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INFORMATION REPORT

FOR: The Commissioners
FROM: *Edward J. Hanrahan*
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Office of Policy Evaluation
Leonard Bickwit, Jr.
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General Counsel
SUBJECT: REPORT OF THE TASK FORCE ON INTERIM OPERATION OF INDIAN POINT
(DOCKET NOS. 50-247 AND 50-286)

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SECY NOTE: Copies of this paper were advanced to each Commissioner on June 12, 1980.
One copy of NUREG-0340, which is referenced in the report has been provided
to each Commissioner for information.

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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 $22\frac{1}{2}$ 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	15300	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	50436
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	705112
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	1202027
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	PO.	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	1120000	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	KOPE CREEK	4773288
31	POINT BEACH	564251	68	CHEROKEE	1308327	105	SHOREHAM	4940868
32	KEWAUNEE	574631	69	NCCUIRE	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

	<u>Population in Highest 22 1/2⁰ Sector(s)</u>
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 water 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.

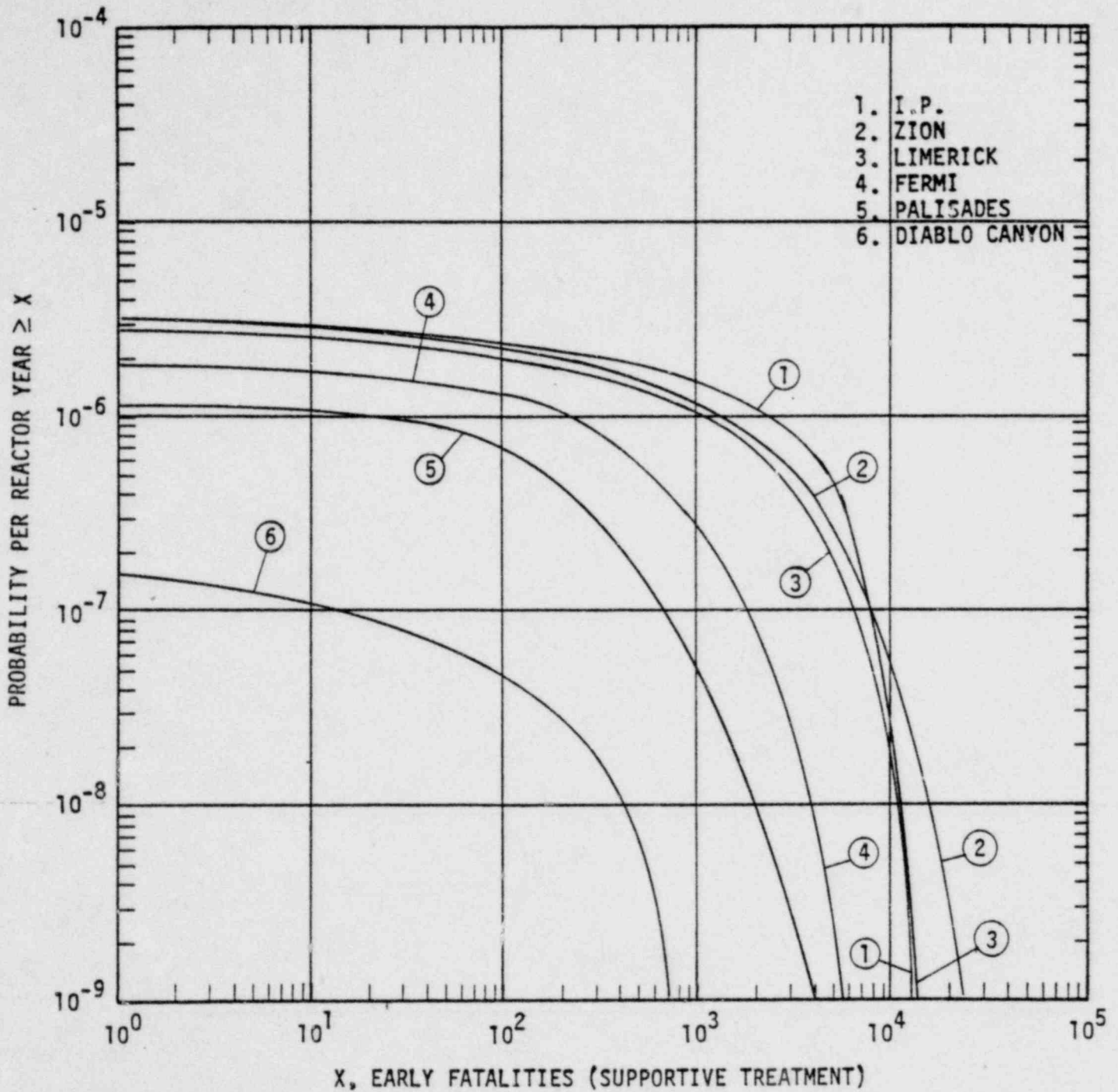
The staff has performed a set of these benchmark calculations using the Surry benchmark reactor with its power increased to 3025 MWT, the rating of Indian Point 3. In general, the risk a reactor poses is proportional to its power level. Six sites were analyzed for this comparison. Four, Indian Point, 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, that is, 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).

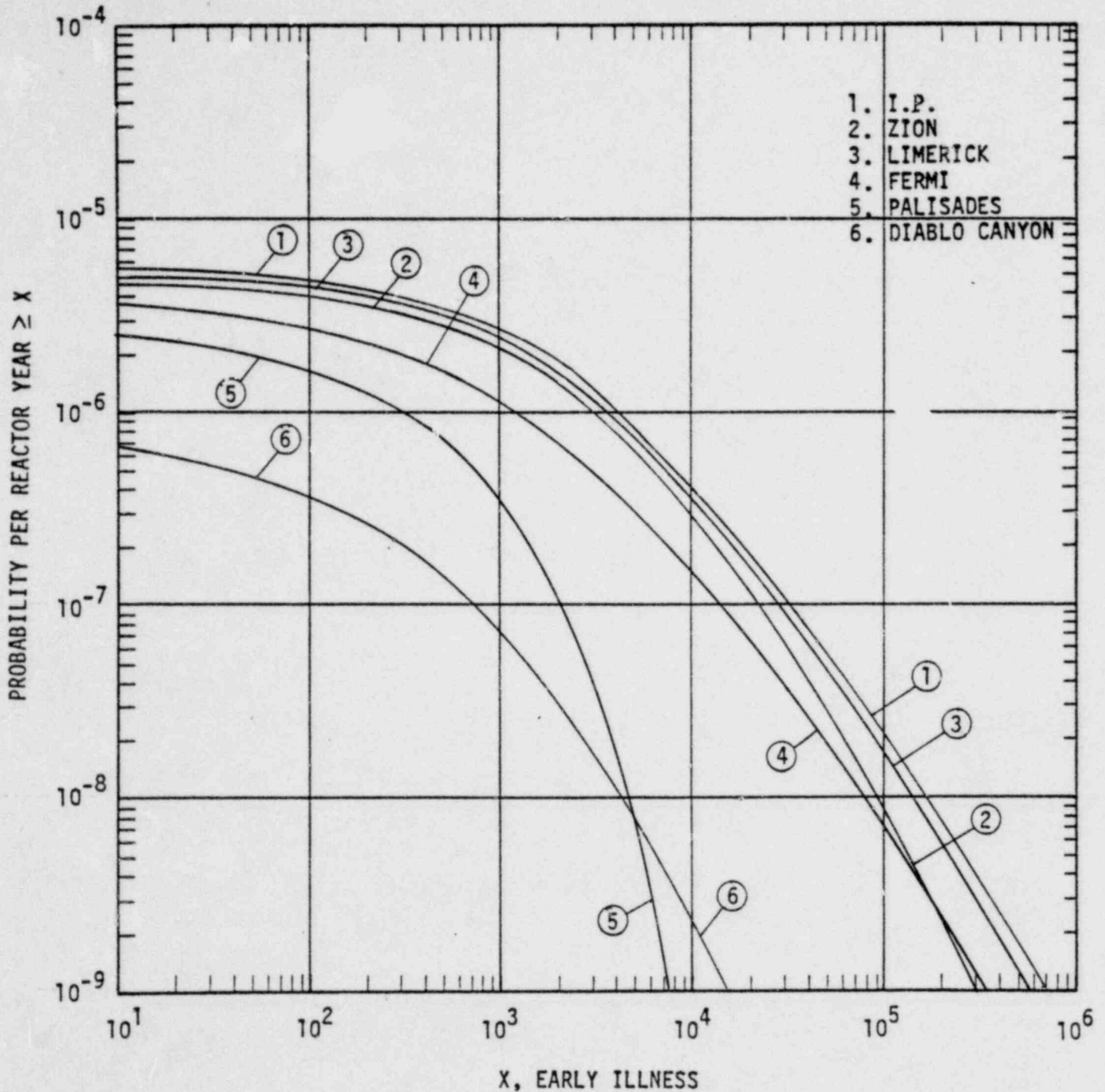
3) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING

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: 1) SURRY DESIGN.

2) I.P. UNIT 3 POWER LEVEL (3025 MWT).

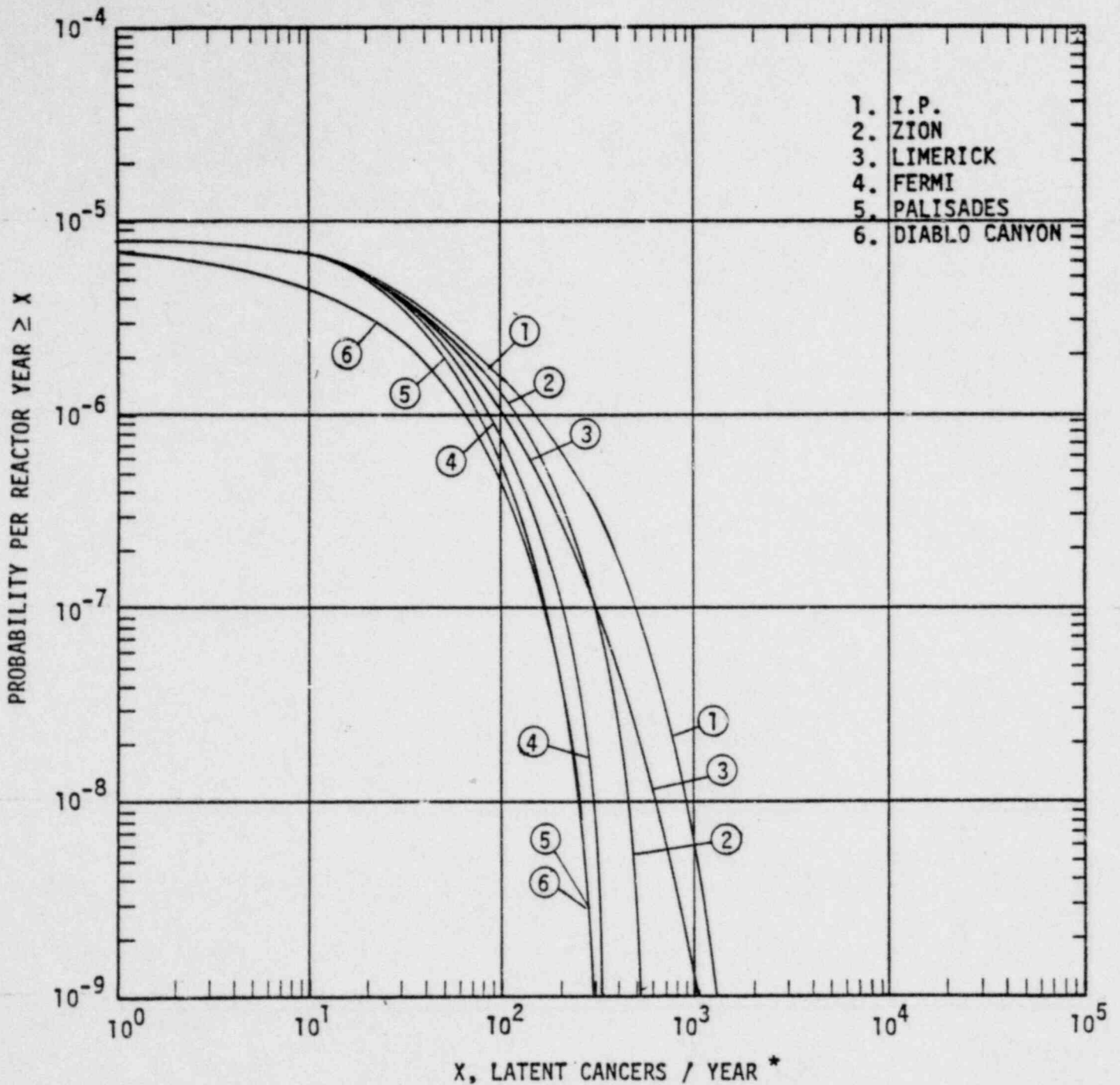
3) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING

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 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).

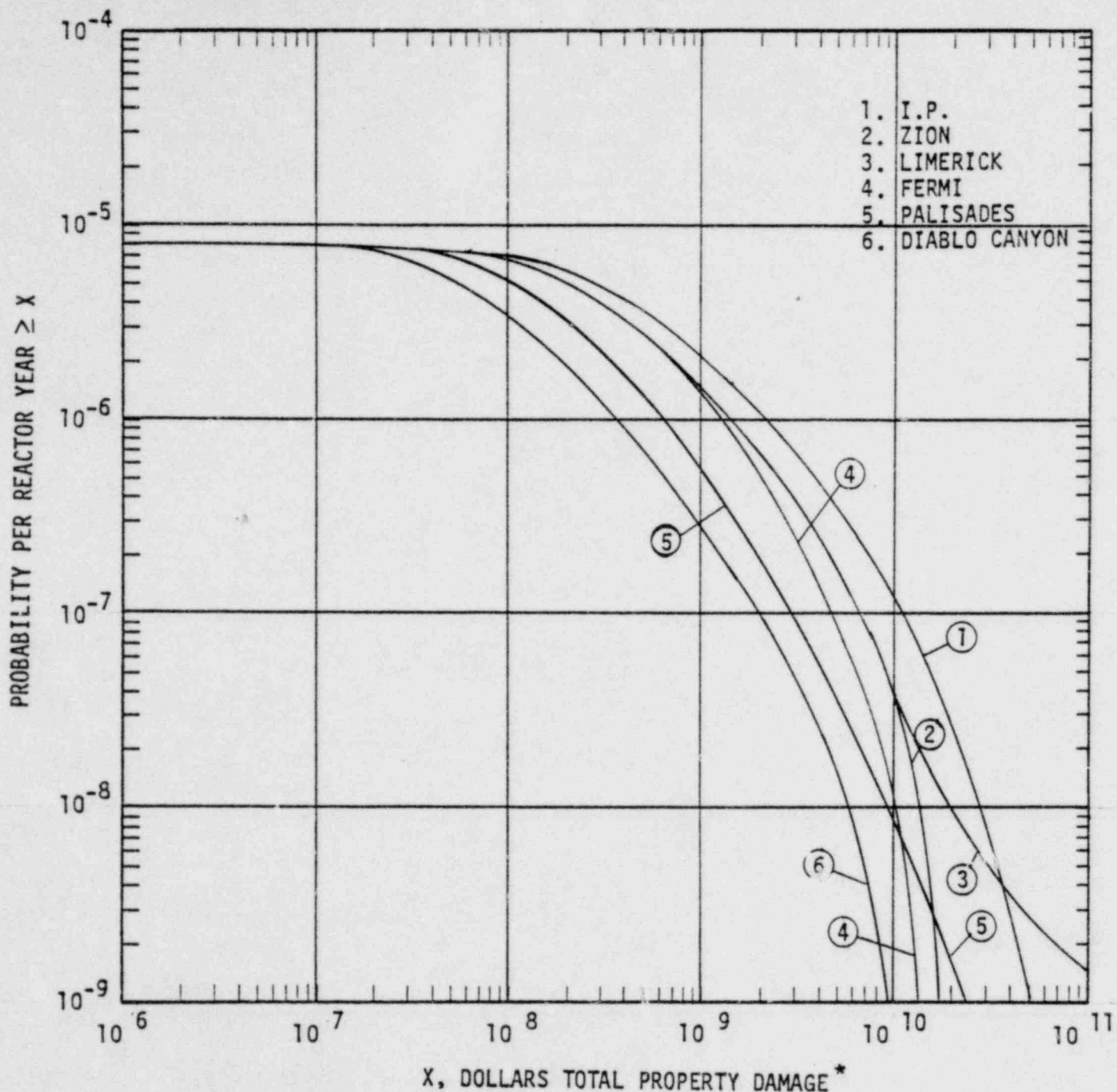
3) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING

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 $10^{-9}/\text{yr}$).

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 (5×10^{-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 3.2×10^{-6} /yr. This means that only $3.2 \times 10^{-6} \div 5 \times 10^{-5} = .064$ 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 of Consequence Oc- currence per yr	Early Fatalities	Early Injuries	Latent Cancer/Yr*	Property Damage \$**
Diablo Canyon		1.6×10^{-5}	2.5×10^{-4}	1.8×10^{-4}	1290
Palisades		2.9×10^{-4}	1.2×10^{-3}	2.7×10^{-4}	2670
Fermi		9.2×10^{-4}	6.3×10^{-3}	3.6×10^{-4}	4780
Limerick		3.5×10^{-3}	1.1×10^{-2}	4.7×10^{-4}	6980
Zion		4.7×10^{-3}	1.2×10^{-2}	4.3×10^{-4}	6030
Indian Point		6.1×10^{-3}	1.5×10^{-2}	5.4×10^{-4}	9550

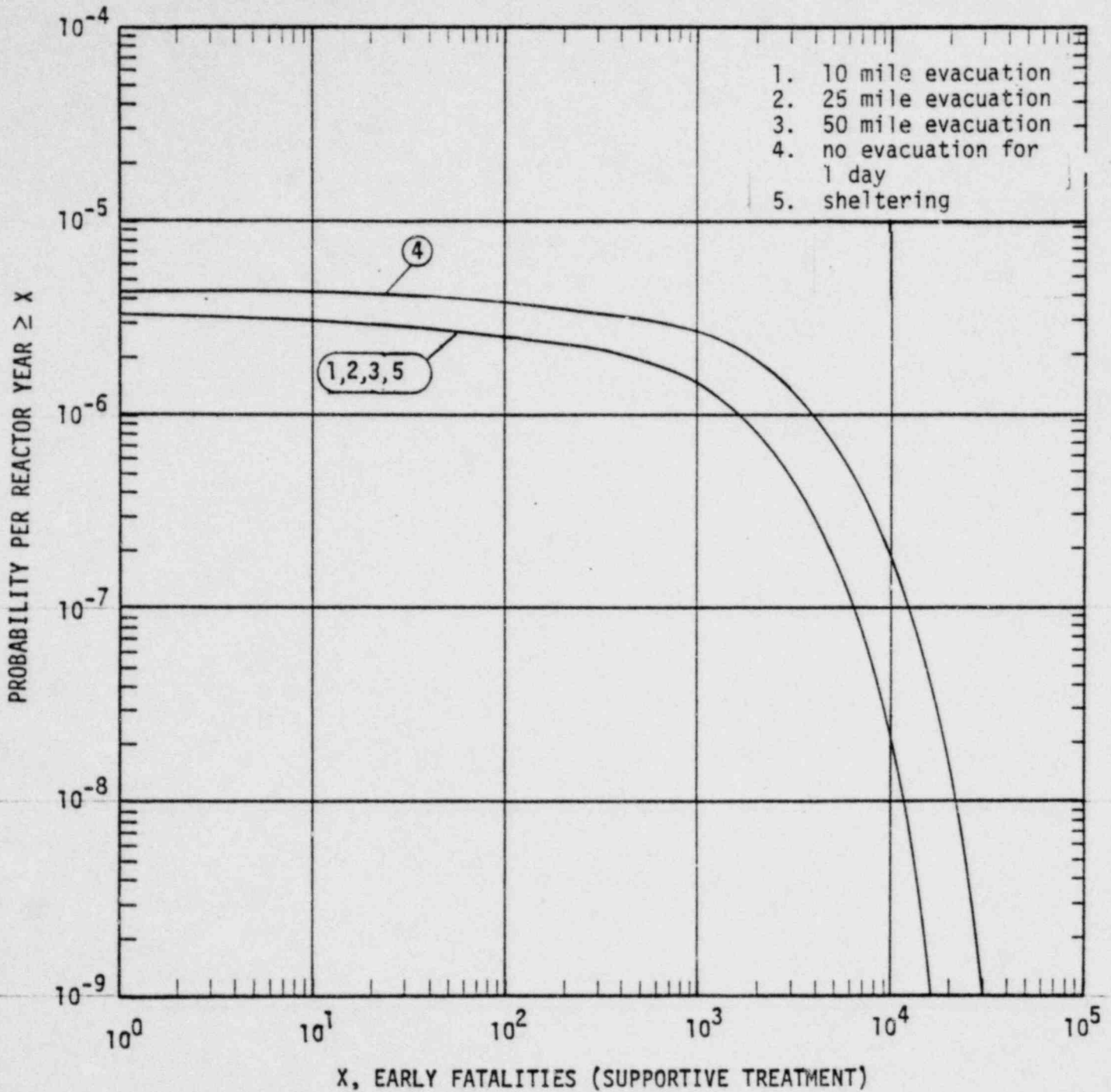
*Total Latent Cancers Would Be 30 Times Higher

**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.

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

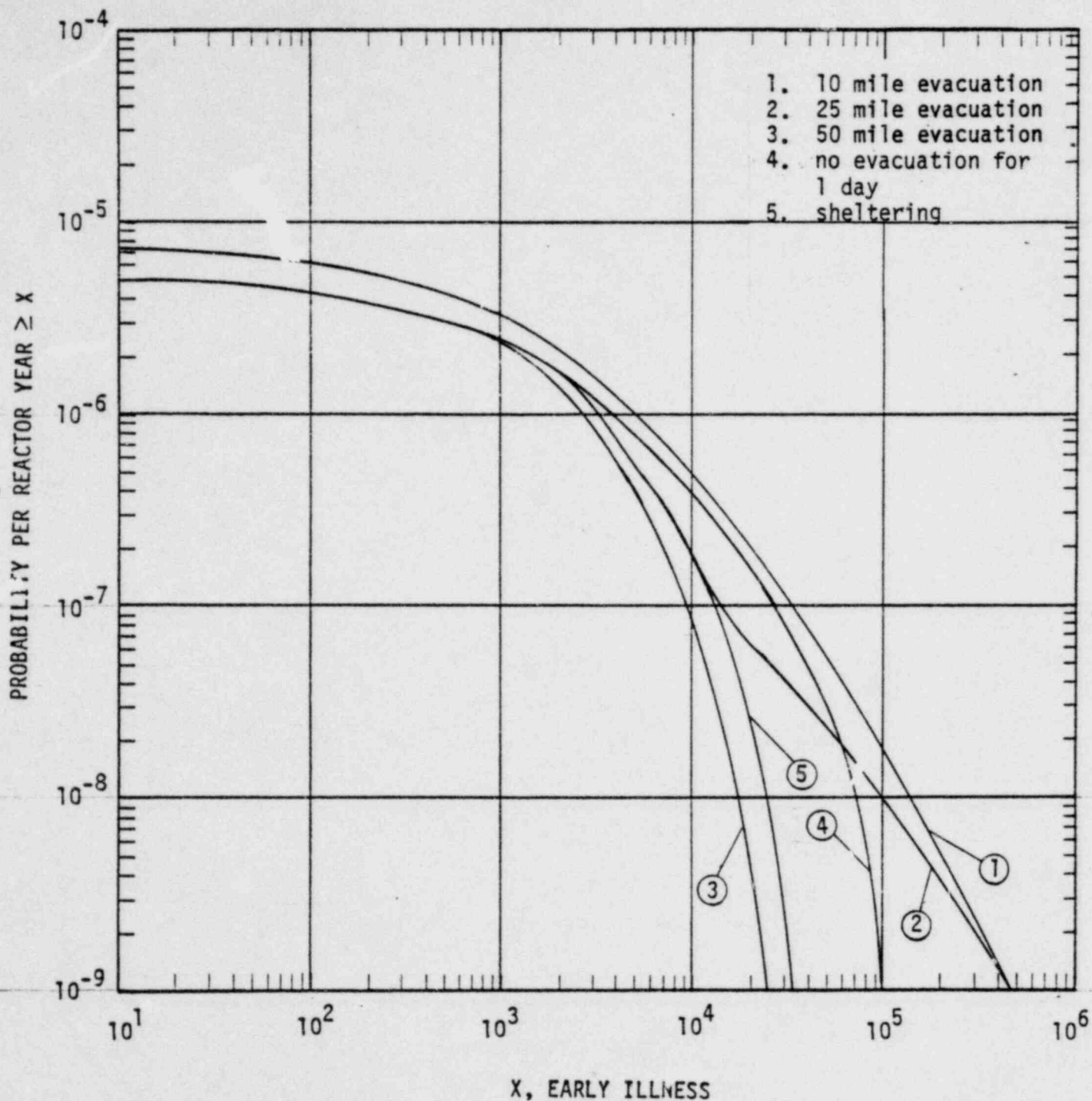
4) INDIAN POINT SITE (POPULATION AND METEOROLOGY)

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 3×10^{-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

<u>Accident Scenario</u>	<u>Sequence Code From WASH-1400</u>	<u>Offsite Consequences Expected</u>
LOCA and failure of ECCS in injection mode	AD S ₁ D S ₂ D	Low to modest ↓
LOCA and failure of ECCS in recirculation mode	AH S ₁ H S ₂ H	
Transient and loss of feedwater or serious failure and no feed and bleed on primary side (X)	TMLX TMKX	
LOCA and loss of containment heat removal with subsequent interactions with ECCS	AG S ₁ G S ₂ G	Intermediate
LOCA and failure of ECCS and containment ESFs in recircu- lation phase due to common cause	AHF S ₁ HF S ₂ HF	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 1×10^5 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.

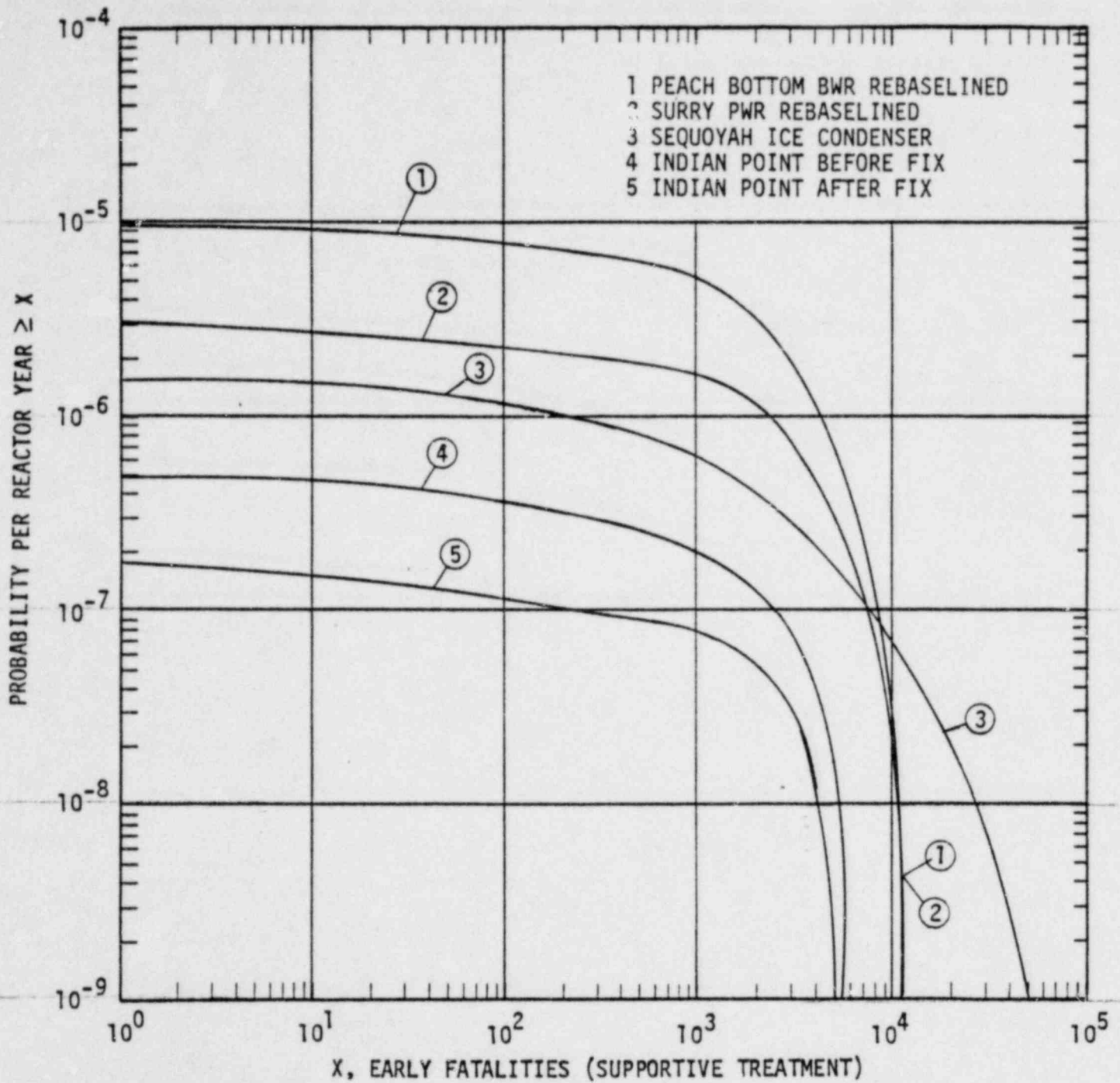
Figures 8, 9 and 10 display the same comparisons for the other risk indicators, early injuries, latent fatalities and property damage.

TABLE 7 - ESTIMATED PROBABILITY OF SEVERE CORE DAMAGE

<u>REACTOR NAME</u>	<u>TYPE</u>	<u>PROBABILITY* OF SEVERE CORE DAMAGE PER REACTOR-YEAR</u>
SURRY	3-loop PWR	6×10^{-5}
PEACH BOTTOM	BWR (Mark I)	3×10^{-5}
SEQUOYAH	4-loop PWR (Ice Condenser)	4×10^{-5}
OCONEE	2-loop PWR	2×10^{-4}
CALVERT CLIFFS	2-loop PWR	2×10^{-4}
CRYSTAL RIVER-3	2-loop PWR	3×10^{-4}
BIBLIS	4-loop PWR	4×10^{-5}
INDIAN POINT	4-loop PWR	1×10^{-5}

* Reflects median values

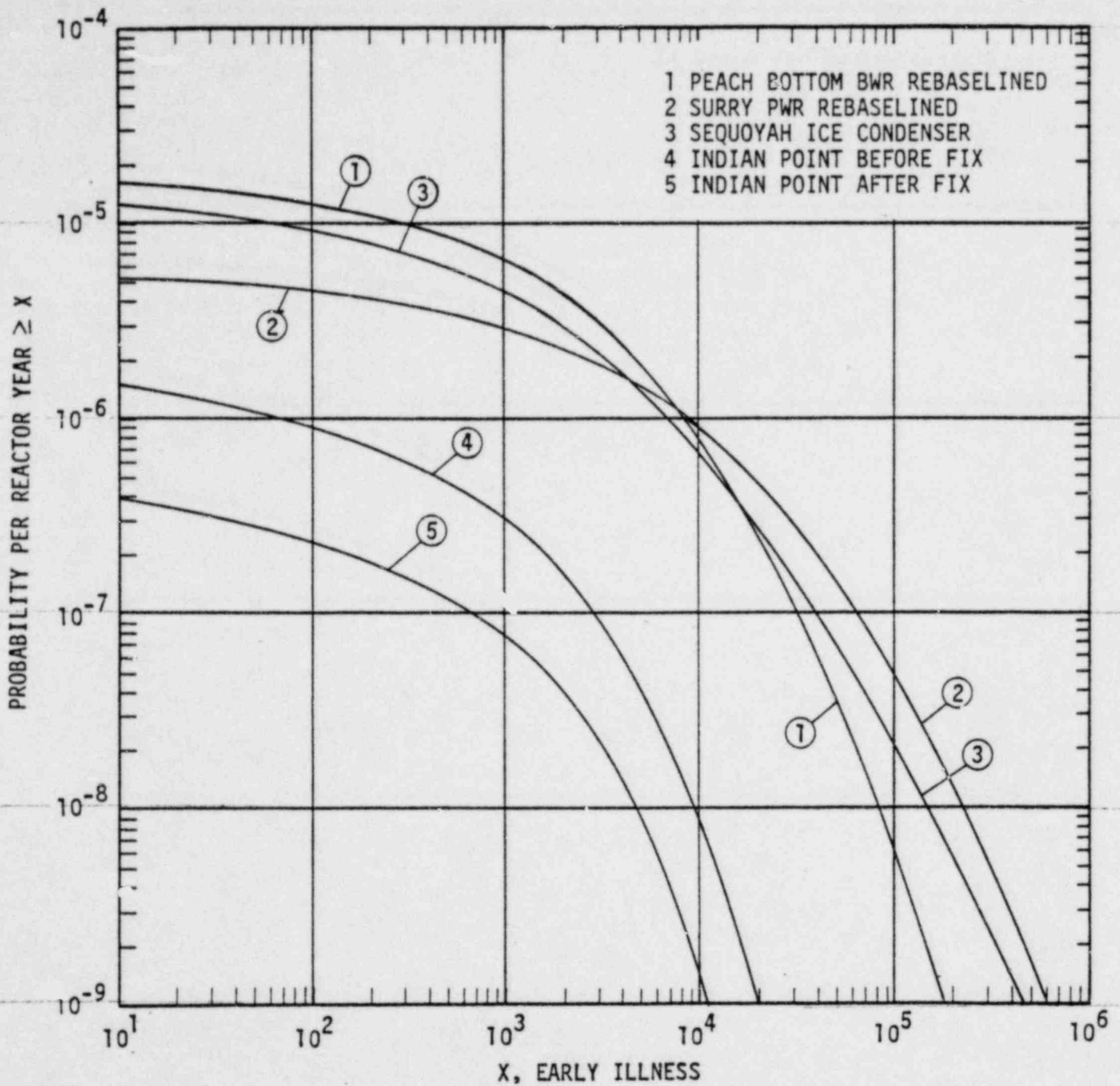
FIGURE 7 - EARLY FATALITY RISK FOR DIFFERENT DESIGNS



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

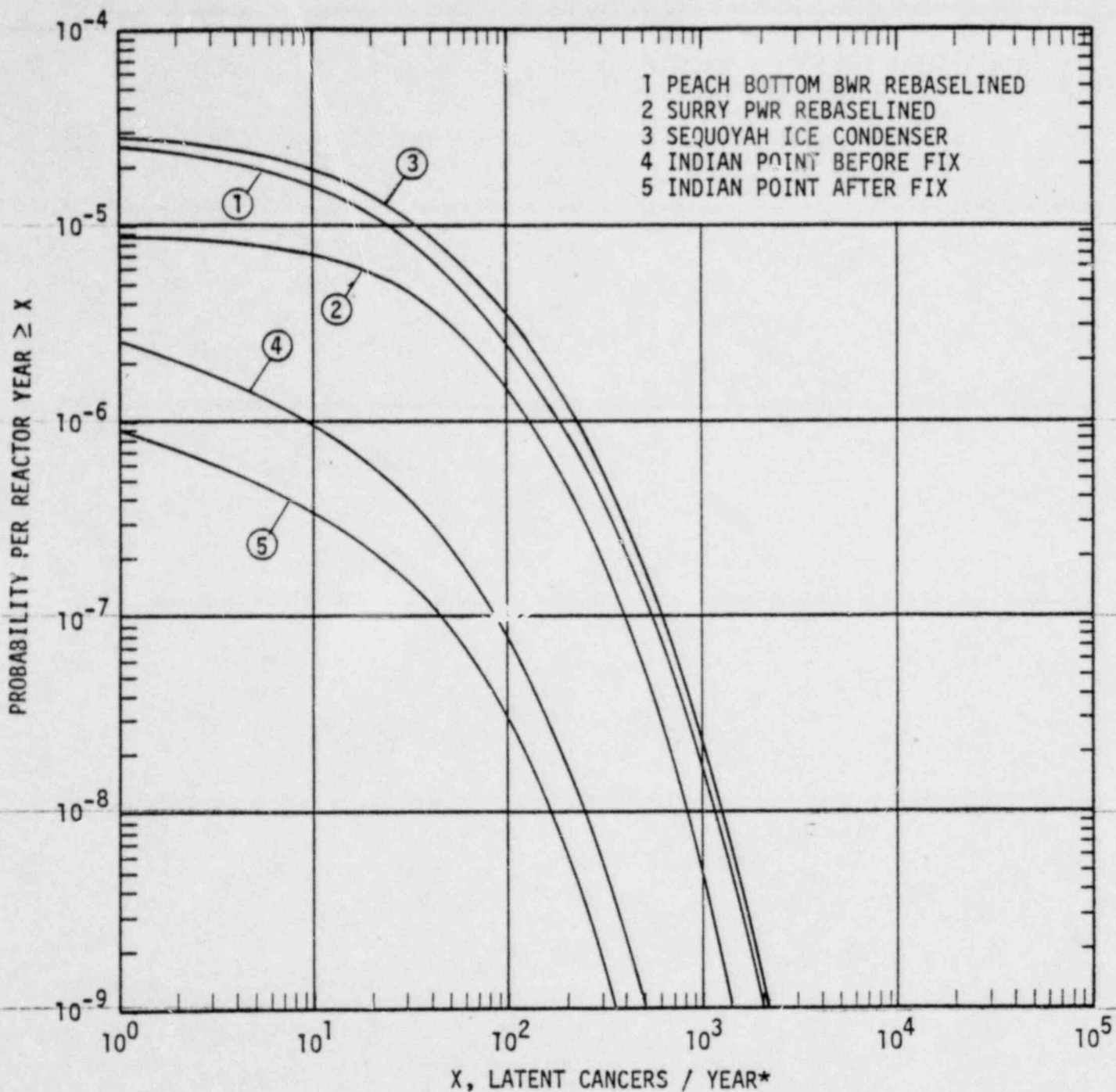
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION

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

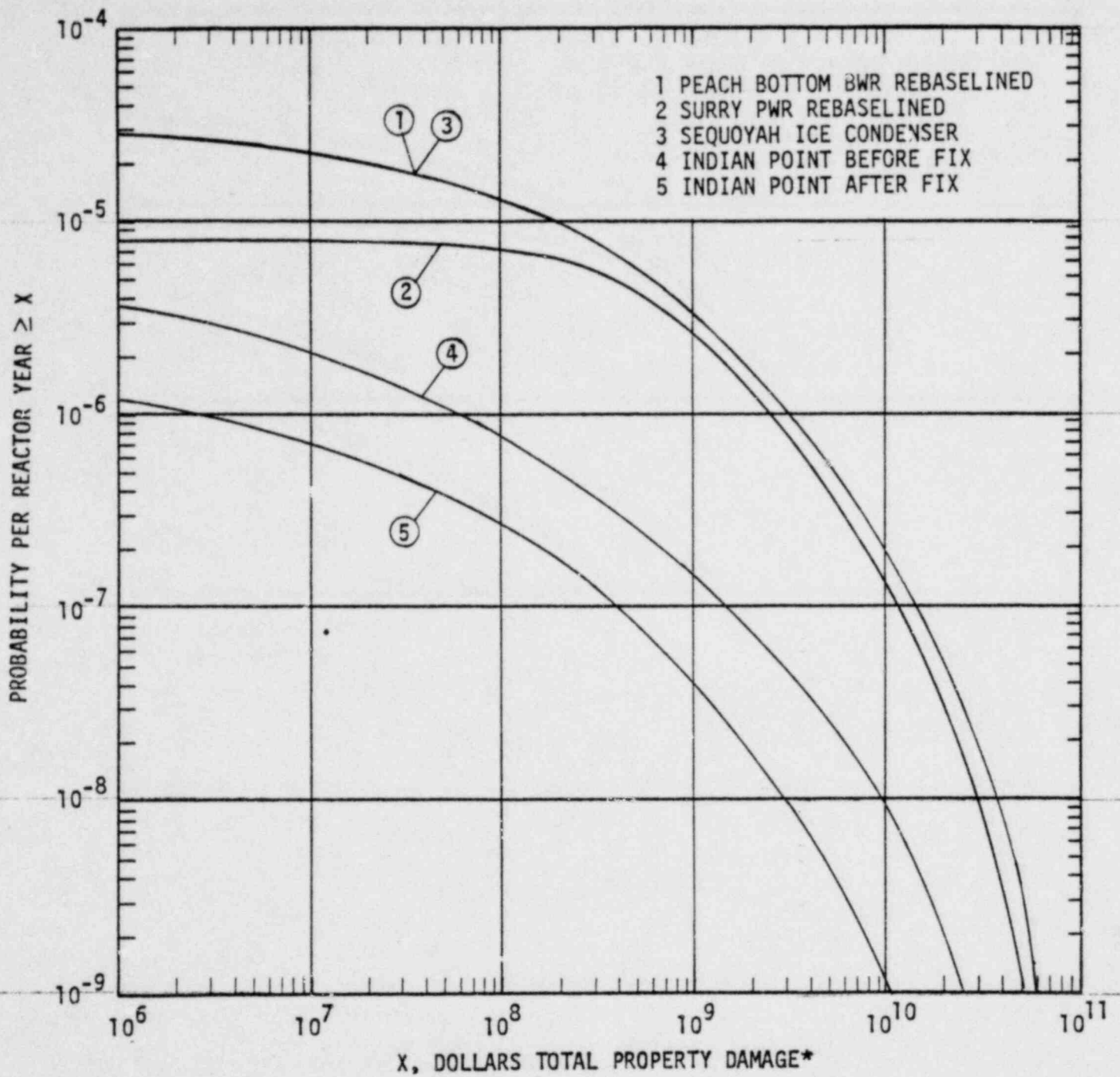


*TOTAL LATENT CANCERS WOULD BE 30 TIMES HIGHER

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	Prob. of Consequence occurring per yr	Early Fatalities	Early Injuries	Latent Cancer/Yr*	Property Damage \$**
IP After Fix		2.2×10^{-4}	2.7×10^{-4}	1.6×10^{-5}	199
IP Before Fix		6.3×10^{-4}	9.5×10^{-4}	4.4×10^{-5}	700
Surry Rebaselined		6.1×10^{-3}	1.5×10^{-2}	5.4×10^{-4}	9550
Sequoyah Ice Condenser		2.7×10^{-3}	2.2×10^{-2}	1.2×10^{-3}	14800
Peach Bottom BWR Rebaselined		1.7×10^{-2}	3.1×10^{-2}	1.1×10^{-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

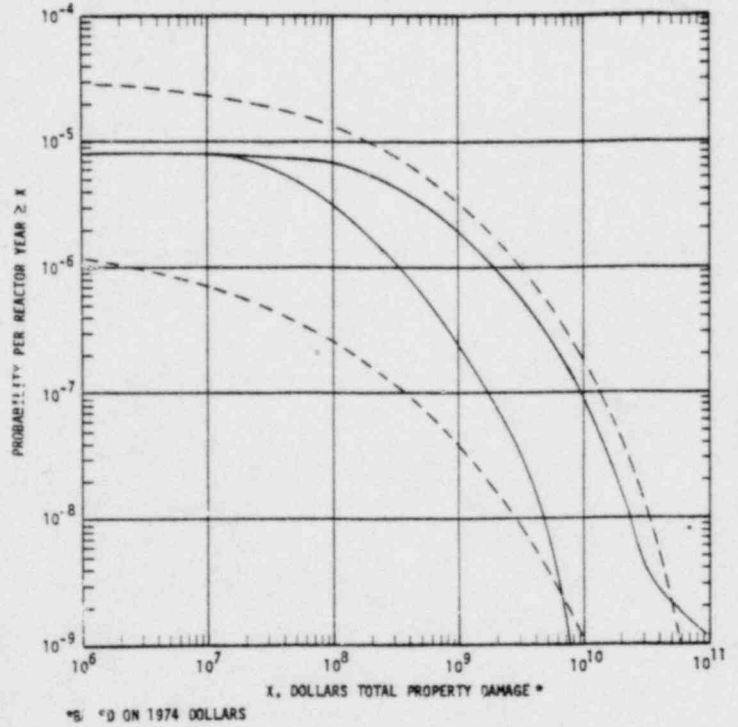
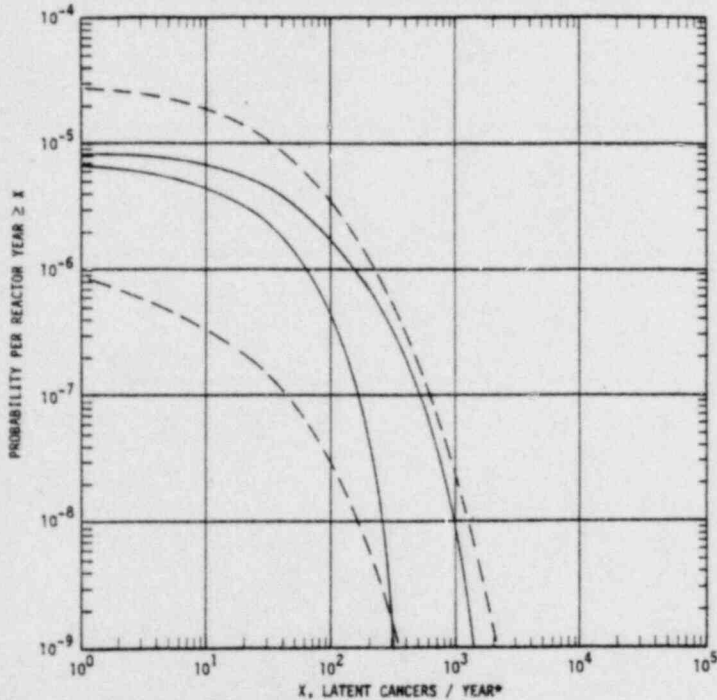
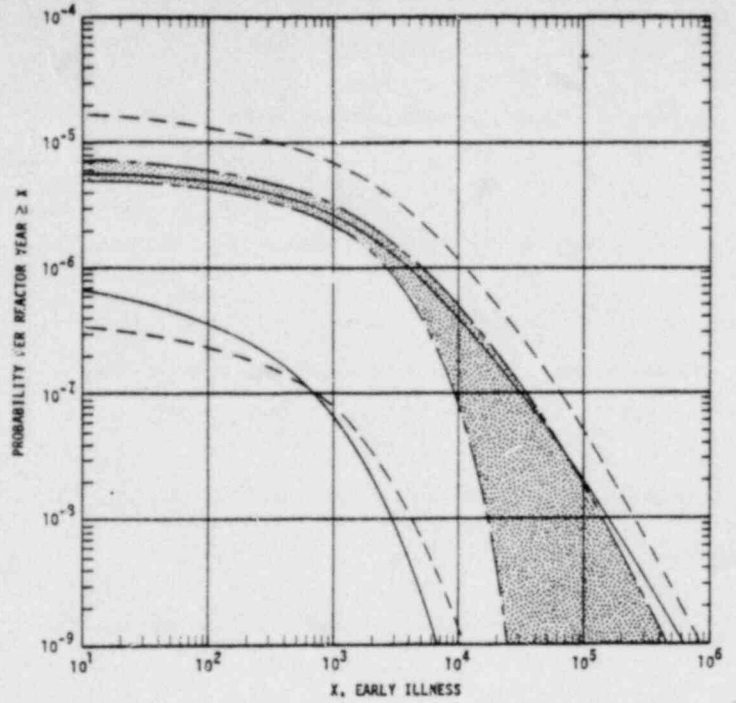
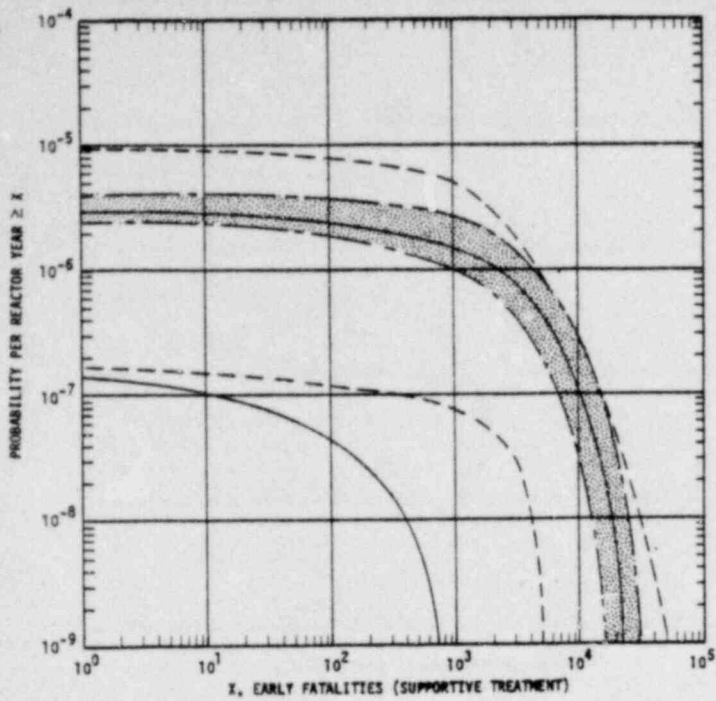
THE SENSITIVITY OF RISK TO VARIATIONS IN SITE, PUBLIC PROTECTION,
AND DESIGN/OPERATING CHARACTERISTICS

In the preceding 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 preceding 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 ELECTRICAL
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 1. 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 2. 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 1

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

	<u>CON ED</u>	<u>PASNY</u>	<u>TOTAL</u>
(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 2

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

	<u>CON ED</u>	<u>PASNY</u>	<u>TOTAL</u>
(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 3. 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 3

Bulk Power Transmission Capability
Above Scheduled Transfers (Mw)

<u>FROM</u>	<u>SUMMER 1980</u>	<u>WINTER 80-81</u>
Upper State New York	500	2200
Pennsylvania-New Jersey-Maryland Interconnection (PJM)	150	50
Long Island Lighting Co. (LILCO)	<u>475</u>	<u>550</u>
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

^{3/} Letter to Hanrahan, op. cit., p. 2.

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 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 operation. 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 for 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 others 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° 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:

UCS (#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 respect 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 risks 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 feed-water 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 Unit 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."

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_0 , that 300 deaths, the seating capacity of the aircraft, will occur. P_0 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_0 , say 0.33 crashes per year, times 300 deaths per crash or 100 deaths per year.

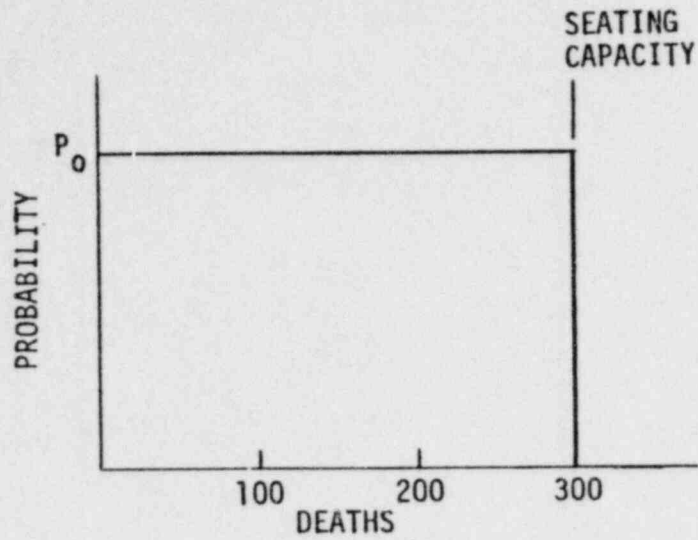
The CCDF can be corrected first to show that the falling aircraft might strike and kill people on the ground. Figure A-2 shows a tail on the CCDF curve reflecting that if the plane crashes, it will most likely not kill many people on the ground. At lower and lower probability, there is the chance of 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.

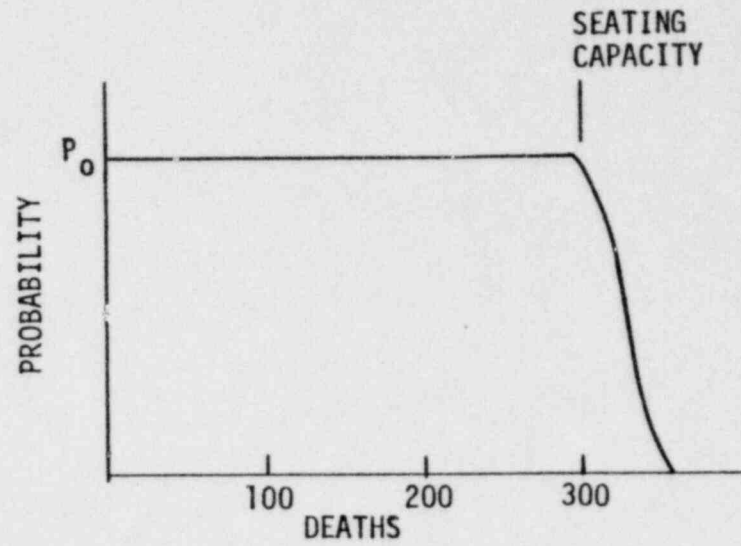
A second stage of refinement in this CCDF can be obtained if the airline gives us figures on the actual passenger loads the aircraft usually carries. If the data are limited, they might simply be reduced to the approximation that on 1/3 of the trips the plane is 1/3 full, on another 1/3 of the trips 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

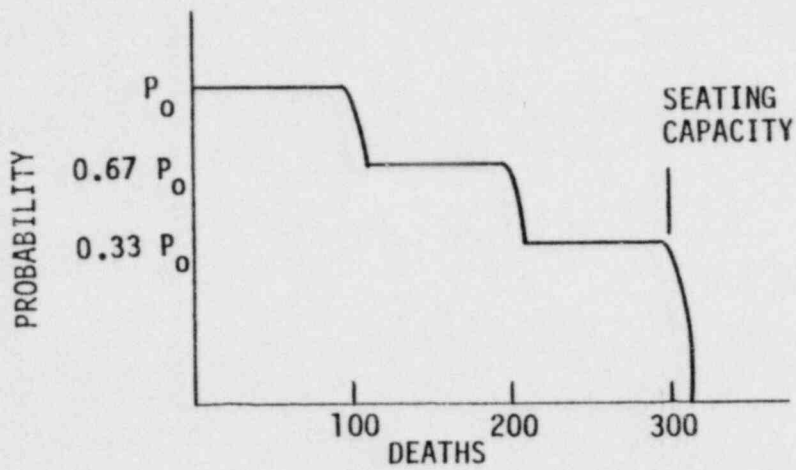


A-1.

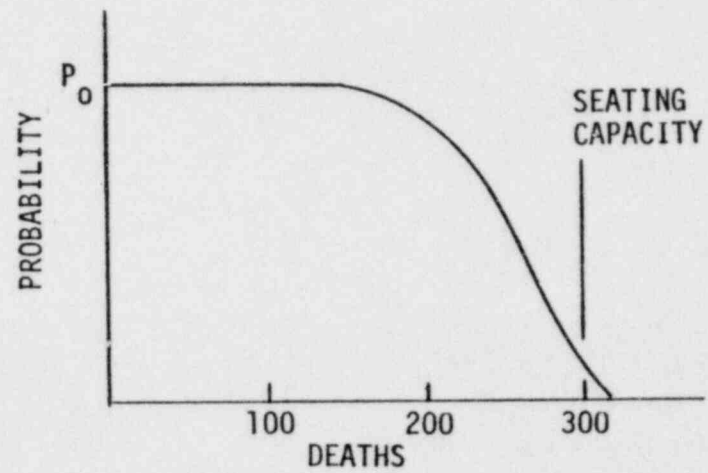


A-2

A-3



A-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 codes¹ 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.

¹It 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 (1×10^{-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 TCX' 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 (α) in the reactor vessel was decreased. This was done to reflect both experimental

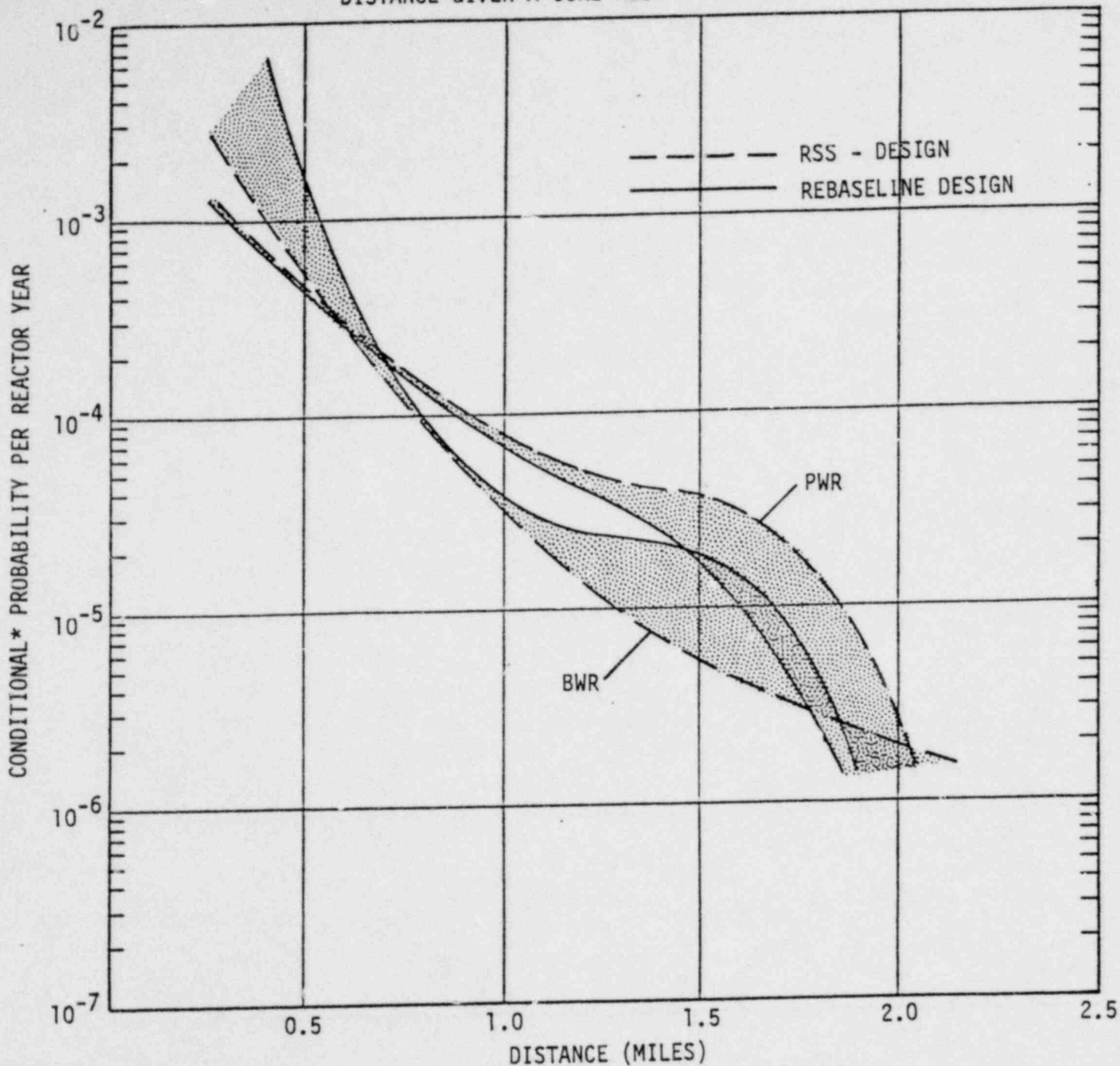
and calculational indications that such explosions are unlikely to occur in those sequences involving small size LOCAs and transients because of the high pressures and temperatures expected to exist within the reactor coolant system during these scenarios. Furthermore, if such an explosion were to occur, 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 TCX'.

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 (a).

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-weighted 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)

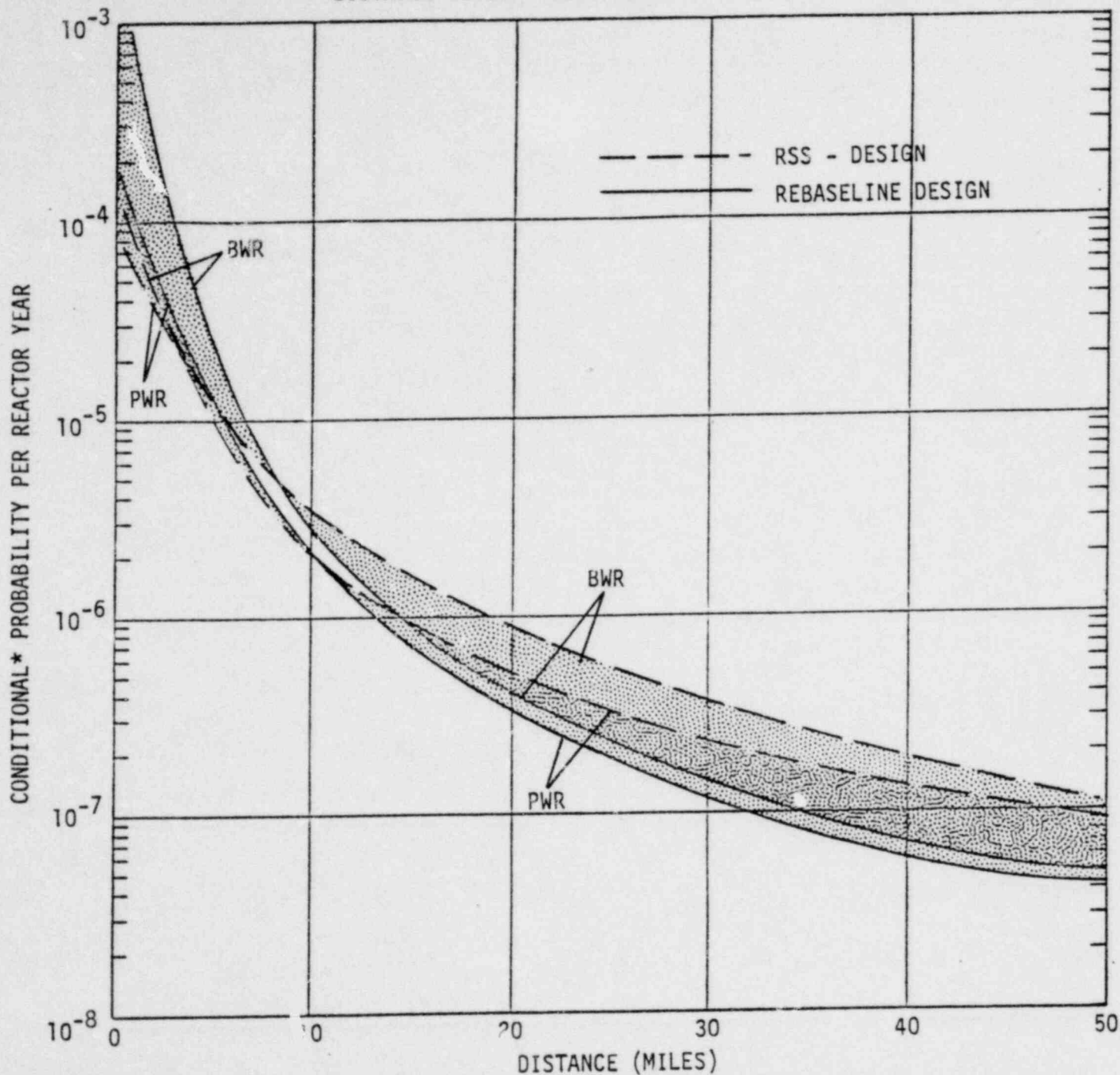
REBASELINE DESIGN

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

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



Department of Energy
Washington, D.C. 20461

MAY 15 1980

Mr. Edward J. Hanrahan, Director
Office of Policy Evaluation
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Hanrahan:

This letter summarizes the views of the Economic Regulatory Administration's Division of Power Supply and Reliability (DPSR) regarding the electric system reliability impact of various modes of operation of nuclear power units Indian Point 2 and 3 as described in your April 28, 1980, letter.

Indian Point 2 is a 849 MW (summer rating) PWR unit owned and operated by the Consolidated Edison Company of New York (CON ED). Indian Point 3 is a 965 MW PWR unit owned and operated by the Power Authority of the State of New York Inc. (PASNY). The units are co-located in Westchester County, 25 miles north of New York City. Both units are included in their respective entity's planned resources available to meet customer demands in 1980.

A shutdown of Indian Point 2 and 3 would impact the reserve capacity in the New York subregion (New York Power Pool) of the Northeast Power Coordinating Council (NPCC). Without Indian Point 2 and 3 the available reserves in the New York Power Pool for the summer 1980 and winter 1980-81 seasons would decline from 46.6 and 58.2 percent to 38.0 and 49.0 percent respectively. This level of reserves is still considered adequate to provide reliable electric service when viewed on a state-wide basis. However, due to limited transmission capability into the New York City and Westchester County areas, the complete shutdown of these two Indian Point units during the 1980 summer peak period could adversely impact the system reliability of CON ED and PASNY.

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. CON ED supplies electricity to over eight million customers. CON ED's summer peak load is about 40 percent higher than its winter peak load due mainly to the widespread use of electric air conditioning. CON ED's projected 1980 summer peak load is 6900 MW.

PASNY does not have any geographically defined "service territory" but serves particular classes of customers in all parts of the State of New York. PASNY's projected summer 1980 peak load is 2503 MW of which less than half is in New York City and Westchester County areas.

The DPSR collected data concerning the latest electric system conditions, maintenance schedules, expected forced outages, and expected peak loads for CON ED and PASNY. The data was compared to historical data contained in various DOE documents, and revised where it was felt necessary. The conclusions drawn in this letter are based upon our analysis of this data.

The adverse impact on reliability due to the status of IP 2 and 3 results from the limited transmission system capability for importing power from other parts of the state, or from neighboring states, into the area in which these units are located. CON ED has bulk power transmission tie lines with neighboring utilities having a megawatt transfer capability (above scheduled transfers) as shown below:

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

Energy transfers from areas outside of PJM or Upper State New York (USNY) would have to rely on the same transmission ties as transfers directly from Upper State New York and PJM. Therefore, the only time capacity available from these outside areas would need to be considered would be if the PJM and USNY areas did not have sufficient capacity to supply the transfer limit. Overall New York State generating capacity will be adequate in this time frame and all possible transfers from the north, up to the limit of the transmission system, could come from this area. PJM also projects sufficient available excess capacity during the 1980 summer to be able to provide the 150 MW of capacity for transfer. Long Island Lighting Company (LILCo) does not have sufficient available capacity this summer to provide the full 475 MW of the transfer capability and would only be able to supply an average of 100 MW. There are no other systems to the east of the area that could supply power over these Long Island transmission connections. There may be some contingency support through the submarine cable from Connecticut but this would be limited to only 145 MW.

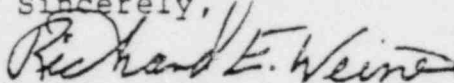
The four possible modes of operation of IP 2 and 3 which DPSP was requested to analyze are detailed below indicating their impact on CON ED and PASNY:

- o Operate IP 2 and 3 at 50 percent capacity for a 3 month period beginning June 1, 1980. The loss of 907 MW from the CON ED and PASNY systems during the summer peak load period along with the expected amounts of forced and scheduled outages, will enable the companies to supply their expected peak loads plus withstand the loss of the largest unit (Ravenswood #3 - 928 MW). This should be adequate for maintaining reliable electric service since normal tie transfers were considered.
- o Shutdown IP 2 and 3 for a three month period beginning June 1, 1980. The loss of 1814 MW from the CON ED and PASNY systems during the summer peak load period will force the New York City and Westchester County areas to depend very heavily upon the transmission interties with neighboring areas. Given the projected loads and expected forced and scheduled outages, the loss of the largest unit (Ravenswood #3 - 928 MW) would force the utilities to use all available capacity and interities to the maximum reasonable extent. Further facility failure, or loads greater than forecast would force the utilities to institute voltage reductions, load curtailments, or other actions as required to prevent widespread loss of customer load. Sustained high loads during the summer period would force CON ED to operate its 1987 MW of combustion turbine generation capacity for longer periods than the units are planned and designed to operate. This mode of operation places the system in a very vulnerable position and is not considered consistent with providing reliable electric service.
- o Operate IP 2 and 3 at 50 percent capacity for a 12 month period beginning June 1, 1980. The loss of 907 MW in the CON ED and PASNY systems for 12 months will have its greatest impact on system reliability during the summer months. This situation is discussed above. During the remainder of the year, CON ED and PASNY will have sufficient capacity to provide reliable electric service to their customers.

- o Shutdown IP 2 and 3 for a 12 month period beginning June 1, 1980. The loss of 1814 MW from CON ED and PASNY will have its greatest impact on system reliability during the summer months as discussed above. During the other months of the year, CON ED and PASNY will have sufficient available capacity on their own systems and from transfers from other areas to provide reliable electric service.

This analysis deals only with electric system reliability and energy supply; it does not consider the need to reduce operating costs and conserve oil or natural gas. The outages of any large non-oil generating unit in Southeastern New York will result in increased costs to the consumers of electricity because of the resulting increased use of low sulfur oil-fired generation. I would appreciate being notified of the decision regarding Indian Point 2 and 3.

Sincerely,



Richard E. Weiner, Director
Division of Power Supply
and Reliability
Economic Regulatory Administration