NUREG-0715

# Task Force Report on Interim Operation of Indian Point

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Abstract

On May 30, 1980, the Commission issued an order establishing a four-pronged approach for resolving the issues raised by the Union of Concerned Scientists' petition regarding the Indian Point nuclear facilities. Among other things a Task Force on Interim Operation was established to address the question of whether Indian Point Units 2 and 3 should or should not be allowed to operate during the pendency of a planned adjudication.

Specifically, the Task Force report deals with two major issues. The first issue relates to accident risk as a function of population density and distribution around the plant. New York City is less than 50 miles to the south of the Indian Point site. The Task Force compared Indian Point risks, e.g., health impacts, property damage with those of other reactor sites and designs, distinguishing between the effects of population densities and of design and other factors. Secondly, the Task Force examined the economic, social and other "non-safety" effects of shutting down or reducing the power levels of either or both reactors. In particular, the Task Force compared projected peak demands for energy with projected available capacity to determine if reducing power levels at Indian Point would affect system reliability in the summer of 1980.

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# INTRODUCTION

This report is submitted in response to Section D. The Task Force on Interim Operation, of the Commission's Order of May 30, 1980, in the Matter of Consolidated Edison Company of New York, Inc. (Indian Point, Unit No. 2) and Power Authority of the State of New York (Indian Point, Unit No. 3). (Docket Nos. 50-247 and 50-286.)

The May 30 Order established an approach, including adjudication, for resolving the issues raised by a petition by the Union of Concerned Scientists (UCS) that called, among other things, for shutdown of Indian Point Units 2 and 3. The Director of the Office of Nuclear Reactor Regulation had issued a decision regarding that petition on February 11, 1980.

Section D of the May 30 Order directed the General Counsel and the Director, Office of Policy Evaluation, to establish a task force to prepare a report to the Commission on information available at this time that bears on the question of whether to permit, prohibit, or curtail operation of Indian Point Units 2 and 3 during pendency of the adjudication. The task force report was to include information on at least certain specified topics listed in the Order. The topics fall into two categories: accident risk considerations (items 1 to 4 of Section D, at pages 6-7 of the Order) and social and economic impact considerations (item 5, at page 7 of the Order).

The accident risk considerations are addressed in Section 1 of this report. Those considerations include comparative site demography; accident risk comparisons; effects of emergency response; and effects of differences between Units 2 and 3, of changes ordered by the Director of NRR, and of power-level reduction. Effects of uncertainties are discussed. Some explanatory details are appended. (Appendices A and B)

Social and economic impact considerations are addressed in Section 2. The principal considerations addressed include effects of shutdown or power reduction on (a) reliability of the electric power supply for the region, including New York City, and (b) sources and cost of electrical energy. Supporting information from the Department of Energy is appended. (Appendix C)

Public comments relevant to interim operation or shutdown, received in response to the Commission's February 15 solicitation of comments, are summarized in Section 3.

The principal contributors to this work were Robert M. Bernero, Roger M. Blond, W. Clark Pritchard, and Merrill A. Taylor, of the Office of Nuclear Regulatory Research; and George Eysymontt and George Sege, of the Office of Policy Evaluation.

## SECTION 1. ACCIDENT RISK CONSIDERATIONS

This section presents estimates of the accident risk posed by operation of the plants in their present condition; a comparison of the risk from other sites and designs; the sensitivity of that risk to emergency protective measures, and the sensitivity of risk to a reduction in power level during operation.

### THE POPULATION DISTRIBUTION

The Indian Point Power Station, with New York City less than 50 miles to the south, has the largest population in its immediate surroundings of any nuclear power station in the United States. "Demographic Statistics Pertaining to Nuclear Power Reactor Sites," NUREG-0348, tabulates all U. S. nuclear power stations according to the total population within a circle of given radius from the reactor. Tables 1, 2, and 3 show the populations at distances of 10, 30 and 50 miles based upon the 1970 census. The region around the Indian Point station is the most densely populated as shown by these data.

When considering reactor accident risk, the population in a given direction, (i.e., in one 22½ degree sector), is often more significant than population density averaged over all directions. Reactors have been ranked by their sector population in Table 4. Here too, Indian Point ranks among the highest. However, a number of other U. S. reactor sites, for example, Zion and Limerick, also have relatively high populations in their vicinity.

TABLE 1

Population Statistics Between 0 and 10 Miles

POPULATION STATISTICS- 1979 REVISION 5/79
BASED ON THE YEAR 1970
POPULATION STATISTICS WITHIN 0-10 MILES
TOTAL NUMBER OF SITES- 111
MINIMUM POPULATION- 0 MAXIMUM POPULATION- 218398
MEAN POPULATION- 36931 MEDIAN POPULATION- 24269
90% PERCENTILE POPULATION- 83557
STANDARD DEVIATION- 39164.6 COEF. OF VARIATION- 1.060

| NO. | SITE NAME | POPULATION | NO. | SITE NAME | POPULATION | NO. | SITE NAME | POPULATION |
|---|---|---|---|---|---|---|---|---|
| 1 | SUNDESERT | 0 | 38 | DAVIS BESSE | 15390 | 75 | OYSTER CREEK | 36797 |
| 2 | WPPSS 2 | 455 | 39 | SKAGIT | 16038 | 76 | FORKED RIVER | 36797 |
| 3 | PEBBLE SPRINGS | 878 | 40 | CALVERT CLIFFS | 16827 | 77 | STERLING | 37705 |
| 4 | PALO VERDE | 1892 | 41 | WOOD | 16889 | 78 | OCONEE | 37831 |
| 5 | WPPSS 1&4 | 2648 | 42 | FORT CALHOUN | 17401 | 79 | MCGUIRE | 39374 |
| 6 | SOUTH TEXAS | 3254 | 43 | PHIPPS BEND | 17665 | 80 | ERIE | 40206 |
| 7 | VOGTLE | 3500 | 44 | RIVER BEND | 19147 | 81 | NEW ENGLAND | 41882 |
| 8 | HATCH | 4803 | 45 | PRAIRIE ISLAND | 19401 | 82 | HUMBOLDT BAY | 45403 |
| 9 | WOLF CREEK | 5260 | 46 | BYRON | 20377 | 83 | GREENE COUNTY | 45786 |
| 10 | COMANCHE PEAK | 5353 | 47 | MARBLE HILL | 20959 | 84 | SAINT LUCIE | 46066 |
| 11 | SUMMER | 5656 | 48 | POINT BEACH | 21073 | 85 | GINNA | 46325 |
| 12 | RANCHO SECO | 6061 | 49 | YANKEE ROWE | 21763 | 86 | SUSQUEHANNA | 50435 |
| 13 | LACROSSE | 6209 | 50 | BRAIDWOOD | 21942 | 87 | PILGRIM | 51203 |
| 14 | DIABLO CANYON | 6302 | 51 | BELLEFONTE | 22709 | 88 | COOK | 53006 |
| 15 | COOPER | 6363 | 52 | ZIMMER | 23023 | 89 | SHOREHAM | 54251 |
| 16 | GRAND GULF | 7245 | 53 | VERMONT YANKEE | 23030 | 90 | HADDAM NECK | 60374 |
| 17 | BIG ROCK POINT | 7551 | 54 | ELK RIVER | 23890 | 91 | TROJAN | 61855 |
| 18 | WATTS BAR | 7674 | 55 | ARKANSAS | 24141 | 92 | MIDLAND | 62000 |
| 19 | NORTH ANNA | 7713 | 56 | NEW HAVEN | 24397 | 93 | CATAWBA | 65901 |
| 20 | HALLAM | 8365 | 57 | SAN ONOFRE | 25725 | 94 | SURRY | 66630 |
| 21 | FORT ST. VRAIN | 8366 | 58 | PEACH BOTTOM | 25984 | 95 | HAVEN | 67981 |
| 22 | TYRONE | 8632 | 59 | MAINE YANKEE | 26000 | 96 | PIQUA | 72560 |
| 23 | YELLOW CREEK | 8828 | 60 | ROBINSON | 26016 | 97 | PERRY | 73600 |
| 24 | CALLAWAY | 8914 | 61 | QUAD-CITIES | 26739 | 98 | DUANE ARNOLD | 79310 |
| 25 | FARLEY | 9528 | 62 | BROWNS FERRY | 27215 | 99 | SEABROOK | 79478 |
| 26 | WPPSS 3&5 | 9767 | 63 | SALEM | 28562 | 100 | BAILLY | 83608 |
| 27 | BRUNSWICK | 10000 | 64 | HOPE CREEK | 28562 | 101 | PATHFINDER | 84117 |
| 28 | BLACK FOX | 10404 | 65 | PALISADES | 29528 | 102 | TURKEY POINT | 88000 |
| 29 | HARTSVILLE | 11340 | 66 | DRESDEN | 31126 | 103 | BONUS | 89000 |
| 30 | CRYSTAL RIVER | 11699 | 67 | CHEROKEE | 31877 | 104 | BEAVER VALLEY | 105000 |
| 31 | CLINTON | 11889 | 68 | DOUGLAS POINT | 32020 | 105 | MILLSTONE | 105619 |
| 32 | CVTR | 12029 | 69 | SEQUOYAH | 32145 | 106 | FERMI | 134206 |
| 33 | SHEARON HARRIS | 12132 | 70 | JAMESPORT | 33200 | 107 | THREE MILE ISLAND | 136400 |
| 34 | MONTICELLO | 12344 | 71 | PERKINS | 34369 | 108 | SHIPPINGPORT | 143371 |
| 35 | KEWAUNEE | 12759 | 72 | WATERFORD | 35678 | 109 | LIMERICK | 152644 |
| 36 | LASALLE | 13343 | 73 | NINE MILE POINT | 36000 | 110 | ZION | 190314 |
| 37 | CARROLL COUNTY | 13999 | 74 | FITZPATRICK | 36000 | 111 | INDIAN POINT | 218398 |

TABLE 2

# Population Statistics Between 0 and 30 Miles

POPULATION STATISTICS- 1979 REVISION 5/79
BASED ON THE YEAR 1970
POPULATION STATISTICS WITHIN 0-30 MILES
TOTAL NUMBER OF SITES= 111
MINIMUM POPULATION= 87 MAXIMUM POPULATION= 3984844
MEAN POPULATION= 531127 MEDIAN POPULATION= 321647
90% PERCENTILE POPULATION= 998939
STANDARD DEVIATION= 645852.0 COEF. OF VARIATION= 1.216

| NO. | SITE NAME | POPULATION | NO. | SITE NAME | POPULATION | NO. | SITE NAME | POPULATION |
|---|---|---|---|---|---|---|---|---|
| 1 | SUNDESERT | 87 | 38 | MAINE YANKEE | 197000 | 75 | SHEARON HARRIS | 495900 |
| 2 | PEBBLE SPRINGS | 4752 | 39 | TROJAN | 197480 | 76 | MILLSTONE | 496143 |
| 3 | PALO VERDE | 20039 | 40 | HAVEN | 208201 | 77 | COOK | 522000 |
| 4 | CRYSTAL RIVER | 32055 | 41 | VERMONT YANKEE | 211630 | 78 | SURRY | 524100 |
| 5 | SOUTH TEXAS | 40950 | 42 | LASALLE | 215680 | 79 | NEW ENGLAND | 563343 |
| 6 | BIG ROCK POINT | 46538 | 43 | PALISADES | 216535 | 80 | DRESDEN | 568123 |
| 7 | COOPER | 58916 | 44 | HALLAM | 218551 | 81 | FORT CALHOUN | 589809 |
| 8 | WOLF CREEK | 61905 | 45 | DUANE ARNOLD | 232995 | 82 | PERKINS | 622997 |
| 9 | COMANCHE PEAK | 65049 | 46 | KEWAUNEE | 245806 | 83 | SUSQUEHANNA | 631467 |
| 10 | ARKANSAS | 76582 | 47 | PRAIRIE ISLAND | 264432 | 84 | JAMESPORT | 655123 |
| 11 | HATCH | 81252 | 48 | BROWNS FERRY | 265532 | 85 | DAVIS BESSE | 672000 |
| 12 | GRAND GULF | 90049 | 49 | MONTICELLO | 271182 | 86 | PERRY | 703553 |
| 13 | HUMBOLDT BAY | 90330 | 50 | STERLING | 275717 | 87 | CATAWBA | 707512 |
| 14 | WPPSS 2 | 92185 | 51 | VOGTLE | 280137 | 88 | MCGUIRE | 805535 |
| 15 | WPPSS 1&4 | 98886 | 52 | GREENE COUNTY | 288026 | 89 | PEACH BOTTOM | 830276 |
| 16 | YELLOW CREEK | 104404 | 53 | YANKEE ROWE | 303271 | 90 | GINNA | 870591 |
| 17 | BRUNSWICK | 108479 | 54 | PHIPPS BEND | 308144 | 91 | PILGRIM | 883583 |
| 18 | DIABLO CANYON | 114014 | 55 | NEW HAVEN | 309178 | 92 | SALEM | 893626 |
| 19 | BELLEFONTE | 114998 | 56 | CLINTON | 334115 | 93 | HOPE CREEK | 893626 |
| 20 | FARLEY | 119394 | 57 | CVTR | 360589 | 94 | PIQUA | 895367 |
| 21 | SAINT LUCIE | 120843 | 58 | BRAIDWOOD | 360694 | 95 | DOUGLAS POINT | 900652 |
| 22 | CALLAWAY | 122389 | 59 | OCONEE | 363543 | 96 | RANCHO SECO | 907789 |
| 23 | WPPSS 3&5 | 124551 | 60 | RIVER BEND | 371036 | 97 | TURKEY POINT | 909916 |
| 24 | PATHFINDER | 135451 | 61 | SUMMER | 378538 | 98 | WATERFORD | 957223 |
| 25 | HARTSVILLE | 135984 | 62 | SAN ONOFRE | 408362 | 99 | THREE MILE ISLAND | 995200 |
| 26 | WOOD | 138451 | 63 | QUAD-CITIES | 415500 | 100 | SEABROOK | 1003843 |
| 27 | LACROSSE | 143321 | 64 | SEQUOYAH | 432375 | 101 | ZIMMER | 1052883 |
| 28 | SKAGIT | 151774 | 65 | FORT ST. VRAIN | 434802 | 102 | ELK RIVER | 120202[illegible] |
| 29 | NORTH ANNA | 152432 | 66 | OYSTER CREEK | 451606 | 103 | ZION | 1262593 |
| 30 | TYRONE | 153801 | 67 | FORKED RIVER | 451606 | 104 | SHIPPINGPORT | 1677889 |
| 31 | WATTS BAR | 161537 | 68 | BONUS | 453000 | 105 | BEAVER VALLEY | 1700000 |
| 32 | POINT BEACH | 187086 | 69 | BYRON | 455409 | 106 | SHOREHAM | 1760382 |
| 33 | CALVERT CLIFFS | 188755 | 70 | MARBLE HILL | 457928 | 107 | HADDAM NECK | 1763975 |
| 34 | ROBINSON | 192140 | 71 | BLACK FOX | 459832 | 108 | BAILLY | 2200000 |
| 35 | NINE MILE POINT | 195143 | 72 | MIDLAND | 470000 | 109 | FERMI | 2371808 |
| 36 | FITZPATRICK | 195143 | 73 | CHEROKEE | 475129 | 110 | LIMERICK | 3836244 |
| 37 | CARROLL COUNTY | 196357 | 74 | ERIE | 483519 | 111 | INDIAN POINT | 3984844 |

TABLE 3

Population Statistics Between 0 and 50 Miles

POPULATION STATISTICS- 1979 REVISION 5/79
BASED ON THE YEAR 1970
POPULATION STATISTICS WITHIN 0-50 MILES
TOTAL NUMBER OF SITES- 111
MINIMUM POPULATION- 7784 MAXIMUM POPULATION-17471479
MEAN POPULATION- 1705750 MEDIAN POPULATION- 948747
90% PERCENTILE POPULATION- 4085400
STANDARD DEVIATION-2196315.2 COEF. OF VARIATION- 1.287

| NO. | SITE NAME | POPULATION | NO. | SITE NAME | POPULATION | NO. | SITE NAME | POPULATION |
|---|---|---|---|---|---|---|---|---|
| 1 | SUNDESERT | 7784 | 38 | SEQUOYAH | 659015 | 75 | SUSQUEHANNA | 1537373 |
| 2 | PEBBLE SPRINGS | 74814 | 39 | CVTR | 661462 | 76 | YANKEE ROWE | 1538765 |
| 3 | HUMBOLDT BAY | 100728 | 40 | PHIPPS BEND | 691304 | 77 | SURRY | 1550000 |
| 4 | BIG ROCK POINT | 128631 | 41 | FORT CALHOUN | 711117 | 78 | PIQUA | 1654093 |
| 5 | ARKANSAS | 150464 | 42 | SUMMER | 724009 | 79 | TURKEY POINT | 1660498 |
| 6 | WOLF CREEK | 165677 | 43 | OCONEE | 730291 | 80 | ZIMMER | 1786790 |
| 7 | CRYSTAL RIVER | 169908 | 44 | CARROLL COUNTY | 733928 | 81 | NEW ENGLAND | 1862933 |
| 8 | COOPER | 171895 | 45 | CLINTON | 768171 | 82 | THREE MILE ISLAND | 1868000 |
| 9 | BRUNSWICK | 174066 | 46 | COMANCHE PEAK | 783124 | 83 | MONTICELLO | 1956232 |
| 10 | WPPSS 1&4 | 181928 | 47 | NORTH ANNA | 827109 | 84 | DAVIS BESSE | 2052000 |
| 11 | WPPSS 2 | 184296 | 48 | NINE MILE POINT | 843725 | 85 | PRAIRIE ISLAND | 2057725 |
| 12 | SOUTH TEXAS | 196206 | 49 | FITZPATRICK | 843725 | 86 | ELK RIVER | 2101115 |
| 13 | DIABLO CANYON | 209444 | 50 | BELLEFONTE | 845838 | 87 | CALVERT CLIFFS | 2305635 |
| 14 | PATHFINDER | 242751 | 51 | HARTSVILLE | 869776 | 88 | ERIE | 2411857 |
| 15 | HATCH | 251612 | 52 | BYRON | 881721 | 89 | PERRY | 2583218 |
| 16 | GRAND GULF | 269314 | 53 | LASALLE | 918803 | 90 | MILLSTONE | 2591658 |
| 17 | CALLAWAY | 299254 | 54 | NEW HAVEN | 921367 | 91 | DOUGLAS POINT | 3167529 |
| 18 | HALLAM | 307945 | 55 | HAVEN | 927246 | 92 | JAMESPORT | 3173531 |
| 19 | SAINT LUCIE | 318784 | 56 | WOOD | 970248 | 93 | HADDAM NECK | 3267732 |
| 20 | FARLEY | 320667 | 57 | PALISADES | 984252 | 94 | OYSTER CREEK | 3290000 |
| 21 | LACROSSE | 321073 | 58 | BONUS | 999000 | 95 | FORKED RIVER | 3290000 |
| 22 | PALO VERDE | 328088 | 59 | MIDLAND | 1000000 | 96 | SAN ONOFRE | 3572478 |
| 23 | YELLOW CREEK | 344716 | 60 | SHEARON HARRIS | 1062200 | 97 | SEABROOK | 3605493 |
| 24 | WPPSS 3&5 | 345935 | 61 | COOK | 1170000 | 98 | SHIPPINGPORT | 3735300 |
| 25 | SKAGIT | 366247 | 62 | TROJAN | 1146188 | 99 | BEAVER VALLEY | 3900000 |
| 26 | TYRONE | 372980 | 63 | VERMONT YANKEE | 1149200 | 100 | BRAIDWOOD | 4088663 |
| 27 | VOGTLE | 456631 | 64 | STERLING | 1154607 | 101 | PEACH BOTTOM | 4121297 |
| 28 | MAINE YANKEE | 486000 | 65 | GINNA | 1215870 | 102 | PILGRIM | 4234545 |
| 29 | ROBINSON | 530817 | 66 | MARBLE HILL | 1245001 | 103 | SALEM | 4773288 |
| 30 | DUANE ARNOLD | 552745 | 67 | CATAWBA | 1245504 | 104 | HOPE CREEK | 4773288 |
| 31 | POINT BEACH | 564251 | 68 | CHEROKEE | 1308327 | 105 | SHOREHAM | 4940868 |
| 32 | KEWAUNEE | 574631 | 69 | MCGUIRE | 1380228 | 106 | FERMI | 5446957 |
| 33 | QUAD-CITIES | 601843 | 70 | RANCHO SECO | 1381581 | 107 | DRESDEN | 6305057 |
| 34 | BROWNS FERRY | 625608 | 71 | GREENE COUNTY | 1383978 | 108 | BAILLY | 6747815 |
| 35 | RIVER BEND | 627983 | 72 | FORT ST. VRAIN | 1396284 | 109 | LIMERICK | 7036199 |
| 36 | BLACK FOX | 641797 | 73 | WATERFORD | 1479345 | 110 | ZION | 7083759 |
| 37 | WATTS BAR | 657836 | 74 | PERKINS | 1506152 | 111 | INDIAN POINT | 17471479 |

TABLE 4

SITES WITH HIGHEST SECTOR POPULATIONS

| | | Population in Highest 22 1/2° Sector(s) |
|---|---|---|
| A. | Based on 1970 census data at 10 miles | |
| 1. | Zion | 65,000; 43,000; 41,000 |
| 2. | Millstone | 39,000 |
| 3. | Duane Arnold | 38,000 |
| 4. | Three Mile Island | 35,000 |
| 5. | Indian Point | 32,000 |
| 6. | Trojan | 32,000 |
| 7. | Beaver Valley | 31,000; 31,000 |
| 8. | Indian Point | 30,000; 30,000 |
| B. | Based on 1970 census data at 30 miles | |
| 1. | Indian Point | 1,500,000; 820,000 |
| 2. | Limerick | 1,300,000; 950,000 |
| 3. | Bailly | 900,000 |
| 4. | Fermi | 800,000; 770,000 |
| 5. | Waterford | 700,000 |
| C. | Based on 1970 census data at 50 miles | |
| 1. | Indian Point | 8,000,000; 2,900,000; 2,300,000 |
| 2. | Dresden | 3,300,000 |
| 3. | Bailly | 3,200,000 |
| 4. | Zion | 3,200,000 |
| 5. | Salem | 2,700,000 |
| 6. | Shoreham | 2,100,000 |
| 7. | Fermi | 2,100,000 |

## REACTOR ACCIDENT RISK PARAMETERS

The accident risk to the public posed by a reactor at a particular site can be analyzed by carefully considering the design and operating characteristics of the reactor plant, the local meteorology, the population distribution around the plant, and the various measures such as sheltering or evacuation which could be taken to reduce the effect of a reactor accident on the public. Ideally, this analysis should be plant and site specific. Experience has already shown that plant design and operating characteristics are not so standardized that it is sufficient to analyze any one reactor, or any one type of reactor, or even any one reactor plant designed by a single supplier. The estimated probabilities and scenarios of reactor accidents are so sensitive to differences in details of component reliability design and procedures, including human errors, that apparently similar plants can be substantially different.

The same need for plant specific analysis holds true for the siting aspects of plants, i.e., the meteorology and especially the demography. Since there exists no exhaustive risk analysis of the Indian Point plants, the following analyses will deal separately with the siting and then the design aspects of the Indian Point plants comparing what we do know of them to similar risk analyses of other U. S. plants. Understanding the overall accident risk of a nuclear power plant or comparison of the risk posed by it to that posed by any other plant requires consideration of the siting as well as the design and operating characteristics of the plant.

SITE ASPECTS

The Reactor Safety Study (WASH-1400), subject, to be sure, to large uncertainties, provides a basic accident risk model which can be used to assess the potential accident risk of a plant, at least in comparison to other plants. The model was developed in the detailed review of only two plants, the Surry pressurized w· ·r reactor (PWR) and the Peach Bottom boiling water reactor (BWR). The Indian Point Unit 2 and 3 reactors are PWRs, furnished by the same nuclear steam system supplier (Westinghouse), but of a larger size and later vintage. To compare reactor sites to one another, the Surry PWR is used as a benchmark and, through the facility of calculation, is moved from site to site calculating the overall risk for four principal risk measures: early fatalities; early (radiation) illnesses; latent cancer fatalities; and public property damage costs. If the power of the benchmark reactor is held constant, then this set of calculations provides a good comparative measure of one site to another.

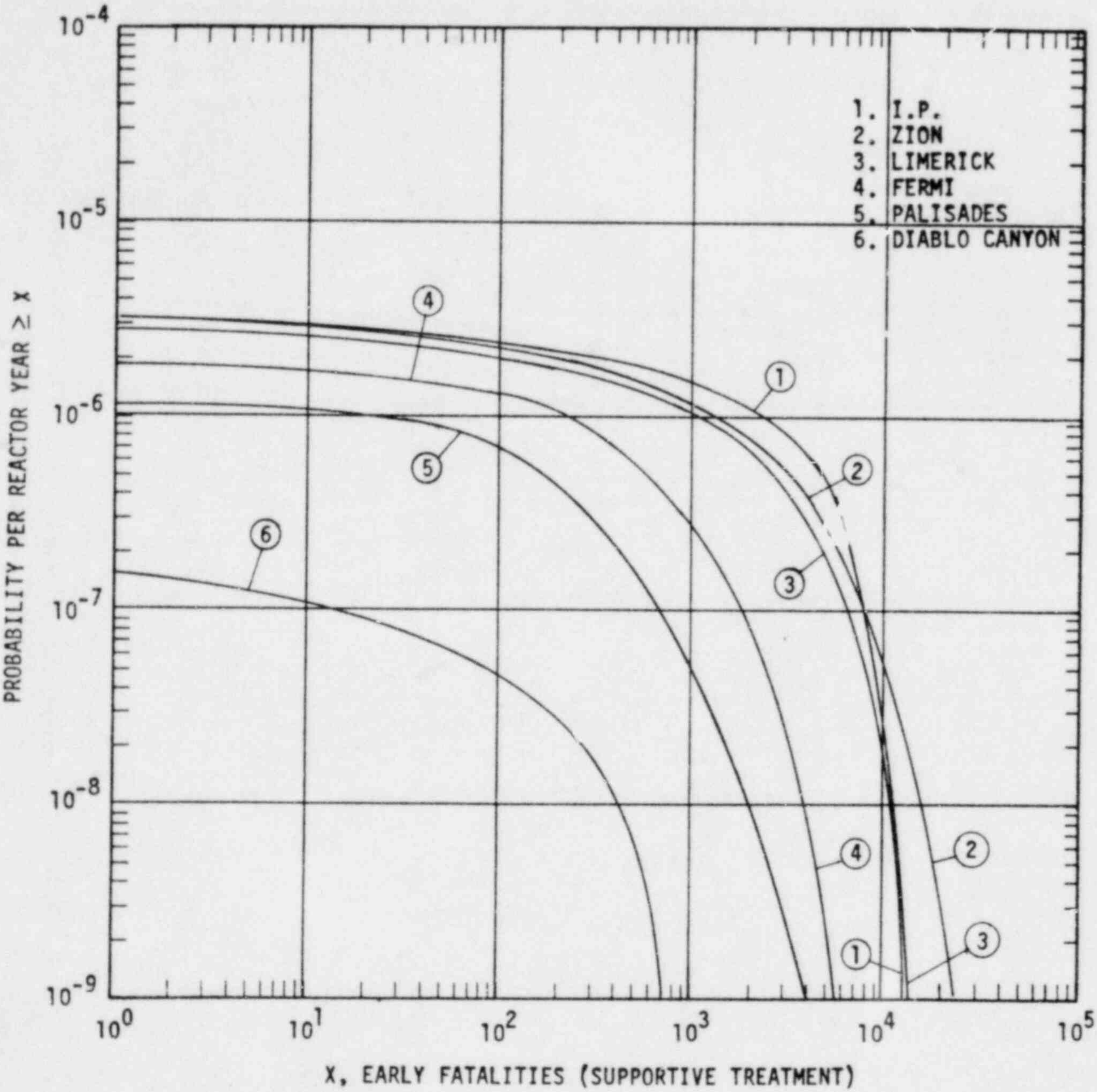
The staff has performed a set of these benchmark calculations using the Surry benchmark reactor with its power increased to 3025 MWT, the rating of Indian Point 3. In general, the risk a reactor poses is proportional to its power level. Six sites were analyzed for this comparison. Four, Indian Point, Zion, Limerick and Fermi, represent sites of relatively high population. One, Palisades, represents what the staff believes is a typical or average population distribution. The last, Diablo Canyon, represents a remote site, thatis, one with relatively low population density. The results of the analyses of the enlarged Surry plant at these six sites are shown in Figures 1 through 4 for the four measures of risk.

The results shown in these figures are the complementary cumulative distribution functions (CCDF)* which are the variation of the consequences of a reactor accident per year with their associated probability of occurrence. The estimated risk of accidents for a given reactor, the product of probabilities and consequences, is the area under the curve. On Figures 1, 2, and 3 are listed the key assumptions about public protective action, namely that people within a 10 mile radius of the plant suffer the entire cloud exposure and then four hours of ground exposure before they are evacuated; people outside the 10 mile radius receive the entire cloud exposure and a subsequent seven day ground exposure assuming normal indoor and outdoor activity.

Before studying the curves consider for a moment the range of consequences that can be caused by a nuclear plant accident. For severe consequences, substantial amounts of radioactive material must be spread out over the surrounding area. The forces ejecting the material and the local meteorology will control how much gets out and how far it will reach. The areas closest to the reactor will stand to receive the highest doses and those farther away, less. The Reactor Safety Study analysis showed that for severe accident releases, only those people within about 10 miles are exposed to fatal doses, beginning at about 300 Rem. Thus, the population within 10 miles of a site will be significant to the early fatality risk for that site; the population beyond 10 miles will not. This was a principal

*The CCDF shows the probability that a consequence will be equalled or exceeded. Appendix A discusses how a CCDF is constructed. For further discussion of the consequence model used in these calculations, please refer to Overview of the Reactor Safety Study Consequence Model (NUREG-0340) and Appendix VI of the Reactor Safety Study (WASH-1400).

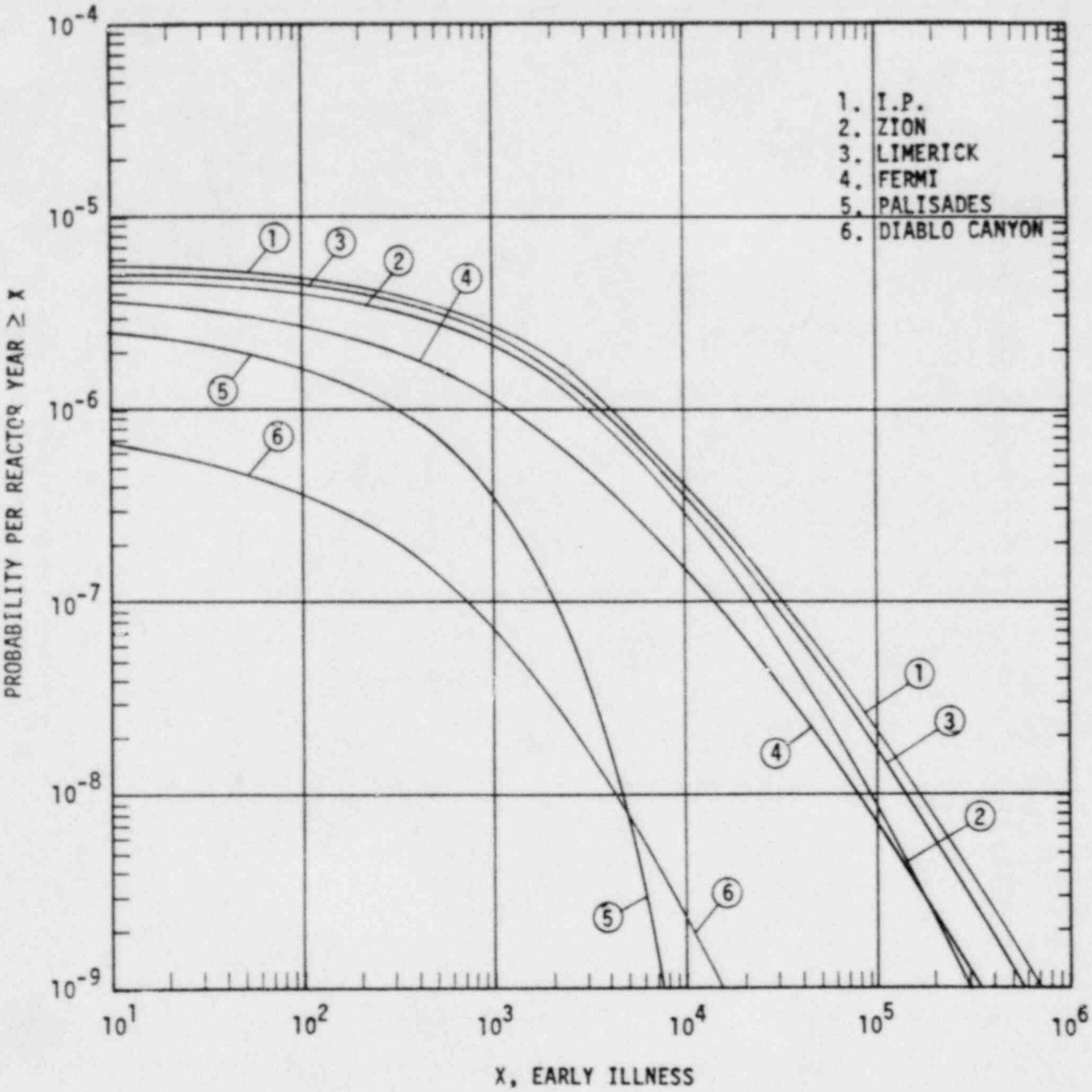
FIGURE 1 - EARLY FATALITY RISK FOR DIFFERENT SITES




NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS:
1) SURRY DESIGN.
2) I.P. UNIT 3 POWER LEVEL (3025 MWT).
3) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE NO SHIELDING
   BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY.
4) WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION.
5) IDENTICAL 91 WEATHER SEQUENCES FOR ALL SITES.

FIGURE 2 - EARLY ILLNESS RISK FOR DIFFERENT SITES




NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS:
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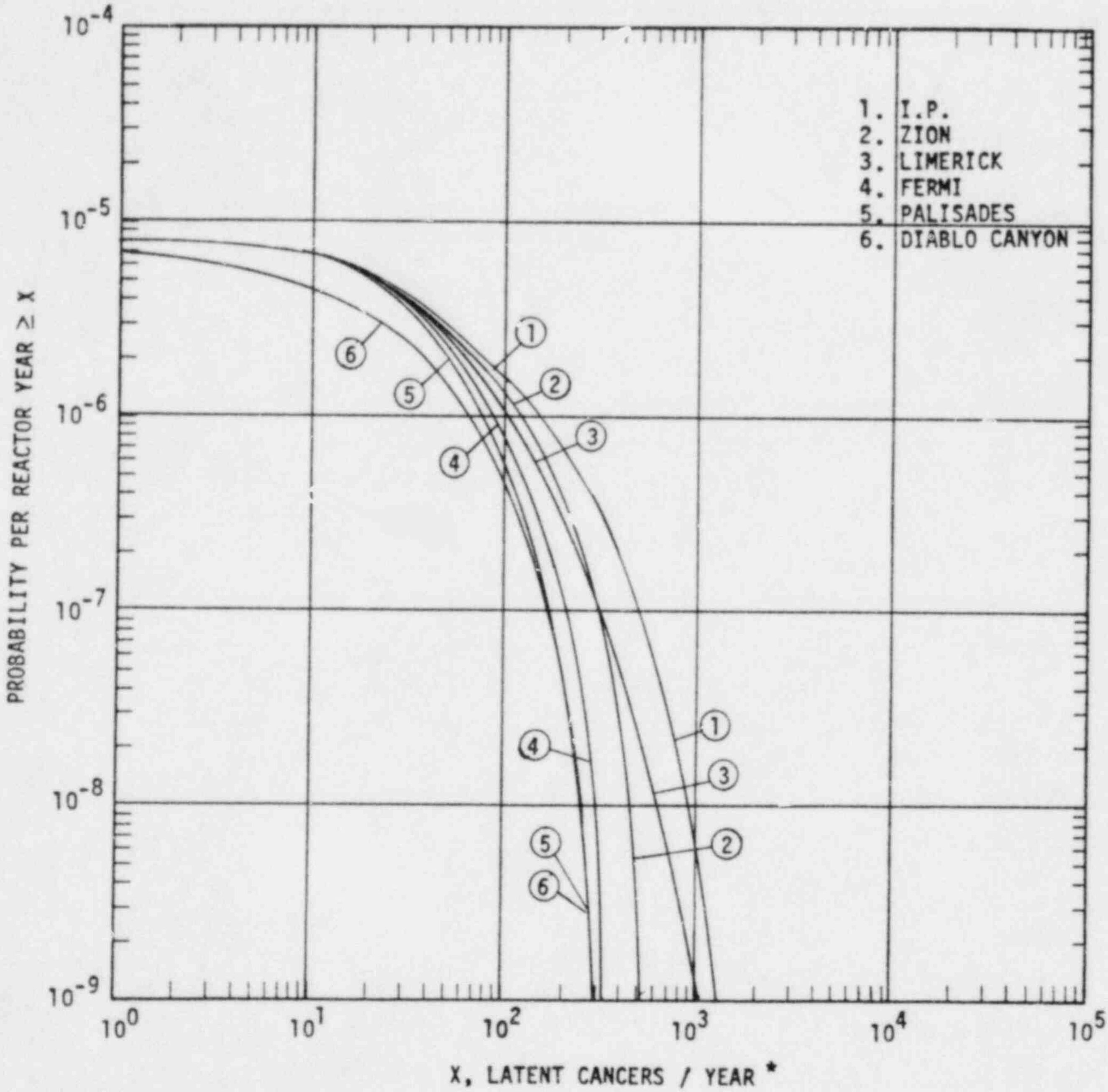
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY.
4) WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION.
5) IDENTICAL 91 WEATHER SEQUENCES FOR ALL SITES.

FIGURE 3 - LATENT CANCER RISK (ANNUAL) FOR DIFFERENT SITES

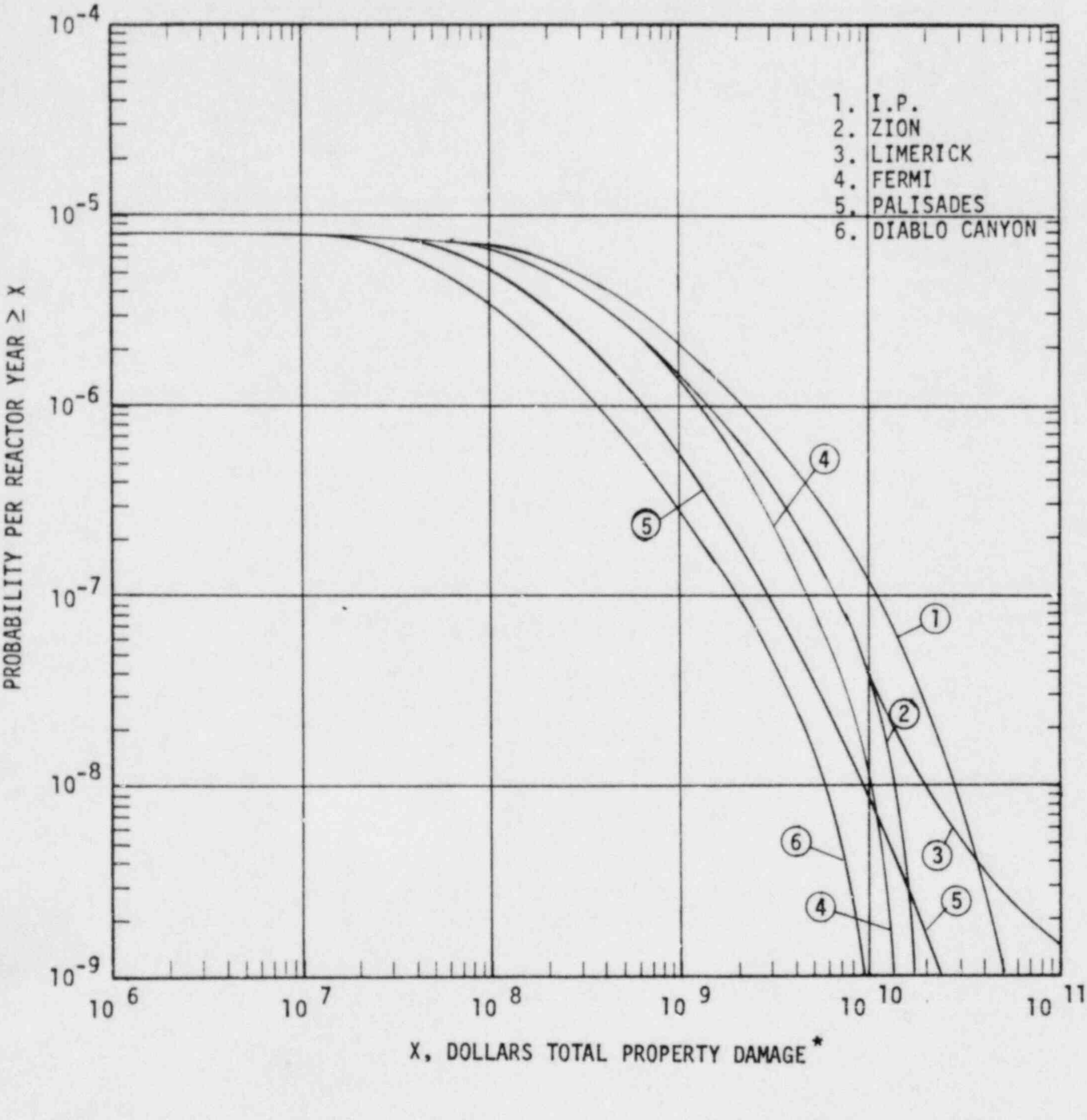



*TOTAL LATENT CANCERS WOULD BE 30 TIMES HIGHER

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS:
1) SURRY DESIGN.
2) I.P. UNIT 3 POWER LEVEL (3025 MWT).
3) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY.
4) WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION.
5) IDENTICAL 91 WEATHER SEQUENCES FOR ALL SITES.

FIGURE 4 - PROPERTY DAMAGE RISK FOR DIFFERENT SITES




* BASED ON 1974 DOLLARS

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS:
1) SURRY DESIGN
2) I.P. UNIT 3 POWER LEVEL (3025 MWT)
3) WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
4) IDENTICAL 91 WEATHER SEQUENCES FOR ALL SITES.

reason for selecting 10 miles as the radius for emergency planning zones (see NUREG-0396, Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants).

Radiation injuries, caused by doses of 50 Rem or more, can reach farther out in the event of a severe reactor accident, to the population as far as 50 miles away. Therefore, the population up to that distance away is significant in estimating the early illness risk; the population beyond 50 miles is not. The estimation of latent cancer fatalities includes even low exposures so populations as far away as 200 miles will significantly influence the latent cancer risk estimate. Thus, for the latent cancer risk, the differences between sites are relatively small since the populations of such large regions are frequently similar.

Figure 1 shows that the three sites with the highest local population density, Indian Point, Zion and Limerick, have essentially the same risk profile for early fatalities. The other sites show progressively lower risks. As was discussed, early fatality risk is dominated by the population within 10 miles of the plant, so the large population of New York City is not a factor here. The absolute values of these risk estimates are subject to large uncertainties but the range should be noted. For low probability--high consequence events, thousands to tens of thousands of early deaths are estimated for most sites.

Early illnesses are defined as radiation exposures in excess of 50 Rem, whole body for an individual. These illnesses or injuries, shown in

Figure 2, are dominated by the size of the population within a 50 mile radius. Thus, New York City is important to the risk of early illness for Indian Point. Zion, Limerick and Fermi also have enough population in the 50 mile range to be comparable to Indian Point as shown by Figure 2. Also for this aspect of risk, the typical Palisades site and the Diablo Canyon site are not very different from each other but are substantially lower than the others. For the sites with higher population density, thousands to hundreds of thousands of early illnesses are projected for the lower probability events.

The latent cancer risk, as shown in Figure 3, is dominated by the population within about a 200 mile radius of the plant. Because of this, the individual site risk curves for latent cancers reflect the character of the region. Remember that Indian Point is outside New York City, Zion outside Chicago on the north shore, Limerick to the northwest of Philadelphia, and Fermi near Detroit. Palisades is on the western side of the Michigan lower peninsula and Diablo Canyon is on the California coast well above Santa Barbara. The latent cancer risk for these sites, and probably all other sites is approximately the same. The number of latent cancer deaths projected is on the order of hundreds per year or thousands per accident for the lower probability events (on the order of $10^{-9}$/yr).

Please note that the latent cancer risk is presented throughout this discussion as latent cancers per year, that is, the average number of cancer deaths that would be expected to occur per year in the population which was exposed to the accident. The total number of latent cancer deaths associated with an accident would be 30 times higher, reflecting the calculated rate of cancer death continuing for a generation. For further discussion of latent cancer risk see NUREG-0340 at page 30.

The curves for property damage are presented in Figure 4. The model still calculates in 1974 dollars; the correction for inflation is probably about a factor of 1.5. The flatness of the curve at the upper left indicates that any accident with substantial releases will cause damage of many millions of dollars. The projected damage for low probability events reaches up into the range of tens of billions of dollars. However, the property damage here does not include damage to the plant. The Three Mile Island accident, which did no offsite property damage, caused several hundred million dollars worth of damage to the plant and replacement power costs, analogous to interdiction costs, on the order of a billion dollars. The property damage risk estimate is directly proportional to population density. With the present property damage model (see NUREG-0340 at page 22) the population out to about 30 miles is significant. However, the use of more strict interdiction and cleanup criteria, as may well be warranted, would make populations beyond that distance important.

The estimated overall probability of core melts for the benchmark reactor (Surry) rebaselined* from WASH-1400 is about one chance out of twenty

*The Reactor Safety Study plants were "rebaselined" for all the analyses presented in this report in order to take into account peer group comments (e.g., the Lewis Committee) and to use better data and analytical techniques which are now available such as the MARCH and CORRAL codes. Further discussion of this rebaselining is presented in Appendix B.

thousand ($5x10^{-5}$) per reactor year. The CCDF curves have been constructed to display the probability vs. consequence relationship for those cases of core melt accidents where offsite harm is done. Note that the majority of core melts are not estimated to do harm offsite. For example, in Figure 1 the benchmark Surry reactor at the Indian Point site is predicted to cause one or more acute fatalities at a frequency of $3.2x10^{-6}$/yr. This means that only $3.2x10^{-6} \div 5x10^{-5}$ = .064 or less than 10 percent of the core melt accidents are predicted to give lethal doses offsite. Conversely about 90 percent of the core melt accidents are not expected to produce lethal doses for that plant. For other plants a larger or smaller fraction of core melt accidents may be expected to cause lethal doses offsite. Our ability to predict how often core melt accidents occur is very limited. However, we are quite reasonably confident from the work so far that most core melt accidents will not give lethal doses offsite. Only certain accident scenarios in the plant, those entailing core meltdown and gross containment failure, coincident with particularly adverse weather conditions, will result in lethal doses or severe offsite ground contamination (i.e., property damage). However, those few core melt accidents that do give lethal doses are likely to do so over a significant area (out to a few miles downwind). If even one person receives a lethal dose offsite, it is quite likely that one thousand will receive a lethal dose. However, in no case are more than a few tens of thousands predicted to receive lethal doses. No combination of weather conditions, ineffectual emergency response and severe accident can be found at any probability that is realistically expected to give lethal doses to as many as one hundred thousand. There are, of course, higher numbers of latent casualties predicted for such accidents, as can be seen in Figure 3.

Consider the differences among the curves; the curves have been constructed on logarithmic scale, which tends to minimize small differences. There are a few perspectives which the CCDFs should clearly provide. For illustrative purposes consider Figure 1; Early Fatality Risk for Different Sites. The probability axis shows the chance of equalling or exceeding a number of early fatalities per reactor year. At 10 fatalities, the range of probabilities for the sites represents the variation between sites of the likelihood of having at least 10 people receive lethal doses. At this level, there is about a factor of 30 difference in probability between the Indian Point and Diablo Canyon sites. Thus, the CCDFs show the variation in probability for given levels of consequences. The CCDFs also give the range of consequences for a given probability level. At the one in one hundred million ($10^{-8}$) probability level, one would expect the Diablo Canyon area population to suffer at least 400 fatalities whereas the number of fatalities estimated at Indian Point would be about 10,000 or more.

In addition to the probability and consequence perspective, the curves give a sense of the importance of the consequences and probabilities. When the curves have a clear knee in them, that is they have an approximately horizontal slope out to some level of consequences and then fall off sharply (see the Indian Point curve in Figure 1, the knee is at about the 4,000 fatalities level) the most important part of the curve is the horizontal portion where one would expect to have about an equal chance of suffering consequences up to about that "knee" level. When the curve drops off, the uncertainties become very large and the importance of perceived differences should be minimized. When the curves do not have a clear knee, as in the case of Indian Point on Figure 2, the probabilities are dropping at about the same rate as the consequences are increasing. This result leaves a question as to the limit of how many consequences could be expected. That is, the low probability-high consequence range (bottom right of curve) is clearly contributing to the overall risk.

The risk curves in Figures 1-4 can be reduced to probability weighted values, or expected consequences and these can be termed the likelihood of the consequence occurring in a year. Table 5 presents these expected consequences. The principal differences between the risks at these sites is seen to be in early fatalities and injuries. The Indian Point site poses about 20 times more risk of early fatality than a typical site such as Palisades. With respect to early injuries, the Indian Point site is about 10 times more risky than Palisades. The differences in other aspects of risk are not so great.

The risks of early fatalities and early illnesses for the Indian Point site alone where only public protective measures are changed are shown in Figures 5 and 6, respectively. For the Indian Point site alone, the sensitivity of early fatalities and early illness to no evacuation at all until a day after the accident, to differences in evacuation radius,

TABLE 5

EXPECTED ANNUAL CONSEQUENCES (RISK)+ FROM 6 SITES
WITH THE SURRY REBASELINED PWR DESIGN

| Site \ Probability-Weighted Consequence per yr | Early Fatalities | Early Injuries | Latent Cancer/Yr* | Property Damage $** |
|---|---|---|---|---|
| Diablo Canyon | $1.6\times10^{-5}$ | $2.5\times10^{-4}$ | $1.8\times10^{-4}$ | 1290 |
| Palisades | $2.9\times10^{-4}$ | $1.2\times10^{-3}$ | $2.7\times10^{-4}$ | 2670 |
| Fermi | $9.2\times10^{-4}$ | $6.3\times10^{-3}$ | $3.6\times10^{-4}$ | 4780 |
| Limerick | $3.5\times10^{-3}$ | $1.1\times10^{-2}$ | $4.7\times10^{-4}$ | 6980 |
| Zion | $4.7\times10^{-3}$ | $1.2\times10^{-2}$ | $4.3\times10^{-4}$ | 6030 |
| Indian Point | $6.1\times10^{-3}$ | $1.5\times10^{-2}$ | $5.4\times10^{-4}$ | 9550 |

*Total Latent Cancers Would Be 30 Times Higher

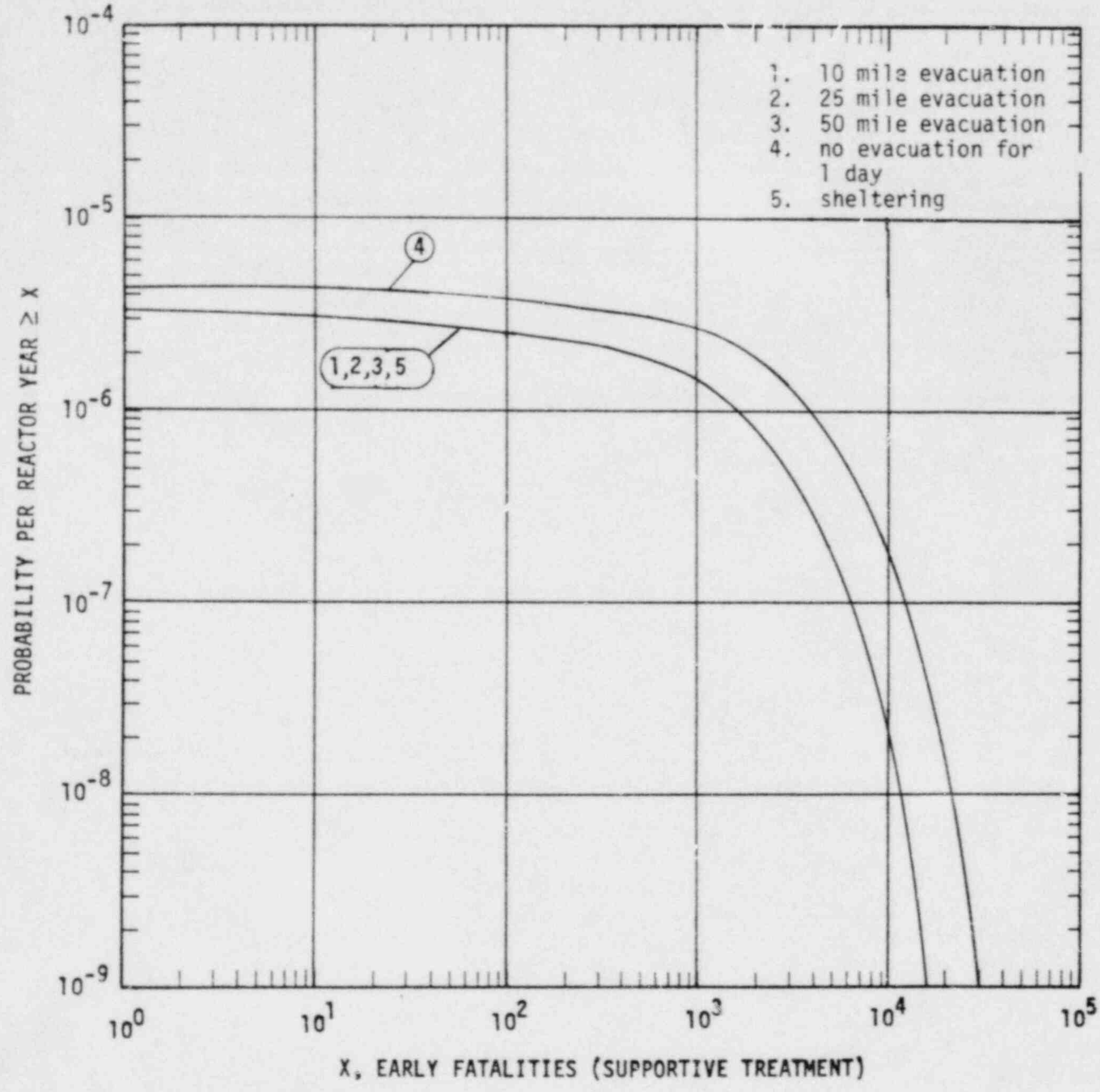
**Based on 1974 Dollars

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS TABLE.

ASSUMPTIONS:
1. SURRY DESIGN.
2. I.P. UNIT 3 POWER LEVEL (3025 MWT).
3. WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE NO SHIELDING
   BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY.
4. WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION.
5. IDENTICAL 91 WEATHER SEQUENCES FOR ALL SITES.

+ The expected annual consequence, or risk, is the sum of the products of the probabilities and consequences (early fatalities; early injuries, etc) for the various accident sequences considered in the study.

FIGURE 5 - EARLY FATALITY RISK AT INDIAN POINT FOR VARIOUS PUBLIC PROTECTION MEASURES




NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

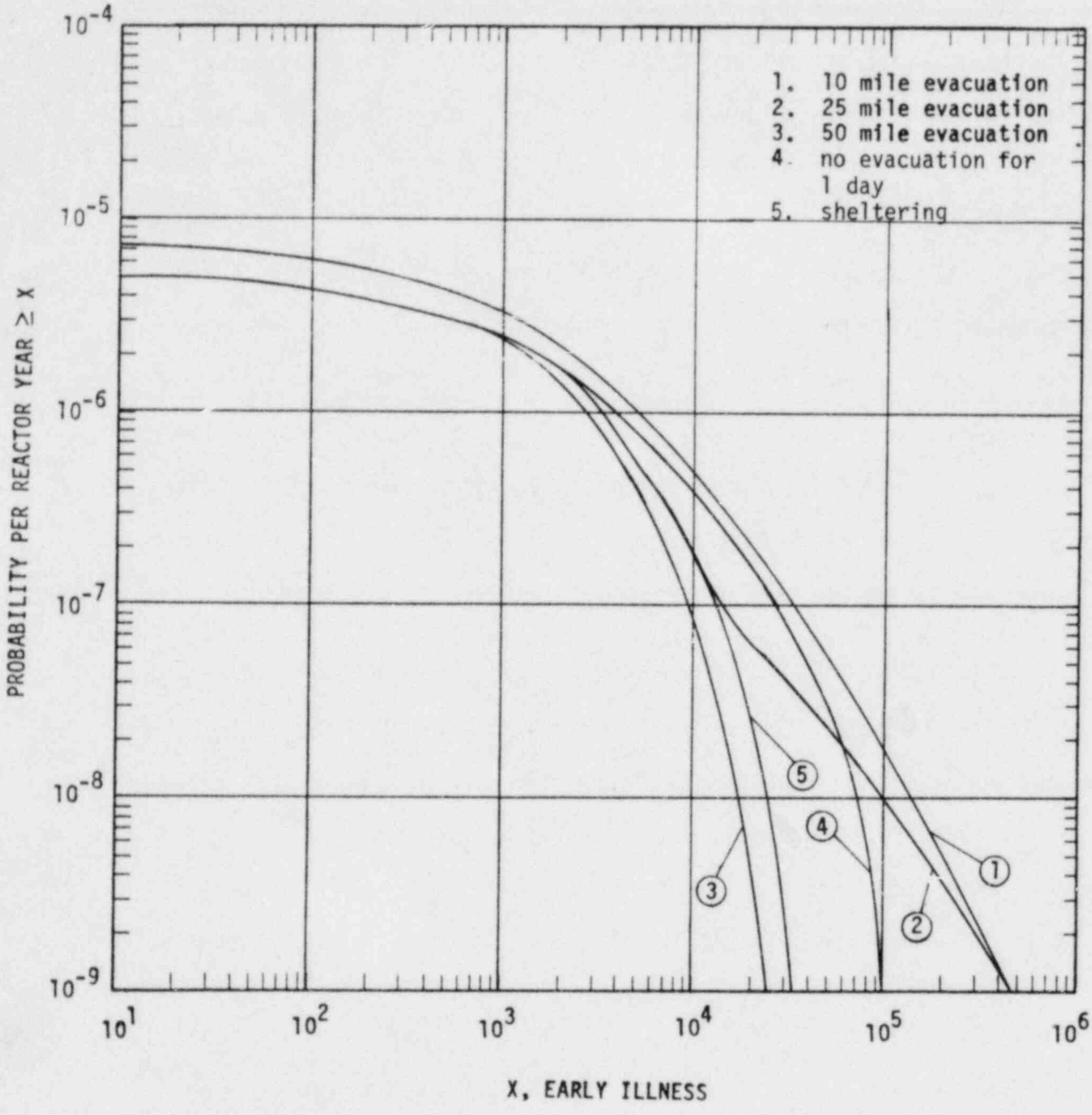
ASSUMPTIONS: 1) SURRY DESIGN.
2) I.P. UNIT 3 POWER LEVEL (3025 MWT).
3) WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
4) INDIAN POINT SITE (POPULATION AND METEOROLOGY)

EVACUATION SCENARIOS - ENTIRE CLOUD EXPOSURE + EITHER 4 HOURS GROUND EXPOSURE, NO SHIELDING WITHIN GIVEN DISTANCE; OR 7 DAYS GROUND EXPOSURE, NORMAL SHIELDING BEYOND GIVEN DISTANCE

NO EVACUATION - ENTIRE CLOUD EXPOSURE + 1 DAY GROUND EXPOSURE, NORMAL SHIELDING

SHELTERING - ENTIRE CLOUD EXPOSURE + 1 DAY GROUND EXPOSURE, SHIELDING ASSUMES BRICK HOUSE WITH NO BASEMENT.

FIGURE 6 - EARLY ILLNESS RISK AT INDIAN POINT FOR VARIOUS PUBLIC PROTECTION MEASURES




NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS: 1) SURRY DESIGN
2) I.P. UNIT 3 POWER LEVEL (3025 MWT)
3) WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
4) INDIAN POINT SITE (POPULATION AND METEOROLOGY)

EVACUATION SCENARIOS - ENTIRE CLOUD EXPOSURE + EITHER 4 HOURS GROUND EXPOSURE, NO SHIELDING WITHIN GIVEN DISTANCE; OR 7 DAYS GROUND EXPOSURE, NORMAL SHIELDING BEYOND GIVEN DISTANCE

NO EVACUATION - ENTIRE CLOUD EXPOSURE + 1 DAY GROUND EXPOSURE, NORMAL SHIELDING

SHELTERING - ENTIRE CLOUD EXPOSURE + 1 DAY GROUND EXPOSURE, SHIELDING ASSUMES BRICK HOUSE WITH NO BASEMENT.

namely, 10, 25 and 50 miles and sheltering were analyzed. For Indian Point, this last would include New York City itself. In Figure 5 for early fatalities, only two curves are shown, one for no evacuation for one day and a second curve representing a range of the public protection options since their differences are too small to distinguish. All evacuations are assumed to include direct exposure of the people to the cloud and then four hours of ground exposure while evacuating. Obviously, if one assumed that the evacuees could leave before suffering less or even any cloud and ground exposure, the risk profile would be drastically lowered. Since early fatalities are dominated by the population within the first 10 miles, evacuating beyond that range produces little reduction in early fatalities.

The early illnesses that could be suffered around the Indian Point site with varying public protection strategies is shown in Figure 6. The lowest risk is with a 50 mile evacuation. The alternative of sheltering for a period of 24 hours and then evacuating selectively appears to provide nearly the same risk reduction for the Indian Point environs. The other alternatives depicted do not appear to offer as much benefit for the low probability-high consequence events.

THE EFFECT OF DESIGN ON RISK AT INDIAN POINT

The extensive use of quantitative risk assessment for U. S. power reactors began with the Reactor Safety Study (RSS), WASH-1400, which studied a 3-loop Westinghouse PWR, Surry, and a General Electric BWR, Peach Bottom. Since the Reactor Safety Study, other reactor risk assessments of somewhat lesser depth have been made. For example, the NRC staff has been pursuing the Reactor Safety Study Methodology Application Program. This program is considering four reactors: Sequoyah, a Westinghouse 4-loop PWR with ice condenser containment; Oconee, a Babcock-Wilcox 2-loop PWR with dry containment; Calvert Cliffs, a Combustion Engineering 2-loop PWR with dry containment; and Grand Gulf, a General Electric BWR with Mark III containment. These designs are being reviewed with application of the Reactor Safety Study event and fault tree techniques. The reports on these studies will not be complete until later this year but some of the preliminary results are available to the staff.

The staff recently began a new program, the Interim Reliability Evaluation Program. The first plant covered in this program is Crystal River 3, a Babcock and Wilcox 2-loop PWR with dry containment. The initial report on this study is now in peer review, and its preliminary results are available to the staff. Also available for comparison are the results of the German reactor risk study of the Biblis B reactor.

The staff used the information gained from these studies to guide a short term risk evaluation of the Indian Point 2 and 3 plants. This evaluation relies heavily on the judgement of the reviewer with respect to the accident sequences being considered and to the parts of the plants involved. The approach was to consider the key accident sequences which involve core meltdown* or containment failure modes that would be expected to dominate risk. The Indian Point plants were briefly reviewed against these scenarios and their designs were surveyed for single point vulnerabilities such as single manual valves or human errors which can trigger or control a significant accident sequence. Particular attention was given to common interactions which could cut across more than one system or be caused by a single initiating event. Rough estimates were made of the likelihood and consequences of various sequences using the data and release characteristics of previous studies, particularly the Reactor Safety Study and its follow-on work, the Methodology Application Program. Prior risk studies showed that a handful of accident scenarios would most likely define and dominate a reasonably complete spectrum of core melt accident scenarios for the PWR design. Table 6 lists the accident scenarios which were so considered and which were among those quantitatively estimated for the Indian Point 2 and 3 study. We found no risk significant differences between the Indian Point 2 and 3 designs.

An estimate of the overall probability of severe core damage or core melt was made for Indian Point 2 and 3 as of December 1979. Then the estimate was revised to reflect those changes that were made or committed to in early 1980. This very preliminary estimate for Indian Point indicates an initial probability of severe core damage of about $3 \times 10^{-5}$

*Here, as in WASH-1400, the terms core meltdown and severe core damage are used interchangeably. The analysis presumes procession to core melting once severe damage is suffered.

TABLE 6

DOMINANT ACCIDENT SEQUENCES

| Accident Scenario | Sequence Code From WASH-1400 | Offsite Consequences Expected |
|---|---|---|
| LOCA and failure of ECCS in injection mode | AD<br>$S_1D$<br>$S_2D$ | Low to modest |
| LOCA and failure of ECCS in recirculation mode | AH<br>$S_1H$<br>$S_2H$ | |
| Transient and loss of feedwater or serious failure and no feed and bleed on primary side (X) | TMLX<br>TMKX | |
| LOCA and loss of containment heat removal with subsequent interactions with ECCS | AG<br>$S_1G$<br>$S_2G$ | Intermediate |
| LOCA and failure of ECCS and containment ESFs in recirculation phase due to common cause | AHF<br>$S_1HF$<br>$S_2HF$ | High |
| LOCA and coupled damage to ECCS and potential bypass of containment | Event V | |
| Transient involving loss of all AC power (or possibly DC) and failure of auxiliary feedwater | TMLB' | |

per year. The improvements made or committed to this year are estimated to reduce that probability by a factor of three to about $1x10^{5}$ per year. For comparison, Table 7 presents the estimated probability of severe core damage for the Indian Point reactors along with similar estimates from the Reactor Safety Study and other studies mentioned previously. The overall effect of the Indian Point improvements is estimated to be a three-fold reduction in the probability of severe core damage if these improvements are successfully implemented. As it turns out, it is not important to this overall analysis to determine whether each of the committed changes has been made and when. The changes committed to are clearly beneficial in reducing risk but it is questionable whether the factor of improvement, three, is statistically significant. The probabilities of severe core damage listed in Table 7 are subject to at least a factor of 5 uncertainty in either direction due to uncertainties in the data upon which all this analysis is based. Therefore, one should be very careful about attaching significance to differences in these estimates which are less than about one order of magnitude.

The effect on risk at the Indian Point site is best seen by comparison of the CCDF's. Figure 7 shows the early fatality risk curves for five different reactor designs, all at the Indian Point site, including the early fatality risk curves estimated for the Indian Point 2 reactor before the 1980 changes and after the 1980 changes.

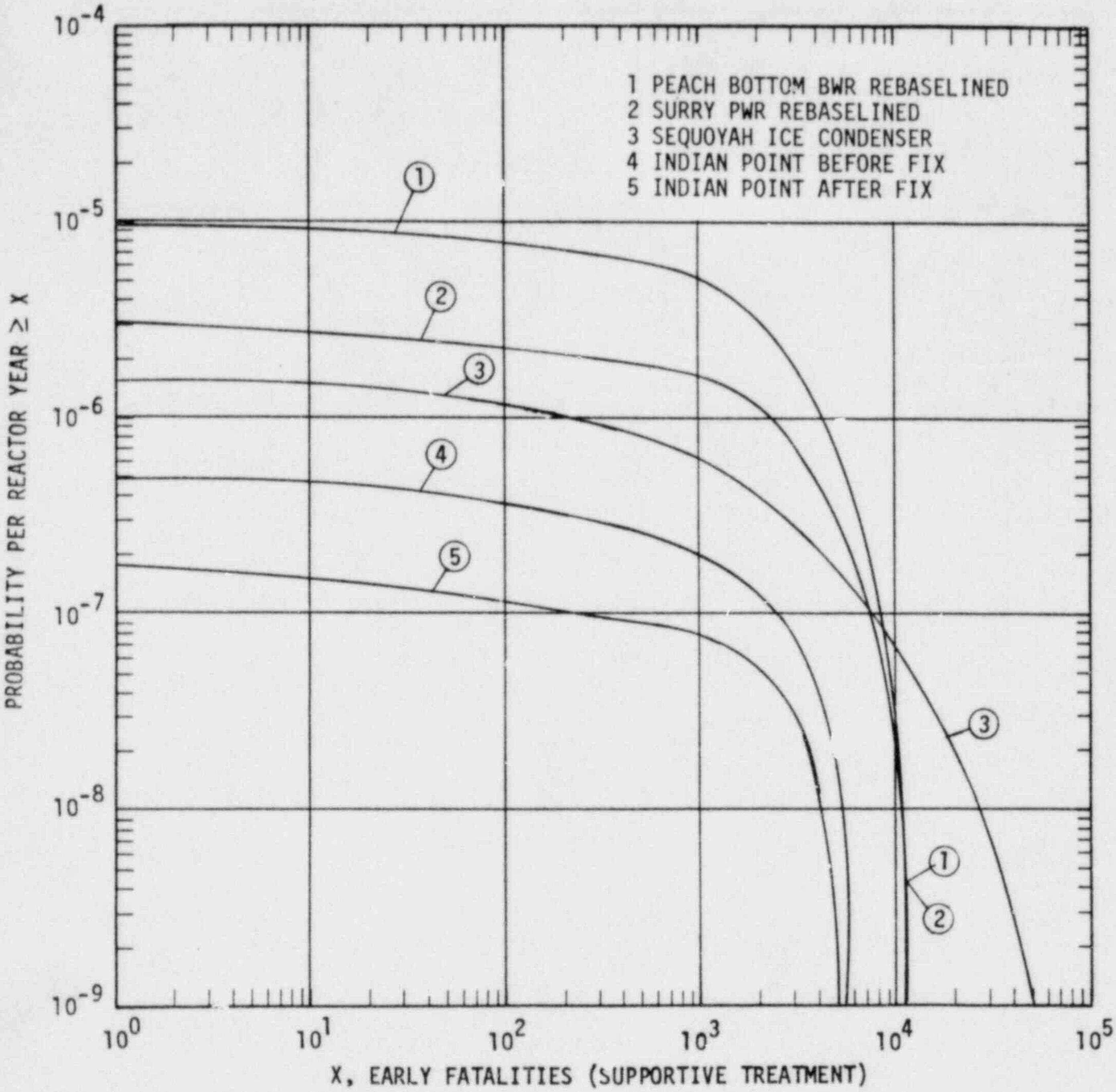
Figures 8, 9 and 10 display the same comparisons for the other risk indicators, early injuries, latent fatalities and property damage.

TABLE 7 - ESTIMATED PROBABILITY OF SEVERE CORE DAMAGE

| REACTOR NAME | TYPE | PROBABILITY* OF SEVERE CORE DAMAGE PER REACTOR-YEAR |
|---|---|---|
| SURRY | 3-loop PWR | $6x10^{-5}$ |
| PEACH BOTTOM | BWR (Mark I) | $3x10^{-5}$ |
| SEQUOYAH | 4-loop PWR (Ice Condenser) | $4x10^{-5}$ |
| OCONEE | 2-loop PWR | $2x10^{-4}$ |
| CALVERT CLIFFS | 2-loop PWR | $2x10^{-4}$ |
| CRYSTAL RIVER-3 | 2-loop PWR | $3x10^{-4}$ |
| BIBLIS | 4-loop PWR | $4x10^{-5}$ |
| INDIAN POINT | 4-loop PWR | $1x10^{-5}$ |

* Reflects median values

FIGURE 7 - EARLY FATALITY RISK FOR DIFFERENT DESIGNS

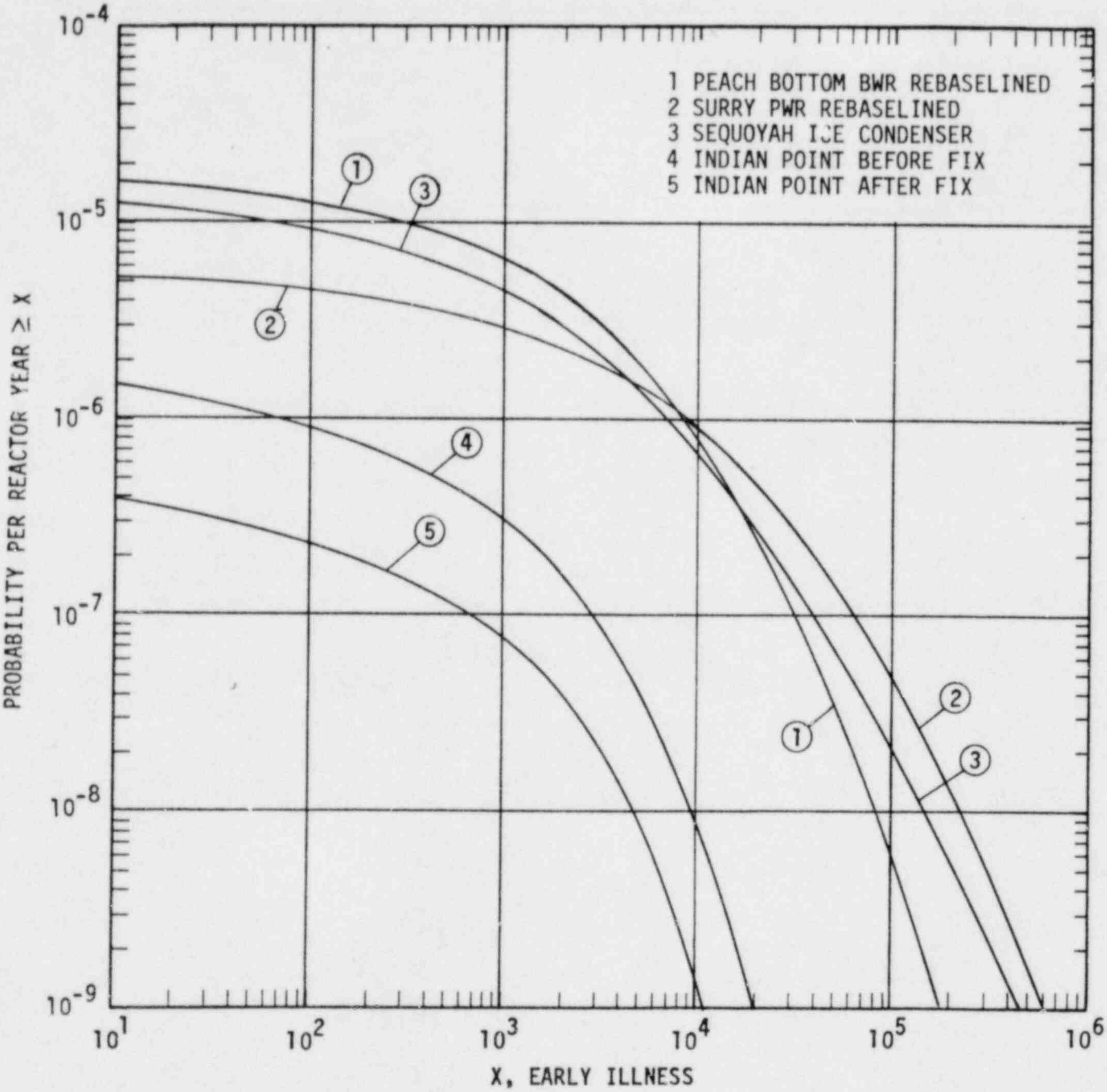



NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)

2) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY

FIGURE 8 - EARLY ILLNESS RISK FOR DIFFERENT DESIGNS

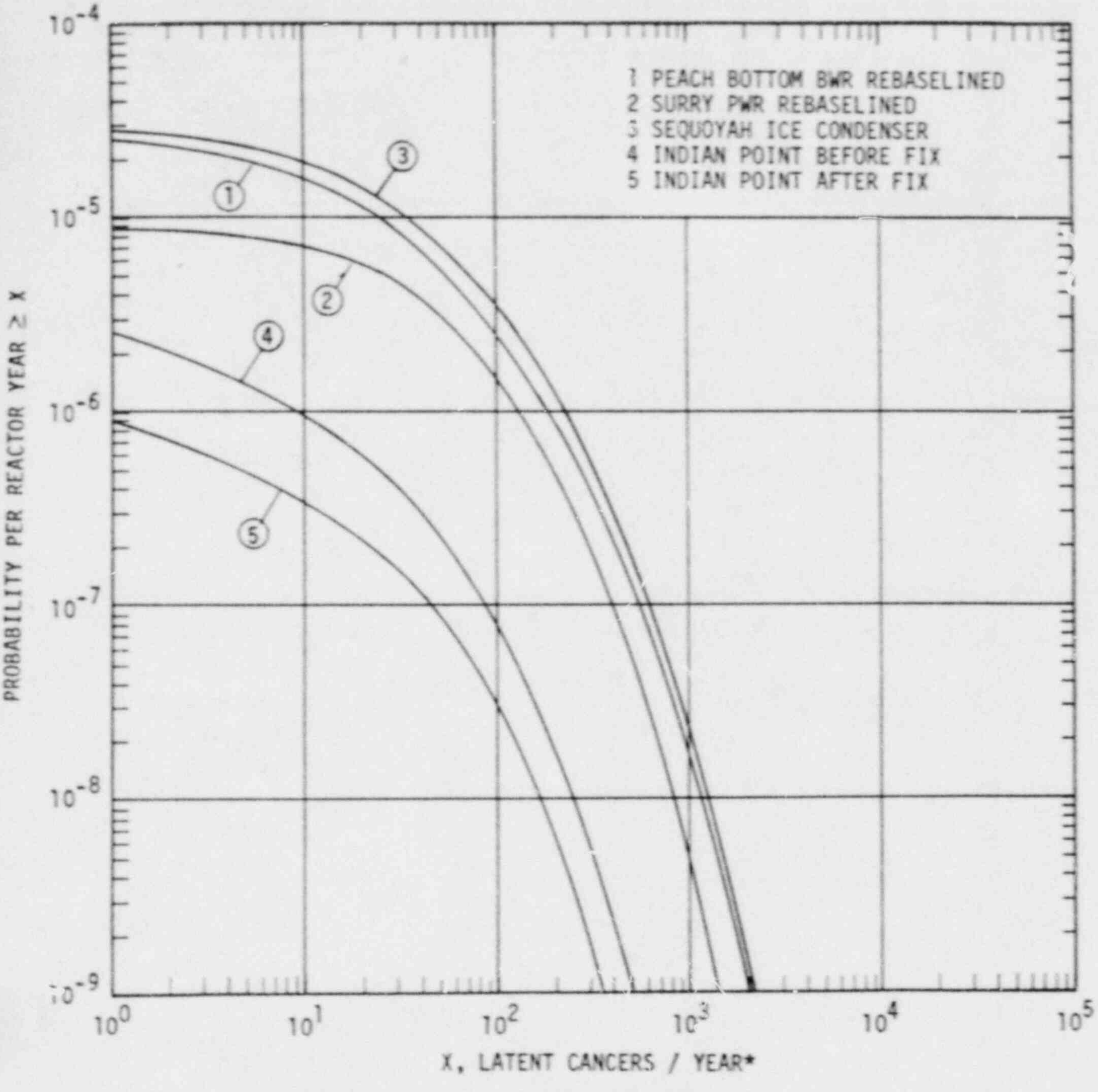



NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)

2) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY

FIGURE 9 - LATENT CANCER RISK FOR DIFFERENT DESIGNS

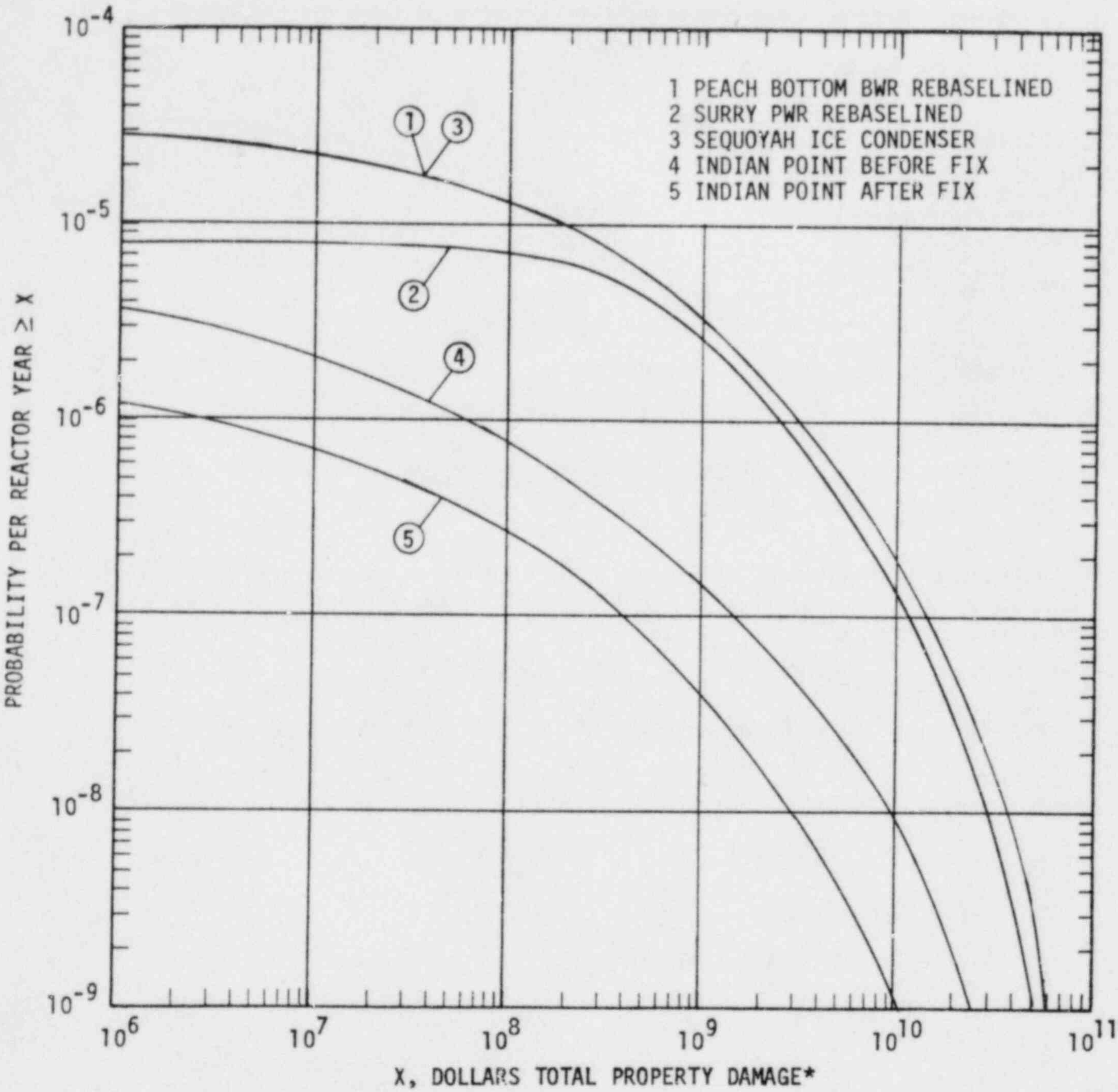



*TOTAL LATENT CANCERS WOULD BE 30 TIMES HIGHER

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE.

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)
2) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY

FIGURE 10 - PROPERTY DAMAGE RISK FOR DIFFERENT DESIGNS




*BASED ON 1974 DOLLARS

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)

The reactor designs whose risk profiles are considered here include the two reactors considered in the Reactor Safety Study, Surry and Peach Bottom; the Sequoyah plant with its ice condenser and the two versions of the Indian Point design. The risk profiles are presented only for these reactors and not the others listed in Table 7 because there was not time to do the others.

When considering the CCDFs presented in Figures 7, 8, 9 and 10, it is important to keep the uncertainties in mind. WASH-1400 assigned an uncertainty of plus or minus a factor of five to analysis such as this. The Lewis Committee questioned that small an uncertainty. We believe it is prudent to consider that these curves have an uncertainty, plus or minus, of about a factor of 10 at the higher probabilities and perhaps as much as a factor of 100 at the lower probabilities. Thus, one can attach significance to the range shown but not to modest differences between curves.

As indicated by the curves, the risk of the Indian Point reactors appears to be even lower compared to the other reactors than the ratio of their core damage probabilities would suggest. Table 8 presents the expected annual consequences or the risk from these five different designs at the Indian Point site. If one postulates that the Surry design is a typical reactor, then "Indian Point After Fix" poses about 30 times less risk of early fatalities, about 50 times less risk of early injuries, about 30 times less risk of latent cancers, and about 50 times less risk of property damage. At this time, not enough is known about the overall risk profile of all the individual plants in the U.S. to say what is typical or even what the range is. The variation of the design and operation parameter done in this analysis was based on information available, not on identifiable bounds.

TABLE 8

EXPECTED ANNUAL CONSEQUENCES (RISK)+ FROM 5 LWR DESIGNS
AT THE INDIAN POINT SITE

| Design \ Probability-weighted consequence per yr | Early Fatalities | Early Injuries | Latent Cancer/Yr* | Property Damage $** |
|---|---|---|---|---|
| IP After Fix | $2.2x10^{-4}$ | $2.7x10^{-4}$ | $1.6x10^{-5}$ | 199 |
| IP Before Fix | $6.3x10^{-4}$ | $9.5x10^{-4}$ | $4.4x10^{-5}$ | 700 |
| Surry Rebaselined | $6.1x10^{-3}$ | $1.5x10^{-2}$ | $5.4x10^{-4}$ | 9550 |
| Sequoyah Ice Condenser | $2.7x10^{-3}$ | $2.2x10^{-2}$ | $1.2x10^{-3}$ | 14800 |
| Peach Bottom BWR Rebaselined | $1.7x10^{-2}$ | $3.1x10^{-2}$ | $1.1x10^{-3}$ | 13500 |

*Total Latent Cancers Would Be 30 Times Higher

**Based on 1974 Dollars

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS TABLE.

ASSUMPTIONS:
1. INDIAN POINT SITE
   METEOROLOGY - 91 WEATHER SEQUENCES
   WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
   UNIT 3 POWER LEVEL (3025 MWT)
2. WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
   NO SHIELDING
   BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
   SHIELDING BASED ON NORMAL ACTIVITY

\+ The expected annual consequence, or risk, is the sum of the products of probabilities and consequences (early fatalities; early injuries, etc) for the various accident sequences considered in the study.

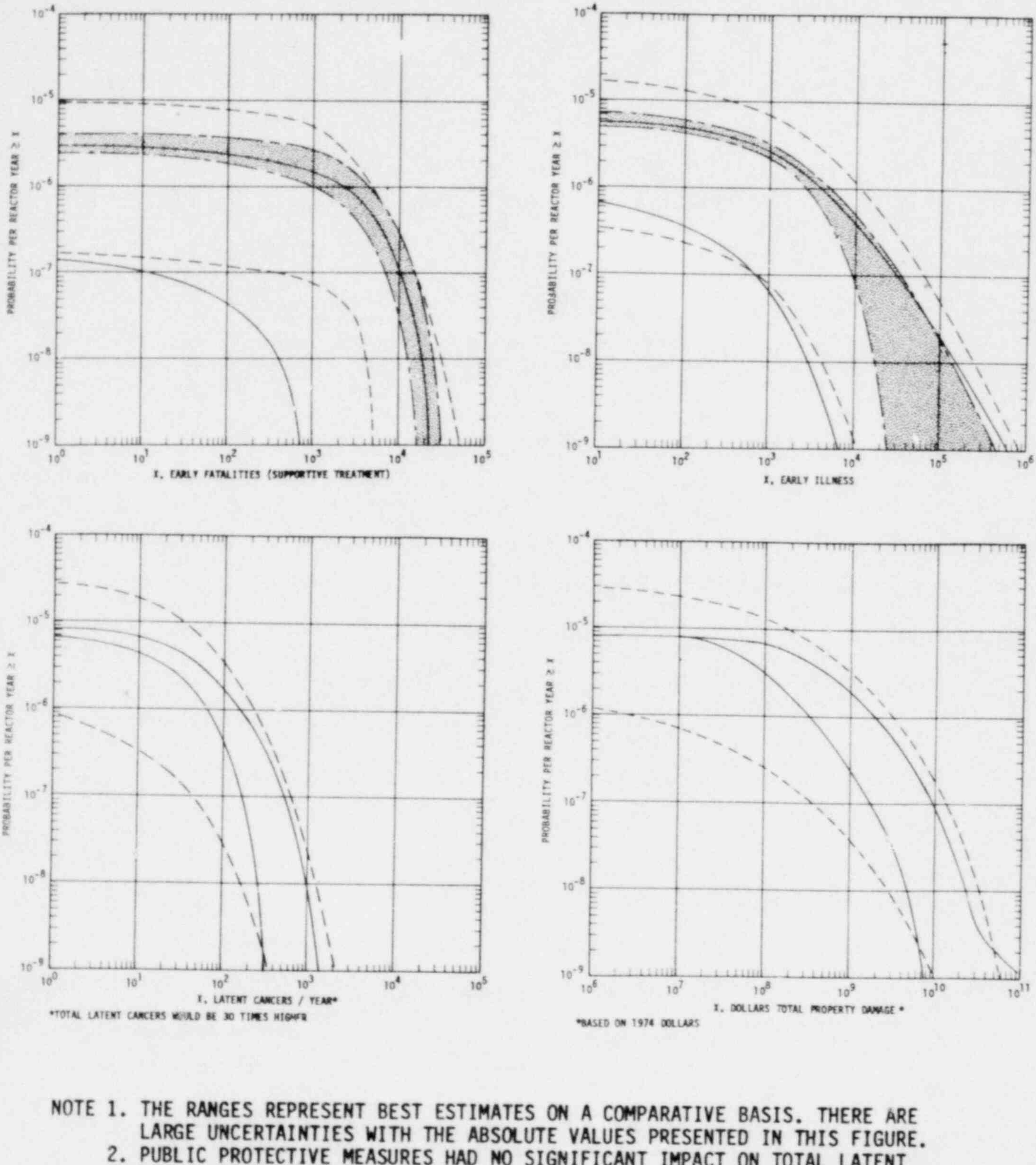
## THE SENSITIVITY OF RISK TO VARIATIONS IN SITE, PUBLIC PROTECTION, AND DESIGN/OPERATING CHARACTERISTICS

In the preceeding sections the risk was considered for variation of three basic parameters, the reactor site, the public protection measures taken, and the different reactor plant design and operating characteristics. For the first, a single reactor design, Surry, was placed at six different sites. The degree of uncertainty in this site comparison is not as great as for the design comparison because, although there are substantial uncertainties in the model, the sites differ only by two relatively well understood parameters, demography and meteorology. The demography differences dominate the comparison. The same degree of uncertainty exists for the public protection measure variation, since no evacuation logistics analysis is made here. The model used for these analyses works just on the demography.

For the design variation there is much greater uncertainty. The comparison of one plant to another involves different levels of study, different dominant accident scenarios, and the use of a great deal more judgment by the analyst. Previous work by the staff in evaluating the reliability of auxiliary feedwater systems in many PWRs was done on a more consistent basis, where each plant received approximately the same depth and scope of analysis. The results of that analysis showed reliability variations for that one important system from plant to plant ranging over two orders of magnitude, about as much as was shown here for site variation.

Figure 11 was drawn to display the range of variation for the three parameters of this analysis. On each of the four graphs shown in Figure 11, the solid lines show the bounds of variation when the same reactor was moved from site to site. The long-short-long lines with shading in the first two graphs show the bounds for variation of public protective action options, all with the pessimistic (or realistic) exposure assumptions described previously. The dashed lines on all four graphs show the range of variation of a few reactor designs that were analyzed. We expect the full range of variation of risk due to design factors from the best to worst plant in the country to be broader than the small sample shown here. Figure 11 suggests that the most significant parameter affecting risk is the design and operation of the plant. The site is a significant variable more for early effects and the public protection options as shown here are the least significant.

FIGURE 11 - RANGES OF RISK VARIATION




NOTE 1. THE RANGES REPRESENT BEST ESTIMATES ON A COMPARATIVE BASIS. THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE.

2. PUBLIC PROTECTIVE MEASURES HAD NO SIGNIFICANT IMPACT ON TOTAL LATENT CANCER OR ON TOTAL PROPERTY DAMAGE.

ESTIMATED RANGE OF CONSEQUENCES FOR VARIOUS DESIGNS CONSIDERED AT INDIAN POINT SITE.

ESTIMATED RANGE OF CONSEQUENCES FOR 6 SITES CONSIDERED WITH SURRY DESIGN.

ESTIMATED RANGE OF CONSEQUENCES FOR VARIOUS PUBLIC PROTECTIVE MEASURES CONSIDERED AT INDIAN POINT SITE.

THE RISK OF AN INDIAN POINT REACTOR COMPARED TO OTHER REACTORS

The preceeding sections examined the risk of the Indian Point site and the Indian Point reactor designs separately. From those examinations it appears that the site is about an order of magnitude more risky than a typical site and the design about as much less risky than a typical design. There is much more certainty in our comparison of the relative site risks than there is in the comparison of the design risks. It is reasonable to conclude that the two about cancel, that is, the overall risk of the Indian Point reactor is about the same as a typical reactor on a typical site. We recognize that such a comparison makes no explicit compensation for the Indian Point risk entailing notably higher consequences even if at lower probability than is typical. It is not unusual in risk aversion to demand lower risk as the potential consequences increase - as the stakes get higher. Accordingly, one might argue that the probability should be more than a magnitude lower if the consequences can be a magnitude higher.

REDUCTION OF OPERATING POWER LEVEL

Obviously, reactor accident risk can be essentially eliminated by shutting down the reactor. Reducing the operating power level can reduce risk in two ways by reducing the potential consequences of an accident and by reducing the probability of an accident occurring or running its course. Reducing the operating power level of a reactor does not reduce the potential consequences proportionately until long after the power level reduction is enforced. A typical PWR core is divided into three sets of fuel assemblies. One set is replaced at each refueling, so that each fuel assembly experiences three operating cycles in its period of use. The accident risk posed by a reactor arises from the inventory of fission products which builds up in these fuel assemblies. Based on the WASH-1400 analysis, about half that risk comes from iodine isotopes with half-lives of no more than eight days. For these iodine isotopes, the equilibrium inventory level is proportional to power level, and is reached in about a month at that power. After about a month, then the iodine contribution to risk is going to be directly proportional to steady state power level.

The other half of the estimated accident risk is dominated by isotopes of elements such as tellurium, cesium and strontium, having fairly long half-lives, e.g., of years. Some of these isotopes never reach an equilibrium level in the fuel as do the short-lived ones but continue to build up in proportion to both power level and the time spent at that level, in essence, in proportion to the number of fissions. Therefore, an operating power level reduction will not proportionately reduce the risk from these isotopes unless there is also a reduction in the fuel burnup allowed.

The reduction of operating power level can also have an effect on accident risk by reducing the fuel operating temperature levels and by reducing the amount of decay heat which must be removed after shutdown. At lower power levels the heat output of the fuel is lower. Since the coolant temperature remains essentially the same as at full power, the result is lower temperature of the fuel and much of the metal surrounding it. The advantage of reduced fuel temperature in an accident is the fact that the fuel has that much more capability of absorbing heat before it reaches severe damage temperature or melts. Thus, the core can tolerate longer periods without proper cooling before damage is done.

Continued operation at reduced power level will also reduce the amount of decay heat generated after shutdown, in proportion to the degree of power reduction. This, as well as lower fuel temperatures, increases the length of time the core can run without proper cooling before damage occurs. With increased tolerance of poor core cooling, there is more time for corrective action by the operators in the event of an accident.

No quantitative analyses were performed to estimate the degree of risk reduction that can be achieved by reduction of the operating power level but, from the factors involved, it appears reasonable to say that risk would be reduced in proportion to the reduction in power level.

## SECTION 2. SOCIAL AND ECONOMIC IMPACT CONSIDERATIONS

### EFFECTS OF AN INDIAN POINT STATION SHUTDOWN ON ELECTRCIAL POWER RELIABILITY IN THE NEW YORK POWER POOL

The New York Power Pool (NYPP) coordinates the generation and delivery of electric power for the State of New York. Its members operate according to certain standards, including the requirements that NYPP members maintain an installed generating capacity reserve equal to 18 percent of maximum one hour net load. There are seven investor-owned and one state owned utility in the NYPP with a total capacity as of Summer 1979, of nearly 30,000 Mw.

Consolidated Edison represents about 31 percent (9400 Mw) and the Power Authority of the State of New York (PASNY) about 22 percent (6700 Mw) of the total capacity of the NYPP. The electric service area of CON ED consists of the five boroughs of New York City and a major part of Westchester County, an area of 600 square miles with over eight million customers. PASNY does not have any geographically defined "service territory" but serves particular classes of customers in all parts of the State of New York.

Southeastern New York State is a summer peaking region. CON ED's summer peak load, in particular, is about 40 percent higher than its winter peak load mainly due to the widespread use of electric air conditioning. The remainder of New York State is a winter peaking region. The total NYPP System peaks in the summer

According to a recent DOE analysis,[1] attached here as Appendix C, the forecast of the combined 1980 summer peak for CON ED and PASNY is 9403 Mw, as shown in Table 9. Total PASNY and CON ED capacity is approximately 16,000 Mw. If Indian Point Units 2 and 3 are removed from the system and an 18 percent reserve margin is added to the forecast summer 1980 CON ED-PASNY peak, there is still an apparent excess capacity of about 3000 Mw.

However, much of PASNY's capacity is not in the Southeastern New York area, but elsewhere in the State. Major PASNY facilities in Southeastern New York include Indian Point #3 and Astoria #6 with a combined megawatt rating of approximately 1740 Mw.

If one assumes that one-half of the projected summer peak demand for the PASNY system originates in the New York City area,[2] and if the location of PASNY's generating capacity is taken into account, then the reserve picture changes considerably as shown in Table 10. It should be noted that of the total capacity of some 9300 Mw nearly 2000 Mw are combustion turbines which are generally not planned or designed for prolonged operation. Given the projected summer load for the Southeastern New York area, the shutdown of Indian Point #2 and #3 would result in insufficient capacity (by some 250 Mw) to maintain an 18 percent reserve. All of the reserve capacity disappears and energy would have to be imported from other parts of the NYPP if scheduled outages, summer capacity reductions and historically experienced forced outages of some 1500 Mw are accounted for. In addition, if the largest unit (Ravenswood #3 - 928 Mw) is lost, the DOE analysis concludes that the utilities would be forced to use all available capacity and interties to the maximum reasonable extent.

1/Letter to Edward J. Hanrahan from Richard Weiner, Director, Division of Power Supply and Reliability, Economic Regulatory Administration, DOE, May 15, 1980.

2/Letter to Hanrahan, op. cit., p.2, DOE states that PASNY's projected summer 1980 peak load is 2503 Mw "of which less than half is in New York City and Westchester County areas".

Table 9

Reserve Situation for the CON ED and PASNY Systems
(Summer, 1980)
(Mw)

| | CON ED | PASNY | TOTAL |
|---|---|---|---|
| (1) Summer Peak, 1980 | 6900 | 2503 | 9,403 |
| (2) Summer Peak, 1980 + 18% reserve margin | 8142 | 2953 | 11,095 |
| (3) Capacity with Indian Point 2, 3 | 9441 | 6740 | 16,181 |
| (4) Capacity without Indian Point 2, 3 | 8592 | 5775 | 14,367 |
| (4)- (2) Apparent Excess Capacity | 450 | 2822 | 3,272 |

Table 10

Revised Reserve Situation for CON ED and
PASNY Systems (Summer, 1980)
(Mw)

| | CON ED | PASNY | TOTAL |
|---|---|---|---|
| (1) Summer Peak, 1980 | 6900 | 1251 | 8,151 |
| (2) Summer Peak, 1980 + 18% reserve | 8142 | 1475 | 9,617 |
| (3) Capacity without Indian Point 2, 3 | 8592 | 775 | 9,367 |
| (3)- (2) Excess Capacity | 450 | -700 | - 250 |

The bulk power transmission tie line capability above scheduled transfers is limited as shown in Table 11. All but LILCO is expected to have sufficient excess capacity in summer 1980 to transfer to the limit of the intertie. LILCO is expected to be able to supply an average of only 100 Mw. There also may be some contingency support through the submarine cable from Connecticut, but this would be limited to only 145 MW.[3/]

Table 11

Bulk Power Transmission Capability
Above Scheduled Transfers (Mw)

| FROM | SUMMER 1980 | WINTER 80-81 |
|---|---|---|
| Upper State New York | 500 | 2200 |
| Pennsylvania-New Jersey-Maryland Interconnection (PJM) | 150 | 50 |
| Long Island Lighting Co. (LILCO) | 475 | 550 |
| TOTAL | 1125 | 2800 |

OTHER EFFECTS OF INDIAN POINT SHUTDOWN

Aside from reliability consideration, the costs to the service area of the CON ED and PASNY Systems of a shutdown of the Indian Point Station include expected increases in cost of service. Indian Point provides electrical energy to the system at a cost in between hydroelectric and oil-fired generation. These types of facilities along with the Fitzpatrick nuclear plant provide almost all of the power for the CON ED and PASNY system. The least expensive method of replacing power lost as a result of the shutdown of Indian Point station appears to be PASNY's hydro facilities

3/ Letter to Hanrahan, op. cit., p. 2.

as well as the purchase of hydro-generated power from the NYPP and Hydro Quebec if available. These facilities, however, are not in the Southeastern New York area, and the transmission facilities into that area are limited according to the DOE analysis.

Assuming that oil-generated power replaces the energy lost by shutting down Indian Point station, it is possible to calculate an upper bound to the economic costs of such an action. If Indian Point station operated at its historic capacity factor of 60 percent, it would produce about 800 million kilowatt-hours per month. Approximately 1.4 million barrels of oil per month would be needed to produce the equivalent amount of oil-fired electricity. At $31 per barrel this would amount to approximately $42 million per month in fuel costs without adjustment for differences in non-fuel operating costs and uranium fuel costs saved. The major impact would be the bill for oil, much of which would likely be imported. This, of course, assumes that none of the energy shortfall could be made available from non-oil generated power.

## SECTION 3. SUMMARY OF PUBLIC COMMENTS

This section summarizes public comments that bear on interim operations. Numbers in parentheses accompanying the comment summaries refer to the comment numbers assigned in SECY-80-168, which contains a full compilation of public comments on the Director of NRR's Indian Point decision received in response to the Commission solicitation of comments. Considerations in the Director's decision that bear on interim operation are also summarized.

### SAFETY ARGUMENTS

#### Director's Decision

The Director relies on two considerations in not ordering interim shutdown for the one to two-year period required to determine and install required additional design safety features:

First, several compensating features for the high population density already exist in the design of Indian Point 2 and 3. These include:

1. Containment weld channels and weld channel pressurization system.
2. Containment penetration pressurization system.
3. Isolation valve seal water system.
4. Extra containment fan cooler capacity.
5. Post-LOCA hydrogen control capability by both recombiner and purge.
6. Third auxiliary feedwater pump, providing added assurance over a twice 100 percent capacity system.

7. HEPA and charcoal filters for containment atmosphere cleanup.
8. Confirmatory actuation signals to power operated valves which are not required to change position.
9. Extra margin in service water and component cooling water supply.
10. Redundant electrical heat tracing on borated systems.

Second, a number of extraordinary interim measures are to be implemented by the licensees -- some immediately and other within various deadlines (30, 60, 90, and 120 days, and 6 months). These measures are specified in Appendix A of the Director's Order. Included among them are matters dealing with modes of operation, shift manning levels, enhanced training of operators, and special containment tests. Some of the numerous specific requirements are:

A. Effective immediately:
   1. Limit power level to keep peak fuel clad temperature at or below $2000^{\circ}$ F under large LOCA conditions.
   2. Operate in base load mode only, without load following.
   3. Have at least two senior operators in the control room during operation or hot shutdown.

B. Within 30 days:
   1. Have vendor representative on site for engineering consultation.
   2. Assure control room habitability under accident conditions.
   3. Enhanced training and retraining provisions.
   4. Special diesel generator tests.

Comments Favoring Interim Shutdown

Commenters' safety arguments for interim shutdown relate to emergency plans, timing of long-term fixes, interim measures, short-term risks, dense population, and psychological impact.

1. Emergency plans:

USC (#85) argues that no plans exist today to evacuate the public within even 10 miles of the site. (#85 at 8 and 13.) Both UCS and Mid-Hudson Nuclear Opponents cite testimony by the County Executive of Westchester County that existing plans are not workable. (#85 at 13 and #86 at 2.) UCS argues that there has never been an assessment of the consequences of a major accident at Indian Point, implying that a basis for emergency planning is lacking, despite NRC's post-TMI commitment to improve emergency planning. (#85 at 8.) They refer to great difficulty of making effective emergency plans for the area due to physical and demographic characteristics. (#85 at 8 and 13.) They further comment that the staff has not clearly found that the licensees' emergency plans comply with the applicable Regulatory Guide (1.101) and that, moreover, Regulatory Guide 1.101 does not require evacuation plans out to 10 miles -- a requirement that will not become operative till 1981. (#85 at 20-21.) They conclude that today, in the absence of effective protection, the risk is too great to permit the plants to operate. (#85 at 34.)

Mid-Hudson Nuclear Opponents (#86) urge interim shutdown in view of the large population density and absence of adequate evacuation plans for a reasonable distance (15 to 25 miles) (#86 at 4).

New York Public Interest Research Group asserts that it would take an estimated two weeks to evacuate the Bronx, whereas only 1-1/2 days would be available in case of a disaster at Indian Point (4-1/2 days with a "core catcher"). (#67 at 4.)

2. Timing of Long-term fixes:

UCS contends that there is no licensee commitment and no requirement established by the Director's order for licensee implementation of the protective-action time-buying provisions (filtered vented containment and core ladle): only a review of possible modifications is required. (#85 at 10-11.) They see evidence of a dispute between the staff and the licensee concerning possible imposition of Class 9 accident related requirements. (#85 at 11-12.) UCS argues that the mere possibility of future protection offers no protection today. (#85 at 11.)

Mid-Hudson Nuclear Opponents refer to post-accident monitoring, aging, and asymmetric LOCA loads as serious unresolved safety issues. They consider it insufficient for control of present risks to merely say that these issues are being examined -- with an unspecified schedule. (#86 at 3.)

3. Interim measures:

UCS comments to the effect that (a) the special safety measures already present at Indian Point 2 and 3 are of little real value and (b) that the special interim measures yet to be implemented (which, in any case, they regard as inadequate for the long term) should not be counted now, because implementation is largely deferred. (#85 at 15-21, 27-34, and passim.) With resepct to the special safety features identified in the Director's Decision as already present, UCS comments specifically on each. (#85 at 15-20.) They impugn each, usually on one or both of two grounds: (a) that they do little or no good -- or are even counterproductive -- and (b) that they merely reflect implementation of present requirements or correction of inadequacies that could not be tolerated anywhere. Thus, for weld channel and penetration pressurization and the isolation-valve seal-water system, they argue that these measures merely compensate for bad welds or leaky valves. (#85 at 16.) For containment atmosphere cleanup, they contend that NRC regulations (Design Criterion 41) require such provision for all plants. (#85 at 18.) Purging for hydrogen control is criticized as counterproductive. ("[T]he staff proposes to seal the containment normally but to vent it during an accident with no capability to filter. . . ") (#85 at 17.) For further interim measures, they argue that they are neither extraordinary nor sufficient, and not yet in place. (#85 at 33 and passim.) The interim measures leave the safety issues raised by UCS unresolved. (#85 at 33.) They stress fire protection, post-accident monitoring, equipment aging, and asymmetric LOCA loads. (#85 at 26-31.)

4. Short-term risks;

UCS asserts: "Little by little, the short-term grows into the long-term." (#85 at 32.)

Dean Corren, of Greater New York Council on Energy, expresses the view that distinction between short-term and long-term risks is "an improper and misleading use of the notion of statistical risk assessment." (#80 at 1.) He contends that any safety improvements that are deemed necessary at all are necessary forthwith. Brooklyn SHAD offers a similar argument. (#63)

Westchester People's Action Coalition views the risks pending completion of fixes as excessive even "for one more day." (#19 at 3.)

5. High population density:

UCS stresses the high population density as an obstacle to effective emergency action. They cite Robert Ryan (NRC's Director of State Programs) as characterizing Indian Point as an "insane" site, "a nightmare from the point of view of emergency preparedness," with difficulties exacerbated by severe traffic problems. (#85 at 8-9.)

Westchester People's Action Coalition argues that dense population inevitably makes Indian Point 10 times more dangerous than the average plant, since plant safety improvements practical at Indian Point should be made nationwide. (#19 at 5.)

Mid-Hudson Nuclear Opponents ask for suspension of the licenses pending the Commission's decision, in view of the large population density and inadequate emergency plans (#86 at 4.)

6. Psychological impact:
Westchester People's Coalition calls for consideration of human responses to minor mishaps, rumors of accidents, or threat of accident. They write of human costs in anxiety and potential panic. (#19 at 3.)

Comments Opposed to Interim Shutdown

Arguments against interim shutdown relate to risk estimates, evacuation, and population density.

1. Risk estimates:
Power Authority of the State of New York (PASNY) (#66) maintains that the staff's risk estimates for Indian Point overstate the risk. (#66 at 17.) They argue that special plant features already existing (identified in the Director's Decision) distinguish Indian Point from average PWR's and lower the Indian Point risks substantially below those derived from WASH-1400. (#66 passim.) They present plots of Indian Point rsks with and without adjustments for plant-specific features. (#66 at Appendix 2.) The plant specific adjustments include elimination of some WASH-1400 sequences that PASNY contends are not significant contributors to core melt probability. These include loss of auxiliary feedwater after shutdown and reactor transient followed by failure of reactor trip. (#66 at 16.)

PASNY also asserts that in-vessel steam explosions now appear less likely than estimated in WASH-1400, so that containment failure due to missiles from such an explosion is also less likely. (#66 at 17.)

2. Evacuation:

Scientists and Engineers for Secure Energy (SE 2) (#62) describes the emergency evacuation of Mississauga, Canada, a city of 240,000, in November 1979, in connection with derailment of a train that included 11 propane tanks. SE 2 cites that experience as showing that massive evacuations are feasible. (#62 at 3.)

Corren (#80) encloses a statement of PASNY before the Committee on Environmental Protection of the New York City Council, dated December 14, 1979, in which PASNY argues evacuability to 10 miles and also argues that a likelihood of evacuation being required for New York City residents under any circumstances is not realistically foreseeable. (Page 6 of PASNY enclosure to #80.)

3. Population density:

SE 2 argues that population density around Indian Point is not unusually high by world reactor siting standards. They cite Canadian, French, British, and Japanese practices of siting reactors in densely populated areas. (#62 at 2-3.)

Differences Between Units 2 and 3

UCS contentions that Indian Point 2 lacks some important safety features of Unit 3 suggest that their arguments for interim shutdown would apply to Unit 2 *a fortiori*. (#85 at 21-23.)

IMPACT ARGUMENTS

The Director's Decision does not reflect consideration of social or economic impacts of interim shutdown.

Comments on this general subject deal with need for power, cost of power, and effect on oil imports.

1. Need for power:

   Westchester People's Action Coalition (#19) contends that Indian Point's power is not needed. They assert that there is 50 percent excess capacity in New York; 30 percent without nuclear facilities. They further assert that there have been no capacity-related blackouts, even though Indian Point Unit 2 has been off-line for four months since last June, and Unit 3 for five. (#19 at 6.) They enclose a New York Times article from which they draw their assertions.

   Dean Corren, of Greater New York Council on Energy (#80) contends that there is no need for the Indian Point capacity. (#80 at 2.) He presents capacity figures that assertedly show that there is a 3,026-MW unutilized excess capacity (on top of an 18 percent reserve over peak demand), as compared with a total Indian Point capacity of 1,838 MW. (Page 3 of first enclosure to #80.) Corren states that Con Ed still claims a 1.8 percent annual peak demand growth, although that growth has slowed to 0.1 percent. He also states that 69.3 percent of the system was idle in 1978, on the average. (Page 4 of first enclosure to #80.) He concludes that ability to meet demand would not be compromised by closing Indian Point 2 and 3. (Page 5 of first enclosure to #80.)

Corren (#80) also encloses statements by UCS and PASNY. The UCS statement (at 1) argues that the Indian Point plants are often out of service, yet New York City does not go dark. The PASNY statement (at 7 and passim) argues need for power on economic (not absolute or reliability) grounds.

2. Cost of power:

Stanley Fink, Speaker of the New York State Assembly, comments that shutdown would cause economic hardship in the Metropolitan New York area. He considers it the responsibility of NRC to work with FERC and others to secure replacement non-oil power at comparable cost, if NRC orders Indian Point temporarily shut down. (#1)

The New York State Building and Construction Trades Council sees a threat to "local economic livelihood" in any Indian Point shutdown. (#7)

PASNY contends that shutdown would be an economic calamity for New York City, costing PASNY's and Con Ed's ratepayers about $700 million in 1980 alone. Increases would escalate with imported oil price increases. (#66 at 20-21.) According to PASNY, 45 percent of the power cost increase would fall on public customers -- New York City and its Metropolitan Transportation Authority (MTA). These entities are already financially hard pressed. MTA's projected $200 million deficit for 1980 would increase by $100 million for increased cost of electricity for subway and commuter rail lines. (#66 at 21.)

Corren estimates that shutdown of Indian Point would cost the average residential ratepayer between $2 and $4 per month. (Pages 11-12 and passim, first enclosure to #80; calculations at Appendix A to that enclosure.) Corren also encloses a concurring analysis by UCS. In addition, he encloses a PASNY statement (with which he takes issue). That PASNY statement is generally consistent with PASNY's comment on the Director's decision. (#66)

3. Oil imports:

Fink states that shutdown of Indian Point would exacerbate the region's dependency on imported oil and calls on NRC to work with FERC and others to secure non-oil replacement power in event of Indian Point shutdown. (#1)

PASNY comments that the region depends on oil and nuclear sources for electric power generation. (#66 at 19.) Indian Point shutdown would require 20 million barrels of imported oil per year for replacement power. (#66 at 20.)

Corren presents a "worst-case" replacement-power-cost estimate of $5.21 per month for an average residential customer, based on oil at $30 per barrel. However, he maintains that replacement fuel is likely to be a more economical mixture of oil, gas, and coal. (Pages 7 and 8 of first enclosure to #80 and Appendix A to that enclosure.)

Corren (#80) encloses a statement by UCS, which contains an estimate that replacement fuel would cause a 0.7 percent increase in total U.S. imported oil consumption. Corren's (#80) last enclosure includes a remark by Commissioner Bradford that nuclear electric generation frees up "residual oil, of which there is something of a surplus anyway."

## References


1. U.S. Nuclear Regulatory Commission, "Comments on Indian Point," USNRC SECY-80-168, March 28, 1980. Available in NRC PDR for inspection and copying for a fee.

2. U.S. Nuclear Regulatory Commission, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites," USNRC Report NUREG-0348, October 1979.*

3. U.S. Nuclear Regulatory Commission, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Executive Summary, WASH-1400 (NUREG-75/014), October 1975.**

4. U.S. Nuclear Regulatory Commission, "Overview of the Reactor Safety Study Consequence Model," USNRC Report, NUREG-0340, September 1977.*

5. U.S. Nuclear Regulatory Commission, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Waster Nuclear Power Plants," USNRC Report NUREG-0396, November 1978.*

6. U.S. Nuclear Regulatory Commission, "Lewis Report - Risk Assessment Review Group," USNRC Report NUREG/CR-0400, September 1978.*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*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.

**Available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

# APPENDIX A

## SAMPLE GENERATION OF A COMPLEMENTARY CUMULATIVE DISTRIBUTION FUNCTION - CCDF

The CCDF is used to present the risk of reactor accidents in the form of a plot of probability vs consequences. The average reader is unaccustomed to studying risk in this form of presentation. To facilitate understanding of the CCDF, consider generating a CCDF for the risk of death from air crash from high altitude using the attached figure.

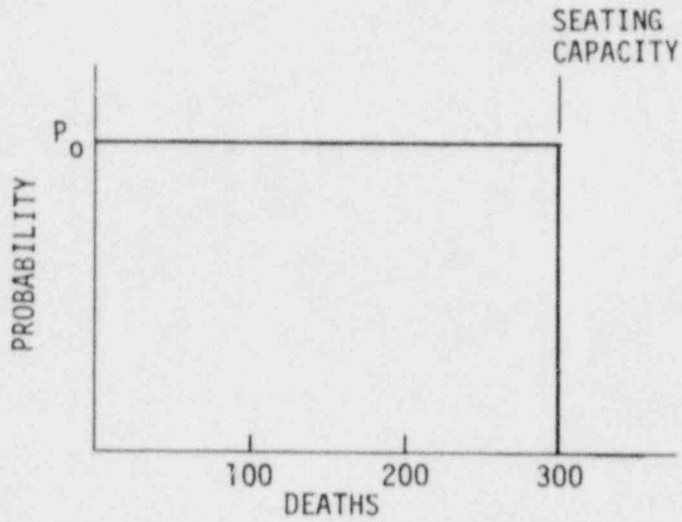
If an airplane crashes from a high altitude, it is virtually certain that all on board will perish. Thus, Figure A-1 is a reasonable first approximation of a CCDF for such a crash; it shows a probability, $P_o$, that 300 deaths, the seating capacity of the aircraft, will occur. $P_o$ is the probability that the plane will crash; 300 is the limit of those on board who will die in a crash. For this simple CCDF curve the expected risk is $P_o$, say 0.33 crashes per year, times 300 deaths per crash or 100 deaths per year.

The CCDF can be corrected first to show that the falling aircraft might strike and kill people on the ground. Figure A-2 shows a tail on the CCDF curve reflecting that if the plane crashes, it will most likely not kill many people on the ground. At lower and lower probability, there is the chance of killing crowds in buildings or gatherings so the curve tails off toward some higher number of deaths. Presumably there is a limit to the ground deaths that can be caused by the crash of a 300 passenger aircraft, perhaps 10,000 or 20,000 if it crashed into a crowded sports stadium. At that limit, the curve would no longer tail off to the right but become a vertical line showing a physical limit analogous to the seating capacity limit.

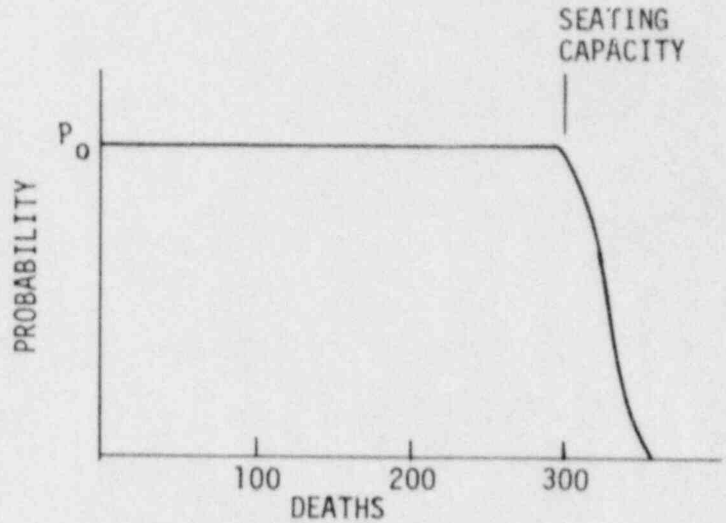
A second stage of refinement in this CCDF can be obtained if the airline gives us figures on the actual passenger loads the aircraft usually carries. If the data are limited, they might simply be reduced to the approximation that on 1/3 of the trips the plane is 1/3 full, on another 1/3 of the trips it is 2/3 full, and on another 1/3 of the trips it is completely full. The CCDF can now be refined as shown in Figure A-3. One hundred deaths occur at probability $P_0$, the probability of crash, because the plane is always at least 1/3 full. At 0.67 $P_0$ the curve shows 200 deaths because the plane is at least 2/3 full 2/3 of the time. And the curve shows 300 deaths at 0.33 $P_0$ because on one third of its flights all seats are filled. We can reflect the probability of ground deaths by putting soft tails on the sharp steps of the curve.

As more accurate flight data are accumulated, the steps in Figure A-3 can be refined into a more accurate curve as shown in Figure A-4. This last curve would represent the most accurate distribution of the likelihood of death from high altitude air crash.

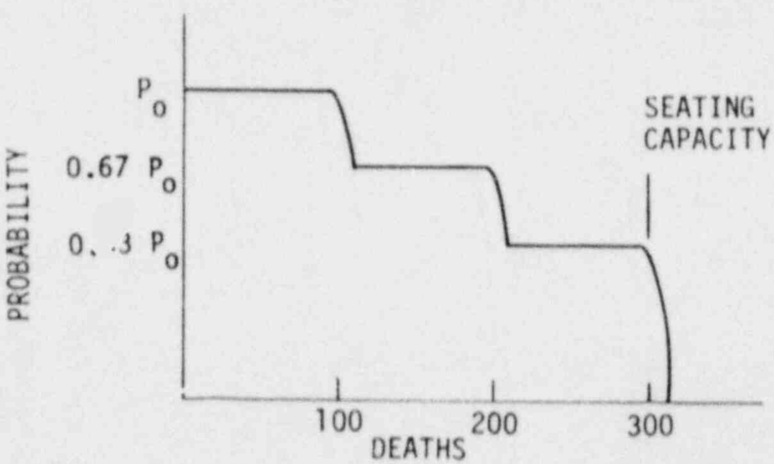
# CCDF FOR AIR CRASH FROM HIGH ALTITUDE



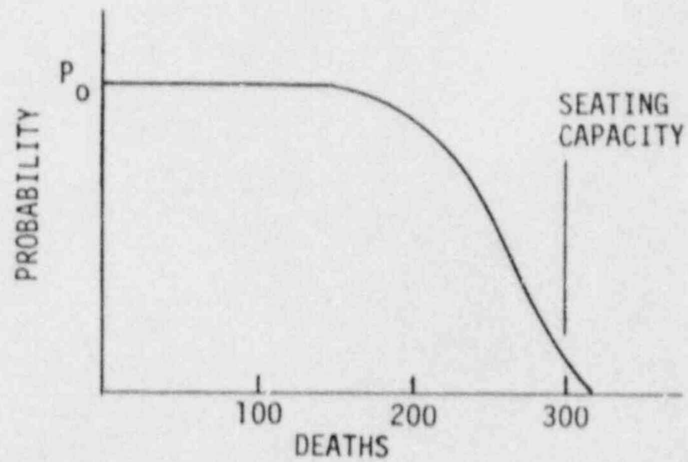

A-1




A-2




A-3




A-4

# APPENDIX B

## REBASELINING OF THE RSS RESULTS

The results of the Reactor Safety Study (RSS) were updated for purposes of this comparative study. The update was done largely to incorporate results of research and development conducted after the October 1975 publication of the RSS and to provide a baseline against which the risk associated with various LWRs could be consistently compared.

Primarily, the rebaselined RSS results reflect use of advanced modeling of the processes involved in meltdown accidents, i.e., the MARCH computer code modeling for transient and LOCA initiated sequences and the CORRAL code used for calculating magnitudes of release accompanying various accident sequences. These codes[1] have led to a capability to predict the transient and small LOCA initiated sequences that is considerably advanced beyond what existed at the time the Reactor Safety Study was completed. The advanced accident process models (MARCH and CORRAL) produced some changes in our estimates of the release magnitudes from various accident sequences in WASH-1400. These changes primarily involved release magnitudes for the iodine, cesium and tellurium families of isotopes. In general, a decrease in the iodines was predicted for many of the dominant accident sequences while some increases in the release magnitudes for the cesium and tellurium isotopes were predicted.

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[1]It should be noted that the MARCH Code was used on a number of scenarios in connection with the TMI-2 recovery efforts and for Post-TMI-2 investigations, e.g., Rogovin) to explore possible alternative scenarios that TMI-2 could have experienced.

Figures B1 and B2 show a comparison of the original RSS and the rebaselined PWR and BWR designs for the individual risk versus distance of early fatalities and latent cancer fatalities, respectively. These figures show the expected values conditioned upon a core melt accident of about one chance in ten thousand reactor years ($1x10^{4}$). This particular conditioned value reflects an average of the core melt probabilities estimated from a number of LWR designs.

Entailed in this rebaselining effort was the evaluation of individual dominant accident sequences as we understand them to evolve rather than the technique of grouping large numbers of accident sequences into encompassing, but synthetic, release categories as was done in WASH-1400. The rebaselining of the RSS also eliminated the "smoothing technique" that was criticized in the report by the Risk Assessment Review Group (sometimes known as the Lewis Report; NUREG/CR-0400).

For rebaselining of the RSS BWR design, the sequence $TC\gamma'$ was explicitly included into the rebaselining results. The accident processes associated with the TC sequence had been erroneously calculated in WASH-1400. For rebaselining of the RSS PWR design, the release magnitudes for the Event V and TMLB' sequences were explicitly calculated and used in the consequence modeling rather than being lumped together into Release Category #2 as was done in WASH-1400.

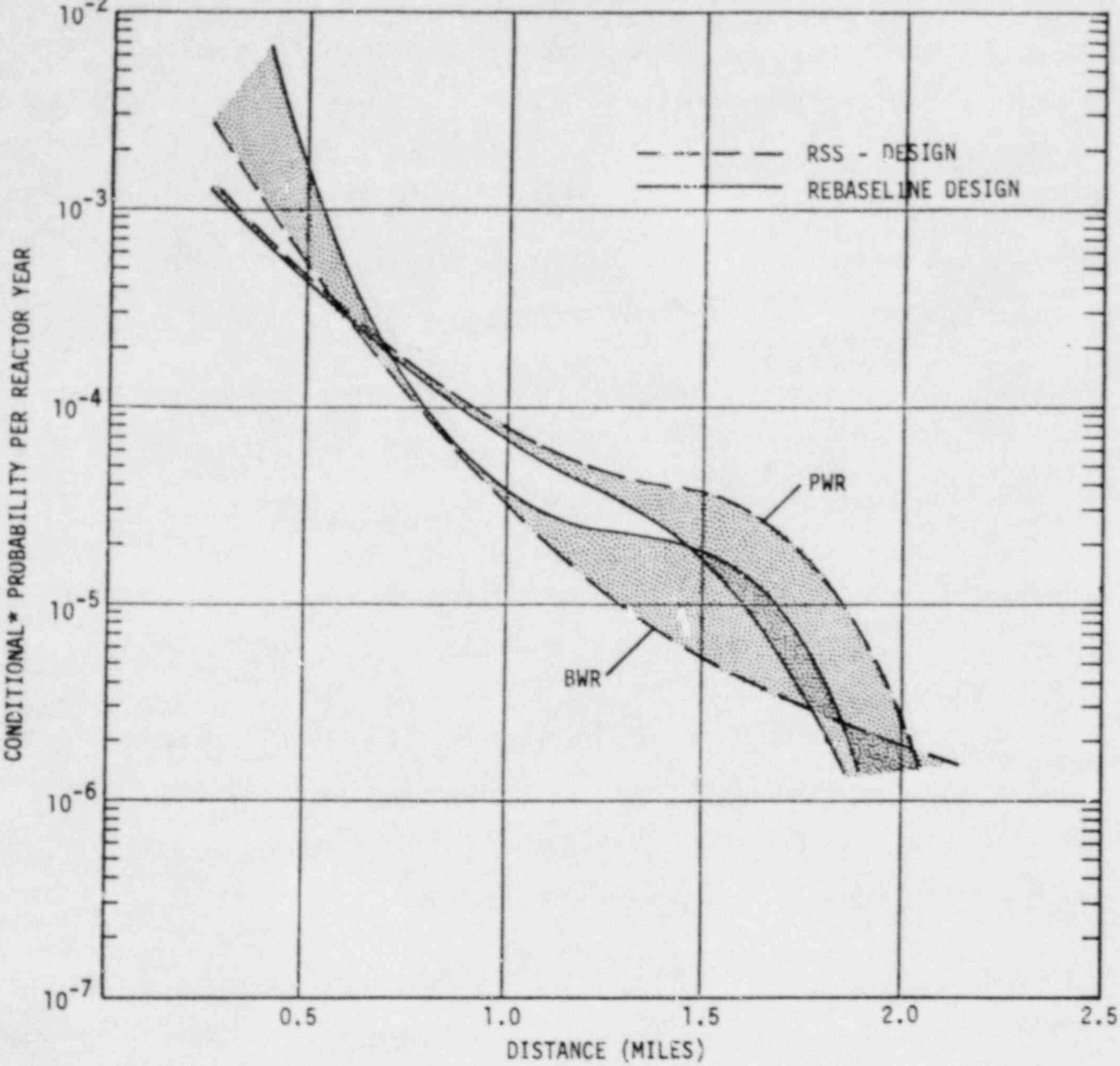
In both of the RSS designs (PWR and BWR) the likelihood of an accident sequence leading to the occurrence of a steam explosion ($\alpha$) in the reactor vessel was decreased. This was done to reflect both experimental and calculational indications that such explosions are unlikely to occur in those sequences involving small size LOCAs and transients because of the high pressures and temperatures expected to exist within the reactor coolant system during these scenarios. Furthermore, if such an explosion were to occur, there are indications that it would be unlikely to produce as much energy and the massive missile-caused breach of containment as was postulated in WASH-1400.

As can be seen from Figures B1 and B2, the net (or overall) change in consequences predicted from the rebaselined RSS results are quite small. In general, the rebaselined results led to slightly increased health impacts being predicted for the RSS BWR design. This is believed to be largely attributable to the inclusions of TC$\gamma'$.

The rebaselined RSS-PWR led to a small decrease in an individual risk of early fatalities and latent cancer fatalities below the original RSS PWR. This is believed to be largely attributable to the decreased likelihood of sequences involving vessel steam explosions ($\alpha$).

In summary, the rebaselining of the RSS results led to small overall differences from the predictions in WASH-1400. It should be recognized that these small differences due to the rebaselining efforts are likely to be far out-weighed by the uncertainties associated with such analyses.

FIGURE B1 - RISK OF EARLY FATALITY TO AN INDIVIDUAL VERSUS DISTANCE GIVEN A CORE MELT*




ASSUMPTIONS: *CORE MELT PROBABILITY ASSUMED TO BE $10^{-4}$/REACTOR YEAR

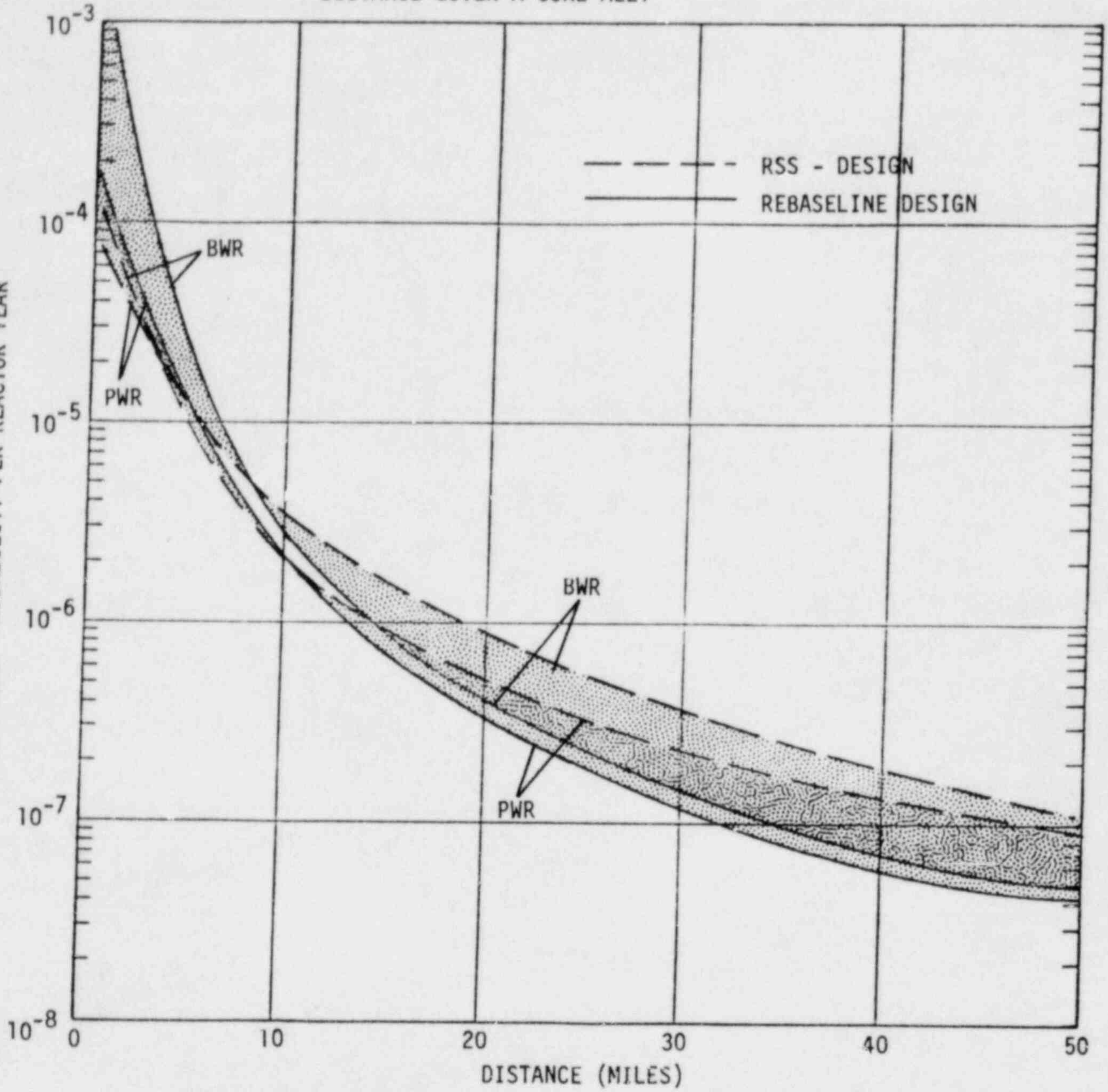
RSS-DESIGN
1. ALL RSS CORE MELT ACCIDENT RELEASE CATEGORIES
2. ALL RSS ASSUMPTIONS (E.G., SMOOTHING)

REBASELINE DESIGN
1. SMOOTHING ELIMINATED
2. EXPLICIT ACCIDENT SEQUENCES
3. NEGLIGIBLE PROBABILITY OF VESSEL STEAM EXPLOSION

EXPECTED CONSEQUENCES FROM 91 WEATHER SEQUENCES WITH 3200 MWT POWER LEVEL

ENTIRE CLOUD EXPOSURE + 24 HOUR GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY

FIGURE B2 - RISK OF LATENT CANCER FATALITY TO AN INDIVIDUAL VERSUS DISTANCE GIVEN A CORE MELT*




ASSUMPTIONS: *CORE MELT PROBABILITY ASSUMED TO BE $10^{-4}$/REACTOR YEAR

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16. ABSTRACT (200 words or less) On May 30, 1980, the Commission issued an order establishing a four-pronged approach for resolving the issues raised by the Union of Concerned Scientists' petition regarding the Indian Point nuclear facilities. Among other things a Task Force on Interim Operation was established to address the question of whether Indian Point Units 2 and 3 should or should not be allowed to operate during the pendency of a planned adjudication.

Specifically, the Task Force report deals with two major issues. The first issue relates to accident risk as a function of population density and distribution around the plant. New York City is less than 50 miles to the south of the Indian Point site. The Task Force compared Indian Point risks, e.g., health impacts, property damage with those of other reactor sites and designs, distinguishing between the effects of population densities and of design and other factors. Secondly, the Task Force examined the economic, social and other "non-safety" effects of shutting down or reducing the power levels of either or both reactors. In particular, the Task Force compared projected peak demands for energy with projected available capacity to determine if reducing power levels at Indian Point would affect system reliability in the summer of 1980.

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