UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
James P. Gleason, Chairman
Dr. Oscar H. Paris
Frederick P. Shon

In the Matter of	
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2)	Docket Nos. 50-247-SP 50-286-SP
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	22 March 1983

UCS/NYPIRG TESTIMONY OF STEVEN C. SHOLLY
ON COMMISSION QUESTION FIVE

Q.01 Would you state your name, position, and business address?

A.01 My name is Steven C. Sholly. I am a Technical Research Associate with the Union of Concerned Scientists, Dupont Circle Building, 1346 Connecticut Avenue, N.W., Suite 1101, Washington, D.C. 20036.

Q.02 Have you prepared a statement of professional qualifications?

A.02 Yes; my statement of professional qualifications is attached to this testimony.

Q.03 What is the purpose of this testimony?

A.03 This testimony addresses Commission Question Five, which states:

D503

"Based on the foregoing [responses to Commission Questions One through Four], how do the risks posed by Indian Point Units 2 and 3 compare with the range of risks posed by other nuclear power plants licensed to operate by the Commission? (The Board should limit its inquiry to generic examination of the range of risks and not go into any site-specific examination other than for Indian Point itself, except to the extent raised by the Task Force [on Interim Operation of Indian Point].)"

Q.04 Is "risk" divisible into component parts? Explain.

A.04 Yes. Risk from the operation of nuclear power plants in general arises from two groups of risk characteristics. The first group of risk characteristics arises from the nature of the reactor site and surrounding environment, and includes meteorology, population distribution and magnitude, and emergency response characteristics. The second group of risk characteristics arises from the design and operation of the plant, and includes the design of the reactor and balance-of-plant (BOP) systems, operator training, maintenance practices, emergency procedures, and management capabilities. It should be noted that the "Task Force" report cited in Commission Question Five divided its preliminary discussion of risk into these groups of risk characteristics [NUREG-0715, pages 7-34].

Following publication of the Reactor Safety Study in 1975, the NRC Staff took the position that there were no statistically significant risk differences between the Surry and Peach Bottom reactors, even though they were different designs, constructed by different architect-engineers, and operated by different utilities. Based on this experience, the Staff did not anticipate that there would be "large differences" in risk among the various designs of light-water reactors covered by the Reactor Safety Study [Gossick, response to Question 4].

Since the Reactor Safety Study, however, it has been generally established by a number of additional probabilistic risk assessments (PRAs) that plant-to-plant differences in design and operation can result in significant differences in the frequency of accidents and the particular event sequences which dominate risk [NUREG/CR-2802, Main Report, page 1; NUREG/CR-1659, Vol. 3, page 6-32; NUREG-0715, page 6].

Even if two reactors are identical in design and maintenance and operated similarly, these two reactors may present differences in risk due to the impact of external events on risk [NUREG/CR-2453, page 70]. Moreover, the Sandia siting study established that site-to-site population and emergency response factors can have a significant impact on risk [NUREG/CR-2239, pages 2-32 through 2-52]. Thus, in combination, we now know that site and design/operation risk characteristics combine to produce a "risk profile" for a reactor/site combination which is unique to each reactor at each site.

Q.05 Once established, can risk profiles for specific reactors at specific sites change?

A.05 Yes. The "risk profile" of a given reactor/site combination may change with time due to: (a) improved knowledge about the site, the reactor, and/or the operation of the reactor; (b) changed conditions (e.g., end-of-operating-life wear-out of components, changing operator training and/or maintenance practices, and changes in site population magnitude and/or distribution); and (c) analysis of additional operating experience which may reveal hitherto unknown safety problems or that actual event frequencies are different than previous estimates based on a more limited data base [Griesmeyer and Okrent, pages 6-7]. Wear-out failures may present a particular problem for systems and components which are expected to last for the life of the reactor since wear-out failures may not be well-represented in the existing data base [NUREG/CR-2453, page 35].

In addition, there will be some reactors for which more than one PRA study will be performed (indeed, Calvert Cliffs Unit 2, Oconee Unit 3, Browns Ferry Unit 1, and Sequoyah Unit 1 will soon fall into this category). It is reasonable to expect that results from PRAs of the same plant by different analysts can have large differences [Griesmeyer and Okrent, page 5]. These differences can also affect relative risk rankings and risk profiles for given reactor/site combinations.

The particulars of the risk profile changes cannot always be predicted, nor can it necessarily be determined in advance in what direction the

change will occur. Thus, a relative risk ranking of reactor/site combinations made on the basis of current knowledge may be changed in the future.

Q.06 Are there any differences in the nature of our knowledge about how site and design/operation risk characteristics influence risk?

A.06 Yes. Our knowledge about how site risk characteristics influence risk is much better understood than our knowledge of how design/operation risk characteristics influence risk. This is due in part to the fact that site risk studies of one form or another have been performed for 91 reactor sites in the U.S. (including some where the planned reactors have since been cancelled). Such a study has been performed in the Sandia siting study, and other site-specific consequence studies have been published in NEPA environmental impact analyses performed by the NRC Staff over the last three years. In addition, there are a large number of sensitivity studies which explore how factors related to meteorology, emergency response, and site demographics affect accident consequences (e.g., NUREG/CR-2239, SAND78-0556, and SAND79-0095).

A important realization regarding our knowledge of how site factors affect risk is that some of the more significant uncertainties are coherent with respect to risk, i.e., such uncertainties will affect all sites in the same manner, thus not greatly affecting relative rankings with respect to risk arising from site factors. This is not true to the same extent with risk factors arising from design and operation of reactors.

Our knowledge of how design/operation risk characteristics affect risk is more limited. Design-specific evaluations have been carried out for less than 15 reactors out of the more than 100 which may eventually be operated in the U.S. The typicality of the reactors which have been evaluated has not been clearly established. Indeed, among the risk-dominant sequences for some of the reactors already analyzed are one or more sequences which involve combinations of factors unique to the plant being analyzed (e.g., the remote manually-initiated auxiliary feedwater system at Calvert Cliffs and the compartment drain operator

error potential leading to common-mode failure of emergency core cooling, containment spray, and residual heat removal systems while in the recirculation mode at Sequoyah).

In addition, the state of knowledge about common-mode failures initiated by "external events" is quite limited. External events analysis has been carried out in PRAs covering only five reactors (Indian Point Units 2 and 3, Zion Units 1 and 2, and Big Rock Point). These results have large uncertainties associated with them, partially because the methodologies leading to the risk estimates from external events are so strongly dependent upon subjective expert judgment. Because of the limited data base on how external events affect risk, it is not possible in advance to accurately predict how inclusion of external events in a PRA of a particular plant will affect the risk estimates for that plant (i.e., whether the risk estimates will be increased or remain the same).

Finally, the comparability of the risk estimates derived from the various existing PRA studies is subject to question. The existing studies were not done using consistent methodologies, data bases, and assumptions, and there are completeness problems of varying degree with all existing PRA studies (for example, all existing PRA studies exclude sabotage as an accident initiator, and most exclude external events such as fires, earthquakes, hurricanes, floods, etc.). Moreover, most of the existing PRA studies (specifically the four RSSMAP and three IREP studies) did not include consequence analyses, but rather use the WASH-1400 release categories (uncorrected for plant-to-plant differences) as risk surrogates.

Q.08 What risk inferences can be drawn from the Sandia siting study (NUREG/CR-2239)?

A.08 There are a number of site risk inferences which can be drawn from the Sandia siting study, particularly on a comparative basis. Before addressing these inferences, however, it is necessary to understand the purposes of the Sandia siting study and the methodology and assumptions used in the study.

Q.09 How was the purpose of the Sandia siting study, and how was it performed?

A.09 The Sandia siting study was intended, among other purposes, to help define the risks associated with existing reactor sites and to examine the dependence of risk on such factors as meteorology, population distribution, and emergency response [NUREG/CR-2239, page 1-1]. To perform the study, Sandia National Laboratories utilized the CRAC2 accident consequence code. CRAC2 is an improved version of the accident consequences code used in the Reactor Safety Study (WASH-1400) and used by the NRC Staff in their Commission Question One testimony in this proceeding as well as in the preparation of recent NEPA environmental impact evaluations. The differences between CRAC and CRAC2 are discussed in the Sandia siting study [NUREG/CR-2239, pages 2-2 through 2-7, and Appendix E].

To perform the consequence calculations for 91 reactor sites, the siting study used site-specific 1970 census population data, regional meteorological data from 29 National Weather Service stations, regional sheltering factors, site-specific annual wind rose data, and site economic data updated from WASH-1400 to reflect inflation and changed economic conditions. The core inventory used in the consequence calculations was the same for each site, and is based on SANDIA-ORIGEN calculations for an end-of-cycle 3412 MWt Westinghouse PWR reactor core with a burnup of 33,000 megwatt-days per metric ton of uranium charged [NUREG/CR-2239, pages 2-5 through 2-6].

In addition, emergency response was modeled with seven generic sets of assumptions. The first three sets of emergency response assumptions assumed evacuation of the area within 10 miles of the reactor radially away from the reactor at a constant speed of 10 miles per hour, with delay times of one, three, and five hours, respectively (evacuees were assumed to travel to a distance of 15 miles from the reactor at which point they were assumed to be able to avoid further plume exposure). A fourth set of assumptions assumed evacuation as above, but at a constant speed of one mile per hour with a delay time of five hours (assumed to model "ineffective" evacuation). The fifth emergency response assumption set assumed no evacuation, but prompt sheltering followed by

relocation after six hours for populations within 10 miles of the site. The sixth set of assumptions modeled no emergency response at all (i.e., normal activities). The seventh set, representing what the study considered to be a "best estimate" for consequence predictions, consisted of a 30%, 40%, and 30% weighting of the first three sets (i.e., 10-mile evacuation at 10 miles per hour with delay times of one, three, and five hours) [NUREG/CR-2239, pages 2-6 through 2-7]. The seventh emergency response set is referred to as "summary evacuation".

Finally, the study used five generic "Siting Source Terms" -- SST-1, SST-2, SST-3, SST-4, and SST-5. The derivation of the source terms is detailed in a separate NRC study [NUREG-0773, especially pages 77-80]. A brief description of the type of accident leading to each of the Siting Source Terms, and the release characteristics and release fractions (taken from the Sandia siting study) is attached to this testimony as Appendix A. The calculations carried out for the Sandia siting study showed clearly that the SST-1 source term is the most significant of the five used in the study.

Q.10 Does the Sandia siting study provide a basis for risk comparisons between reactor sites?

A.10 Yes. In fact, the study specifically included a sensitivity study of risk to population density and distribution variations among the 91 sites evaluated. These results are conditional on an SST-1 release category assuming "summary evacuation" at all 91 sites. While the results obtained do not necessarily reflect actual reactor/site combinations, quite useful risk inferences can nonetheless be drawn from the results if the limitations of the study are taken into account.

The study presented conditional CCDF curves for early fatalities, early injuries, and latent fatalities for all 91 sites considered. These plots are attached to this testimony as Appendix B. The early fatality results show the greatest variability, with range of a factor of over 100 for the probability of one early fatality, and a range of more than a factor of 1000 for the number of early fatalities at a conditional probability of 10^{-3}.

The early injury results show substantial variability from site-to-site, but less variability than for early fatalities. For early injuries, a range of about a factor of 50 is shown for the probability of one early injury, and a range of more than 500 is shown for the number of early injuries at a conditional probability of 10^{-3}.

The latent cancer fatality results show much less variability than either early fatalities or early injuries. The probability of at least 10 latent cancer fatalities is nearly 1.0 for all sites, while the range in the number of latent cancer fatalities at a conditional probability of 10^{-3} is about a factor of ten.

The study explains these ranges in terms of the variability in population densities at 20, 50, and 200 miles for early fatalities, early injuries, and latent cancer fatalities, respectively. For areas within 20 miles of the 91 sites considered, the population densities ranged from one to 710 persons per square mile (a factor of 710 from highest to lowest population densities at 20 miles). For areas within 50 miles, the population densities ranged from 10 to 2100 persons per square mile (a factor of 210 from highest to lowest). Finally, for areas within 200 miles, the population densities ranged from 14 to 335 persons per square mile (a factor of 24 from highest to lowest) [NUREG/CR-2239, page 2-37].

More easily readable early fatality, early injury, and latent cancer fatality results (displayed in CCDF plots for groups of six sites) are shown in Figures C-1 through C-18 of the Sandia siting study, reproduced here and attached to this testimony as Appendix C. When the results for Indian Point from Figure C-8 are compared with the corresponding plots for all 91 sites evaluated, it is clear that the Indian Point curves fall in the upper range for each of the three consequences shown.

Further observations regarding the nature of the risk posed by the 91 sites evaluated can be obtained by a detailed examination of the CCDF curves in Figures C-1 through C-18 of the study (Appendix C to this

testimony). The Indian Point site is among a small group of sites for which the conditional probability of any early fatalities at all is 0.3 or greater for the SST-1 release category. The sites falling into this category are: Braidwood, Indian Point, Limerick, and Zion.

There are few sites where the number of early fatalities at a conditional probability of one in ten (i.e., 10^{-1}) exceeds 1000. These sites are: Indian Point, Limerick, McGuire, Midland, and Zion. There are also few sites (but more than for early fatalities) where the number of early injuries at a conditional probability of 10^{-1} exceeds 1000. These sites are: Bailly, Beaver Valley, Catawba, Fermi, Forked River, Haddam Neck, Indian Point, Limerick, McGuire, Midland, Millstone, Oyster Creek, Perkins, Perry, St. Lucie, Shoreham, Susquehanna, Three Mile Island, Waterford, Zimmer, and Zion. There are few sites where the number of latent cancer fatalities at a conditional probability of 10^{-1} exceeds 10,000. These sites are: Forked River, Indian Point, Limerick, and Oyster Creek.

There are also a limited number of sites for which the calculated early fatalities conditional on an SST-1 release exceed 10,000 at a conditional probability of 10^{-3}. The sites falling into this category are: Beaver VAlley, Catawba, Forked River, Haddam Neck, Indian Point, Limerick, McGuire, Oyster Creek, Peach Bottom, Perkins, Salem, Sequoyah, Shoreham, Surry, Susquehanna, Three Mile Island, and Zimmer. For early injuries, there ar a limited number of sites for which early injuries exceed 100,000 at a conditional probability of 10^{-3}. These sites are: Beaver Valley, Fermi, Indian Point, Limerick, and Zimmer.

In addition, when the mean results conditional on an SST-1 release and assuming summary evacuation are examined, the results for Indian Point are among the largest. This data is summarized in Table C-1 of the siting study, and is attached to this testimony as Appendix D. For mean early fatalities, the sites with the highest number of mean early fatalities (in descending order) are: Limerick (970), Indian Point (830), Zion (520), Midland (320), and Three Mile Island and Millstone (240). For mean early injuries, the sites with the highest number of

mean early injuries (in descending order) are: Indian Point (3600), Limerick (2800), Zion (1600), and Bailly, Beaver Valley, and Three Mile Island (1200). For mean latent cancer fatalities, the sites with the highest number of mean latent cancer fatalities (in descending order) are: Indian Point (8100), Limerick (5400), Forked River and Oyster Creek (4400), Zion (4000), and Three Mile Island (3500).

What these results demonstrate is that for a given release magnitude, there are a very few reactor sites in the U.S. for which large consequences are possible even under unfavorable combinations of wind direction and rain, and that Indian Point is one of these few sites.

Q.11 Do these results reflect actual power level?

A.11 No. However, a separate Sandia National Laboratories report has performed corrections of the mean results for power level by numerical scaling based on the electrical output of the reactor. In addition to providing results for mean early fatalities, early injuries, and latent cancer fatalities, the results in the separate report (NUREG/CR-2723) include mean population exposure (in person-rem) and mean offsite property damage. Based on results scaled to reflect actual reactor size and conditional on an SST-1 release with summary evacuation, the following rankings are provided (in descending order):

EARLY FATALITIES: Limerick (917), Indian Point-3 (716), Indian Point-2 (648), Zion (512), Millstone-3 (250), Midland-2 (233)

EARLY INJURIES: Indian Point-3 (3136), Indian Point-2 (2837), Limerick (2675), Zion (1571), Millstone-3 (1019), Fermi (955), Beaver Valley (915), Three Mile Island-1 (849)

LATENT CANCER FATALITIES: Indian Point-3 (6988), Indian Point-2 (6321), Limerick (5087), Forked River (4420), Zion (3968), Braidwood (3240), Millstone-3 (3234), Sequoyah (3103)

POPULATION EXPOSURE (MILLIONS OF PERSON-REM): Indian Point-3 (108), Indian Point-2 (97.4), Forked River (85.5), Limerick (77.6), Zion (60.9)

OFFSITE PROPERTY DAMAGE (BILLIONS OF DOLLARS): Indian Point-3 (10.2), Indian Point-2 (9.20), Limerick (6.23), Zion (4.80), Forked River (4.06)

Thus, even when the Sandia siting study results are corrected for power level by scaling the results, the conditional consequences of a large release of radioactivity at the Indian Point reactors are high compared with reactors located at 90 other sites in the U.S. Indeed, the scaled results for the indicated consequences show Indian Point Units 2 and 3 to have the highest mean consequences (conditional on SST-1) except for early fatalities, where the Indian Point reactors rank number 2 and 3 behind the Limerick reactors (which tie for number 1 since they are the same power level).

Q.12 Did the Sandia siting study provide conditional consequence results for consequences other than those cited above?

A.12 In one sense, the answer is yes. The siting study included numerous sensitivity studies using the Indian Point population and New York City weather as a "base case" for sensitivity calculations on, among other consequences, the maximum distances for early fatalities and early injuries, and the distance and areal extent of interdiction areas.

In another sense, however, the answer is no. The only consequence results that the siting study itself presented for all sites examined were early fatalities, early injuries, and latent cancer fatalities. The financial consequences study (NUREG/CR-2723) provided some additional information (mean population dose and offsite property damage levels). The CRAC2 computer printouts, which are at the NRC's Public Document Room (although some sites, Indian Point included, are missing from the files), contain much more information. It was not feasible to reproduce the data from these files in time for inclusion in this testimony. The data contained in the CRAC2 computer printouts (from which the siting and financial consequences studies were compiled) display early consequences for the seven emergency response scenarios identified above, and latent and financial consequences in general which were apparently assumed by the siting study to be independent of emergency response.

In addition to providing conditional consequence data on consequences other than those specified in the siting and financial consequences study, the CRAC2 computer printouts contain the data from which CCDF curves for each consequence may be constructed. These CCDF data are not truncated at a conditional probability of 10^{-3} as are the data reported in the siting study, but extend to the "maximum calculated value" from the CRAC2 output (an example of such CCDF curves, released by the NRC in response to a FOIA request, is attached to this testimony as Appendix E). It was the "maximum calculated value" data that was released by the staff of the Subcommittee on Oversight and Investigation of the House Committee on Interior and Insular Affairs on November 1, 1982.

Q.13 What rationale did the Sandia siting study provide for truncating its results at a conditional probability of 10^{-3}?

A.13 The siting study provided the following explanation [NUREG/CR-2239, pages 2-34 through 2-34]:

> "The figures have been truncated at conditional probabilities of 10^{-3} (one in a thousand releases). This was done because consequence probabilities and magnitudes for improbable events (those with conditional probabilities less than 10^{-3}) are very uncertain. A large part of this uncertainty is due to the assumption of an evacuation only within 10 miles. Because of this assumption, all persons beyond 10 miles were assumed to be exposed to deposited radionuclides for 1 day, regardless of dose rate[a]. Any emergency actions taken beyond 10 miles (e.g., sheltering or prompt relocation) would significantly mitigate the consequences of low-probability, high consequence events."
>
> [a] Under some meteorological conditions, the 1-day bone marrow dose at 10 miles can exceed 1000 rem.

Q.14 Does the Sandia siting study provide any indication as to the range of consequences predicted for the "low-probability, high consequence events" identified above for varying emergency response actions taken beyond 10 miles?

A.14 Yes. Section 2.5 of the siting study contains sensitivity studies conducted using a "base case" example of Indian Point population and wind rose, New York City population, an 1120 MWe reactor, and an SST-1 release. Table 2.5-6 in the siting study summarizes the results of

these sensitivity calculations (attached to this testimony as Appendix F). Table 2.5-6 provides mean and "99th percentile" results (i.e., consequences at a conditional probability of 10^{-3}). Results are presented for a total of eight scenarios:

Evacuation:	5-hour delay, 1 mph
	5-hour delay, 10-mph
	3-hour delay, 10-mph
	1-hour delay, 10 mph
	Summary evacuation
Sheltering:	24-hour relocation time
	12-hour relocation time
	6-hour relocation time

In addition, results are presented for all eight of these scenarios for response distances of 0, 5, 10, 15, and 25 miles.

Q.15 Are the "maximum calculated value" results of any use?

A.15 Yes. Considering the uncertainties involved, the "maximum calculated value" results from the Sandia siting study may be considered as crude, upper-bound estimates of "worst-case" results (the "maximum calculated value" results for early fatalities, early injuries, and latent cancer fatalities for the reactors with the largest calculated consequences, scaled for reactor power level, are attached to this testimony as Appendix G). It is clear from the sensitivity studies cited above (and displayed in Appendix E to this testimony) that implementation of emergency response beyond 10 miles can substantially reduce these consequences in many cases (if such actions are feasible). The most significant conclusion to be drawn from the "maximum calculated value" results and the sensitivity study cited above is that these results generally show the value of emergency response beyond 10 miles for low-probability, high consequence accidents.

Q.16 Turning now to that portion of risk related to design and operational characteristics, what probabilistic risk assessments (PRAs) have been performed for U.S. nuclear reactors?

A.16 WASH-1400 (the Reactor Safety Study) represented the first application of PRA methodology to commercial nuclear power plants. That study

evaluated accident sequences for the Surry and Peach Bottom reactors, and examined accident consequences at six hypothetical sites.

In addition, there are seven published PRAs sponsored by the NRC which have been performed in two programs. The first of these programs is the "Reactor Safety Study Methodology Applications Program" (RSSMAP) in which PRA studies of the following reactors were performed: Grand Gulf Unit 1, Sequoyah Unit 1, Calvert Cliffs Unit 2, and Oconee Unit 3. The second program of NRC-sponsored PRAs is the "Interim Reliability Evaluation Program" (IREP) in which PRA studies of the following reactors were performed: Browns Ferry Unit 1, Arkansas Nuclear One Unit 1, and Crystal River Unit 3. An IREP report on Millstone Unit 1 is apparently in printing and should be available soon; in addition, an IREP report on Calvert Cliffs Unit 2 is under preparation, the purpose of which is to "benchmark" the RSSMAP and IREP report results.

There are also several industry-sponsored PRA studies that have been published. These PRAs evaluated: Indian Point Units 2 and 3, Zion Units 1 and 2, Big Rock Point, and Limerick Units 1 and 2. In addition, the NRC Staff reports that industry-sponsored PRA studies are underway for the following reactors: Oconee, Midland, Browns Ferry, Susquehanna, LaSalle, Seabrook, Millstone Unit 3, Palisades, Shoreham, Pilgrim, Sequoyah, Yankee Rowe, McGuire, Oyster Creek, and Calvert Cliffs [Dircks, page 17].

Q.17 To what extent do the results from the Reactor Safety Study (WASH-1400) establish the "range of risks" cited by the Commission in Commission Question Five as the basis of comparison for the risks posed by Indian Point Units 2 and 3?

A.17 WASH-1400 did not attempt to establish a "range of risks". The results of the study were generalized from the evaluation of the design of only two reactors (Surry and Peach Bottom) to represent the risk posed by the first 100 reactors anticipated to be licensed for operation in the U.S. Accident consequences were evaluated for only six sites, and none of these sites were actual U.S. reactor sites, but rather were synthesized from population data from the sites for the first 100 reactors. The 68

sites for the first 100 reactors were assigned to a composite on the basis of comparable meteorology. The NRC Staff recognized soon after the Reactor Safety Study was published that meaningful conclusions could not be drawn on the sensitivity of the WASH-1400 results to differences in site characteristics [Gossick, response to Question 4].

An NRC-sponsored review of the WASH-1400 report by the Risk Assessment Review Group (RARG) concluded that it could not determine whether the absolute probabilities in WASH-1400 were high or low, but that the error bounds on those estimates were generally "greatly understated" [NUREG/CR-0400, page viii]. A 1981 paper by Erdmann, et. al. [Erdmann], attempted to account for some of the more significant uncertainties in the WASH-1400 early fatality results by using an informal delphi approach (collective engineering judgment of experts in the PRA field). This effort excluded sabotage and external events, as did the original WASH-1400 study, however.

The authors concluded that the median accident frequency value from WASH-1400 should be reduced by a factor of 12 and that the median accident consequence value should be reduced by a factor of 5 below the WASH-1400 values. The authors placed uncertainties of a factor of 20 on accident frequency (as opposed to a factor of 5 in WASH-1400) and a factor of 15 on accident consequences (as opposed to a factor of 4 in WASH-1400) [Erdmann, page 379]. A graph showing the WASH-1400 uncertainty band and the Erdmann, et. al., uncertainty band for WASH-1400 early fatality estimates for 100 reactors is attached to this testimony as Appendix H [Erdmann, page 379]. While establishing a greater uncertainty band, it is not clear that this effort answers fully the RARG's criticisms. It is also not clear that this effort establishes the "range of risks" sought by the Commission. The authors of the paper ascribe an uncertainty factor of only three for the scaling of the WASH-1400 results from two to 100 reactors [Erdmann, page 376], whereas the RSSMAP and IREP results show a much greater variability.

A Sandia Laboratories sensitivity study (which substituted actual site demographic data from a number of sites for the composite data in

WASH-1400) showed a much greater range of early fatality results for specific sites than the range of uncertainty assigned to the early fatality results in WASH-1400 [SAND78-0556, pages 42 and 44]. In addition, it should be noted that the CCDF curves presented in WASH-1400 were for the first 100 reactors in aggregate, not for individual reactors. Thus, it is clear that the WASH-1400 results do not establish the "range of risks" sought by the Commission as a basis for comparison of the risks posed by Indian Point Units 2 and 3.

Q.18 To what extent do the RSSMAP results account for the inadequacies of WASH-1400 noted above?

A.18 The RSSMAP PRAs evaluated a greater range of reactor and containment designs, but there are problems with making comparisons of the results of the RSSMAP studies with other PRA results.

The RSSMAP studies were specifically intended for comparison with the WASH-1400 results and were initiated in an attempt to ascertain the risk-dominating accident sequences for a broader spectrum of LWR designs than considered in WASH-1400. The RSSMAP studies were not intended, however, "to be an absolute determination of risk" [NUREG/CR-1659, Vol. 1, pages 1-1 and 3-4].

Thus, while the RSSMAP reports extend the WASH-1400 type of analysis to other LWR designs, it is clear that many of the WASH-1400 problems remain. The RSSMAP results were not intended to stand alone as determinants of absolute risk for the four plants studied in the program. It would therefore be inappropriate to attach consequence analyses to the RSSMAP results and assert that the combined results represent complete risk studies for the RSSMAP plants. Moreover, due to differences between the RSSMAP studies and WASH-1400, it is not clear that the core melt and release category frequency results are strictly comparable since somewhat different methodologies were used to calculate these values, and the release categories do not reflect plant-specific differences.

Q.19 What are some of the limitations on the RSSMAP studies?

A.19 The RSSMAP studies were limited in several respects. The RSSMAP studies are grounded strongly in the results of WASH-1400; a group of transients and LOCAs similar to those in WASH-1400 served as the initiating events in the RSSMAP studies. Specific sequences such as pressurized thermal shock and multiple steam generator tube rupture were not analyzed in either WASH-1400 or the RSSMAP reports. Indeed, the Oconee RSSMAP report noted that a more in-depth analysis would require treatment of a larger group of transient initiators [NUREG/CR-1659, Vol. 2, page 6-44]. Further, as was the case with WASH-1400, the RSSMAP studies excluded external events, sabotage, and fires as initiating events [NUREG/CR-1659, Vol. 2, page iv; Vol. 3, page vi; Vol. 4, pages 2-1 and 2-8].

Another strong link between WASH-1400 and the RSSMAP reports is the fact that the RSSMAP analyses are based on the WASH-1400 conclusion that system failure probabilities are dominated by only a few types of failures [NUREG/CR-1659, Vol. 2, page 2-6]. Thus, rather than develop elaborate fault tree models and conduct exhaustive searches for component and system failure modes, detailed evaluations were conducted only for those systems which appeared in the dominant accident sequences in WASH-1400. Detailed evaluation of the remaining systems in the RSSMAP plants were performed only if it could not be readily shown that the systems did not appear in sequences contributing significantly to risk [NUREG/CR-1659, Vol. 1, page 3-1]. The RSSMAP studies acknowledge that this methodology does not provide assurance that all failure modes were identified [NUREG/CR-1659, Vol. 3, page 2-6].

There are other linkages between WASH-1400 and the RSSMAP studies. The human error and component failure data bases in WASH-1400 were adopted without modification for the RSSMAP studies with a few exceptions were adequate plant-specific data was available [NUREG/CR-1659, Vol. 2, page iii; Vol. 3, page v; Vol. 4, page v]. The WASH-1400 release categories were also adopted unchanged in the RSSMAP reports and were used as risk surrogates -- the RSSMAP reports contain no accident consequence analyses. The calculation of the release category frequencies in the

RSSMAP studies also used the much-criticized "curve smoothing" technique that was applied to the WASH-1400 results [NUREG/CR-1659, Vol. 2, page 2-7; Vol. 3, pages 2-6 through 2-7; Vol. 4, page 2-5].

Despite the above similarities, however, there are differences between WASH-1400 and the RSSMAP studies which make risk comparisons between the two sets of studies questionable. The RSSMAP studies were conducted based on information contained in the Final Safety Analysis Reports, Technical Specifications, and selected plant procedures for the plants analyzed [NUREG/CR-1659, Vol. 1, page v; Vol. 3, page iii]. The Grand Gulf RSSMAP study was even more limited since that plant was under construction at the time of the analysis and the Technical Specifications and operating procedures were not available [NUREG/CR-1659, Vol. 4, pages iii and 6-32].

The studies generally indicate that a more thorough analysis would require additional information, including as-built plant drawings and diagrams, all plant procedures, and direct personal contacts at the utilities and reactor vendors for the purpose of answering questions [see, e.g., NUREG/CR-1659, Vol. 2, page 6-44]. It is important to recognize that performing the RSSMAP analyses on such a limited data base is conservative, since the FSAR analyses are typically more conservative than typical PRA practice. FSARs are licensing documents for which a conservative approach is standard, whereas PRAs typically are based on a realistic approach to such important factors as system success criteria. It must be recognized, therefore, that the RSSMAP results are likely to overestimate core melt and release category frequencies. When comparing the RSSMAP results against other PRA results, therefore, it must be kept in mind that the RSSMAP results will tend to bias such risk comparisons by making plants whose PRAs conducted using more realistic assumptions look less risky in comparison to the RSSMAP plants. Indeed, the NRC's Executive Director for Operations expressed the following reservation regarding cross-PRA comparisons, "(O)ne should recognize that these probabilistic analyses were not performed using consistent methodology and assumptions; therefore, the

direct comparison of one with the other may well be invalid." [Dircks, pages 10-11]

In addition, the core melt and release characteristics for the RSSMAP studies were performed using the MARCH and CORRAL-2 computer risk codes. MARCH had not been written at the time of the WASH-1400 analyses [MARCH was publicly released in October 1980; NUREG-0773, page 17], and a different version of CORRAL was used in WASH-1400 [NUREG/CR-1659, Vol. 2, pages iii through iv; Vol. 3, pages v through vi; Vol. 4, page v]. Further, rather than express component unavailabilities as WASH-1400 did (with a median unavailability and associated tolerance bounds), the RSSMAP reports expressed component unavailabilities as "point estimates". This was justified on the bases that the RSSMAP results were intended for comparison with the WASH-1400 results and not as absolute determinations of risk [NUREG/CR-1659, Vol. 1, page 3-4], and that the additional effort of estimating error bounds was not necessary for the RSSMAP/WASH-1400 comparisons or for the identification of risk-dominant accident sequences [NUREG/CR-1659, Vol. 2, page 2-7; Vol. 3, pages 2-6 through 2-7; Vol. 4, page 2-5].

Other differences are different treatment of in-vessel steam explosion phenomena [NUREG/CR-1659, Vol. 1, page 4-6; Vol. 2, page 5-8; Vol. 3, pages 5-10 through 5-11; Vol. 4, page 5-11], and different modeling of systems interaction [NUREG/CR-1659, Vol. 3, page 2-7; Vol. 4, page 2-6].

Q.20 Do the IREP studies account for the inadequacies of the Reactor Safety Study noted above?

A.20 No. As with the RSSMAP studies, the IREP reports also exclude sabotage, external events, and fires as accident initiators. The IREP studies are reliability studies and while inferences regarding risk-dominant accident sequences are drawn from the analyses, detailed risk analyses were not performed nor was it intended that they be performed. The analyses leading to the grouping of accident sequences into release categories relied on previous studies of similar plants. Based on a recognition of these limitations, the release category frequencies from the IREP studies were not used as input to accident consequence

calculations. According to the Browns Ferry IREP report [NUREG/CR-2802, Main Report, page 95], "(T)he quantitative results must be regarded as incomplete from a risk point of view." The Browns Ferry IREP report further explains [NUREG/CR-2802, Main Report, page 2]: "The principal product obtained is the integrated engineering logic presented in the plant and system models and the insights into plant features contributing significantly to risk -- not the specific values computed for accident frequencies."

In addition to the above limitations, there are completeness problems with the IREP studies. Coupling of faults with design, fabrication, or environmental conditions was not treated explicitly in the studies. Further, the technique used to link the root cause of an initiating event with system faults may have missed multiple fault scenarios that can both initiate a transient and degrade the performance of one or more safety systems. Human errors of commission were in general not included in the analyses, nor was consideration given to systematic bias arising from morale or management practices at the plants [NUREG/CR-2802, Main Report, page 95; NUREG/CR-2787, Vol. 1, page 8-84]. Only accidents initiated at full power were considered. Partial failures (such as degraded bus voltage) are also not modeled [NUREG/CR-2453, pages 31-32].

Unlike the RSSMAP reports, the IREP results are not intended for comparison with WASH-1400 results. Among the reasons given for this by the IREP studies are differences in reactor design, advances in the state-of-the-art in PRA, and differences in the data bases used by the respective studies [NUREG/CR-2515, Vol. 1, page I-i].

Further, the IREP studies are dependent on the RSSMAP and WASH-1400 results for accident sequence phenomenology data. The Crystal River Unit 3 IREP and the Arkansas Nuclear One Unit 1 IREP studies are both based on the MARCH and CORRAL-2 results from the Oconee Unit 3 RSSMAP report. The Browns Ferry IREP is based on the Grand Gulf RSSMAP and WASH-1400 Peach Bottom results; few plant-specific MARCH calculations were performed and no plant-specific CORRAL-2 calculations were

performed [NUREG/CR-2787, Vol. 1, page 8-41; NUREG/CR-2515, Vol. 1, pages I-i, 1-4, and 2-22; NUREG/CR-2802, Main Report, page 85].

Thus, as was the case with the RSSMAP report results, the IREP results do not adequately address the problems with the WASH-1400 results. The IREP results are admittedly incomplete from a risk standpoint, and were not intended as predictors of absolute risk. Therefore, the IREP studies are not useful in establishing the "range of risk" sought by the Commission in response to Commission Question Five.

Q.21 To what extent do the published industry-sponored PRAs overcome the difficulties with WASH-1400 cited above and establish the "range of risks" posed by U.S. reactors?

A.21 Only to a very limited extent. The Big Rock Point PRA addresses a very small reactor (72 MWe). Comparison of the Big Rock Point PRA results with the results of IPPSS is therefore largely academic, and the Big Rock Point PRA does little to establish the "range of risks" posed by operating U.S. LWRs.

The Limerick PRA has limitations which make risk comparisons between the risk results for Limerick and Indian Point questionable. It should be noted that the Limerick plants have not been licensed to operate by the NRC; license hearings are in progress, however. On more substantive matters, the Limerick PRA is intended to provide a comparison of the Limerick reactors with the WASH-1400 BWR results; the Limerick study is self-descrived as an "updated Reactor Safety Study (WASH-1400) type of analysis" [Limerick PRA, page 1-19].

External events, sabotage, and fire were excluded from the Limerick PRA analysis. The study makes the explicit assumption that current NRC requirements on seismic design, equipment separation, environmental qualification, and security are adequate to reduce the probability of core melt accidents to below that from other causes [Limerick PRA, pages 1-16 and 1-19]. The study does not concern itself with possible design flaws, assuming that "(c)omponent design meets the requirements needed for proper operation". The study further assumes that operator

training, supervision, and plant maintenance are adequate for proper plant operation [Limerick PRA, page 1-31].

The Limerick PRA used the WASH-1400 consequence model (CRAC) essentially unchanged. Thus, the protective actions modeled do not reflect a site-specific analysis of protective action implementation capabilities, but rather assume that evacuation occurs within a five-mile radius and to a distance of 25 miles downwind in a 45-degree sector. The modeling also assumed a constant evacuation speed of 1.2 mph and no delay time before evacuation [Limerick PRA, pages E-14 through E-15 and E-25].

The results from the Limerick PRA, while accounting for site-specific population size and distribution, was intended for comparison with WASH-1400 BWR results. The Limerick PRA does not reflect adequate site-specific modeling of protective actions in its consequence model. In addition, as noted above, the study excludes fires and external events as initiating events. Comparison of the Limerick PRA results with the results in IPPSS is therefore questionable. Further, the Limerick PRA does not establish the "range of risks" posed by licensed U.S. LWRs, either alone or in conjunction with the only other existing site- and plant-specific PRA in this category -- the Zion Probabilistic Safety Study.

The Zion Probabilistic Safety Study (ZPSS) was performed by Pickard, Lowe, and Garrick and Westinghouse, and is very similar to the IPPSS study. The methodologies used in both studies were identical. Thus, the results of ZPSS and IPPSS can be compared within their uncertainties. However, the ZPPSS study does not itself establish the "range of risks" posed by operating U.S. LWRs. Comparison of IPPSS results with ZPSS would provide a comparison with two additional plants only -- Zion Units 1 and 2.

Q.22 Is it possible to validly compare accident frequencies, release category frequencies, or overall risk (as determined by available PRA studies) from just internal initiating events?

A.22 No. Comparisons of accident sequence frequencies would be misleading

since similar accident sequences contribute to risk quite differently for different reactor/site combinations. Comparisons of release category frequencies (as in the RSSMAP and IREP reports) is not appropriate since the release categories could be very different from plant-to-plant (the IREP and RSSMAP reports adopted the WASH-1400 release categories unmodified). Risk comparisons based on only internal accident initiators may lead to misunderstandings regarding the risk of one plant relative to other plants, partially because risk from external events can be dominant for a given plant.

A more useful comparison might be on the basis of the reliability of safety functions. Such data, taken from FSARs, is available for a number of important safety systems, including emergency power, interfacing systems (for Event V comparisons), containment cooling, auxiliary feedwater, ECCS, and decay heat removal. These data are available in computerized form to the NRC Staff for 69 reactors [NUREG/CR-2069]. Even comparisons of safety function reliability are questionable, however, since the significance of a given safety function to risk can vary considerably from plant to plant. To make relative risk judgments among plants, one would need an integrated conparison of the sort provided by consistently performed PRAs. The existing PRA studies were in general not consistently performed; different methodologies were used, assumptions differed, the scope of the studies differed, and the degree of conservatism employed in modeling differed. Thus, using safety function reliability as a surrogate for risk involves very large uncertainties.

Q.23 What comparisons are able to be made regarding that portion of risk deriving from "external events"?

A.23 WASH-1400 generally concluded that "external events" were not important contributors to risk. The Risk Assessment Review Group, among others, questioned the correctness of this assessment [NUREG/CR-0400, page ix].

There are only a few plants for which external events have been studied in the context of a PRA. The only published PRA studies to date which have included external events are for Indian Point Units 2 and 3, Zion

Units 1 and 2, and Big Rock Point. This is a very limited data base from which to make judgments, especially if the intent is to extend the conclusions of these studies to reactor/site combinations generally. What the limited number of external events analyses have shown, however, is that "external events" can be important contributors to risk for some plants. It is generally conceded, however, that risk estimates from "external events" have larger uncertainties associated with them than risk estimates associated with "internal events".

It should be noted that while fires are generally grouped with "external events", the fires of principal risk concern are those occurring within the plant. Thus, PRA studies limited to "internal events" do not necessarily consider <u>all</u> internally-initiated risks.

Q.24 Can valid risk inferences be drawn from comparisons of safety systems among plants?

A.24 No. As noted above, one might compare the reliability of safety <u>functions</u> using established data bases for this purpose, however, the risk significance of safety functions varies from plant to plant. For example, emergency AC power is not of equal concern for all plants; some plants have multiple external AC power sources available. Additionally, combustible gas control may not be of the same risk significance for large dry PWR containments as it is for other types of containments. Comparisons of safety function reliability, in the absence of an integrated assessment of all safety functions, might miss important intersystem dependencies [NUREG/CR-2453, pages 72-73].

Consequently, while comparisons of the reliability of safety functions can be made, their risk significance is open to question [NUREG/CR-2453, page 73]. As noted previously, what is needed to establish risk is an integrated assessment of safety function reliability represented by consistently performed PRA studies. Consistency among the PRA studies for which comparisons are being attempted is very important due to possible differences in methodology, assumptions, and failure rate data bases. Even when consistently-performed PRA studies are available for safety function reliability comparisons, the sources of uncertainty in

the PRAs for the plants being compared must be carefully examined to ascertain the sources of the uncertainties and their impact on the validity of the comparisons being attempted.

Q.25 Is a comparison of the Indian Point results with the Commission's Safety Goals germane to answering Commission Question Five?

A.25 Not in my view. The principal reason for this view is that what the Commission ultimately seeks to assess through this proceeding is the acceptability of the risks posed by operation of Indian Point Units 2 and 3. Comparisons of risk estimates with the safety goals does not accomplish this goal for two principal reasons.

First, the safety goals are incomplete estimators of risk. As noted by the Risk Assessment Review Group report [NUREG/CR-0400, page ix], there are many accident consequences associated with risk. The safety goals focus on early fatality, latent cancer, and core melt risks. Other consequences which can be important risk considerations are land contamination, financial impacts, and non-fatal injuries and disease. Second, the safety goals are intended as assessment tools, not as "speed limits" [Bernero, page 7]. A given plant may exceed a safety goal and still not represent an unacceptable risk; by the same token, a given plant might meet all of the safety goals, and still represent an unaccaptable risk. Moreover, the safety goals are subject to change after the two-year evaluation period which the Commission has adopted [Safety Goal Policy Statement, pages 2, 4, and 8].

It should also be noted that during the two-year evaluation period, the Commission has stated that the safety goals and quantitative design objectives should not be used in the licensing process, and are not to be litigated in NRC hearings. The Commission has explicitly limited the use of the safety goals during the evaluation period to: (a) examiniation of proposed and existing regulatory requirements; (b) establishment of research priorities; (c) resolution of generic safety issues; and (d) definition of the relative importance of issues as they arise [Safety Goal Policy Statement, page 7].

Q.26 Are there any conclusions regarding the risk of Indian Point relative to the "range of risks" that can be drawn on the basis of information now available?

Q.26 Yes. On a site basis, and considering the actual power level of the Indian Point reactors, it is clear that the Indian Point site is an outlier with respect to other plants. The only sites which approach the conditional mean consequences (based on an SST-1 release) for Indian Point are Zion and Limerick. These two sites are also outliers.

Considering design/operation risk characteristics, valid comparisons are not feasible at present. There are a limited number of PRA studies available, the typicality of the reactors upon which the studies have been performed has yet to be established. Furthermore, the existing PRA studies vary significantly with regards to completeness, assumptions, methodology, data base, and degree of conservatism employed in the analyses. It would be premature to attempt to rank reactors according to absolute risk because of these factors.

Q.27 What implications do these conclusions have regarding the desirability of accident mitigation systems?

A.27 Serious consideration of such mitigation systems for Indian Point should be undertaken. It will be a period of several years (at best) before it will be reasonable to attempt relative risk rankings based on absolute risk projections. Waiting for such results, if they are indeed available in a few years, poses an uncertain risk to the population surrounding Indian Point. It is clear that the risk associated with site risk characteristics is high compared to other sites. The risk associated with design/operation risk characteristics is highly uncertain, but provides no firm basis for delaying in-depth consideration of accident mitigative measures.

Q.28 Does consideration of mitigative measures using a value of $1,000 per person-rem averted provide a reasonably complete view of the value of mitigation systems?

A.28 No. Use of the $1,000 per person-rem averted standard reflects a hidden judgment that reactors in general are already adequately safe. It would

require a very high core melt probability combined with a large conditional probability of a large release to make even the most modest add-on mitigation system "cost-effective" under the $1,000 per person-rem averted standard.

Moreover, the standard does not reflect the degree to which fatalities, injuries, and offsite property damage is caused by the person-rem exposure. This is especially true with respect to consequences whose magnitude is determined on a threshold basis, since the number of persons suffering the early consequences is strongly determined by population magnitude and distribution near the plant site. Using the CRAC2 results from the Sandia siting study, D.R. Strip of Sandia National Laboratories has examined this issue by ranking the number of sites versus the number of early fatalities, early injuries, latent cancer fatalities, and offsite property damage per thousand person-rem of exposure for an SST-1 release. The results for early fatalities per thousand person-rem span more than a factor of 1,000. The results for early injuries span a factor of about 70. The results for offsite property damage span about a factor of 10. The results for latent cancer fatalities span a range of less than a factor of 2 [NUREG/CR-2899, page 6]. Thus, the results of applying the 1,000 per person-rem averted will not be consistent from site-to-site.

In addition, with regard to latent cancers, the present use of the $1,000 per person-rem standard involves two problems. First, the reevaluation of the atomic bomb casualty data is ongoing; Radford asserts that this reevaluation will support the linear hypothesis for cancer risk coefficients, rather than the present model which is based on a liner-quadratic formulation [Radford, page 14]. Second, the present consequences model calculates only fatal cancers; there would also be a large number of non-fatal cancers whose costs would not be assessed under the $1,000 per person-rem standard. In addition, the costs associated with treatment of early injuries would not be assessed as the standard is presently designed.

The value of a mitigation system for Indian Point should also be

considered in conjunction with cost estimates for an early shutdown of the reactors. The costs of an early shutdown of Indian Point Units 2 and 3 will be addressed in testimony by Vince Taylor, to be filed under Commission Question Six in this proceeding.

Q.29 Does this conclude your testimony?

A.29 Yes.

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A P P E N D I X   A

Tables 2.3.1-1 and 2.3.1-2, "Brief Descriptions Characterizing the Accident Groups Within the NRC "Accident Spectrum", and, "NRC Source Terms for Siting Analysis"; taken from pages 2-12 and 2-13, NUREG/CR-2239, Technical Guidance for Siting Criteria Development, Sandia National Laboratories, December 1982.

Table 2.3.1-1. Brief Descriptions Characterizing the Accident Groups Within the NRC "Accident Spectrum" [22]

Group 1	Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach of containment.
Group 2	Severe core damage. Containment fails to isolate. Fission product release mitigating systems (e.g., sprays, suppression pool, fan coolers) operate to reduce release.
Group 3	Severe core damage. Containment fails by basemat melt-through. All other release mitigation systems function as designed.
Group 4	Modest core damage. Containment systems operate in a degraded mode.
Group 5	Limited core damage. No failures of engineered safety features beyond those postulated by the various design basis accidents. The most severe accident in this group assumes that the containment functions as designed following a substantial core melt.

Table 2.3.1-2. NRC Source Terms for Siting Analysis

Release Characteristics[a]	Source Term				
	SST1	SST2	SST3	SST4	SST5
Accident Type	Core Melt	Core Melt	Core Melt	Gap Release	Gap Release
Containment Failure Mode	Overpressure	H_2 Explosion or Loss of Isolation	-	-	-
Containment Leakage	Large	Large	1%/day	1%/day	0.1%/day
Time of Release (hr)	1.5	3	1	0.5	0.5
Release Duration (hr)	2	2	4	1	1
Warning Time (hr)	0.5	1	0.5	-	-
Release Height (meters)	10	10	10	10	10
Release Energy	0	0	0	0	0
Inventory Release Fractions					
Xe-Kr Group	1.0	0.9	6×10^{-3}	3×10^{-6}	3×10^{-7}
I Group	0.45	3×10^{-3}	2×10^{-4}	1×10^{-7}	1×10^{-8}
Cs-Rb Group	0.67	9×10^{-3}	1×10^{-5}	6×10^{-7}	6×10^{-8}
Te-Sb Group	0.64	3×10^{-2}	2×10^{-5}	1×10^{-9}	1×10^{-10}
Ba-Sr Group	0.07	1×10^{-3}	1×10^{-6}	1×10^{11}	1×10^{-12}
Ru Group	0.05	2×10^{-3}	2×10^{-6}	0	0
La Group	9×10^{-3}	3×10^{-4}	1×10^{-6}	0	0

a. As defined in the Reactor Safety Study [1].

A P P E N D I X   B

Figure 2.4.2-1, page 2-33, NUREG/CR-2239, Technical Guidance for Siting Criteria Development, Sandia National Laboratories, December 1982. Provides early fatalities, early injuries, and latent cancer fatalities CCDFs (conditional on an SST-1 release) for 91 U.S. reactor sites.

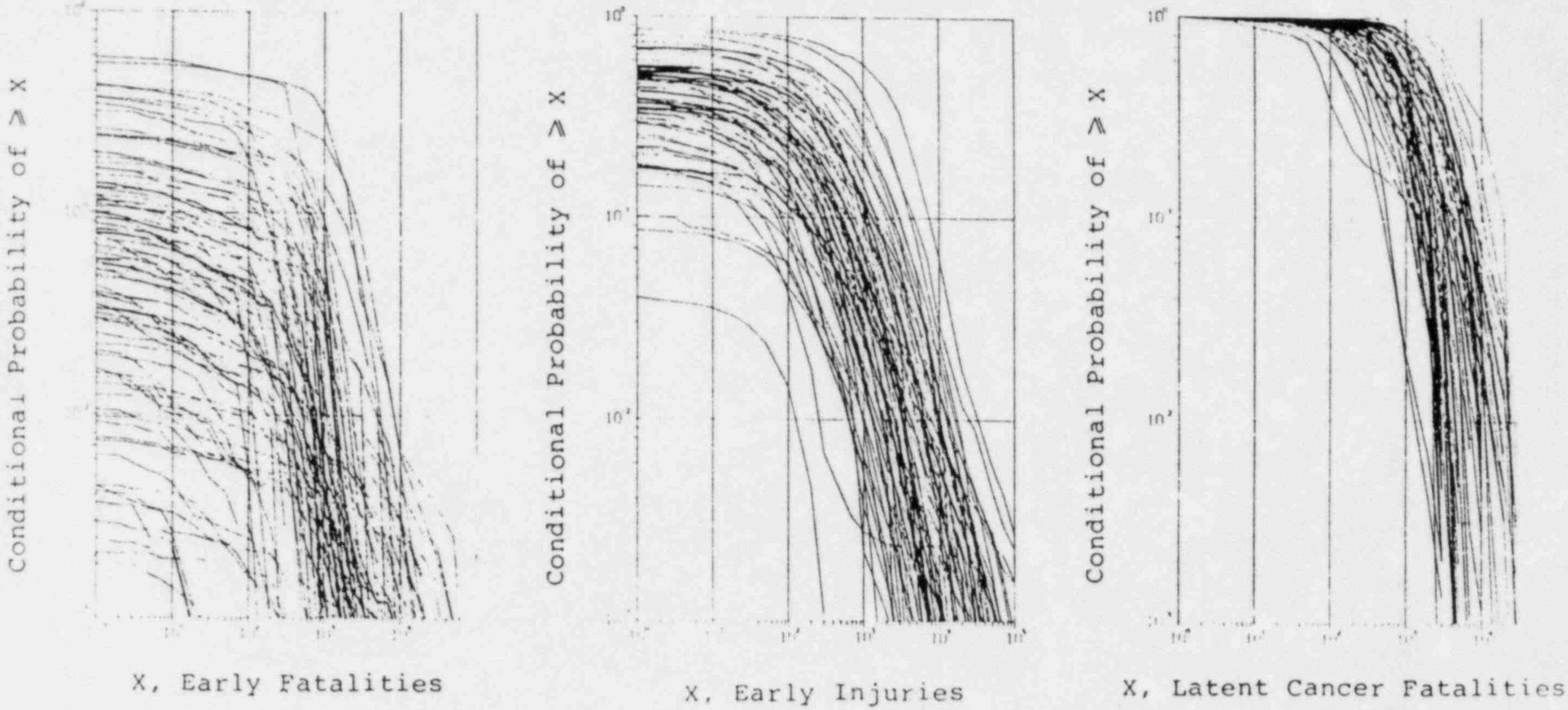

Figure 2.4.2-1. (a) Early Fatality, (b) Early Injury, and (c) Latent Cancer Fatality CCDFs Conditional on an SST1 Release at all 91 Current U.S. Reactor Sites. Assumptions: 1120 MWe reactor, Summary Evacuation, representative meteorology. Range of means: early fatalities 0.4 to 970, early injuries 4 to 3600, and latent cancer fatalities 230 to 8100.

A P P E N D I X C

Figures C-1 through C-18, NUREG/CR-2239, <u>Technical Guidance for Siting Criteria Development</u>, Sandia National Laboratories, December 1982. Provides CCDF curves, in groups of six sites, for early fatalities, early injuries, and latent cancer fatalities conditional on an SST-1 release. Values not corrected for actual power level; assumes an 1120 MWe reactor at each site.

Note: These CCDFs <u>do not</u> represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

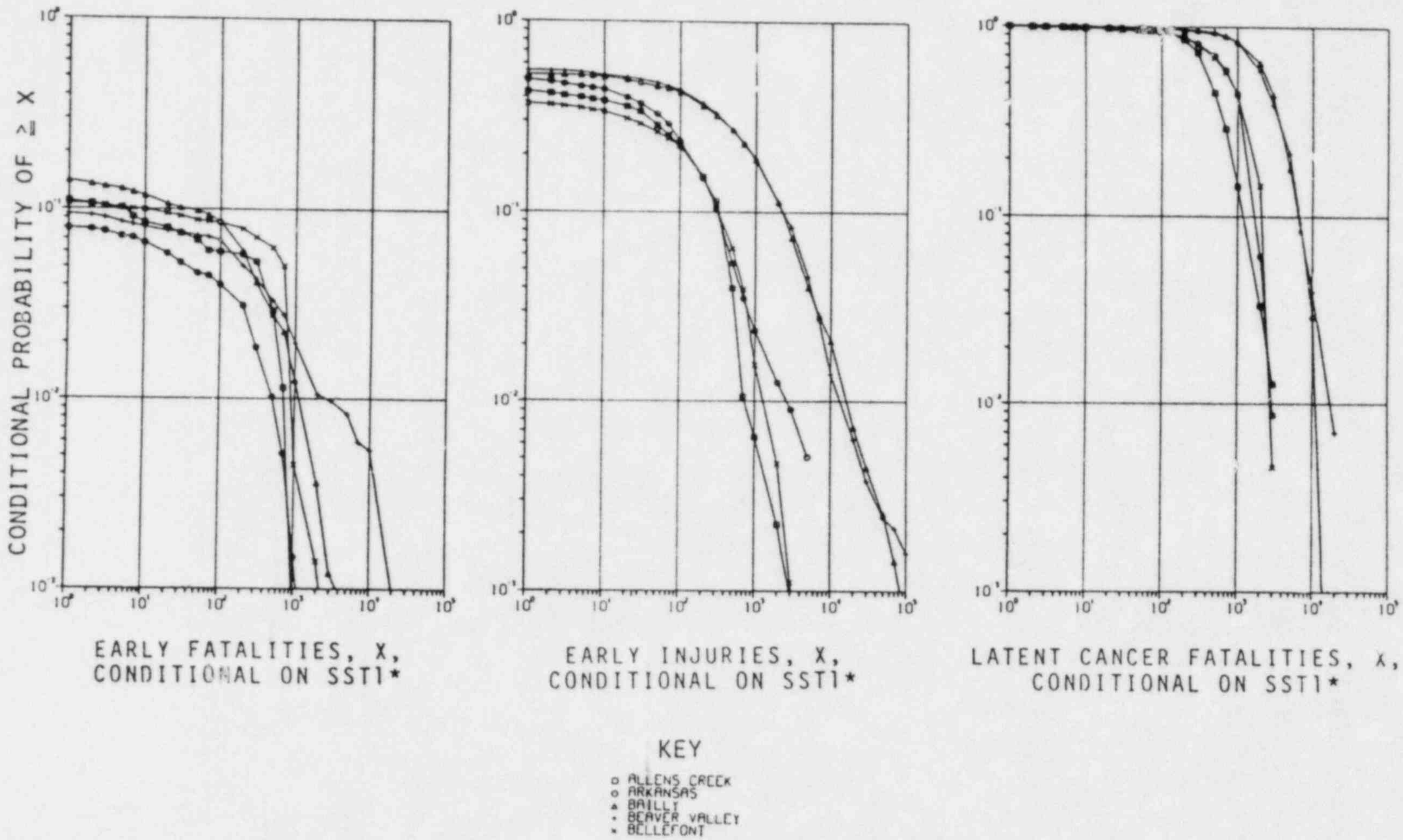

Figure C-1: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs <u>do not</u> represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

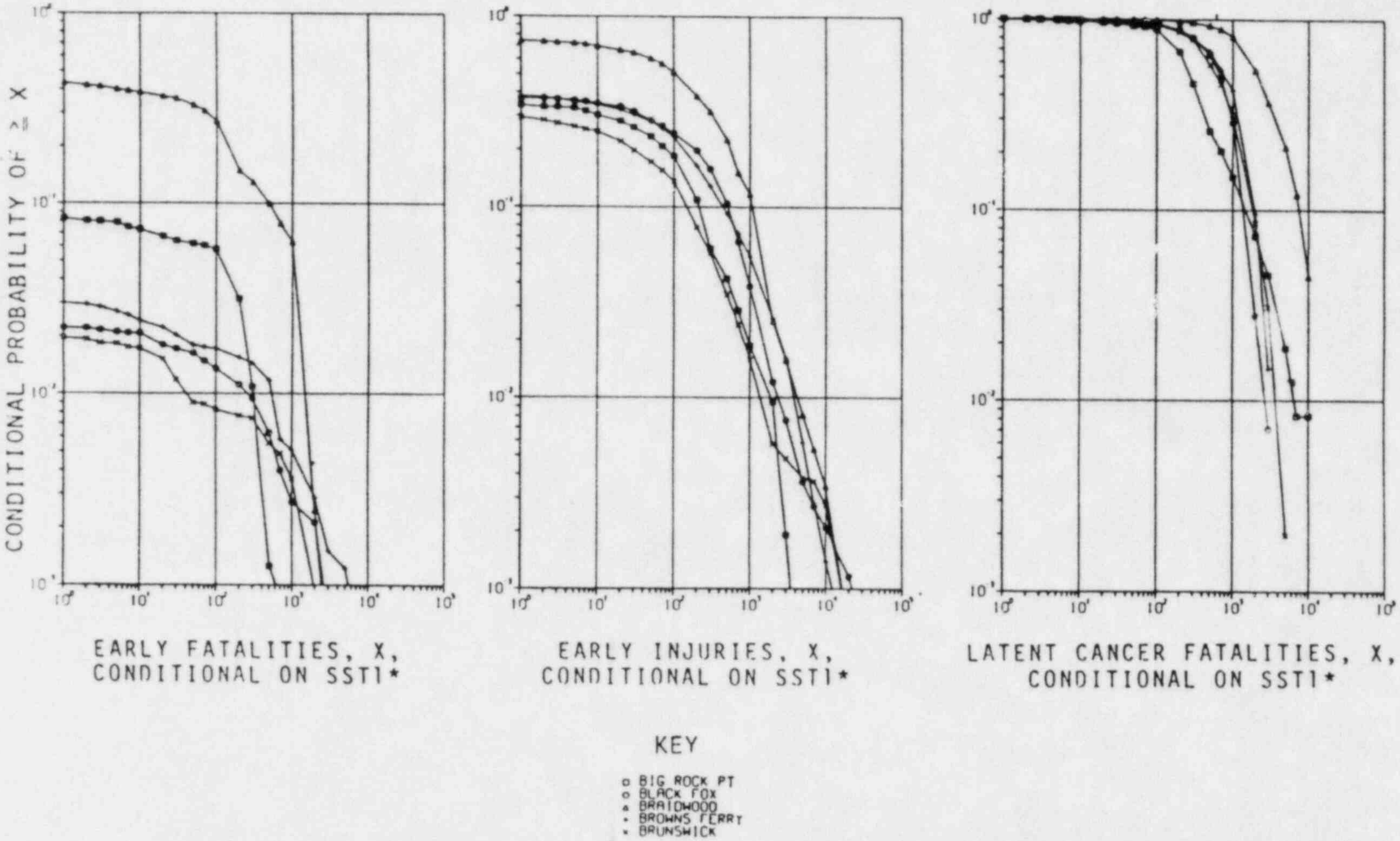

Figure C-2: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

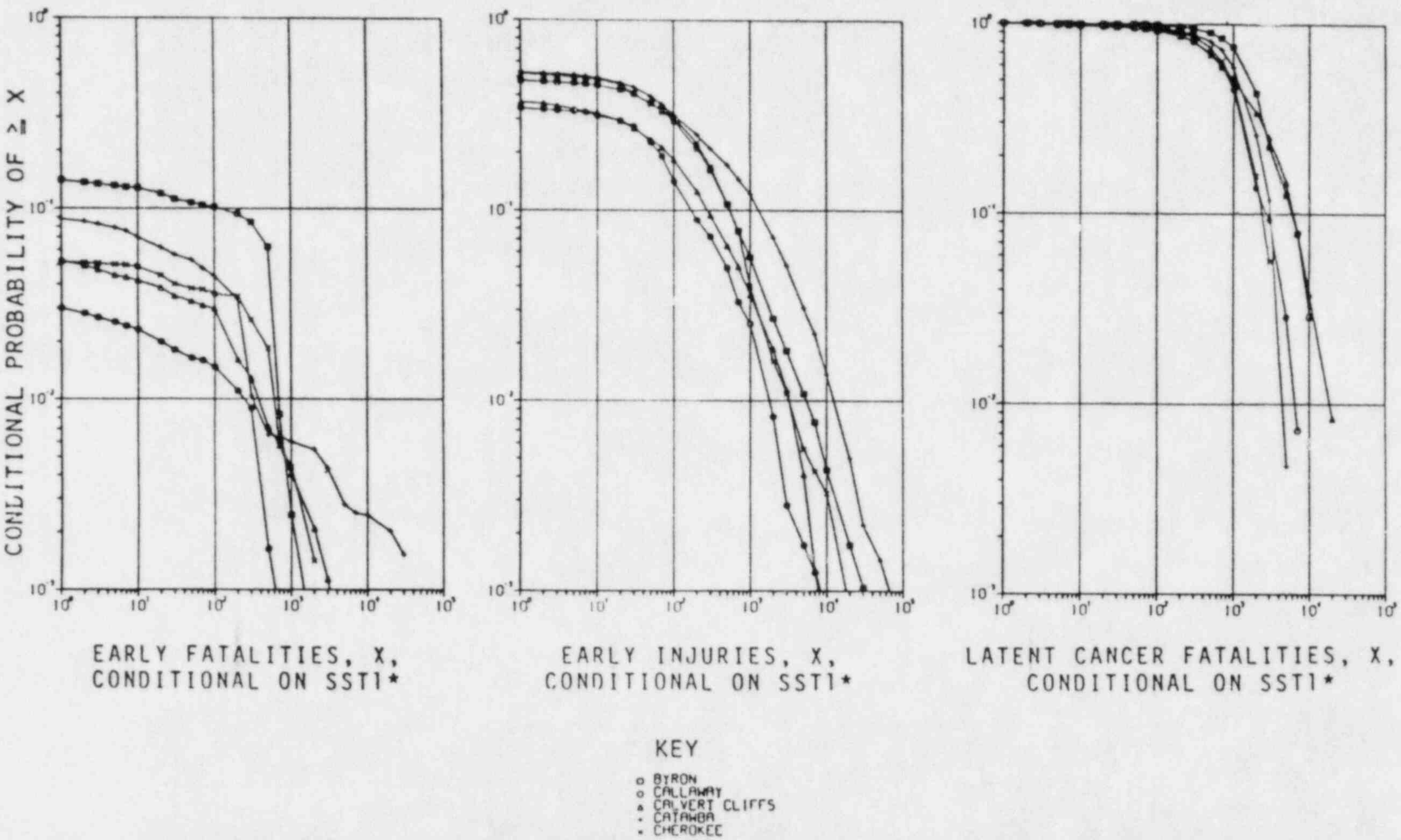

Figure C-3: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

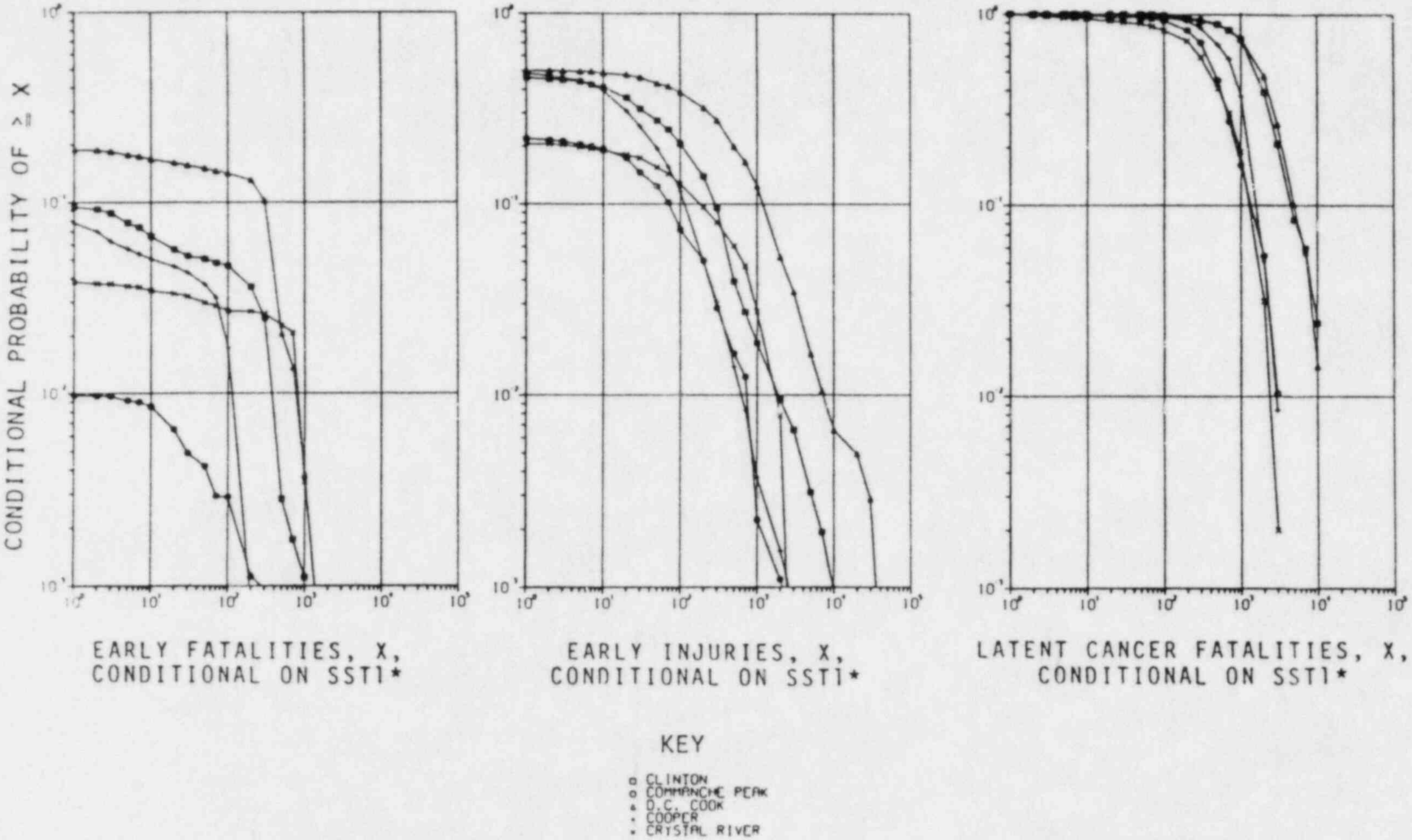

Figure C-4: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

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*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

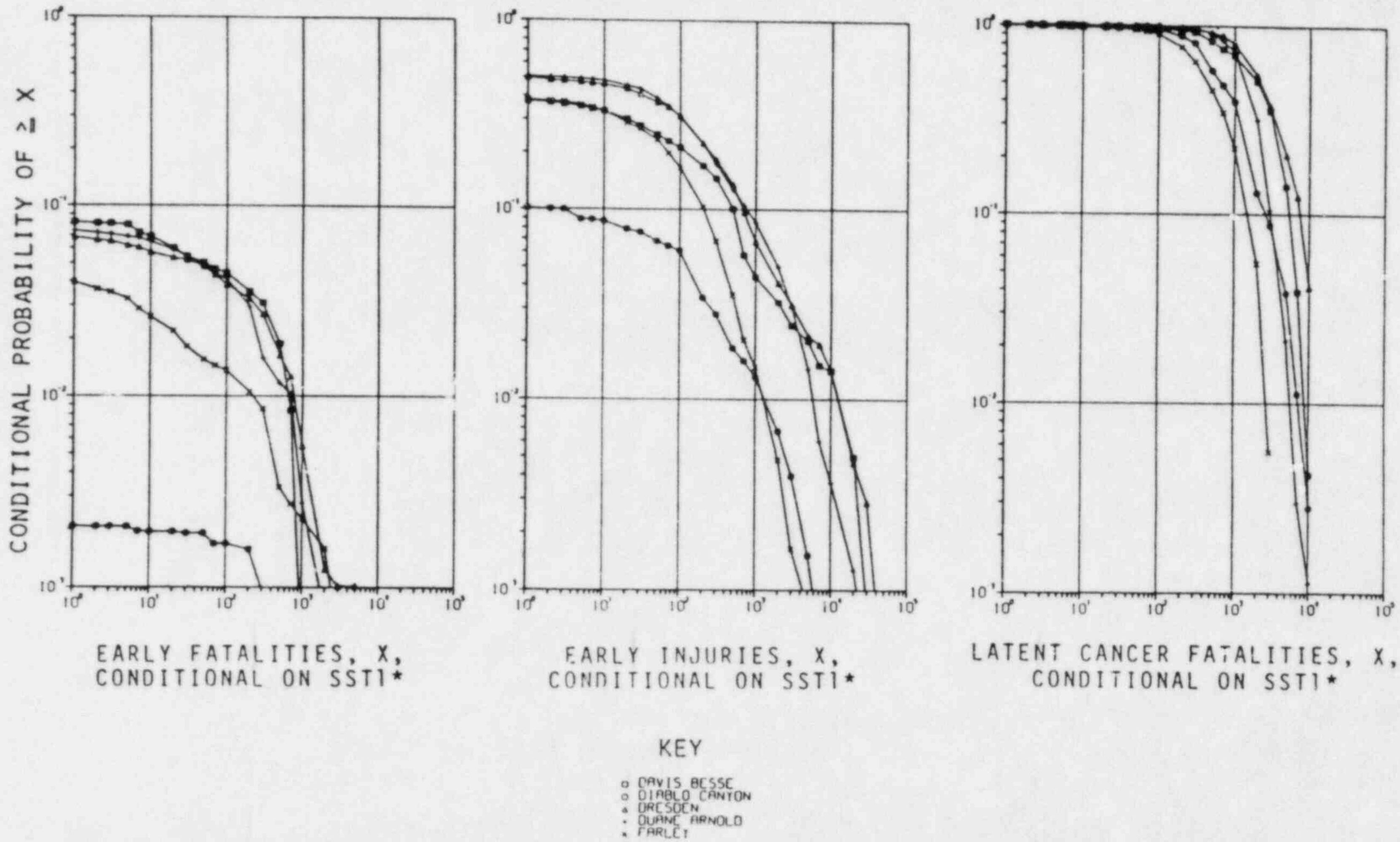

Figure C-5: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

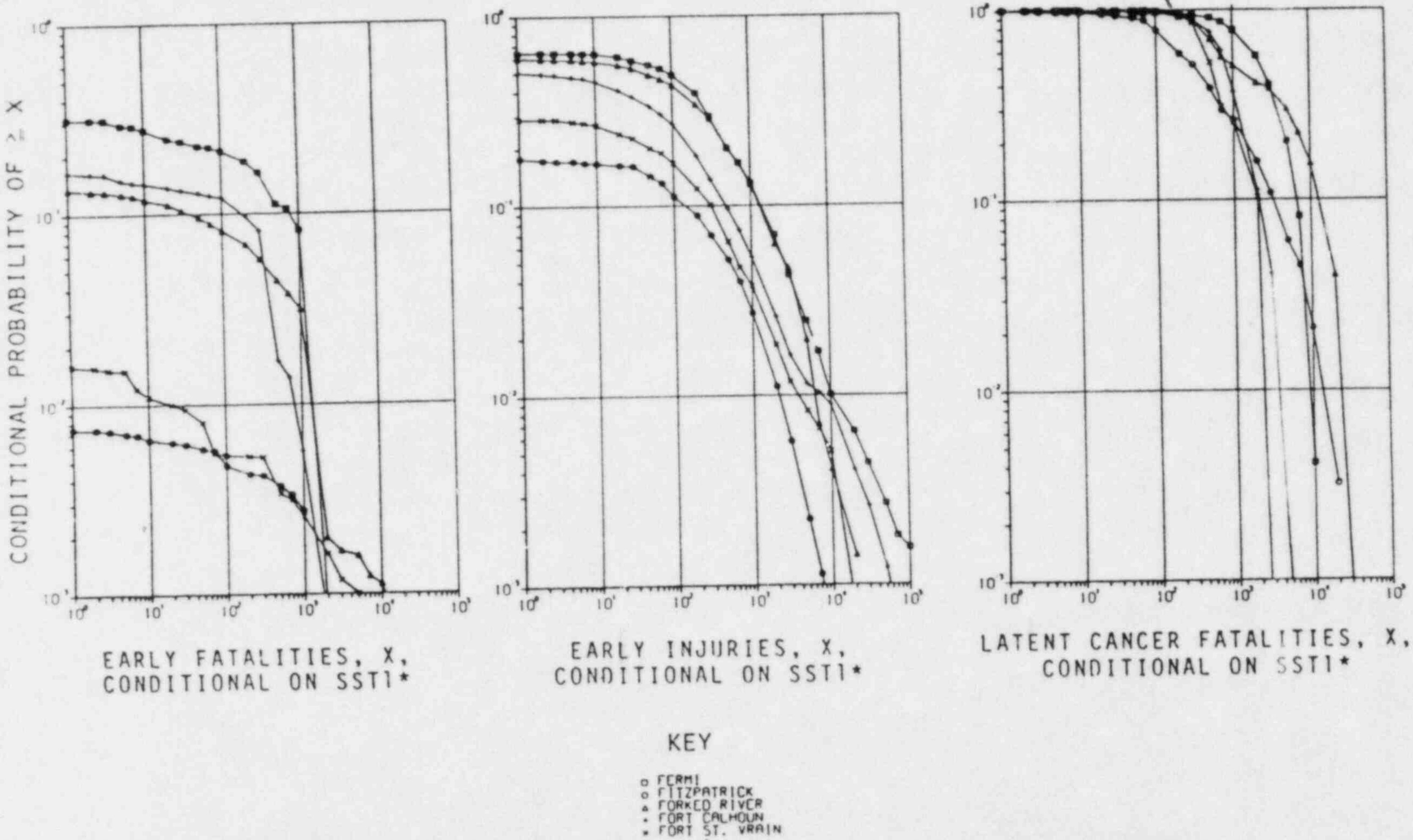

Figure C-6: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

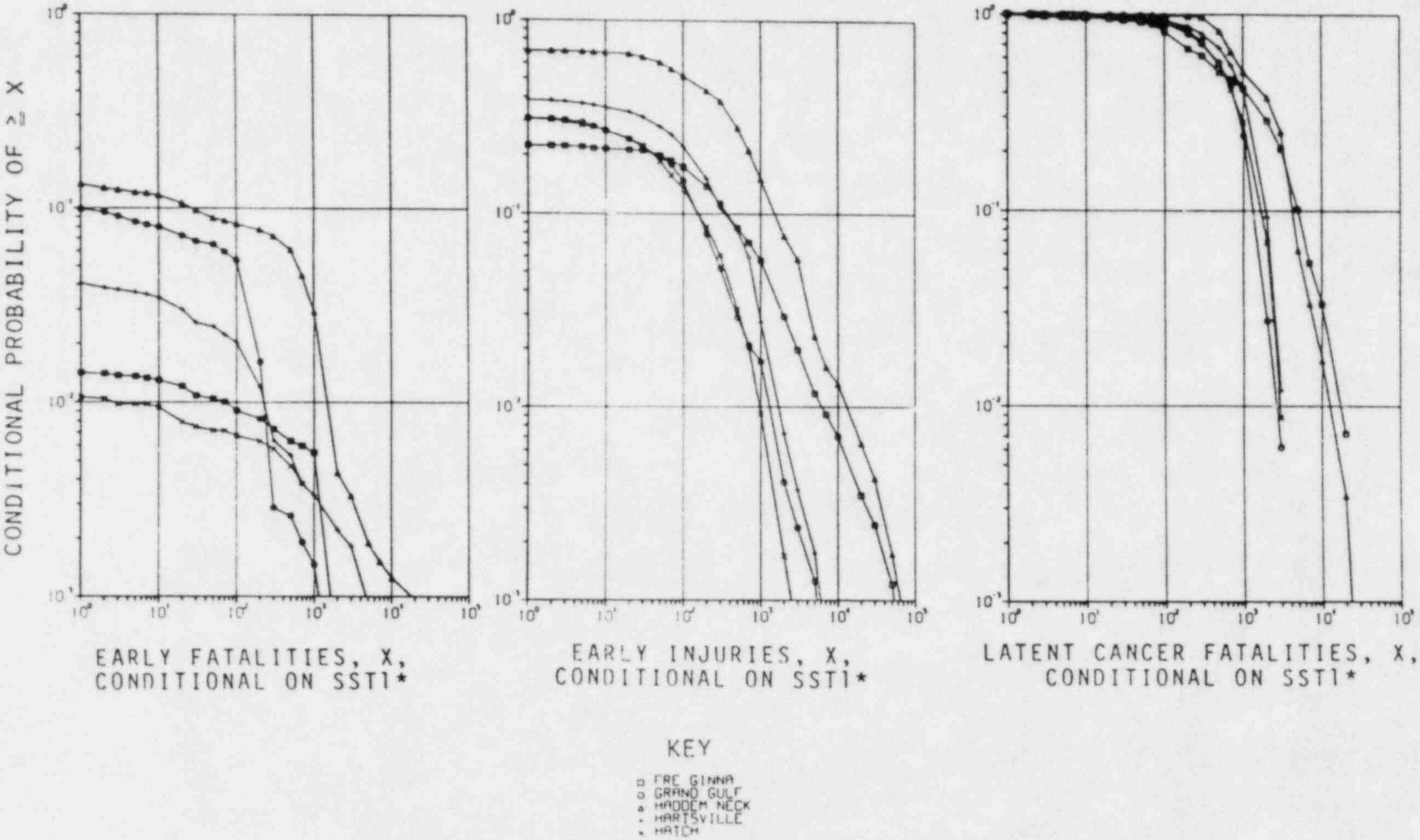

Figure C-7: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release. Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

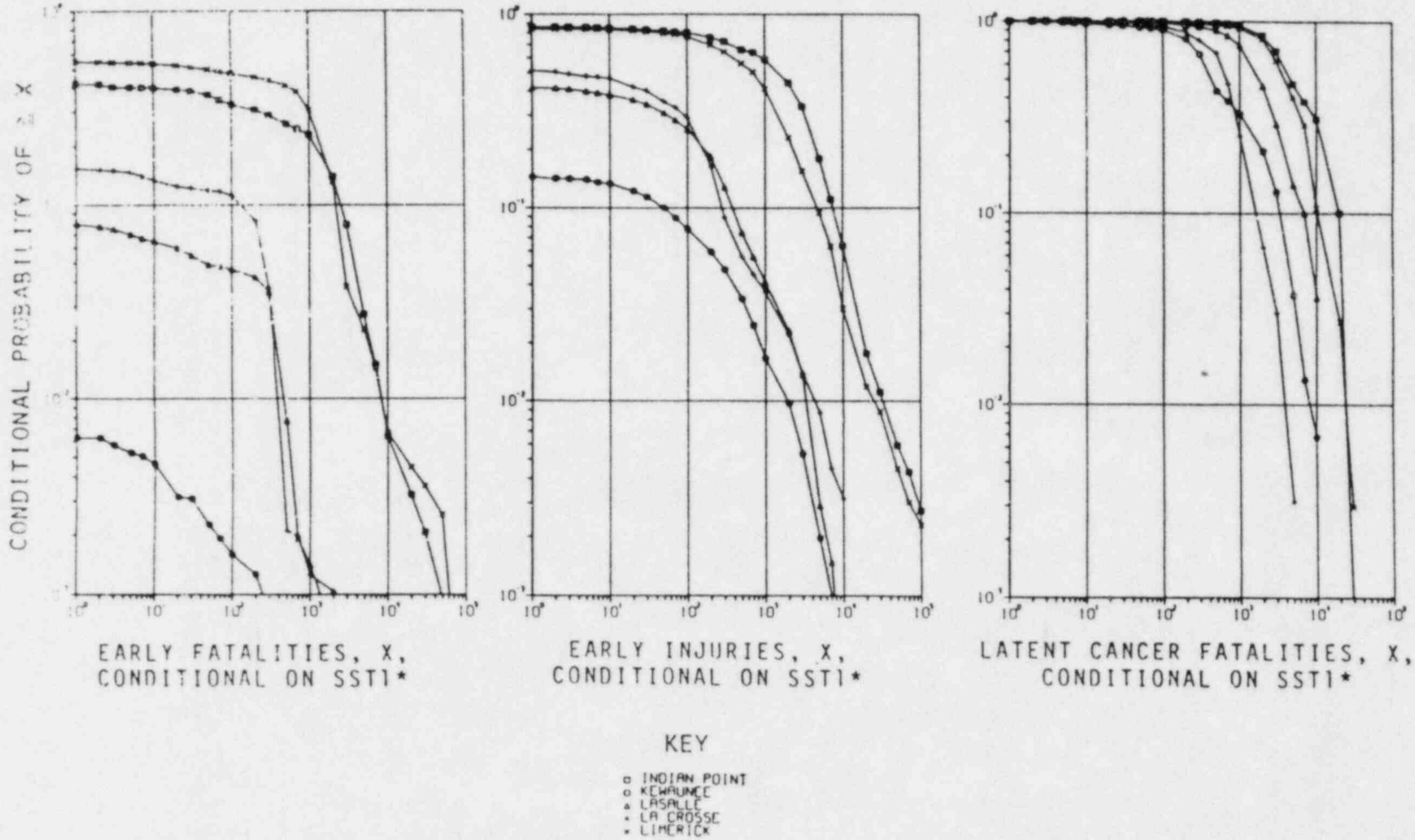

Figure C-8: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

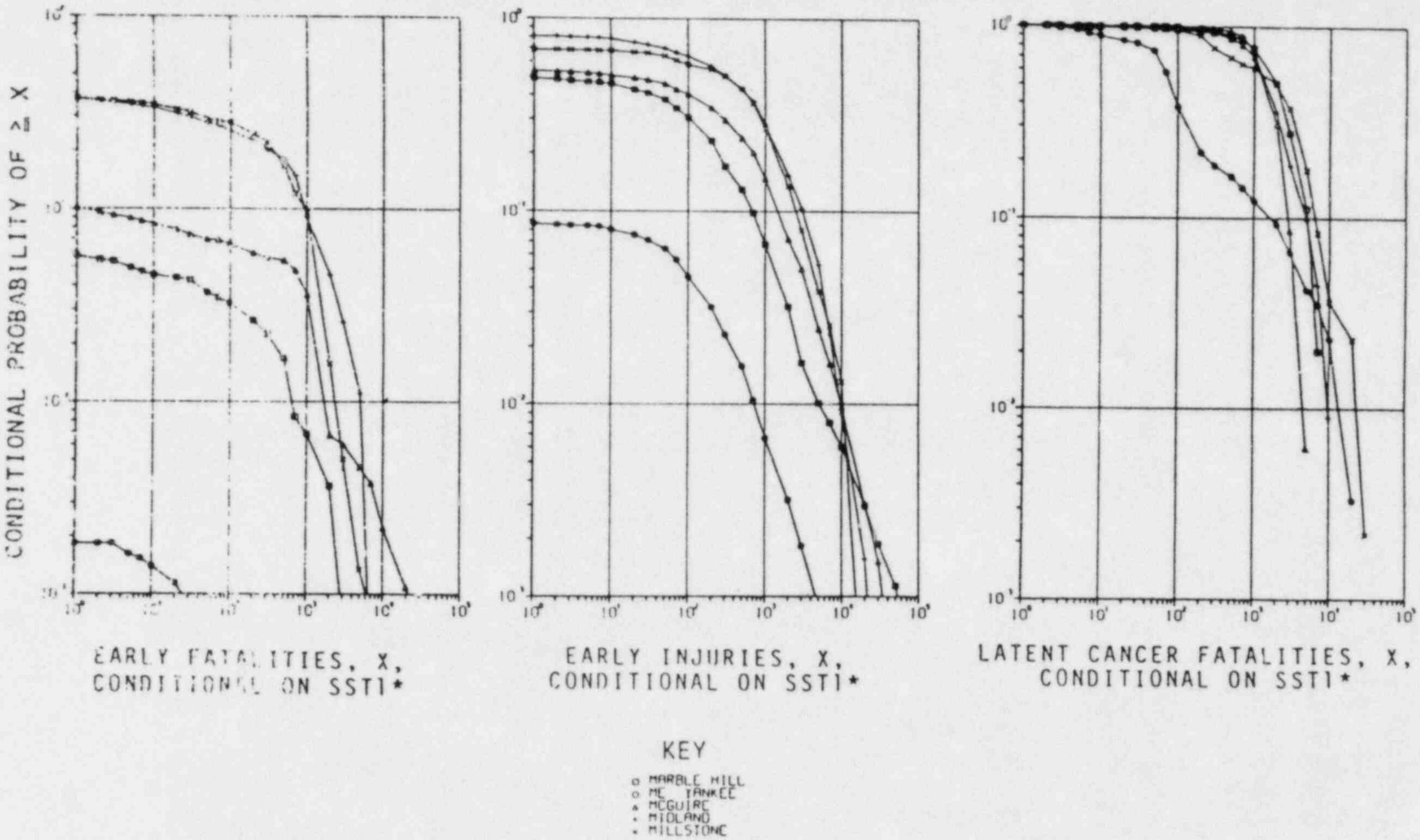

Figure C-9: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

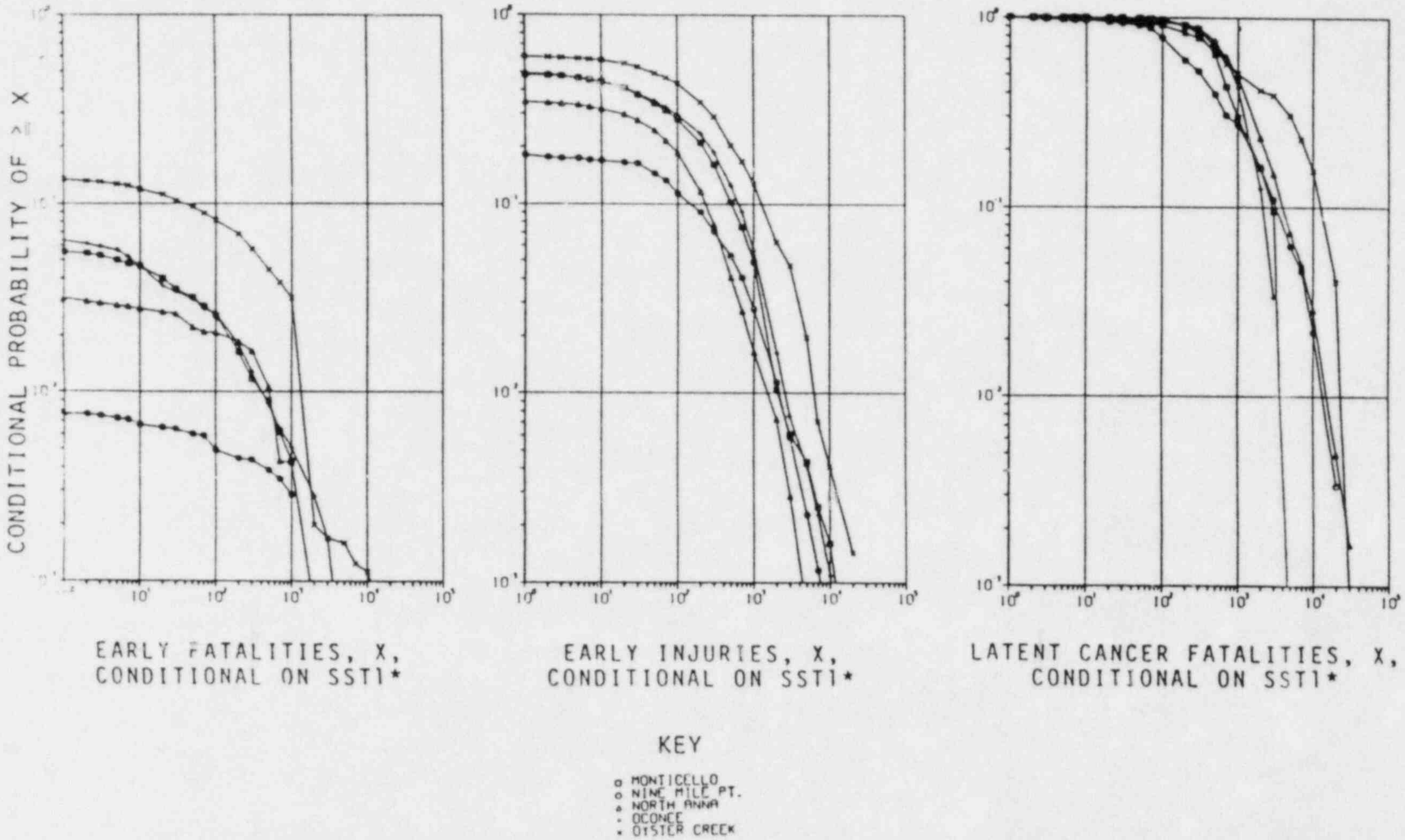

Figure C-10: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

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*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

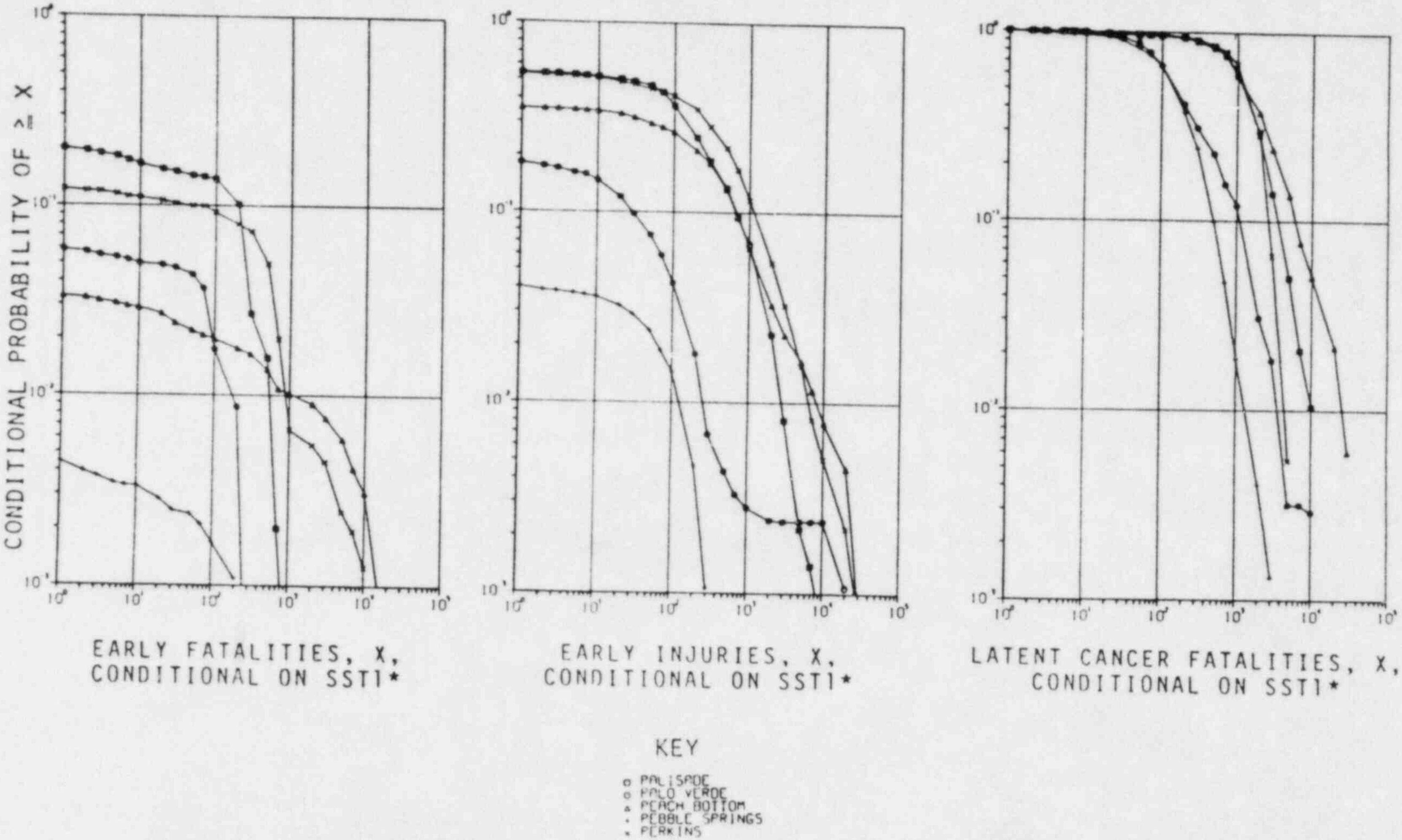


Figure C-11: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

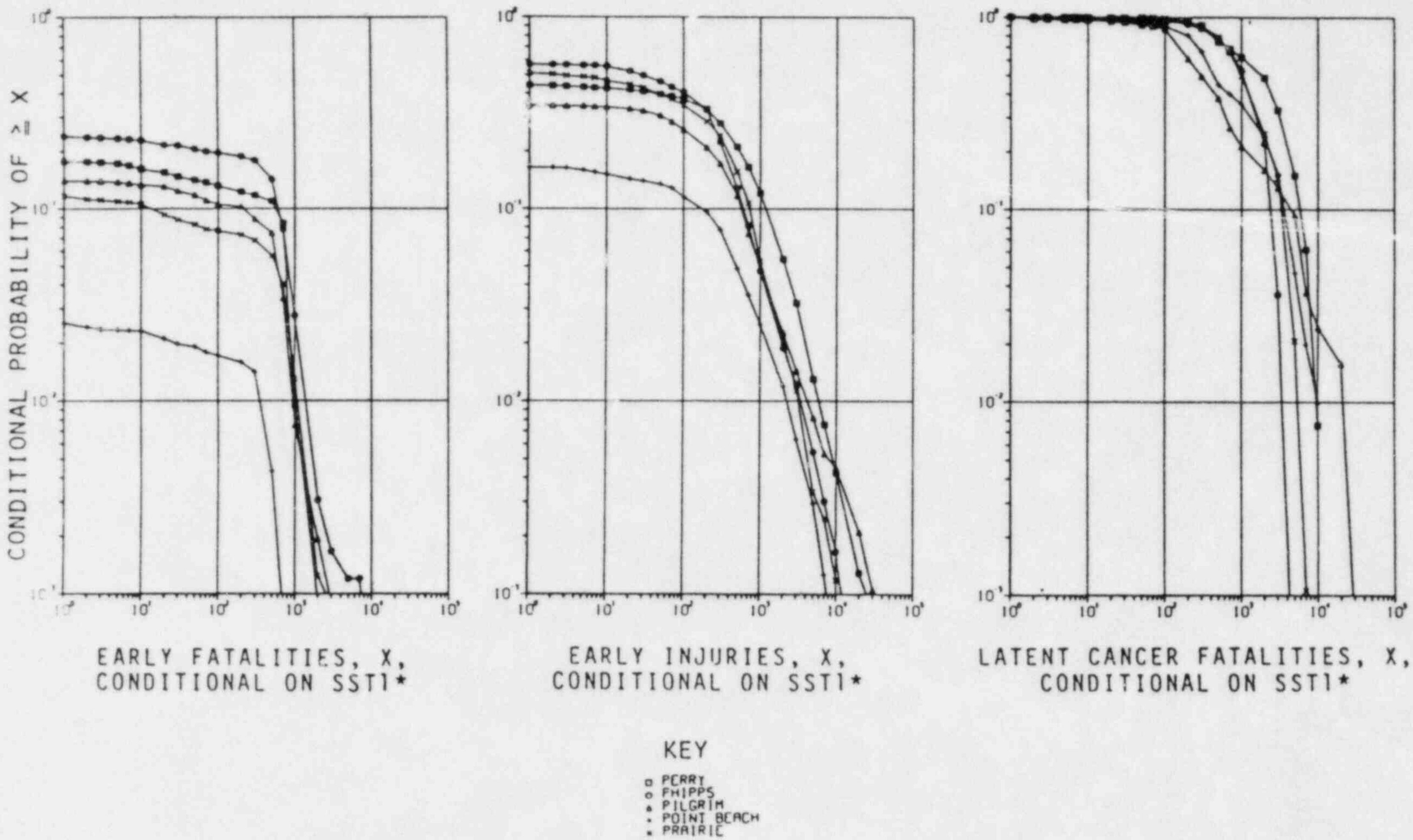


Figure C-12: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

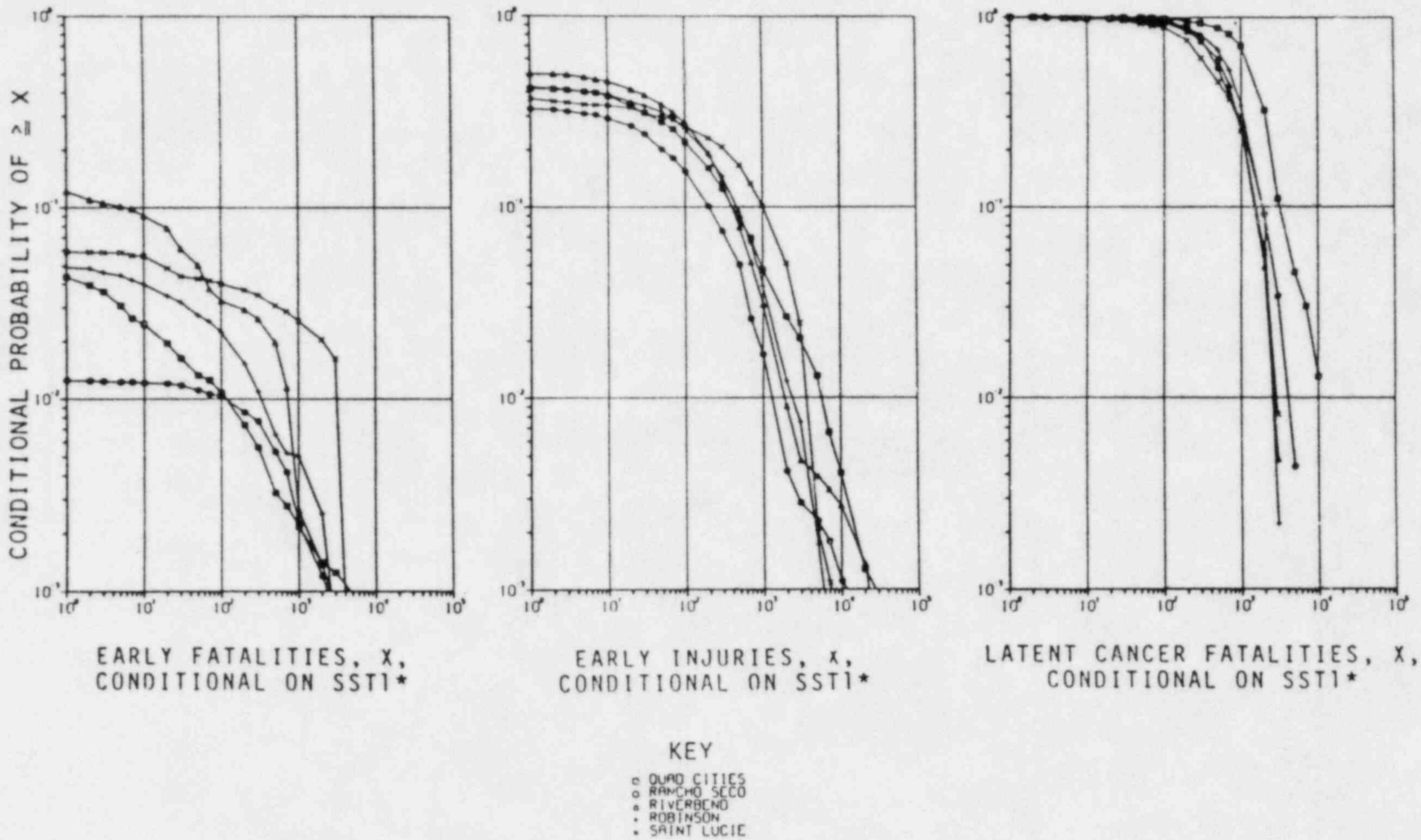


Figure C-13: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

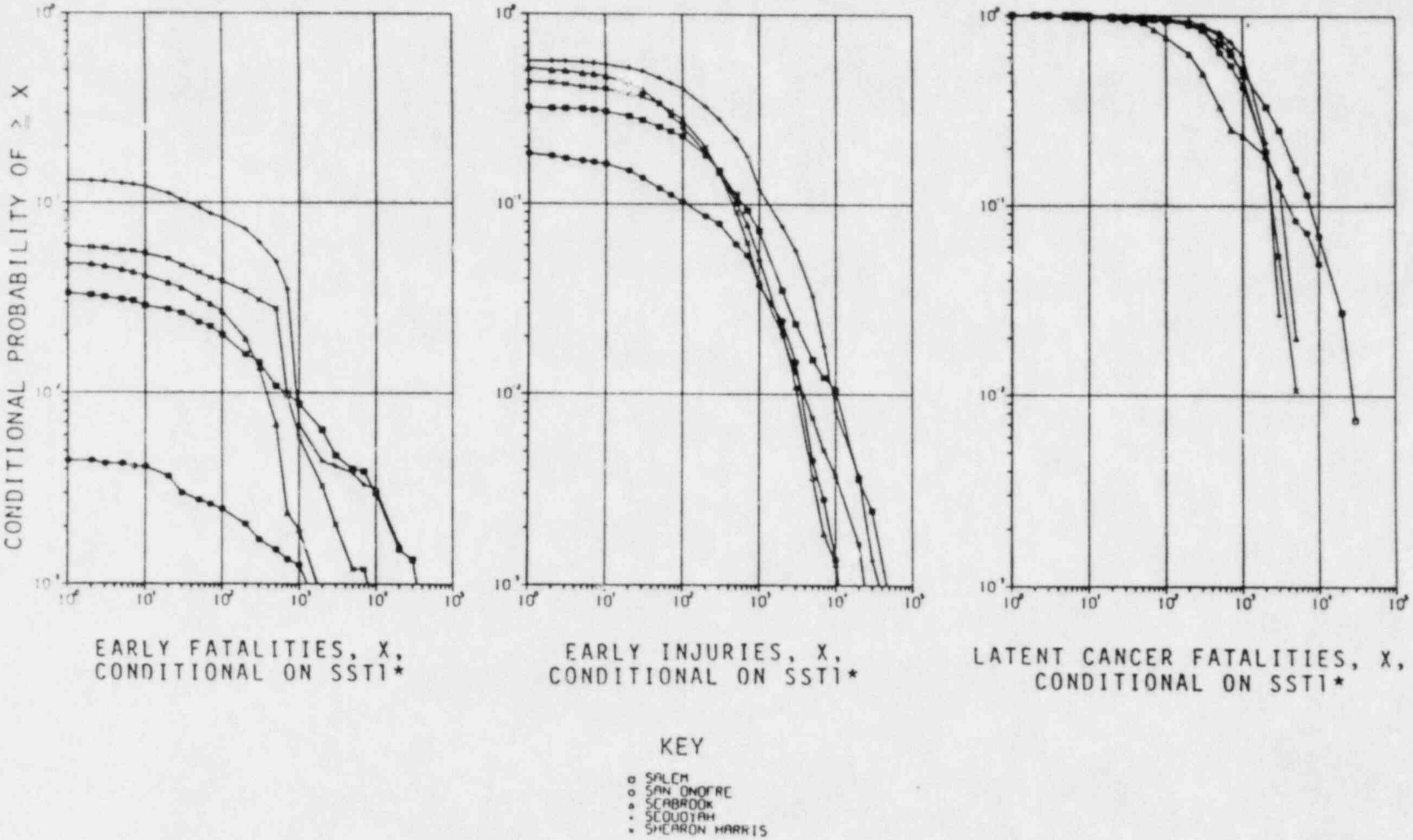


Figure C-14: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

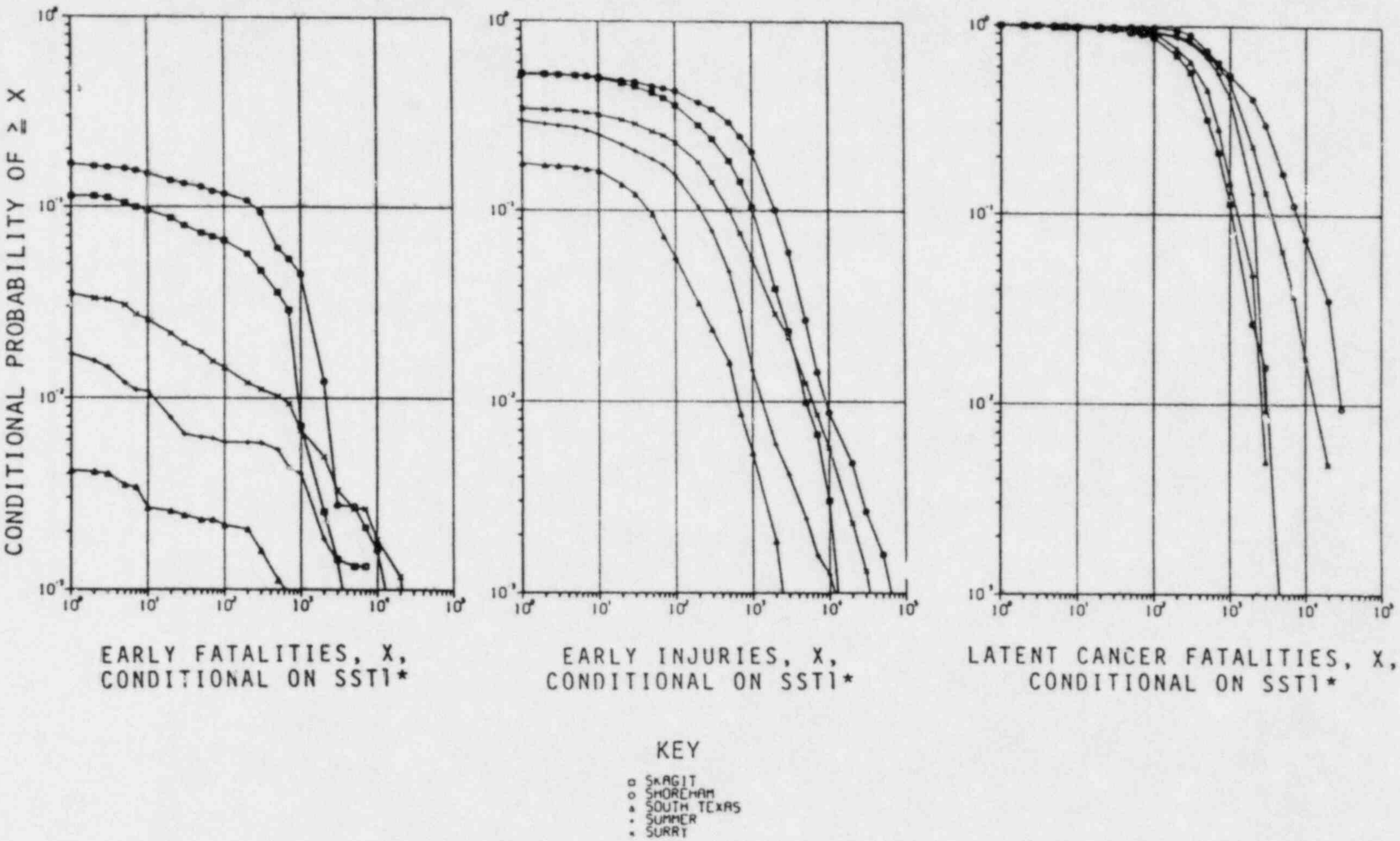


Figure C-15: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

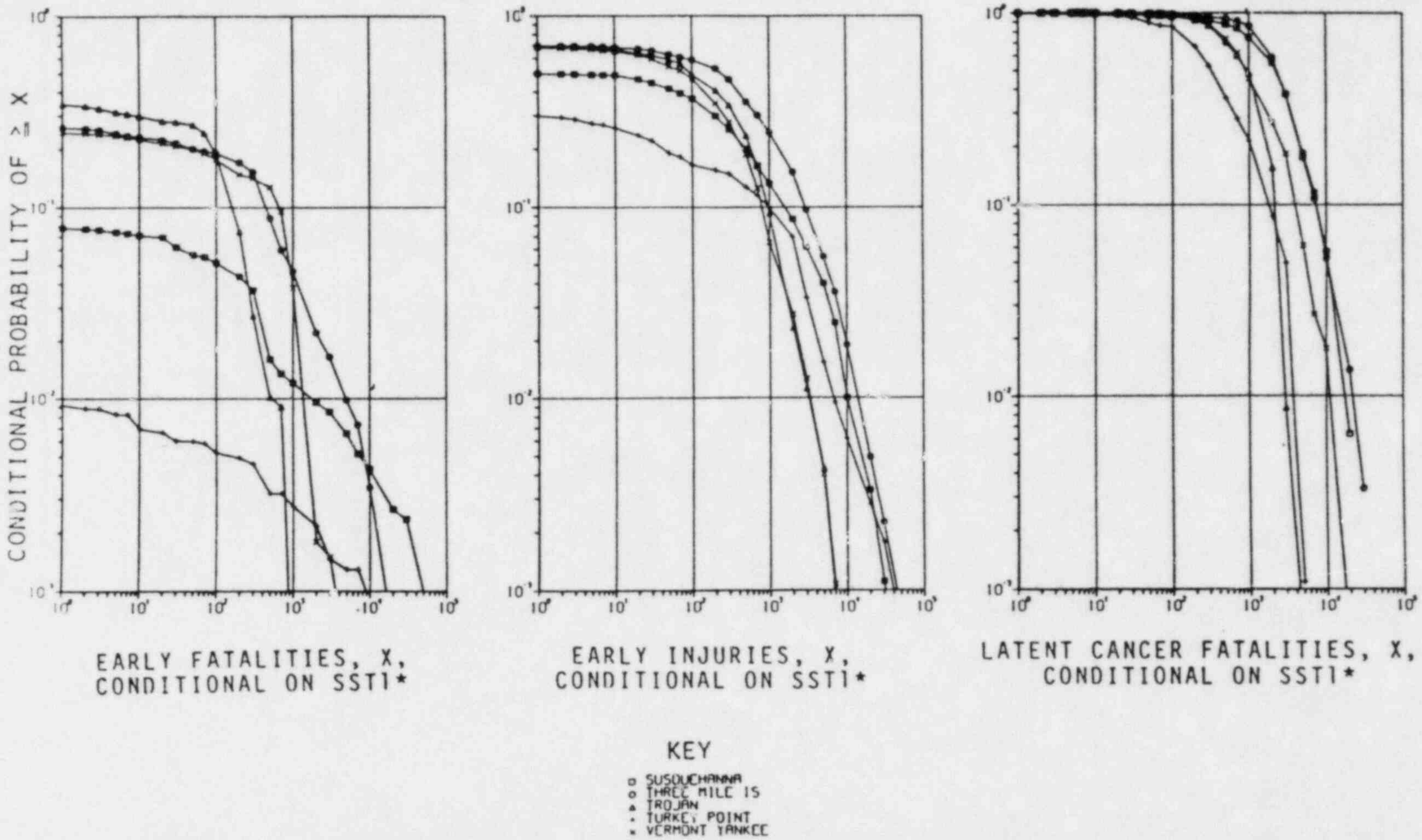


Figure C-16: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

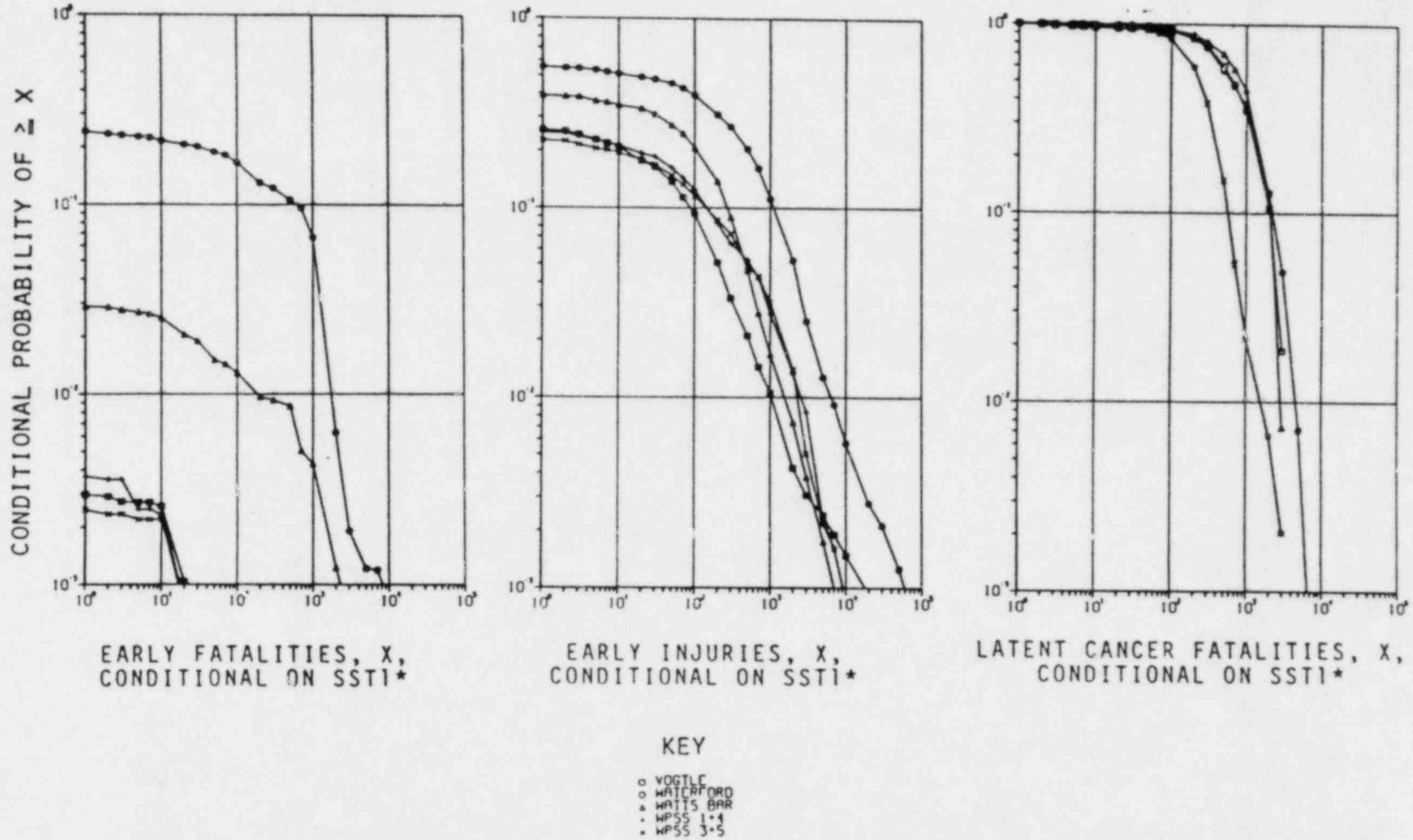


Figure C-17: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

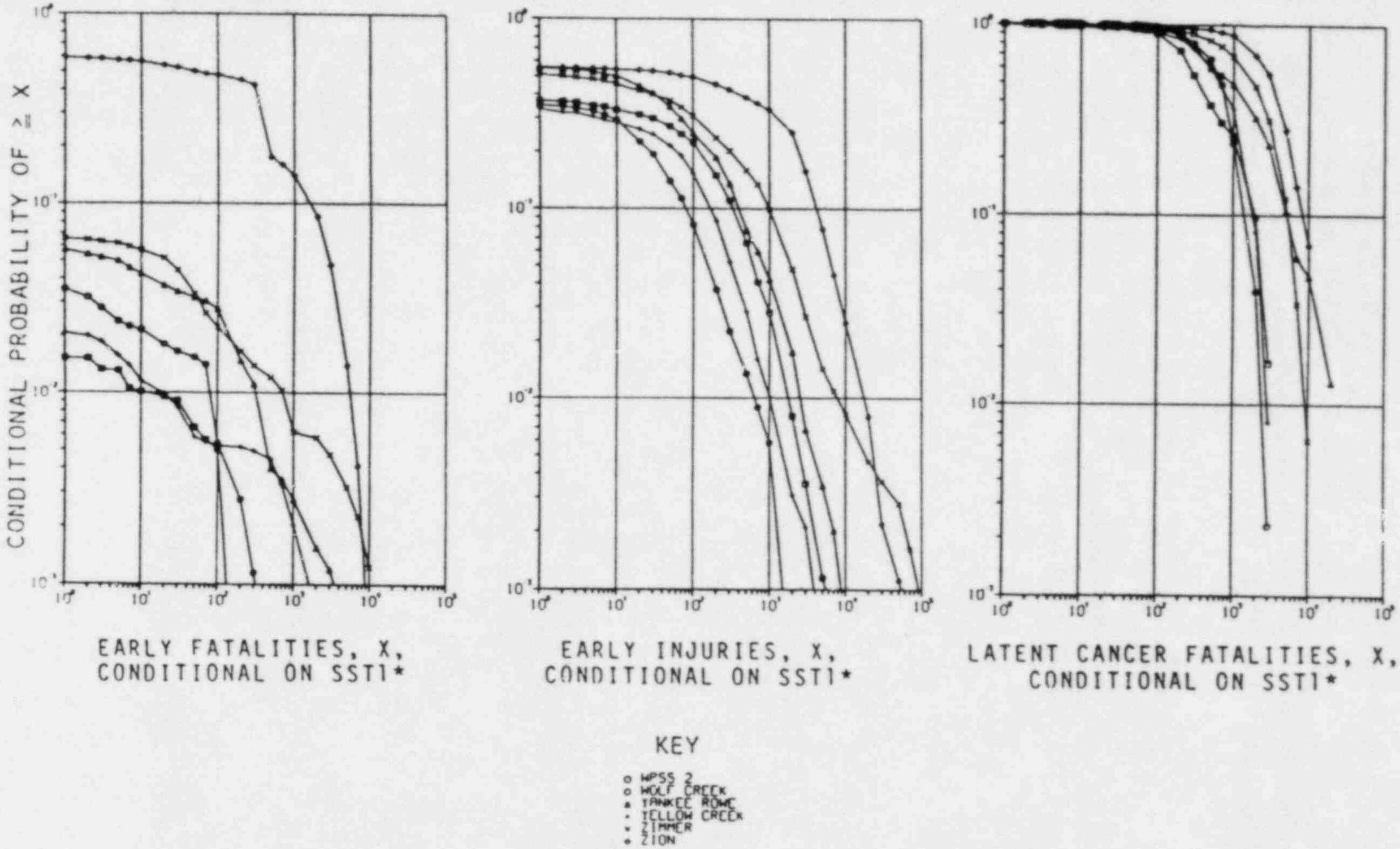


Figure C-18: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2

A P P E N D I X   D

Table C-1 from NUREG/CR-2239, Technical Guidance for Siting Criteria Development, Sandia National Laboratories, December 1982. Presents mean consequence values for early fatalities, early injuries, and latent cancer fatalities for 91 U.S. reactor sites, conditional on SST-1, SST-2, and SST-3 releases. Values not corrected for actual power level; assumes an 1120 MWe reactor at each site.

Table C-1. Mean Number (Per Reactor-Year) of Early Fatalities, Early Injuries and Latent Cancer Fatalities for each of 91 Sites, for SST1, SST2, or SST3 Accident Source Terms.

Assumptions:

(1) Standard 1120 MWe PWR

(2) Summary Evacuation

(3) Actual Site Population and Wind rose

(4) Best Estimate Meteorology

	Mean Early Fatalities*			Mean Early Injuries*			Mean Latent Cancer Fatalities*		
	SST1	SST2	SST3	SST1	SST2	SST3	SST1	SST2	SST3
Allens Creek	$31xP_1$	$0xP_2$	$0xP_3$	$93xP_1$	$0.9xP_2$	$0xP_3$	$620xP_1$	$49xP_2$	$0.2xP_3$
Arkansas	$17xP_1$	$0xP_2$	$0xP_3$	$150xP_1$	$0.2xP_2$	$0xP_3$	$950xP_1$	$82xP_2$	$0.3xP_3$
Bailly	$58xP_1$	$0xP_2$	$0xP_3$	$1200xP_1$	$0.5xP_2$	$0xP_3$	$3300xP_1$	$260xP_2$	$0.9xP_3$
Beaver Valley	$150xP_1$	$0xP_2$	$0xP_3$	$1200xP_1$	$0.4xP_2$	$0xP_3$	$3400xP_1$	$200xP_2$	$0.6xP_3$
Bellefonte	$63xP_1$	$0.08xP_2$	$0xP_3$	$110xP_1$	$5.6xP_2$	$0xP_3$	$1000xP_1$	$70xP_2$	$0.3xP_3$
Big Rock Pt.	$15xP_1$	$0xP_2$	$0xP_3$	$90xP_1$	$0.5xP_2$	$0xP_3$	$680xP_1$	$53xP_2$	$0.2xP_3$
Black Fox	$13xP_1$	$0xP_2$	$0xP_3$	$220xP_1$	$0.01xP_2$	$0xP_3$	$780xP_1$	$69xP_2$	$0.3xP_3$
Braidwood	$160xP_1$	$0.05xP_2$	$0xP_3$	$420xP_1$	$10xP_2$	$0xP_3$	$3200xP_1$	$240xP_2$	$0.9xP_3$
Browns Ferry	$25xP_1$	$0xP_2$	$0xP_3$	$220xP_1$	$0.03xP_2$	$0xP_3$	$970xP_1$	$69xP_2$	$0.3xP_3$
Brunswick	$12xP_1$	$0xP_2$	$0xP_3$	$120xP_1$	$0.01xP_2$	$0xP_3$	$890xP_1$	$98xP_2$	$0.4xP_3$

*Detailed Probabilistic Risk Assessments (PRAs) have not been performed for all reactors. Therefore, consequence calculations were performed in this study using Siting Source Terms (SSTs) defined by NRC (see Section 2.3.1, Chapter 2). By adjusting the probabilities associated with each of the source terms, the set can be made to approximately represent any current LWR design. Based on currently available PRAs, NRC has suggested that representative probabilities for the SSTs are: P_1 for SST1 = 1×10^{-5}, P_2 for SST2 = 2×10^{-5}, and P_3 for SST3 = 1×10^{-4}. There are very large variations (factors of 10 to 100) in the accident probabilities associated with a specific design.

Caution should be used when applying these numbers. Probability times consequence is not an adequate representation of risk; it provides only a common measure for comparative purposes (i.e., rank ordering). The Complementary Cumulative Distribution Functions (shown in Figure C-1 through C-18) are a better representation of risk.

Table C-1. (continued)

	Mean Early Fatalities*			Mean Early Injuries*			Mean Latent Cancer Fatalities*		
	SST1	SST2	SST3	SST1	SST2	SST3	SST1	SST2	SST3
Byron	$54xP_1$	$0.09xP_2$	$0xP_3$	$330xP_1$	$4.3xP_2$	$0xP_3$	$2500xP_1$	$190xP_2$	$0.7xP_3$
Callaway	$10xP_1$	$0xP_2$	$0xP_3$	$100xP_1$	$0.04xP_2$	$0xP_3$	$1200xP_1$	$97xP_2$	$0.3xP_3$
Calvert Cliffs	$18xP_1$	$0xP_2$	$0xP_3$	$170xP_1$	$0.08xP_2$	$0xP_3$	$2400xP_1$	$120xP_2$	$0.4xP_3$
Catawba	$100xP_1$	$0xP_2$	$0xP_3$	$710xP_1$	$0.2xP_2$	$0xP_3$	$1500xP_1$	$110xP_2$	$0.4xP_3$
Cherokee	$27xP_1$	$0xP_2$	$0xP_3$	$250xP_1$	$0.1xP_2$	$0xP_3$	$1200xP_1$	$76xP_2$	$0.3xP_3$
Clinton	$16xP_1$	$0xP_2$	$0xP_3$	$130xP_1$	$0.7xP_2$	$0xP_3$	$2300xP_1$	$170xP_2$	$0.7xP_3$
Comanche Peak	$1.3xP_1$	$0xP_2$	$0xP_3$	$37xP_1$	$0xP_2$	$0xP_3$	$640xP_1$	$49xP_2$	$0.2xP_3$
Cooper	$4.7xP_1$	$0xP_2$	$0xP_3$	$47xP_1$	$0.09xP_2$	$0xP_3$	$900xP_1$	$81xP_2$	$0.3xP_3$
Crystal River	$21xP_1$	$0xP_2$	$0xP_3$	$88xP_1$	$0.9xP_2$	$0xP_3$	$590xP_1$	$66xP_2$	$0.3xP_3$
Davis-Besse	$21xP_1$	$0xP_2$	$0xP_3$	$420xP_1$	$0.6xP_2$	$0xP_3$	$2600xP_1$	$160xP_2$	$0.5xP_3$
Diablo Canyon	$4.7xP_1$	$0xP_2$	$0xP_3$	$50xP_1$	$0xP_2$	$0xP_3$	$1200xP_1$	$98xP_2$	$0.4xP_3$
Donald C. Cook	$55xP_1$	$0.04xP_2$	$0xP_3$	$590xP_1$	$2.2xP_2$	$0xP_3$	$2500xP_1$	$120xP_2$	$0.4xP_3$
Dresden	$42xP_1$	$0xP_2$	$0xP_3$	$540xP_1$	$0.3xP_2$	$0xP_3$	$3300xP_1$	$260xP_2$	$0.9xP_3$
Duane Arnold	$21xP_1$	$0xP_2$	$0xP_3$	$380xP_1$	$0.4xP_2$	$0xP_3$	$1700xP_1$	$190xP_2$	$0.8xP_3$
Fermi	$160xP_1$	$0.08xP_2$	$0xP_3$	$970xP_1$	$7.1xP_2$	$0xP_3$	$3000xP_1$	$200xP_2$	$0.6xP_3$
Fitzpatrick	$5.0xP_1$	$0xP_2$	$0xP_3$	$110xP_1$	$0.06xP_2$	$0xP_3$	$1200xP_1$	$57xP_2$	$0.2xP_3$
Forked River	$84xP_1$	$0xP_2$	$0xP_3$	$530xP_1$	$0.8xP_2$	$0xP_3$	$4400xP_1$	$200xP_2$	$0.6xP_3$
Fort Calhoun	$50xP_1$	$0.1xP_2$	$0xP_3$	$440xP_1$	$3.0xP_2$	$0xP_3$	$1100xP_1$	$110xP_2$	$0.4xP_3$
Ft. St. Vrain	$15xP_1$	$0xP_2$	$0xP_3$	$220xP_1$	$0xP_2$	$0xP_3$	$810xP_1$	$82xP_2$	$0.3xP_3$
Ginna	$11xP_1$	$0xP_2$	$0xP_3$	$370xP_1$	$0.1xP_2$	$0xP_3$	$1900xP_1$	$89xP_2$	$0.3xP_3$
Grand Gulf	$14xP_1$	$0xP_2$	$0xP_3$	$73xP_1$	$0.7xP_2$	$0xP_3$	$700xP_1$	$60xP_2$	$0.3xP_3$
Haddem Neck	$110xP_1$	$0xP_2$	$0xP_3$	$890xP_1$	$1.2xP_2$	$0xP_3$	$2100xP_1$	$160xP_2$	$0.5xP_3$

*See footnote, page C-2.

Table C-1. (continued)

	Mean Early Fatalities*			Mean Early Injuries*			Mean Latent Cancer Fatalities*		
	SST1	SST2	SST3	SST1	SST2	SST3	SST1	SST2	SST3
Hartsville	$19xP_1$	$0xP_2$	$0xP_3$	$140xP_1$	$0.04xP_2$	$0xP_3$	$970xP_1$	$64xP_2$	$0.2xP_3$
Hatch	$4xP_1$	$0xP_2$	$0xP_3$	$62xP_1$	$0.04xP_2$	$0xP_3$	$770xP_1$	$64xP_2$	$0.2xP_3$
Hope Creek	$120xP_1$	$0xP_2$	$0xP_3$	$440xP_1$	$0xP_2$	$0xP_3$	$3000xP_1$	$160xP_2$	$0.5xP_3$
Indian Pt.	$830xP_1$	$0.08xP_2$	$0xP_3$	$3600xP_1$	$18xP_2$	$0xP_3$	$8100xP_1$	$590xP_2$	$1.8xP_3$
Joseph M. Farley	$12xP_1$	$0xP_2$	$0xP_3$	$85xP_1$	$0.03xP_2$	$0xP_3$	$670xP_1$	$56xP_2$	$0.2xP_3$
Kewaunee	$1.2xP_1$	$0xP_2$	$0xP_3$	$78xP_1$	$0xP_2$	$0xP_3$	$1200xP_1$	$70xP_2$	$0.3xP_3$
LaCrosse	$32xP_1$	$0xP_2$	$0xP_3$	$200xP_1$	$1.8xP_2$	$0xP_3$	$850xP_1$	$58xP_2$	$0.2xP_3$
La Salle	$26xP_1$	$0xP_2$	$0xP_3$	$180xP_1$	$0.6xP_2$	$0xP_3$	$2800xP_1$	$200xP_2$	$0.7xP_3$
Limerick	$970xP_1$	$2.2xP_2$	$0xP_3$	$2800xP_1$	$6.6xP_2$	$0xP_3$	$5400xP_1$	$370xP_2$	$1.3xP_3$
Maine Yankee	$4.1xP_1$	$0xP_2$	$0xP_3$	$34xP_1$	$0xP_2$	$0xP_3$	$770xP_1$	$29xP_2$	$0.1xP_3$
Marble Hill	$28xP_1$	$0xP_2$	$0xP_3$	$420xP_1$	$0xP_2$	$0xP_3$	$2400xP_1$	$180xP_2$	$0.7xP_3$
McGuire	$130xP_1$	$0xP_2$	$0xP_3$	$680xP_1$	$0xP_2$	$0xP_3$	$1600xP_1$	$130xP_2$	$0.5xP_3$
Midland	$320xP_1$	$0.2xP_2$	$0xP_3$	$1100xP_1$	$1.3xP_2$	$0xP_3$	$2200xP_1$	$130xP_2$	$0.5xP_3$
Millstone	$240xP_1$	$0.02xP_2$	$0xP_3$	$990xP_1$	$4.5xP_2$	$0xP_3$	$3200xP_1$	$160xP_2$	$0.6xP_3$
Monticello	$12xP_1$	$0xP_2$	$0xP_3$	$200xP_1$	$0.08xP_2$	$0xP_3$	$1100xP_1$	$98xP_2$	$0.4xP_3$
Nine Mile Pt.	$5.2xP_1$	$0xP_2$	$0xP_3$	$110xP_1$	$0.06xP_2$	$0xP_3$	$1200xP_1$	$58xP_2$	$0.2xP_3$*
North Anna	$14xP_1$	$0xP_2$	$0xP_3$	$92xP_1$	$0.08xP_2$	$0xP_3$	$1800xP_1$	$75xP_2$	$0.3xP_3$
Oconee	$2.0xP_1$	$0xP_2$	$0xP_3$	$240xP_1$	$0.03xP_2$	$0xP_3$	$1100xP_1$	$70xP_2$	$0.3xP_3$
Oyster Creek	$84xP_1$	$0xP_2$	$0xP_3$	$530xP_1$	$0.8xP_2$	$0xP_3$	$4400xP_1$	$200xP_2$	$0.6xP_3$
Palisades	$37xP_1$	$0.02xP_2$	$0xP_3$	$250xP_1$	$1.3xP_2$	$0xP_3$	$1700xP_1$	$90xP_2$	$0.3xP_3$
Palo Verde	$5.8xP_1$	$0xP_2$	$0xP_3$	$59xP_1$	$0.2xP_2$	$0xP_3$	$450xP_1$	$26xP_2$	$0.09xP_3$
Peach Bottom	$97xP_1$	$0xP_2$	$0xP_3$	$400xP_1$	$0.02xP_2$	$0xP_3$	$2800xP_1$	$140xP_2$	$0.4xP_3$

*See footnote, page C-2.

Table C-1. (continued)

	Mean Early Fatalities*			Mean Early Injuries*			Mean Latent Cancer Fatalities*		
	SST1	SST2	SST3	SST1	SST2	SST3	SST1	SST2	SST3
Pebble Springs	$0.41xP_1$	$0xP_2$	$0xP_3$	$3.7xP_1$	$0xP_2$	$0xP_3$	$230xP_1$	$18xP_2$	$0.07xP_3$
Perkins	$98xP_1$	$0xP_2$	$0xP_3$	$520xP_1$	$2.1xP_2$	$0xP_3$	$1500xP_1$	$120xP_2$	$0.5xP_3$
Perry	$95xP_1$	$0.07xP_2$	$0xP_3$	$520xP_1$	$4.2xP_2$	$0xP_3$	$2500xP_1$	$160xP_2$	$0.6xP_3$
Phipps Bed	$170xP_1$	$0.3xP_2$	$0xP_3$	$300xP_1$	$16xP_2$	$0xP_3$	$1300xP_1$	$82xP_2$	$0.3xP_3$
Pilgrim	$71xP_1$	$0.02xP_2$	$0xP_3$	$300xP_1$	$2.4xP_2$	$0xP_3$	$1500xP_1$	$85xP_2$	$0.3xP_3$
Pt. Beach	$7.7xP_1$	$0xP_2$	$0xP_3$	$110xP_1$	$0.3xP_2$	$0xP_3$	$1400xP_1$	$77xP_2$	$0.3xP_3$
Prairie Is.	$56xP_1$	$0xP_2$	$0xP_3$	$260xP_1$	$2.4xP_2$	$0xP_3$	$1400xP_1$	$110xP_2$	$0.4xP_3$
Quad Cities	$17xP_1$	$0xP_2$	$0xP_3$	$290xP_1$	$0.04xP_2$	$0xP_3$	$1900xP_1$	$170xP_2$	$0.7xP_3$
Rancho Seco	$15xP_1$	$0xP_2$	$0xP_3$	$110xP_1$	$0.02xP_2$	$0xP_3$	$870xP_1$	$87xP_2$	$0.3xP_3$
River Bend	$31xP_1$	$0xP_2$	$0xP_3$	$200xP_1$	$0.2xP_2$	$0xP_3$	$750xP_1$	$60xP_2$	$0.2xP_3$
Robinson	$16xP_1$	$0xP_2$	$0xP_3$	$170xP_1$	$0.01xP_2$	$0xP_3$	$880xP_1$	$59xP_2$	$0.2xP_3$
St. Lucie	$77xP_1$	$0xP_2$	$0xP_3$	$310xP_1$	$0.6xP_2$	$0xP_3$	$700xP_1$	$69xP_2$	$0.4xP_3$
Salem	$120xP_1$	$0xP_2$	$0xP_3$	$440xP_1$	$0xP_2$	$0xP_3$	$3000xP_1$	$160xP_2$	$0.5xP_3$
San Onofre	$11xP_1$	$0xP_2$	$0xP_3$	$150xP_1$	$0xP_2$	$0xP_3$	$1800xP_1$	$150xP_2$	$0.5xP_3$
Seabrook	$13xP_1$	$0xP_2$	$0xP_3$	$210xP_1$	$0.04xP_2$	$0xP_3$	$1000xP_1$	$54xP_2$	$0.2xP_3$
Sequoyah	$110xP_1$	$0xP_2$	$0xP_3$	$690xP_1$	$0.6xP_2$	$0xP_3$	$1300xP_1$	$95xP_2$	$0.3xP_3$
Shearon Harris	$40xP_1$	$0xP_2$	$0xP_3$	$260xP_1$	$0.4xP_2$	$0xP_3$	$1300xP_1$	$110xP_2$	$0.4xP_3$
Shoreham	$140xP_1$	$0xP_2$	$0xP_3$	$870xP_1$	$1.9xP_2$	$0xP_3$	$3400xP_1$	$170xP_2$	$0.5xP_3$
Skagit	$50xP_1$	$0xP_2$	$0xP_3$	$370xP_1$	$0.4xP_2$	$0xP_3$	$500xP_1$	$49xP_2$	$0.2xP_3$
South Texas	$5.2xP_1$	$0xP_2$	$0xP_3$	$32xP_1$	$0xP_2$	$0xP_3$	$610xP_1$	$43xP_2$	$0.2xP_3$
Surry	$65xP_1$	$0xP_2$	$0xP_3$	$330xP_1$	$0xP_2$	$0xP_3$	$1700xP_1$	$95xP_2$	$0.3xP_3$
Susquehanna	$180xP_1$	$0xP_2$	$0xP_3$	$700xP_1$	$0.2xP_2$	$0xP_3$	$3300xP_1$	$150xP_2$	$0.5xP_3$

*See footnote, page C-2.

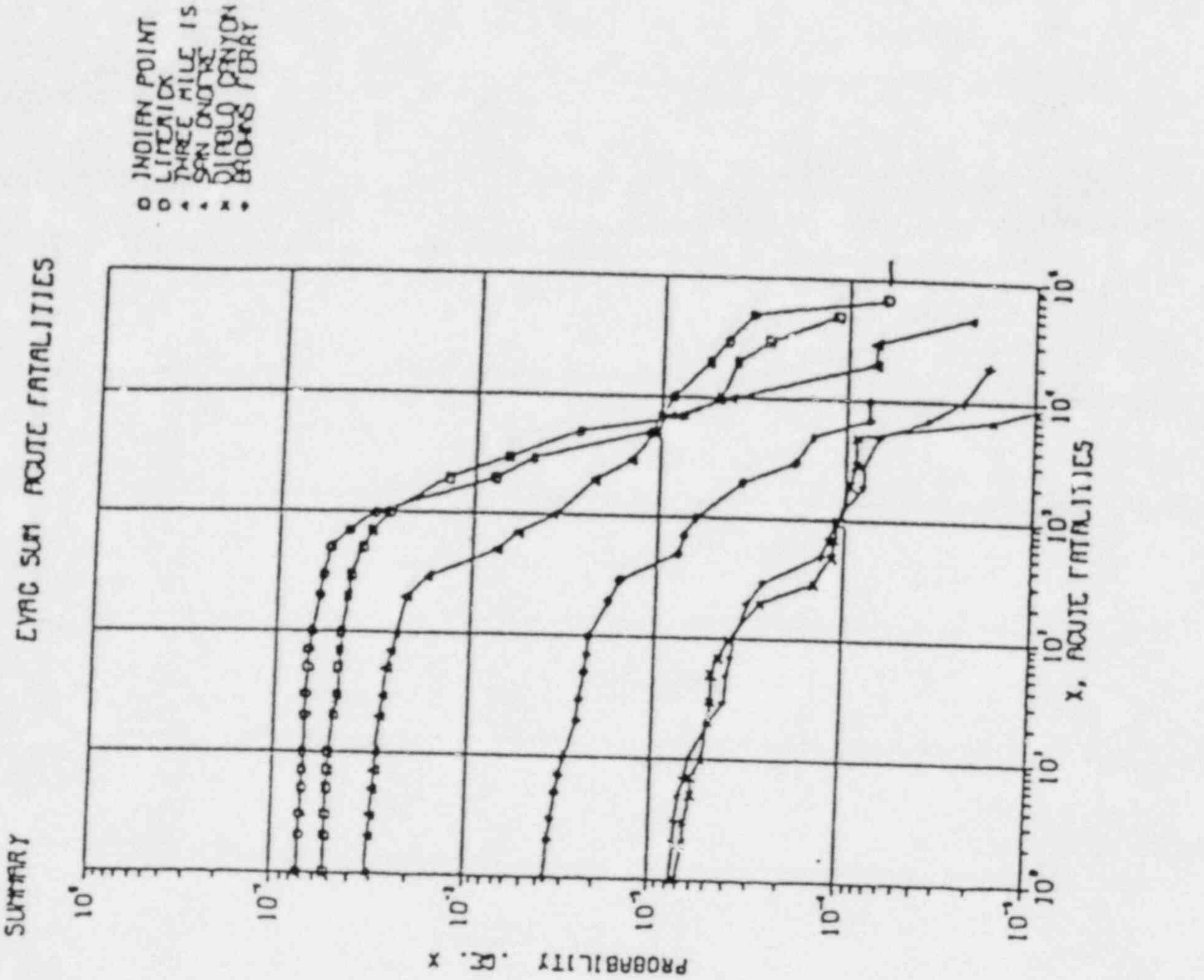
Table C-1. (continued)

	Mean Early Fatalities*			Mean Early Injuries*			Mean Latent Cancer Fatalities*		
	SST1	SST2	SST3	SST1	SST2	SST3	SST1	SST2	SST3
Three Mile Island	$240xP_1$	$0xP_2$	$0xP_3$	$1200xP_1$	$4.5xP_2$	$0xP_3$	$3500xP_1$	$170xP_2$	$0.6xP_3$
Trojan	$46xP_1$	$0.1xP_2$	$0xP_3$	$350xP_1$	$3.8xP_2$	$0xP_3$	$1100xP_1$	$73xP_2$	$0.3xP_3$
Turkey Pt.	$31xP_1$	$0xP_2$	$0xP_3$	$460xP_1$	$0xP_2$	$0xP_3$	$690xP_1$	$83xP_2$	$0.4xP_3$
Vermont Yankee	$130xP_1$	$0xP_2$	$0xP_3$	$320xP_1$	$4.4xP_2$	$0xP_3$	$1800xP_1$	$72xP_2$	$0.3xP_3$
Virgil Summer	$12xP_1$	$0xP_2$	$0xP_3$	$120xP_1$	$0xP_2$	$0xP_3$	$1000xP_1$	$63xP_2$	$0.2xP_3$
Vogtle	$0.07xP_1$	$0xP_2$	$0xP_3$	$85xP_1$	$0xP_2$	$0xP_3$	$900xP_1$	$70xP_2$	$0.3xP_3$
WPPSS 1,4	$0.1xP_1$	$0xP_2$	$0xP_3$	$110xP_1$	$0xP_2$	$0xP_3$	$310xP_1$	$37xP_2$	$0.2xP_3$
WPPSS 2	$1.0xP_1$	$0xP_2$	$0xP_3$	$120xP_1$	$0xP_2$	$0xP_3$	$720xP_1$	$53xP_2$	$0.2xP_3$
WPPSS 3,5	$0.1xP_1$	$0xP_2$	$0xP_3$	$110xP_1$	$0xP_2$	$0xP_3$	$310xP_1$	$37xP_2$	$0.2xP_3$
Waterford	$170xP_1$	$0.2xP_2$	$0xP_3$	$580xP_1$	$8.3xP_2$	$0xP_3$	$990xP_1$	$93xP_2$	$0.4xP_3$
Watts Bar	$13xP_1$	$0xP_2$	$0xP_3$	$110xP_1$	$0.02xP_2$	$0xP_3$	$1000xP_1$	$66xP_2$	$0.3xP_3$
Wolf Creek	$2.4xP_1$	$0xP_2$	$0xP_3$	$34xP_1$	$0.04xP_2$	$0xP_3$	$760xP_1$	$70xP_2$	$0.3xP_3$
Yankee Rowe	$18xP_1$	$0xP_2$	$0xP_3$	$180xP_1$	$0.05xP_2$	$0xP_3$	$2300xP_1$	$100xP_2$	$0.2xP_3$
Yellow Creek	$5.6xP_1$	$0xP_2$	$0xP_3$	$68xP_1$	$0xP_2$	$0xP_3$	$850xP_1$	$63xP_2$	$0.3xP_3$
Zimmer	$46xP_1$	$0xP_2$	$0xP_3$	$670xP_1$	$0.4xP_2$	$0xP_3$	$2400xP_1$	$170xP_2$	$0.6xP_3$
Zion	$520xP_1$	$4.1xP_2$	$0xP_3$	$1600xP_1$	$32xP_2$	$0xP_3$	$4000xP_1$	$330xP_2$	$1.2xP_3$

*See footnote, page C-2.

A P P E N D I X E

Sample CCDF curve from CRAC2 printout for Sandia siting study carried past a conditional probability of one in a thousand; from NRC response to FOIA request from Charles Elliott.


SUMMARY
EVAC SUM ACUTE FATALITIES
INDIAN POINT
LIMERICK
THREE MILE IS
SAN ONOFRE
DIABLO CANYON
BROWNS FERRY
PROBABILITY .GE. X
X, ACUTE FATALITIES

A P P E N D I X F

Table 2.5-6 from NUREG/CR-2239, Technical Guidance for Siting Criteria Development, Sandia National Laboratories, December 1982, page 2-47.

Table 2.5-6. Dependence of Early Fatalities and Early Injuries on Response Distance for Eight Emergency Response Scenarios. Results are Conditional on an SST1 Release

Emergency Response		Response Distance (mi)									
Type	Characteristics	0[a]	5	10	15	25	0[a]	5	10	15	25
		Mean Early Fatalities					Mean Early Injuries				
Evacuation	5 hr delay, 1 mph	3,600	2,100	1,900	1,800	1,800	6,300	6,200	5,300	5,100	4,700
	5 hr delay, 10 mph	3,600	1,600	1,400	1,300	1,250	6,300	6,000	4,300	3,300	2,500
	3 hr delay, 10 mph	3,600	1,200	920	860	790	6,300	5,800	4,000	3,000	2,200
	Summary Evacuation	3,600	1,100	830	780	700	6,300	5,500	3,600	2,700	1,800
	1 hr delay, 10 mph	3,600	440	180	110	40	6,300	4,600	2,500	1,500	700
Sheltering[b]	24 hr relocation	3,600	c	1,200	c	c	6,300	c	4,100	c	c
	12 hr relocation	3,600	c	750	c	c	6,300	c	3,800	c	c
	6 hr relocation	3,600	830	560	490	420	6,300	5,600	3,700	2,700	1,800
		99th Percentile Early Fatalities[d]					99th Percentile Early Injuries[d]				
Evacuation	5 hr delay, 1 mph	18,000	16,000	14,000	12,000	11,000	41,000	41,000	40,000	41,000	28,000
	5 hr delay, 10 mph	18,000	14,000	10,000	9,400	8,800	41,000	40,000	34,000	26,000	10,000
	3 hr delay, 10 mph	18,000	11,000	8,000	7,300	7,000	41,000	40,000	32,000	26,000	10,000
	Summary Evacuation	18,000	11,000	8,300	7,600	7,200	41,000	40,000	33,000	26,000	9,400
	1 hr delay, 10 mph	18,000	7,000	1,400	1,200	1,000	41,000	39,000	30,000	24,000	5,200
Sheltering[b]	24 hr relocation	18,000	c	9,300	c	c	41,000	c	34,000	c	c
	12 hr relocation	18,000	c	7,500	c	c	41,000	c	33,000	c	c
	6 hr relocation	18,000	9,300	5,500	4,900	4,500	41,000	40,000	32,000	25,000	11,000

Assumptions: 1120 MWe reactor, SST1 release, Indian Point population and wind rose, New York City Meteorology.

a. No emergency response. b. Northeast Regional Shielding Factors. c. Not calculated. d. Consequence magnitude equalled or exceeded following 1 out of every 100 releases.

A P P E N D I X G

Rankings of "Maximum Calculated Values" from CRAC2 Output from Sandia siting study printouts, corrected for reactor power level, extracted from, "List of Sites With the Highest Scaled Consequences Based on NRC CRAC2 Accident Consequence Analysis", November 1, 1982, Subcommittee on Oversight and Investigation, House Committee on Interior and Insular Affairs

EARLY FATALITIES		EARLY INJURIES		LATENT CANCER FATALITIES	
Salem-1/2	100,000	Limerick-1/2	610,000	Salem-1/2	40,000
Waterford-3	96,000	Fermi-2	340,000	Millstone-3	38,000
Limerick-1/2	74,000	Waterford-3	279,000	Peach Bottom-2/3	37,000
Peach Bottom-2/3	72,000	Perry-1/2	180,000	Shoreham	35,000
Susquehanna-1/2	67,000	Indian Point-3	167,000	Limerick-1/2	34,000
Indian Point-3	50,000	Beaver Valley-1/2	156,000	Millstone-2	33,000
Indian Point-2	46,000	Zion-1/2	155,000	North Anna-1/2	29,000
Three Mile Island-1	42,000	Marble Hill-1/2	150,000	Susquehanna-1/2	28,000
Catawba-1/2	42,000	Indian Point-2	141,000	Millstone-1	28,000

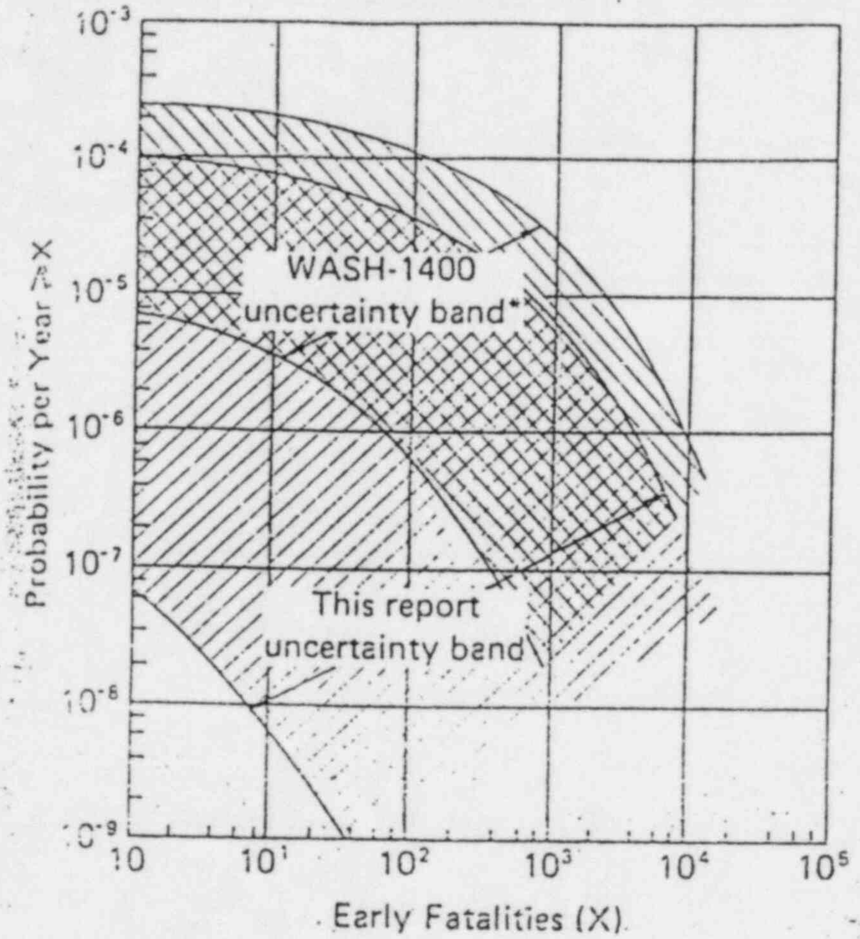


Fig. 1. Probability distribution for early fatalities per year for 100 reactors.

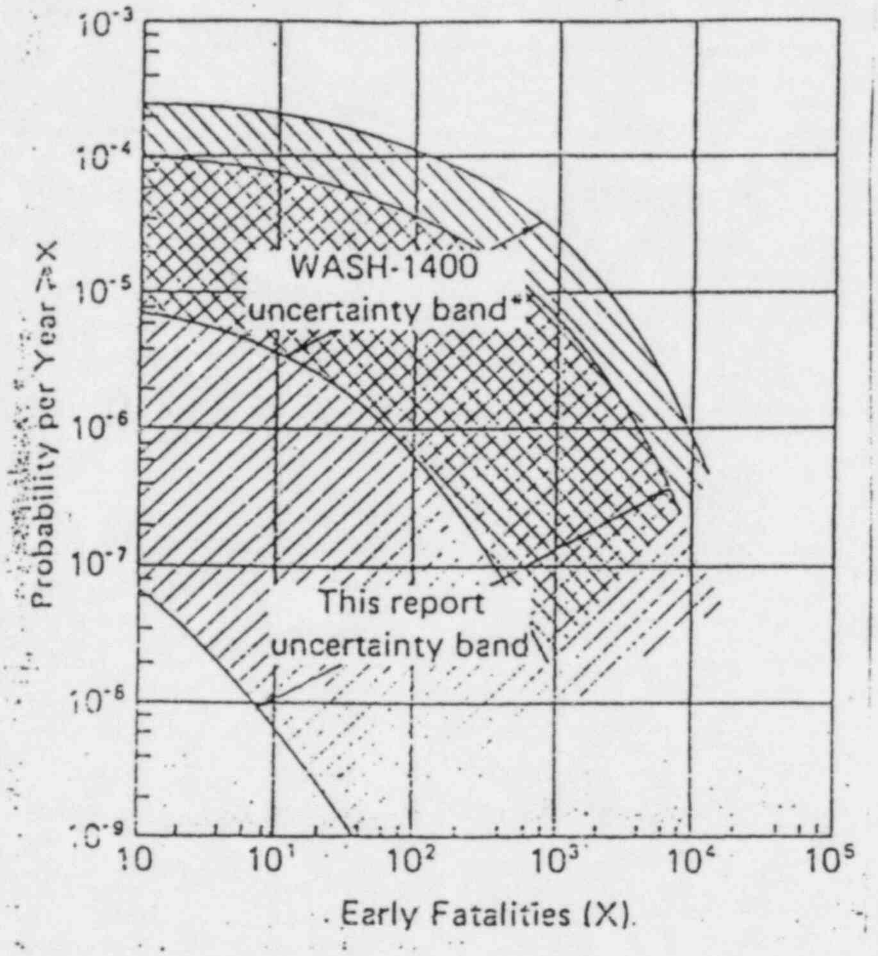


Fig. 1. Probability distribution for early fatalities per year for 100 reactors.

A P P E N D I X H

From Erdmann, 1981, page 379; comparison of WASH-1400 and reestimated median value and uncertainty band for early fatality estimates for 100 reactors. (enlarged)

STATEMENT OF PROFESSIONAL QUALIFICATIONS - STEVEN C. SHOLLY

My name is Steven C. Sholly. I am a Technical Research Associate with the Union of Concerned Scientists (UCS), 1346 Connecticut Avenue, N.W., Suite 1101, Washington, D.C., 20036. I joined the UCS staff in February 1981. My responsibilities at UCS include monitoring technical developments in a number of fields related to nuclear reactor safety, including radiological emergency planning, severe accident research, probabilistic risk assessment and accident consequence analysis, accident mitigation systems, and systems interaction. My responsibilities at UCS also include writing articles for UCS's quarterly report Nucleus, responding to inquiries from the media and from citizens groups, and researching NRC and other technical literature on a variety of topics related to nuclear reactor safety. My most recent articles published in Nucleus include "The Probability of a Core Melt Accident" [Vol. 4, No. 3, Fall 1982], and "The Consequences of a Nuclear Reactor Accident" [Vol. 4, No. 4, Winter 1983].

Prior to joining the UCS staff, I served as Research Coordinator and later as Project Director of the TMI Public Interest Research Center (TMIPIRC), 1037 Maclay Street, Harrisburg, PA, 17103. At TMIPIRC, I was responsible for directing research and public education activities. I also attended the TMI-1 Restart proceeding before the Atomic Safety and Licensing Board and kept TMIPIRC member groups and the public informed on the hearings through press conferences and periodic reports. While at TMIPIRC, I authored a report on the then-proposed venting of Krypton-85 gas from the containment of the damaged TMI-2 reactor. I was also responsible for monitoring the

progress of the cleanup of the TMI-2 reactor.

I was awarded a Bachelor of Science degree in Education from Shippensburg State College, Shippensburg, PA, in August 1975. My majors were Earth and Space Science and General Science, and I took a minor in Environmental Education. I have also completed graduate courses at Shippensburg State College in land use planning.

In addition to the work job experience detailed above, I taught Earth and Space Science and Environmental Science for two years at the junior high school level, and operated wastewater treatment facilities for two years. In the latter capacity, I served as Chief Process Operator at the Derry Township Municipal Authority's treatment facility in Hershey, PA, where I was responsible for directing and monitoring the biological performance of a 5-MGD tertiary wastewater treatment plant.

I have published several articles in Nucleus and have also published an article in the Journal of Geological Education on determining Mercalli earthquake intensities from media accounts of historical earthquakes ["Determining Mercalli Intensities from Newpaper Reports", Journal of Geological Education, Vol. 25, pages 105-106, 1977]. I have testified in several Congressional hearings, most recently in December 1982 before the Subcommittee on Oversight and Investigation of the House Interior and Insular Affairs Committee on steam generator operating experience and accident hazards. I have also testified on filtered vented containment systems and compartment venting systems (Commission Question Two), and accident consequences (Commission Question One) in the Indian Point special investigation proceeding.

Updated 15 March 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2) POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	Docket Nos. 50-247 SP 50-286 SP 22 March 1983

CERTIFICATE OF SERVICE

I hereby certify that a single copy of UCS/NYPIRG TESTIMONY OF STEVEN C. SHOLLY ON COMMISSION QUESTION FIVE was served upon the following by deposit in the U.S. mail, first class postage prepaid, this 22th day of March 1983, except where noted otherwise by asterisks.

Steven C. Sholly

*Nunzio Palladino, Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*John Ahearne, Commissioner
U.S. Nuclear Regulatory Commission
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*James Asselstine, Commissioner
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*Thomas Roberts, Commissioner
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*James P. Gleason, Esq., Chairman
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Dr. Oscar H. Paris
Administrative Judge
Atomic Safety and Licensing
Board
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