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in the FES-CP. This early program has been updated and expanded; it is presented in Section 6.1.5 of the applicant's ER-OL and is summarized here in Tables 5.8 through 5.11.

The applicant states that the preoperational program will have been implemented at least 2 years before initial criticality of Unit 1 to document background levels of direct radiation and concentrations of radionuclides that exist in the environment. The preoperational program will continue up to initial criticality of Unit 1, at which time the operational radiological monitoring program will commence.

The staff has reviewed the preoperational environmental monitoring plan of the applicant and finds that it is acceptable as presented. The current NRC staff position is that a total of about 40 dosimetry stations (or continuously recording dose-rate instruments) should be placed as follows: an inner ring of stations in the general area of the site boundary and an outer ring in the 6 to 8 km (4 to 5 mile) range from the site with a station in each sector of each ring (16 sectors x 2 rings = 32 stations). The remaining eight stations should be placed in special interest areas such as population centers, nearby residences and schools, and in two or three areas to serve as control stations. The station locations have been reviewed by the NRC staff and are specified in Table 5.9.

5.9.3.4.2 Operational

The operational offsite radiological-monitoring program is conducted to provide data on measurable levels of radiation and radioactive materials in the site environs in accordance with 10 CFR 20 and 50. It assists and provides backup support to the effluent-monitoring program recommended in RG 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants."

The applicant states that the operational program will in essence be a continuation of the preoperational program described above, with some periodic adjustment of sampling frequencies in expected critical exposure pathways--such as increasing milk sampling frequency and deletion of fruit, vegetable, soil, and gamma radiation survey samples. The proposed operational program will be reviewed prior to plant operation. Modification will be based upon anomalies and/or exposure pathway variations observed during the preoperational program.

The final operational-monitoring program proposed by the applicant will be reviewed in detail by the NRC staff, and the specifics of the required monitoring program will be incorporated into the operating license Radiological Technical Specifications.

5.9.4 Environmental Impacts of Postulated Accidents

5.9.4.1 Plant Accidents

The staff has considered the potential radiological impacts on the environment of possible accidents at the Limerick Generating Station, Units 1 and 2, in accordance with a Statement of Interim Policy published by the Nuclear Regulatory Commission on June 13, 1980 (45 FR 40101-40104). The following discussion reflects the staff's considerations and conclusions.

NUCLEAR REGULATORY COMMISSION

Docket No. 50-352/353 Official Exh. No. 29

In the matter of Philadelphia Electric

Staff ✓ IDENTIFIED ✓

Applicant RECEIVED ✓

Intervenor REJECTED

Cont'g Off'r

Contractor DATE 5/22/84

Other Witness

Reporter Ann Riley

Table 5.8 Preoperational radiological environmental monitoring program summary

Year	Sample type	No. of stations	Analysis	Frequency of analysis
1982 (partial)	Direct radiation	48	Gamma dose	Monthly
	Air (particulate & iodine)	17	Radioiodine (I-131)	---
			Gross beta	Weekly
			Gamma isotopic composite	Monthly
	Surface water	5	Gamma isotopic	Monthly
			Tritium composite	Quarterly
			Gross beta (soluble & insoluble)	Monthly
	Drinking water	5	Gamma isotopic	Monthly
			Tritium composite	Quarterly
			Gross beta (soluble & insoluble)	Monthly
	Groundwater	2	Gamma isotopic	Semi-annually
			Tritium	Semi-annually
	Sediment	3	Gamma isotopic	Semi-annually
	Fish	3	Gamma isotopic	Semi-annually
	Vegetation	1	Radioiodine	Monthly when available
	Milk	12	Radioiodine (I-131)	Quarterly
			Gamma isotopic	Quarterly
	Small game	1	Gamma isotopic	Annually
1983 (partial)	Direct radiation	48	Gamma dose	Monthly
	Air (particulate & iodine)	17	Gross beta	Weekly
			Gamma isotopic composite	Monthly

Table 5.8 (continued)

Year	Sample type	No. of stations	Analysis	Frequency of analysis
1983	Surface water	5	Gamma isotopic	Monthly
			Tritium composite	Quarterly
			Gross beta (soluble & insoluble)	Monthly
	Drinking water	5	Gamma isotopic	Monthly
			Tritium composite	Quarterly
			Gross beta (soluble & insolube)	Monthly
	Groundwater	2	Gamma isotopic	Semi-annually
			Tritium	Semi-annually
	Sediment	3	Gamma isotopic	Semi-annually
	Fish	3	Gamma isotopic	Semi-annually
	Vegetation	1	Radioiodine	Monthly during growing season
	Milk	12	Radioiodine (I-131)	Quarterly
	Small game	1	Gamma isotopic	Annually
1984	Direct radiation	48	Gamma dose	Monthly
	Air (particulate & iodine)	17	Radioiodine (I-131)	Weekly (7 stations)
			Gross beta	Weekly
			Gamma isotopic composite	Monthly

Table 5.8 (continued)

Year	Sample type	No. of stations	Analysis	Frequency of analysis
1984	Surface water	5	Gamma isotopic	Monthly
			Tritium composite	Quarterly
			Gross beta (soluble & insoluble)	Monthly
	Drinking water	5	Gamma isotopic	Monthly
			Tritium composite	Quarterly
			Gross beta (soluble & insoluble)	Monthly
	Groundwater	2	Gamma isotopic	Semi-annually
			Tritium	Semi-annually
	Sediment	3	Gamma isotopic	Semi-annually
	Fish	3	Gamma isotopic	Semi-annually
	Vegetation	1	Radioiodine	Monthly during growing season
	Milk	13	Radioiodine (I-131)	Bi-weekly during grazing season, monthly at other times (4 stations)
				Monthly analysis only (9 stations)
			Gamma isotopic	Quarterly
	Small Game	1	Gamma isotopic	Annually

Source: ER-OL Table 6.1-45, through Revision 17, February 1984

Table 5.9 Preoperational radiological environmental monitoring program station locations

Location description	Code	Sector	Distance (km)
TLD (inner ring)			
Evergreen & Sanatoga Rd., N sector site boundary	36S1	N	0.97
Sanatoga Rd., NNE sector site boundary	3S1	NNE	0.97
Possum Hollow Rd.	5S1	NE	0.64
Limerick Training Center	7S1	ENE	0.80
Keen Rd.	10S1	E	0.80
Limerick Information Center	11S1	ESE	0.80
Longview Rd., SE sector site boundary	14S1	SE	0.97
Longview Rd., SSE sector site boundary	16S2	SSE	0.97
Railroad tracks along Longview Rd.	18S1	S	0.48
Impounding basin, SSW sector site boundary	21S1	SSW	0.80
Transmission tower, SW sector site boundary	23S2	SW	0.80
WSW sector site boundary	25S1	WSW	0.80
Met tower 2 site	26S3	W	0.64
WNW sector site boundary	29S1	WNW	0.80
NW sector site boundary	32S1	NW	0.97
Met tower 1 site	34S2	NNW	0.97
TLD (outer ring)			
Ringing Rock substation	35F1	N	6.8
Laughing Waters GSC	2E1	NNE	8.2
Neiffer Rd.	4E1	NE	7.4
Pheasant Rd. Game Farm site	7E1	ENE	6.8
Transmission corrider, Royersford Rd.	10E1	E	6.3
Trappe substation	10F3	ESE	8.8
Vaughn substation	13E1	SE	6.9
Pikeland substation	16F1	SSE	7.9

Table 5.9 (continued)

Location description	Code	Sector	Distance (km)
Snowden substation	19D1	S	5.8
Sheeder substation	20F1	SSW	8.4
Porters Mill substation	24D1	SW	6.3
Transmission corrider, Hoffecker & Keim Sts.	25D1	WSW	6.4
Transmission corrider, W. Cedarville Rd.	28D2	W	6.1
Prince St.	29E1	WNW	7.9
Poplar substation	31D2	NW	6.3
Yarnell Rd.	34E1	NNW	7.4
TLD (control stations and other selected locations)			
Sanatoga substation	2B1	NNE	2.4
Birch substation	5H1	NE	42
Pottstown landing field	6C1	ENE	3.4
Reed Rd.	9C1	E	3.5
King Rd.	13C1	SE	4.7
3508 Market St., Philadelphia	13H3	SE	45
Spring City substation	15D1	SE	5.1
Linfield substation	17B1	S	2.6
Planebrook substation	18G1	S	21
Ellis Woods Rd.	20D1	SSW	5
Manor substation	22G1	SW	28
Old Schuylkill Rd.	26B1	W	2.7
Yost Rd.	29B1	WNW	2.9
Lincoln substation	31D1	NW	4.8
Friedensburg substation	32G1	NW	25
Pleasantview Rd.	35B1	NNW	3.1
Dairy farms			
	5C1	NE	4.2
	9E1	E	6.6
	9G1	E	18
	10B1	ESE	1.8
	10C1	ESE	4.5

Table 5.9 (continued)

Location description	Code	Sector	Distance (km)
	11E1	ESE	7.9
	17C2	S	4.0
	17D1	S	5.8
	18C1	S	3.1
	21B1	SW	2.7
	22F1	SW	16
	25B1	WSW	2.1
	36E1	N	7.6
<u>Air particulate and iodine</u>			
Sanatoga substation	2B1	NNE	2.4
Pottstown landing field	6C1	ENE	3.4
Reed Rd.	9C1	E	3.5
Keen Rd.	10S3	E	0.80
Limerick Information Center	11S1	ESE	0.80
King Rd.	13C1	SE	4.7
2301 Market St., Philadelphia	13H4	SE	46
Longview Rd., SE sector site boundary	14S1	SE	0.97
Spring City substation	15D1	SE	5.1
Linfield substation	17B1	S	2.6
Ellis Woods Rd.	20D1	SSW	5
Manor substation	22G1	SW	28
Old Schuylkill Rd.	26B1	W	2.7
Yost Rd.	29B1	WNW	2.9
Lincoln substation	31D1	NW	4.8
Met tower 1	34S2	NNW	0.97
Pleasantview Rd.	35B1	NNW	3.1
<u>Vegetation</u>			
Limerick Information Center garden	11S1	ESE	0.80
<u>Fish</u>			
Upstream of Limerick (Keim St. bridge to Hanover St. bridge)	29C1*		

Table 5.9 (continued)

Location description	Code	Sector	Distance (km)
Downstream of Limerick discharge	20S1*		
Middle of Vincent pool upstream to Pigeon Creek	16C5*		
<u>Game</u>			
Fricks Lock, Limerick vicinity	26S5*		
<u>Sediment</u>			
Upriver from Limerick discharge	33A2*		
Linfield bridge area	16B2*		
Vincent Dam pool area	16C4*		
<u>Water sampling stations</u>			
Surface water:			
Limerick intake	24S1*		
Fricks Lock boat house	24S2*		
Linfield bridge	16B2*		
Philadelphia Suburban Water Company	15F5*		
Perkiomen pumping station	10F2*		
<u>Drinking water</u>			
Philadelphia Suburban Water Company	15F4*		
Phoenixville Water Works	15F7*		
Citizens Home Water Company	16C2*		
Pottstown Water Authority	28F3*		
Belmont Water Works (Philadelphia)	13H2*		
<u>Well Water</u>			
Limerick Information Center	11S1*		
<u>Well Water</u>			
S sector farm near site	18A1*		

*See ER-OL Figures 6.1-23 through 6.1-29 for details.

Source: ER-OL Table 6.1-46, through Revision 17, February 1984

Table 5.10 Detection capabilities for environmental sample analyses

Sample type	Analysis	Sensitivity LLD*	Nonroutine reporting levels	Units
Surface water	Gross beta (insol)	4	200	pCi/l
	Gross beta (sol)	4	200	
	Tritium	2000	20000	
	Gamma			
	Mn-54	15	1000	
	Fe-59	30	400	
	Co-58	15	1000	
	Co-60	15	300	
	Zn-65	30	300	
	Zr-95	30	400	
	Nb-95	15	400	
	Cs-134	15	30	
	Cs-137	18	50	
	Ba-140	60	200	
	La-140	15	200	
Drinking water	Gross beta (insol)	4	200	pCi/l
	Gross beta (sol)	4	200	
	Tritium	2000	20000	
	Gamma			
	Mn-54	15	1000	
	Fe-59	30	400	
	Co-58	15	1000	
	Co-60	15	300	
	Zn-65	30	300	
	Zr-95	30	400	
	Nb-95	15	400	
	Cs-134	15	30	
	Cs-137	18	50	
	Ba-140	60	200	
	La-140	15	200	
Well water	Tritium	2000	20000	pCi/l
	Gamma			
	Mn-54	15	1000	
	Fe-59	30	400	
	Co-58	15	1000	
	Co-60	15	300	
	Zn-65	30	300	
	Zr-95	30	400	
	Nb-95	15	400	
	Cs-134	15	30	
	Cs-137	18	50	
	Ba-140	60	200	

Table 5.10 (continued)

Sample type	Analysis	Sensitivity LLD*	Nonroutine reporting levels	Units
Milk	I-131	1	3	pCi/l
	Gamma			
	Cs-134	15	60	
	Cs-137	18	70	
	Ba-140	60	300	
	La-140	15	300	
Food products	Gamma			
	I-131	0.06	0.1	pCi/g(wet)
	Cs-134	0.06	1.0	
	Cs-137	0.08	2.0	
Game	Gamma			
	Cs-134	0.06		pCi/g(wet)
	Cs-137	0.08		
Fish	Gamma			
	Mn-54	0.130	30	pCi/g(wet)
	Fe-59	0.260	10	
	Co-58	0.130	30	
	Co-60	0.130	10	
	Zn-65	0.260	20	
	Cs-134	0.130	1	
	Cs-137	0.150	2	
Sediment	Gamma			
	Cs-134	0.150		pCi/g(dry)
	Cs-137	0.180		
Air particulates	Gross beta	0.01		pCi/m^3
	Gamma			
	Cs-134	0.05	10	
	Cs-137	0.06	20	
Air iodine	I-131	0.07	0.9	pCi/m^3
Direct radiation	TLD	RG 4.15		mrad/std month

*LLD is the "a priori" lower limit of detection, defined as the smallest concentration of radioactive material in a sample (picocuries per unit of mass or volume) that will yield a net count, above system background, that will be detected with 95% probability, with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

Source: ER-OL Table 6.1-47, through Revision 17, February 1984

Table 5.11 Environmental sampling and measuring equipment

Sample type of measurement	Equipment
Airborne particulate & radioiodine	Continuous air pump that passes approximately 1 cfm through filter paper and charcoal cartridge
Surface water (composite)	Automatic composite sampler
Drinking water (composite)	Automatic composite sampler
Direct radiation	Thermoluminescent dosimeter
Fish	Trap net, seine, hook and line, electro fishing apparatus and/or equivalent equipment

Source: ER-OL Table 6.1-48, Revision 17, February 1984

Section 5.9.4.2 deals with general characteristics of nuclear power plant accidents, including a brief summary of safety measures provided to minimize the probability of their occurrence and to mitigate their consequences if they should occur. Also described are the important properties of radioactive materials and the pathways by which they could be transported to become environmental hazards. Potential adverse health effects and impacts on society associated with actions to avoid such health effects also are identified.

Next, actual experience with nuclear power plant accidents and their observed health effects and other societal impacts are described. This is followed by a summary review of safety features of the Limerick station and of the site that act to mitigate the consequences of accidents.

The results of calculations of the potential consequences of accidents that have been postulated in the design basis are then given. Also described are the results of calculations for the Limerick site using contemporary probabilistic methods and their inherent uncertainties to estimate the possible impacts and the risks associated with severe accident sequences of low probability of occurrence.

5.9.4.2 General Characteristics of Accidents

The term "accident," as used in this section, refers to any unintentional event not addressed in Section 5.9.3 that results in a release of radioactive materials into the environment. The predominant focus, therefore, is on events that can lead to releases substantially in excess of permissible limits for normal operation. Normal release limits are specified in the Commission's regulations at 10 CFR 20, and 10 CFR 50, Appendix I.

There are several features that combine to reduce the risk associated with accidents at nuclear power plants. Safety features provided for in design, construction, and operation comprise the first line of defense and are to a very large extent devoted to the prevention of the release of radioactive materials from their normal places of confinement within the plant. There are also a number of additional lines of defense that are designed to mitigate the consequences of failures in the first line. These safety features are designed taking into consideration the specific locations of radioactive materials within the plant; their amounts; their nuclear, physical, and chemical properties; and their relative tendency to be transported into and for creating biological hazards in the environment. Descriptions of these features for Limerick Units 1 and 2 may be found in the applicant's FSAR and in the staff's Safety Evaluation Report (SER, NUREG-0991). The most important mitigative features are described in Section 5.9.4.4(1) below.

(1) Fission Product Characteristics

By far the largest inventory of radioactive material in a nuclear power plant is produced as a byproduct of the fission process and is located in the uranium oxide fuel pellets in the reactor core in the form of fission products. During periodic refueling shutdowns, the assemblies containing these fuel pellets are transferred to a spent-fuel storage pool so that the second largest inventory of radioactive material is located in this storage area. Much smaller inventories of radioactive materials also are normally present in the water that circulates in the reactor coolant system and in the systems used to process gaseous and liquid radioactive wastes in the plant.

All these radioactive materials exist in a variety of physical and chemical forms. Their potential for dispersion into the environment depends not only on mechanical forces that might physically transport them, but also upon their inherent properties, particularly their volatility. The majority of these materials exist as nonvolatile solids over a wide range of temperatures. Some, however, are relatively volatile solids and a few are gaseous in nature. Such characteristics have a significant bearing upon the assessment of the environmental radiological impact of accidents.

The gaseous materials include radioactive forms of the chemically inert noble gases krypton and xenon. These have the highest potential for release into the atmosphere. If a reactor accident were to occur involving degradation of the fuel cladding, the release of substantial quantities of these radioactive gases from the fuel is a virtual certainty. Such accidents are of low frequency, but are considered credible events (see Section 5.9.4.3). It is for this reason that the safety analysis of each nuclear power plant incorporates a hypothetical design-basis accident that postulates the release of the entire contained inventory of radioactive noble gases from the fuel in the reactor vessel into the containment structure. If these gases were further released to the environment as a possible result of failure of safety features, the hazard to individuals from these noble gases would arise predominantly through the external gamma radiation from the airborne plume. The reactor containment structure and other features are designed to minimize this type of release.

Radioactive forms of iodine are formed in substantial quantities in the fuel by the fission process and in some chemical forms may be quite volatile. For these reasons, they have traditionally been regarded as having a relatively high potential for release (1) from the fuel at higher than normal temperatures, or (2) from defects in fuel pins. If radioiodines are released to the environment, the principal radiological hazard associated with the radioiodines is incorporation into the human body and subsequent concentration in the thyroid gland. Because of this, the potential for release of radioiodines to the atmosphere is reduced by the use of special structures, components, and systems designed to retain the iodine. The chemical forms in which the fission product radioiodines are found are generally solid materials at room temperatures, so they have a strong tendency to condense (or "plate out") upon cooler surfaces. In addition, most of the iodine compounds are quite soluble in or chemically reactive with water. Although these properties do not inhibit the release of radioiodines from degraded fuel, they do act to mitigate the release both to and from containment structures that have large internal surface areas and that contain large quantities of water as a result of an accident. The same properties affect the behavior of radioiodines that may "escape" into the atmosphere. Thus, if rainfall occurs during a release, or if there is moisture on exposed surfaces (for example, dew), the radioiodines will show a strong tendency to be absorbed by the moisture. Although less volatile than many iodine compounds, virtually all cesium and rubidium (alkali metals) compounds are soluble in or react strongly with water, and would behave similarly in the presence of moisture. In addition, the more volatile iodine compounds are capable of reacting with vegetation and traces of organic gases and pollen normally present in air, while many alkali metal compounds are capable of reacting with siliceous materials such as concrete, glass and soil.

Other radioactive materials formed during the operation of a nuclear power plant have lower volatilities and by comparison with the noble gases, iodine and alkali metals have a much smaller tendency to escape from degraded fuel unless the temperature of the fuel becomes very high. By the same token, if such materials escape by volatilization from the fuel, they tend (1) to condense quite rapidly to solid form again when they are transported to a region of lower temperature and/or (2) to dissolve in water when it is present. The former mechanism can have the result of producing some solid particles of sufficiently small size to be carried some distance by a moving stream of gas or air. If such particulate materials are dispersed into the atmosphere as a result of failure of the containment barrier, they will tend to be carried downwind and deposit on surfaces by gravitational settling or by precipitation (fallout), where they will become "contamination" hazards in the environment.

All of these radioactive materials exhibit the property of radioactive decay with characteristic half-lives ranging from fractions of a second to many days or years (see Table 5.11a). Many of them decay through a sequence or chain of decay processes, and all eventually become stable (nonradioactive) materials. The radiation emitted during these decay processes is the reason that they are hazardous materials. As a result of radioactive decay, most fission product elements transmute into other elements. Iodines transmute into noble gases, for example, while the noble gases transmute into alkali metals. Because of this property, fission products which escape into the environment as one element may later become a contamination hazard as a different element.

(2) Exposure Pathways

The radiation exposure (hazard) to individuals is determined by their proximity to the radioactive materials, the duration of exposure, and factors that act to

Table 5.11a Activity of radionuclides in a Limerick reactor core at 3458 MWt (WASH-1400 basis)

Group/radionuclide	Radioactive inventory (millions of Ci)	Half-life (days)
A. NOBLE GASES		
Krypton-85	0.6	3,950
Krypton-85m	30	0.183
Krypton-87	50	0.0528
Krypton-88	70	0.117
Xenon-133	200	5.28
Xenon-135	40	0.384
B. IODINES		
Iodine-131	90	8.05
Iodine-132	100	0.0958
Iodine-133	200	0.875
Iodine-134	200	0.0366
Iodine-135	200	0.280
C. ALKALI METALS		
Rubidium-86	0.03	18.7
Cesium-134	8	750
Cesium-136	3	13.0
Cesium-137	5	11,000
D. TELLURIUM-ANTIMONY		
Tellurium-127	6	0.391
Tellurium-127m	1	109
Tellurim-129	30	0.048
Tellurim-129m	6	34.0
Tellurium-131m	10	1.25
Tellurium-132	100	3.25
Antimony-127	7	3.88
Antimony-129	40	0.179
E. ALKALINE EARTHS		
Strontium-89	100	52.1
Strontium-90	4	11,030
Strontium-91	100	0.403
Barium-140	200	12.8
F. COBALT AND NOBLE METALS		
Cobalt-58	0.8	71.0
Cobalt-60	0.3	1,920
Molybdenum-99	200	2.8
Technetium-99m	200	0.25

Table 5.11a (Continued)

Group/radionuclide	Radioactive inventory (millions of Ci)	Half-life (days)
F. <u>COBALT AND NOBLE METALS</u> (Continued)		
Ruthenium-103	100	39.5
Ruthenium-105	100	0.185
Ruthenium-106	30	366
Rhodium-105	50	1.50
G. <u>RARE EARTHS, REFRACTORY OXIDES AND TRANSURANICS</u>		
Yttrium-90	4	2.67
Yttrium-91	100	59.0
Zirconium-95	200	65.2
Zirconium-97	200	0.71
Niobium-95	200	35.0
Lanthanum-140	200	1.67
Cerium-141	200	32.3
Cerium-143	100	1.38
Cerium-144	100	284
Praseodymium-143	100	13.7
Neodymium-147	60	11.1
Neptunium-239	2000	2.35
Plutonium-238	0.06	32,500
Plutonium-239	0.02	8.9×10^6
Plutonium-240	0.02	2.4×10^6
Plutonium-241	4	5,350
Americium-241	0.002	1.5×10^5
Curium-242	0.5	163
Curium-244	0.03	6,630

Note: The above grouping of radionuclides corresponds to that in Table 5.11c. The listed inventory has been rounded to one significant digit to reflect its accuracy in describing the Limerick core. All calculations, however, were done using the CRAC data file at much higher precision.

shield the individual from the radiation. Pathways that lead to radiation exposure nazards to humans are generally the same for accidental as for "normal" releases. These are depicted in Figure 5.4. There are two additional possible pathways that could be significant for accident releases that are not shown in Figure 5.4. One of these is the fallout onto open bodies of water of radioactivity initially carried in the air. The second would be unique to an accident that results in temperatures inside the reactor core sufficiently high to cause uncontrolled or unmitigated melting and subsequent penetration of the basemat underlying the reactor by the molten core debris. This situation could create the potential for the release of radioactive material into the hydrosphere through contact with groundwater, and may lead to external exposure to radiation and to internal exposures if radioactive material is inhaled or ingested from contaminated food or water.

It is characteristic of the transport of radioactive material by wind or by water that the material tends to spread and disperse, like a plume of smoke from a smokestack, becoming less concentrated in larger volumes of air or water. The results of these natural processes are to lessen the intensity of exposure to individuals downwind or downstream of the point of release, but to increase the number who may be exposed. The bulk of radioactive releases is more likely to reach the atmosphere than to reach streams or groundwater. For a release into the atmosphere, the degree to which dispersion reduces the concentration in the plume at any downwind point is governed by the turbulence characteristics of the atmosphere, which vary considerably with time and from place to place. This fact, taken in conjunction with the variability of wind direction and the presence or absence of precipitation, means that accident consequences are very much dependent upon the weather conditions existing at the time of the accident.

(3) Health Effects

The cause-and-effect relationships between radiation exposure and adverse health effects are quite complex (National Research Council, 1979; Land, 1980), but they have been studied exhaustively in comparison to many other environmental contaminants.

Whole-body radiation exposure resulting in a dose greater than about 10 rems for a few persons and about 25 rems for nearly all people over a short period of time (hours) is necessary before any physiological effects to an individual are clinically detectable. Doses about 7 or more times larger than the latter dose also received over a relatively short period of time (hours to a few days), can be expected to cause some fatal injuries. At the severe but extremely low probability end of the accident spectrum, exposures of these magnitudes are theoretically possible for persons in close proximity to such accidents if measures are not or cannot be taken to provide protection, such as sheltering or evacuation.

Lower levels of exposures also may constitute a health risk, but the ability to define a direct cause-and-effect relationship between any given health effect and a known exposure to radiation is difficult, given the backdrop of the many other possible reasons why a particular effect is observed in a specific individual. For this reason, it is necesary to assess such effects on a statistical basis. Such effects include randomly occurring cancer in the exposed population and genetic changes in future generations after exposure of a prospective parent. The occurrence of cancer itself is not necessarily indicative of fatality, however. Occurrences of cancer in the exposed population may begin to develop only after a lapse of 1 to 15 years (latent period) from the time of exposure and then continue over a period of about 30 years (plateau period). However, in the case of exposure to fetuses (in utero), occurrences of cancer may begin to develop at birth (no latent period) and end at age 10 (that is, the plateau period is 10 years). The health consequences model used was based on the 1972 BEIR I Report of the National Academy of Sciences (NAS, 1972).

Most authorities agree that a reasonable, and probably conservative, estimate of the randomly occurring number of health effects of low levels of radiation exposure to a large number of people is within the range of about 10 to 500 potential cancer deaths per million person-rems (although zero is not excluded by the data). The range comes from the latest NAS BEIR III Report (1980), which also indicates a probable value of about 150. This value is virtually identical to the value of about 140 used in the NRC health-effects models. In addition, approximately 220 genetic changes per million person-rems would be projected over succeeding generations by models suggested in the BEIR III report. This also compares well with the value of about 260 per million person-rems used by the NRC staff, which was computed as the sum of the risk of specific genetic defects and the risk of defects with complex etiology.

(4) Health Effects Avoidance

Radiation hazards in the environment tend to disappear by the natural processes of radioactive decay and weathering. However, where the decay process is slow, and where the material becomes relatively fixed in its location as an environmental contaminant (such as in soil), the hazard can continue to exist for a relatively long period of time--months, years, or even decades. Thus, a possible consequential environmental societal impact of severe accidents is the avoidance of the health hazard rather than the health hazard itself, by restrictions on the use of the contaminated property or contaminated foodstuffs, milk, and drinking water. The potential economic impacts that this avoidance can cause are discussed below.

5.9.4.3 Accident Experience and Observed Impacts

As of February 1983, there were 76 commercial nuclear power reactor units licensed for operation in the United States at 52 sites, with power-generating capacities ranging from 50 to 1180 megawatt electric (MWe). (Limerick Units 1 and 2 are designed for 1055 MWe per unit). The combined experience with all these units represents approximately 500 reactor years of operation over an elapsed time of about 20 years. Accidents have occurred at several of these facilities (Oak Ridge National Laboratory, 1980; NUREG-0651). Some of these have resulted in releases of radioactive material to the environment ranging from very small fractions of a curie to a few million curies. None is known to have caused any radiation injury or fatality to any specific member of the public, nor any significant individual or collective public radiation exposure, nor any significant contamination of the environment. This experience base is not large enough to permit a reliable quantitative statistical inference for predicting accident probabilities. It does, however, suggest that significant environmental impacts caused by accidents are very unlikely to occur over time periods of a few decades.

Melting or severe degradation of reactor fuel has occurred in only one of these units, during the accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979. In addition to the release to the environment of a few million curies of noble gases, mostly xenon-133, it has been estimated that approximately 15 curies of radioiodine also were released to the environment at TMI-2 (NRC Special Inquiry Group, 1980). This amount represents an extremely minute fraction of the total radioiodine inventory present in the reactor at the time of the accident. No other radioactive fission products were released to the environment in measurable quantity. It has been estimated that the maximum cumulative offsite radiation dose to an individual was less then 100 mrems (NRC Special Inquiry Group, 1980; President's Commission on the Accident at Three

Mile Island, 1979). The total population exposure has been estimated to be in the range from about 1000 to 5300 person-rems. This exposure could produce between none and one additional fatal cancer over the lifetime of the population. The same population receives each year from natural background radiation about 240,000 person-rems. Approximately a half-million cancers are expected to develop in this group over their lifetimes (NRC Special Inquiry Group, 1980; President's Commission on the Accident at Three Mile Island, 1979), primarily from causes other than radiation. Trace quantities (barely above the limit of detectability) of radioiodine were found in a few samples of milk produced in the area. No other food or water supplies were impacted.

Accidents at nuclear power plants also have caused occupational injuries and a few fatalities, but none attributed to radiation exposure. Individual worker exposures have ranged up to about 5 rems as a direct consequence of reactor accidents (although there have been higher exposures to individual workers as a result of other unusual occurrences). However, the collective worker exposure levels (person-rem) are a small fraction of the exposures experienced during normal routine operations that average about 440 to 1300 preson-rems in a PWR and 790 to 1660 person-rems in a BWR per reactor-year.

Accidents also have occurred at other nuclear reactor facilities in the United States and in other countries (Oak Ridge National Laboratory, 1980; NUREG-0651). Because of inherent differences in design, construction, operation, and purpose of most of these other facilities, their accident record has only indirect relevance to current nuclear power plants. Melting of reactor fuel occurred in at least seven of these accidents, including the one in 1966 at the Enrico Fermi Atomic Power Plant, Unit 1. Fermi Unit 1 was a sodium-cooled fast breeder demonstration reactor designed to generate 61 MWe. This accident did not release any radioactivity to the environment. The damages were repaired and the reactor reached full power 4 years following the accident. It operated successfully and completed its mission in 1973.

A reactor accident in 1957 at Windscale, England, released a significant quantity of radioiodine, approximately 20,000 curies, to the environment (United Kingdom Atomic Energy Office, "Accident at Windscale," 1957). This reactor, which was not operated to generate electricity, used air rather than water to cool the uranium fuel. During a special operation to heat the large amount of graphite in this reactor (characteristic of a graphite-moderated reactor), the fuel overheated and radioiodine and noble gases were released directly to the atmosphere from a 123-m (405-foot) stack. Milk produced in a 518-km^2 (200-mi^2) area around the facility was impounded for up to 44 days. The United Kingdom National Radiological Protection Board estimated that the releases may have caused about 260 cases of thyroid cancer, about 13 of them fatal, and about 7 deaths from other cancers or hereditary diseases (NRPB-R135, Crick and Linsley, 1982). This kind of accident cannot occur in a water moderated- and -cooled reactor like Limerick, however.

5.9.4.4 Mitigation of Accident Consequences

Pursuant to the Atomic Energy Act of 1954, the NRC conducted a safety evaluation of the application to operate Limerick Units 1 and 2 (NUREG-0991). Although NUREG-0991 contains more detailed information on plant design, the principal design features are addressed in the following section.

(1) Design Features

Limerick Units 1 and 2 are essentially identical. Each unit contains features designed to prevent accidental release of fission products from the fuel and to lessen the consequences should such a release occur. These accident-preventive and mitigative features are referred to collectively as engineered safety features (ESF). To establish design and operating specifications for ESF, postulated events referred to as design-basis accidents are analyzed.

An emergency core cooling system (ECCS) is provided to supply cooling water to the reactor core during an accident to prevent or minimize fuel damage. Means of removing heat energy from the containment to mitigate its overpressurization following an accident are also provided.

The containment system itself is a passive ESF, designed to prevent direct escape of released fission products to the environment. The Limerick containment structures consist of an inner primary containment and an outer secondary containment. The primary containment is designed to withstand internal pressures resulting from reactor accidents. The secondary containment surrounds the primary containment and includes all equipment outside primary containment that could handle fission products in the event of an accident. The secondary containment is designed to collect, delay, and filter any leakage from the primary containment before its release to the environment for all events up to and including those of design basis severity, and for some events of greater severity.

The secondary containment encloses plant areas that are accessible and, therefore, ventilated during normal operation. When a release of radioactivity is detected, normal ventilation is automatically isolated, and two ESFs--standby gas treatment system (SGTS) and reactor enclosure recirculation system (RERS)--assume control of air flow within and from the secondary containment. The SGTS and RERS filter the secondary containment atmosphere and exhaust sufficient filtered air to establish and maintain an internal pressure less than the outside atmospheric pressure. This negative pressure is to be sufficient to prevent unfiltered air leakage from the building. Radioactive iodine and particulate fission products would be substantially removed from the SGTS and RERS flow by safety-grade activated charcoal and high-efficiency particulate air filters. A filtered exhaust system also encloses the spent fuel pool.

The main steamlines pass through the secondary containment in going from the reactor to the turbine building. Any leakage of the main steamline isolation valves, therefore, could pass through those lines without being intercepted by the SGTS and RERS. To prevent this passage, a leakage control system is designed to collect main steamline isolation valve leakage and direct it into the secondary containment atmosphere and sumps, so that any airborne emissions are processed by the SGTS and RERS.

All mechanical systems mentioned above are designed to perform their functions given single failures, are qualified for their anticipated accident environments, and are supplied with emergency power from onsite diesel generators if normal offsite and station power is interrupted.

Much more extensive discussion of these design features may be found in the applicant's FSAR and the staff's SER (NUREG-0991). In addition, the implementation of the lessons learned from the TMI-2 accident--in the form of improvements in design, procedures, and operator training--will significantly reduce the likelihood of a degraded core accident that could result in large releases of fission products to the containment. The applicant will be required to meet the TMI-related requirements specified in NUREG-0737. As noted in Section 5.9.4.5(7), the relative improvement in safety from these actions has not been quantified in this statement.

(2) <u>Site Features</u>

The NRC's reactor site criteria, 10 CFR 100, require that the site for every power reactor have certain characteristics that tend to reduce the risk and potential impact of accidents. The discussion that follows briefly describes the Limerick site characteristics and how they meet these requirements.

First, the site has an exclusion area, as required by 10 CFR 100. The total site area is about 241 ha (595 acres). The exclusion area, located within the site boundary, is a circular area with a minimum distance of 762 meters (2500 feet) from the center of Unit 1 and Unit 2 to the exclusion area boundary. There are no residents within the exclusion area. The applicant owns all surface and mineral rights in the exclusion area and has the authority, as required by 10 CFR 100, to determine all activities in this area. Several state-maintained roads traverse the area, allowing access to the plant and to the Schuylkill River. One railroad and the Schuylkill River traverse the exclusion area. The Schuylkill River, including that section within the exclusion area, is used for recreational activities such as boating and fishing. In the event of an emergency, the applicant has made arrangements with Pennsylvania State Police to control access to and activities on the Schuylkill River and the roads traversing the exclusion area. The applicant also has made arrangements with Conrail for authority to control activities on the railroad traversing the exclusion area.

Second, beyond and surrounding the exclusion area is a low population zone (LPZ), also required by 10 CFR 100. The LPZ for the Limerick site is a circular area with a 1.27-mile (2.04-km) radius. Within this zone, the applicant must ensure that there is a reasonable probability that appropriate protective measures could be taken on behalf of the residents in the event of a serious accident. The applicant has indicated that 1177 persons lived within a 1.27-mile (2.04-km) radius in 1980. The major source of seasonal transients within the same 1.27-mile (2.04-km) radius of the site are the patrons of the Countryside Swim Club, which is located 1.2 miles west-southwest. The 1980 industrial employee population within the LPZ was 87 persons.

In case of a radiological emergency, the applicant has made arrangements to carry out protective actions, including evacuation of personnel in the vicinity of the plant (see also the following section on emergency preparedness).

Third, 10 CFR 100 also requires that the distance from the reactor to the nearest boundary of a densely populated area containing more than about 25,000 residents be at least one and one-third times the distance from the reactor to the outer boundary of the LPZ. Because accidents of greater potential hazards than those commonly postulated are highly improbable, although conceivable, it was considered desirable to add the population center distance requirement in 10 CFR 100 to provide for protection against excessive doses to people in large centers. Pottstown borough, with a 1980 population of 22,729, located 1.7 miles northwest of the site, is the nearest population center. This population center distance is at least one and one-third times the LPZ distance. The population density within a 30-mile (48.2-km) radius of the site was 1215 people/mi^2 (3147 people/km^2) in 1980 and is projected to increase to about 1966 people/mi^2 (5092 people/km^2) by the year 2020.

The safety evaluation of the Limerick site has also included a review of potential external hazards, that is, activities offsite that might adversely affect the operation of the nuclear plant and cause an accident. The review encompassed nearby industrial and transportation facilities that might create explosive, fire, missile or toxic gas hazards. The risk to the Limerick station from such hazards has been found to be negligible. A more detailed discussion of the compliance with the Commission's siting criteria and the consideration of external hazards is in the Limerick SER (NUREG-0991).

(3) Emergency Preparedness

The emergency preparedness plans, including protective action measures for Limerick station and environs, are in an advanced, but not yet fully completed stage. In accordance with the provisions of 10 CFR 50.47, effective November 3, 1980, no operating license will be issued to the applicant unless a finding is made by the NRC that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Among the standards that must be met by these plants are provisions for two emergency planning zones (EPZs); a plume exposure pathway EPZ of about 10 miles (16 km) in radius and an ingestion exposure pathway EPZ of about 50 miles (80 km) in radius. Other standards include appropriate ranges of protective actions for each of these zones, provisions for dissemination to the public of basic emergency planning information, provisions for rapid notification of the public during a serious reactor emergency, and methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences in the EPZs of a radiological emergency condition.

NRC and the Federal Emergency Management Agency (FEMA) have agreed that FEMA will make a finding and determination as to the adequacy of state and local government emergency response plans. NRC will determine the adequacy of the applicant's Emergency Response Plans with respect to the standards listed in 10 CFR 50.47(b), the requirements of Appendix E to 10 CFR 50, and the guidance contained in NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980. After the above determinations by NRC and FEMA, the NRC will make a finding in the licensing process as to the state of preparedness. The NRC staff findings will be reported in a supplement to the SER. Although the presence of adequate and tested emergency plans cannot prevent an accident, it is the staff's judgment that such plans when implemented can mitigate the consequences to the public if an accident should occur.

5.9.4.5 Accident Risk and Impact Assessment

(1) Design-Basis Accidents

As a means of ensuring that certain features of the Limerick facility meet acceptable design and performance criteria, both the applicant and the staff have analyzed the potential consequences of a number of postulated accidents. Some of these could lead to significant releases of radioactive materials to the environment, and calculations have been performed to estimate the potential radiological consequences to persons off site. For each postulated initiating event, the potential radiological consequences cover a considerable range of values, depending upon the particular course taken by the accident and related conditions, including wind direction and weather prevalent during the accident.

In the Limerick safety analysis and evaluation, three categories of accidents have been considered by the applicant and the staff. These categories are based on probability of occurrence and include (1) incidents of moderate frequency (events that can reasonably be expected to occur during any year of operation); (2) infrequent accidents (events that might occur once during the lifetime of the plant); and (3) limiting faults (accidents not expected to occur but that have the potential for significant releases of radioactivity). The radiological consequences of incidents in the first category, also called anticipated operational occurrences, are discussed in Section 5.9.3. Some of the initiating events postulated in the second and third categories for the Limerick units are shown in Table 5.11b. These events are designated design-basis accidents in that specific design and operating features such as described in Section 5.9.4.4(1) are provided to limit their potential radiological consequences. Approximate radiation doses that might be received by a person at the

Table 5.11b Approximate doses during a 2-hour exposure at the exclusion area boundary*

Accidents and faults	Duration of release	Whole-body dose (rems)	Thyroid dose (rems)
INFREQUENT ACCIDENTS			
Category 2			
Fuel-handling accident	<2 hours	0.5	1
LIMITING FAULTS			
Category 3			
Main steamline break	<2 hours	1	80
Control rod drop	hours-days	0.1	0.7
Large-break LOCA	hours-days	5	300

*2500 feet (762 m) from centers of Unit 1 or 2. All numbers have been rounded to one significant digit.

exclusion area boundary are also shown in the table, along with a characterization of the duration of the releases. The results shown in the table reflect a conservative estimate of the potential upper bound of individual radiation exposures from the initiating accidents in Table 5.11b for the purpose of implementing the provisions of 10 CFR 100 and are reported in the staff's Safety Evaluation Report (SER, NUREG-0991). For these calculations, pessimistic (conservative) assumptions are made as to the course taken by the accident and the prevailing conditions. These assumptions include conservatively large amounts of radioactive material released by the initiating events, additional single failures in equipment, operation of ESFs in a degraded mode,* and very poor meteorological dispersion conditions. The results of these calculations show that radioiodine releases have the potential for offsite exposures ranging up to about 300 rems to the thyroid. For such an exposure to occur, an individual would have to be located at a point on the site boundary where the radioiodine concentration in the plume has its highest value and inhale at a breathing rate characteristic of jogging for a period of 2 hours during very poor atmospheric dispersion conditions. The health risk to an individual receiving such a thyroid exposure is the potential appearance of benign or malignant thyroid nodules in about 1 out of 10 cases, and the development of a fatal cancer in about 4 out of 1000 cases.

The staff experience has been that realistic dose estimates for a spectrum of accidents up to and including those as severe as design-basis accidents would result in values considerably lower than the design-basis accidents established for the purpose of implementing the provisions of 10 CFR Parts 50 and 100 as reviewed in the staff's SER.

None of the calculations of the impacts of design-basis accidents described in this section take into consideration possible reductions in individual or population exposures as a result of any protective actions.

(2) Probabilistic Assessment of Severe Accidents

In this and the following three sections, there is a discussion of the probabilities and consequences of accidents of greater severity than the design-basis accidents discussed in the previous section. As a class, they are considered less likely to occur, but their consequences could be more severe for both the plant itself and for the environment. These severe accidents (heretofore frequently called Class 9 accidents) can be distinguished from design-basis accidents in two primary respects: they all involve substantial physical deterioration of the fuel in the reactor core to the point of melting, and they involve deterioration of the capability of the containment structure to perform its intended function of limiting the release of radioactive materials to the environment. It should be understood that even the very severe reactor accidents, unlike weapons, would not result in blast and in high pressure- and high temperature-related consequences to the offsite public or to the environment.

The assessment methodology employed is essentially as described in the reactor safety study (RSS, WASH-1400) which was published in 1975 (NUREG-75/014), but includes improvements in the assessment methodology that were developed after

*The containment system, however, is assumed to prevent leakage in excess of that which can be demonstrated by testing, as provided in 10 CFR 100.11(a).

publication of the RSS* (such as better thermal-hydraulic models, more precise core melt phenomenology and containment response analysis). The assessment is also plant and site specific.

In the Limerick Environmental Report--Operating License stage (ER-OL) Revision 12, April 1983, the applicant has presented a plant- and site-specific probablistic assessment of severe accidents, including the effects of external events such as fires and earthquakes. The details of the applicant's analysis are contained in a supporting document, "Limerick Generating Station Severe Accident Risk Analysis (LGS-SARA)," which also includes information from the applicant's earlier submittal "Limerick Generating Station Probabilistic Risk Assessment (LGS-PRA)." As a direct result of the applicant's efforts in performing the probabilistic assessment, several risk reduction modifications to the plant design were implemented during its construction. These modifications have been reviewed by the staff and are incorporated into the staff's analysis. The NRC staff contracted with the Brookhaven National Laboratory (BNL) to review portions of the LGS-SARA. The results of BNL's review of LGS-PRA is reported in NUREG/CR-3028, and that of the earthquake and fire hazards from the SARA is summarized in the draft report attached to the staff's letter to the applicant dated August 31, 1983. By letter dated March 13, 1984 the applicant informed the staff that errors in the LGS-SARA consequence analysis had been discovered. The staff has determined that correction of the applicant's errors will not change the conclusions contained herein. The results of an independent staff analysis of severe accidents are summarized below. Neither the applicant's analysis nor the staff's analysis includes the potential effects of sabotage; such an analysis is considered to be beyond the state of the art of probabilistic risk assessment. However, the staff judges that the additional risks from severe accidents initiated by sabotage are within the uncertainties of risks presented for the severe accidents considered here.

Accident sequences initiated by both internal and external causes that are used in the staff analysis are described in Appendix H to this report, based on information provided by BNL. Accident sequences are grouped into "release categories" based upon similarities of the sequences regarding core-melt accident progression, containment failure characteristics, and the parameters of atmospheric release of radionuclides required for consequence analysis.

Included in the list of potential accident initiators that are called external events are fires and earthquakes. The staff concurs with the SARA findings that the hazards due to other external events such as floods, tornadoes, transportation accidents, industrial accidents, and turbine missiles do not contribute significantly to the risk from severe accidents.

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*However, there are large uncertainties in the assessment methodology and the results derived from its application. A discussion of the uncertainties is provided in section 5.9.4.5(7). Large uncertainties in event frequencies and other areas of risk analysis arise, in part, from similar causes in all plant and site assessments; hence the results are better used in carefully constructed comparisons rather than as absolute values. External event frequencies used here are, however, more representative of the Limerick site than those used in the RSS.

Table 5.11c provides information used in the staff's consequence assessment for each specific release category and summarizes the BNL analysis described in Appendix H. The information includes time estimates from termination of the fission process during the accident until the beginning of release to the environment (release time), duration of the atmospheric release, warning time for offsite evacuation, and estimates of the energy associated with the release, height of the release location above the ground level, and fractions of the core inventory (see Table 5.11a) of seven groups of radionuclides in the release. The radionuclide release fractions shown in Table 5.11c were derived using WASH-1400 radiochemistry assumptions of fission product releases from fuel and their attenuation through various elements of the primary system and containment such as the suppression pool and aerosol transport in the containment building as described in Appendix H. The number in parentheses following the designation of each release category in Table 5.11c indicates its relative rank in terms of the magnitude of the core-fraction of cesium estimated to be in the release. Cesium was chosen because of its biological significance.

The BNL-calculated mean value (i.e., the point estimate or the best estimate) of probability associated with each release category used in the staff analysis, is shown in Table 5-11d (see Appendix H and Section 5.9.4.5(7)). In this table, the probability of each accident sequence or release category is shown in two separate parts based on the cause of the accident. One contribution to the probability is ascribed to the accident-initiating events that include plant internal causes, fires, and earthquakes of low to medium severity (effective peak ground acceleration less than 0.4 g; that is, Modified Mercalli (MM) intensity scale VIII or lower) (see Appendix H). In Table 5.11c of the DES supplement release fractions for four release categories were found to be in error (IV-T/DW, IV-T/WW, IV-T/WW and IV-A/DW) and these have been corrected. The second contribution to the probability is ascribed to very severe regional earthquakes (effective peak ground acceleration equal to or greater than 0.4 g; that is, MM intensity scale IX or higher) (see Appendix H) as potential cause of reactor accidents, which would also alter offsite conditions adversely to seriously hamper emergency responses that would mitigate the consequences of such accidents. (Appendix I provides a description of potential offsite damages from earthquakes of various intensities.) As in the RSS, there are substantial uncertainties in these probabilities. This is due, in part, to difficulties associated with the quantification of human error and to inadequacies (1) in the data base on failure rates of individual plant components (NUREG/CR-0400), and (2) in the data base on external events and their effects on plant systems and components that are used to calculate the probabilities.

Analyses of risks have indicated that reactor accidents having mean likelihoods of less than 10^{-9} per reactor-year (i.e., less than once in a billion reactor years), even considering the uncertainties of such estimates, are unlikely to contribute substantially to estimated risks. For this reason, and because of the low probabilities of occurrence of these accidents, the staff has omitted from any further discussion the Table 5-11c accidents and release categories for which the mean probability in Table 5-11d is estimated to be less than 10^{-9} per reactor-year.

The magnitudes (curies) of radioactivity release to the atmosphere for each accident sequence or release category are obtained by multiplying the release

Table 5.11c Summary of the atmospheric release specifications used in consequence analysis for Limerick Units 1 and 2[a]

Release category[b]	Release time (hr)	Release duration (hr)	Warning time for evacuation (hr)	Energy release (10^6 Btu/hr)	Release height (m)	Fractions of Core Inventory Released: Xe-Kr	Organic I[c]	Inorganic I	Cs-Rb	Te-Sb	Ba-Sr	Ru[d]	La[e]
I-T/DW(22)*	5	0.5	4	100	30	1	7(-3)**	2(-3)	2(-2)	8(-2)	1(-3)	5(-3)	1(-3)
I-T/WW(25)	5	0.5	4	100	30	1	7(-3)	1(-4)	3(-4)	1(-3)	2(-5)	7(-5)	1(-5)
I-T/$\overline{WW}$(24)	5	0.5	4	100	30	1	7(-3)	2(-4)	9(-4)	2(-3)	8(-5)	1(-4)	3(-5)
I-T/SE(14)	2	0.5	1	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
I-T/HB(20)	2	0.5	1	100	30	1	--	2(-1)	6(-2)	1(-1)	7(-3)	8(-2)	1(-5)
I-T/LGT(26)***	2	3	0	1	30	0.7	--	3(-3)	1(-4)	5(-4)	2(-5)	3(-5)	6(-6)
I-T/$\overline{LGT}$(18)	2	3	0	1	30	0.7	--	2(-2)	1(-1)	5(-2)	2(-3)	3(-3)	6(-4)
II-T/WW(8)	20	4	5	1	30	1	7(-3)	7(-1)	3(-1)	2(-1)	4(-2)	4(-2)	3(-3)
II-T/SE(14)	30	0.5	7	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
III-T/WW(10)	3	1	2	100	30	1	7(-3)	8(-2)	2(-1)	6(-1)	2(-2)	4(-2)	7(-3)
III-T/SE(5)	2	0.5	1	100	30	1	--	4(-1)	5(-1)	5(-1)	5(-2)	5(-1)	3(-3)
III-T/HB(20)	2	0.5	1	100	30	1	--	2(-1)	6(-2)	1(-1)	7(-3)	8(-2)	1(-5)
III-T/LGT(26)	0.5	4	0	1	30	0.7	--	3(-3)	1(-4)	5(-4)	2(-5)	3(-5)	6(-6)
III-T/$\overline{LGT}$(18)	0.5	4	0	1	30	0.7	--	2(-2)	1(-1)	5(-2)	2(-3)	3(-3)	6(-4)
IV-T/DW(2)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	9(-2)	7(-3)
IV-T/WW(4)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	8(-2)	6(-3)
IV-T/$\overline{WW}$(3)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	9(-2)	7(-3)
IV-T/SE(5)	2	0.5	2	100	30	1	--	4(-1)	4(-1)	5(-1)	5(-2)	5(-1)	3(-3)
I-S/DW(23)	5	0.5	4	100	30	1	7(-3)	3(-3)	5(-3)	3(-3)	6(-4)	3(-4)	4(-4)
IV-A/DW(1)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	9(-2)	7(-3)
IS-C/DW(13)	0	3	0.4	1	30	1	7(-3)	8(-2)	1(-1)	6(-1)	7(-3)	8(-2)	7(-3)
IS-C/SE(14)	1	0.5	1	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
IS-$\overline{C}$/DW(12)	1	3	1	1	30	1	7(-3)	8(-2)	1(-1)	6(-1)	8(-3)	1(-1)	7(-3)
IS-$\overline{C}$/SE(14)	2	0.5	2	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
S-H20/$\overline{WW}$(11)	3	5	3	1	30	1	7(-3)	1(-1)	2(-1)	3(-1)	1(-2)	5(-2)	4(-3)
S-H20/SE(5)	4	0.5	4	100	30	1	--	4(-1)	4(-1)	5(-1)	5(-2)	5(-1)	3(-3)
S-$\overline{H20}$/$\overline{WW}$(9)	3	4	3	1	30	1	7(-3)	3(-1)	3(-1)	4(-1)	3(-2)	6(-2)	5(-3)

[a]See Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

[b]See Appendix H for designations and descriptions of the release categories.

[c]Organic iodine is added to inorganic iodine for consequence calculations because organic iodine is likely to be converted to inorganic or particulate forms during environmental transport.

[d]Includes Ru, Rh, Co, Mo, Tc.

[e]Includes Y, La, Zr, Nb, Ce, Pr, Nd, NP, Pu, Am, Cm.

*Number in parentheses indicates relative ranking of the release category according to cesium fraction.

**7(-3) = 7 x 10^{-3} = 0.007.

***This release category is combined with III-T/LGT in consequence analysis.

Table 5.11d Summary of the calculated mean (point estimate) probabilities of atmospheric release categories

Release category	Probability of the release category initiated by internal causes, fires, and low to moderately severe earthquakes (per reactor-year)	Probability of the release category initiated by severe earthquakes (per reactor-year)
I-T/DW	2(-5)*	6(-7)
I-T/WW	2(-5)	5(-7)
I-T/$\overline{\mathrm{WW}}$	2(-6)	6(-8)
I-T/SE	8(-9)	2(-10)***
I-T/HB	8(-7)	2(-8)
I-T/LGT**	2(-5)	5(-7)
I-T/$\overline{\mathrm{LGT}}$	2(-5)	6(-7)
II-T/WW	2(-6)	2(-8)
II-T/SE	4(-10)***	4(-10)***
III-T/WW	2(-6)	4(-7)
III-T/SE	3(-10)***	7(-11)***
III-T/HB	3(-8)	7(-9)
III-T/LGT	7(-7)	2(-7)
III-T/$\overline{\mathrm{LGT}}$	9(-7)	2(-7)
IV-T/DW	2(-7)	5(-8)
IV-T/WW	2(-7)	4(-8)
IV-T/$\overline{\mathrm{WW}}$	2(-8)	5(-9)
IV-T/SE	3(-11)***	1(-11)***
I-S/DW	4(-8)	0
IV-A/DW	5(-9)	0
IS-C/DW	1(-8)	1(-7)
IS-C/SE	1(-12)***	1(-11)***
IS-$\overline{\mathrm{C}}$/DW	1(-7)	9(-7)
IS-$\overline{\mathrm{C}}$/SE	1(-11)***	9(-11)***
S-H20/$\overline{\mathrm{WW}}$	1(-8)	4(-8)
S-H20/SE	1(-12)***	4(-12)***
S-$\overline{\mathrm{H20}}$/$\overline{\mathrm{WW}}$	1(-8)	4(-7)
Total probability per reactor-year	9(-5)	5(-6)

*2(-5) = 2×10^{-5} = .00002

**This release category is combined with III-T/LGT in consequence analysis.

***Any release category with probability less than 10^{-9} per reactor-year is omitted from consequence analysis because of its low probability and insignificant contribution to risks.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

fractions shown in Table 5-11c by the amounts that would be present in the core at the time of the hypothetical accident and by depletion factors as a result of inplant radioactive decay during the release time. The core inventory of radionuclides are shown in Table 5.11a for Units 1 and 2 at a core thermal power level of 3458 MWt. This is the power level used in the FSAR for analysis of radiological consequences and is used here instead of the 3293 MWt expected maximum power to correct for power density variations and instrument error in measurement of power levels normally present in operating reactors. The 54 nuclides shown in the table represent those (of the hundreds actually expected to be present in the operating plant) that are potentially major contributors to the health and economic effects of severe accidents. They were selected on the basis of the half-life of the nuclide, consideration of the health effects of daughter products, and the approximate relative offsite dose contribution.

The potential radiological consequences of these releases have been calculated by the computer code CRAC, based on the consequence model used in the RSS (see NUREG-0340), adapted and modified as described below to apply to a specific site. The essential elements are shown in schematic form in Figure 5.4a. Environmental parameters specific to the site of Limerick station have been used and include

(1) meteorological data for the site representing a full year (1976) of consecutive hourly measurements and seasonal variations with good data recovery characteristics (annual average probabilities of wind blowing into 16 directions of the compass are shown in Table 5.11e)

(2) projected population for the year 2000 extending throughout regions of 80-km (50-mile) and 563-km (350-mile) radius from the site

(3) the habitable land fraction within a 563-km (350-mile) radius

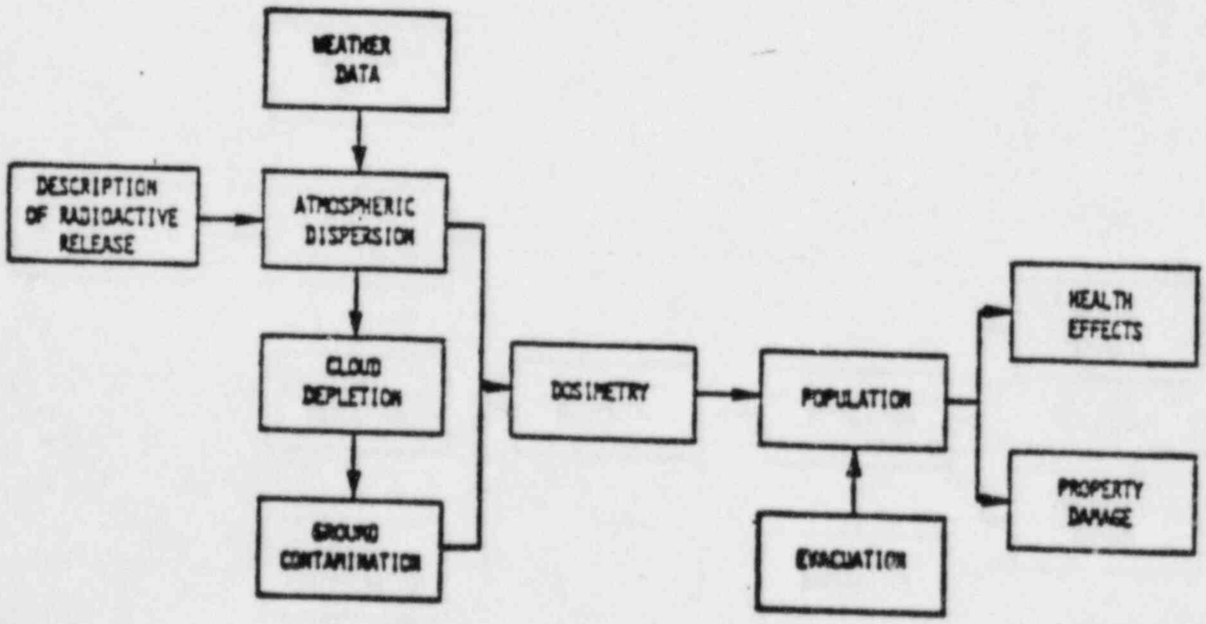

Figure 5.4a Schematic outline of consequence model

Table 5.11e Annual average wind-direction probabilities for the Limerick site based on data for the year 1976

Wind blowing toward the direction	Probability (fraction of the year)
N	0.07
NNE	0.07
NE	0.06
ENE	0.05
E	0.10
ESE	0.16
SE	0.11
SSE	0.04
S	0.04
SSW	0.03
SW	0.03
WSW	0.04
W	0.07
WNW	0.03
NW	0.04
NNW	0.06
Total	1.00

(4) land-use statistics on a countywide basis within and statewide basis outside of a 80-km (50-mile) region, including farm land values, farm product values including dairy production, and growing season information, for the counties, the State of Pennsylvania and each surrounding state within the 563-km (350-mile) region

For the region beyond 563 km (350 miles), the U.S. average population density was assumed.

The calculation was extended out to 3200 km (2000 miles) from the site, to account for the residual radionuclides that would remain in the atmosphere at large distances, with rain assumed in the interval between 563 km and 3200 km to deplete the plume of all non-noble-gas inventory. To obtain a probability distribution of consequences, calculations were performed assuming the occurrence of each release category at each of 91 different "start" times distributed throughout a 1-year period. Each calculation utilized site-specific hourly meteorological data and seasonal information for the period following each "start" time.

The consequence model was also used to evaluate the consequence reduction benefits of offsite emergency response such as evacuation, relocation, and other protective actions. Early evacuation and relocation of people would considerably reduce the exposure from the radioactive cloud and the contaminated ground in the wake of the cloud passage. The evacuation model used (see Appendix J) has been revised from that used in the RSS for better site-specific application. In the staff calculation, three sets of assumptions were made about the short-term emergency response that would likely be undertaken to minimize the severe accident health effects from early or short-term radiological exposure. Table 5.11f lists the assumptions and parameters for each emergency response scenario evaluated.

The first set of parameters assumes evacuation of the population within 10 miles (16 km). The effective evacuation speed in Table 5.11f is based on an evaluation made by the applicant's contractor, NUS Corporation, in an evacuation time estimate study (NUS, 1980). The estimate of the delay time before evacuation in the same study has been rejected by the applicant in LGS-SARA and, therefore, is not used in the staff analysis. Instead, the value of delay time in Table 5.11f is a staff assumption and is based partly on considerations of the NRC requirement regarding prompt notification of the public of the emergency, and partly on the staff judgment regarding the time people would take preparing for evacuation after being notified of the emergency, for a high population density site, during normal to moderately adverse conditions such as snow, ice, hurricane, low to moderately severe earthquakes (up through MM intensity scale VIII), etc. The values of delay time before evacuation and effective evacuation speed used in the staff analysis are assumed only to be average values. Within the 10-mile emergency planning zone there normally would be some facilities (such as nursing homes, hospitals, prisons, schools, etc.) where special equipment or personnel may be required to effect evacuation, and there may be some people who choose not to evacuate. Therefore, actual effectiveness could be greater or less than that characterized by the average values. Because special consideration will be given in emergency planning for Limerick to any unique aspects of dealing with special facilities, it is not expected that actual evacuation effectiveness would be very much less than that modeled by the average values used here. For areas beyond 10 miles (16 km), however, the parameters selected reflect the assumptions that an extension of emergency response would occur during a large accident and people would be advised to leave areas that would be considered to be highly contaminated (see below for criterion), i.e., people would relocate. Relocation of the public from the highly contaminated areas beyond 10 miles (16 km) is assumed to take place 12 hours after plume passage. The criterion for this relocation is whether the projected 7-day ground dose to the total bone marrow, as projected by field measurements, would exceed 200 rems (which is only slightly above the average threshold exposure for potential early fatality with minimal medical treatment); otherwise people in highly contaminated areas are assumed to be relocated within 7 days. The offsite emergency response mode characterized by these assumptions is designated Evac-Reloc.

The second set of parameters reflects the hypothesis that the planned evacuation may not take place in a real situation for one or more reasons such as short warning time, indecision regarding whether to evacuate or not because of uncertain plant conditions, or adverse site conditions that would cause long delay before evacuation. In lieu of evacuation, it was assumed that people in the footprint of the plume within 10 miles (16 km) would leave the area (i.e., relocate) 6 hours after plume passage. This 6-hour relocation time is similar to the time for evacuation assumed in the first set based on 2 hours delay and about 2.5 miles per hour evacuation speed. Beyond 10 miles (16 km), relocation

Table 5.11f Emergency response assumptions for each reactor unit

Emergency response set no.*	Evacuation distance (mi)**	Delay time (hr)	Effective evacuation speed (mph)	Effective downwind distance moved*** (mi)	Relocation zone size (mi) Zone A†	Relocation zone size (mi) Zone B†	Zone B ~~relocation~~ relocation time (hr)	Zone B relocation dose criterion (bone marrow dose projected for 7 days) (rems)	Shielding protection factor (fraction): During evacuation, plume/ground	Shielding protection factor (fraction): Other times, plume/ground
1	10	2	2.5	15	0	>10	12	200	1¶/0.5¶	0.75¶¶/0.33¶¶
2	N/A††	N/A	N/A	N/A	10†††	>10	12	200	N/A	0.75¶¶/0.33¶¶
3	N/A	N/A	N/A	N/A	0	>0	24	200	N/A	1.0¶¶¶/0.5¶¶¶

*Sets 1, 2, and 3 are also identified as Evac-Reloc, Early Reloc, and Late Reloc, respectively, in text, tables, and figures.

**To change miles to km, multiply the values shown by 1.609.

***An artificial parameter used only to represent a realistic path-length for each evacuee over which radiation exposure to the evacuee is calculated in the CRAC code.

†Zone A is the 10-mile plume exposure pathway emergency planning zone; Zone B is the area outside Zone A.

††N/A - Not Applicable.

†††Relocation takes place 6 hours after ground contamination.

¶During evacuation, automobiles are assumed to provide essentially no shielding to gamma rays from the plume and some shielding to gamma rays from the contaminated ground. The selected values of shielding protection factors for the plume and the ground during evacuation are taken from Table VI 11-13 of Appendix VI of WASH-1400.

¶¶At other times than during evaucation, shielding protection factors are the average values representative of normal activities of the people during which some people are indoors and some are outdoors. The selected values of the shielding protection factors for the plume and the ground for this situation are taken from Table VI 11-13 of Appendix VI of WASH-1400.

¶¶¶During an abnormal situation in the site region caused by a external event such as a severe earthquake, it is assumed that many of the buildings may not remain habitable to provide shielding protection to the people against gamma rays from the plume. So, the shielding factor for the plume is taken to be 1. However, the nature of the ground surface is assumed to become altered by debris and possibly mud/slush/water generated from a severe earthquake. So, the ground shielding factor (provided by the altered ground and whatever building structures that would still have remained intact) of 0.5 was selected for this scenario, which is about midway between the values 0.33 for normal situation and 0.7 for an ordinary and uncovered ground surface.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties.

was assumed as in the previous set of assumptions. The offsite emergency response mode characterized by these assumptions is designated Early Reloc and was used for an alternative risk analysis.

The third set of parameters reflects a radiological emergency response situation hampered by a severe type of external event, such as a severe regional earthquake, which would seriously limit the ability to evacuate, and would also eliminate or reduce the shielding protection that the public would otherwise experience. However, relocation of the public from highly contaminated areas 24 hours after plume passage was assumed. The criterion for this relocation was the same as in the first set of assumptions, but relocation was assumed to extend outward from the site exclusion area boundary (762 meters, as opposed to the 10-mile (16-km) EPZ boundary); otherwise people are assumed to be relocated within 7 days. The offsite emergency response mode characterized by this third set of assumptions is designated Late Reloc.

The environmental protective actions considered as part of relatively long-term offsite emergency response to reduce health effects from chronic exposure include: (1) either complete denial of use (interdiction), or permitting use only at a later time after appropriate decontamination, of food stuffs such as crops and milk; (2) decontamination of severely contaminated land and property when it is considered to be economically feasible to lower the levels of contamination to protective action guide (PAG) levels*; and (3) denial of use (interdiction) of severely contaminated land and property for varying periods of time until the contamination levels are reduced by radioactive decay and weathering to such values that land and property can be economically decontaminated as in (2) above. These actions would reduce radiological exposures and health effects to the people from immediate and/or subsequent use of or living in the contaminated environment, but would also result in economic costs to implement them. Lowering the PAG levels would lower the delayed health effects but would increase costs.

Estimates of meteorology-averaged societal consequences of several types conditional upon occurrence of each release category in Table 5.11c are tabulated in Appendix K. For each release category, separate estimates are provided using each of the offsite emergency response modes in Table 5.11f. These conditional mean values are of use only in judging the relative severity of each release category and they cannot be used directly for risk assessment without simultaneous association with the probability of the release category to which the consequences are due. Therefore, in the following paragraphs, the impacts of severe accidents in the Limerick reactors are appropriately weighted by their probabilities.

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*PAG levels used in CRAC analyses are not to be confused with those drafted by the U.S. Environmental Protection Agency (EPA-520/1-75-001, September 1975), or by the U.S. Department of Health and Human Services (47 FR 47073, October 22, 1982), for reactor accidents. PAG levels used in CRAC are defined in Table VI 11-6 of WASH-1400, and were based on the recommendations of the former U.S. Federal Radiation Council and the British Medical Research Council. However, for control of long-term external irradiation, the PAG level for urban areas in WASH-1400 Table VI.11-6 was used in CRAC for all areas (urban and rural).

The consequences and risks* of severe accidents in the Limerick reactors initiated by plant internal causes, fires, and low to moderately severe earthquakes were evaluated using the release categories in Table 5.11c, the corresponding probabilities in Table 5.11d, and the parameters of the Evac-Reloc mode of offsite emergency response in Table 5.11f. The consequences and risks of accidents initiated by very severe regional earthquakes that could also affect the offsite conditions so as to seriously hamper evacuation or early relocation were evaluated using the accident parameters in Table 5.11c, the corresponding probabilities in Table 5.11d, and the parameters of the Late Reloc mode of offsite emergency response in Table 5.11f. Finally, the overall evaluation of consequences and risks of reactor accidents at Limerick from internal causes, fires, and low to high severity earthquakes is made by combining the results for Evac-Reloc and Late Reloc offsite emergency response modes.

The results of the staff calculations using the consequence model are radiological doses to individuals and to populations, health effects that might result from these exposures, costs of implementing protective actions and costs associated with property damage by radioactive contamination, and land area that would be subject to long-term interdiction. These results are presented and discussed below. Breakdowns for each type of consequence in terms of contributions from accidents initiated by severe earthquakes and from accidents initiated by other causes considered in the analysis are presented in Appendix L.

An alternative overall evaluation of consequences and risk in which the Evac-Reloc mode of offsite emergency response is replaced by the Early Reloc mode is presented in Appendix M. The staff critique of the principal aspects of the applicant's consequence analysis in the Environmental Report-Operating License stage (ER-OL), which is identified to be the same as in LGS-SARA, is provided in Appendix N.

There are large uncertainties in each facet of the estimates of consequences both in the staff analysis and the applicant's analysis (see Section 5.9.4.5(7)).

(3) Dose and Health Impacts of Atmospheric Releases

The results of the staff calculations of the environmental dispersion of radioactive releases to the atmosphere and the radiological dose to people and health impacts performed for the Limerick station and site are presented in the form of probability distributions in Figures 5.4b through 5.4f and are included in the impact summary Table 5.11g. The graphs in Figures 5.4b through 5.4f (and in similar Figures 5.4g and 5.4h introduced later) display a type of probability distribution called a complementary cummulative distribution function (CCDF). CCDFs are intended to show the relationship between the probability of a particular type of consequence being equaled or exceeded and the magnitude of the consequence. These graphs are useful in visualizing the degree to which the probability of occurrence of consequences decreases as the magnitude of the consequence increases. Probability per reactor-year** is the chance that a given event would occur or a given consequence magnitude would be exceeded in 1 year ✗

*Risk of a particular kind of consequence is to be understood as the average value of several estimates of the product of magnitude of the particular consequence and its associated probability.

** See *, p. 5-91.

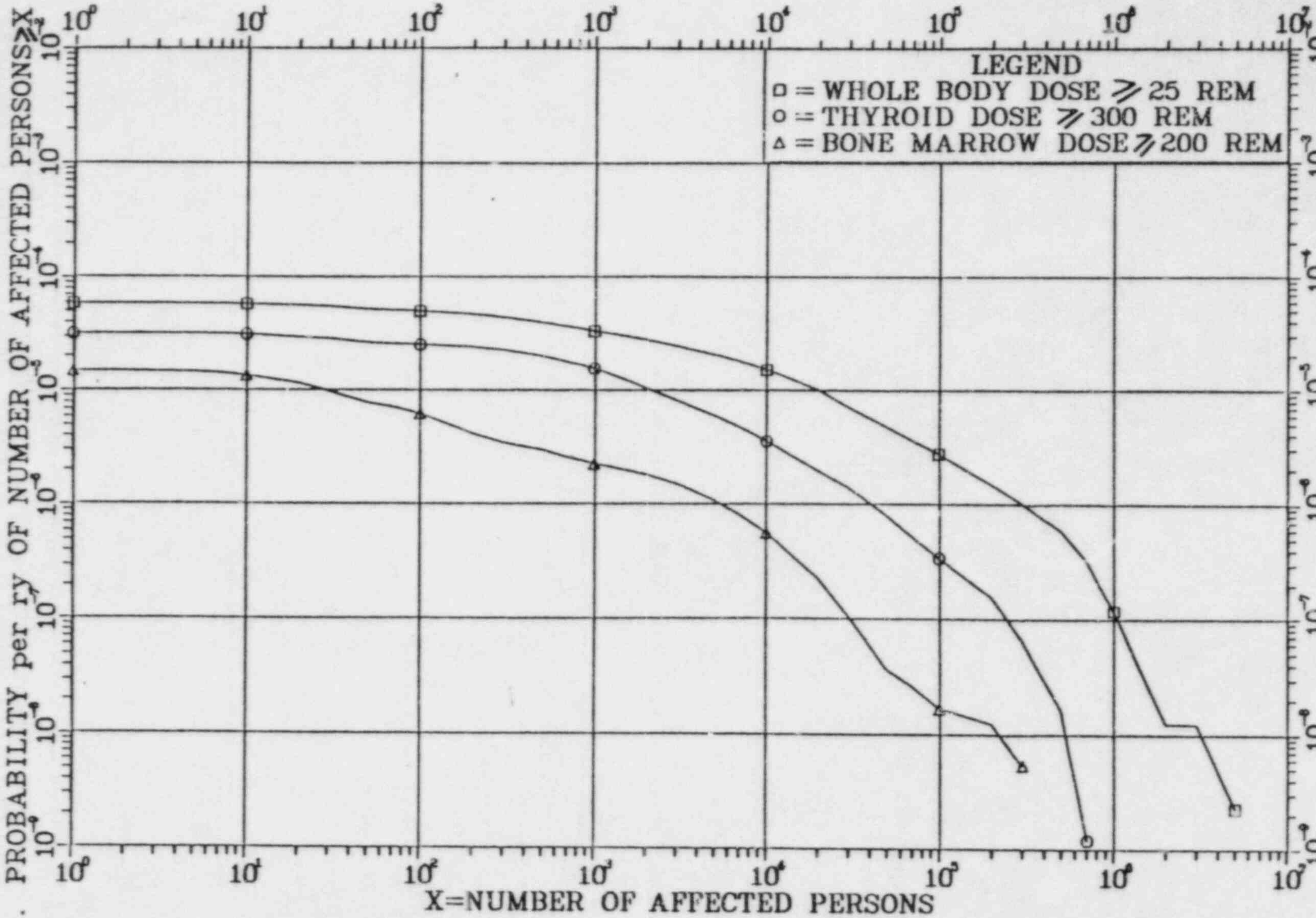

Figure 5.4b Probability distributions of individual dose impacts

NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

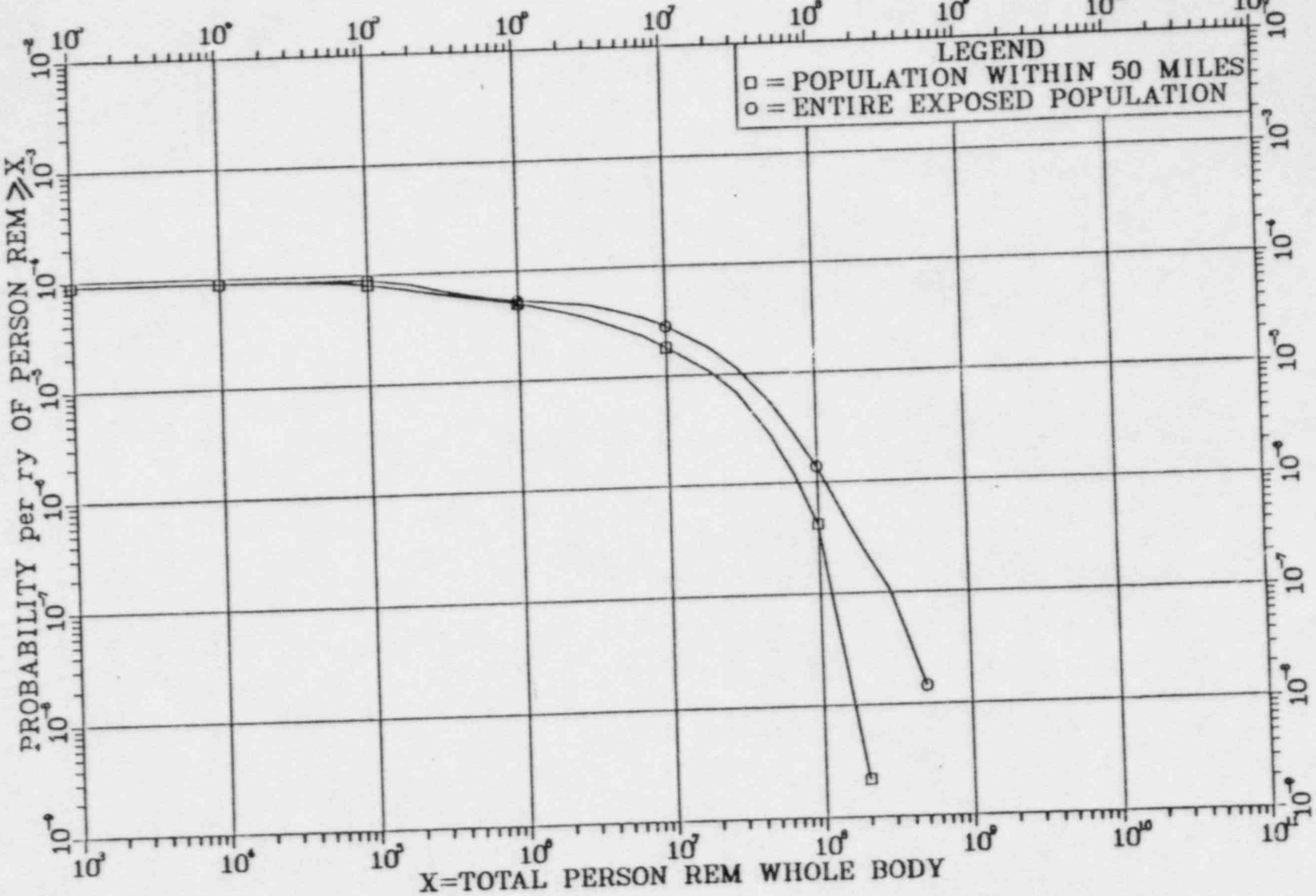

Figure 5.4c Probability distributions of population exposures

NOTES: 1. The average annual dose to the population within 50 miles resulting from natural background radiation is about 800,000 person-rems.

2. See Section 5.9.4.5(7) for a discussion of uncertainties.

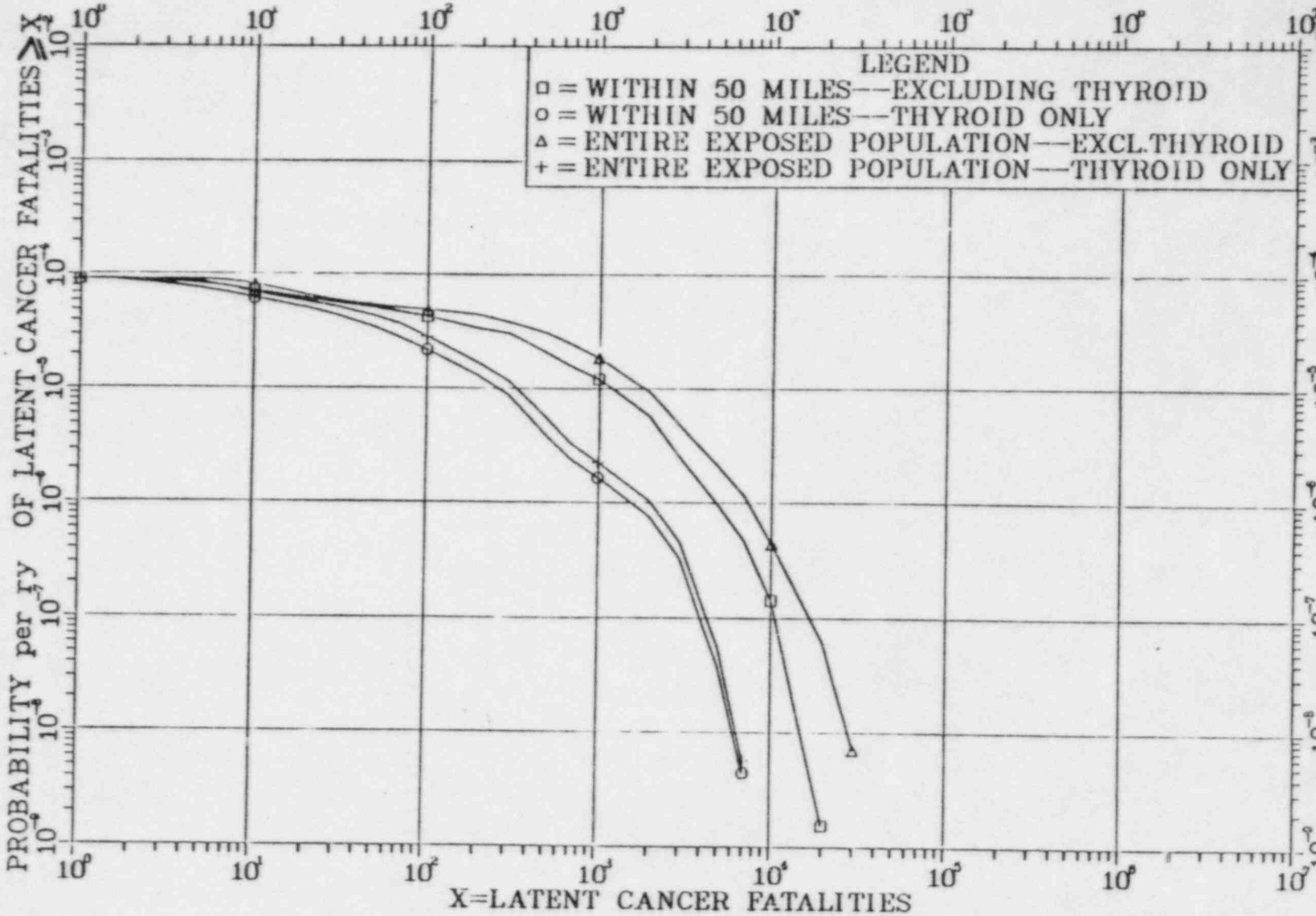

Figure 5.4d Probability distributions of cancer fatalities

NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

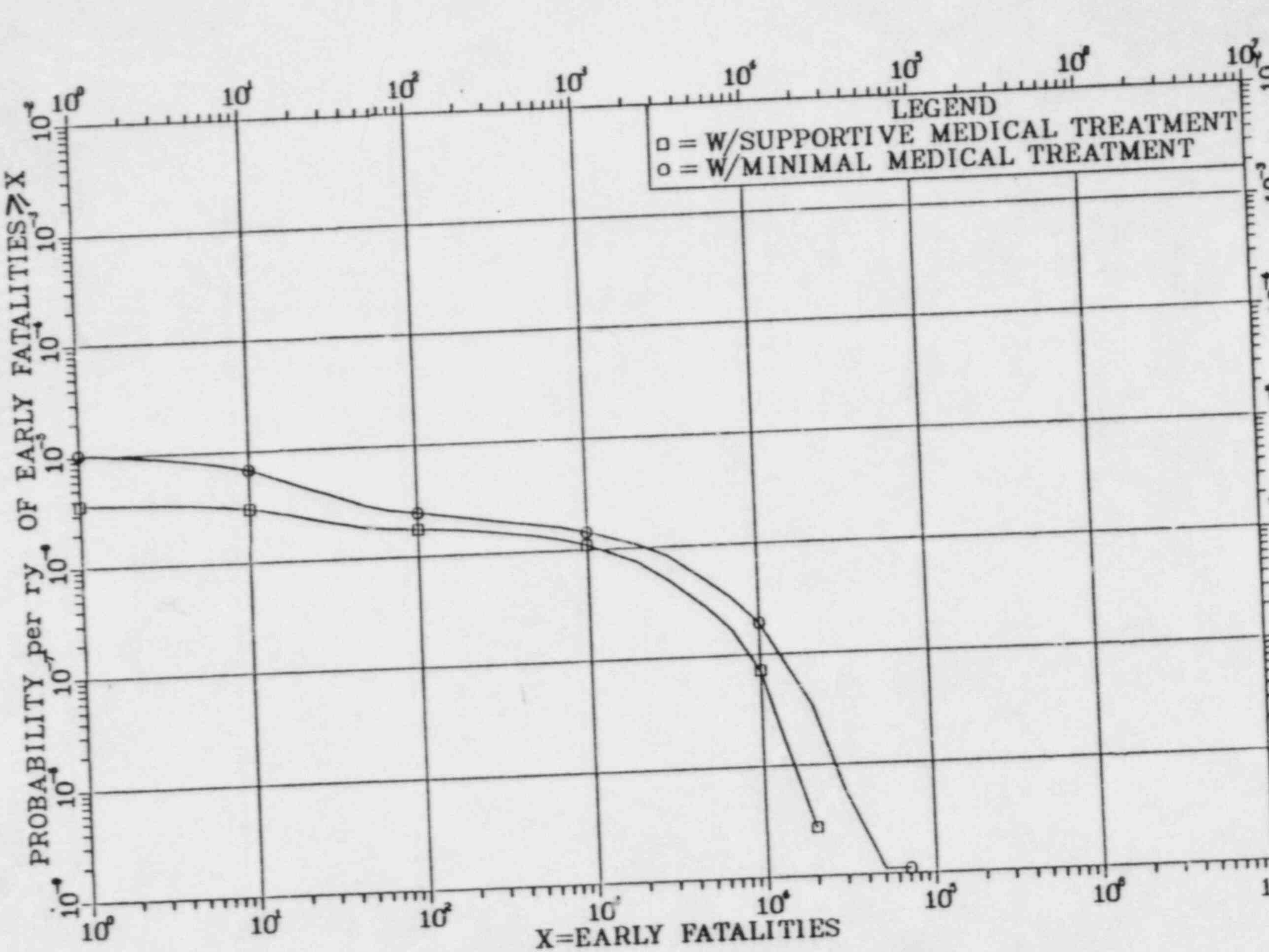

Figure 5.4e Probability distribution of early fatalities

NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

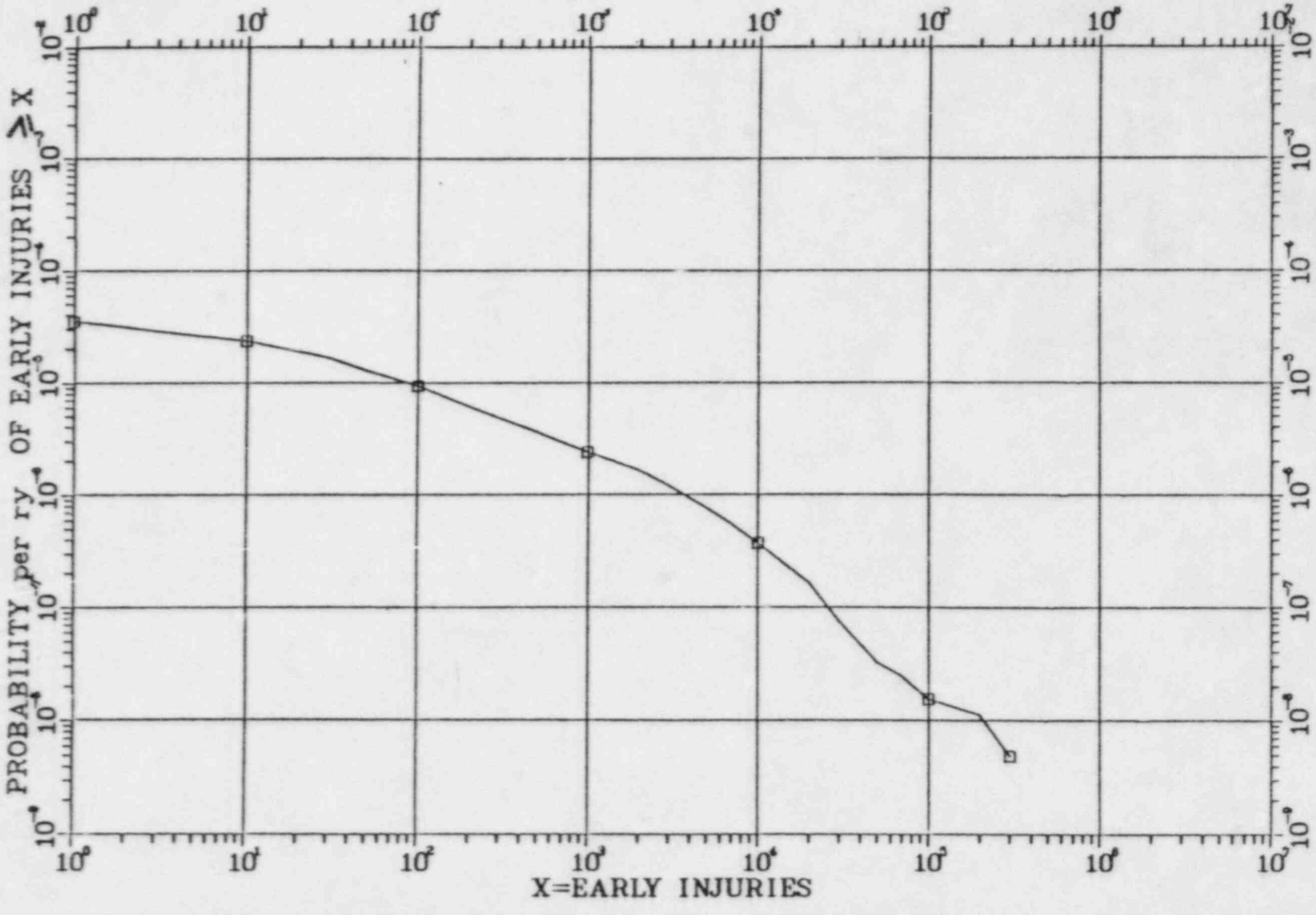

Figure 5.4f Probability distribution of early injuries

NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

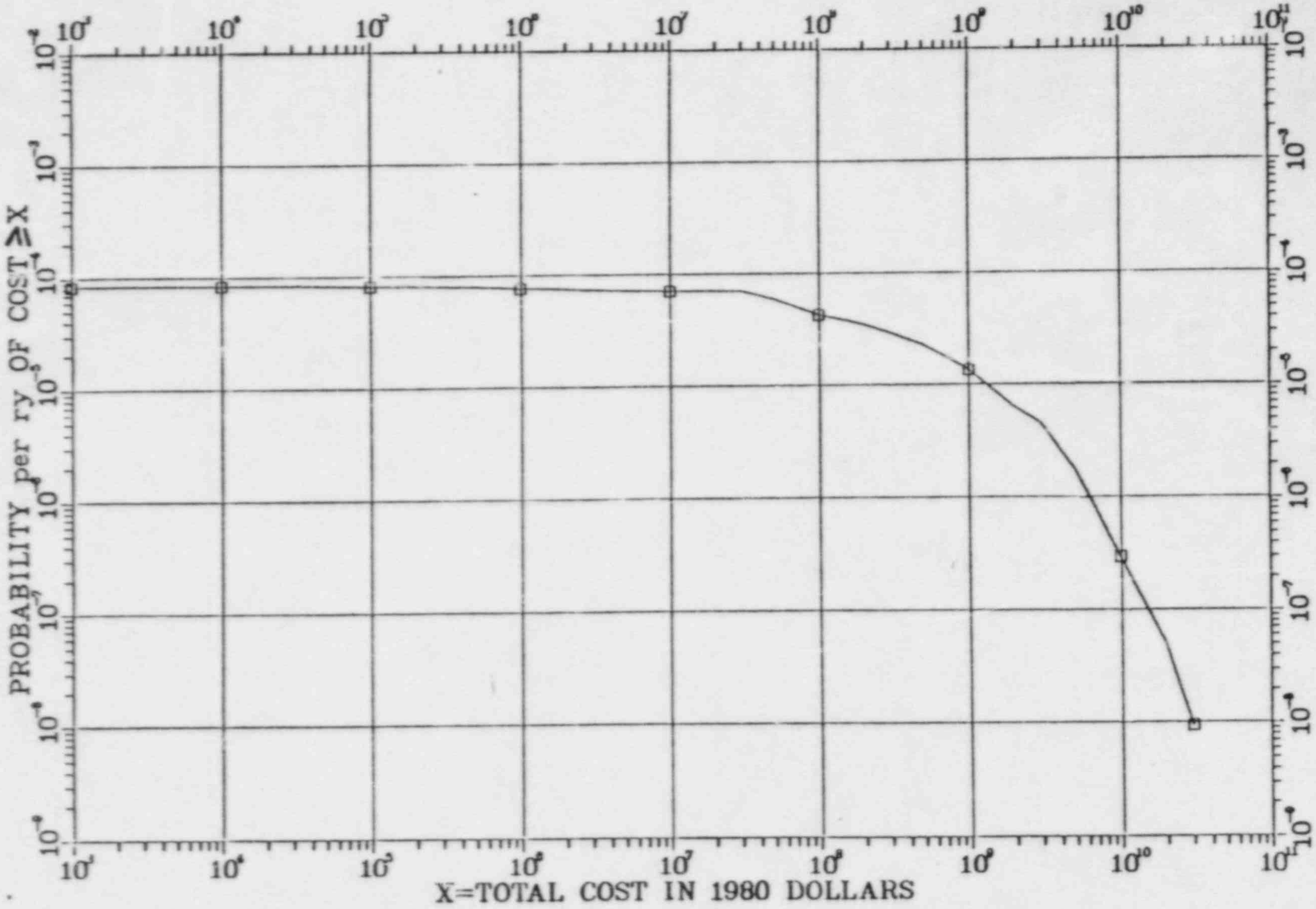

Figure 5.4g Probability distribution of cost of mitigation measures

NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

Table 5.11g Summary of environmental impacts and probabilities

Probability of impact per reactor-year	Persons exposed over: 300 rems thyroid dose	Persons exposed over: 200 rems total marrow dose	Persons exposed over: 25 rems whole body dose	Population exposure, whole body (million person-rems)*: 50 miles (80 km)	Population exposure, whole body (million person-rems)*: Total	Latent cancer fatalities (persons), Excluding thyroid: 50 miles (80 km)	Latent cancer fatalities (persons), Excluding thyroid: Total	Latent cancer fatalities (persons), Thyroid: 50 miles (80 km)	Latent cancer fatalities (persons), Thyroid: Total	Early fatalities (persons): With supportive medical treatment	Early fatalities (persons): With minimal medical treatment	Early injuries (persons)	Cost of offsite mitigation measures (millions of 1980 $)	Land area for long-term interdiction (millions of m^2)**
10^{-4}	0	0	0	0	0	0	0	0	0	0	0	0	0	0
10^{-5}	2(3)	3(1)	2(4)***	2(1)	3(1)	1(3)	2(3)	3(2)	3(2)	0	1(0)	9(1)	1(3)	4(1)
5×10^{-6}	7(3)	2(2)	5(4)	4(1)	5(1)	2(3)	3(3)	4(2)	6(2)	0	2(1)	2(2)	3(3)	7(1)
10^{-6}	4(4)	5(3)	3(5)	7(1)	1(2)	5(3)	7(3)	2(3)	2(3)	1(3)	2(3)	4(3)	6(3)	1(2)
10^{-7}	2(5)	3(4)	1(6)	1(2)	3(2)	1(4)	2(4)	4(3)	4(3)	9(3)	1(4)	3(4)	2(4)	3(2)
10^{-8}	5(5)	2(5)	3(6)	2(2)	5(2)	2(4)	3(4)	6(3)	6(3)	2(4)	3(4)	2(5)	3(4)	7(2)
See Figure	5.4b	5.4b	5.4b	5.4c	5.4c	5.4d	5.4d	5.4d	5.4d	5.4e	5.4e	5.4f	5.4g	5.4h

*About 260 cases of genetic effects may occur in the succeeding generations per million person-rem to the exposed generation.

**About 2.6 million square meters equals 1 square mile.

***$2(4) = 2 \times 10^4 = 20000$.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

of operation for one reactor. Different accident releases and atmospheric dispersion conditions, source-term magnitudes, and dose effects result in wide ranges of calculated magnitudes of consequences. Similarly, probabilities of equaling or exceeding a given consequence magnitude would also vary over a wide range because of varying probabilities of accidents and dispersion conditions.** Therefore, the CCDFs are presented as logarithmic plots in which numbers varying over a large range can be conveniently shown on a graph scaled in powers of 10. For example, a consequence magnitude of 10^6 means a consequence magnitude of one million (1 followed by six zeroes); a probability of 10^{-6} per reactor-year means a chance of 1 in one million or one millionth (0.000001) per reactor-year. All release categories shown in Table 5.11c contribute to the results; the consequences from each are weighted by its associated probability (Table 5.11d). For these calculations, the Evac Reloc mode of offsite emergency response was assumed for accidents initiated by causes internal to the plant, by fires and by low to moderately severe earthquakes; and Late Reloc mode of offsite emergency response was assumed for accidents initiated by very severe regional earthquakes (see Table 5.11f).

Figure 5.4b shows the probability distribution for the number of persons who might receive whole-body doses equal to or greater than 25 rems, total bone marrow doses equal to or greater than 200 rems, and thyroid doses equal to or greater than 300 rems from early exposure;*** all on a per reactor-year basis. The 200-rem total bone marrow dose figure corresponds, approximately, to a threshold value for which hospitalization would be indicated for the treatment of radiation injury. The 25-rem whole-body dose (which has been identified earlier as the lower limit for a clinically observable physiological effect in nearly all people) and the 300-rem thyroid dose figures correspond to the Commission's guideline values for reactor siting in 10 CFR 100.

Figure 5.4b shows in the left-hand portion that there are, approximately, 60 chances in 1 million (6×10^{-5}) per reactor-year that one or more persons may receive doses equal to or greater than any of the doses specified. The fact that the three curves run almost parallel in horizontal lines initially shows that if one person were to receive such doses, the chances are about the same that up to 10 would be so exposed. The chances of larger numbers of persons being exposed at those levels are seen to be considerably smaller. For example, the chances are less than 1 in 1 million (10^{-6}) that 10,000 or more people might receive doses of 200 rems or greater. A majority of the exposures reflected in this figure would be expected to occur to persons within a 40-km (25-mile) radius of the plant. Virtually all would occur within a 160-km (100-mile) radius.

Figure 5.4c shows the probability distribution for the total population exposure in person-rems; that is, the probability per reactor-year that the total population exposure will equal or exceed the values given. Most of the population

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*ry in the plots means reactor-year.

**See (7) below for further discussion of areas of uncertainty.

***Early exposure to an individual includes external doses from the radioactive cloud and the contaminated ground, and the dose from internally deposited radionuclides from inhalation of contaminated air during the cloud passage. Other pathways of exposures are excluded.

exposure up to 100 million person-rems would occur within 80-km (50 miles) but very severe releases would result in exposure to persons beyond the 80-km (50-mile) range, as shown.

For perspective, population doses shown in Figure 5.4c may be compared with the annual average dose to the population within 80 km (50 miles) of the Limerick site resulting from natural background radiation of about 800,000 person-rems, and to the anticipated annual population dose to the general public (total U.S.) from normal plant operation of about 80 person-rems (both units, excluding plant workers) (Appendix D of the environmental statement, Tables D.7 and D.9).

Figure 5.4d represents the statistical relationship between population exposure and the induction of fatal cancers that might appear over a period of many years following exposure. The impacts on the total population and the population within 80 km (50 miles) are shown separately. Further, the fatal latent cancer estimates have been subdivided into those attributable to exposures of the thyroid and all other organs. The majority of latent cancer (including thyroid) fatalities would occur within 80 km (50 miles) of the plant.

Figure 5.4e shows probability distributions of early fatalities. Two curves are shown representing benefits of two types of medical treatment (supportive and minimal; see Appendix J of this ~~supplement~~ FES and Appendix F of Appendix VI of WASH-1400) that would likely be given to individuals receiving excessive doses to the total bone narrow from early exposure. One curve shows the results considering the benefit of the supportive medical treatment. The early fatalities with supportive medical treatment are predicted to be essentially all within 32 km (20 miles) of the site. The other curve shows the results including the benefit of minimal medical treatment. The early fatalities with minimal medical treatments are predicted to be essentially all within 80 km (50 miles) of the site. As discussed in Appendix J, because it is conceivable that for very severe but low probability accidents, some of the people requiring supportive medical treatment may not actually receive it, the likely probability distribution of the early fatalities would be between the two curves shown in Figure 5.4e.

Figure 5.4f shows the probability distributions of early injuries that may result from acute radiation exposure. The cases of early injuries are predicted to be all within 160 km (100 miles) of the site.

An additional potential pathway for doses resulting from atmospheric release is from fallout onto open bodies of water. This pathway has been investigated in the NRC analysis of the Fermi Unit 2 plant, which is located on Lake Erie, and for which appreciable fractions of radionuclides in the plume could be deposited in the Great Lakes (NUREG-0769). It was found that for the Fermi site, the indicated individual and societal doses from this pathway were smaller than the interdicted doses from other pathways. Further, the individual and societal liquid pathway doses could be substantially eliminated by the interdiction of the aquatic food pathway in a manner comparable to interdiction of the terrestrial food pathway in the present analysis. Because Limerick is not on a large surface water body, the fraction of radioactive material that could fall out in nearby rivers, streams, or lakes would be correspondingly reduced. The staff has also considered fallout onto and runoff and leaching into water bodies in connection with a study of severe accidents at the Indian Point reactors in southeastern New York (Written staff testimony on Commission Question 1, Section III.D by Richard Codell on Liquid Pathway Considerations for the Indian Point ASLB Special Hearing, June 1982-April 1983). In this study empirical models were developed based upon considerations of radionuclide data collected in the New York City water supply system as a result of fallout from atmospheric weapons tests. As with the Fermi study, the Indian Point evaluation indicated that the uninterdicted risks from this pathway were fractions of the interdicted risks from other pathways. Further, if interdicted in a manner similar to interdiction assumed for other pathways, the liquid pathway risk from fallout would be a very small fraction of the risks from other pathways. Considering the LGS and the regional meteorology and hydrology, the staff sees nothing to indicate that the liquid pathway contribution to the total accident risk would be significantly greater than found for Fermi 2 and Indian Point. This water pathway would be of small importance compared to the results presented here for fallout onto land.

(4) Economic and Societal Impacts

As noted in Section 5.9.4.2, the various measures for avoiding adverse health effects, including those resulting from residual radioactive contamination in the environment, are possible consequential impacts of severe accidents. Calculations of the probabilities and magnitudes of such impacts for Limerick station and environs also have been made. (NUREG-0340 describes the model used.) Unlike the radiation exposure and health effect impacts discussed above, impacts associated with avoiding adverse health effects are more readily transformed into economic impacts.

The results are shown as the probability distribution for cost of offsite mitigating actions in Figure 5.4g and are included in the impact summary Table 5.11g. The factors contributing to these estimated costs include the following:

- evacuation costs
- value of crops contaminated and condemned
- value of milk contaminated and condemned
- costs of decontamination of property where practical
- indirect costs resulting from the loss of use of property and incomes derived therefrom

The last-named costs would derive from the necessity for interdiction to prevent the use of property until it is either free of contamination or can be economically decontaminated.

Figure 5.4g shows that at the extreme end of the accident spectrum these costs could exceed tens of billions of dollars, but that the probability that this would occur is exceedingly small (less than one chance in 10 million per reactor-year).

Additional economic impacts that can be monetized include costs of related health effects, cost of regional industrial impacts, costs of decontamination of the facility itself, and the costs of replacement power. Probability distributions for these impacts have not been calculated, but they are included in the discussion of risk considerations in Section 5.9.4.5(6) below.

As an additional impact of environmental contamination, Figure 5.4h shows the probability distribution of severely contaminated land area in square meters (about 2.6 million square meters equals 1 square mile) that would not be returned to use by decontamination, because decontamination procedures would not be very effective. Such areas would be marked for long-term interdiction (more than 30 years). At the extreme end of the accident spectrum, Figure 5.4h shows that such areas could be as large as several hundreds of square miles, but the probability that this could occur is extremely small (less than 1 chance in 10 million per reactor-year). This impact is also included in Table 5.11g.

The geographical extent of the kinds of impacts discussed above, as well as many other types of impacts, is a function of several factors. For example, the dispersion conditions and wind direction following a reactor accident, the type of accident, and the magnitude of the release of radioactive material are all important in determining the geographical extent of such impacts. Because of these large inherent uncertainties, the values presented herein are mean values of the important types of risk based upon the methodology employed in the accident consequence model (NUREG-0340) and do not indicate specific geographical areas.

(5) Releases to Groundwater

A groundwater pathway for radiation exposure to the public and environmental contamination that would be unique for severe reactor accidents was identified in Section 5.9.4.2(2) above. Consideration has been given to potential environmental impacts of this pathway for the Limerick station. The penetration of the basemat of the containment building can release molten core debris to the strata beneath the plant. The soluble radionuclides in the debris can be leached and transported with groundwater to downgradient domestic wells used for drinking water or the surface water bodies used for drinking water, aquatic food, and recreation. Releases of radioactivity to the groundwater underlying the site could also occur via depressurization of the containment atmosphere and releases of radioactive ECCS and suppression pool water through the failed containment.

An analysis of the potential consequences of a liquid pathway release of radioactivity for generic sites was presented in the "Liquid Pathway Generic Study" (LPGS) (NUREG-0440). The LPGS compares the risk of accidents involving the liquid pathway (drinking water, irrigation, aquatic food, swimming, and shoreline usage) for four conventional, generic, land-based nuclear plants and for a floating nuclear plant for which the nuclear reactor would be mounted on a barge and moored in a water body. Parameters for each generic land-based site were chosen to represent averages for a wide range of real sites and were thus "typical", but represented no real sites in particular. The discussion in this section is a summary of an analysis performed to compare the liquid pathway consequences of a postulated accident at the Limerick site with that of the generic small-river land-based site considered in the LPGS. The comparison is made on the basis of population doses from drinking contaminated water, eating contaminated fish, and such shoreline uses as recreation. The parameters that were evaluated include the amounts and rate of release of radioactive materials to the ground, ground water travel time, sorption on geological media, surface water transport, drinking water usage, aquatic food consumption, and recreation area usage.

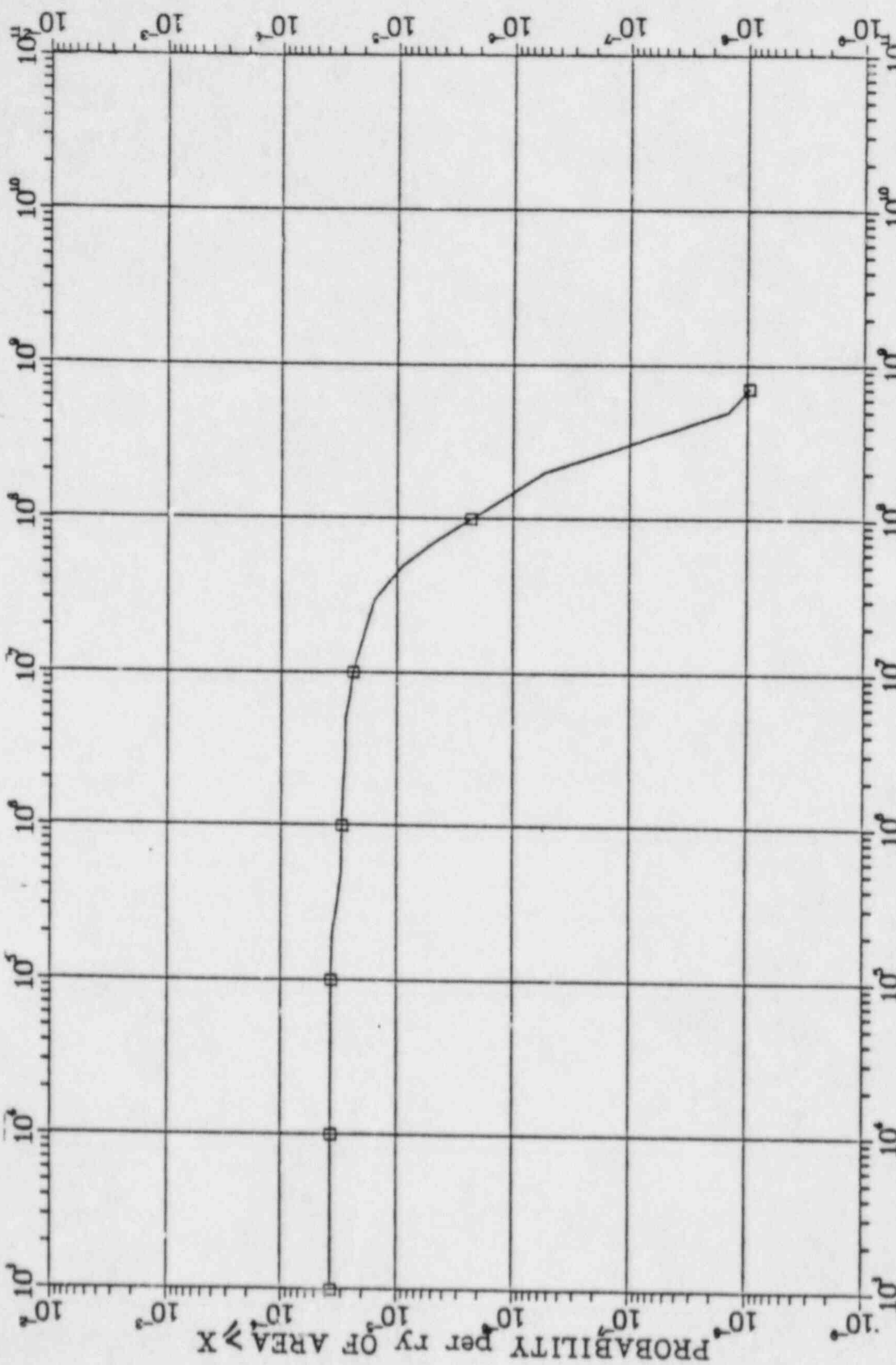

Figure 5.4h Probability distribution of land area interdiction

NOTE: See Section 5.9.4.5(7) for discussion of uncertainties.

All of the reactors considered in the LPGS were Westinghouse pressurized water reactors (PWRs) with ice condenser containments. There are likely to be significantly different mechanisms and probabilities of releases of radioactivity for the Limerick boiling water reactor (BWR). The staff is not aware of any studies which indicate the probabilities or magnitudes of liquid releases for BWRs. The source term used for Limerick in this comparison is assumed to be equal to that used in the LPGS.

Doses to individuals and populations were calculated in the LPGS without consideration of interdiction methods such as isolating the contaminated groundwater or denying use of the water. In the event of surface water contamination, alternative sources of water for drinking, irrigation, and industrial uses would be expected to be found, if necessary. Commercial and sports fishing, as well as many other related activities, could be restricted. The consequences would, therefore, be largely economic and social rather than radiological. In any event, the individual and population doses for the liquid pathway range from fractions to very small fractions of those that can arise from the airborne pathways.

The Limerick site is about 244 meters (800 feet) from and 33.5 meters (110 feet) above the Schuylkill River. The aquifer underlying the site is composed of red shale, sandstone, and siltstone. Most of the groundwater movement in the aquifer follows secondary openings that have developed following the deposition of the beds. The most important openings are nearly vertical joint planes; they cross each other at various angles throughout the beds. Where these joints are present, they provide an interconnected series of channels through which groundwater can flow, giving the material a low to moderate permeability.

The weathered upper bedrock in the power block area has been removed and the small fracture zones in the remaining rock have been filled with concrete. Should a core melt accident occur at the Limerick site and the leached radionuclides find a path through the concrete basemat, the tight bedrock beneath the basemat would tend to confine the effluent and greatly limit its transport downgradient. For the purposes of this analysis, however, the radioactive effluent was conservatively assumed to travel immediately through the underlying rock and move downgradient toward the river.

The applicant performed an analysis of the liquid pathway release following a postulated core melt accident and determined a groundwater travel time of 3.28 years from the reactor building to the Schuylkill River The groundwater travel time calculated for the LPGS generic site was 0.61 years.

The staff has evaluated the applicant's groundwater travel time calculation and the data used to choose the pertinent parameters and considers the applicant's analyses to be conservative. The average bedrock permeability, estimated from site permeability tests, is 65 m (214 feet) per year, and the effective porosity is estimated to be 0.05. The groundwater gradient likely to exist after plant construction is estimated to be no greater than 0.025, based on well hydrographs at the site. From these values, the staff estimates a groundwater travel time of 7.5 years for the 244 meters to the river.

It was demonstrated in the LPGS that for holdup times on the order of years, virtually all the liquid pathway population dose results from Sr-90 and Cs-137.

Therefore, only these two radionuclides are considered in the remainder of this analysis.

The radionuclides Sr-90 and Cs-137 usually move much slower than groundwater because of the effects of sorption (ion-exchange) on the geologic media. However, most of the measured values of the retardation effects of sorption are applicable only to soil or pulverized rock. There is only limited data available on retardation in fractured geologic media. At the Limerick site, however, the fractures in the siltstone and sandstone are partially filled with calcite, sand, and clay. Hence, part of the flow path would be through porous media, and ion exchange can be expected to retard the movement of radionuclides to the Schuylkill River. Based on measured retardation related distribution coefficients (Kd) for similar rock types and soil (Isherwood, 1981), a Kd of 2 was selected for Sr-90 and a Kd of 20 for Cs-137. Both Kd values selected are on the low side of representative values and are, therefore, considered to be conservative. A total porosity of 25% was selected as representative of the fractured and filled media through which the radioactive effluent would travel. From these values, retardation coefficients of 20 for Sr-90 and 193 for Cs-137 were determined as being reasonably conservative for the transport media. The calculated radionuclide travel time is then 150 years for Sr-90 and 1447.5 years for Cs-137. The radionuclide travel times for Sr-90 and Cs-137 in the LPGS are 5.7 years and 51 years, respectively. As a result of radioactive decay, the estimated amount of Sr-90 entering the Schuylkill River would be reduced to about 3% of the amount determined in the LPGS. The amount of Cs-137 would be about 14 orders of magnitude less than that in the LPGS, and its contribution to population dose via the various pathways (drinking water, fish consumption, and recreation activities) need not be considered further.

The primary pathway for Sr-90 to humans is through drinking water. Comparison of drinking water population doses will be based upon the ratio of population served to river flow, which takes into account the effects of dilution. Downstream of the Limerick site, there are approximately 1.9 million people using the Schuylkill River as a drinking water supply. The average flow in Schuylkill River is about 1900 ft^3/sec resulting in a population to flow ratio of 1000 people/ft^3/sec. The corresponding ratio in the LPGS for a small river site is about 32 people/ft^3sec. Hence, for a similar release to a river, the total drinking water dose at Limerick without a change in drinking water supply, would be about 30 times worse. However, since the concentration of Sr-90 entering the water would be only 3% of that of the LPGS, the total drinking water dose is roughly equivalent to that determined in the LPGS. The staff concludes that population dose as a result of the liquid pathway contribution at the Limerick site would be about the same as that from the generic site.

The staff recognizes that, because of the differences in design of the Limerick reactor as compared to the reactor design analyzed in the LPGS, a different inventory of radionuclides could be released following a core melt accident and postulated breach of the basemat. This uncertainty, along with uncertainties in the amount of radionuclides that could be released, could result in a different dose comparison than the one presented. However, the staff also considers the potential for a release through the basemat at the Limerick site following a core melt accident to be significantly less than that for the design considered in the LPGS. Therefore, the total risk from the liquid pathway is still estimated to be less than or about the same order as that in the LPGS.

In conclusion, Limerick should be considered about equal in regard to risk from the liquid pathway (groundwater) in comparison to other land-based sites. In addition, the long groundwater travel time ensures that mitigation measures such as slurry walls, grouting, dewatering, and other measures can be completed in time to protect downstream drinking water and fisheries. A comprehensive discussion of accident mitigation measures has been presented by V. A. Harris (Harris, 1982).

(6) Risk Considerations

The foregoing discussions have dealt with both the frequency (or likelihood of occurrence) of accidents and their impacts (or consequences). Because the ranges of both factors are quite broad and uncertain (see (7) below), it also is useful to combine them to obtain average measures of environmental risks. Such averages can be particularly instructive as an aid to the comparison of radiological risks associated with accident releases with risks associated with normal operational releases and with other forms of risks.

A common way in which this combination of factors is used to estimate risk is to multiply probabilities by the consequences. The resultant risk is then expressed as a measure of consequences per unit of time. Such a quantification of risk does not mean that there is universal agreement that peoples' attitudes about risks, or what constitutes an acceptable risk, can or should be governed solely by such a measure. However, it can be a contributing factor to a risk judgment, although not necessarily a decisive factor.

Table 5.11h shows average values of societal risk estimates associated with population dose, early fatalities with two types of medical treatment (minimal and supportive), early injuries, latent cancer fatalities, costs for evacuation and other protective actions, and land area for long-term interdiction. These average values are obtained by summing the probabilities multiplied by the consequences over the entire range of the distributions. Because the probabilities are on a per-reactor-year basis, the averages shown also are on a per-reactor-year basis.

Incremental risks per reactor-year of early fatality (with two types of medical treatment) and latent cancer fatality associated with spatial intervals up to 50 miles (80 km) from the Limerick reactors are shown in Appendix L.

The population exposures and latent cancer fatality risks may be compared with those from normal operation shown in Appendix D and Section 5.9.3.2 of this statement. The comparison (excluding exposure to station personnel) shows that the accident risks are up to 30 times higher. For a different perspective, the latent cancer (including thyroid) fatality risks of 3×10^{-4} persons per reactor-year within 1 mile (1.6 km) of the site exclusion area boundary (EAB) (based on data in Table L.4 in Appendix L) and 5×10^{-2} persons per reactor-year within the 50-mile (80-km) region (from Table 5.11h) may be compared with such risks from causes other than reactor accidents. Approximately 3000 persons are projected to live within 1 mile (1.6 km) from the EAB and 7 million persons are projected to live within the 50-mile (80-km) region in the year 2000. The background cancer mortality rate is 1.9×10^{-3} cancer fatality per person per year

Table 5.11h Estimated values of societal risks from severe accidents, per reactor-year

Consequence type		Estimated risk within the 50-mile region	Estimated risk within the entire region
1.	Early fatalities with Supportive medical treatment (persons)	5(-3)*	5(-3)
2.	Early fatalities with minimal medical treatment (persons)	8(-3)	8(-3)
3.	Early injuries (persons)	2(-2)	2(-2)
4.	Latent cancer fatalities (excluding thyroid) (persons)	4(-2)	7(-2)
5.	Latent thyroid cancer fatalities (persons)	1(-2)	1(-2)
6.	Total person-rems	7(2)	[illegible](3)
7a.	Cost of offsite mitigation measures (1980 $)	5(4)	5(4)
7b.	Regional industrial impact costs (1980 $)		5(4)***
7c.	Plant costs (1980 $)	1(5)	
8.	Land area for long-term interdiction (m^2)**	1(3)	1(3)

*5(-3) = 5×10^{-3} = .005
**About 2.6 million m^2 equals to 1 mi^2.
***Excludes costs of crop and milk interdiction, which are included in 7a.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

in the U.S (American Cancer Society, 1981). Therefore, at this rate, about 6 background cancer fatalities per year are expected in the population within 1 mile (1.6 km) of the EAB, and 10,000 background cancer fatalities in the population within the 50-mile (80-km) region in the year 2000. Thus, the risk of cancer fatality from reactor accidents at Limerick is small compared to the risk of normal occurrence of such fatality.

The ratio of latent cancer fatality risk from reactor accidents at Limerick to the population living within 50 miles of the plant in the year 2000 to the cancer fatality risk in the same population from all other causes is 5×10^{-6} (5×10^{-2}/10,000) on a per reactor-unit basis.

There are no early fatality, early injury, long-term land interdiction, or economic risks associated with protective actions and decontamination for normal releases; but these risks can be associated with large accidental releases. For perspective and understanding of the meaning of the early fatality risk of 5×10^{-3} persons per reactor-year with supportive medical treatment and 8×10^{-3} persons per reactor-year with minimal medical treatment (from Table 5.11h), the staff notes that occurrences of early fatalities with supportive and minimal medical treatments would be contained, approximately, within the 20-mile (32-km) and 50-mile (80-km) regions, respectively. The number of persons projected to live within these regions in the year 2000 are 0.8 million and 7 million, respectively. The background risk for the average individual in the U.S. is 5×10^{-4} accidental death per year (NUREG/CR-1916). Therefore, the expected number of non-Limerick accidental fatalities per year within the 20-mile (32-km) and 50-mile (80-km) regions are 400 and 4000, respectively, in the year 2000. Thus, the risk of early fatality with supportive or minimal medical treatment from reactor accidents at Limerick is extremely small compared with that from non-Limerick accidents. For an added perspective, the risk of early fatality within 1 mile (1.6 km) of the exclusion area boundary (EAB) from reactor accidents may be compared with early fatality risk from nonnuclear accidents in the same region. From Tables L.2 and L.3 in Appendix L, the Limerick risks of early fatality with supportive or minimal medical treatments are 5×10^{-4} persons per reactor-year and 6×10^{-4} persons per reactor-year, respectively, in this region. At the average rate of 5×10^{-4} nonnuclear accidental death per individual per year in the U.S., the number of nonnuclear accidental fatalities in the population of 3000 projected to live within 1 mile (1.6 km) from the EAB in the year 2000 would be 2 per year. This also shows that the early fatality risk from reactor accidents at Limerick is expected to be small compared with risk of nonnuclear accidental deaths.

The ratio of (1) risk of early fatality with minimal medical treatment from reactor accidents at Limerick to an average individual living within a mile of the site exclusion area boundary to (2) the risk to the same individual of accidental death from all other causes, is 3×10^{-4} ($6 \times 10^{-4}/3000 \div 2/3000$) on a per reactor-unit basis.

To provide a reasonable bound to the role of evacuation in risk estimates from the release categories not initiated by severe earthquakes, as well as to assess the sensitivity of risks from these release categories with respect to uncertainties in executing an evacuation, an analysis of these release categories was made by assuming the Early Reloc mode of offsite emergency response (see Table 5.11f). Results of the analysis are provided in Appendix M. These results, when combined with those previously calculated for the release categories initiated by severe eathquakes, show only slight increases in the risks of latent cancer and early fatalities and also corroborate the preceding conclusions that these risks from Limerick reactor accidents are small compared with the background risks from nonnuclear causes.

Figure 5.4i shows the calculated risk of whole-body dose to an individual from early exposure as a function of the downwind distance from the plant. The values are on a per-reactor-year basis and all release categories contributed to the dose, weighted by their associated probabilities. For purposes of comparison the risk of receiving a whole body dose of 99 mrems per year from

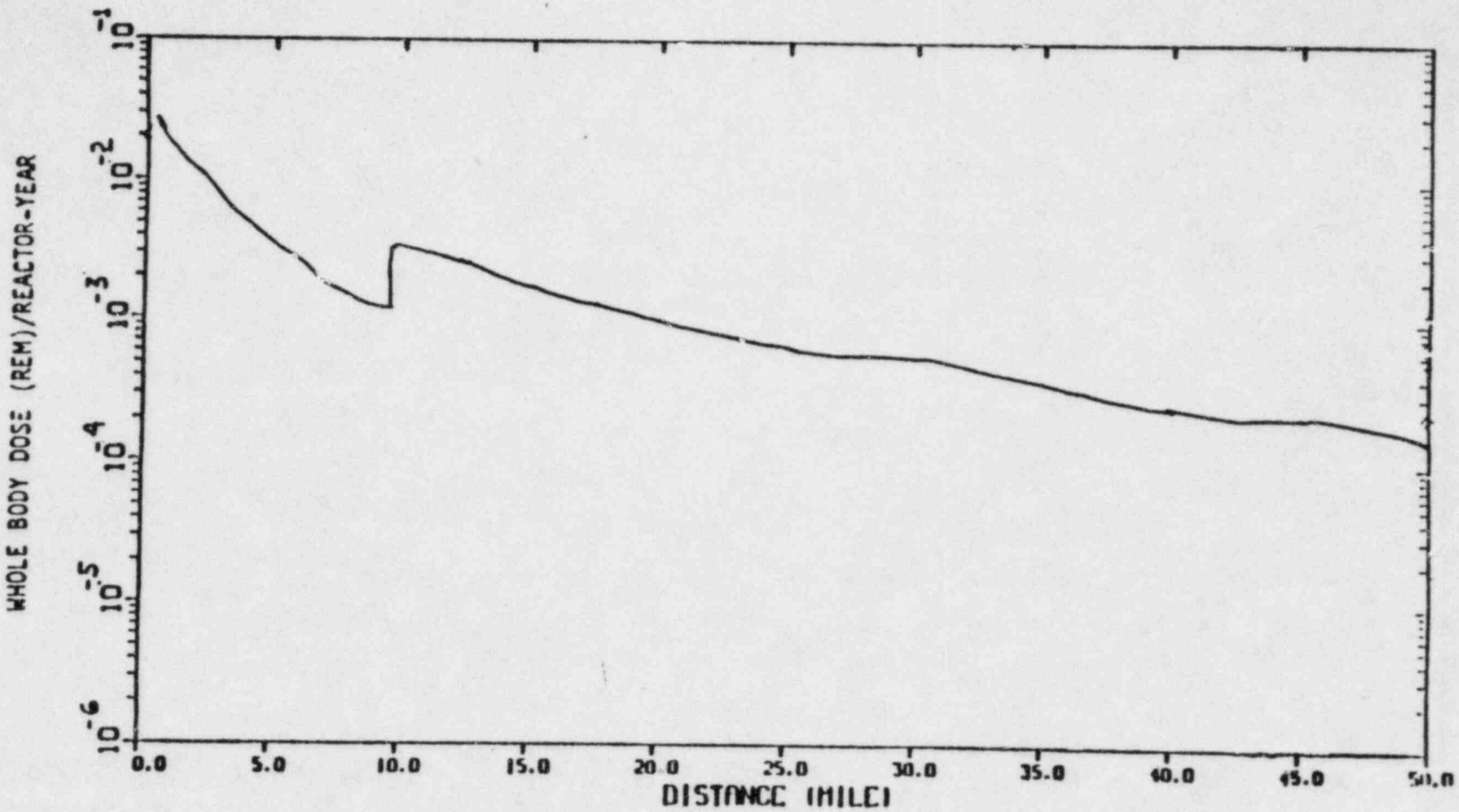

Figure 5.4i Risk of downwind individual dose versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

natural background is a virtual certainty for any individual living in the Limerick site region (see Table D.7 in Appendix D).

Figures 5.4j, 5.4k, and 5.4l, respectively, display risk to an individual of early fatality, early injury, and latent cancer fatality, all from early exposure, as functions of distance from the Limerick reactors and on a per-reactor-year basis. The curves in these figures were generated without regard to the differences in the likelihood of wind blowing in different directions (the staff used 16 direction sectors of the compass). To obtain risk curves for a specific direction (1 out of the 16), all values on the curves along the vertical axis must be multiplied by 16P, where P is the annual average probability of the wind blowing toward the direction of interest. The values of P for the Limerick site derived from 1976 meteorological data are shown in Table 5.11e. For comparison to early fatality risk to an individual from Limerick reactor accidents, the following nonnuclear risks, per year, of accidental fatality to an individual living in the United States may be noted (National Research Council, 1979, p. 577): automobile accident 2.2×10^{-4}, falls 7.7×10^{-5}, drowning 3.1×10^{-5}, burning 2.9×10^{-5}, and firearms 1.2×10^{-5}. For comparison to the estimated latent cancer fatality risk to an individual from the Limerick reactor accidents, it should be noted that the risk of cancer fatality to an individual in the U.S. from nonnuclear causes is 1.9×10^{-3} per year (American Cancer Society, 1981).

The economic risk associated with evacuation and other protective actions could be compared with property damage costs associated with alternative energy generation technologies. The use of fossil fuels, coal, or oil, for example, would emit substantial quantities of sulfur dioxide and nitrogen oxides into the atmosphere and, among other things, lead to environmental and ecological damage through the phenomenon of acid rain (National Research Council, 1979, pp. 559-560). In the judgment of the staff, this effect has not been sufficiently quantified to draw a useful comparison at this time.

The staff has also considered the health care costs resulting from hypothetical accidents in a generic model developed by the Pacific Northwest Laboratory (Nieves, 1982). Based upon this generic model, the staff concludes that such costs may be a fraction of the offsite costs evaluated herein, but that the model is not sufficiently constituted for application to a specific reactor site.

A severe accident that requires the interdiction and/or decontamination of land areas is likely to force numerous businesses to temporarily or permanently close. These closures would have additional economic effects beyond the contaminated areas through the disruption of regional markets and sources of supplies. Estimates of these risks were made using: (1) the RSS consequence model (Appendix VI, WASH-1400) and (2) the regional input-output modeling system (RIMS II), developed by the Bureau of Economic Analysis (BEA).

The industrial impact model developed by BEA is based on contamination levels of a physically affected area defined by the RSS consequence model. Contamination levels define an interdicted area immediately surrounding the plant, followed by an area of decontamination, an area of crop interdiction, and finally an area of milk interdiction.

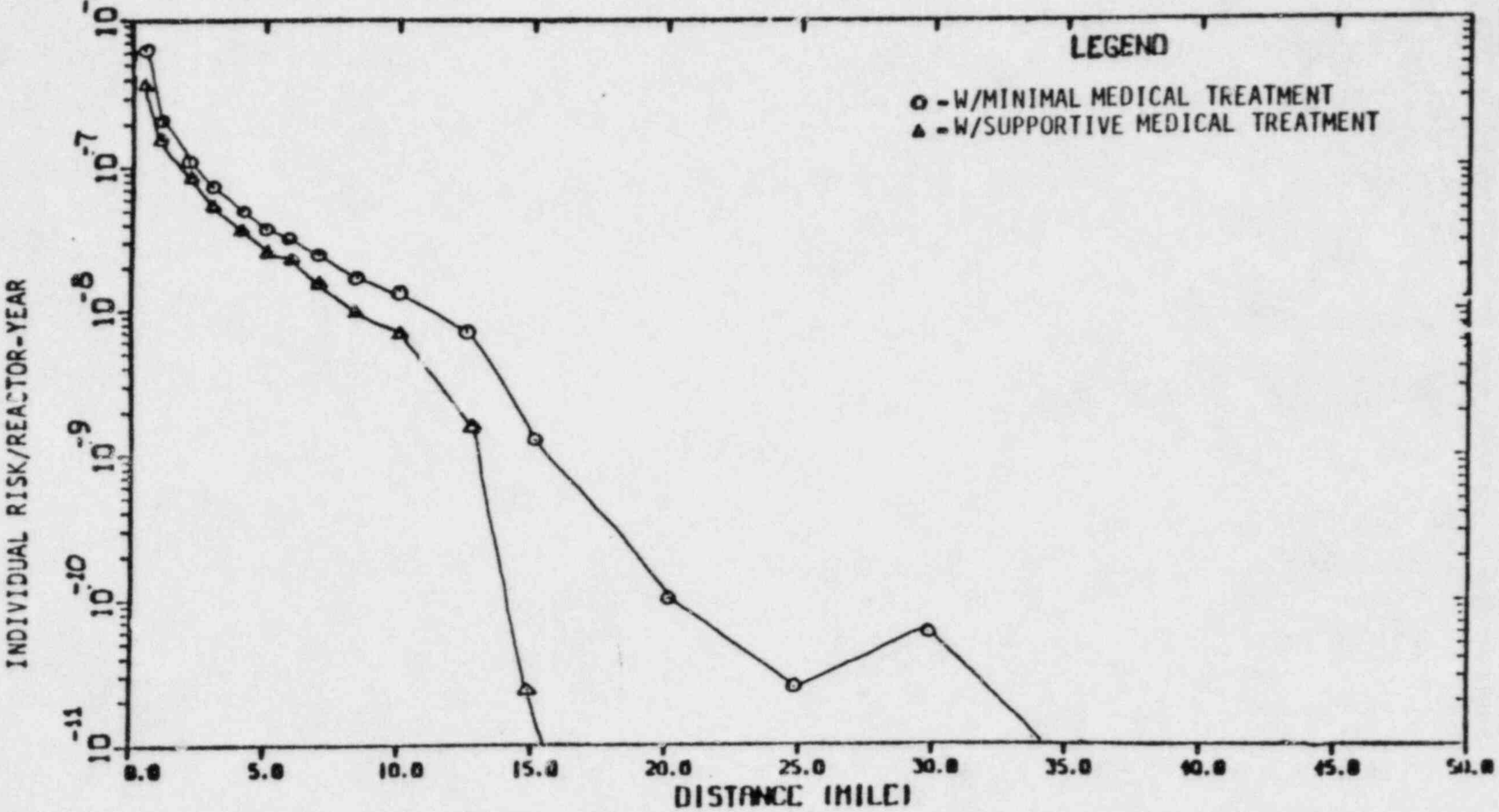

Figure 5.4j Individual risk of early fatality versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

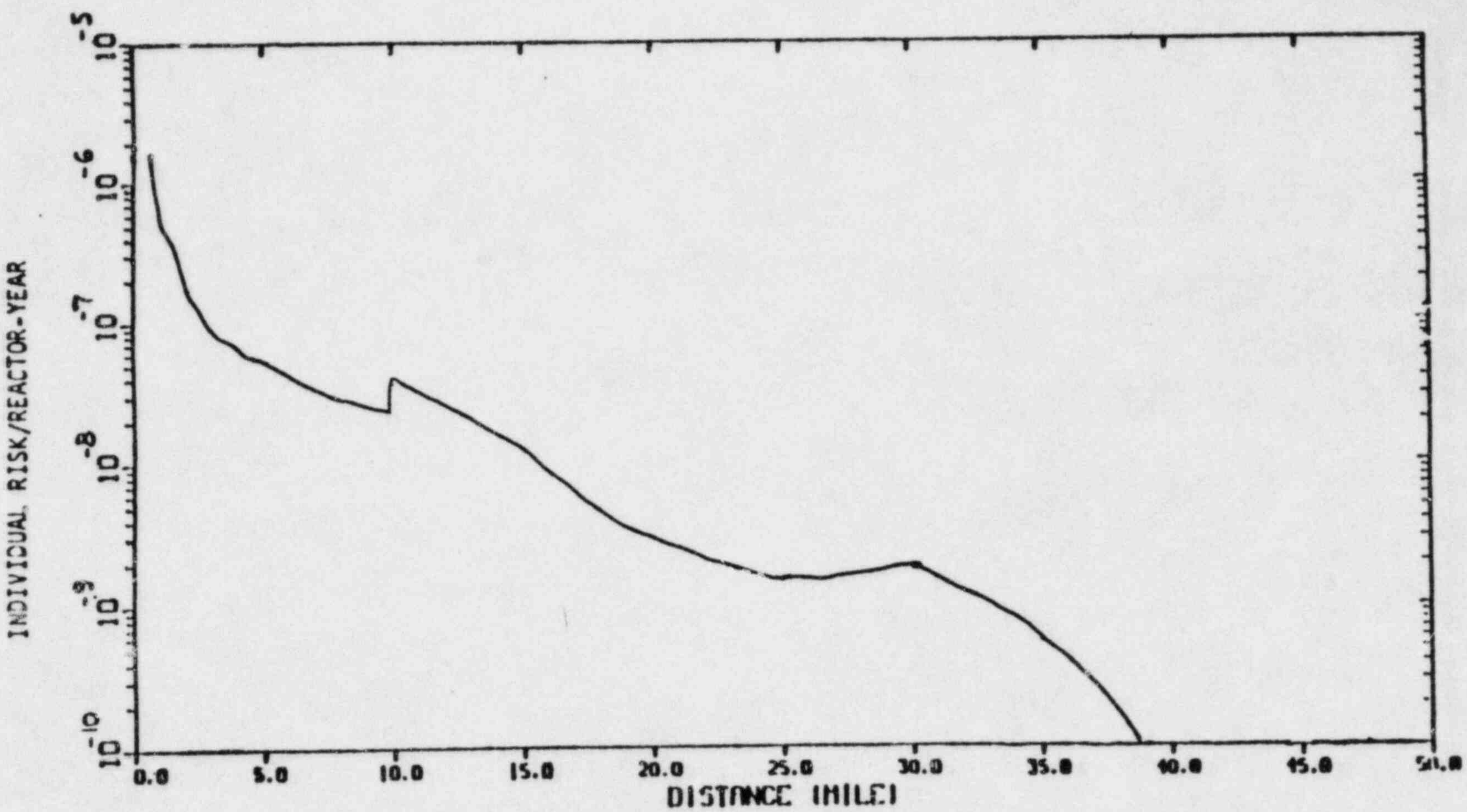

Figure 5.4k Individual risk of early injury versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

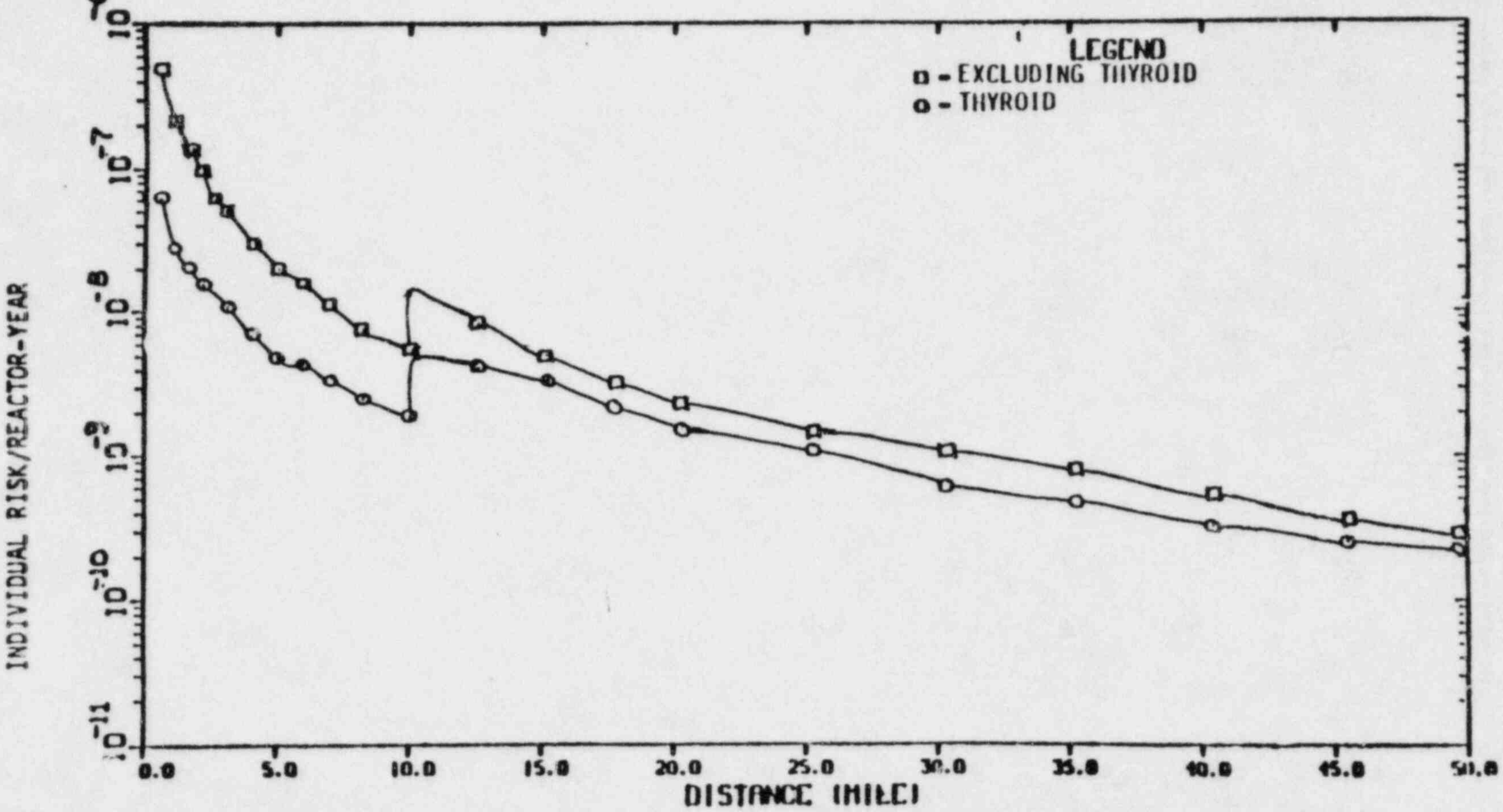

Figure 5.41 Individual risk of latent cancer fatality versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

Specific assumptions used in the analysis are

(1) In the interdicted area, all industries would lose total production for more than a year.

(2) In the decontamination zone, there would be a 3-month loss in nonagricultural output; a 1-year loss in all crop output (except there would be no loss in greenhouse, nursery, and forestry output); a 3-month loss in dairy output; and a 6-month loss in livestock and poultry output.

(3) In the crop interdiction area, there would be no loss in nonagricultural output; a one-year loss in agricultural output (except there would be no loss in greenhouse, nursery, and forestry output); no loss in livestock and poultry output; and a 2-month loss of dairy output.

(4) In the milk interdiction zone, there would be only a 2-month loss in dairy output.

The estimates of industrial impacts are made for an economic study area that consists of a physically affected area and a physically unaffected area. An accident that causes an adverse impact in the physically affected area (for example, the loss of agricultural output) could also adversely affect output in the physically unaffected area (for example, food processing). In addition to the direct impacts in the physically affected area, the following additional impacts could occur in the physically unaffected area:

(1) decreased demand (in the physically affected area) for output produced in the physically unaffected area

(2) decreased availability of production inputs purchased from the physically affected area

Only the impacts occuring during the first year following an accident are considered. The longer term consequences are not considered because they will vary widely depending on the level and nature of efforts to mitigate the accident consequences and to decontaminate the physically affected areas.

The estimates assume no compensating effects, such as the use of unused capacity in the physically unaffected area to offset the initial lost production in the physically affected area or income payments to individuals displaced from their jobs that would enable them to maintain their spending habits. These compensating effects would reduce the industrial impacts. Realistically, these compensating effects would occur over a lengthy period. The estimates using no compensating effects are the best measures of first year economic impacts.

The output loss risk can be estimated by mutiplying the probabilities of the release categories representative of those in Table 5.11c by the probability of the wind blowing in various directions and the associated consequences. The overall risk associated with these release categories was then estimated as the sum of the individual products. The estimated overall risk values using output losses as the measure of accident consequences, expressed in a per reactor-year basis, is $50,000 (1980 dollars) per reactor-year. This includes $2000 as the cost of crop and milk interdictions calculated in CRAC runs for consequence analysis. The corresponding expected employment loss is between two and three jobs per reactor-year. Half of the total risk per reactor-year is accounted for by the cases of wind blowing toward the east-southeast. The risk is least severe with the wind blowing toward the east-southwest. Because of the economic mix of the entire region, the composition of impacts consists of 85% nonagricultural impacts, 4% agricultural impacts, and 11% indirect impacts of decreased exports and supply constraints.

There are other economic impacts and risks that can be monetized but that are not included in the cost calculations discussed earlier. These are accident impacts on the facility itself that result in added costs to the public (ratepayers, taxpayers, and/or shareholders). These costs would be for decontamination and repair or replacement of the facility, and replacement power. Experience with such costs is currently being accumulated as a result of the Three Mile Island accident. If an accident occurs during the first full year of Limerick Unit 1 operation (1985), the economic penalty associated with the initial year of the unit's operation is estimated at $1500 million for decontamination and restoration, including replacement of the damaged nuclear fuel. This is based on a conservative (high) 10% escalation of the $950 million cost in 1980 dollars estimated for Three Mile Island (EMD-81-106). Although insurance would cover $300 million or more of the $1500 million, the insurance is not credited against the $1500 million because the $300 million times the risk probability should theoretically balance the insurance premium. In addition, staff estimates additional fuel costs of $50 million (1985 dollars) for replacement power during each year Limerick Unit 1 was being restored. This estimate assumes conservatively (high cost) that two-thirds of the energy that would have been forthcoming from the unit (assuming 55% capacity factor) would be replaced by coal-fired generation and one-third by oil-fired generation. Assuming the nuclear unit does not operate for 8 years, the total additional replacement power costs would be approximately $400 million in 1985 dollars.

The probability of a core melt or severe reactor damage is assumed to be as high as 10^{-4} per reactor year (this accident probability is intended to account for all severe core damage accidents leading to large economic consequences for the owner, not just those leading to significant offsite consequences).

Multiplying the previously estimated costs of $1900 million for an accident to Limerick Unit 1 during the initial year of its operation by the above 10^{-4} probability results in an economic risk of approximately $190,000 (in 1985 dollars or $120,000 in 1980 dollars) applicable to Limerick Unit 1 during its first year of operation. This is also aproximately the economic risk (in 1985 dollars) to Limerick Unit 1 during the second and each subsequent year of its operation. Although nuclear units depreciate in value and may operate at reduced capacity factors so that the economic consequences of an accident become less as the units become older, this is conservatively (high cost) considered to be offset by a slightly higher escalation rate than discount rate.

The economic risk to Limerick Unit 2 (in 1985 dollars) is also approximately $190,000 (or $120,000 in 1980 dollars) during the first year and each subsequent year of operation because of the balancing effect of escalation and the present-worth discount factor.

(7) Uncertainties

The probabilistic risk assessment discussed above has been based mostly on the methodology in the RSS, which was published in 1975 (NUREG-75/014). Although substantial improvements have been made in various facets of the RSS methodology since this publication was issued, there are still large uncertainties in the results of the analysis presented above because of the uncertainties associated with the likelihoods of the accident sequences and containment failure modes leading to the release categories, the source terms for the release categories, and the estimates of environmental consequences.

Relatively more important contributors to uncertainties in the results presented in this supplement are as follows:

Probability of Occurrence of Accident

If the probability of a release category were to be changed by a certain factor, the probabilities of various types of consequences from that release category would also change exactly by the same factor. Thus, an order of magnitude uncertainty in the probability of a release category would result in an order of magnitude uncertainty in both societal and individual risks stemming from the release category. As in the RSS, there are substantial uncertainties in the probabilities of the release categories. This is due, in part, to difficulties associated with the quantification of human error and to inadequacies in (1) the data base on failure rates of individual plant components, and (2) the data base on external events and their effects on plant systems and components that are used to calculate the probabilities.

Severe earthquakes are one cause of accidents. Uncertainties in the estimates of probabilities of severe earthquake induced core melt sequences are judged to be very large because of (1) the relatively sparse data base on severe earthquakes in the eastern U.S. and (2) the unavailability of an acceptably precise and definite procedure to quantify seismically induced accident sequences. In LGS-SARA, the spectrum of probabilities of seismically induced core melt sequences varied over a wide range (several orders) of magnitudes. However, the mean (point or best estimate) probabilities of seismically induced core melt accident sequences used in the staff analysis (which essentially came from LGS-SARA) are within the range of probabilities developed in LGS-SARA, and are within a factor of about 6 of the upper end of the spectrum of probabilities in LGS-SARA. Thus, the point estimates of seismic probabilities used to evaluate risks are more representative of Limerick than WASH-1400 values, and consider the applicant's estimate of the range of seismic frequency uncertainty. The staff has concluded that the high and low values of the range should not be characterized as 95% and 5% limits, but rather as a representative range of the seismic sequence frequencies, which incorporates a large part (but not necessarily all) of the uncertainties with such events. This statement reflects the staff's view that the rigorous definition of seismic hazard and its uncertainty at low probabilities is beyond the state-of-the-art at this time and should be recognized as such. Different studies would not necessarily yield equivalent results. For example, an interium report to be published "Seismic Hazard Characterization of the Eastern U.S." of an ongoing study being carried out by Lawrence Livermore National Laboratory (LLNL) for the NRC shows seismic hazard calculations for the Limerick site which overlap, but are not necessarily coincident with, the range of seismic hazard assumed in LGS-SARA.

The median (50%) hazard calculated in the interim LLNL report is within, but near the high end of, the range of hazard curves utilized in LGS-SARA. Additional studies of seismic hazard in the eastern U.S. are being carried out by such groups as the Electric Power Research Institute. Given the highly judgmental nature of seismic hazard calculations, there is not reason to believe that these studies or the final LLNL report would not show differences in estimated seismic hazard and uncertainty between themselves and the LGS-SARA, particularly at the low probabilities being calculated for Limerick. The staff believes that only the use of a full range of seismic probabilities in risk analysis would be appropriate. However, to keep the risk analysis manageable, the staff has used the point estimates of probabilities of seismically induced release categories in the risk analysis, and has provided below a discussion of uncertainty in the risk estimates arising from the use of point estimates of probabilities.

Inspection of the results shown in Tables L-1a and b and M-1a and b indicates that with the use of the mean values of probabilities of the severe earthquake initiated release categories, these release categories contribute: (1) dominantly (about 4 to 30 times higher) to the risks of early fatality; (2) about equally to the risk of early injury; and (3) much less to the other types of risks--all compared to the contributions from the release categories initiated by causes other than severe earthquakes. If, instead of using the mean probabilities, the staff had used the values of probabilities of earthquake-initiated release categories from the high estimates, then: (1) the total risks of early fatality would be increased by a factor of about 6 (because the high estimates of probabilities of the earthquake-initiated release categories are about 6 times higher than the mean values); (2) the total risk of early injury would be increased by a factor of about 4; and (3) the other types of risks would be increased by factors of about 2. On the other hand, if the staff had used the low estimates of probabilities of the earthquake-initiated release categories (which are lower than the mean values by several orders of magnitudes), then the contributions to the risks from these release categories would be negligible compared to those from the release categories initiated by causes other than severe earthquakes. Therefore, use of the full range of probabilities of earthquake-initiated release categories would result in spreads in the staff's risk estimates; values of the risks would fall within ranges of about one-thirtieth to about 6 times the values depicted in Tables 5-11h, L-1a and b, and M-1a and b. We do not mean to imply that higher risk estimates are more appropriate than the median, mean or lower estimates. Indeed the most significant earthquake damage anywhere within the vicinity of the Limerick Site, in the two to three hundred years during which we have records, are fallen chimneys 50 kilometers away during an earthquake at Wilmington, Delaware in 1871 whose magnitude can be estimated to have been less than 5.0. We certainly cannot exclude from the range of reasonable assumptions the judgment that there essentially is no risk to the public resulting from earthquake-induced damage at the seismically-engineered nuclear power plant at Limerick during its operating life.

Overall, accident probabilities may be expressed in terms of the probability of core melt, and considered an important measure of the likelihood of environmental and human impacts from severe reactor accidents. To provide some perspective on the uncertainty in such estimates, Figure 5.4m compares the estimate of core melt probabilities and their uncertainties based on contemporary PRA-based estimates for several different reactors. Except for Limerick, the results presented on Figure 5.4m are taken directly from published PRAs without modification (Rowsome and Blond, 1982). The results for Limerick are based on staff contractor estimates for Limerick (NUREG-3028). The PRAs were not necessarily performed using consistent methodologies or assumptions, and some of the PRAs evaluate designs that have subsequently been altered. Caution should be exercised when using these results because there are very large uncertainties in these analyses. No attempt has been made to adjust the results to compensate for inconsistency of approach or methods. Therefore, the appropriateness of the comparison may be in question. However, all of the studies have analyzed, in roughly the same manner, the so-called "internally" initiated events.

Quantity and Chemical Form of Radioactivity Released

The models used in these calculations contain approximations to describe the physical behavior of the radionuclides which affects the transport within the reactor vessel and other plant structures and the amounts of release. This relates to the quantity and chemical form of each radionuclide species that would be released from a reactor unit during a particular accident sequence. Such releases would originate in the fuel and would be attenuated by physical and chemical processes in route to being released to the environment. Depending on the accident sequence, attenuation in the reactor vessel, the primary cooling system, the containment, and adjacent buildings would influence both the magnitude and chemical form of radioactive releases. The releases of radionuclides to the environment, called source terms, used in the staff analysis were determined using the RSS methodology applicable to a BWR of Peach Bottom design; therefore, the RSS methodology may not have been fully appropriate for the Limerick BWRs. Information available in NUREG-0772 and from the latest research activities sponsored by the Commission and the industry indicates that source terms used in the staff analysis cannot be much higher in the maximum, but could be substantially lower. Some lower source term values could be higher also, primarily because of the manner in which the source term was evaluated for early releases using the RSS methodology. The impact of lesser values of source terms would be substantially lower estimates of health effects, particularly early fatalities and injuries. The source terms resulting from the applicants PRA would, for example, yield significantly lower estimates of risk than those used by the staff in this report. The NRC staff anticipates better source term information at the end of 1984 when the staff's Accident Source Term Program Office and the American Physical Society complete their studies.

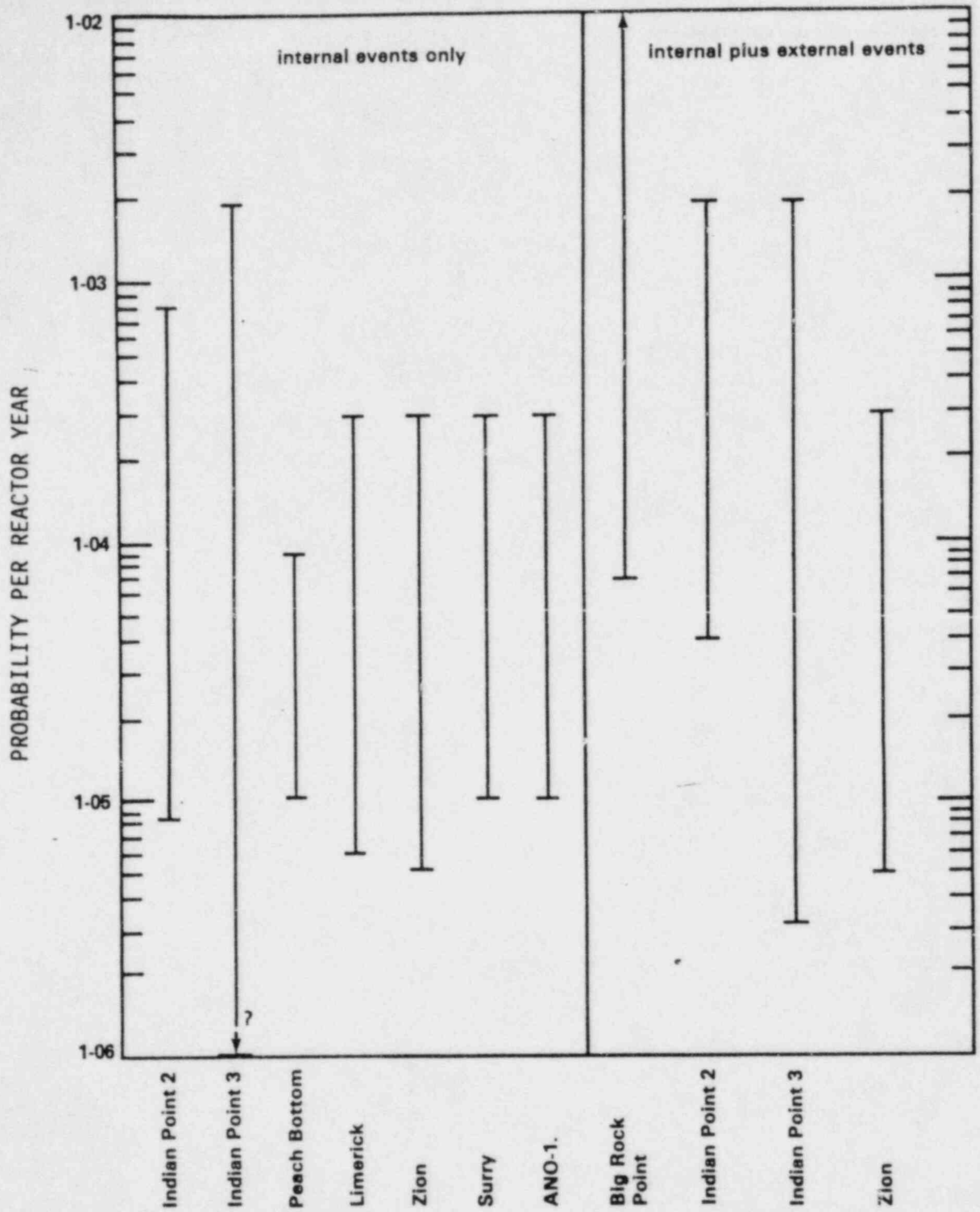

Figure 5.4m Core melt probability uncertainty bounds for internal events and internal plus external events

Atmospheric Dispersion Modeling for the Radioactive Plume Transport, Including the Physical and Chemical Behavior of Radionuclides in Particulate Form in the Atmosphere

This uncertainty is due to differences between the modeling of the atmospheric transport of radioactivity in gaseous and particulate states in the CRAC code and the actual transport, diffusion and deposition or fallout that would occur during an accident (including the effects of precipitation). The phenomenon of plume rise because of heat that is associated with the atmospheric release, effects of precipitation on the plume, and fallout of particulate matter from the plume all have considerable impact on both the magnitude of early health consequences and the distance from the reactor to which these consequences would occur. The staff judgment is that these factors can result in substantial overestimates or underestimates of both early and later effects (health and economic).

Errors of Completeness, Modeling, Arithmetic, and Omission

This area of lumped uncertainty includes such topics as the omission of a model of sabotage, modeling errors in event trees, common cause failures other than those originating in external events or fires, improvements in design or operating criteria undertaken or to be undertaken by the applicant, potential errors in the different models used to assess risks, statistical errors, and arithmetic errors. The impact on risk estimates of this class of uncertainty could be large, but is unknown and virtually impossible to quantify accurately (Rowsome, 1982). Because of the depth to which the applicant and the staff have considered risks for Limerick, however, uncertainties of this type are not expected to be as large as for other reactors for which less comprehensive probabilistic risk assessments have been performed.

Other areas that have substantial but relatively less effect on uncertainty than the preceeding items are

- Duration and Energy of Release, Warning Time, and Inplant Radionuclide Decay Time

  The assumed release duration, energy of release, and the warning and the inplant radioactivity decay times may differ from those that would actually occur during a real accident.

  For a relatively long duration (greater than a half-hour) of an atmospheric release, the actual cross-wind spread (the width) of the radioactive plume that would develop would likely be larger than the width calculated by the dispersion model in CRAC. However, the effective width of the plume is calculated in the code using a plume expansion factor that is determined by the release duration. For a given quantity of radionuclides in a release, the plume and, therefore, the area that would come under its cover would become wider if the release duration were made longer. In effect, this would result in lower air and ground concentrations of radioactivity but a greater area of contamination.

The thermal energy associated with the release affects the plume rise phenomenon, which results in relatively lower air and ground concentrations in the closer-in regions and relatively higher concentrations as a result of fallout in the more distant regions. Therefore, if a large amount of thermal energy were associated with a release containing large fractions of core inventory of radionuclides, the distance from the reactor over which early health effects may occur is likely to be increased.

Warning time before evacuation has considerable impact on the effectiveness of offsite emergency response. Longer warning times would improve the effectiveness of the response.

The time from reactor shutdown until the beginning of the release to the environment (atmosphere), known as the time of release, is used to calculate the depletion of radionuclides by radioactive decay within the plant before release. The depletion factor for each radionuclide (determined by the radioactive decay constant and the time of release) multiplied by the release fraction of the radionuclide and its core inventory determines the actual quantity of the radionuclide released to the environment. Longer release times would result in release of fewer curies to the environment for given values of release fractions.

The first three of the parameters discussed above can have significant impacts on accident consequences, particularly early consequences. The staff judgment is that the estimates of early consequences and risks could be substantially exceeded, or could be substantial overestimates, because of uncertainties in the first three parameters.

Meteorological Sampling Scheme Used

The meteorological sequences used with the selected 91 start times (sampling) in the CRAC code may not adequately represent all meteorological variations that may occur over the life of the plant. This factor is judged to produce greater uncertainties for early effects and less for latent effects.

Emergency Response Effectiveness

The modeling assumptions of the emergency response of the people residing around the Limerick site may not correspond to what would happen during an actual severe reactor accident. Included in these considerations are such subjects as evacuation effectiveness under different circumstances, possible sheltering and its effectiveness, and the effectiveness of population relocation. The staff judgment is that the uncertainties associated with emergency response effectiveness could cause large uncertainties in estimates of early health consequences. The uncertainties in estimates of latent health consequences and costs are considered smaller than those of early health consequences. A limited sensitivity analysis in this area is presented in Appendix M. It indicates that for release categories initiated by causes other than severe earthquakes, the risk of early fatality with supportive or minimal medical treatment would be increased by factors of less than 5, if people from within the plume exposure pathway EPZ would not evacuate to evade the plume but would wait for the plume to leave the area and then relocate from the contaminated ground after a time interval equal to the evacuation time assumed for the Limerick site. Under the same assumptions, increases in risks of other health effects would be less. However, the increase in risks of all health effects from release categories initiated by all causes (severe earthquakes and other causes) taken together would be within about 20%.

- Dose Conversion Factors and Dose Response Relationships for Early Health Consequences, Including Benefits of Medical Treatment

  There are many uncertainties associated with estimates of dose and early health effects on individuals exposed to high levels of radiation. Included are the uncertainties associated with the conversion of contamination levels to doses, relationships of doses to health effects, and considerations of the availability of what was described in the RSS as supportive medical treatment (a specialized medical treatment program of limited resources that would minimize the early health effect consequences of high levels of radiation exposure following a severe reactor accident). The staff analysis shows that the variation in estimates of early fatality risks stemming from considerations of supportive medical treatment alone is less than a factor of 3 for the Limerick site.

- Dose Conversion Factors and Dose Response Relationships for Latent Health Consequences

  In comparison to early health effects, there are even larger uncertainties associated with dose estimates and latent (delayed and long-term) health effects on individuals exposed to lower levels of radiation and on their succeeding generations. Included are the uncertainties associated with conversion of contamination levels to doses and doses to health effects. The staff judgment is that this category has a large uncertainty. The uncertainty could result in relatively small underestimates of consequences, but it also could result in substantial overestimates of consequences. (Note: radiobiological evidence on this subject does not rule out the possibility that low level radiation could produce zero consequences.)

- Chronic Exposure Pathways, Including Environmental Decontamination and the Fate of Deposited Radionuclides

  Uncertainties are associated with chronic exposure pathways to people from long-term use of the contaminated environment. Uncertainty also arises from the possibility that the protective action guide levels that may actually be used for interdiction or decontamination of the exposure pathways may differ from those assumed in the staff analysis. Further, uncertainty arises as a result of the lack of precise knowledge about the fate of the radionuclides in the environment as influenced by such natural processes as runoff, weathering, etc. The staff's qualitative judgment is that the uncertainty from these considerations is substantial.

- Economic Data and Modeling

  There are uncertainties in the economic parameters and economic modeling, such as costs of evacuation, relocation, medical treatment, cost of decontamination of properties, and other costs of property damage. Uncertainty in this area could be substantial.

- Fission Product Inventory

  The fission product inventory presented in Table 5.11a is an approximation of that which would be present after extended operation at maximum power. The amount of each isotope listed will, in fact, vary with time in a manner dependent upon the fuel management scheme and the power history of the core. The actual inventory at the time of an accident could not be much larger for any isotope than the amount in Table 5.11a, but, especially for long-lived fission products, could be substantially smaller.

The means for quantitative evaluation of the uncertainties in a probabilistic risk analysis such as the type presented here are not well developed. The staff, however, has attempted to identify all sources of uncertainty, and to assess the net effect upon the uncertainty of the risk estimates. Based upon the insight gained from the review of similar PRAs for Indian Point and Zion, it is the judgment of the staff that the risk estimates for Limerick could be too low by a factor of about 40 or too high by a factor of about 400. The risk estimates are equal to the integrals of the corresponding probability distributions of the consequences (CCDFs). As a result, errors in probabilities and consequences are partially offset. Because of the magnitude of uncertainties, the staff has concluded that estimates of the absolute magnitudes of probabilities, consequences, and risks do not provide an accident perspective unless the uncertainties are also considered.

When the accident at Three Mile Island occurred in March 1979, the accumulated experience record was about 400 reactor-years. It is of interest to note that this was within the range of frequencies estimated by the RSS for an accident of this severity (National Research Council, 1979, p. 553). It should also be noted that the Three Mile Island accident has resulted in a very comprehensive evaluation of similar reactor accidents by a number of investigative groups both within and outside of the NRC. Actions to improve the safety of nuclear power plants have resulted from these investigations, including those from the President's Commission on the Accident at Three Mile Island and from NRC staff investigations and task forces. A comprehensive "NRC Action Plan Developed as a Result of the TMI-2 Accident" (NUREG-0660, Vol I) collects the various recommendations of these groups and describes them under the subject areas of: Operational Safety; Siting and Design; Emergency Preparedness and Radiation Effects; Practices and Procedures; and NRC Policy, Organization, and Management. NUREG-0737, "Clarification of TMI Action Plan Requirements," and Supplement 1 to NUREG-0737 identified those requirements that were approved for implementation. The action plan presents a sequence of actions, some already taken, that results in a gradually increasing improvement in safety as individual actions are completed. The Limerick units are receiving and will receive the benefit of these actions on the schedule discussed in the SER. The improvement in safety from these actions has not been quantified, however.

(8) Comparison of Limerick Risks with Other Plants

To provide a perspective as to how the Limerick reactors compare in terms of risks from severe accidents with some of the other nuclear power plants that are either operating or that are being reviewed by the staff for possible issuance of a license to operate, the estimated risks from severe accidents for several nuclear power plants (including those for Limerick) are shown in Figures 5.4n through 5.4v for three important categories of risk. The values for individual

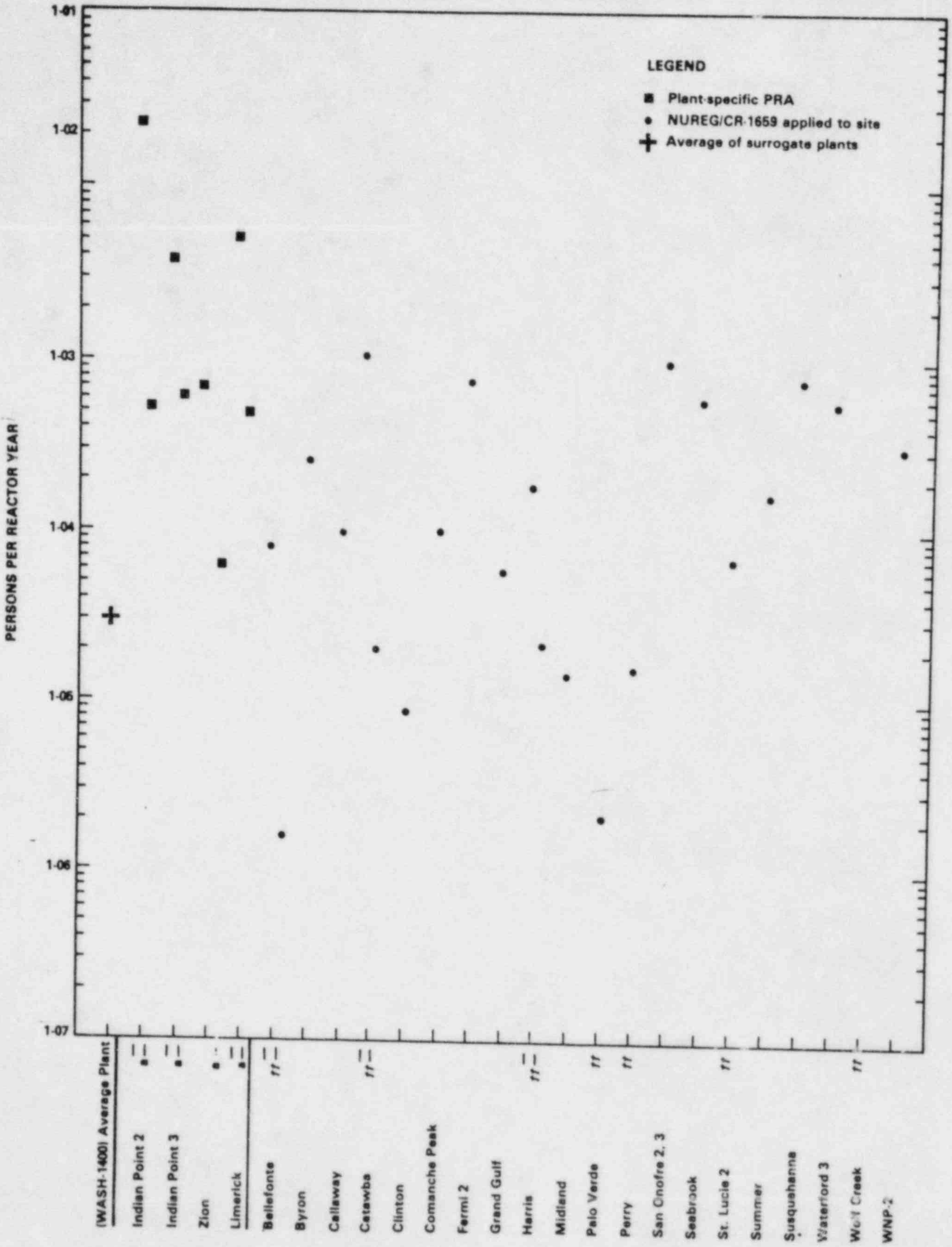

Figure 5.4n Estimated early fatality risk with supportive medical treatment (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate. See footnotes following Figure 5.4v.

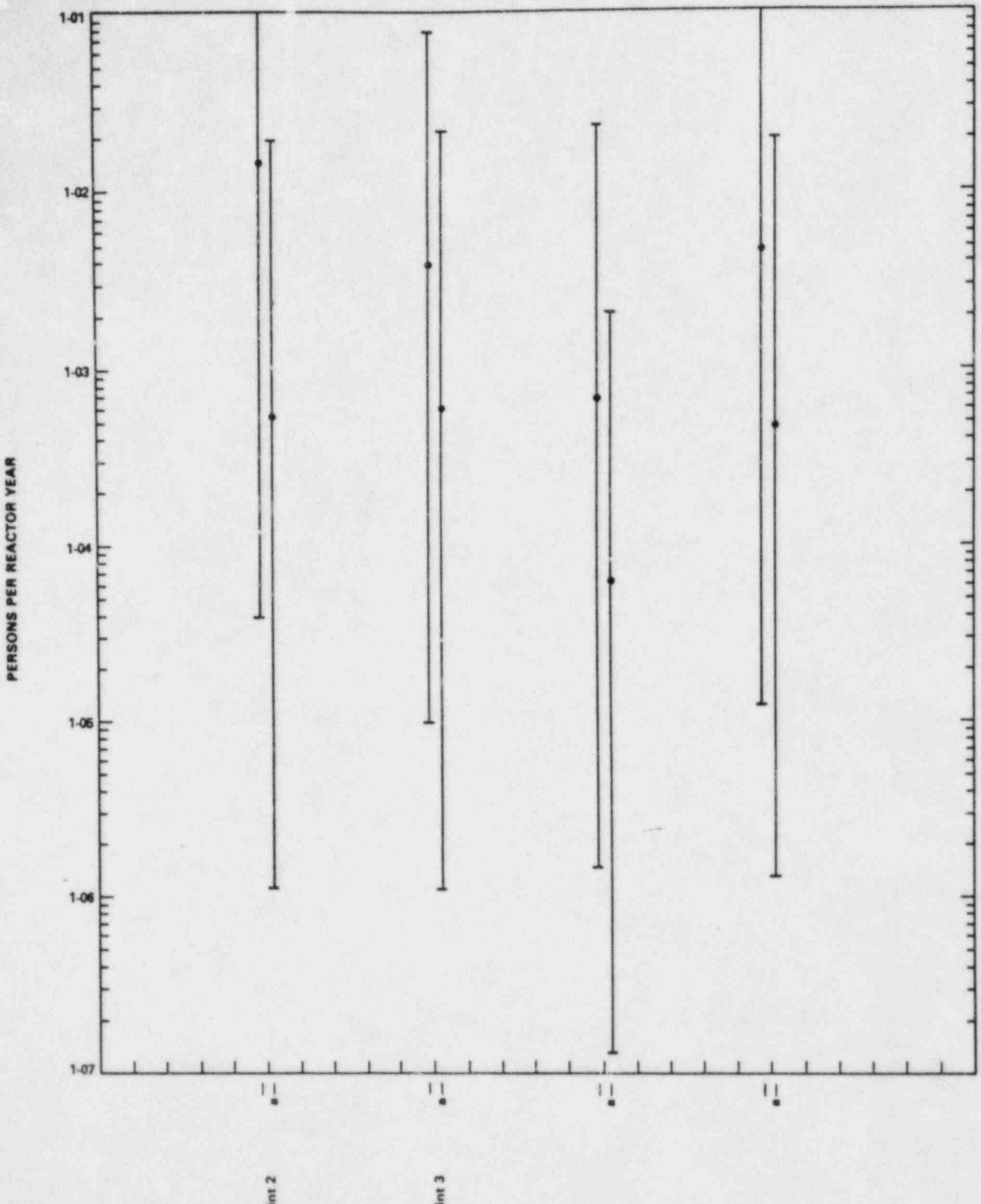

Figure 5.4o Estimated early fatality risk with supportive medical treatment (persons) from severe reactor accidents for nuclear power plants having plant-specific PRAs, showing estimated range of uncertainties. See footnotes following Figure 5.4v.

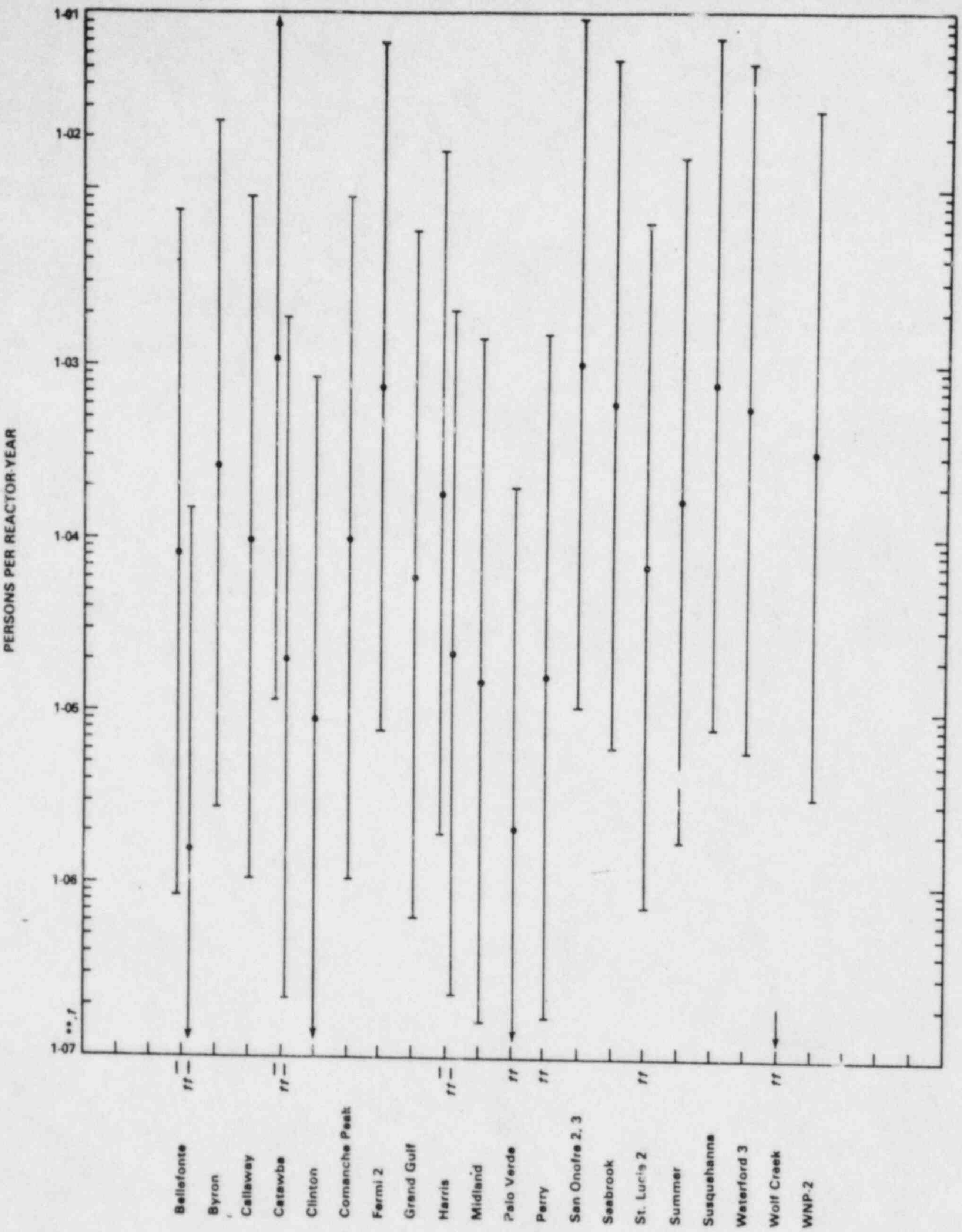

Figure 5.p Estimated early fatality risk with supportive medical treatment (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate for which site-specific applications of NUREG/CR-1695 accident releases have been used to calculate off-site consequences. Bars are drawn to illustrate effect of uncertainty range discussed in text. See footnotes following Figure 5.4v.

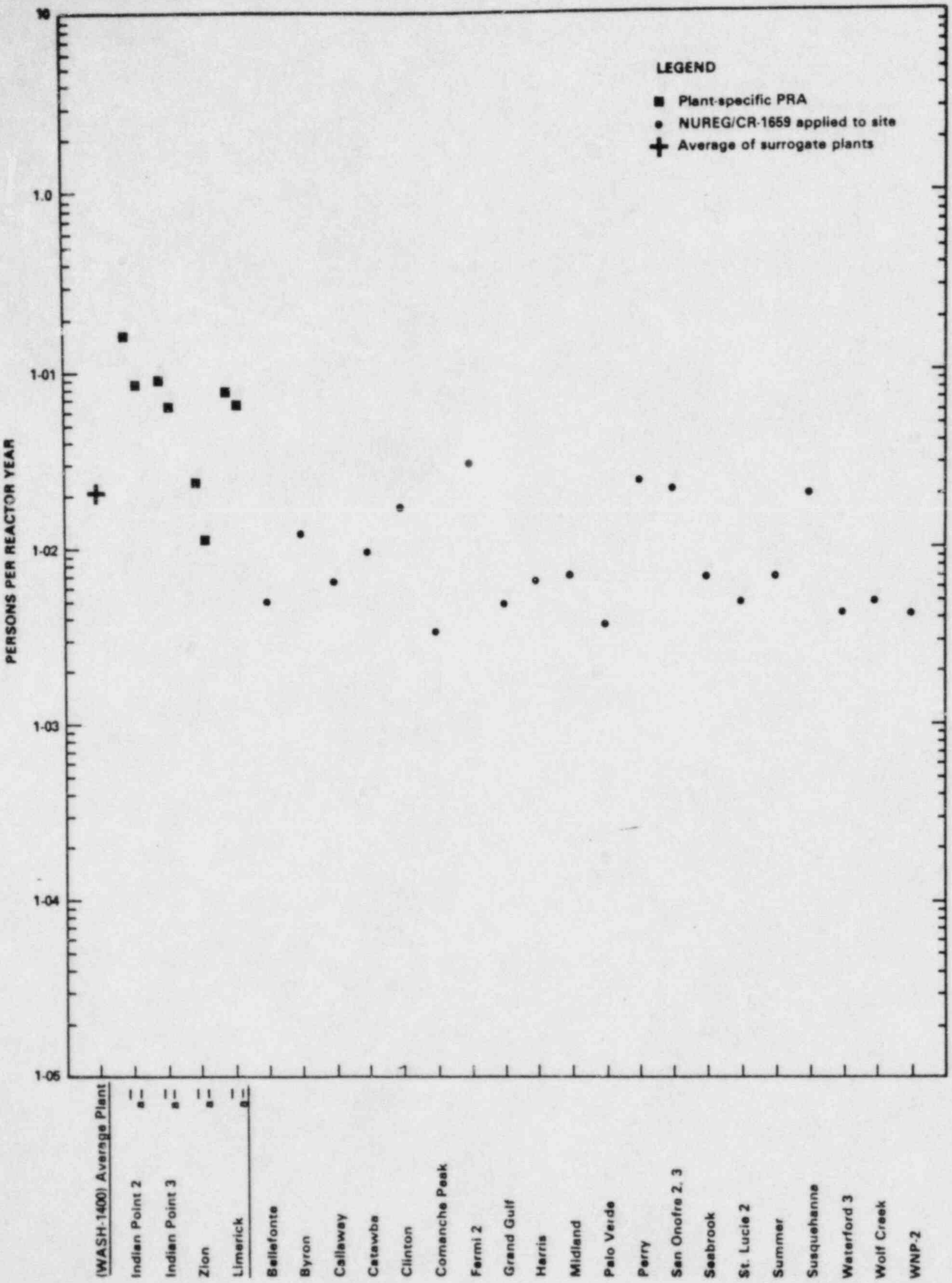

Figure 5.4q Estimated latent cancer fatality risk, excluding thyroid (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate. See footnotes at the end of Figure 5.4v.

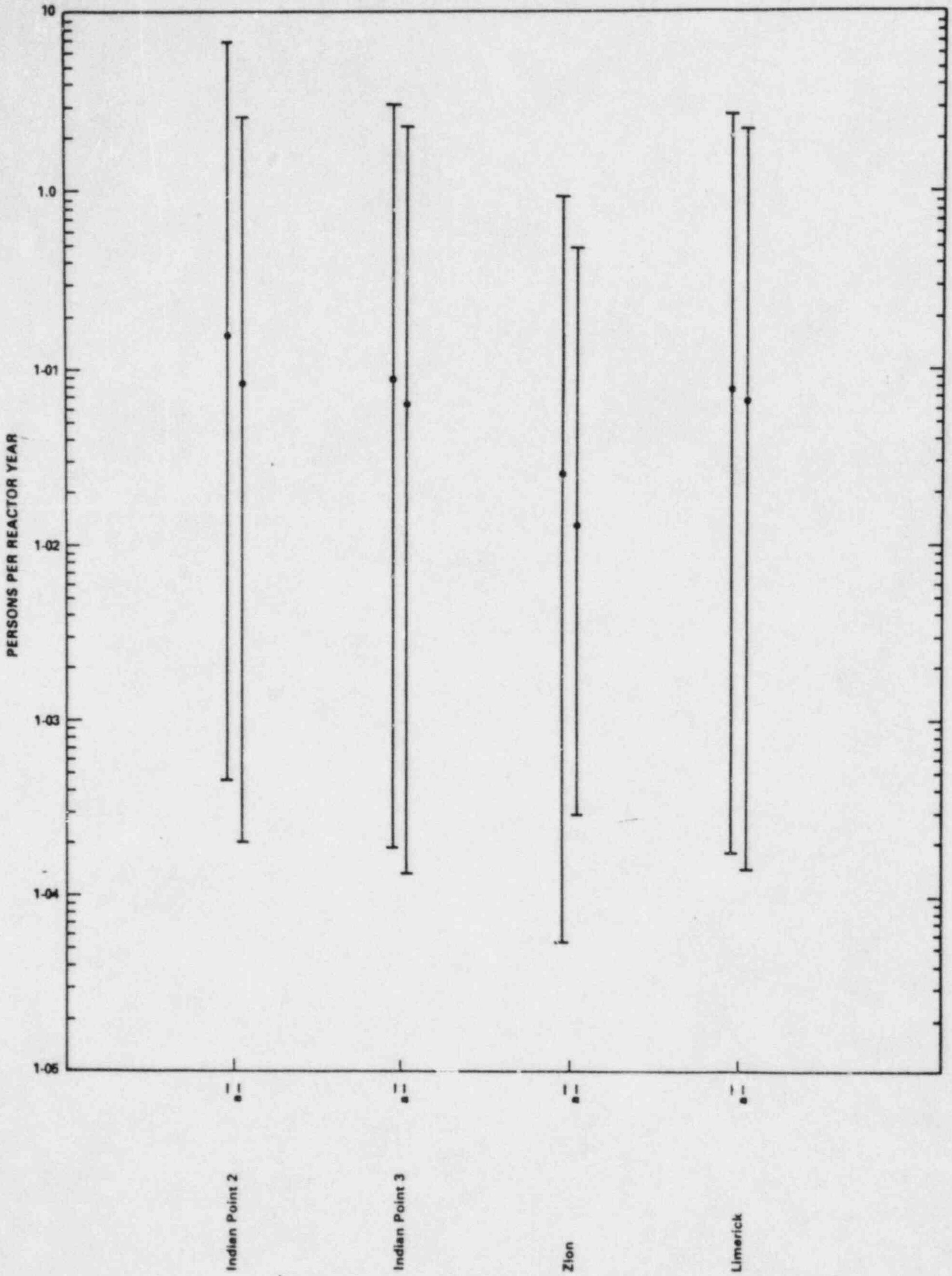

Figure 5.4r Estimated latent cancer fatality risk, excluding thyroid (persons) from severe reactor accidents for nuclear power plants having plant-specific PRAs, showing estimated range of uncertainties. See footnotes at the end of Figure 5.4v.

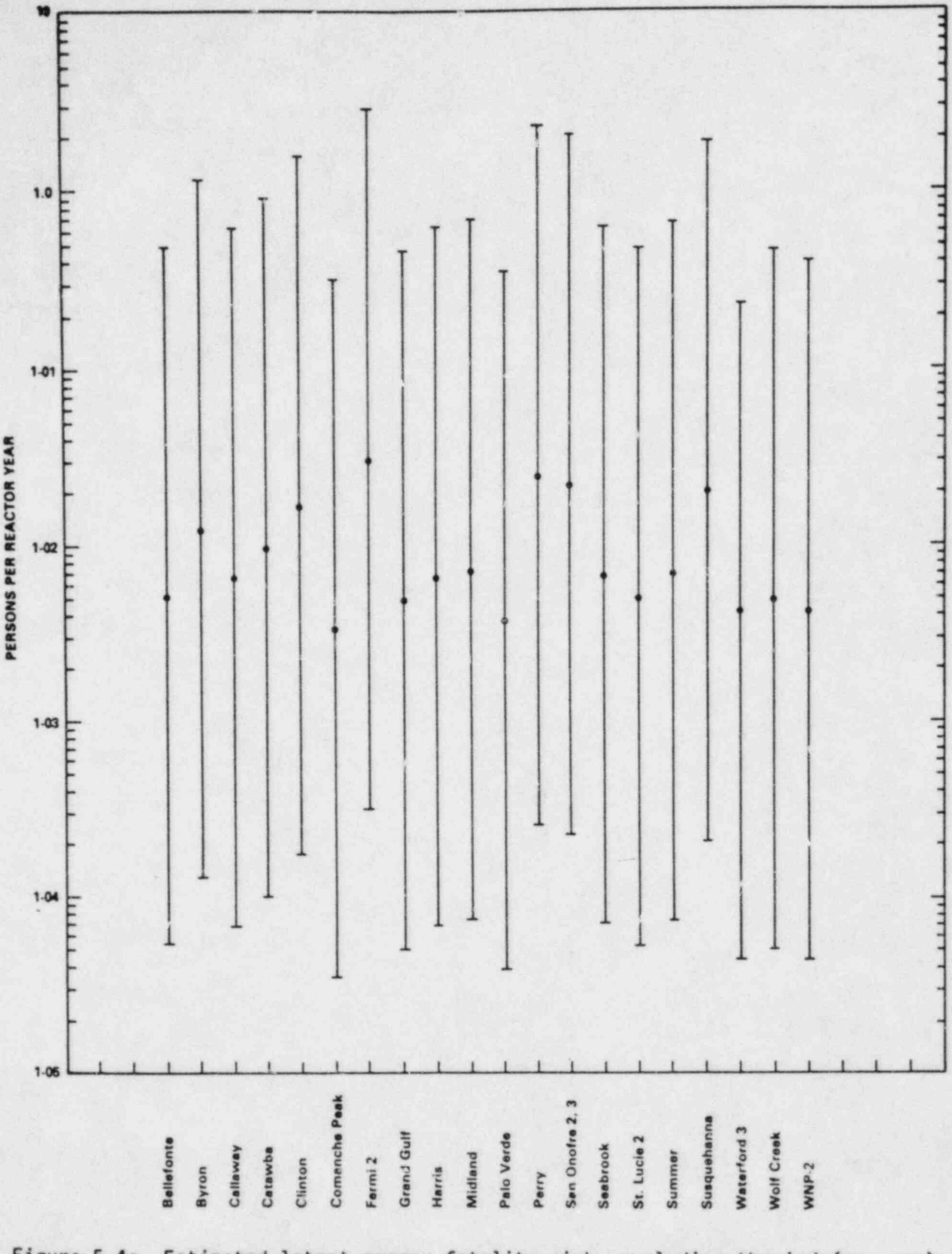

Figure 5.4s Estimated latent cancer fatality risk, excluding thyriod (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate for which site-specific applications of NUREG/CR-1695 accident releases have been used to calculate off-site consequences. Bars are drawn to illustrate effect of uncertainty range discussed in text. See footnotes at the end of Figure 5.4v.

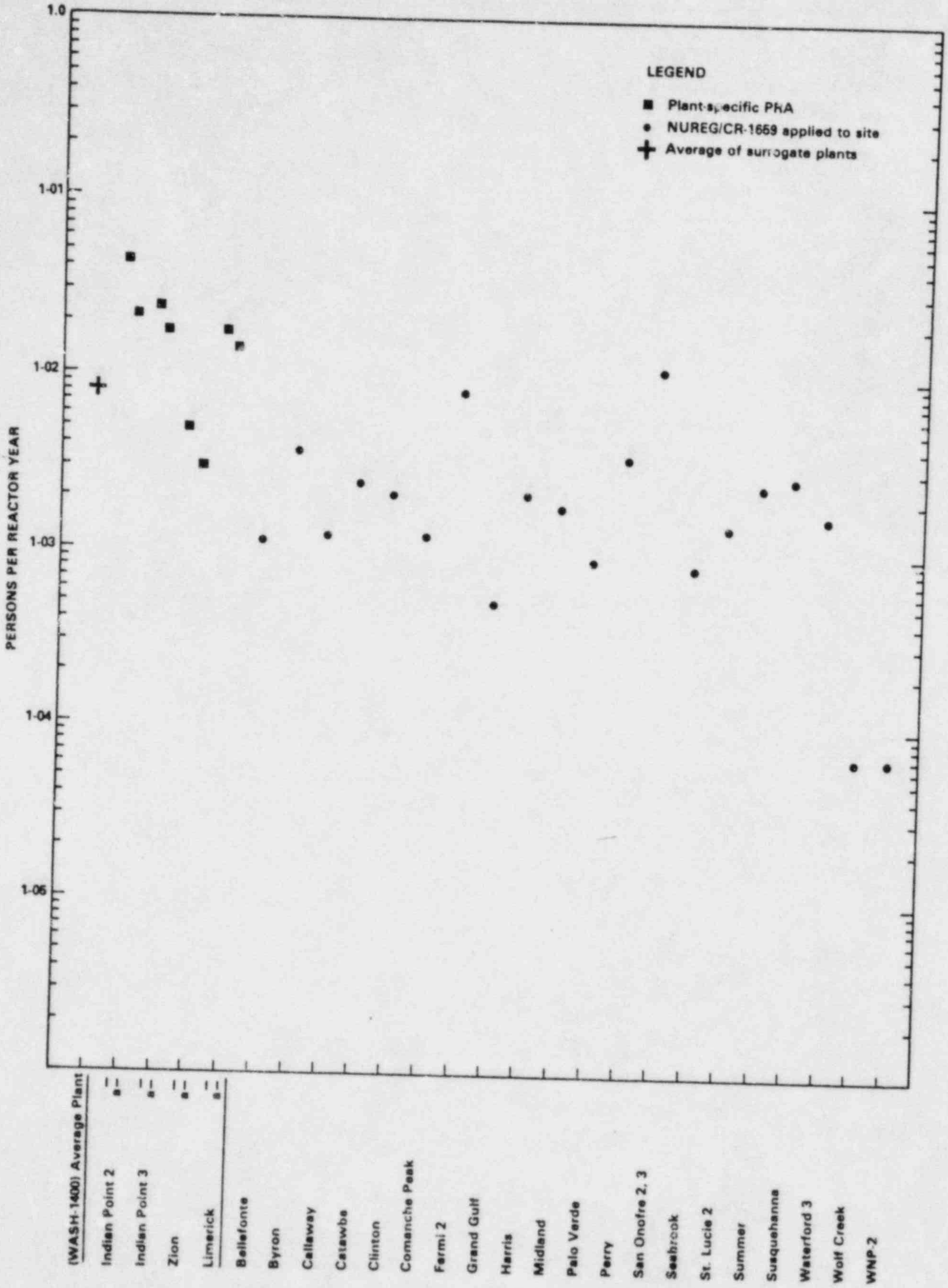

Figure 5.4t Estimated latent thyroid cancer fatality risk (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate. See footnotes at the end of Figure 5.4v.

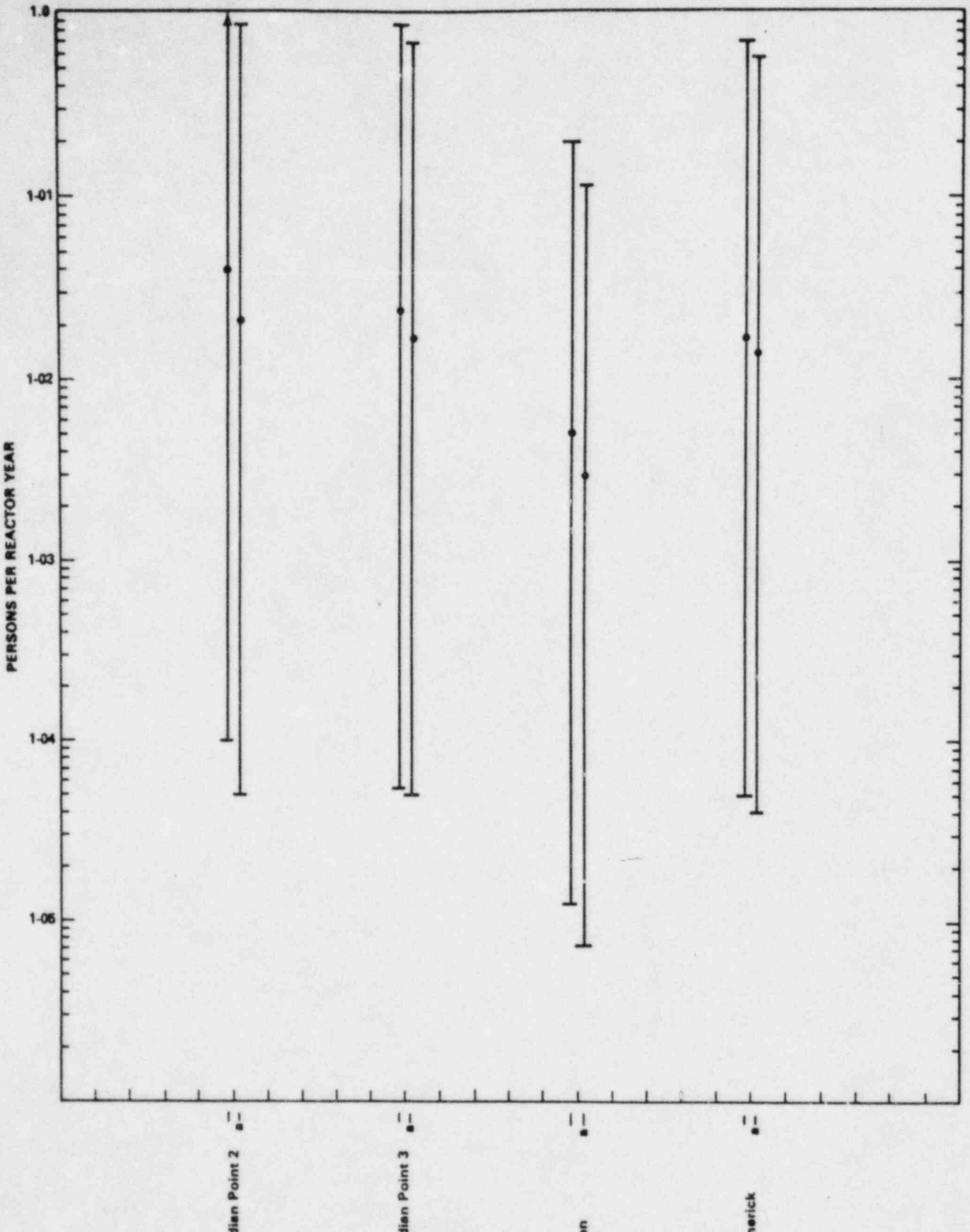

Figure 5.4u Estimated latent thyroid cancer fatality risk (persons) from severe reactor accidents for nuclear power plants having plant-specific PRAs, showing estimated range of uncertainties. See footnotes at the end of Figure 5.4v.

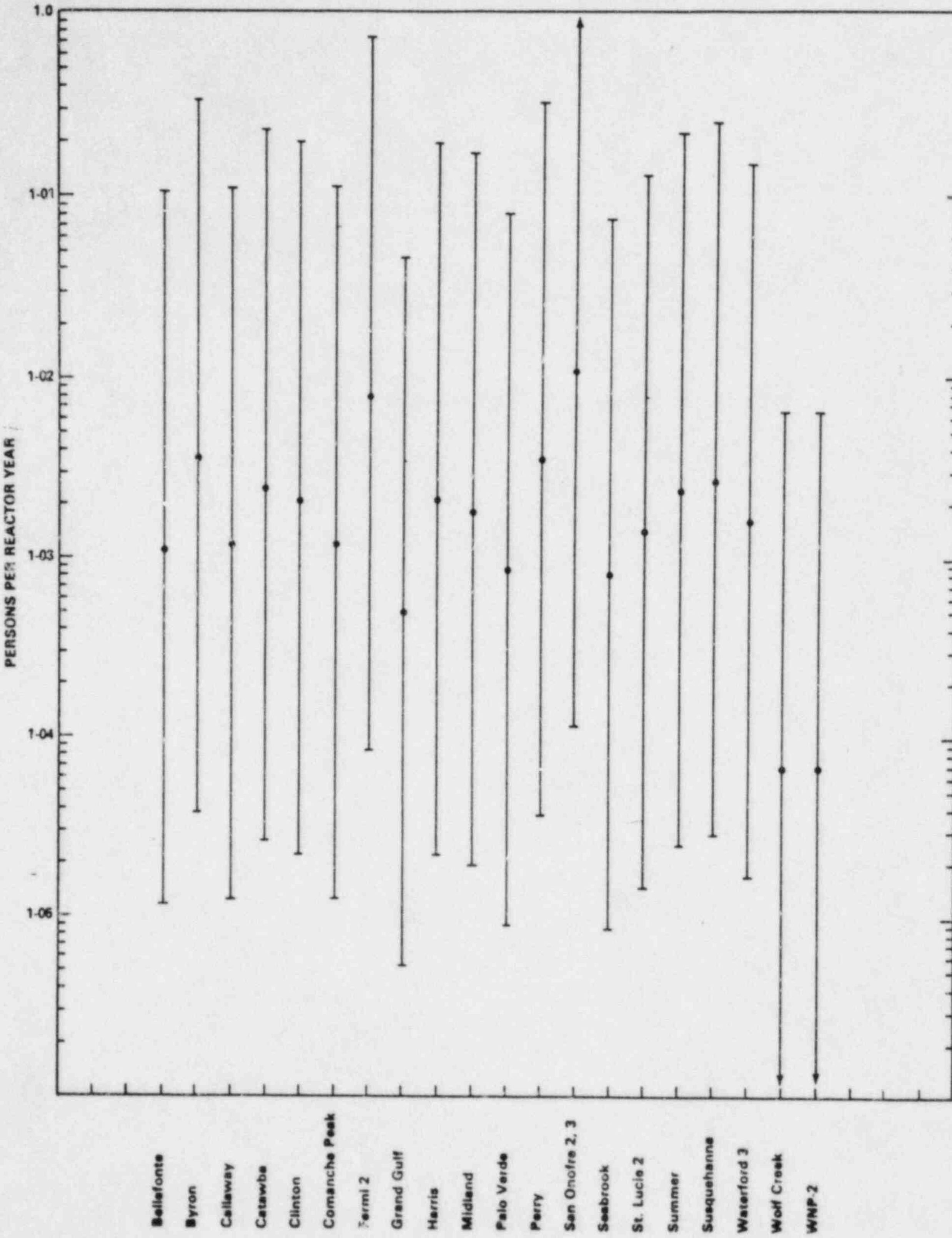

Figure 5.4v Estimated latent thyroid cancer fatality risk (persons) from severe reactor accidents from several nuclear power plants either operating or receiving consideration for issuance of license to operate for which site-specific applications of NUREG-1695 accident releases have been used to calculate off-site consequences. Bars are drawn to illustrate effect of uncertainty range discussed in text. See footnotes on following page.

Notes for Figures 5.4n through 5.4v

- Except for Indian Point, Zion, and Limerick, risk analyses for other plants in these figures are based on WASH-1400 generic source terms and probabilities for severe accidents and do not include external event analyses. Any or all of the values could be under or over-estimates of the true risks.
- $1\text{-}01 = 1 \times 10^{-1}$

†Assumes evacuation to 25 miles.

††With evacuation within 10 miles and relocation from 10-25 miles.

[a]Excluding severe earthquakes and hurricanes.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties.

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plants are based upon three types of estimates: from the RSS (labeled WASH-1400 Average Plant), from independent staff reviews of contemporary probabilistic risk assessments (Indian Point 2 and 3, Zion and Limerick), and from generic applications of RSSMAP accident sequences to reactor sites for environmental statements by the staff (for 21 nuclear power plants). The RSS risk estimates were intended to illustrate the general level of risk from a variety of plant designs at a variety of sites, and these estimates appear in Figures 5.4n, q and t as point estimates along with the corresponding point estimates obtained by the other types of analysis. Figures 5.4o, r and u show the range of uncertainty that is estimated for those four plants for which a plant-specific probabilistic risk assessment has been performed. Figures 5.4p, s and v are included to illustrate the effect uncertainties of a factor of 100 would have upon comparison amongst risk estimates using a fixed set of accident sequences, but site-specific meteorology and population. The display of risk in three sets of figures is intended to allow comparison of risks similarly evaluated, and to allow an overall comparison of risks to be made among all types of risk evaluations available. Figures 5.4n through 5.4v indicate that the estimated Limerick risks may be higher than those for some plants, and lower than those for several other plants but, except for early fatalities at the Wolf Creek site, not by a margin that would exceed the uncertainties in the estimates themselves. Similarly, Figure 5.4m, which compares core melt probabilities for Limerick with several other reactors, indicates that the estimated likelihood of a core melt accident at Limerick is roughly the same as for several operating reactors. Furthermore, any or all of the estimates of risk could be under or overestimates.

5.9.4.6 Conclusions

The foregoing sections consider the potential environmental impacts from accidents at Limerick station. These have covered a broad spectrum of possible accidental releases of radioactive materials into the environment by atmospheric and liquid pathways. Included in the considerations are postulated design-basis accidents and more severe accident sequences that lead to a severely damaged reactor core or core melt. The applicant also considered similar accidents in the ER-OL. The staff has considered the technical merits of the applicant's assessment and the uncertainties involved, and agrees in several areas and disagrees in several other areas (see Appendix N). Notable disagreements are in the area of source terms and offsite emergency response modeling. For several sequences the staff's source terms are considerably higher; the offsite emergency response modeling is site specific and more pessimistic for severe earthquake conditions in the site region than that modeled by the applicant. As a result, the applicant's risk estimates are substantially lower than the staff estimates. In both the applicant's and the staff's analyses of accident risk, however, there are very large uncertainties.

This section documents the staff's use of PRA in its inquiry into the environmental impacts of reactor accidents. The staff's inquiry into the implications of the risk assessments for reactor design and operation; to wit, questions of compliance with the reactor safety regulations and the questions of whether plant-specific vulnerabilities to severe accidents warrant requirements more stringent than the norm, will be documented elsewhere.

The environmental impacts that have been considered include potential radiation exposures to individuals and to the population as a whole, the estimated likelihood of core melt accidents, the risk of near- and long-term adverse health effects that such exposures could entail, and the potential economic and societal consequences of accidental contamination of the environment. These impacts could be severe, but the likelihood of their occurrence is judged to be small and comparable to that of other reactors. This conclusion is based on (1) the fact that considerable experience has been gained with the operation of similar facilities without significant degradation of the environment, (2) the fact that, to obtain a license to operate, the Limerick station must comply with the applicable Commission regulations and requirements, (3) a comparison with the estimated core melt probabilities of other reactors, and (4) a probabilistic assessment of the risk based upon the methodology developed in the RSS, improvements on the RSS methodology including external event analysis, and a sensitivity analysis of offsite emergency response modeling. The overall assessment of environmental risk of accidents, assuming protective actions, shows that the risks of population exposure and latent cancer fatality are within a factor of 30 of those from normal operation. Accidents have a potential for early fatalities and economic costs that cannot arise from normal operations; however, the risks of early fatality from potential accidents at the site are small in comparison with risks of early fatality from other human activities in a comparably sized population, and the accident risk will not add significantly to population exposure and cancer risks. Accident risks from Limerick are expected to be a small fraction of the risks the general public incurs from other sources. Further, the best estimate calculations show that the risks of potential reactor accidents at Limerick are within the range of such risks from other nuclear power plants.

Based on the foregoing considerations of environmental impacts of accidents, which have not been found to be significant, the staff has concluded that there are no special or unique circumstances about the Limerick site and environs that would warrant consideration of alternatives for Limerick Units 1 and 2.

5.10 Impacts from the Uranium Fuel Cycle

The Uranium Fuel Cycle rule, 10 CFR 51.20 (44 FR 45362), reflects the latest information relative to the reprocessing of spent fuel and to radioactive waste

6 EVALUATION OF THE PROPOSED ACTION

6.1 Unavoidable Adverse Impacts

The staff has reassessed the physical, social, biological and economic impacts that can be attributed to the operation of the Limerick generating station. These impacts are summarized in Table 6.1.

The applicant is required to adhere to the following conditions for the protection of the environment:

(1) Before engaging in any additional construction or operational activities that may result in any significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in this statement, the applicant will provide written notification of such activities to the Director of the Office of Nuclear Reactor Regulation and will receive written approval from that office before proceeding with such activities.

(2) The applicant will carry out the environmental monitoring programs outlined in Section 5 of this statement, as modified and approved by the staff and implemented in the Environmental Protection Plan and Technical Specifications that will be incorporated in the operating licenses.

(3) If an adverse environmental effect or evidence of irreversible environmental damage is detected during the operating life of the plant, the applicant will provide the staff with an analysis of the problem and a proposed course of action to alleviate it.

6.2 Irreversible and Irretrievable Commitments of Resources

There has been no change in the staff's assessment of this impact since the earlier review except that the continuing escalation of costs has increased the dollar values of the materials used for constructing and fueling the plant.

6.3 Relationship Between Short-Term Use and Long-Term Productivity

There have been no significant changes in the staff's evaluation for the Limerick generating station since the construction permit stage environmental review.

6.4 Benefit-Cost Summary

6.4.1 Summary

Sections below describe the economic, environmental and socioeconomic benefits and costs that are associated with the operation of the Limerick generating station. They are summarized in Table 6.1.

Table 6.1 Benefit-cost summary for Limerick

Primary impact and effect on population or resources	Quantity (Section)*	Impacts**
BENEFITS		
Direct		
Electrical energy	10 billion kWh/yr	Large
Additional generating capacity	2110 MWe (design rating) (Sec. 6.4.1)	Large
COSTS		
Environmental		
Damages suffered by other water users		
Surface water consumption	(Sec. 5.3.2)	Small
Surface water contamination	(Sec. 5.3.2)	Small
Groundwater consumption	(Sec. 4.3.2)	None
Groundwater contamination	(Sec. 4.3.2)	None
Damage to aquatic resources		
Impingement and entrainment	(Sec. 5.5.2)	Small
Thermal effects	(Secs. 5.3.2 & 5.5.2)	Small
Chemical discharges	(Sec. 5.3.2)	Small
Diversion flow effects (East Branch)	(Sec. 5.5.2.3)	Moderate
Damage to terrestrial resources		
Station operations	(Sec. 5.5)	Small
Transmission line maintenance	(Sec. 5.5.1)	Small
Adverse socioeconomic effects		
Loss of historic or archeological resources	(Sec. 5.7)	Moderate
Increased demands on public facilities and services	(Sec. 5.8)	Small
Increased demands on private facilities and services	(Sec. 5.8)	Small
Noise	(Sec. 5.12)	Moderate-Small
Adverse nonradiological health effects		
Water quality changes	(Sec. 5.3.2)	None
Air quality changes	(Sec. 5.4)	

*See footnotes at end of table.

Table 6.1 (Continued)

Primary impact and effect on population or resources	Quantity (Section)*	Impacts**
Adverse radiological health effects		
Routine operation	(Sec. 5.9.3)	Small
Design basis accidents	(Sec. 5.9.4)	Small
Severe accident risks	(Sec. 5.9.4)	Small
Uranium fuel cycle	(Sec. 5.10)	Small

*Where a particular unit of measure for a benefit/cost category has not been specified in this statement or where an estimate of the magnitude of the benefit/cost under consideration has not been made, the reader is directed to the appropriate section of this report for further information.

**Subjective measure of costs and benefits is assigned by reviewers, where quantification is not possible: "Small" = impacts that in the reviewers' judgments, are of such minor nature, based on currently available information, that they do not warrant detailed investigations or considerations of mitigative actions; "Moderate" = impacts that in the reviewers' judgments are likely to be clearly evident (mitigation alternatives are usually considered for moderate impacts); "Large" = impacts that in the reviewers' judgments, represent either a severe penalty or a major benefit. Acceptance requires that large negative impacts should be more than offset by other overriding project considerations.

6.4.2 Benefits

A major benefit to be derived from the operation of the Limerick station is the approximately 10 billion kWh of baseload electrical energy that will be produced annually (this projection assumes that both units will operate at an annual average capacity factor of 55%). The addition of the plant will also improve the applicant's ability to supply system load requirements by contributing 2110 MW of generating capacity to the Philadelphia Electric Company system (1055 MW from Unit 1 in 1985 and 1055 MW from Unit 2 in 1989).

6.4.3 Costs

No significant socioeconomic costs are expected from either the operation of the Limerick generating station or from the number of station personnel and their families living in the area. The socioeconomic impacts of a severe accident could be large; however, the probability of such an accident is small.

6.5 Conclusion

As a result of its analysis and review of potential environmental, technical, and social impacts, the NRC staff has prepared an updated forecast of the effects of operation of the Limerick generating station. The NRC staff has determined that the Limerick generating station can be operated with minimal environmental impact. To date, no new information has been obtained that alters the overall favorable balancing of the benefits of station operation versus the environmental costs that resulted from evaluations made at the construction permit stage.

6.6 Reference

U.S. Nuclear Regulatory Commission, NUREG-0586, "Draft Generic Environmental Impact Statement on Decommissioning Nuclear Facilities," January 1981.

APPENDIX H

LIMERICK ACCIDENT SEQUENCES AND RELEASE CATEGORIES USED IN CONSEQUENCE ANALYSIS

For the purpose of performing accident consequence analyses for the Limerick DES and FES, the staff requested Brookhaven National Laboratory (BNL) to help develop specifications of atmospheric release of radionuclides from severe accidents in the Limerick reactors based on the applicant's two probabilistic risk analyses (PRAs), Limerick Generating Station Probabilistic Risk Assessment (LGS-PRA)[1] and the Limerick Generating Station Severe Accident Risk Analysis (LGS-SARA).[2] The specifications included (1) identification of core-melt accident sequences leading to atmospheric release initiated by internal causes, fires, and earthquakes; (2) probabilities of the sequences; and (3) quantities and forms of radionuclides (source terms) and the other parameters necessary for appropriate characterization of atmospheric release from these sequences.

The ground rules recommended by the staff for the BNL analysis relate to the method of estimating source terms. There has been significant research activity in this area sponsored by both industry and the Commission since the publication of the Reactor Safety Study (RSS)[3] in 1975. Updated fission product source term assessment methods are currently being developed and are receiving extensive peer review. However, it is the judgment of the staff that the application of the evolving methodologies for assessment of source terms in licensing activities before they are thoroughly and carefully appraised would be premature. Therefore, the staff requested that BNL use the RSS prescriptions of fission product release from the damaged fuel, primary system holdup, credit for decontamination by suppression pool scrubbing, and fallout, plateout, and transport of radionuclides in the containment leading to atmospheric release. These RSS prescriptions are explained below.

In the RSS methodology, quantities of fission products released from the core material were based on four release components: gap, melt, oxidation, and vaporization. The gap release is modeled as a single event and is assumed to occur at accident initiation as the result of rupture of fuel cladding. It consists mostly of activity that would be released to void spaces within the fuel rods during normal reactor operation, and rapid depressurization of contained gases provides the driving force for escape. The melt release occurs from the fuel while it first heats to melting and becomes molten. High gasflows in the core during this period sweep the activity out of the core region. The melt release is divided into 10 equally sized releases evenly spaced between the time of core melt and the time of core slump. The oxidation release is modeled as a single release that occurs when the reactor pressure vessel (RPV) head fails and is the result of oxidation of that fraction of the core debris that is assumed to interact with water on the diaphragm floor or to fall into the suppression pool. Finely divided fuel material is scattered into an oxygen atmosphere and undergoes extensive oxidation, which liberates specific fission products. The vaporization release is assumed to start after vessel failure when core-concrete interactions begin. Turbulence caused by internal convection and melt sparging by gaseous decomposition products of concrete produce the driving forces for escape. The vaporization release is divided into 20 parts, 10 releases of exponentially decreasing magnitude in the first half hour followed by 10 more releases, also of exponentially decreasing magnitude, during the next 1½ hours.

Also in the RSS methodology, no specific credit for attenuation of fission products released from the RPV to containment building is allowed in the primary system. Thus, all the fission products released during the gap and melt release phases are assumed to enter the containment building.

For fission product attenuation as a result of scrubbing by water in the suppression pool, a decontamination factor (DF) of 100 is used for the subcooled pools and a DF of 1 is used for the saturated pools. (Noble gases and organic iodine are not subject to pool scrubbing.)

In the RSS methodology, the fission product transport within the containment building volumes is predicted using the CORRAL-II code. This code is used in conjunction with the fission product release model, pool scrubbing model, and the MARCH code.

As stated earlier, in the source term assessment made by BNL for use in the Limerick DES, only the RSS methodolgy was used. Use of the RSS methodology for Limerick may have resulted in over-estimates of source terms for some accident sequences and underestimates of source terms for others. However, because the evolving methodologies have not been fully appraised, the staff used its current practice of following the RSS source term assessment methodology in licensing evaluations. On balance, however, the staff has concluded that the risks estimated using the RSS source term methodology are reasonable, particularly when considered within the overall numerical uncertainties discussed in Section 5.9.4.5(7).

The staff worked with BNL during the analysis, and the final results have been reviewed by the staff and found adequate. Following the staff's guidelines, BNL developed 27 release categories for use in the Limerick DES. The same 27 release categories have also been used in the staff analysis in the FES. Characteristics of these release categories are shown in Table 5.11c and their likelihoods (point estimates of mean annual probabilities) in Table 5.11d. As noted in Section 5.9.4.5(2), source terms associated with four of the release categories in Table 5.11c, and probabilities of some of the release categories in Table 5.11d include revisions made after publication of the DES. For identification and quantification of these release categories, BNL considered (1) the sequence of events and conditions that could lead to core melt (accident damage states); (2) the containment building failure modes and radionuclide release paths; and (3) the actual characterization of radionuclide releases to the environment. Procedures used for identification of these release categories and their brief descriptions are summarized below.

Initially 67 plant damage states were identified for the Limerick reactors. Subsequently, however, 10 surrogate damage states were found to encompass these original 67 damage states. This was possible because many of the original damage states were found to be very similar in terms of the core-melt accident progression and containment failure characteristics. Table H.1 gives a brief description of each of the surrogate damage states and uses simple designators to identify the damage states for easy reference. The first six of the surrogate damage states given in Table H.1 include damage states discussed in LGS-PRA and NUREG/CR-3028,[4] but they also include the damage states initiated by fires and low to moderately severe earthquakes discussed in LGS-SARA. The last four of the surrogate damage states in Table H.1 include damage states discussed exclusively in LGS-SARA. Mean probabilities per reactor-year assigned to the 10 surrogate damage states are shown in Table H.2.

Using the 10 surrogate damage states, BNL performed analyses to determine the Limerick containment failure modes and radionuclide release characteristics using the MARCH/CORRAL computer code system*. Seven containment failure modes and release paths were identified (see Table H.3) and analyzed. They can be subdivided into leakage failures and structural failures. The leakage failures prevent the more catastrophic structural failure and, in some of the cases, make effective use of the standby gas treatment system (see Section 5.9.4.4(1)). The structural failures result in release pathways that either (1) bypass the suppression pool by failing the drywell or by causing the suppression pool to drain or (2) pass through the suppression pool. The mechanisms for developing these release pathways are overpressure from steam or noncondensibles, overpressure from hydrogen burns (for the containment deinerted cases), seismic (earthquake) failure of structures and systems, and steam explosion-induced failures. Analyses showed that there could be only 40 combinations of the 10 surrogate damage states and the 7 containment failure modes (and release paths) with non-zero probabilities (having any possibility of occurrence). The other 30 combinations were considered as essentially impossible.

The 40 combinations of surrogate damage states and containment failure modes (and leakage paths) were further reduced because the accident progressions resulting in radionuclude release to the atmosphere associated with a number of them are very similar. This resulted in 27 release categories for consequence analysis. These release categories are described in Table H.4. It should be noted that the labeling of each release category has been made both in terms of the surrogate damage state and the matching containment failure mode or leakage path.

As stated earlier, specifications (including the source terms) of each of the 27 release categories developed by BNL are shown in Table 5-11c. The timing of the radionuclide release, energy of release, duration of release, and warning time for evacuation shown in Table 5.11c were based on the MARCH analysis. The time of release is defined as the time of containment failure for those cases in which the meltdown would take place in an intact containment building. For those cases, when the containment building would fail prior to core damage, the time of release is defined as the start of core melting. The duration of release is defined as the time for the containment building to blowdown to

*The MARCH computer code used includes a new decay heat model based on the ANS-5.1-1979 standard. The 1979 standard produces an integrated decay heat over the first hour after the reactor shutdown about 20% greater than the 1971 standard used in the previous BNL review (NUREG/CR-3028)[4] of the LGS-PRA. The main effect of the new decay heat model has been the change in timing of major events during the progression of the accidents. The time to core meltdown, core slump, reactor pressure vessel failure, and containment failure predicted using the new decay heat model are significantly earlier than in NUREG/CR-3028.

atmospheric pressure. However, if the building fails first (meltdown into a failed containment building), the duration of release was defined to be from the start of core melting to the completion of vaporization release. The warning time is defined as the time period between the start of the core melt and the time of containment failure. If the containment building fails first, the warning time was defined as the time from the time of containment failure to the start of core melt. The energy of release is the energy release rate associated with the release at the time of containment failure. In those cases where the release could be spread out over many hours, the energy of release would be low. The height of release was chosen to be 25 m (82 ft) in all cases.

Following the guidelines provided by the staff, BNL subdivided the mean probability of each release category initiated by earthquakes into two parts. One part was associated with the release category that would be initiated by very severe earthquakes (effective peak ground acceleration equal to or in excess of 0.4g*), and the other part was associated with the same release category initiated by low to moderately severe earthquakes (effective peak ground acceleration less than 0.4g). The latter part was added to the mean probability of the same release category initiated by internal causes and fires. The rearranged mean probability for each release category is shown in Table 5.11d.

The purpose of such breakdown was to aid in making appropriate assumption regarding offsite emergency response in the consequence analysis. It was the judgment of the staff that earthquakes resulting in effective peak ground acceleration equal to or greater than about 0.4g would be of severity of Modified Mercalli (MM) intensity scale IX or worse.** Earthquakes of MM intensity scale IX or higher would be likely to seriously hamper the offsite emergency response efforts. (See Appendix I for description of offsite damages likely to be caused by earthquakes of various MM intensity scales.)

There are substantial uncertainties in the estimated mean probabilities shown in Table 5.11d. Further, the mean probability of a release category is not necessarily the representative of the full spectrum of values of its probability. Particularly for seismically induced release categories, values of probabilities span several orders of magnitudes between low and high estimates. However, it is the judgment of the staff that the use of the mean probabilities in consequence analysis, supplemented by discussion of uncertainties resulting from this use, provides a reasonable risk perspective. For discussion of uncertainties see Section 5.9.4.5(7).

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*g stands for acceleration due to gravity and is numerically about 32 feet per second per second.

**The lack of actual recording associated with this intensity and the controversy surrounding the definition of effective peak ground acceleration made the choice of 0.4g imprecise. A sensitivity analysis performed with a range of values of effective peak ground acceleration such as 0.35g to 0.5g would have been more appropriate. However, it was the staff's judgment that breakdown of probabilities of seismically induced release categories using several values from the range 0.35g to 0.5g of effective peak ground acceleration would not have resulted in probability sets very different from those obtained by using 0.4g.

REFERENCES

1. Letter, from PECo to NRC, submitting operating license application and a report, "Limerick Generating Station, Probabilistic Risk Assessment," March 17, 1981.

2. Letter from E. J. Bradley, PECo, to A. Schwencer, NRC, submitting report "Limerick Generating Station, Severe Accident Risk Assessment," April 21, 1983.

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

4. I. A. Papazoglou, et al., "Review of Limerick Generating Station Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, NUREG/CR-3028, February 1983.

Table H.1 Description of surrogate damage states

Designator	Description
I-S	These are LOCA (loss-of-coolant accident)-initiated sequences (medium and small pipe breaks only) involving loss-of-coolant inventory makeup. They would result in a relatively fast core melt, with the containment intact at the time of core melt.
I-T	These are sequences initiated by transient events* involving loss-of-coolant inventory makeup. Core melt is expected to be relatively fast and the containment to be intact at the time of core melt.
II-T	These are transient- or LOCA-intiated sequences involving loss of containment heat removal or inadvertent steam relief valve opening accidents with inadequate heat removal capability. Core melt is expected to be relatively slow as a result of the lower decay power level, with the containment failing before core melt.
III-T	These sequences are transients involving loss of scram (fast shut-down of reactor) function and inability to provide coolant makeup, large LOCAs with insufficient coolant makeup, transients with loss of heat removal, and long-term loss-of-coolant inventory makeup. Core melt is expected to be relatively fast, and the containment intact at core melt.
IV-T	These sequences are transients that involve loss of scram function and a loss of containment heat removal or all reactivity control, but with coolant makeup capability. Core melt is expected to be relatively fast with the containment failing before core melt because of overpressure.
IV-A	As above but initiated by large LOCAs.
IS-$\overline{C}$	These sequences are seismically (earthquake) induced sequences that lead to failure of the coolant inventory/makeup systems and a breach of wetwell integrity with the reactor scrammed. Core melt is expected to be fast, with the containment failing before core melt because the residual heat removal (RHR) system suction lines are severed.
IS-C	As above, but coupled with a loss of the scram function.
S-H2O	These sequences are seismically induced reactor vessel failures (plus random reactor-vessel failure), coupled with immediate con-tainment failure. Core melt is fast, with the vessel and contain-ment both failed at the time of core melt. This sequence assumes the vessel break is high, which would allow water to be retained in the bottom of the vessel before core slump.

*See next page for footnote.

Table H.1 (Continued)

Designator	Description
S-$\overline{H2O}$	As above, but with a vessel failure location that results in complete draining of the water from the vessel.

*In general, the term reactor transient applies to any significant deviation from the normal operating values of any of the key reactor operating parameters. More specifically, transient events can be assumed to include all those situations (except for the LOCA, which is treated separately) that could lead to fuel heat imbalances. When viewed in this way, transients cover the reactor in its shutdown condition as well as in its various operating conditions. The shutdown condition is important in the consideration of transients because many transient conditions result in shutdown of the reactor, and decay heat removal systems are needed to prevent fuel heat imbalances as a result of core decay heat.

Transients may occur as a consequence of an operator error or the malfunction or failure of equipment. Many transients are handled by the reactor control system, which would return the reactor to its normal operating condition. Others would be beyond the capability of the reactor control system and would require reactor shutdown by the reactor protection system to avoid damage to the reactor fuel.

In safety analyses, the principal areas of interest are increases in reactor core power (heat generation), decreases in coolant flow (heat removal), and increases in reactor coolant system (RCS) pressure. Any of these could potentially result from a malfunction or failure, and they represent a potential for damage to the reactor core and/or the pressure boundary of the RCS. The analysis of reactor transients has been directed at identifying those malfunctions or failures that can cause core melting or rupture of the RCS pressure boundary. Regardless of the way in which transients might cause core melting, the consequences are essentially the same; that is, the molten core would be inside the containment and would follow the same course of events as a molten core that might result from a LOCA.

Each potential transient is assessed to fall into either one of two general categories, the anticipated (likely) transients and the unanticipated (unlikely) transients. The large majority of potential transients are those that have become commonly known as anticipated transients. All other transients are considered to fall into the unanticipated transients category. The relatively low probability (unanticipated) transients can be eliminated from the risk determination because their potential contribution to risk is small compared to that of the more likely (anticipated) transients that would produce the same consequences.

The anticipated transient initiators for which successful reactor scram could be accomplished have been divided into five groups for analysis of the Limerick reactors. These groups are

Table H.1 (Continued)

(1) transients resulting in turbine trip
(2) transients leading to isolation of the reactor vessel from the main condenser, a main steamline isolation valve (MSIV) closure, and loss of feedwater
(3) transients resulting from loss of offsite power
(4) transients resulting from inadvertent open relief valve (IORV)
(5) orderly and controlled manual shutdown

Thirty-seven BWR transients identified from operating experience data are listed in Table 2.9 of NUREG/CR-3028[4] and are included in the first four of the above groups. If the reactor protection system fails to scram the reactor after an initiating event in any of the first four transient groups, then an anticipated transient without scram (ATWS) condition results. The following four groups of ATWS initiators were, therefore, considered:

(1) turbine trip ATWS
(2) MSIV closure ATWS
(3) loss of offsite power ATWS
(4) IORV ATWS

Table H.2 Mean (point estimate) probabilities of surrogate damage states by initiating events

Surrogate damage state	Probability per reactor-year			
	Internal causes	Fires	Low to moderately severe earthquakes (EPA* < 0.4g)**	Severe earthquakes (EPA* $\geq$ 0.4g)**
I-S	8(-8)***			
I-T	8(-5)	3(-6)	9(-7)	2(-6)
II-T	4(-6)		1(-8)	4(-8)
III-T	3(-6)		8(-8)	7(-7)
IV-T	3(-7)		2(-8)	1(-7)
IV-A	5(-9)			
IS-$\overline{C}$			1(-7)	9(-7)
IS-C			1(-8)	1(-7)
S-H2O	1(-8)			4(-8)
S-$\overline{H2O}$	1(-8)			4(-7)
TOTAL	9(-5)	3(-6)	1(-6)	4(-6)

*EPA stands for effective peak ground acceleration.

**g stands for an acceleration equal that due to gravity and is numerically equal to 32 feet per second per second

***8(-8) = 8 x 10^{-8}

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table H.3 Containment failure mode and release path notation

Designator	Description
DW	Containment failure via overpressurization. Failure location in the drywell.
WW	Containment failure via overpressurization. Failure location in the wetwell above the suppression pool.
$\overline{WW}$	Containment failure via overpressurization. Failure location in the wetwell below the suppression pool resulting in loss of suppression pool water.
SE	Failure via in-vessel steam explosion-generated missiles.
HB	Failure via hydrogen burning during the periods when the containment atmosphere is de-inerted. This failure mode also includes hydrogen detonation and ex-vessel steam explosion failure modes, which are of very low frequency.
LGT	Containment leakage rates sufficiently low to allow the standby gas treatment system (SGTS) to operate effectively.
$\overline{LGT}$	Containment leakage rates so high that the SGTS is ineffective.

Table H.4 Description of the release categories

Category	Description
1. RELEASE CATEGORIES ASSOCIATED WITH SURROGATE DAMAGE STATE I-T	The damage state I-T is defined in Table H.1 and basically consists of transients with loss-of-coolant inventory makeup. Core melt in such situations is expected to be relatively fast and occurs within an intact containment. After vessel failure, the majority of the core materials are retained on the diaphragm floor below the reactor vessel. Containment failure occurs via gradual overpressurization (except for SE, HB, LGT and $\overline{\text{LGT}}$ release -- see Table H.3) several hours after vessel failure as a result of core/concrete interactions.
I-T/DW	This release category assumes an overpressure failure in the drywell wall. The gap and melt releases would be directed to the suppression pool and subjected to a DF of 100 (water is subcooled) before they reach the wetwell airspace. The vaporization release would be directed to the drywell without any pool scrubbing. All fission products in the drywell and wetwell would be subjected to agglomeration and settling as predicted by the CORRAL code before vessel failure, several hours after the pressure vessel failure.
I-T/WW	This release category assumes a failure in the wetwell above the suppression pool. The gap, melt, and vaporization releases would be released to the drywell and wetwell as described above. The only difference is that when the containment fails, fission products in the drywell must pass through the downcomers and suppression pool before they are released to the atmosphere.

Table H.4 (Continued)

Category	Description
I-T/$\overline{WW}$	This release category assumes a failure in the wetwell below the suppression pool, which drains the water. The gap, melt, and vaporization releases would be released to the containment as described above. The only difference is that at containment failure the suppression pool would be drained so that fission products in the drywell no longer have to pass through the suppression pool (as in the I-T/WW release path) before they are released to atmosphere.
I-T/SE	This release category results from an in-vessel steam explosion-generated missile. BNL assumed this occurs at core slump and opens a direct path from the primary system to atmosphere. In the LGS-PRA, this failure mode was similar to RSS release category BWR-1. The release corresponds to an anticipated transient without scram sequence analyzed in Appendix V of the RSS, in which the steam explosion was assumed to occur after only 13% of the core had melted. Consequently, most of the melt release would be released to containment without pool scrubbing. However, BNL used a steam explosion release that more appropriately reflects BNL's analysis of the sequence.
I-T/HB	This release category could result from hydrogen burn failures during the time when the containment atmosphere is deinerted. BNL used the same release category as in the LGS-PRA, but reduced the core fraction associated with the oxidation releases in a manner consistent with WASH-1400. (Note in the LGS-PRA, this release category was representative of ex-vessel steam explosions.)

Table H.4 Description of the release categories

Category	Description
I-T/LGT and I-T/$\overline{LGT}$	These release categories result from containment leakage and assume that the SGTS operates (LGT), or that it does not operate ($\overline{LGT}$). BNL used the LGS-PRA releases, but changed the timing to correspond to the BNL MARCH analysis.
2. RELEASE CATEGORIES ASSOCIATED WITH SURROGATE DAMAGE STATE II-T	The damage state II-T is defined in Table H.1 and basically assumes loss of containment heat removal. Eventually, the containment would fail and cause the loss of inventory makeup. As the containment would fail prior to core melt and the suppression pool is saturated (DF of 1), the location of containment failure (DW, WW or $\overline{WW}$ -- see Table H.3) is of rather less importance than it is for the I-T damage states.
II-T/WW	This release category assumes a failure in the wetwell above the suppression pool. The melt release would be directed to the suppression pool, but would not be subjected to pool decontamination because the water would be saturated. The vaporization release would be directed to the drywell, then through the downcomers to the wetwell air space, and finally to the atmosphere. This one failure location was also used to represent failures in the drywell (DW) and wetwell below the suppression pool ($\overline{WW}$). This assumption is reasonable because, as the pool is saturated, the different flow paths would not result in significant differences in calculated release fractions (see IV-T below).
II-T/SE	This release category results from an in-vessel steam explosion generated missile. The release path used in the LGS-PRA, which was taken from Appendix V of the RSS, was considered appropriate and was used. Differences relate only to the timing, which now corresponds to the present analysis of a II-T damage state.

Table H.4 Description of the release categories

Category	Description
3. RELEASE CATEGORIES ASSOCIATED WITH SURROGATE DAMAGE STATE III-T	The damage state III-T corresponds to a transient event coupled with loss of scram function (see Table H.1). Core melt would be rapid and into an intact containment. Containment failure is predicted to occur after vessel failure as a result of overpressurization. However, the suppression pool would be saturated so that the gap, melt, and vaporization releases would not be subjected to decontamination by the pool. Consequently, again (as for the II-T damage state) one failure location was used to represent the three potential locations.
III-T/WW	This release category is similar to the I-T/WW sequence; however, because the pool is saturated, the melt release would not be subjected to pool scrubbing.
III-T/SE	The steam explosion release category used in the LGS-PRA was considered appropriate and was used. Differences in conditions postulated were related only to timing, which was made consistent with a MARCH thermal-hydraulics analysis.
III-T/HB, III-T/LGT and III-T/$\overline{LGT}$	These release categories are also considdered as possible and would be similar to I-T/HB, I-T/LGT and I-T/$\overline{LGT}$, respectively.
4. RELEASE CATEGORIES ASSOCIATED WITH SURROGATE DAMAGE STATE IV-T	The damage state IV-T is defined in Table H.1 and essentially consists of ATWS sequences in which continued coolant makeup was postulated to result in overpressurization failure of the containment before core melt. The suppression pool would be saturated for these sequences and hence the DF would be unity.
IV-T/DW, IV-T/WW and IV-T/$\overline{WW}$	For these release categories, the impacts of the three potential failure locations (DW, WW, and $\overline{WW}$) were analyzed. Because of the saturated pool, similar release fractions were estimated. These calculations support the use of only one failure location for the II-T and III-T damage states. The release paths (DW, WW, and $\overline{WW}$) for the three locations are discussed in detail above.

Table H.4 Description of the release categories

Category	Description
IV-T/SE	The steam explosion release category used in the LGS-PRA for Class III (damage state III-T) was considered appropriate to this damage state. Consequently, this release category is used, with the timing changed to be consistent with the BNL MARCH analysis.
5. RELEASE CATEGORIES ASSOCIATED WITH SURROGATE DAMAGE STATES I-S AND IV-A	The damage states I-S and IV-A are defined in Table H.1 and correspond to LOCA-initiated sequences. They were calculated to have a low frequency but, because of differences in flow paths relative to transients, were analyzed separately.
I-S/DW	This release category would result in the release of the melt and vaporization releases to the drywell, thus bypassing pool scrubbing. However, because the containment would fail several hours after vessel failure, the release fractions are not significantly different from the I-T/DW flow path (in which the gap and melt releases were subjected to suppression pool scrubbing.)
IV-A/DW	This release category is similar to IV-T/DW except that the initiating event is a large LOCA.
6. RELEASE CATEGORIES ASSOCIATED WITH SURROGATE DAMAGE STATES IS-C AND IS-$\overline{C}$	The damage states IS-C and IS-$\overline{C}$ are defined in Table H.1 and could be induced by earthquakes. The RHR suction lines could be severed, resulting in partial loss of the suppression pool. The gap and melt releases would be directed to the suppression pool and subjected to decontamination (the water would be subcooled and the DF = 100) before release via the severed RHR suction lines. The vaporization release would be directed to the drywell and then flow through the downcomers into the wetwell. However, as the suppression pool would be drained below the downcomer outlet, the vaporization release would not be subject to pool scrubbing. The difference between IS-C and IS-$\overline{C}$ relates to the scram function and does not influence the flow paths; only the timing of the sequence is affected.

Table H.4 Description of the release categories

Category	Description
IS-C/DW and IS-$\overline{C}$/DW	The failure mode for these release categories was considered to be similar to a DW mode in LPG-SARA. However, this should not be interpreted as a failure location in the drywell. Rather, for release analysis purposes, a containment failure of the type DW is postulated.
IS-C/SE and IS-$\overline{C}$/SE	For these release categories, the in-vessel steam explosion failures were assumed to be similar to the I-T/SE release. Only the timing was altered to reflect the MARCH analysis.
7. RELEASE CATEGORIES ASSOCIATED WITH SURROGATE DAMAGE STATES S-H20 AND S-$\overline{H20}$	The damage states S-H20 and S-$\overline{H20}$ are defined in Table H.1; they also would be earthquake induced. The RHR suction lines would be severed, but the vessel also could fail at the start of the accident. Thus, the core would melt into a failed containment and none of the releases would be subjected to pool scrubbing. The only differences between the S-H20 and S-$\overline{H20}$ sequences relate to the location of possible failure in the vessel. For the S-H20 sequence, water would remain in the vessel and be available for interacting with core debris as slumping occurs. This would affect movement of the fission products and allow the potential for an in-vessel steam explosion. The S-$\overline{H20}$ damage state involves a failure of the vessel so that the water would be completely drained at the start of the accident. Thus, there would be no in-vessel debris/water interaction and no potential for an in-vessel steam explosion.
S-H20/$\overline{WW}$, S-$\overline{H20}$/SE and S-$\overline{H20}$/$\overline{WW}$	These release categories are considered possible. Assignment of $\overline{WW}$ failure mode to damage states S-H20 and S-$\overline{H20}$ relates only to similarity of fission product release path and lack of suppression pool scrubbing, rather than the actual failure location.

APPENDIX I

DESCRIPTION OF POTENTIAL OFFSITE DAMAGES FROM EARTHQUAKES OF VARIOUS INTENSITIES, ACCORDING TO THE MODIFIED MERCALLI INTENSITY SCALE OF 1931

[Adapted from Seiberg's Mercalli-Cancani scale, modified and condensed.]

I.
- a. Not felt, except rarely under especially favorable circumstances. Under certain conditions, at and outside the boundary of the area in which a great shock is felt.
- b. Sometimes birds or animals reported uneasy or disturbed.
- c. Sometimes dizziness or nausea experienced.
- d. Sometimes trees, structures, liquids, bodies of water may sway, doors swing very slowly.

II.
- a. Felt indoors by few, especially on upper floors, or by sensitive, or nervous persons.
- b. Sometimes hanging objects may swing, especially when delicately suspended.
- c. Sometimes trees, structures, liquids, bodies of water may sway, doors swing very slowly.
- d. Sometimes birds or animals reported uneasy or disturbed.
- e. Sometimes dizziness or nausea experienced.

III.
- a. Felt indoors by several persons.
- b. Motion, usually rapid vibration.
- c. Sometimes not recognized to be an earthquake at first.
- d. Duration estimated in some cases.
- e. Vibration like that due to passing of light or lightly loaded trucks or heavy trucks some distance away.
- f. Hanging objects may swing slightly.
- g. Movements may be appreciable on upper level of tall structures.
- h. Standing motorcars rocked slightly.

IV.
- a. Felt indoors by many, outdoors by few.
- b. Awakened few, especially light sleepers.
- c. Frightened no one, unless apprehensive from previous experience.
- d. Vibration like that due to passing of heavy or heavily loaded trucks.
- e. Sensation like heavy body striking building, or falling of heavy objects inside.
- f. Rattling of dishes, windows, doors; glassware and crockery clink and clash.
- g. Creaking of walls, frame, especially in the upper range of this grade.

h. Hanging objects swing in numerous instances.
i. Liquids in open vessels slightly disturbed.
j. Standing motorcars rocked noticeably.

V.
a. Felt indoors by practically all; outdoors by many or most.
b. Outdoors direction estimated.
c. Awakened many or most.
d. Frightened few, slight excitment, a few ran outdoors.
e. Buildings trembled throughout.
f. Dishes, glassware broken to some extent.
g. Windows cracked in some cases, but not generally.
h. Vases, small or unstable objects overturned, in many instances, with occasional falls.
i. Hanging objects, doors, swing generally or considerably.
j. Pictures knocked against walls or swung out of place.
k. Doors, shutters opened or closed abruptly.
l. Pendulum clocks stopped, started, or ran fast, or slow.
m. Small objects, furnishings moved, the latter to a slight extent.
n. Liquids spilled in small amounts from well-filled open containers.
o. Trees, bushes shaken slightly.

VI.
a. Felt by all, indoors and outdoors.
b. Frightened many; excitement general; some alarm; many ran outdoors.
c. Awakened all.
d. Persons made to move unsteadily.
e. Trees, bushes shaken slightly to moderately.
f. Liquid set in strong motion.
g. Small bells rang--church, chapel, school, etc.
h. Damage slight in poorly built buildings.
i. Fall of plaster in small amount.
j. Plaster cracked somewhat, especially fine cracks (in) chimneys in some instances.
k. Dishes, glassware broken in considerable quantity, also some windows.
l. Knickknacks, books, pictures fall.
m. Furniture overturned in many instances.
n. Moderately heavy furnishings moved.

VII.
a. Frightened all; general alarm, all ran outdoors.
b. Some, or many, found it difficult to stand.
c. Noticed by persons driving motorcars.
d. Trees and bushes shaken moderately to strongly.
e. Waves on ponds, lakes, and running water.
f. Water turbid from stirred-up mud.
g. Incaving to some extent of sand or gravel stream banks.
h. Large church bells, etc. rang.
i. Suspended objects quiver.
j. Damage negligible in buildings of good design and construction.

k. Damage slight to moderate in well-built ordinary buildings; considerable in poorly built or badly designed buildings, adobe houses, old walls (especially without mortar), spires, etc.
l. Chimneys cracked to considerable extent, walls to some extent.
m. Fall of plaster in considerable to large amounts; also some stucco falls.
n. Numerous windows broken; furniture to some extent.
o. Loosened brickwork and tiles shaken down.
p. Weak chimneys broken at the roofline (sometimes damaging roofs).
q. Cornices fall from towers and high buildings.
r. Bricks and stones dislodged.
s. Heavy furniture overturned, with damage from breaking.
t. Considerable damage to concrete irrigation ditches.

VIII. a. Fright general; alarm approaches panic.
b. Persons driving motorcars disturbed.
c. Trees shaken strongly; branches, trunks broken off, especially palm trees.
d. Sand and mud ejected in small amounts.
e. Temporary and permanent changes in flow of springs and wells; dry wells renewed flow, temperature changes in spring and well waters.
f. Damage slight in structures (brick) built especially to withstand earthquakes.
g. Damage considerable in ordinary substantial buildings: partial collapse, racked; tumbled down wooden houses in some cases; threw out panel walls in frame structures; decayed piling broken off.
h. Walls fall.
i. Cracked, broke solid stone walls seriously; wet ground to some extent, also ground on steep slopes.
j. Chimneys, columns, monuments, factory stacks, towers twist, fall.
k. Very heavy furniture moved conspicuously, overturned.

IX*. a. Panic general
b. Ground cracked conspicuously.
c. Damage considerable in (masonry) structures built especially to withstand earthquakes.
d. Some wood frame houses built especially to withstand earthquakes, thrown out of plumb.
e. Damage great in substantial (masonry) buildings, some collapse in large part; wholly shifted frame buildings off foundations, racked frames.
f. Damage serious to reservoirs.
g. Underground pipes sometimes broken.

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*It is the staff's judgment that MM Intensity Scale of IX and higher would be associated with effective peak ground acceleration of about or greater than 0.4g.

X. a. Ground cracked, especially when loose and wet, up to widths of several inches; fissures up to a yard in width parallel to canal and stream banks.
b. Landslides considerable from river banks and steep coasts.
c. Sand and mud shifted horizontally on beaches and flat land.
d. Level of water in wells changed.
e. Water thrown on banks of canals, lakes, rivers, etc.
f. Damage serious to dams, dikes, embankments.
g. Damage severe to well-built wooden structures and bridges, some destroyed.
h. Dangerous cracks developed in excellent brick walls.
i. Most masonry and frame structures destroyed, also their foundations.
j. Railroad rails bent slightly.
k. Pipelines buried in earth torn apart or crushed endwise.
l. Open cracks and broad wavy folds in cement pavements and asphalt road surfaces.

XI. a. Many and widespread disturbances in ground, varying with ground material.
b. Broad fissures, earth slumps, and land slips in soft, wet ground.
c. Water ejected in large amounts charged with sand and mud.
d. Sea-waves (tidal waves) of significant magnitude.
e. Damage severe to wood frame structures, especially near shock centers.
f. Damage great to dams, dikes, embankments, often for long distances.
g. Few, if any, masonry structures remained standing.
h. Large, well-built bridges destroyed by the wrecking of supporting piers, or pillars.
i. Yielding wooden bridges affected less.
j. Railroad rails bent greatly and thrust endwise.
k. Pipelines buried in earth put completely out of service.

XII. a. Damage total--practically all works of construction damaged greatly or destroyed.
b. Disturbances in ground great and varied, numerous shearing cracks.
c. Landslides, falls of rock of significant character, slumping of river banks, etc., numerous and extensive.
d. Large rock masses wrenched loose, torn off.
e. Fault slips in firm rock, with notable horizontal and vertical offset displacements.
f. Water channels, surface and underground, disturbed and modified greately.
g. Lakes dammed, waterfalls produced, rivers deflected, etc.
h. Waves seen on ground surfaces (actually seen, probably, in some cases).
i. Lines of sight and level distorted.
j. Objects thrown upward into the air.

APPENDIX J

CONSEQUENCE MODELING CONSIDERATIONS

J.1 Evacuation Model

"Evacuation," used in the context of offsite emergency response in the event of substantial amount of radioactivity release to the atmosphere in a reactor accident, denotes an early and expeditious movement of people to avoid exposure to the passing radioactive cloud and/or to acute ground contamination in the wake of the cloud passage. It should be distinguished from "relocation" which denotes a post-accident response to reduce exposure from long-term ground contamination. The Reactor Safety Study (RSS) (WASH-1400, NUREG-75/014) consequence model contains provision for incorporating radiological consequence reduction benefits of public evacuation. The benefits of a properly planned and expeditiously carried out public evacuation would be manifested in a reduction of early health effects associated with early exposure; namely, in the number of cases of early fatality (see Section J-2) and acute radiation sickness that would require hospitalization. The evacuation model originally used in the RSS consequence model is described in WASH-1400 as well as in NUREG-0340. However, the evacuation model that has been used herein is a modified version (SAND 78-0092) of the RSS model and is, to a certain extent, oriented toward site emergency planning by inclusion of site-specific delay time before evacuation and effective evacuation speed as model parameters. The modified version is incorporated into the current version of the CRAC code (and the CRAC2 code which is a modified version of CRAC) and is briefly outlined below.

The model assumes that people living within portions of a circular area with a specified radius (such as the 10-mile (16-km) plume exposure pathway Emergency Planning Zone (EPZ)), with the reactor at the center, would evacuate if an accident should occur involving imminent or actual release of significant quantities of radioactivity to the atmosphere.

Significant atmospheric releases of radioactivity would in general be preceded by one or more hours of warning time (postulated as the time interval between the awareness of impending core melt and the beginning of the release of radioactivity from the containment building)--although for some specific release categories the warning time could be less than an hour. For the purpose of calculation of radiological exposure, the model assumes that those people who would potentially be under the radioactive cloud that would develop following the release would leave their residences after a specific amount of delay time* and then evacuate. The delay time is reckoned from the beginning of the warning time and is recognized as the sum of the time required by the reactor operators to notify the responsible authorities; the time required by the authorities to interpret the data, decide to evacuate, and direct the people to evacuate; and the time required for the people to mobilize and get underway.

*Assumed to be of constant value which would be the same for all evacuees.

The model assumes that while leaving the area each evacuee would move radially out and in the downwind direction* with an average effective speed** (obtained by dividing the zone radius by the average time taken to clear the zone after the delay time) over a fixed distance** from the evacuee's starting point. The fixed distance used in the analysis discussed in Section 5.9.4.5(2) was selected to be 15 miles (24 km) (which is 5 miles (8 km) more than the 10-mile (16-km) plume exposure pathway EPZ radius). After reaching the end of the travel distance, the evacuee is assumed to receive no further radiation exposure. In a real evacuation, paths of evacuees would be dictated by the site road net-work. However, each segment of actual trajectory of an evacuee would project a component in the downwind direction which, in the consequence model, is assumed to be radial. Therefore, each evacuee's actual motion would have a component of motion along the radial downwind direction. The evacuation model assumption that evacuees originating from areas that would come under the radioactive cloud would move radially out over a certain distance amounts to only an artifice for dose calculation: as if the evacuee's radiological exposure is due to their component motion along the radial downwind direction (over a component path length which is assumed to be 15 miles).

The model incorporates a finite length of the radioactive cloud in the downwind direction; this would be determined by the product of the duration over which the atmospheric release would take place and the average windspeed during the release. It is assumed that the front and the back of the cloud formed would move with an equal speed, which would be the same as the prevailing windspeed; therefore, its length would remain constant. At any time after the release, the concentration of radioactivity is assumed to be uniform over the length of the cloud. If the delay time would be less than the warning time, then all evacuees would have a head start, i.e., the cloud would be trailing behind the evacuees initially. On the other hand, if the delay time would be more than the warning time, then, depending on initial locations of the evacuees there are possibilities that (1) an evacuee would still have a head start, (2) the cloud would already be overhead when an evacuee starts out to leave, or (3) an evacuee would be initially trailing behind the cloud. However, this initial picture of cloud-people disposition would change as the evacuees travel, depending on the relative speeds and positions between the cloud and people. It is possible that the cloud and an evacuee would overtake one another one or more times before the evaucee would reach his or her destination. In the model, the radial position of an evacuating person, while stationary or in transit, is compared to the front and the back of the cloud as a function of time to determine a period of exposure to airborne radionuclides. The model calculates the time periods during which people are exposed to radionuclides on the ground while they are stationary and while they are evacuating. Because radionuclides would be deposited continually from the cloud as it passed a given location, a person while under the cloud would be exposed to ground contamination less concentrated than if the cloud had completely passed. To account for this reasonably, the revised model assumes that persons are exposed to the total ground contamination when completely passed by the cloud; to one half the calculated concentration when they are anywhere under the cloud; and to no concentration when they are in front of the cloud.

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*In the RSS consequence model and the CRAC and CRAC2 codes, the radioactive cloud is assumed to travel radially outward only.

**Assumed to be a constant value for all evacuees.

The model provides for use of different values of the shielding protection factors for exposure from airborne radioactivity and contaminated ground for stationary and moving evacuees during delay and transit periods.

The model has the same provision for calculation of the economic cost associated with implementation of evacuation as in the original RSS model. For this purpose, the model assumes that for atmospheric releases of durations 3 hours or less, all people living within a circular area of 5-mile (8-km) radius centered at the reactor, plus all people within a 90° angular sector within the plume exposure pathway EPZ and centered on the the downwind direction, will evacuate and temporarily relocate. However, if the duration of release exceeds 3 hours, the cost of evacuation is based on the assumption that all people within the entire plume exposure pathway EPZ would evacuate and temporarily relocate. For either of these situations, the cost of evacuation and relocation is assumed to be $225 (1980 dollar) per person, which includes cost of food and temporary sheltering for a period of 1 week.

J.2 Early Health Effects Model

The medical advisors to the RSS (WASH-1400, Appendix IV, Section 9.2.2, and Appendix F) proposed three alternative dose-mortality relationships that can be used to estimate the number of early fatalities that might result in an exposed population. These alternatives characterize different degrees of postexposure medical treatment from "minimal," to "supportive," to "heroic"; they are more fully described in NUREG-0340. There is uncertainty associated with both the mortality relationships (NUREG/CR-3185), and the availability and efficacy of different classes of medical treatment (Elliot, 1982). Estimates of the early fatility risks using the dose-mortality relationship that is based upon the supportive treatment alternative are presented in the texts of Section 5.9.4.5. This implies the availability of medical care facilities and services for those exposed in excess of 175 rems, the approximate level that the medical advisors to the RSS indicated would be indicative of the potential need for more than minimum services to reduce early fatality risks. At the extreme low probability end of the spectrum (i.e., at the 1 chance in 100 million per reactor-year level), the number of persons involved might exceed the capacity of facilities for such services, in which case the number of early fatalities might have been underestimated. To gain perspective on this element of uncertainty, the staff has also performed calculations using the most pessimistic dose-mortality relationship based upon WASH-1400 medical experts' estimated dose-mortality relationship for minimal medical treatment and using identical assumptions regarding offsite emergency response as made in Section 5.9.4.5. These results are also presented in Section 5.9.4.5. The staff has also considered the uncertainties associated with the WASH-1400 dose-mortality relationship for minimal medical treatment and has concluded that early fatality risk estimates as bounded by the uncertainties discussed in Section 5.9.4.5(7) are reasonable. This is because it is inconceivable that a major reactor accident at Limerick would not be followed by a mobilization of medical services, services which can be expected to reduce mortality risks to less than those indicated by the WASH-1400 description of minimal medical treatment.

J.3 References

Elliot, D.A., Task 5 letter report from Dr. D. A. Elliot of Andrulis Research Corp. to Ms. A. Chu, NRC Project Officer, on Technical Assistance Contract No. NRC-03-82-128, December 13, 1982.

Sandia Laboratories, "A Model of Public Evacuation for Atmospheric Radiological Releases," SAND-78-0092, June 1978.

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

---, NUREG-0340, "Overview of the Reactor Safety Study Consequences Model," October 1977.

---, NUREG/CR-3185, "Critical Review of the Reactor Safety Study Radiological Health Effects Model," Sandia Laboratories (also SAND-82-7081) March 1983.

APPENDIX K

CONDITIONAL MEAN VALUES OF ACCIDENT CONSEQUENCES

The conditional mean values of potential societal consequences of several kinds from each release category in Table 5.11c are shown in Table K.1. These means were calculated by the CRAC code and represent averages of each kind of consequence for each release category over the spectrum of the Limerick site meterological conditions. "Conditional" mean values are so called because these mean values are conditional upon the occurrence of the accidents represented by the release categories. Probabilities of release categories have not been factored into these mean value estimates. The conditional mean values are provided for a perspective only; they are devoid of much importance without simultaneous association of probabilities of the release categories to which the mean values are due. They are useful, however, in judging the relative importance of different sequences.

Table K.1 is useful for risk calculations. It can be used to calculate the risk of any particular kind of consequence (shown in the table) from any of the listed release categories by simply multiplying the conditional mean value of the given consequence by the probability per reactor-year (Table 5.11d) of the release category to which the mean value is due. It can also be used to calculate the risk of any particular kind of consequence from a group of release categories by calculating the sum of the products of the conditional mean values of the consequence and the probabilities of the respective release categories in the group; the group may include some or all of the release categories.

Table K.1 Conditional mean values of societal consequences from individual release categories for three alternative offsite emergency response modes

Consequence Category	Offsite Emergency Response Mode	Release Categories									
		I-T/DW	I-T/WW	I-T/$\overline{WW}$	I-T/SE*	I-T/HB	I-T/$\overline{LGT}$	II-T/WW	III-T/WW	III-T/HB	III-T/LGT
1. Early fatalities with supportive medical treatment (persons)	Evac-Reloc	0	0	0	2(2)**	1(1)	5(-1)	0	0	1(1)	0
	Early Reloc	1(0)	0	0	7(1)	1(1)	1(0)	2(2)	3(1)	1(1)	0
	Late Reloc	3(1)	5(-1)	5(-1)	---	1(2)	5(1)	2(3)	4(2)	2(2)	2(-2)
2. Population receiving in excess of 200 Rems total marrow dose from early exposure (persons)	Evac-Reloc	0	0	0	2(3)	4(2)	4(1)	5(2)	2(3)	4(2)	3(0)
	Early Reloc	1(1)	0	0	1(3)	3(2)	2(1)	2(3)	2(3)	3(2)	0
	Late Reloc	1(2)	3(0)	1(0)	--	1(3)	9(2)	5(3)	7(3)	1(3)	5(0)
3. Early injuries (persons)	Evac-Reloc	4(1)	0	0	3(3)	5(2)	5(1)	6(2)	3(3)	5(2)	5(0)
	Early Reloc	5(1)	1(-2)	2(-2)	3(3)	4(2)	4(1)	2(3)	3(3)	4(2)	8(-1)
	Late Reloc	2(2)	2(0)	1(0)	--	1(3)	6(2)	3(3)	6(3)	1(3)	9(0)
4. Delayed cancer fatalities (excluding thyroid) (persons)	Evac-Reloc	6(2)	1(1)	4(1)	.6(3)	2(3)	1(3)	4(3)	4(3)	2(3)	2(1)
	Early Reloc	6(2)	3(1)	5(1)	6(3)	2(3)	1(3)	4(3)	4(3)	2(3)	3(1)
	Late Reloc	7(2)	3(1)	5(1)	--	2(3)	1(3)	4(3)	4(3)	2(3)	3(1)
5. Delayed thyroid cancer fatalities (persons)	Evac-Reloc	1(2)	2(1)	2(1)	8(2)	6(2)	2(2)	1(3)	9(2)	6(2)	1(1)
	Early Reloc	1(2)	2(1)	2(1)	8(2)	6(2)	2(2)	1(3)	1(3)	6(2)	2(1)
	Late Reloc	2(2)	2(1)	2(1)	--	7(2)	2(2)	1(3)	1(3)	7(2)	2(1)
6. Total person-rems	Evac-Reloc	1(7)	5(5)	8(5)	4(7)	2(7)	2(7)	6(7)	6(7)	2(7)	4(5)
	Early Reloc	1(7)	5(5)	9(5)	4(7)	2(7)	2(7)	6(7)	6(7)	2(7)	5(5)
	Late Reloc	1(7)	5(5)	1(6)	--	2(7)	3(7)	7(7)	7(7)	3(7)	6(5)
7. Cost of offsite mitigation measures (1980 dollars)	Evac-Reloc	3(8)	5(7)	6(7)	2(9)	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
	Early Reloc	2(8)	2(6)	3(6)	2(9)	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
	Late Reloc	2(8)	2(6)	3(6)	--	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
8. Land area for long-term interdiction (m^2)	Evac-Reloc	1(6)	2(4)	3(4)	7(7)	2(7)	3(7)	1(8)	6(7)	2(7)	0
	Early Reloc	1(6)	2(4)	3(4)	7(7)	2(7)	3(7)	1(8)	6(7)	2(7)	0
	Late Reloc	1(6)	2(4)	3(4)	--	2(7)	3(7)	1(8)	6(7)	2(7)	0

*This release category has a probability less than 10^{-9} per reactor-year to be initiated by severe earthquakes; it is not analyzed with Late Reloc mode for its insignificant contribution to risks due to its low probability.

**2(2) = 2×10^2 = 200.

***These release categories are initiated by plant internal causes; therefore, the Late Reloc mode does not apply.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table K.1 (Continued)

Consequence Category	Offsite Emergency Response Mode	Release Categories									
		III-T/$\overline{LGT}$	IV-T/DW	IV-T/WW	IV-T/$\overline{WW}$	I-S/DW***	IV-A/DW***	IS-C/DW	IS-$\overline{C}$/DW	S-H2O/$\overline{WW}$	S-$\overline{H2O}$/$\overline{WW}$
1. Early fatalities with supportive medical treatment (persons)	Evac-Reloc	6(-1)	6(2)	5(2)	6(2)	0	7(2)	3(2)	1(2)	0	0
	Early Reloc	1(0)	1(3)	1(3)	1(3)	0	1(3)	7(2)	7(2)	2(2)	6(2)
	Late Reloc	7(1)	4(3)	4(3)	4(3)	--		3(3)	3(3)	2(3)	3(3)
2. Population receiving in excess of 200 Rems total marrow dose from early exposure (persons)	Evac-Reloc	5(1)	5(3)	4(3)	4(3)	0	4(3)	2(3)	2(3)	4(2)	4(2)
	Early Reloc	3(1)	6(3)	5(3)	4(3)	5(-1)	5(3)	3(3)	3(3)	1(3)	2(3)
	Late Reloc	1(3)	1(4)	1(4)	1(4)	--	--	9(3)	9(3)	5(3)	8(3)
3. Early injuries (persons)	Evac-Reloc	6(1)	5(3)	4(3)	3(3)	0	3(3)	2(3)	2(3)	5(2)	6(2)
	Early Reloc	4(1)	5(3)	4(3)	4(3)	5(-1)	3(3)	3(3)	3(3)	2(3)	2(3)
	Late Reloc	7(2)	7(3)	6(3)	7(3)	--	--	6(3)	6(3)	3(3)	5(3)
4. Delayed cancer fatalities (excluding thyroid) (persons)	Evac-Reloc	1(3)	5(3)	5(3)	5(3)	2(2)	5(3)	4(3)	4(3)	3(3)	4(3)
	Early Reloc	1(3)	5(3)	5(3)	5(3)	2(2)	5(3)	4(3)	4(3)	3(3)	4(3)
	Late Reloc	1(3)	6(3)	6(3)	6(3)	--	--	4(3)	4(3)	3(3)	4(3)
5. Delayed thyroid cancer fatalities (persons)	Evac-Reloc	2(2)	2(3)	2(3)	2(3)	3(1)	2(3)	9(2)	9(2)	7(2)	1(3)
	Early Reloc	2(2)	2(3)	2(3)	2(3)	3(1)	2(3)	9(2)	1(3)	8(2)	1(3)
	Late Reloc	2(2)	2(3)	2(3)	2(3)	--	--	1(3)	1(3)	8(2)	1(3)
6. Total person-rems	Evac-Reloc	2(7)	8(7)	7(7)	6(7)	3(6)	8(7)	5(7)	5(7)	4(7)	6(7)
	Early Reloc	2(7)	8(7)	8(7)	8(7)	3(6)	8(7)	5(7)	5(7)	5(7)	6(7)
	Late Reloc	3(7)	9(7)	8(7)	9(8)	--	--	6(7)	6(7)	5(7)	7(7)
7. Cost of offsite mitigation measures (1980 dollars)	Evac-Reloc	1(9)	5(9)	5(9)	5(9)	9(7)	5(9)	2(9)	2(9)	2(9)	3(9)
	Early Reloc	1(9)	5(9)	5(9)	5(9)	4(7)	5(9)	2(9)	2(9)	2(9)	3(9)
	Late Reloc	1(9)	5(9)	5(9)	5(9)	--	--	2(9)	2(9)	2(9)	3(9)
8. Land area for long-term interdiction (m^2)	Evac-Reloc	3(7)	1(8)	1(8)	2(8)	3(5)	1(8)	5(7)	6(7)	5(7)	8(7)
	Early Reloc	3(7)	1(8)	1(8)	2(8)	3(5)	1(8)	5(7)	6(7)	5(7)	8(7)
	Late Reloc	3(7)	1(8)	1(8)	2(8)	--	--	5(7)	6(7)	5(7)	8(7)

APPENDIX L

CONSEQUENCES AND RISKS OF RELEASE CATEGORIES INITIATED BY SEVERE EARTHQUAKES AND THOSE OF RELEASE CATEGORIES INITIATED BY OTHER CAUSES

Probability distributions of accident consequences and probability-weighted values of these consequences (i.e., risks) are presented and discussed in Sections 5.9.4.5(3), 5.9.4.5(4), and 5.9.4.5(6). The results presented in those sections were the combined results from release categories initiated by internal causes, fires and low to moderately severe earthquakes, and from release categories initiated by severe earthquakes. The severe earthquake initiated release categories were analyzed with the assumption of late relocation (Late Reloc) mode of offsite emergency response (see Section 5.9.4.5(2) and Table 5.11f). Release categories initiated by causes other than severe earthquakes were analyzed with the assumption of evacuation and relocation (Evac-Reloc) mode of offsite emergency response (see Section 5.9.4.5(2) and Table 5.11f). A separate display of radiological contributions to the overall results (presented in sections cited above) from release categories initiated by severe earthquakes and from release categories initiated by causes other than severe earthquakes is provided here. Additionally, breakdowns of societal consequences of early fatalities and latent cancer fatalities in terms of contributions from spatial intervals up to 50 miles (80 km) from the Limerick reactors are also presented.

Figures L.1 through L.20 display the breakdowns of each of the graphical plots presented in Figures 5.4b through 5.4l in the sections cited above into two components--one ascribed to the severe earthquakes and the other ascribed to the other causes. In Figures L.1 through L.20, the graphical plots of Figures 5.4b through 5.4l are reproduced for easy reference.

Tables L.1a and b provide a breakdown of each category of risk shown in Table 5.11h into the two components as stated above. From these tables it is apparent that the release categories initiated by severe earthquakes are the dominant contributors to the risk of early fatality (with supportive or minimal medical treatment). These release categories contribute almost equally as the release categories initiated by other causes to the risk of early injury. However, the release categories initiated by causes other than severe earthquakes are the dominant contributors to the other types of risk in Tables L.1a and b.

Table L.2 shows the contributions to the risk of early fatality with supportive medical treatment from the spatial intervals within 50 miles (80 km) of the plant. Contributions from each spatial interval is also broken down into component contributions ascribed to severe earthquakes and the other causes.

Table L.3 shows similar results for early fatality as in Table L.2, but with minimal medical treatment.

Table L.4 shows the risk of latent cancer fatality in similar fashion as in Table L.2 for early fatality. Latent cancer fatality risks shown in Table L.4 include risks of both thyroid and nonthyroid cancer fatalities.

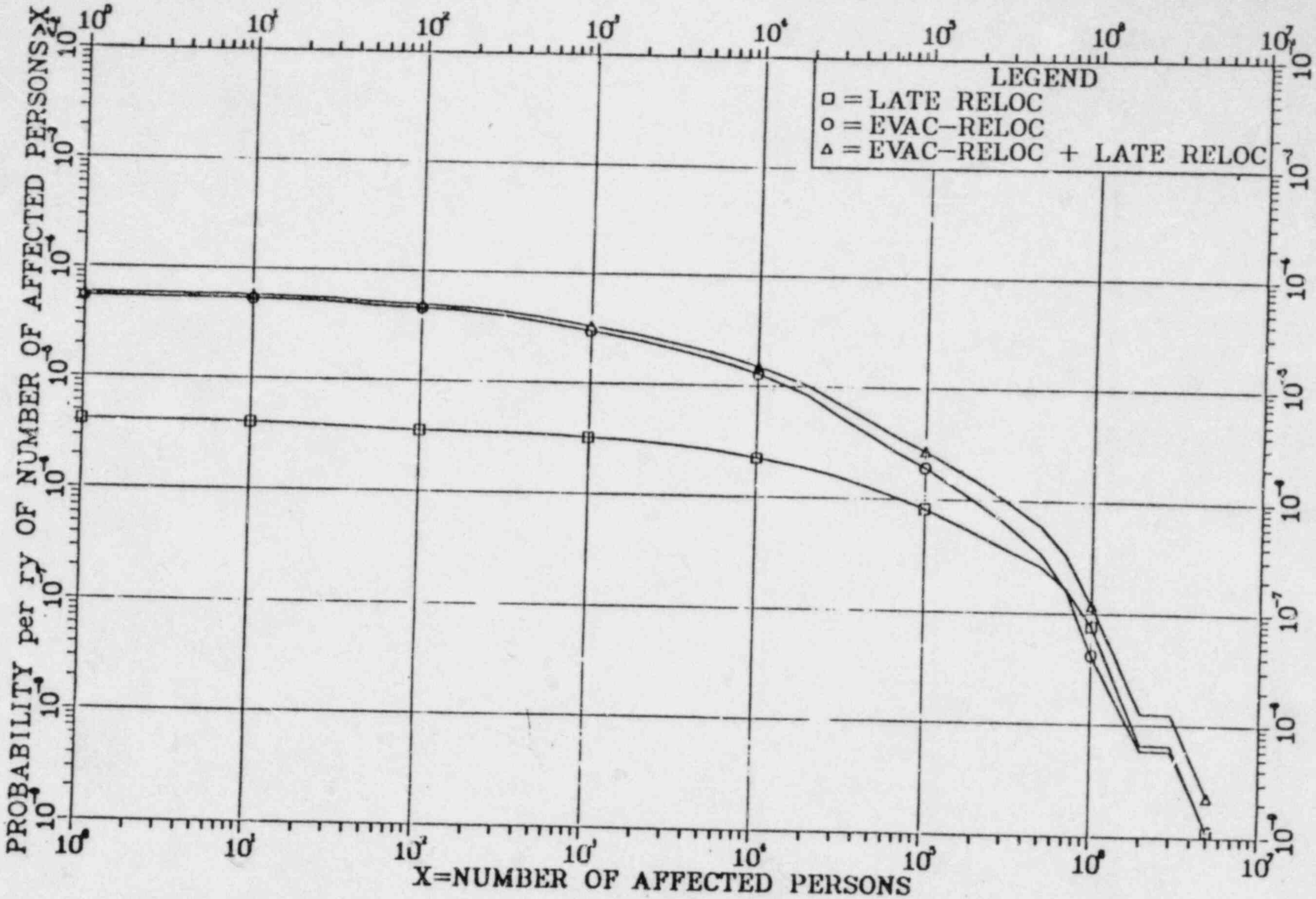


Figure L.1 Probability distribution of population with whole body dose greater than or equal to 25 rems

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

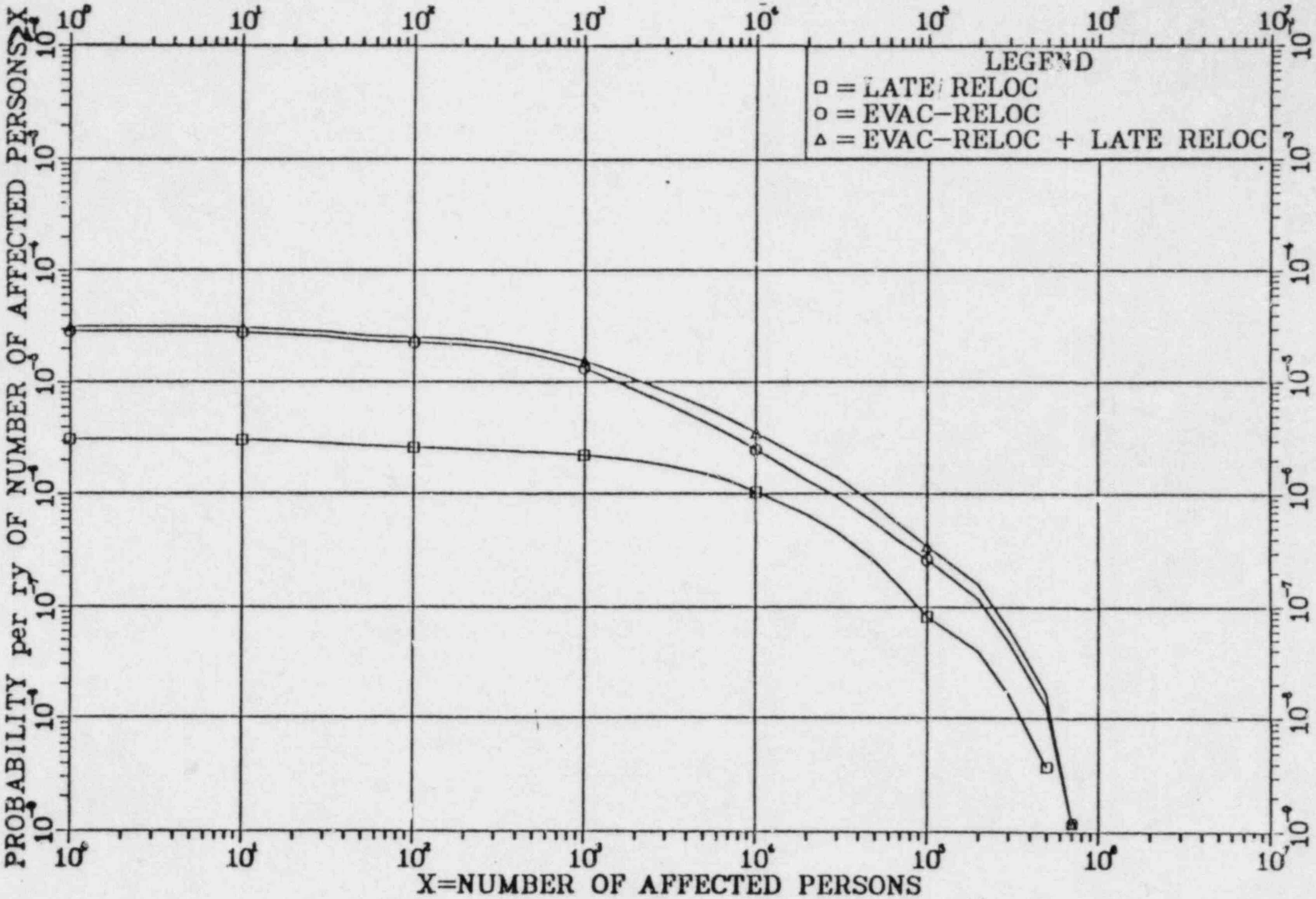
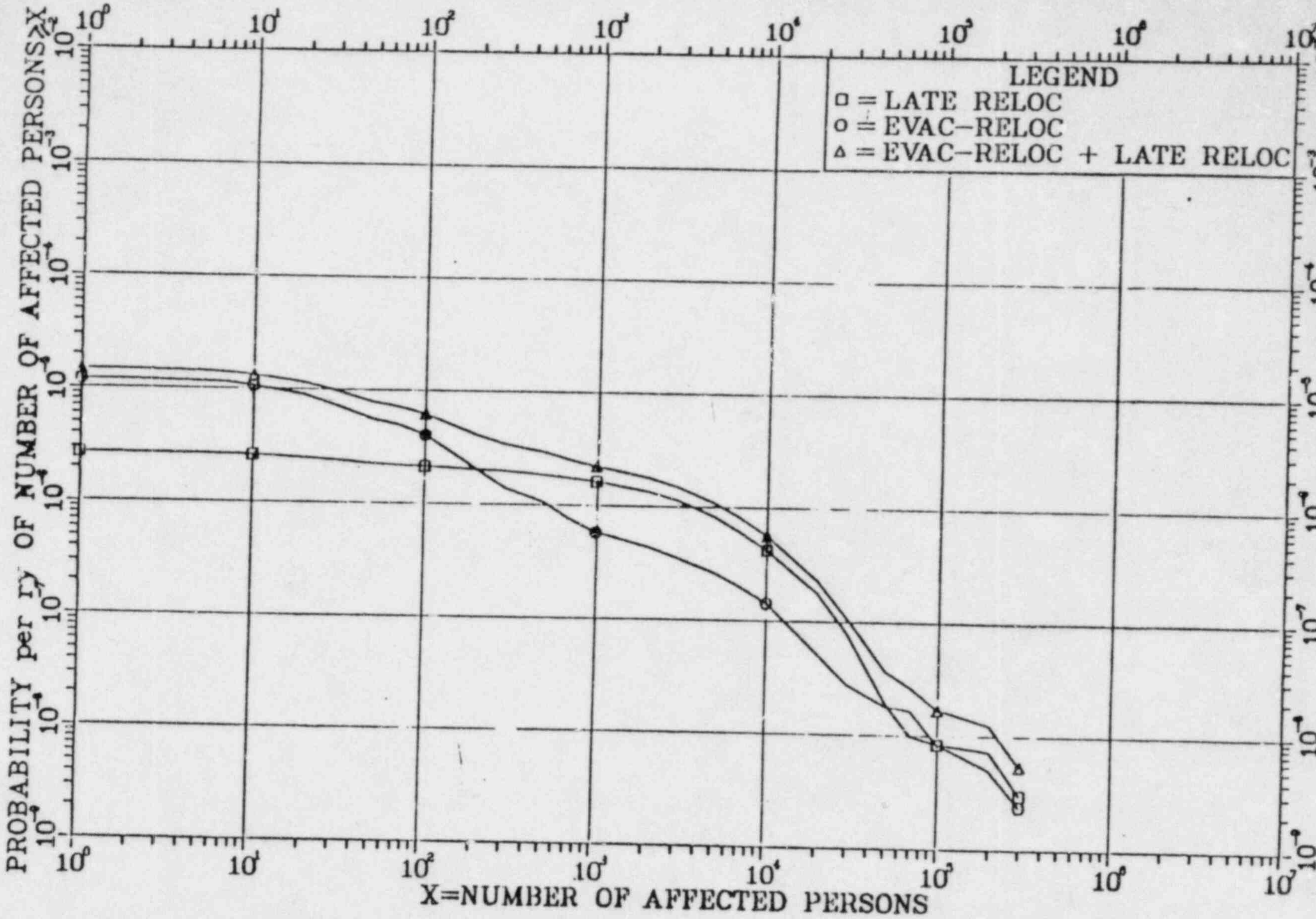


Figure L.2 Probability distribution of population with thyroid dose greater than or equal to 300 rems

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.



.Figure L.3 Probability distribution of population with total bone marrow dose greater than or equal to 200 rems

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

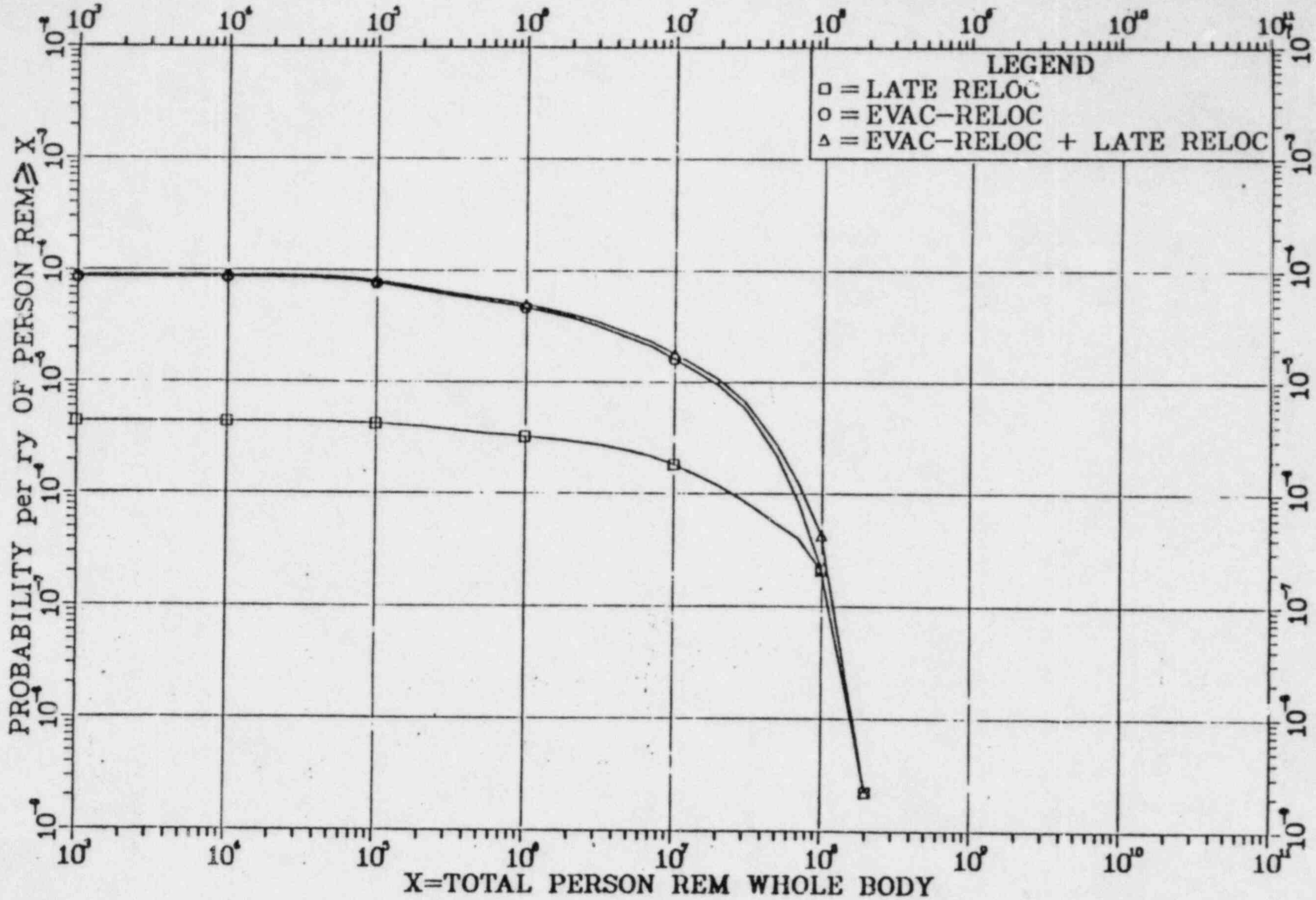


Figure L.4 Probability distribution of population exposure within 50 miles (80 km)

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

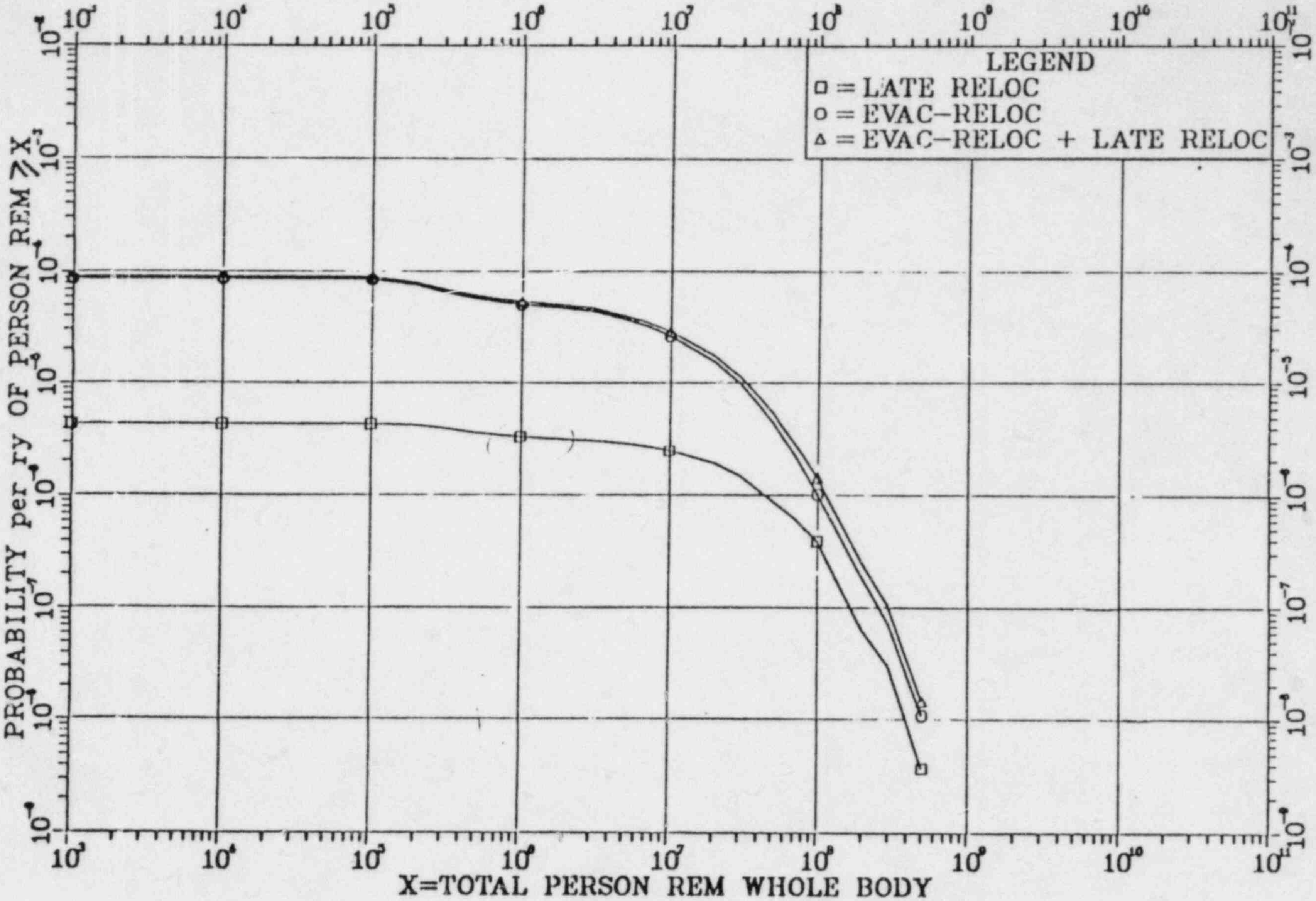


Figure L.5 Probability distribution of population exposure, entire region

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

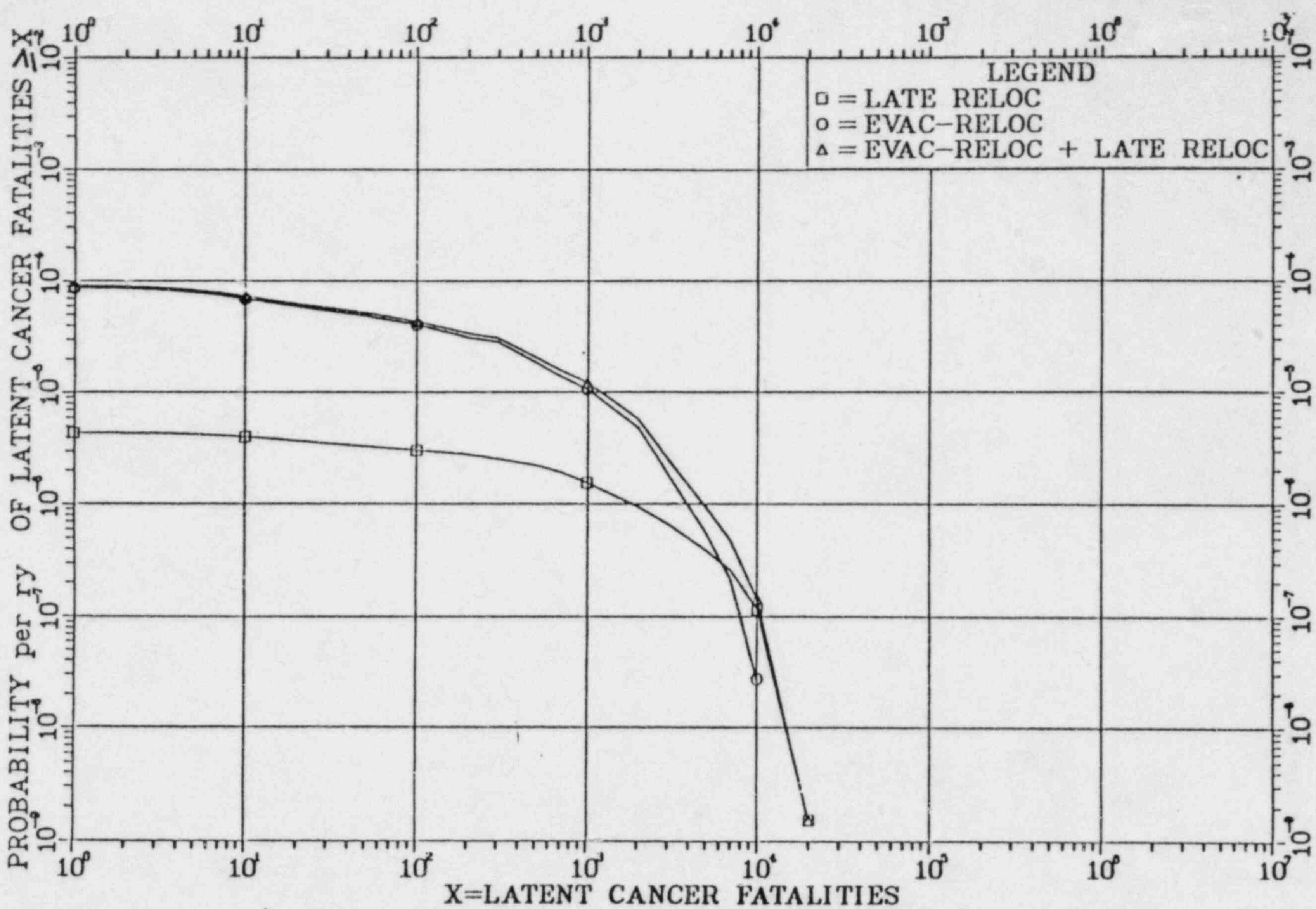


Figure L.6 Probability distribution of latent cancer fatalities, excluding thyroid within 50 miles (80 km)

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

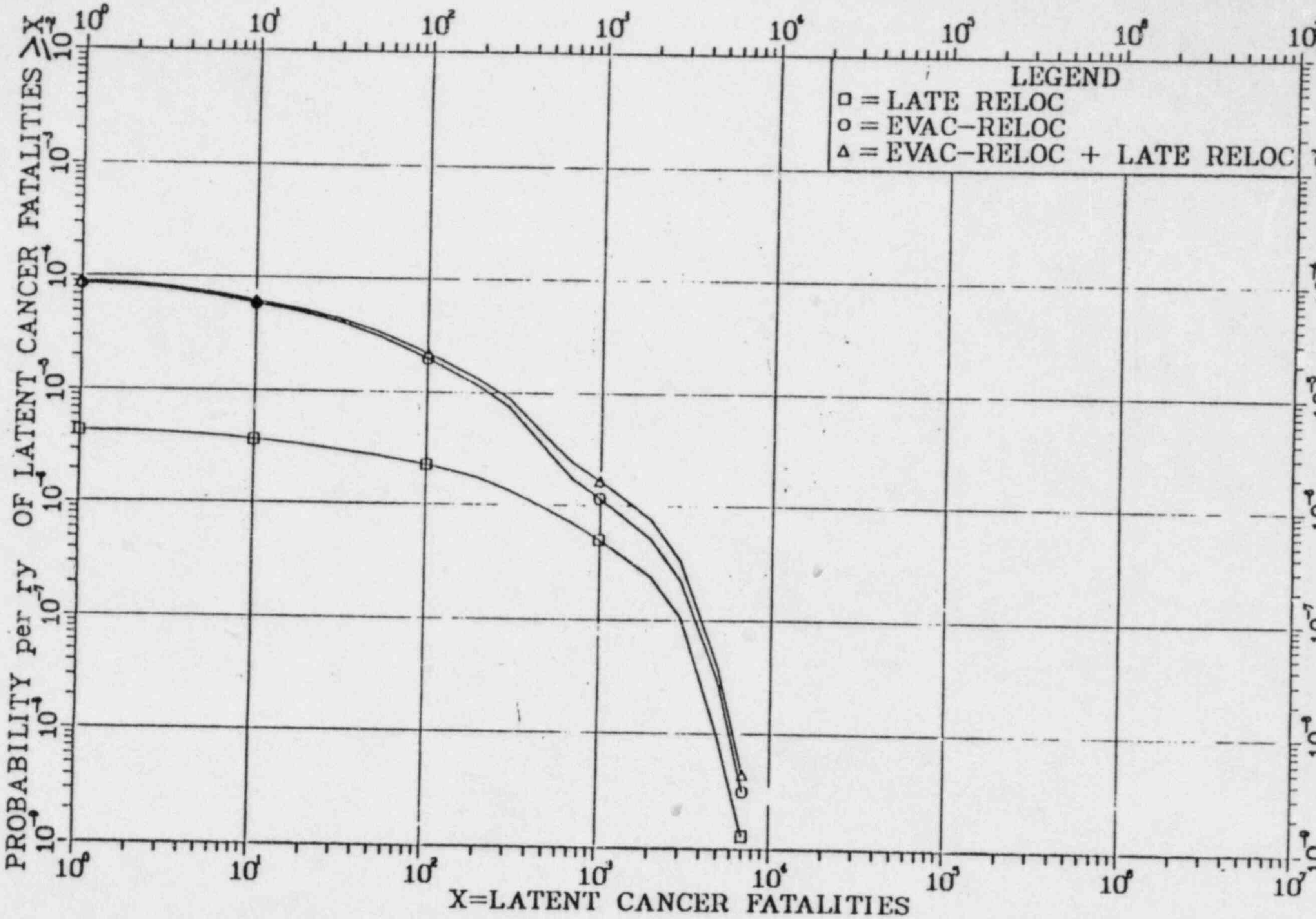


Figure L.7 Probability distribution of latent thyroid cancer fatalities, 50 miles (80 km)

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

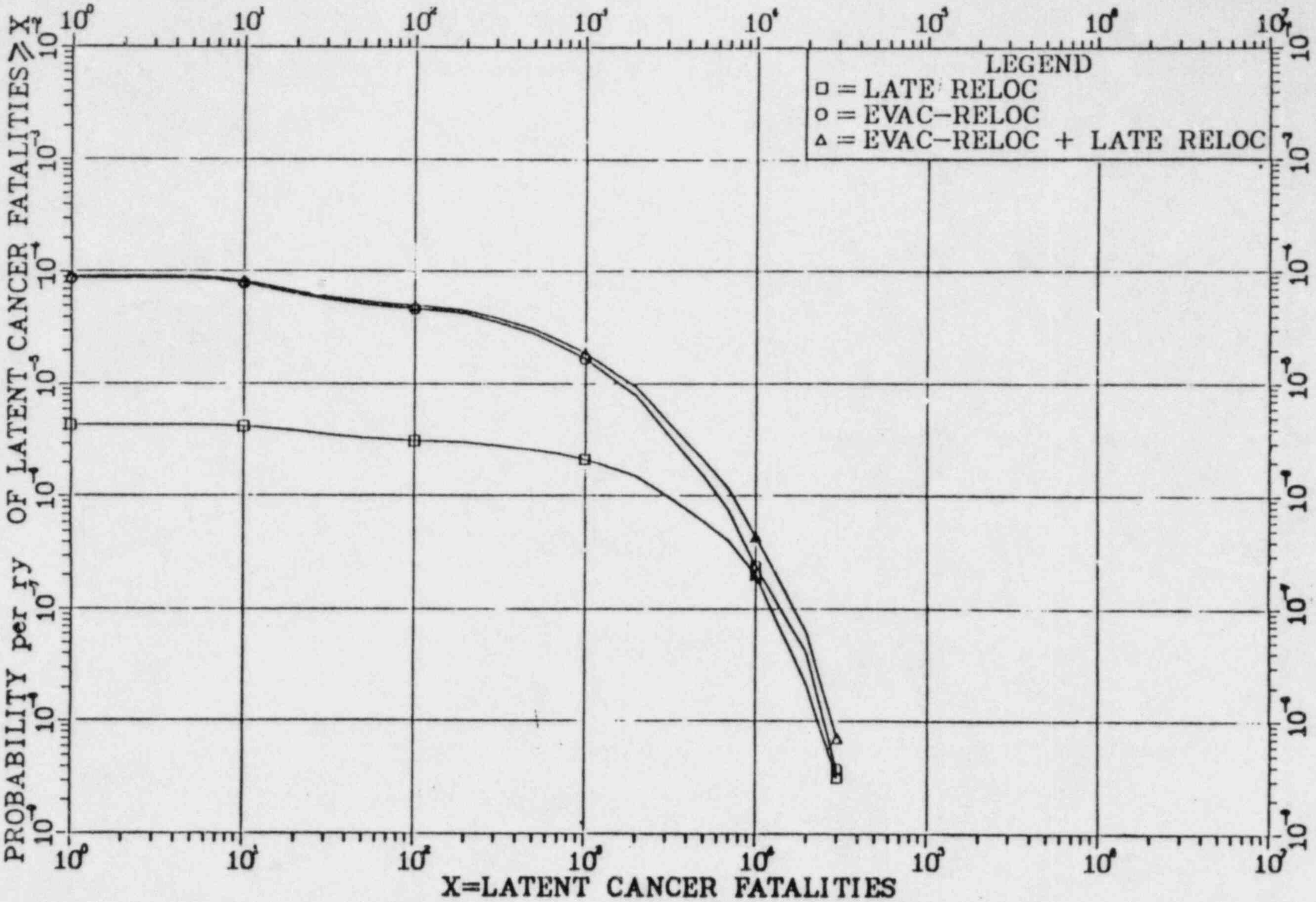


Figure L.8 Probability distribution of latent cancer fatalities, excluding thyroid, entire population

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

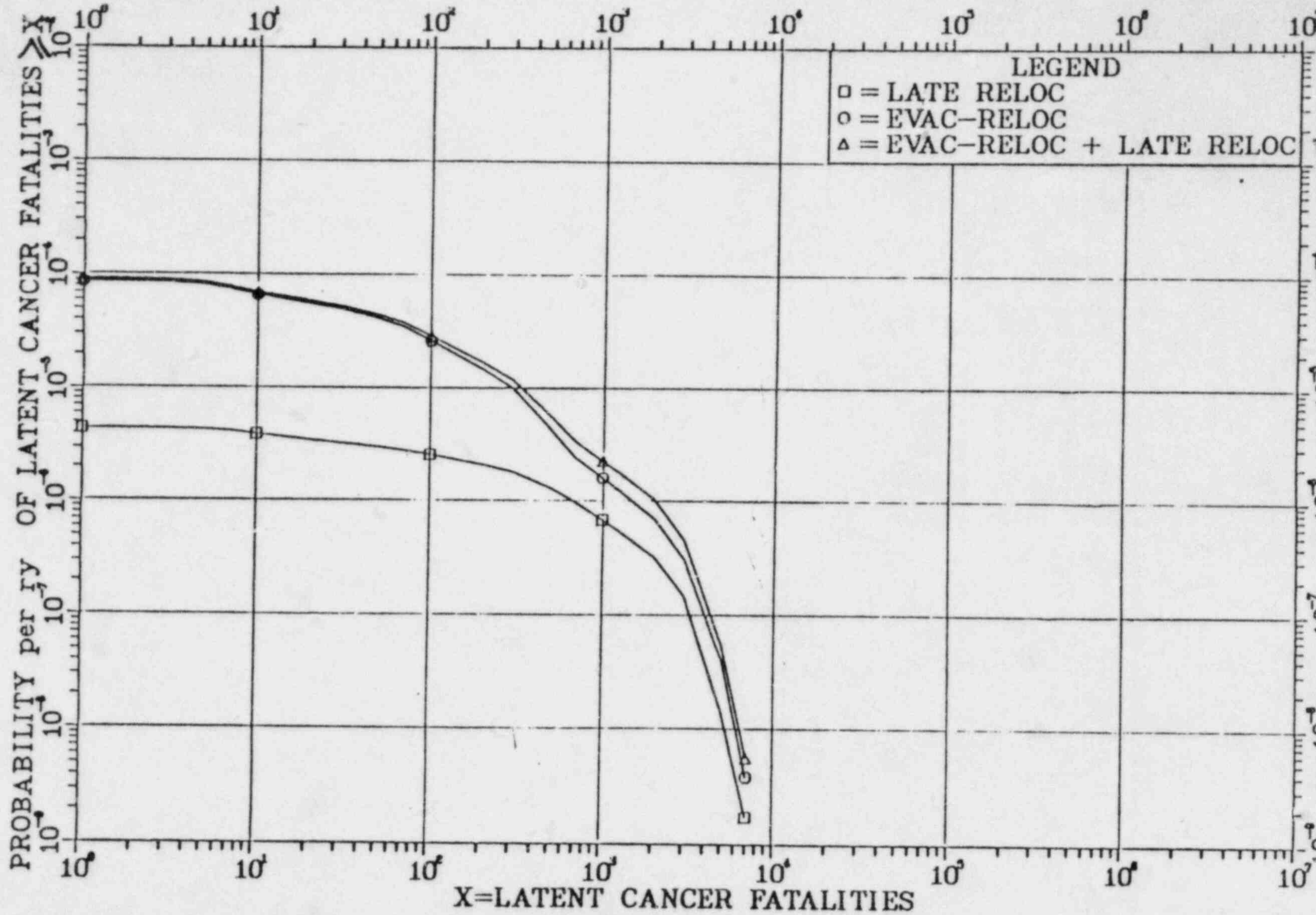


Figure L.9 Probability distribution of latent thyroid cancer fatality, entire population

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

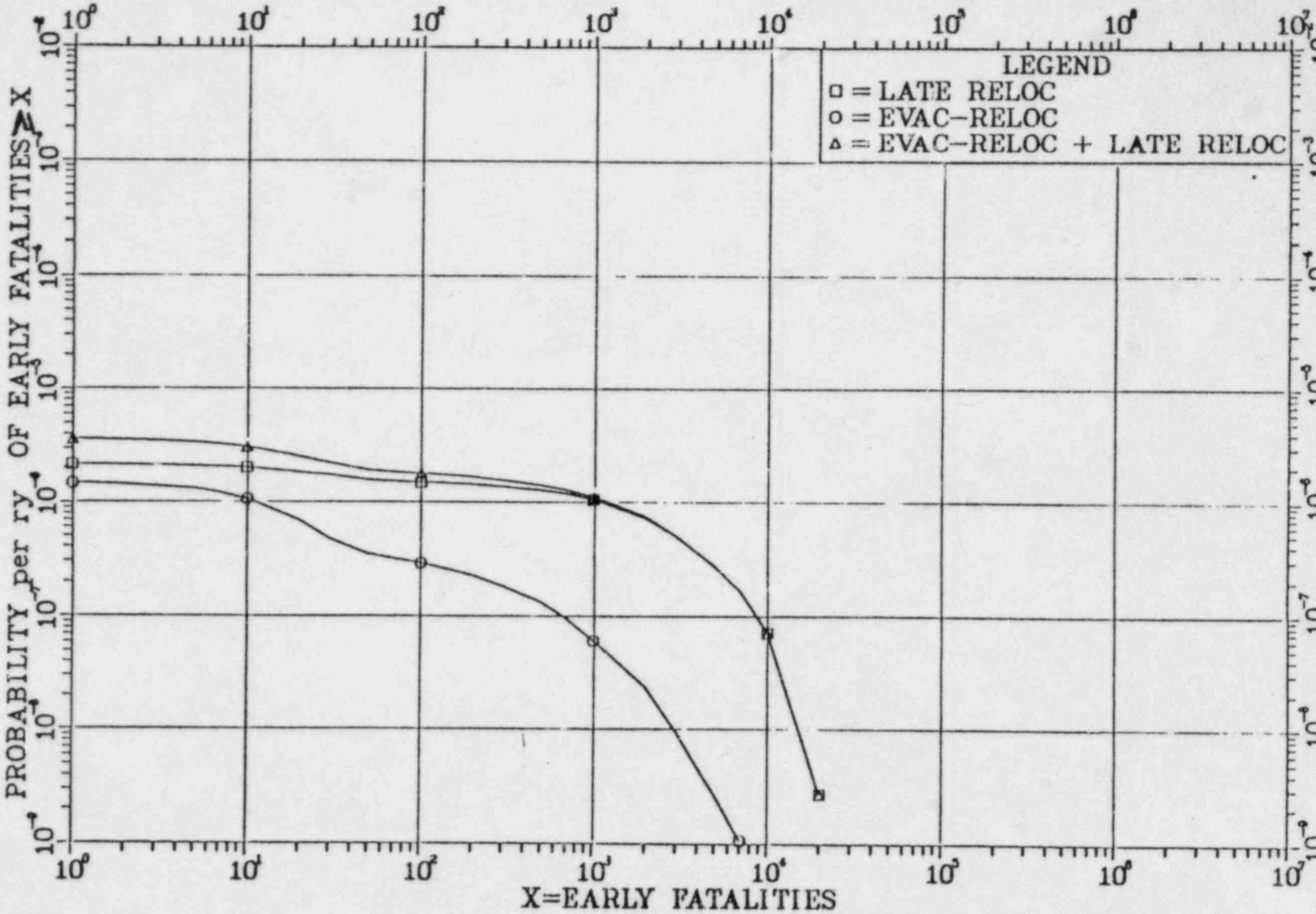


Figure L.10 Probability distribution of early fatality with supportive medical treatment

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

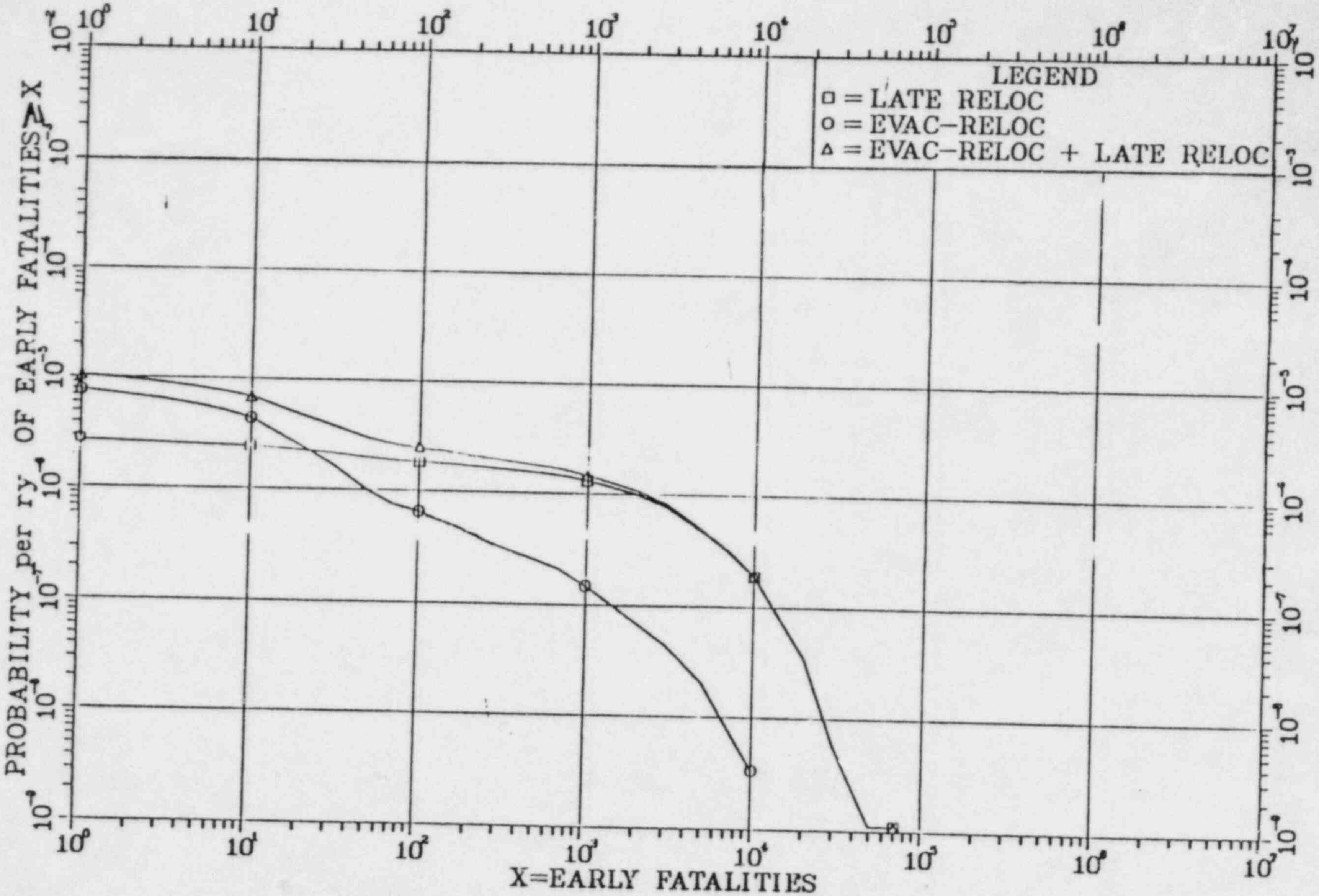


Figure L.11 Probability distribution of early fatalities with minimal medical treatment

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

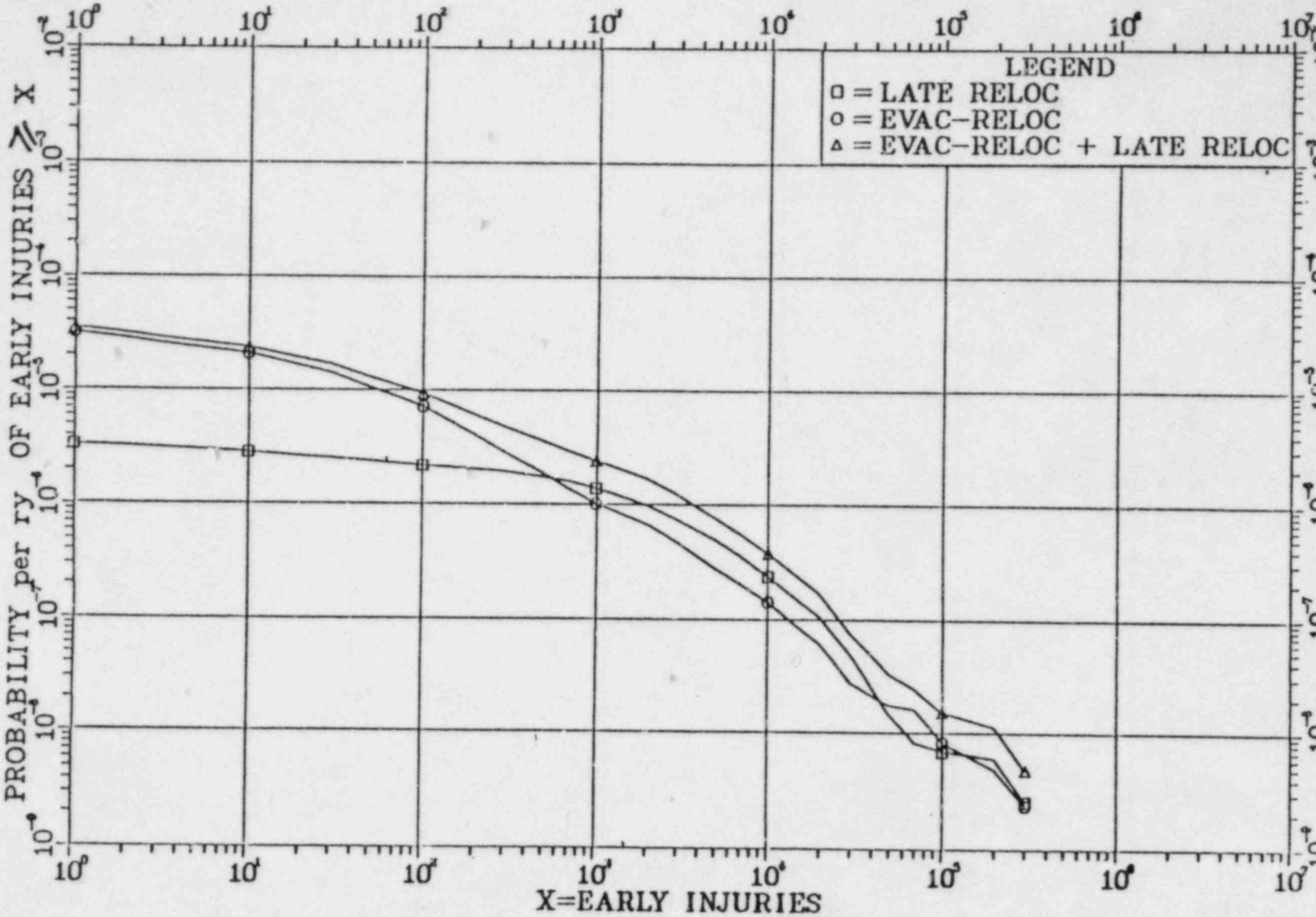


Figure L.12 Probability distribution of early injuries

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

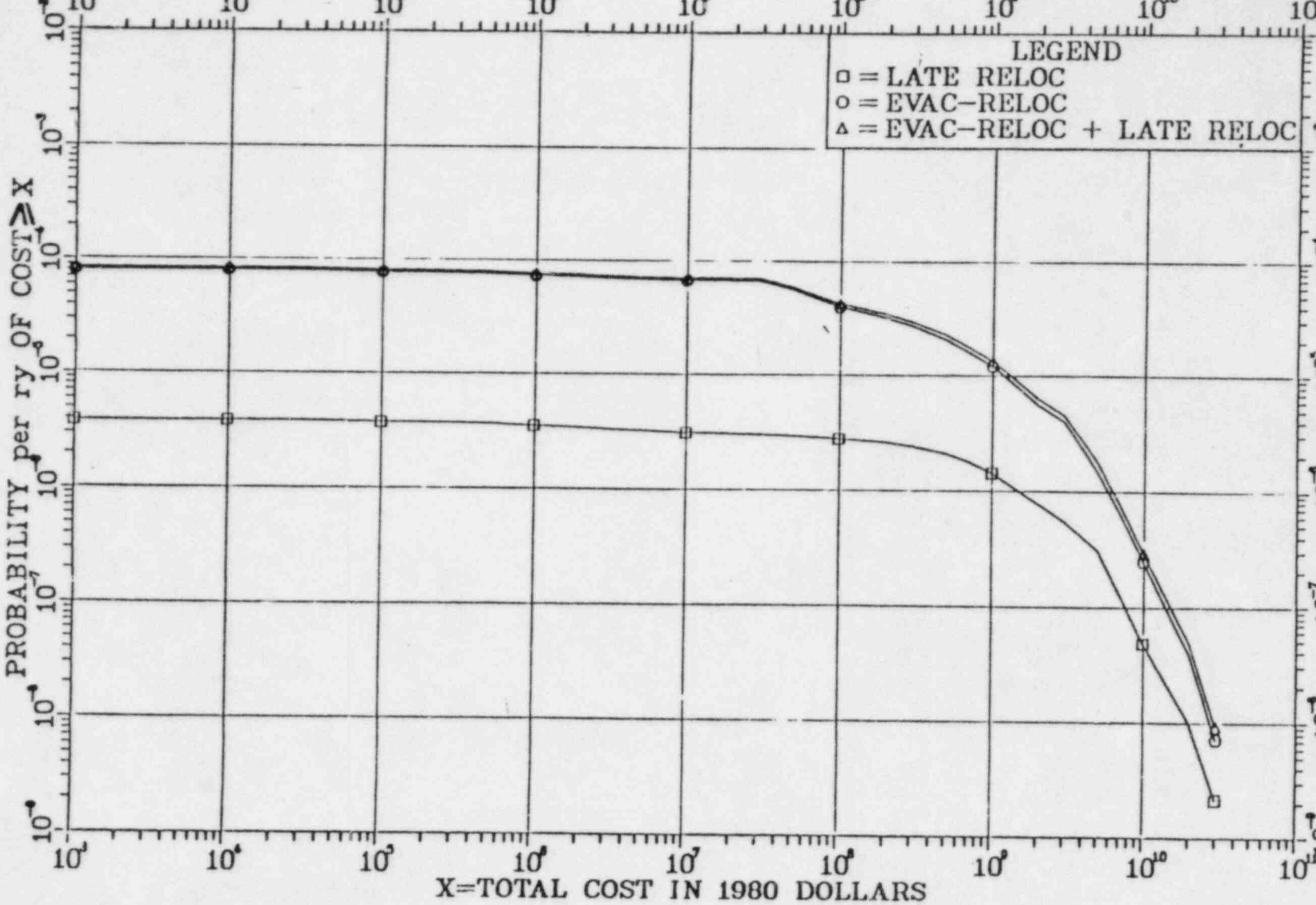


Figure L.13 Probability distribution of mitigation measures cost

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

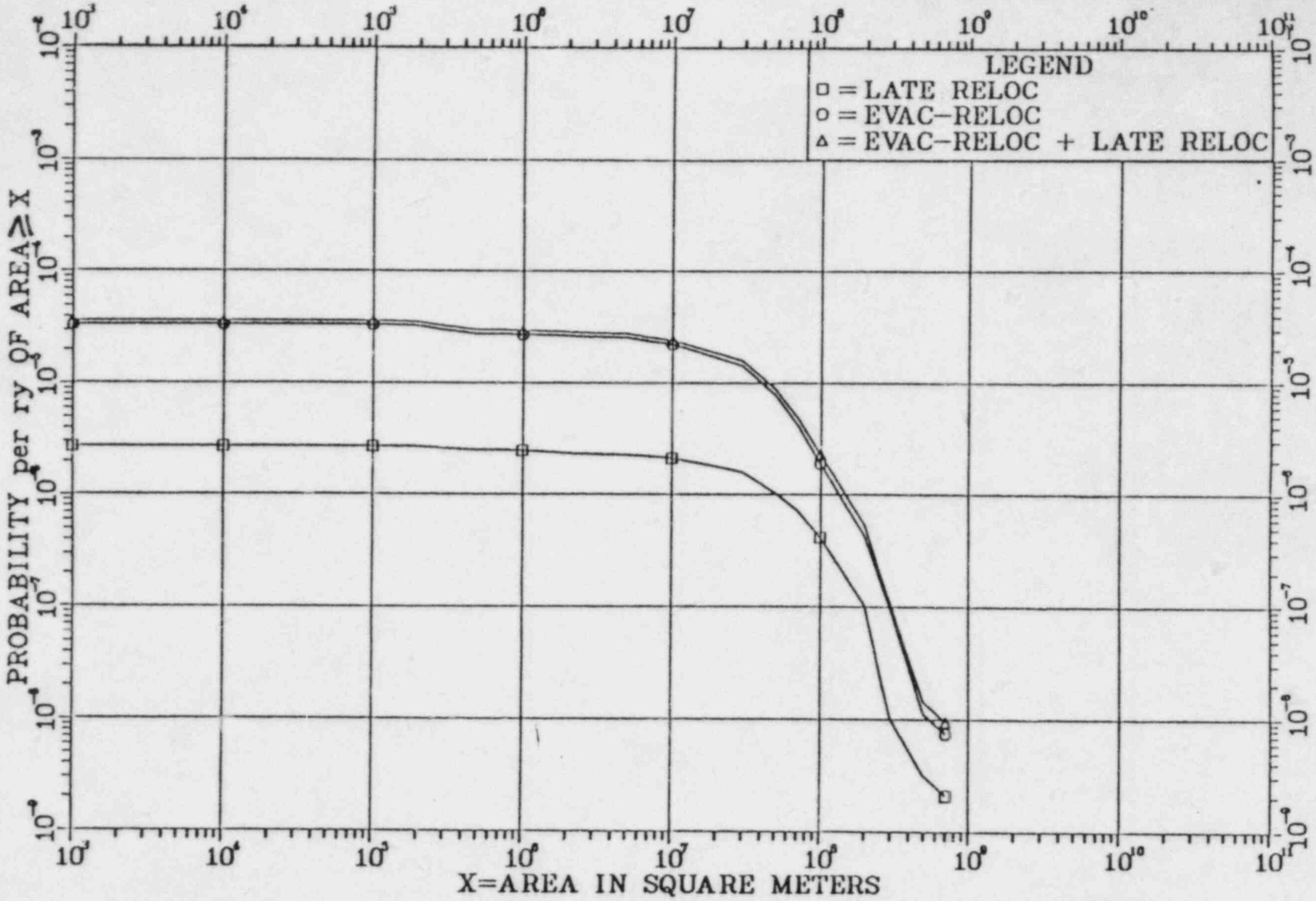


Figure L.14 Probability distribution of land area interdiction

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

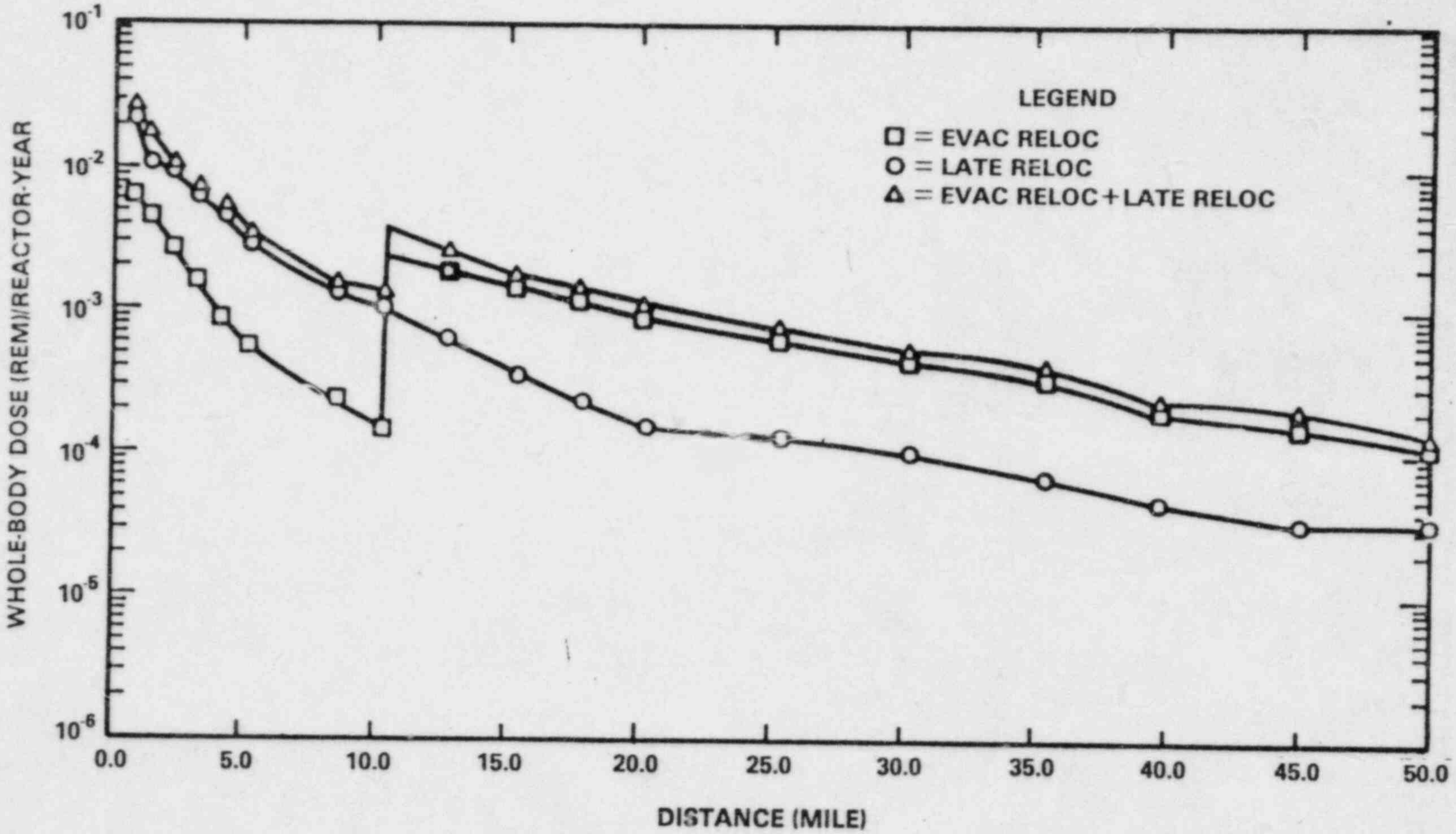


Figure L.15 Individual risk of downwind dose versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

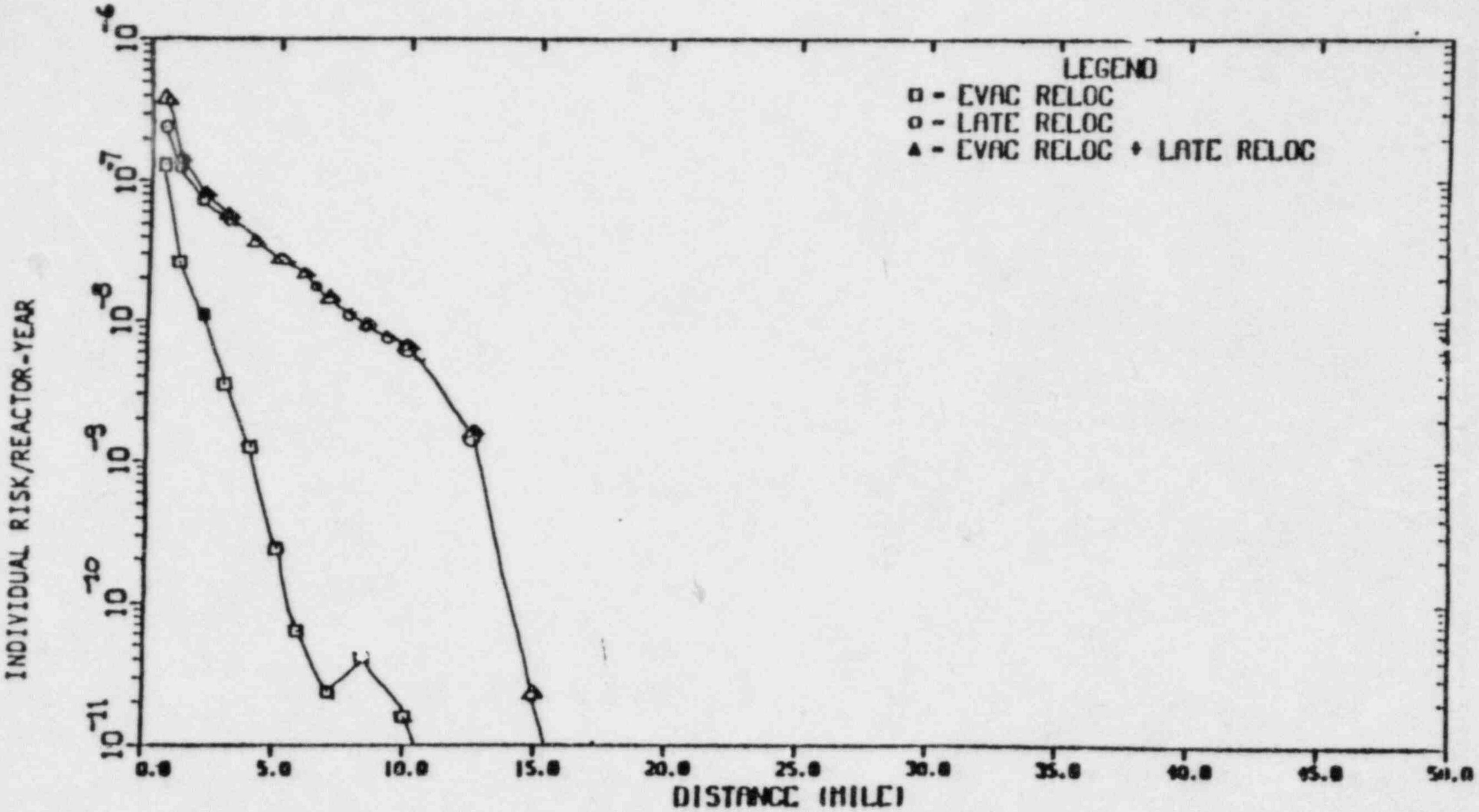


Figure L.16 Individual risk of early fatality with supportive medical treatment versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

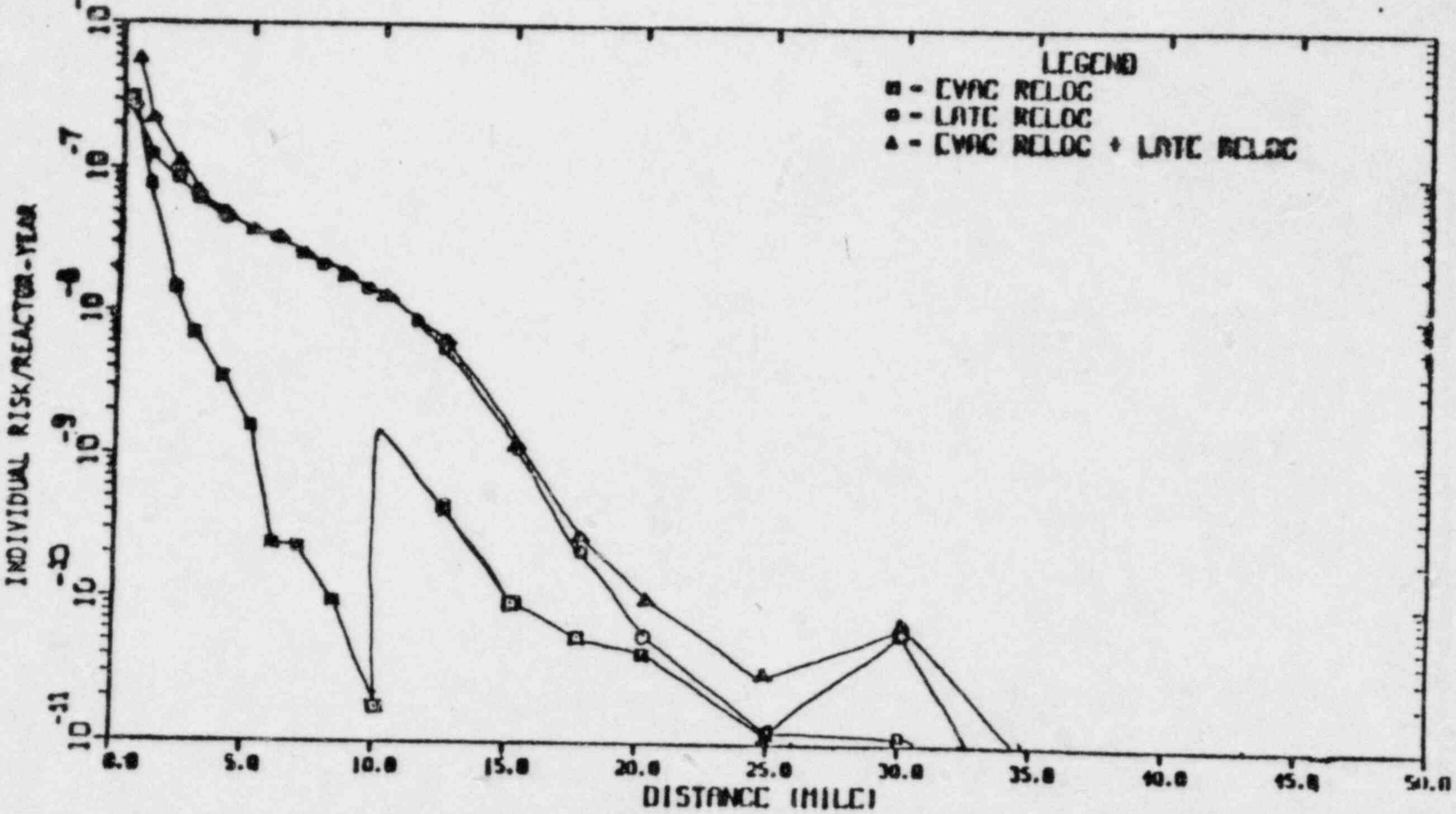


Figure L.17 Individual risk of early fatality with minimal medical treatment versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

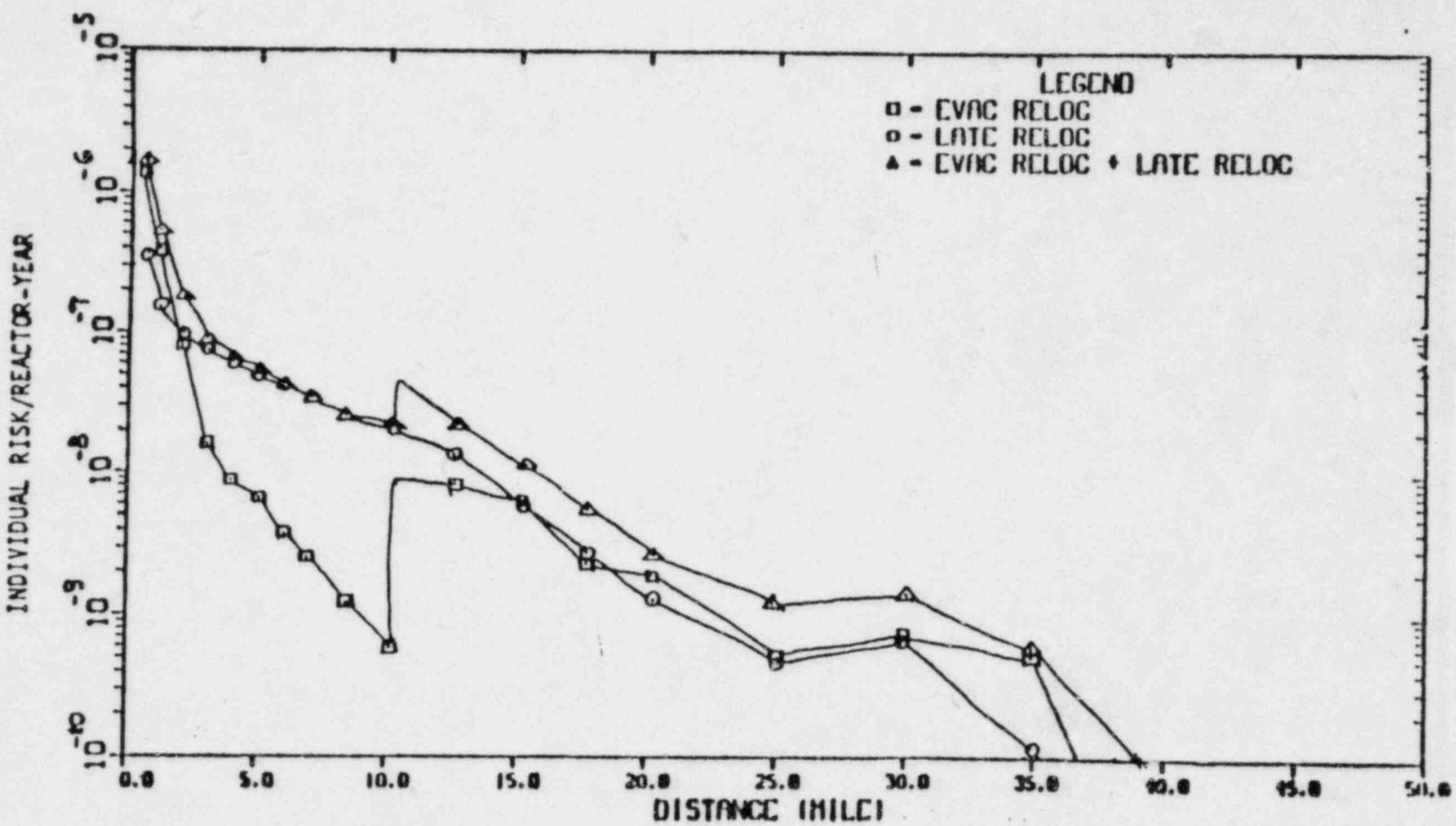


Figure L.18 Individual risk of early injury versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

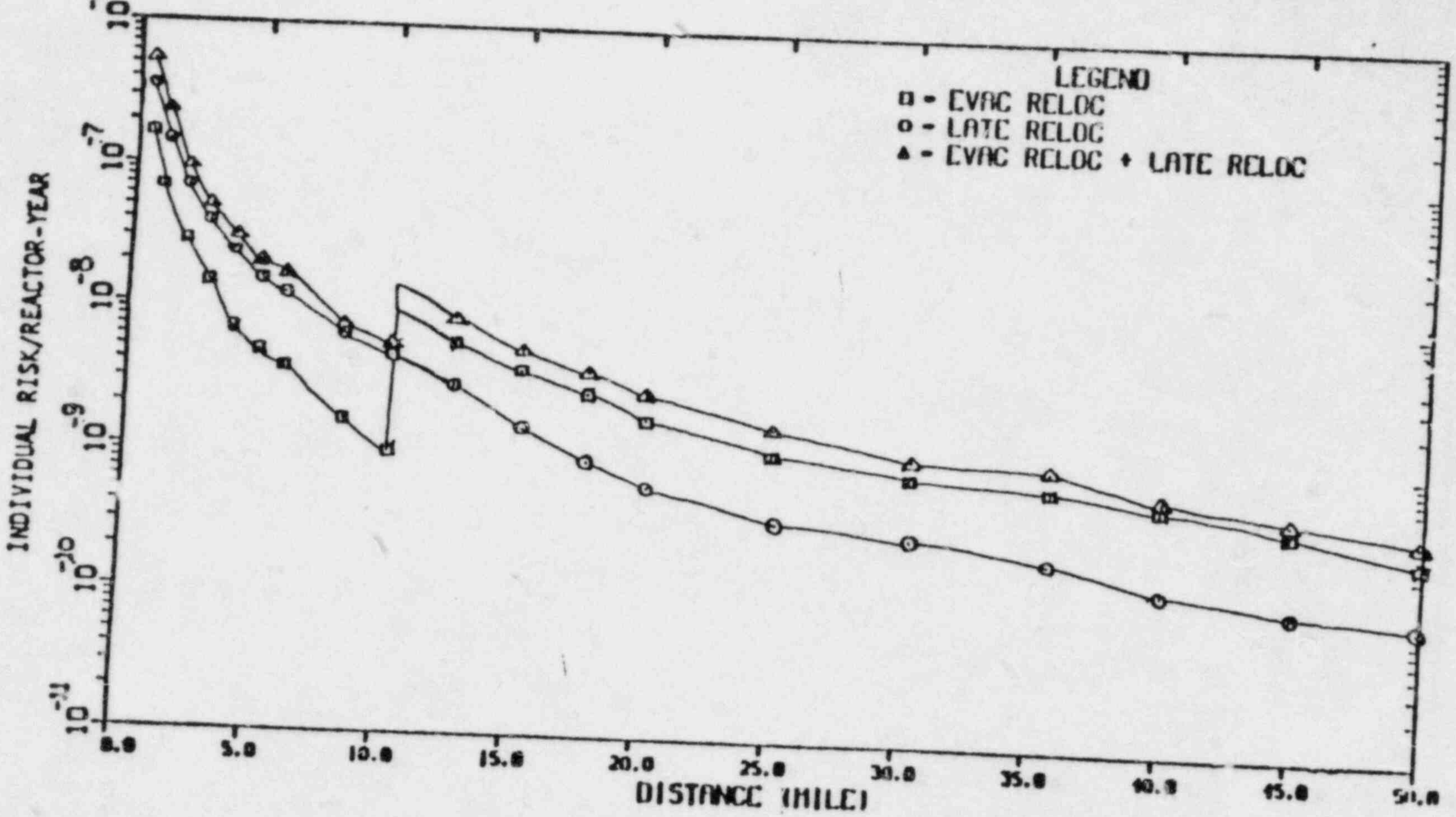


Figure L.19 Individual risk of latent cancer fatality (excluding thyroid) versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

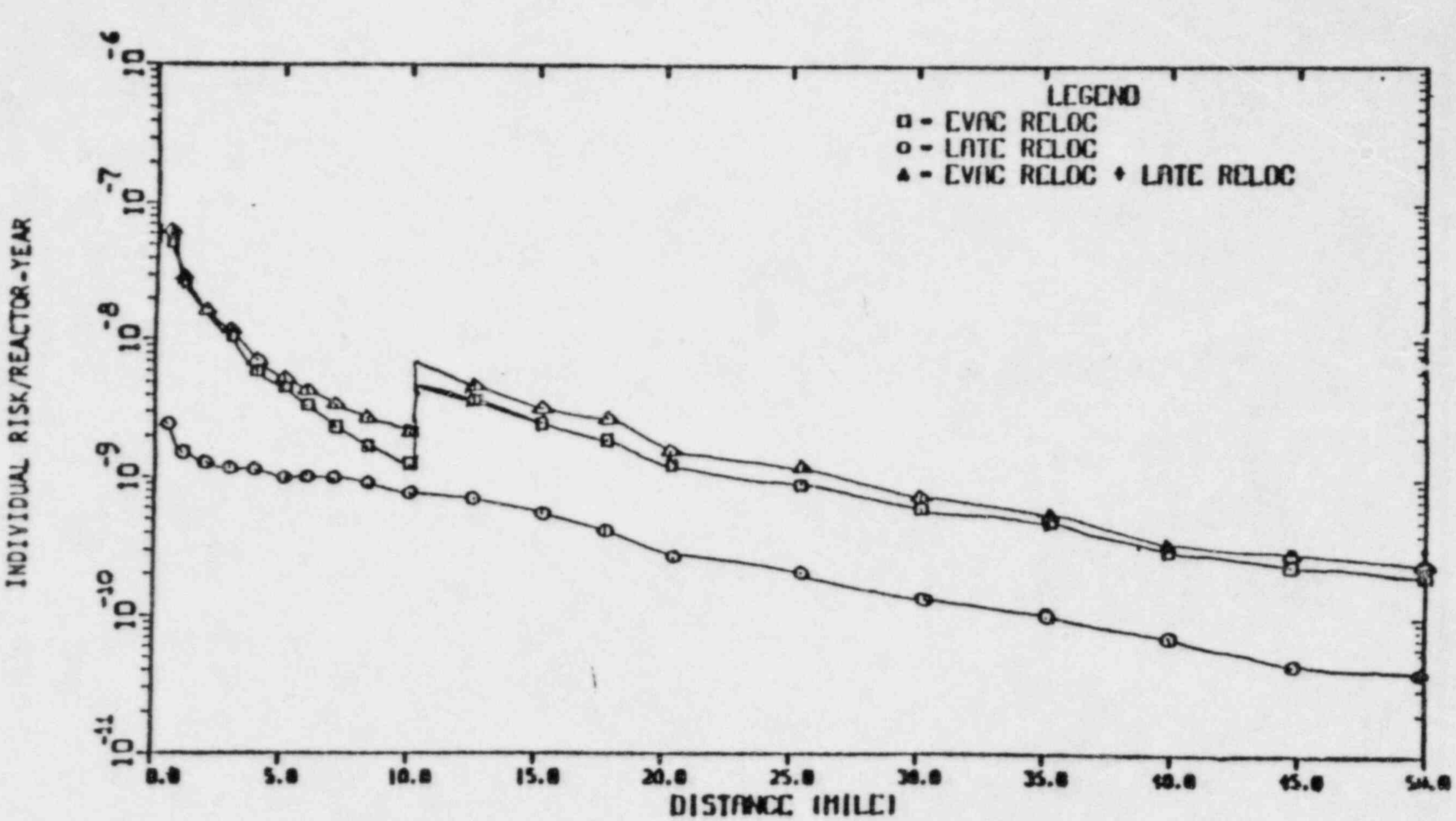


Figure L.20 Individual risk of latent thyroid cancer fatality versus distance

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties

Table L.1a Societal risks within 50 miles (80 km) of Limerick site with Evac-Reloc* and Late Reloc* offsite emergency response modes

Consequence type	Risk per reactor-year: From causes other than severe earthquakes (Evac-Reloc)	From severe earthquakes (Late Reloc)	Total
1. Early fatalities with supportive medical treatment (persons)	2(-4)**	5(-3)	5(-3)
2. Early fatalities with minimal medical treatment (persons)	7(-4)	8(-3)	8(-3)
3. Early injuries (persons)	1(-2)	1(-2)	2(-2)
4. Latent cancer fatalities (excluding thyroid) (persons)	4(-3)	7(-3)	4(-2)
5. Latent thyroid cancer fatalities (persons)	9(-3)	2(-3)	1(-2)
6. Total person-rems	6(2)	9(1)	7(2)
7. Cost of offsite mitigation measures (1980 dollars)	4(4)	5(3)	5(4)
8. Land area for long-term interdiction (square meters)	1(3)	1(2)	1(3)

*See Section 5.9.4.5(2).
**2(-4) = 2 x 10^{-4} = .0002

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table L.1b Societal risks within the entire region of Limerick site with Evac-Reloc* and Late Reloc* offsite emergency response modes

Consequence type	Risk per reactor-year		
	From causes other than severe earthquakes (Evac-Reloc)	From severe earthquakes (Late Reloc)	Total
1. Early fatalities with supportive medical treatment (persons)	2(-4)**	5(-3)	5(-3)
2. Early fatalities with minimal medical treatment (persons)	7(-4)	8(-3)	8(-3)
3. Early injuries (persons)	1(-2)	1(-2)	2(-2)
4. Latent cancer fatalities (excluding thyroid) (persons)	6(-2)	1(-2)	7(-2)
5. Latent thyroid cancer fatalities (persons)	1(-2)	2(-3)	1(-2)
6. Total person-rems	1(3)	1(2)	1(3)
7. Cost of offsite mitigation measures (1980 dollars)	5(4)	6(3)	5(4)
8. Land area for long-term interdiction (square meters)	1(3)	2(2)	1(3)

*See Section 5.9.4.5(2).
**2(-4) = 2 x 10^{-4} = .0002

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table L.2 Contributions to risk of early fatality with supportive medical treatment from spatial intervals within 50 miles (80 km) of Limerick site with Evac-Reloc* and Late Reloc* offsite emergency response modes

	Risk per reactor-year		
Spatial interval from (mi) - to (mi)†	From causes other than severe earthquakes (Evac-Reloc) (persons)	From servere earthquakes (Late Reloc) (persons)	Total (persons)
0.0 - 0.5**	2(-5)***	4(-5)	6(-5)
0.5 - 1.0	1(-5)	6(-5)	8(-5)
1.0 - 1.5****	4(-5)	3(-4)	3(-4)
1.5 - 2.0	4(-5)	3(-4)	4(-4)
2.0 - 2.5	4(-5)	4(-4)	4(-4)
2.5 - 3.0	2(-5)	3(-4)	4(-4)
3.0 - 3.5	3(-5)	6(-4)	6(-4)
3.5 - 4.0	2(-5)	5(-4)	6(-4)
4.0 - 4.5	6(-6)	3(-4)	3(-4)
4.5 - 5.0	2(-6)	3(-4)	3(-4)
5.0 - 6.0	9(-7)	3(-4)	3(-4)
6.0 - 7.0	4(-7)	2(-4)	2(-4)
7.0 - 8.5	1(-6)	3(-4)	3(-4)
8.5 - 10.0	6(-7)	2(-4)	2(-4)
10.0 - 12.5	2(-6)	3(-4)	3(-4)
12.5 - 15.0	2(-8)	2(-6)	2(-6)
15.0 - 17.5	3(-8)	5(-8)	8(-8)
17.5 - 20.0	4(-8)	0	4(-8)
20.0 - 25.0	0	0	0
25.0 - 30.0	0	7(-7)	7(-7)
30.0 - 35.0	0	0	0
35.0 - 40.0	0	0	0
40.0 - 45.0	0	0	0
45.0 - 50.0	0	0	0
Total	2(-4)	5(-3)	5(-3)

†To change miles to km, multiply the values shown by 1.609.
*See Section 5.9.4.5(2).
**This circular zone includes the Site Exclusion Area.
***2(-5) = 2 x 10^{-5} = .00002
****93% of the area of this annulus is included within an annulus 1-mile wide outside of the site exclusion area boundary.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table L.3 Contributions to risk of early fatality with minimal medical treatment from spatial intervals within 50 miles (80 km) of Limerick site with Evac-Reloc* and Late Reloc* offsite emergency response modes

Spatial interval from (mi) - to (mi)†	Risk per reactor-year: From causes other than severe earthquakes (Evac-Reloc) (persons)	From servere earthquakes (Late Reloc) (persons)	Total (persons)
0.0 - 0.5**	5(-5)***	4(-5)	1(-4)
0.5 - 1.0	4(-5)	7(-5)	1(-4)
1.0 - 1.5****	8(-5)	3(-4)	4(-4)
1.5 - 2.0	6(-5)	4(-4)	5(-4)
2.0 - 2.5	7(-5)	5(-4)	6(-4)
2.5 - 3.0	5(-5)	4(-4)	5(-4)
3.0 - 3.5	6(-5)	8(-4)	8(-4)
3.5 - 4.0	5(-5)	7(-4)	8(-4)
4.0 - 4.5	2(-5)	4(-4)	4(-4)
4.5 - 5.0	2(-5)	4(-4)	4(-4)
5.0 - 6.0	1(-5)	4(-4)	5(-4)
6.0 - 7.0	3(-6)	4(-4)	4(-4)
7.0 - 8.5	3(-6)	5(-4)	5(-4)
8.5 - 10.0	7(-7)	5(-4)	5(-4)
10.0 - 12.5	9(-5)	1(-3)	1(-3)
12.5 - 15.0	9(-6)	1(-4)	1(-4)
15.0 - 17.5	1(-5)	5(-5)	7(-5)
17.5 - 20.0	1(-5)	2(-5)	3(-5)
20.0 - 25.0	1(-5)	1(-5)	3(-5)
25.0 - 30.0	2(-5)	2(-4)	2(-4)
30.0 - 35.0	1(-5)	9(-6)	2(-5)
35.0 - 40.0	7(-8)	1(-6)	1(-6)
40.0 - 45.0	3(-8)	8(-7)	9(-7)
45.0 - 50.0	3(-6)	3(-7)	3(-6)
Total	7(-4)	8(-3)	8(-3)

†To change miles to km, multiply the values shown by 1.609.
*See Section 5.9.4.5(2).
**This circular zone includes the Site Exclusion Area.
***5(-5) = 5 x 10^{-5} = .00005
****93% of the area of this annulus is included within an annulus 1-mile wide outside of the site exclusion area boundary.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table L.4 Contributions to risk of latent cancer (including thyroid) fatality from spatial intervals within 50 miles (80 km) of Limerick site with Evac-Reloc* and Late Reloc* offsite emergency response modes

	Risk per reactor-year		
Spatial interval from (mi) - to (mi)†	From causes other than severe earthquakes (Evac-Reloc) (persons)	From severe earthquakes (Late Reloc) (persons)	Total (persons)
0.0 - 0.5**	3(-5)***	1(-6)	3(-5)
0.5 - 1.0	5(-5)	3(-6)	5(-5)
1.0 - 1.5****	2(-4)	2(-5)	2(-4)
1.5 - 2.0	2(-4)	3(-5)	2(-4)
2.0 - 2.5	2(-4)	4(-5)	3(-4)
2.5 - 3.0	2(-4)	4(-5)	2(-4)
3.0 - 3.5	3(-4)	9(-5)	4(-4)
3.5 - 4.0	3(-4)	9(-5)	4(-4)
4.0 - 4.5	1(-4)	5(-5)	2(-4)
4.5 - 5.0	1(-4)	5(-5)	2(-4)
5.0 - 6.0	2(-4)	8(-5)	2(-4)
6.0 - 7.0	2(-4)	8(-5)	2(-4)
7.0 - 8.5	2(-4)	1(-4)	4(-4)
8.5 - 10.0	2(-4)	1(-4)	4(-4)
10.0 - 12.5	3(-3)	8(-4)	4(-3)
12.5 - 15.0	1(-3)	2(-4)	1(-3)
15.0 - 17.5	2(-3)	4(-4)	3(-3)
17.5 - 20.0	2(-3)	4(-4)	2(-3)
20.0 - 25.0	7(-3)	1(-3)	8(-3)
25.0 - 30.0	1(-2)	2(-3)	2(-2)
30.0 - 35.0	6(-3)	1(-3)	7(-3)
35.0 - 40.0	5(-3)	8(-4)	6(-3)
40.0 - 45.0	2(-3)	3(-4)	2(-3)
45.0 - 50.0	2(-3)	3(-4)	2(-3)
Total	5(-2)	9(-3)	5(-2)

†To change miles to km, multiply the values shown by 1.609.
*See Section 5.9.4.5(2).
**This circular zone includes the Site Exclusion Area.
***3(-5) = 3×10^{-5} = .00003
****93% of the area of this annulus is included within an annulus 1-mile wide outside of the site exclusion area boundary.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

APPENDIX M

AN ALTERNATIVE EVALUATION OF THE RELEASE CATEGORIES INITIATED BY CAUSES OTHER THAN SEVERE EARTHQUAKES

The results presented in Sections 5.9.4.5(3), 5.9.4.5(4), and 5.9.4.5(6) and in Appendix L include contributions from the release categories initiated by severe earthquakes, and from the release categories initiated by internal causes, fires, and low to moderately severe earthquakes. The release categories not initiated by severe earthquakes were analyzed with the assumption of Evac-Reloc offsite emergency response mode (see Section 5.9.4.5(2) and Table 5.11f). To provide a reasonable bound to the role of evacuation in risk estimates from the latter release categories, as well as to display sensitivity of risks from these release categories with respect to pertubations in evacuation, an analysis of these release categories was made assuming the Early Reloc mode of offsite emergency response described in Section 5.9.4.5(2). The results of this analysis are provided in this appendix. Only the probability-weighted societal consequences (i.e., the societal risks) resulting from this alternative evaluation are presented below.

Tables M.1a and b are similar to Tables L.1a and b, respectively, in Appendix L. The numbers in the second columns of Tables M.1a and b are the estimates of risks of various kinds from the release categories initiated by causes other than severe earthquakes evaluated with the Early Reloc mode of offsite emergency response. The numbers in the third columns are reproduced from the third columns of Tables L.1a and b and are the estimates of risks ascribed to the severe earthquake-induced release categories as before. The numbers in the fourth columns represent alternative estimates of overall risks (for comparison with those shown in Table 5.11h) from release categories initiated by all causes, and are the sums of the numbers in the preceding columns for each risk type.

Number in parentheses in Tables M.1a and b below the entry for each type of risk (health effects and population exposure only) is the ratio of the risk estimate in these tables and the corresponding risk estimate in Tables L.1a and b. This ratio is indicative of the sensitivity of each type of risk to the choice between the Evac-Reloc and Early Reloc modes of offsite emergency response for the release categories initiated by causes other than severe earthquakes.

From inspection of the ratios (see above), it is apparent that the risk of early fatality (with supportive or minimal medical treatment) is most sensitive to the choice of emergency response mode. The risk of early fatality is about 3 to 4 times as large for the Early Reloc mode as that for the Evac-Reloc mode for release categories not initiated by severe earthquakes. However, because the risk of early fatality is dominated by the release categories initiated by severe earthquakes, the overall risk of early fatality with supportive or minimal medical treatment is only about 20% higher for the choice of the Early Reloc over the Evac-Reloc mode. The other types of risks in

Tables M.1a and b are less sensitive to the choice between the Early Reloc and Evac-Reloc modes.

Tables M.2, M.3, and M.4, respectively, display the contributions to the risks of early fatality with supportive medical treatment and with minimal medical treatment, and latent cancer (including thyroid) fatality from the spatial intervals within 50 miles (80 km) of the plant.

Table M.1a Societal risks within 50 miles (80-km) of Limerick site with Early Reloc* and Late Reloc* offsite emergency response modes

Consequence type	Risk per reactor-year		
	From causes other than severe earthquakes (Early Reloc)	From severe earthquakes (Late Reloc)	Total
1. Early fatalities with supportive medical treatment (persons)	1(-3)** (4)	5(-3)	6(-3) (1)
2. Early fatalities with minimal medical treatment (persons)	2(-3) (3)	8(-3)	1(-2) (1)
3. Early injuries (persons)	1(-2) (1)	1(-2)	2(-2) (1)
4. Latent cancer fatalities, excluding thyroid (persons)	4(-2) (1)	7(-3)	4(-2) (1)
5. Latent thyroid cancer fatalities (persons)	1(-2) (1)	2(-3)	1(-2) (1)
6. Total person-rems	6(2) (1)	9(1)	7(2) (1)
7. Cost of offsite mitigation measures (1980 dollars)	4(4)	5(3)	4(4)
8. Land area for long-term interdiction (square meters)	1(3)	1(2)	1(3)

*See Section 5.9.4.5(2).
**1(-3) = 1×10^{-3} = .001

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table M.1b Societal risks within the entire region of Limerick site with Early Reloc* and Late Reloc* offsite emergency response modes

Consequence type	Risk per reactor-year		
	From causes other than severe earthquakes (Early Reloc)	From severe earthquakes (Late Reloc)	Total
1. Early fatalities with supportive medical treatment (persons)	1(-3)** (4)	5(-3)	6(-3) (1)
2. Early fatalities with minimal medical treatment (persons)	2(-3) (3)	8(-3)	1(-2) (1)
3. Early injuries (persons)	1(-2) (1)	1(-2)	2(-2) (1)
4. Latent cancer fatalities, excluding thyroid (persons)	6(-2) (1)	1(-2)	7(-2) (1)
5. Latent thyroid cancer fatalities (persons)	1(-2) (1)	2(-3)	2(-2) (1)
6. Total person-rems	1(3) (1)	1(2)	1(3) (1)
7. Cost of offsite mitigation measures (1980 dol' rs)	5(4)	6(3)	5(4)
8. Land area for long-term interdiction (square meters)	1(3)	2(2)	1(3)

*See Section 5.9.4.5(2).
**1(-3) = 1 x 10^{-3} = .001

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table M.2 Contributions to risk of early fatality with supportive medical treatment from spatial intervals within 50 miles (80 km) of the Limerick site with Early Reloc* and Late Reloc* offsite emergency response modes

Spatial interval from (mi) - to (mi)†	Risk per reactor-year From causes other than severe earthquakes (Early Reloc) (persons)	From severe earthquakes (Late Reloc) (persons)	Total (persons)
0.0 - 0.5**	6(-5)***	4(-5)	1(-4)
0.5 - 1.0	6(-5)	6(-5)	1(-4)
1.0 - 1.5****	2(-4)	3(-4)	5(-4)
1.5 - 2.0	2(-4)	3(-4)	5(-4)
2.0 - 2.5	1(-4)	4(-4)	5(-4)
2.5 - 3.0	1(-4)	3(-4)	4(-4)
3.0 - 3.5	1(-4)	6(-4)	7(-4)
3.5 - 4.0	9(-5)	5(-4)	6(-4)
4.0 - 4.5	3(-5)	3(-4)	3(-4)
4.5 - 5.0	3(-5)	3(-4)	3(-4)
5.0 - 6.0	2(-5)	3(-4)	3(-4)
6.0 - 7.0	6(-6)	2(-4)	3(-4)
7.0 - 8.5	2(-6)	3(-4)	3(-4)
8.5 - 10.0	6(-7)	2(-4)	2(-4)
10.0 - 12.5	2(-6)	3(-4)	3(-[illegible])
12.5 - 15.0	2(-8)	2(-6)	2(-6)
15.0 - 17.5	3(-8)	5(-8)	8(-8)
17.5 - 20.0	4(-8)	0	4(-8)
20.0 - 25.0	0	0	0
25.0 - 30.0	0	7(-7)	7(-7)
30.0 - 35.0	0	0	0
35.0 - 40.0	0	0	0
40.0 - 45.0	0	0	0
45.0 - 50.0	0	0	0
Total	1(-3)	5(-3)	6(-3)

†To change miles to km, multiply the values shown by 1.609.
*See Section 5.9.4.5(2).
**This circular zone includes the Site Exclusion Area.
***6(-5) = 6×10^{-5} = .00006
****93% of the area of this annulus is included within an annulus 1-mile wide outside of the site exclusion area boundary.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table M.3 Contributions to risk of early fatality with minimal medical treatment from spatial intervals within 50 miles (80 km) of the Limerick site with Early Reloc* and Late Reloc* offsite emergency response modes

	Risk per reactor-year		
Spatial interval from (mi) - to (mi)†	From causes other than severe earthquakes (Early Reloc) (persons)	From severe earthquakes (Late Reloc) (persons)	Total (persons)
0.0 - 0.5**	8(-5)***	4(-5)	1(-4)
0.5 - 1.0	1(-4)	7(-5)	2(-4)
1.0 - 1.5****	3(-4)	3(-4)	7(-4)
1.5 - 2.0	3(-4)	4(-4)	7(-4)
2.0 - 2.5	3(-4)	5(-4)	8(-4)
2.5 - 3.0	2(-4)	4(-4)	6(-4)
3.0 - 3.5	3(-4)	8(-4)	1(-3)
3.5 - 4.0	3(-4)	7(-4)	1(-3)
4.0 - 4.5	1(-4)	4(-4)	5(-4)
4.5 - 5.0	3(-5)	4(-4)	4(-4)
5.0 - 6.0	6(-5)	4(-4)	5(-4)
6.0 - 7.0	3(-5)	4(-4)	4(-4)
7.0 - 8.5	2(-5)	5(-4)	6(-4)
8.5 - 10.0	2(-5)	5(-4)	5(-4)
10.0 - 12.5	9(-5)	1(-3)	1(-3)
12.5 - 15.0	9(-6)	1(-4)	1(-4)
15.0 - 17.5	1(-5)	5(-5)	7(-5)
17.5 - 20.0	1(-5)	2(-5)	3(-5)
20.0 - 25.0	1(-5)	1(-5)	3(-5)
25.0 - 30.0	2(-5)	2(-4)	2(-4)
30.0 - 35.0	1(-5)	9(-6)	2(-5)
35.0 - 40.0	7(-8)	1(-6)	1(-6)
40.0 - 45.0	3(-8)	8(-7)	9(-7)
45.0 - 50.0	3(-6)	3(-7)	3(-6)
Total	2(-3)	8(-3)	1(-2)

†To change miles to km, multiply the values shown by 1.609.
*See Section 5.9.4.5(2).
**This circular zone includes the Site Exclusion Area.
***8(-5) = 8 × 10^{-5} = .00008
****93% of the area of this annulus is included within an annulus 1-mile wide outside of the site exclusion area boundary.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table M.4 Contributions to risk of latent cancer (including thyroid) fatality from spatial intervals within 50 miles (80 km) of the Limerick site with Early Reloc* and Late Reloc* offsite emergency response modes

Spatial interval from (mi) - to (mi)†	Risk per reactor-year: From causes other than severe earthquakes (Early Reloc) (persons)	From severe earthquakes (Late Reloc) (persons)	Total (persons)
0.0 - 0.5**	4(-5)***	1(-6)	4(-5)
0.5 - 1.0	7(-5)	3(-6)	7(-5)
1.0 - 1.5****	3(-4)	2(-5)	3(-4)
1.5 - 2.0	3(-4)	3(-5)	3(-4)
2.0 - 2.5	4(-4)	4(-5)	4(-4)
2.5 - 3.0	4(-4)	4(-5)	4(-4)
3.0 - 3.5	6(-4)	9(-5)	7(-4)
3.5 - 4.0	6(-4)	9(-5)	7(-4)
4.0 - 4.5	3(-4)	5(-5)	3(-4)
4.5 - 5.0	3(-4)	5(-5)	3(-4)
5.0 - 6.0	4(-4)	8(-5)	4(-4)
6.0 - 7.0	3(-4)	8(-5)	4(-4)
7.0 - 8.5	5(-4)	1(-4)	6(-4)
8.5 - 10.0	5(-4)	1(-4)	6(-4)
10.0 - 12.5	3(-3)	8(-4)	4(-3)
12.5 - 15.0	1(-3)	2(-4)	1(-3)
15.0 - 17.5	2(-3)	4(-4)	3(-3)
17.5 - 20.0	2(-3)	4(-4)	2(-3)
20.0 - 25.0	7(-3)	1(-3)	8(-3)
25.0 - 30.0	1(-2)	2(-3)	2(-2)
30.0 - 35.0	6(-3)	1(-3)	7(-3)
35.0 - 40.0	5(-3)	8(-4)	6(-3)
40.0 - 45.0	2(-3)	3(-4)	2(-3)
45.0 - 50.0	2(-3)	3(-4)	2(-3)
Total	5(-2)	9(-3)	6(-2)

†To change miles to km, multiply the values shown by 1.609.
*See Section 5.9.4.5(2).
**This circular zone includes the Site Exclusion Area.
***4(-5) = 4 x 10^{-5} = .00004
****93% of the area of this annulus is included within an annulus 1-mile wide outside of the site exclusion area boundary.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

APPENDIX N

CRITIQUE OF APPLICANT'S CONSEQUENCE ANALYSIS IN LIMERICK GENERATING STATION ENVIRONMENTAL REPORT-OPERATING LICENSE (ER-OL)

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CRITIQUE OF APPLICANT'S CONSEQUENCE ANALYSIS IN LIMERICK GENERATING STATION ENVIRONMENTAL REPORT-OPERATING LICENSE (ER-OL)

In the ER-OL, a total of 11 source terms (or release categories) were used. Some of these release categories were the result of binning (or grouping) of several individual source terms. In some of the bins, the member source terms had very dissimilar release characteristics and release fractions, and the source terms selected to represent the bins were considered by the staff to be unrepresentative of the bins. For this reason, the staff did not use the ER-OL binning of the source terms and chose to use a greater number and more consistent set of release categories in its consequence analysis. However, the 11 different sets of release fractions (source terms) used in the ER-OL and the 27 release categories used in the staff analysis are intended to encompass an equivalent number of combinations of the plant damage states and containment failure modes.

The point estimates of radionuclide release fractions for the 11 source terms in the ER-OL are generally lower and warning times for evacuation associated with some of these source terms are longer than those for the release categories used in the staff analysis. However, exact comparison of source term specifications between those in ER-OL and in the staff analysis is difficult because of the different numbers of source terms used in the two analyses.

The point estimates of probabilities of the source terms in the ER-OL add up to 6×10^{-6} per reactor-year for seismic causes and 4×10^{-5} per reactor-year for non-seismic causes. The staff analysis uses the same total value for the point estimates of the probabilities of the seismically induced release categories; however, the staff's total of the point estimates of the probabilities of non-seismically induced release categories is 9×10^{-5} per reactor-year.

The consequence analysis in the ER-OL used the CRAC2 computer code, which is a modified version of the CRAC code used in the Reactor Safety Study (WASH-1400 NUREG-75/014). Both CRAC2 and the staff version of CRAC (1980) incorporate the same evacuation model which is revised from that used in WASH-1400. The revised evacuation model is capable of incorporating people's delay time before evacuation in addition to their speed during evacuation. Both the codes are also capable of modeling a variety of offsite emergency response options--such as shelter and relocation--in addition to evacuation separately or in combination. CRAC2 incorporates a modified scheme for sampling the weather data in addition to the usual sampling schemes of CRAC. However, using the modified weather sampling scheme of CRAC2 and the stratified sampling scheme of CRAC, both the codes produced almost identical results, within likely uncertainty bands, in international benchmark exercises for comparison of codes used in consequence analysis. Therefore, the use of CRAC2 in the ER-OL is acceptable to the staff. However, the staff chose to use CRAC for its independent consequence analysis for two reasons:

(1) Although CRAC and CRAC2 produced almost equal results, within likely uncertainty bands, for benchmark problems, there are some differences in results produced by the two codes for other problems which have yet to be properly explained. A detailed comparison between CRAC and CRAC 2 has been sponsored by the staff at Oak Ridge National Laboratory. After the differences between the two codes are understood, the staff may use CRAC2, with or without any additional modifications, in future applications.

(2) The other reason for using CRAC in the staff analysis is that the staff has used the 1980 version of CRAC in severe accident consequence analyses in the environmental statements issued after July 1, 1980, pursuant to the Commission Statement of Interim Policy, June 13, 1980 (45 FR 40101-40104). The staff has provided a comparison of risk estimates for Limerick with those made using CRAC in environmental statements for other plants and the use of CRAC2 could prove inconsistent.

Five years' worth of meteorological data (from 1972 to 1976) was used in the ER-OL consequence analysis after some modifications were made to CRAC2, which normally uses only 1 year of meteorological data. In response to the staff question as to the degree of improvement achieved by using 5 years of data, the applicant provided a comparison of CRAC2 runs for sample problems using each of the five 1-year data periods separately with those using data for the entire 5-year period. The comparison did not show much difference between these runs. Further, in response to the staff question regarding the adequacy of the number of weather sequences sampled from 5 years of data, the applicant presented a comparison of CRAC2 runs for sample problems with increased weather sequence samples. No appreciable difference as a result of the increased sampling was noticed. Therefore, the use of 5 years' worth of meteorological data and the sampling scheme in the ER-OL are acceptable to the staff.

The ER-OL analysis used a core inventory of radionuclides (excluding activation products) calculated for a BWR at a power level of 3293 MWt. However, the staff analysis used 105% of this power level (3458 MWt), and calculated the core inventory based upon WASH-1400 estimates of fission and activation product distributions. The use of a lower power level would result in lower offsite consequences.

The ER-OL analysis used an estimated population distribution for the year 2000 up to 500 miles (800 km) from the plant, and economic data related to land use on county-wide basis up to 50 miles (80 km) and on a state-wide basis outside 50 miles (80 km). These are acceptable to the staff, although staff used its own estimates of inputs to the CRAC code. The other economic data in the ER-OL are not site specific, but they are site-specific in the staff analysis.

For releases not caused by severe earthquakes, the ER-OL analysis used a generic set of parameters for evacuation within the 10-mile (16-km) Emergency Planning Zone (EPZ): 1-, 3-, and 5-hour delay times with probabilities of 0.3, 0.4, and 0.3, respectively, and 10 mph (16 km per hour) for effective evacuation speed. Because this is not site specific, it is unacceptable to the staff. A study prepared by the NUS Corporation for the applicant in 1980 provides a basis for the estimate of effective evacuation speed of about 2.5 mph (4 km per hour), considering the road network and the expected traffic loading for evacuation from the 10-mile (16-km) EPZ during emergency. The estimate of the site-specific delay time of about 5 hours made in the NUS study was rejected by the applicant because the study did not take into account the early warning system that would be required for notification of emergency before the plant would be licensed for operation. The staff recognizes the applicant's position. However, in lieu of any available estimate of delay time for the site, the staff assumed a delay time of 2 hours, which is consistent with similar estimates for other high population density sites. The ER-OL assumed a maximum distance of 20 miles (32 km) traveled by the evacuees; however staff used 15 miles (24 km) for this distance, as it has for other plants. The staff assumptions of 2 hours for delay time, 2.5 mph (4 km per hour) of evacuation speed, and a travel distance of 15 miles (24 km) are applied to the situations of releases as a result of plant-internal causes, fires, and low to moderately severe earthquakes (see Section 5.9.4.5(2) for an alterative to the assumption of evacuation from the 10-mile (16-km) EPZ). For these situations, the ER-OL also assumes relocation of people from the 10- to 25-mile (16- to 40-km) region 12 hours after passage of radioactive plume. Although a similar assumption has been made by the staff in the consequence analyses in the environmental statements for several other plants, the staff judgment is that this assumption for a site with high population density would not be appropriate because the large number of people that would be involved in the 10- to 25-mile (16- to 40-km) region would make this scenario unrepresentative. Instead, the staff analysis assumes that outside of the 10-mile (16-km) EPZ, only people from the highly contaminated areas (see Section 5.9.4.5(2)) would be relocated 12 hours after plume passage. Shielding factors used in the ER-OL are: (1) the same as in the staff analysis during evacuation, (2) higher than the staff's during delay before evacuation, and (3) lower than the staff's during waiting before relocation. The values used by the staff are the same as those used in WASH-1400. The impact of differences in shielding factors used in the ER-OL from those in WASH-1400 is difficult to assess, although it is not likely to be substantial.

For releases caused by severe earthquakes, the ER-OL assumes evacuation from the 10-mile (16-km) EPZ after a 3-hour delay with an effective speed of 0.5 meter/sec, and relocation from 10- to 25-mile (16- to 40-km) region 24 hours after plume passage. However, the severity of earthquakes assumed is Modified Mercalli intensity scale of IX or higher, and it is the judgment of the staff that earthquakes of such severity would cause very extensive damage in the site region that would seriously hamper the evacuation. Therefore, the staff assumed no evacuation for these situations but, instead, assumed relocation of people from highly contaminated areas 24 hours after plume passage. Shielding factors used by the staff are also more pessimistic. The ER-OL analysis assumed an effective peak ground acceleration of 0.61g or more to be associated with Modified Mercalli intensity scales of IX of higher. However, the staff used 0.4g as the dividing line, although it recognizes that there is lack of actual recordings of effective peak ground accelerations associated with the intensity scales. It is the staff's judgment that although a range of effective peak ground acceleration of 0.35g to 0.5g would be more appropriate, the results of consequence analysis are not sensitive to the choice of values within a range of 0.35g to 0.5g. Therefore, the staff used only the single value of 0.4g. The ER-OL assumptions regarding the offsite emergency response during severe earthquake conditions as well as the assumption of 0.61 g as the dividing line for classification of less severe and very severe earthquakes result in lower estimates of risks from seismically induced source terms.

The ER-OL point estimates of risk from the 11 source terms and the staff's point estimates of risks from 27 release categories are as follows:

Type of risk	Risk per reactor-year	
	ER-OL*	Staff
1. Early fatalities with supportive medical treatment (persons)	3(-4)**	5(-3)
2. Latent cancer fatalities excluding thyroid (persons)		
50-mile (80-km) region	2(-2)	4(-2)
Entire region	3(-2)	7(-2)
3. Latent thyroid cancer fatalities (persons)		
50-mile (80-km) region	5(-3)	1(-2)
Entire region	6(-3)	1(-2)
4. Whole body person-rems		
50-mile (80-km) region	300	700
Entire region	500	1000
5. Cost of offsite mitigation measures (1980 dollars)	20,000	50,000

*On March 13, 1984, the applicant informed the staff that the ER-OL consequence calculations are being revised and that the revised calculations will not result in significant changes in the results currently presented in the ER-OL. Based upon the applicant's explanation of the source of the error, the staff judges that the impact of these revisions will be relatively small.

**$3(-4) \times 10^{-4} = .0003$. Estimated numbers were rounded to one significant digit only for the purpose of this table.

In the ER-OL, an uncertainty analysis on risks is provided with respect to four major parameters

(1) probability of each source term

(2) magnitude and other release characteristics of some of the dominant source terms

(3) evacuation and sheltering parameters

(4) dose-response relationships for early fatality with three types of medical treatment

The first of these parameters was treated by system analysis and standard methods of combining uncertainties. The other three were treated by a sensitivity study using the CRAC2 code to provide a large number of conditional CCDFs for the 11 different sets of release fractions (source terms). These CCDFs were used to define the upper and lower conditional CCDFs for the source terms. The upper and lower CCDFs were combined probabilistically with the uncertainity distribution on source term probabilities in order to generate the uncertainty bands on the overall CCDFs.

The variations used in source term parametrization were mostly subjective. For offsite emergency response the evacuation speed was varied from 2.5 to 10 mph (4 to 16 km an hour), while the delay time before evacuation ranged from 1 to 5 hours. For severe earthquake conditions, no variation in the parameters of offsite emergency response was made. For the 10- to 25-mile (16- to 40-km) region, sheltering in basements for 24 hours followed by rapid relocation was used; the 25-mile (40-km) distance was also extended to 50 miles (80 km). Considering that the state of the art of uncertainty assessment in consequence analysis is not well developed, this method of uncertainty analysis in the ER-OL is acceptable to the staff. However, the lack of any variation in the pessimistic direction in offsite emergency response parameters for the severe earthquake conditions and too many variations in the optimistic direction for nonsevere earthquake conditions, and the lack of variation in the source terms to encompass some of the high values of the release fractions as used in the staff analysis, lead the staff to disagree with the upper estimates of the overall CCDFs in the ER-OL.

By letter, dated March 13, 1984, PECo states that errors had been discovered in the ER-OL consequence analysis. PECo has further stated that these errors, when corrected, will not significantly alter the ER-OL conclusions.

The staff also performed a limited sensitivity analysis. With respect to variation of probability of earthquake-induced release categories, the staff concluded that the staff's point estimates of risks could be exceeded by factors of up to 6, but could also be lower by factors up to 3. With respect to parameters of offsite emergency response the overall risks could be increased by up to 20%. With respect to medical treatment, the risk of early fatality could have a spread within factors of 2 to 3. The staff has not performed a sensitivity study with respect to probabilities of release categories initiated by causes other than severe earthquakes, source terms, and other elements that contribute to uncertainties. Based upon the insight gained from review of similar PRAs for Indian Point and Zion, it is the judgment of the staff that the staff's Limerick risk estimates could be too low by a factor of about 40 or too high by a factor of about 400.