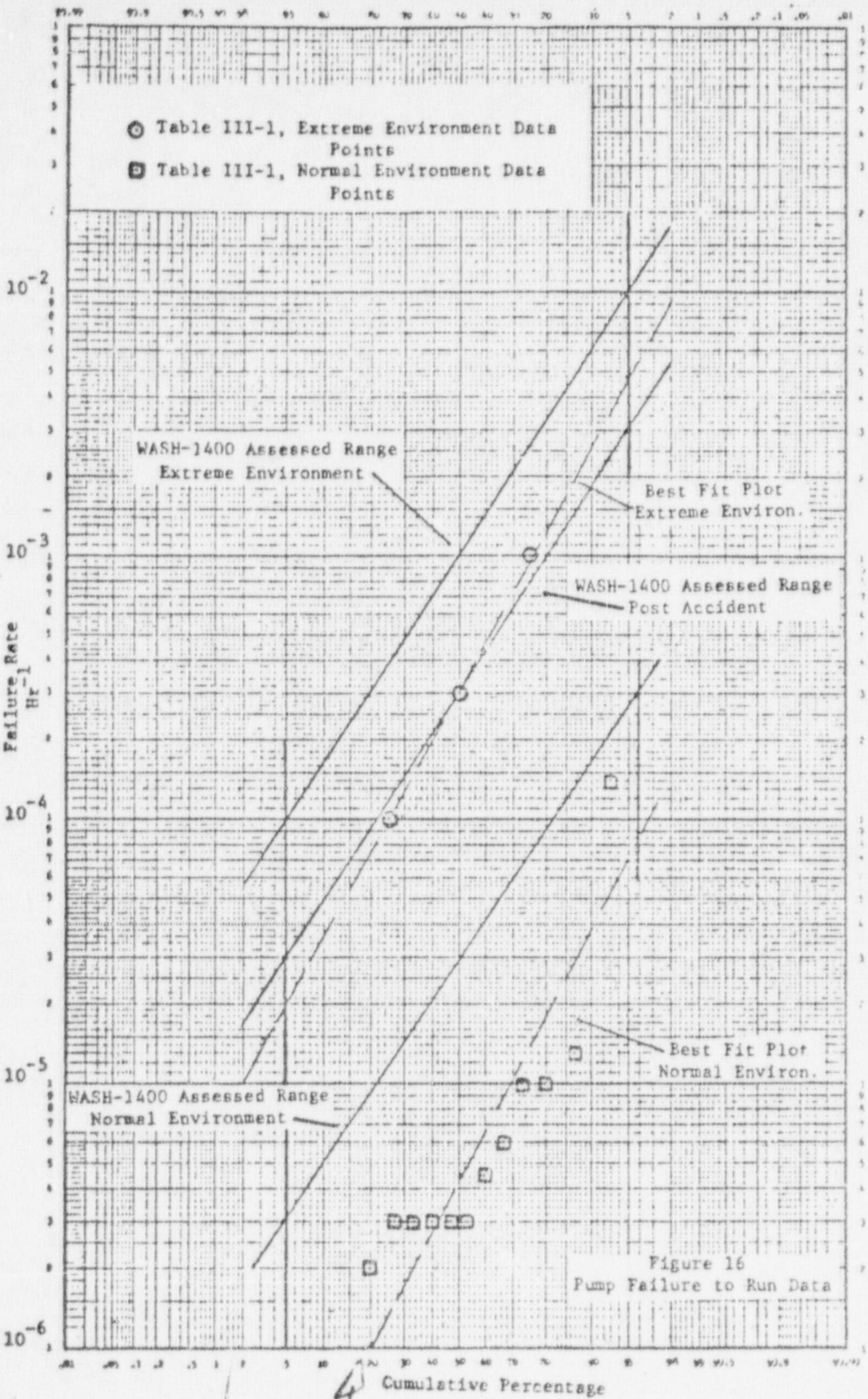

Figure 16
Pump Failure to Run Data

3×10^{-4}

4×10^{-5}

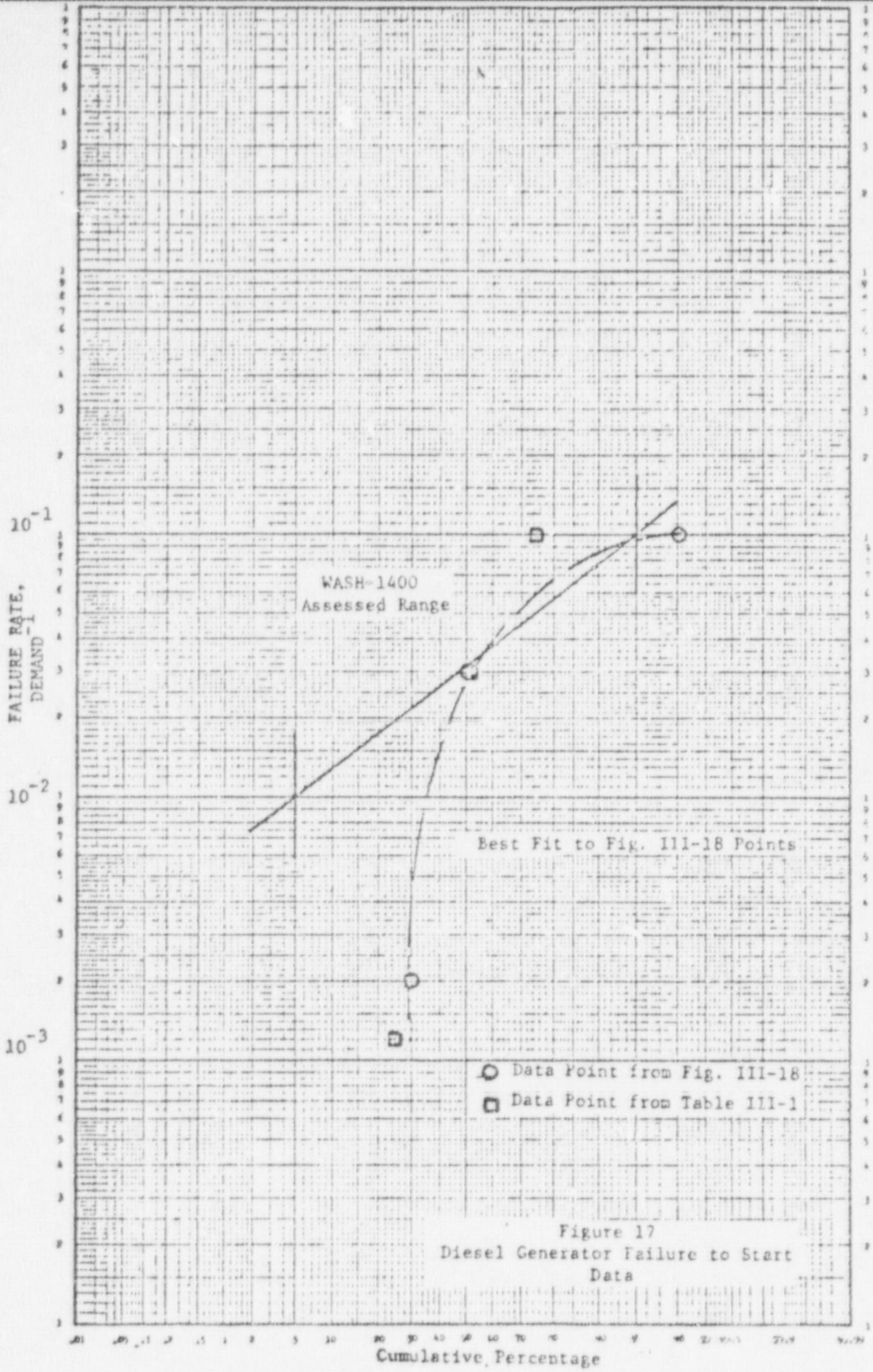

Figure 17
Diesel Generator Failure to Start Data

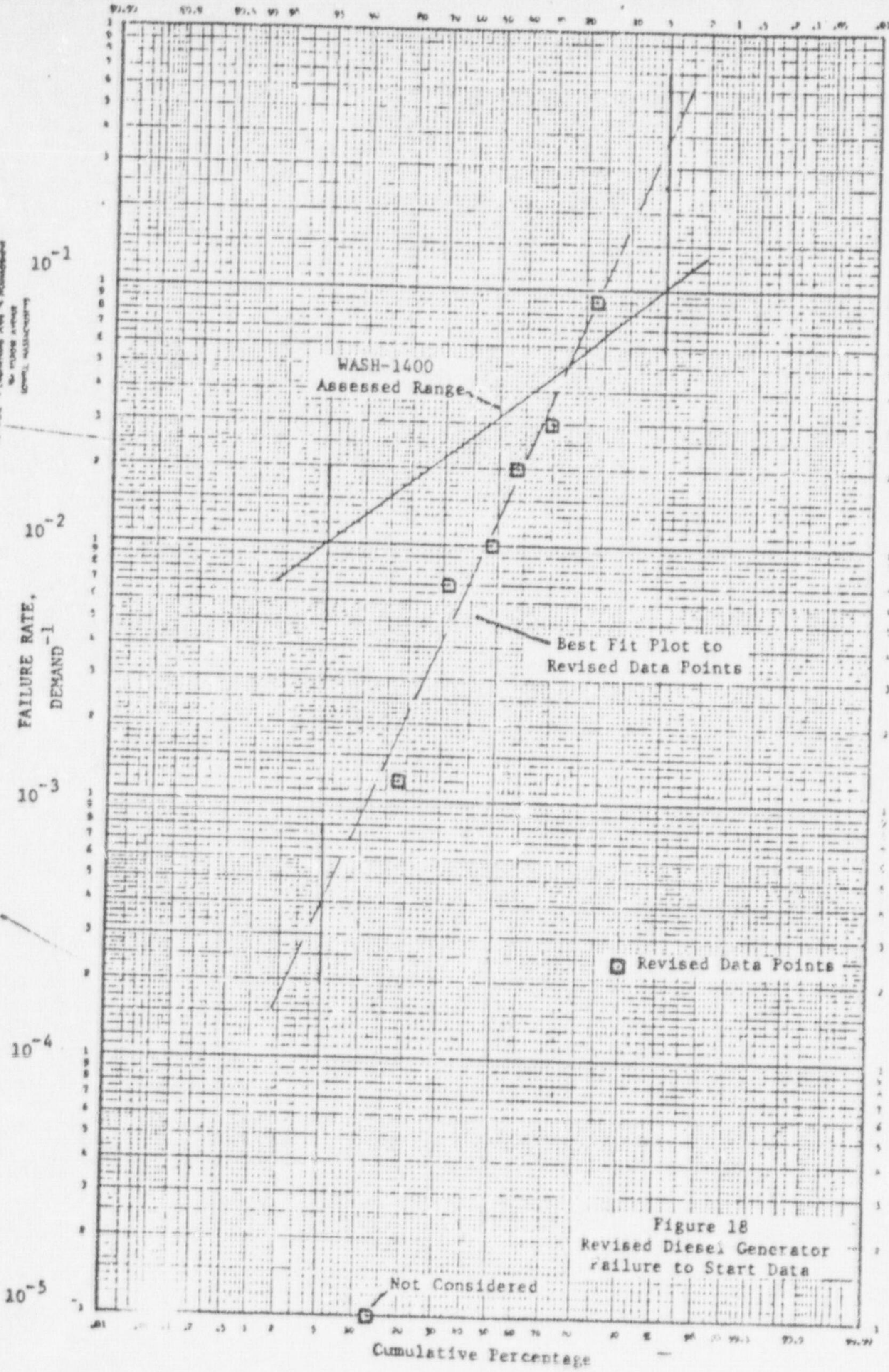

Figure 18
Revised Diesel Generator
Failure to Start Data

5×10^{-2}

10^{-2}

Notes for Discussion with ACRS Working Group on the N.R.C.'s Reactor

Safety Study

Frank von Hippel
Program on Nuclear Policy Alternatives
Center for Environmental Studies
Princeton University

(January 4, 1977)

I. Introduction

It may be useful to you for me to indicate the nature of the background on which I will base my comments on Reactor Safety Study:

i) I participated in the A.P.S. Light Water Reactor Safety Study during the course of which my subcommittee reviewed the treatment in the RSS of the long term consequences of reactor accidents.

ii) I discussed the usefulness of the final Reactor Safety Study report in my testimony before the Subcommittee on Energy and Environment of the House Interior Committee on June 11, 1976.

iii) More recently in response to requests for advice from the New Jersey Department of Environmental Protection, the New York City Commission on Public Health, and the California Energy Resources Conservation and Development Commission my colleague, Dr. Beyea, and I have reviewed the treatment of the short term consequences of reactor accidents in the final RSS study.

The organization of my remarks reflects this chronological development of my involvement with the RSS study.

II. The Long Term Consequences of Reactor Accidents

The A.P.S. review identified certain errors in the treatment in the Draft RSS report of the long term consequences of reactor accidents. Although I am not fully satisfied with the corrections which were made in some cases and have not been able to adequately review the new and much expanded analysis in the final report, I think that the numbers in the final report are probably now of the right order of magnitude.
Figure 1 is my summary of the major errors identified in the Draft RSS report and the corresponding changes in the final report. I made this

transparency to demonstrate in my Congressional testimony the importance of peer review to studies of this type.

III. The Usefulness of the RSS Report for Policy Purposes

The primary purpose of the RSS report was to put the reactor safety issue into perspective. Here I believe that it failed for a number of reasons:

1) It Made Misleading Comparisons of Nuclear Accident Risks to Other Risks:

In Figure 2 I show the famous comparison of reactor hazards with natural events in the Executive Summary. This comparison appears to show that the hazard to the U.S. population from 100 reactors is of the same order of magnitude as that from meteors.

Unfortunately the comparison made here excludes from consideration all consequences of reactor accidents other than short term fatalities. In Figure 3 I show for the same probability level where Figure 2 shows 10 fatalities what some of the other consequences of reactor accidents were calculated to be in the RSS report. It appears to me that to say that 10 people died would be to inadequately report such events.

2) It Did Not Adequately Acknowledge the Great Uncertainties of the Probability Calculations:

My work in reviewing this aspect of the RSS report is rather thin. There are other people at this meeting who have examined these questions in greater depth than I. I would like, however, to make a few remarks:

i) For a system as complex as a nuclear power plant it is difficult to believe that the probabilities of certain accident consequences can be calculated within a factor of 5 as asserted in the RSS Executive Summary.

In the final RSS report* the graph shown in Figure 4 is offered in substantiation of the factor of 5 error. It shows the ratio of observed failure rates to predictions for about 45 systems as compiled by A.E. Green and A.J. Bourne of the National Centre of Systems Reliability in England. One of the members of our group, Mr. Robert Weatherwax, is a reliability engineer and was interested in this graph so he wrote to Mr. Green and asked him for a list of the systems to which these points correspond. Dr. Bourne wrote back providing references to those cases where the numbers had been published. Figure 5 shows a compilation of this data.** It will be seen that the systems are mostly simple pieces of electronic or mechanical equipment where the failure rates range from 0.13 to 220 per year. It would appear to me that it is hardly justified to take success within a factor of five in predicting the failures of a system which is expected to fail of the order of once a year and expect the same success for events involving systems containing thousands of components of this type in intimate interactions with human beings and each other and which are predicted to have probabilities of failure of the order of once in ten thousand years.

ii) There appears to be areas in which the RSS group decided that a particular accident initiator would contribute little to the total probability of melt-down accidents without adequate analysis. Let me give as an example the case of earthquakes. The estimates made in the RSS report*** of the probability of failing two safety systems were calculated on the basis of numbers given in a paper written by N.M. Nemark. This paper was not

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*Appendix XI, p. XI 3-4.

**References: (1) A.J. Bourne, "Reliability Assessment of Technological Systems," 1971; (2) A.R. Eames, Nuclear Engineering, March 1976; (3) G. Hensley, Measurement and Control, April 1968; (4) A.J. Bourne, UKAEA Report SRS/GR/5.

***Main report, Section 5.4.1, pp., 66-68.

published so I wrote to the RSS group to get a copy. The paper turned out to be draft of a report prepared for the N.R.C. Office of Regulatory Research but apparently not completed. The draft is 24 pages long - 12 pages of text. The first 9 pages of the text contains general discussion of such matters as the seismic responses of a building in Los Angeles, the response to an underground nuclear test of two small reinforced concrete buildings, the damage to various buildings and a brick chimney observed in some recent California earthquakes, the relationship between ground accelerations and the Mercalli earthquake intensity scale, and the appropri ateness of the N.R.C. design response spectrum. Finally in the last three pages there is the analysis from which the RSS drew its numbers. It turns out not to be an analysis at all but a "guesstimate" based on judgements of the relative conservativism of the seismic designs of conventional structures and nuclear power plants. To reproduce these three pages would have taken up about one page in 2400 of the final RSS report. By not publishing it the RSS made the basis of its conclusions on accident initiation by earthquakes relatively inaccessible to peer review. I would be interested to learn the judgement of the ACRS as to whether these three pages of Nemark's report provide an adequate basis for the RSS conclusion that earthquakes are not important as accident initiators.

iii) In some analyses the RSS study has not addressed obvious questions. An example here is the analysis of the implications of the Brown's Ferry fire.* The purpose of this discussion was to justify the neglect of fires as accident initiators. The analysis is extremely narrowly defined, however, and ignores many of the complexities and potentialities of the real event. For example, when the possiblity is

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*Appendix XI, pp. XI 3-50 through XI 3-58.

considered that control over the pressure relief valves could have been lost valves earlier in the accident than actually occurred, rendering it impossible to maintain the water level over the core with the single operational high pressure injection system, it was assumed that the operators would know of four possible repairs that they could make and that they would work on them all efficiently and simultaneously. In this way it was possible to calculate that, although each of the repairs would take a time of the same order as the two hours before the water level could have fallen below acceptable levels, the probability of failing to make all of these repairs in this time was only one in three hundred although in the actual event it took four hours before the first system was repaired. This efficiency effort was assumed to occur despite the fact that a fire was raging and that, at one stage during the accident, the control room was filled with smoke to the point where respirators were required. These assumptions would appear to fly in the face of the conclusions of the discussion of human reliability in Section 6 of Appendix III - yet there is no discussion of these questions. There is also the question of whether, at a reactor where construction of neighboring units was not in progress - ar at night - whether all the expertise assumed to be present would in fact have been.

It is for such reasons that I am extremely skeptical about the accuracy of the probability calculations made in the RSS report and that I have not changed my subscription to the conclusion in the APS Study report that "based on our experience with problems of this nature involving very low probabilities, we do not now have confidence in the presently calculated absolute values of the probabilities."

So I would say that, while the information in the RSS report could be used to help put the reactor safety controversy into perspective, the actual effort by the RSS group to do so is so misleading in terms of its comparisons to other hazards and the certainty which it claims for its probability calculations as to make the report useless for policy-making purposes in its present form.

IV. Information for Siting Decisions and Contingency Planning

Recently Dr. Beyea and I have been asked to provide advice to the State of New Jersey and California, which are trying to develop nuclear reactor siting policies, and the City of New York, which is trying to develop contingency plans in case of a reactor accident at Indian Point.

A constant question has been: "How far away from a major nuclear reactor accident would the population be in danger?" This information is at the core of the RSS group's calculations so we started by seeing if we could extract it from the report. All we found was three figures in Appendix VI which are reproduced in Figures 6, 7 and 8. Figures 6 and 7 show the one day doses to different organs of the population downwind for an unspecified accident for two different weather conditions. Figure 8 shows the corresponding probabilities of death for different distances downwind. Under the worse of the two weather conditions (F type weather) the lethal distance is about eight miles.

Does this mean that the population is safe beyond this distance? To find out we had to reproduce the RSS consequence computer program and test the sensitivity of its results to the various parameters. The following figures show results obtained by Dr. Beyea. As the curves in Figure 9 show, the reuslts are quite sensitive to these parameters within the range of their uncertainties as given in the RSS report and, for particularly unfortunate combinations of initial plume height and deposition velocity, seven day bone marrow doses can reach 250 rem up to 40 miles downwind. Figure 10 shows the upper limit envelope to the one day and seven day doses calculated for time invariant weather. It will be seen that evacuation beyond 20 miles could be important. Figure 11 shows a contour plot in the deposition plume height parameter space for the 1 day bone marrow dose at 17 miles. The horizontal limits of the box are defined by the limits on the deposition velocity given in the RSS study. The vertical scale assumes plume rises of 0 to 250 meters. In the RSS study the plume rise was calculated from the heat content of the released gases using a smoke-stack formula but we are doubtful about the applicability of this formula - as was the RSS group in the first draft of their report - so we treat plume rise as a parameter.

If we treat every point within the parameter box as equally probable, Figure 12 shows the 10 percent probability limit curves and the median curve for the seven day dose for F type weather. It will be seen that the accident consequences are not adequately represented by the median curve - the full parameter space has to be explored.

V. Recommendations

In conclusion I think that it is evident that I believe that the RSS is provocative but not yet truely useful for policy-making purposes.

In the short term I think that the most useful thing which could be done would be to withdraw the Executive Summary and to write a new one which more accurately reflects the findings of the Reactor Safety Study and their uncertainties, and which puts them into a more balanced perspective.

Then I think that outside reviews should be commissioned of certain parts of the analysis. I would include on the list: the treatment of earthquakes, the treatment of the Brown's Ferry fire, the handling of uncertainties, the inhalation exposure - dose coefficient, the treatment of the special hazards associated with accidents at sites such as Indian Point and Zion, the treatment of radioactive contamination of water bodies, the assumptions made concerning the possibilities for decontamination of large areas, the assumptions made concerning the evacuation of people 10 to 40 miles away, the treatment of the degradation of the performance of emergency equipment under accident conditions, and the relevance of the findings for the two reactors which were studied to the other 98 reactors to which they were extended. I am sure that other people would have other items to add to this list -- as would I if I had more time.

Finally I think that there are a number of areas where additional work will have to be done. These would include: studies of the type that Jan Beyea has done to facilitate the development of siting criteria and contingency plans, comparing the long term effects of reactor accidents to those of potential accidents involving chemical carcinogens and mutagens, developing an understanding of the problems which would be encountered if it were necessary to decontaminate large areas, a detailed study of the vulnerability of a plant such as Diablo Canyon to earthquake damage, calculating strong upper bounds on the probabilities of large consequence accidents, and examination if there are any design changes such as a filtered pressure release system on containments which could substantially reduce these upper bounds.

OF THE DRAFT RASMUSSEN REPORT

EFFECT	CHANGE IN FINAL REPORT
WHOLE BODY RADIATION DOSE TO POPULATION DOWNWIND (CAUSE OF CANCER AND GENETIC DEFECTS)	INCREASED TENFOLD
RADIATION DOSE TO LUNGS OF POPULATION DOWNWIND (OMITTED IN DRAFT AS CAUSE OF CANCER)	NOW LARGEST CAUSE OF CANCER
THYROID TUMORS	INCREASED THREEFOLD
NATURAL DURATION OF RADIOACTIVE LAND CONTAMINATION	INCREASED TENFOLD
WATER CONTAMINATION BY STRONTIUM-90	INCREASED ONE THOUSANDFOLD

FIGURE # 1

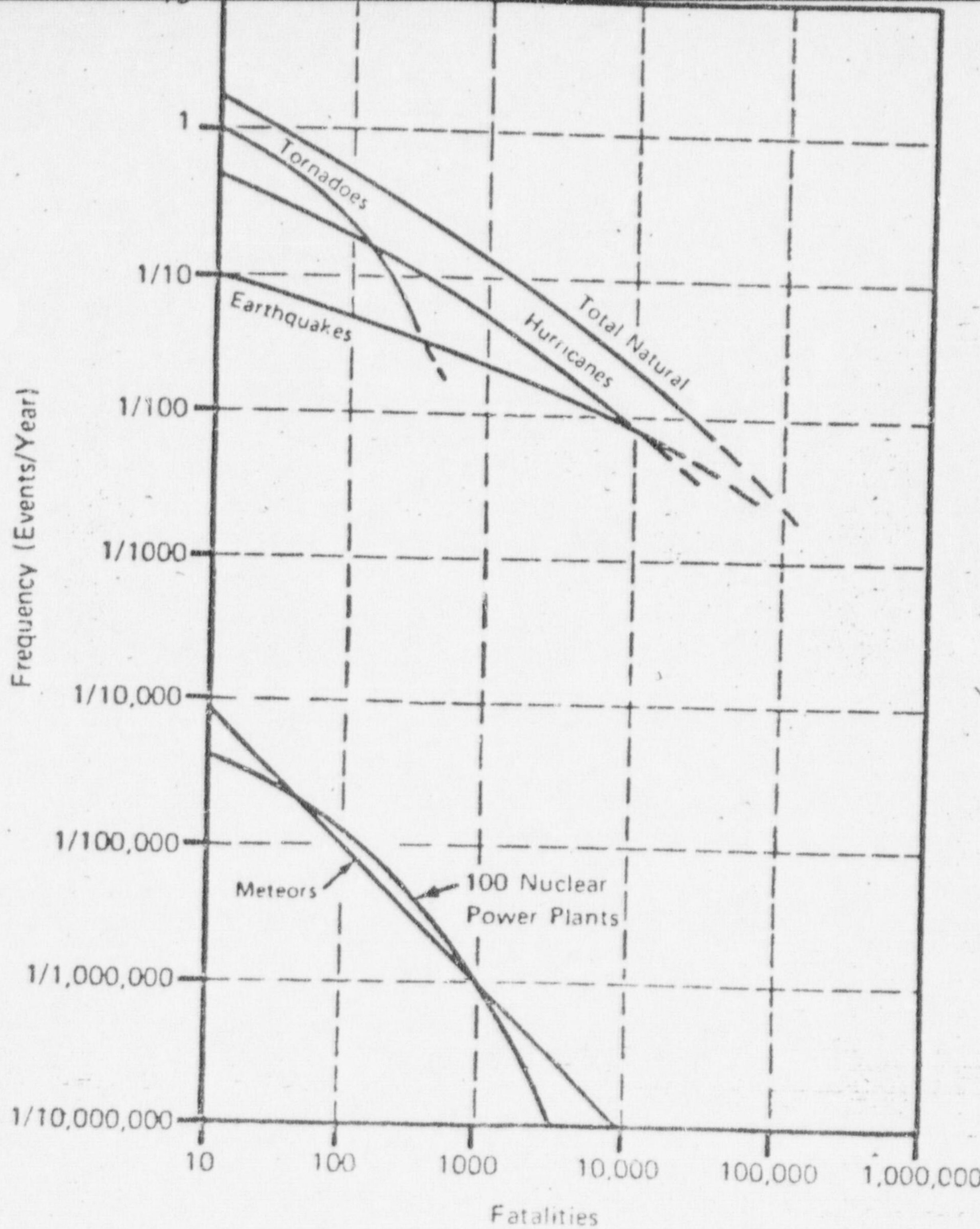


FIGURE 1-2 Frequency of Fatalities due to Natural Events

Figure 2

OTHER CONSEQUENCES OF NUCLEAR REACTOR ACCIDENTS WHICH HAVE THE SAME LIKELIHOOD AS 10 SHORT TERM FATALITIES

7,000 CANCER DEATHS

4,000 GENETIC DEFECTS

60,000 THYROID TUMOR CASES

3,000 SQUARE MILES OF LAND CONTAMINATED WITH RADIOACTIVITY ABOVE ACCEPTABLE LEVELS

MASSIVE WATER CONTAMINATION

(BASED ON APPENDICES VI AND VII OF THE RASMUSSEN REPORT)

FIGURE 3

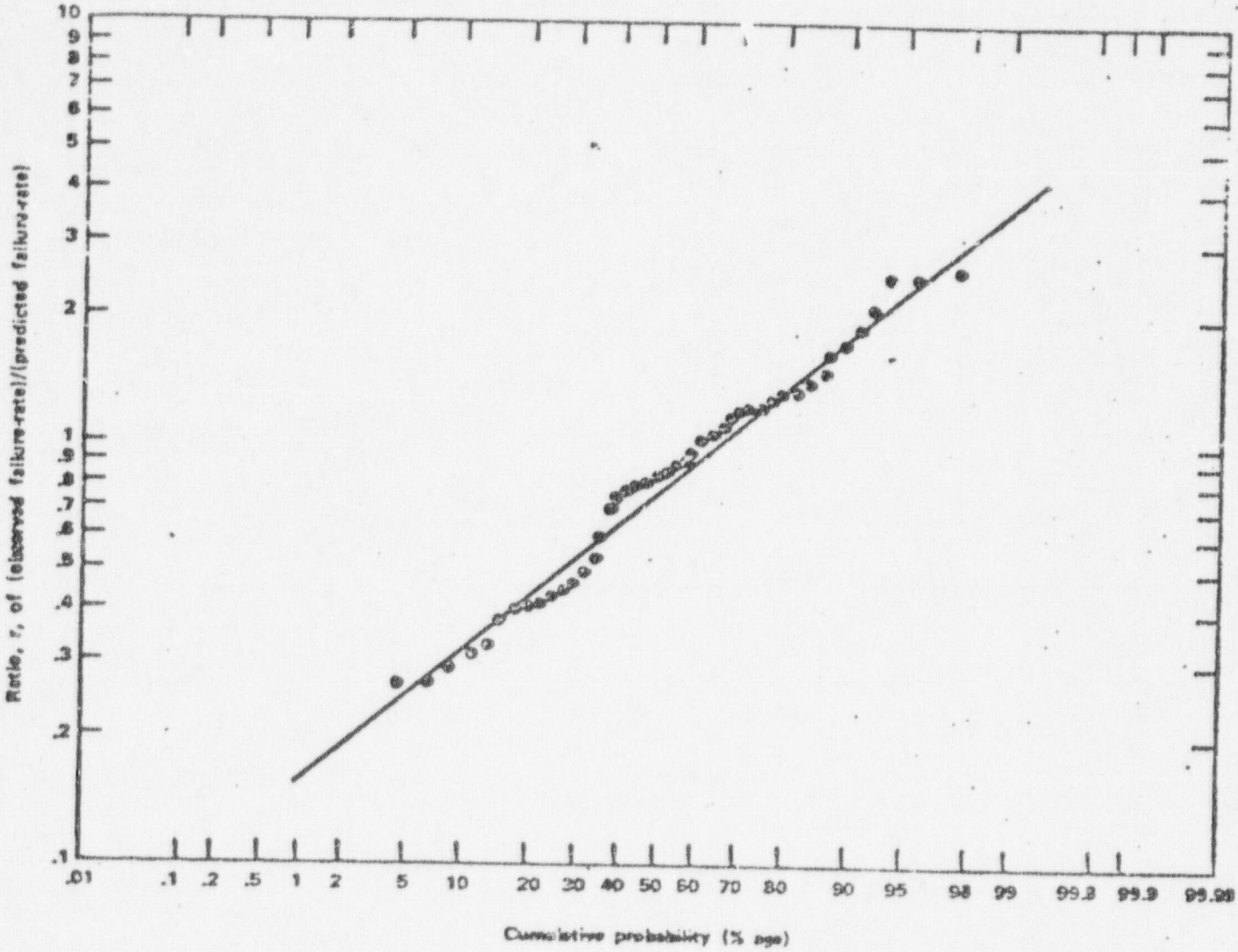

Ratio, r, of (observed failure-rate)/(predicted failure-rate)
Cumulative probability (% age)

Figure 4

Pneumatic transmitting flowmeter	0.6	(1)
Pressure switch	0.14	(1)
Oxygen analyser	2.5	(1)
Ring balance controller	0.38	(1)
Pneumatic valve	0.25	(1)
Temperature trip amplifier, Type A.	2.6	(1)
Temperature trip amplifier, Type B	1.7	(2)
Criticality monitor, Type B	0.34	(1)
Criticality monitor, Type C	0.13	(1)
Pulse amplifier	5.0	(1)
Pulse discriminator	7.1	(1)
Gamma monitor	0.85	(1)
Radiation monitor	0.34	(2)
Pulse channel	17.4	(2)
Log-period meter	1.8	(2)
Differential pressure amplifier	0.76	(2)
Three-term controller	0.38	(2)
Pneumatic relay	0.17	(2)
Ring balance meter	0.15	(2)
Variable area flowmeter	0.68	(3)
Gas analyser	2.5	(3)
Criticality monitor	0.39	(3)
Diecasting machine:		
Tool system	220	(4)
Tool closing and locking system	21	(4)
Metal injecting system	111	
Operating control system	60	(4)
Ancillary systems	67	(4)

Figure 5

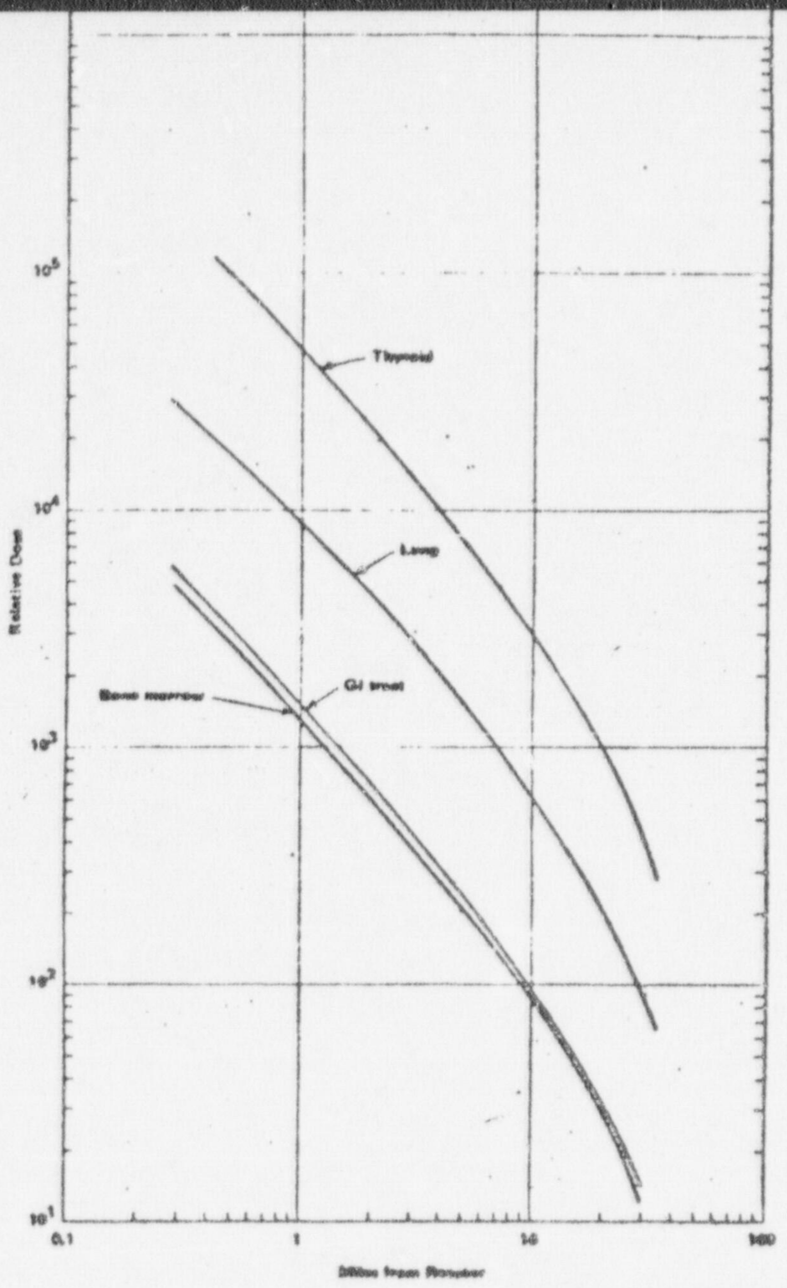


FIGURE VI 13-6 Total organ doses versus distance from reactor for hypothetical weather; stability A, wind speed of 0.5 m/sec. Thyroid dose = 1-day ground + external cloud dose + 30-day inhalation dose; Lung dose = 1-day ground + external cloud dose + 1-year inhalation dose; GI tract dose = 1-day ground + external cloud dose + 7-day inhalation dose (the GI tract dose is the dose to the regenerative cells of the lower large intestine); bone marrow dose = 1-day ground + external cloud dose + ½(7-day inhalation + 30-day inhalation dose)

Figure 6

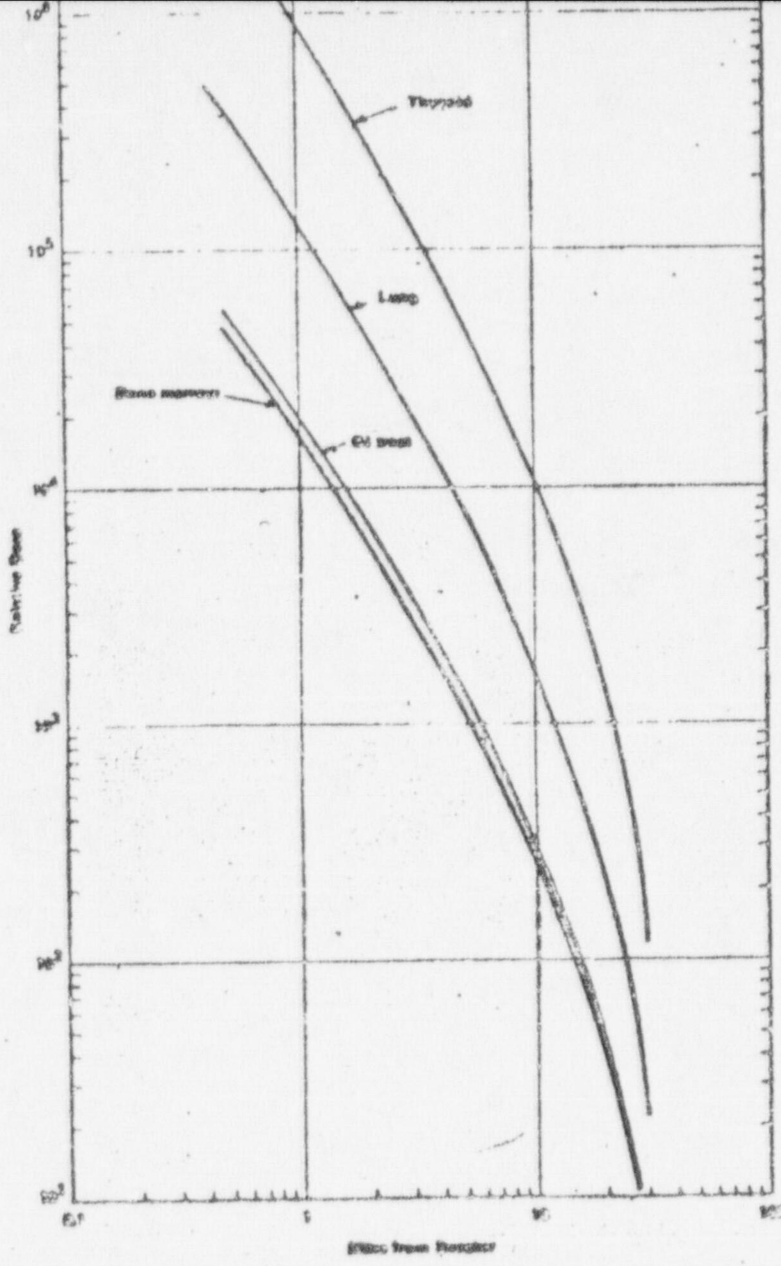


FIGURE VI 13-8 Total organ doses versus distance from reactor for hypothetical weather; stability F, wind speed = 2.0 m/sec. Thyroid dose = 1-day ground + external cloud dose + 30-day inhalation dose; Lung dose = 1-day ground + external cloud dose + 1-year inhalation dose; GI tract dose = 1-day ground + external cloud dose + 7-day inhalation dose (the GI tract dose is the dose to the regenerative cells of the lower large intestine); bone marrow dose = 1-day ground + external cloud dose + ½(7-day inhalation + 30-day inhalation dose)

Figure 7

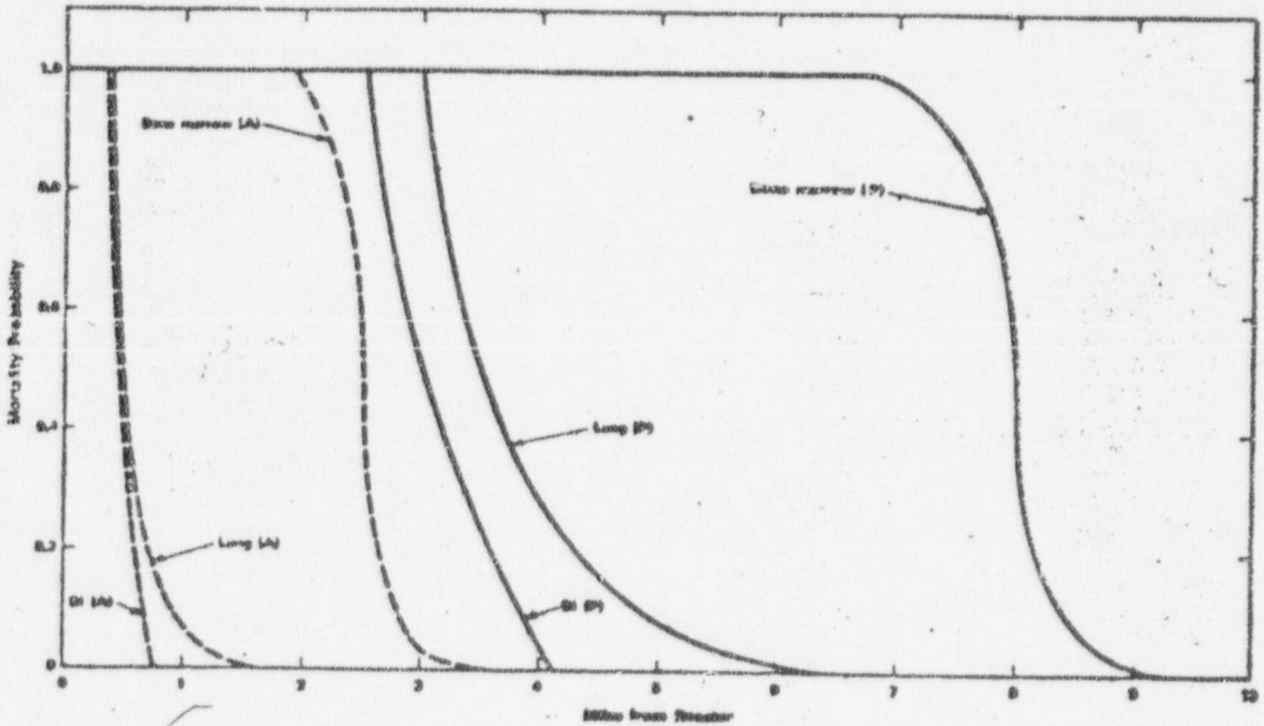
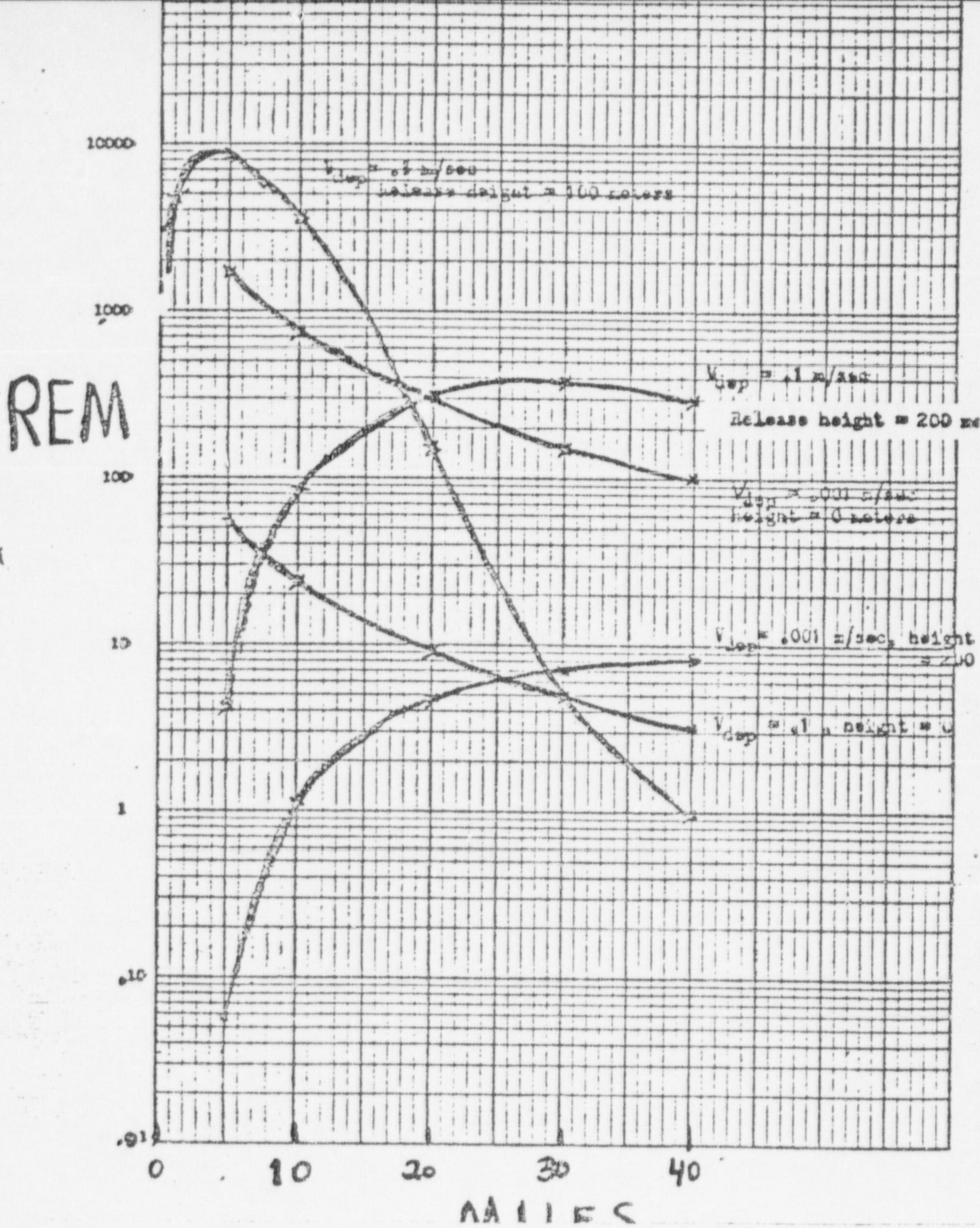


FIGURE VI 13-7 Mortality probability for an affected population versus distance from reactor for two hypothetical weathers: stability category A, wind speed = 0.5 m/sec; stability category F, wind speed = 2.0 m/sec.

Figure 8


REM
10000
1000
100
10
1
.10
.01
0
10
20
30
40
Release height = 200 m
.001 m/sec, height
= 200

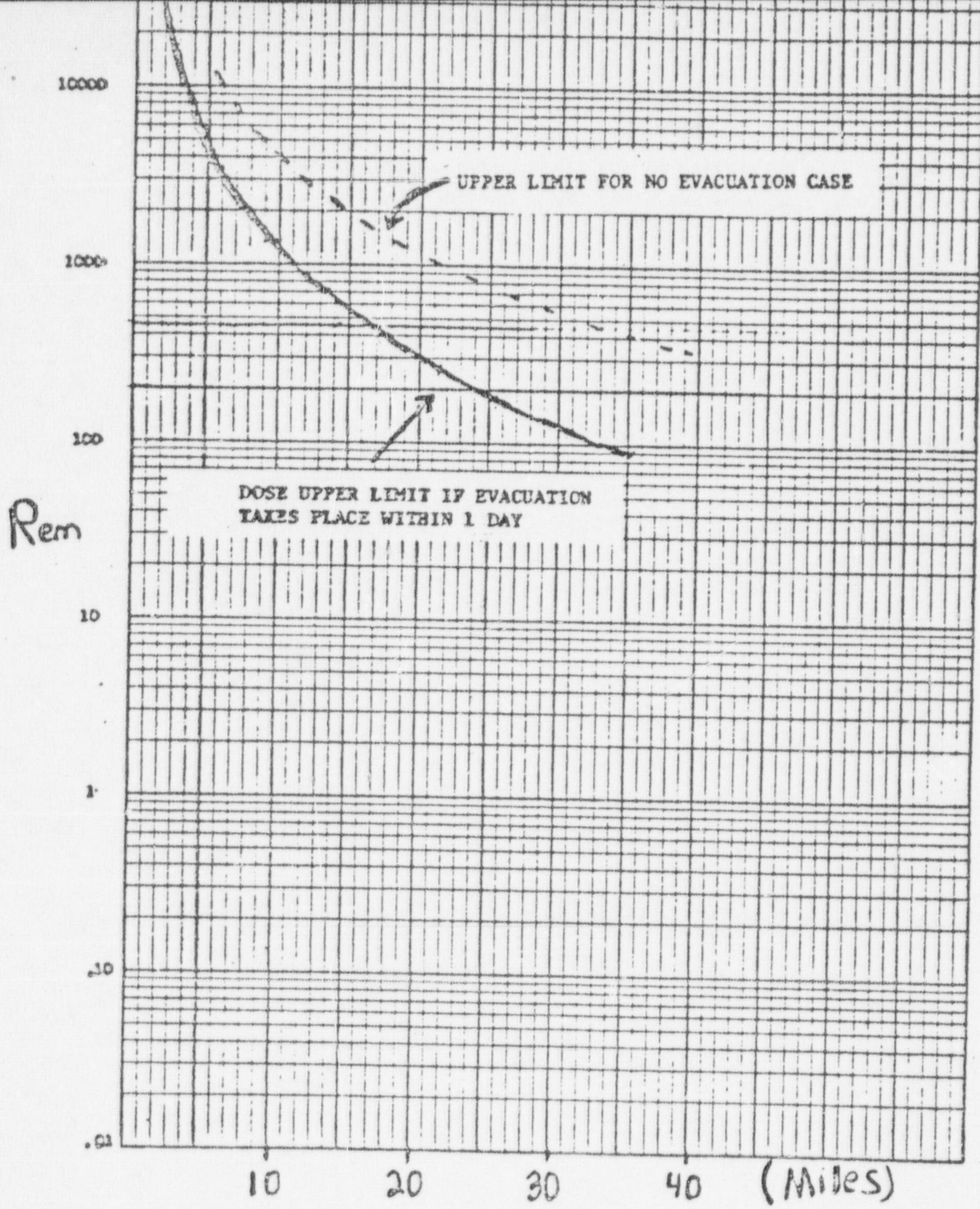
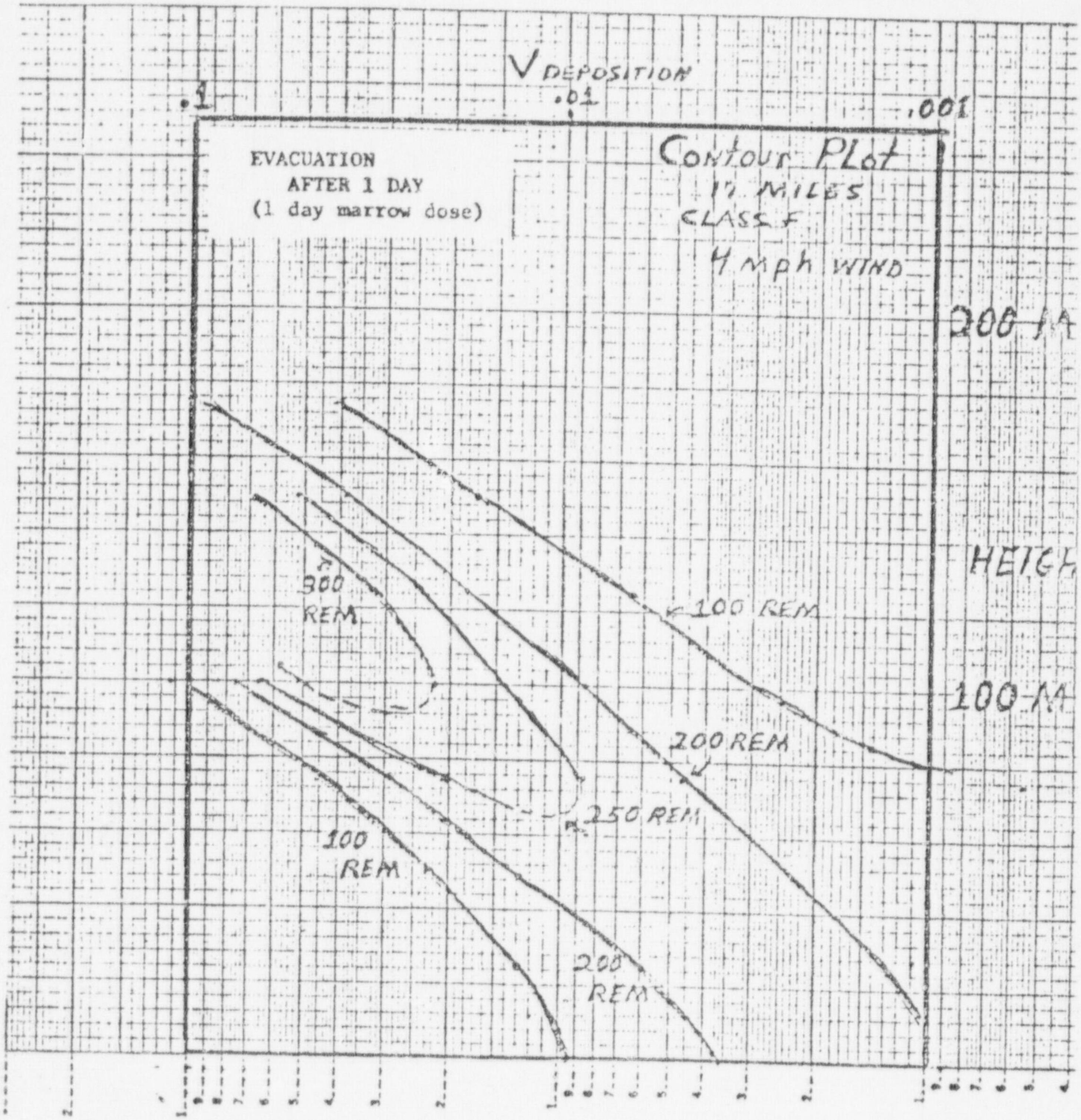
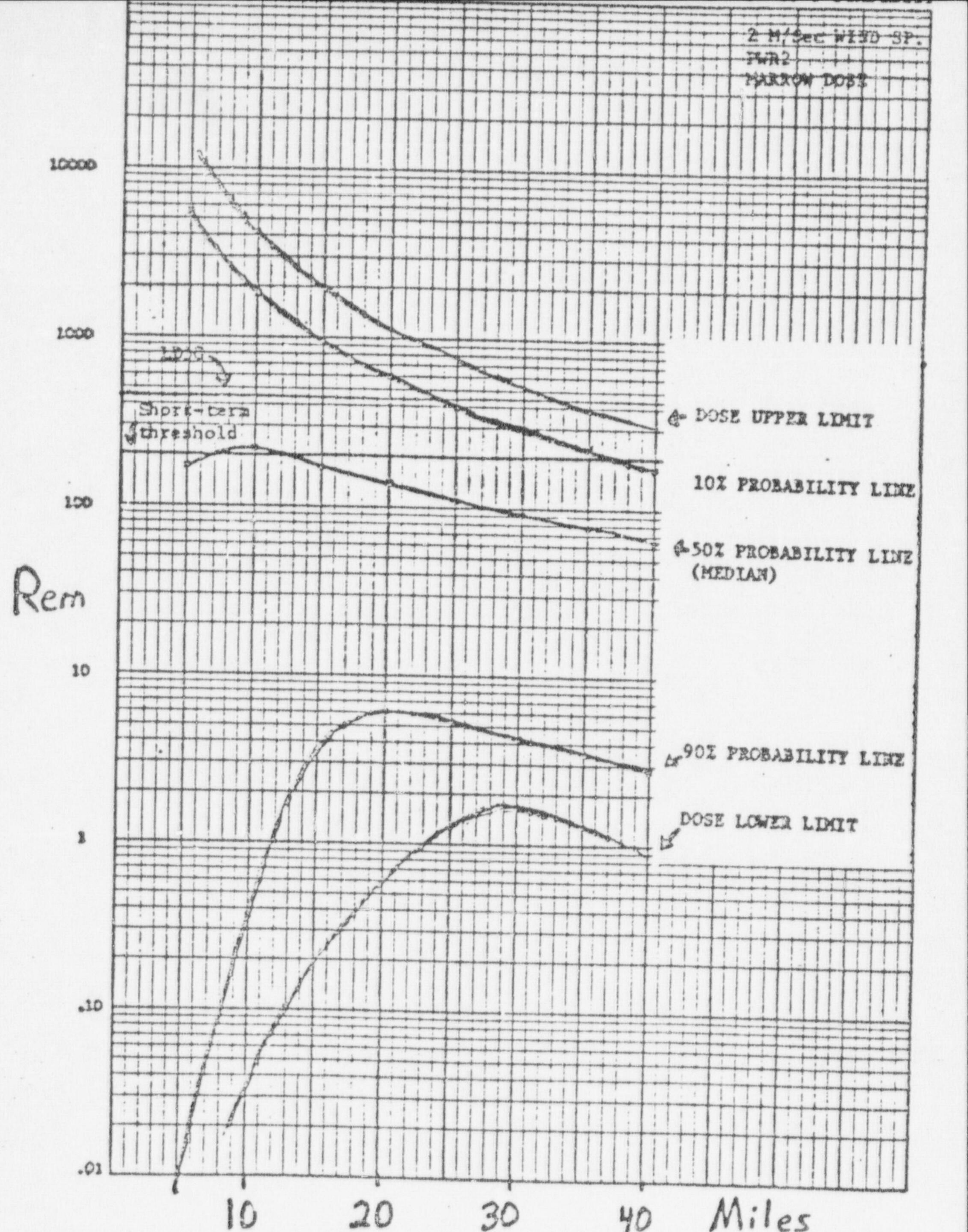

10000
1000
100
10
1
.10
.01
Rem
UPPER LIMIT FOR NO EVACUATION CASE
DOSE UPPER LIMIT IF EVACUATION
TAKES PLACE WITHIN 1 DAY
10
20
30
40
(Miles)

Fig 10



V DEPOSITION
.1
.01
.001
EVACUATION
AFTER 1 DAY
(1 day marrow dose)
Contour PLot
17 MILES
CLASS F
4 mph WIND
200 M
HEIGH
100 M
300
REM
100 REM
200 REM
250 REM
100
REM
200
REM



2 M/Sec WIND SP.
PWR2
MARROW DOSE
10000
1000
100
10
1
.10
.01
Rem
LDSO
Short-term threshold
DOSE UPPER LIMIT
10% PROBABILITY LINE
50% PROBABILITY LINE (MEDIAN)
90% PROBABILITY LINE
DOSE LOWER LIMIT
10
20
30
40
Miles

Additional

Documents Made Available to the Reactor Safety Study Working Group ~~for~~ during the January 4, 1977, Meeting*

1. Estimate of the Cancer Risk Due to Nuclear-Electric Power Generation, October 1976, U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, D.C. 20460 (ORP/CSO-76-2)

2. 40 CFR 190 Environmental Radiation Protection Requirements for Normal Operations of Activities in the Uranium Fuel Cycle, Final Environmental Statement, Volumes I & II, U.S. Environmental Protection Agency, Office of Radiation Programs (EPA 520/4-76-016)

3. Transmittal of Comments on the Nuclear Regulatory Commission's revised draft environmental impact statement for the Atlantic Generating Station, Units 1 & 2 (Construction Stage), Ocean and Atlantic Counties, New Jersey, from the United States Department of the Interior, December 29, 1976 (ER 76/1025)

√4. Transmittal of Comments on the Nuclear Regulatory Commission's Part III of the draft environmental statement and draft liquid pathway generic study on the proposed manufacture of floating nuclear power plants (STN 50-437), Duval County, Florida, from the United Stated Department of the Interior, December 23, 1976 (ER 76/1010)

5. Yellin, J. "The Nuclear Regulatory Commission's Reactor Safety Study: reply." The Bell Journal of Economics,

6. Wilson, R. "Yellin's Review of the Reactor Safety Study." The Bell Journal of Economics, Vol. 7, No. 2, (Autumn 1976), pp. 701-710.

* One office copy is available for inspection at the ACRS office.

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